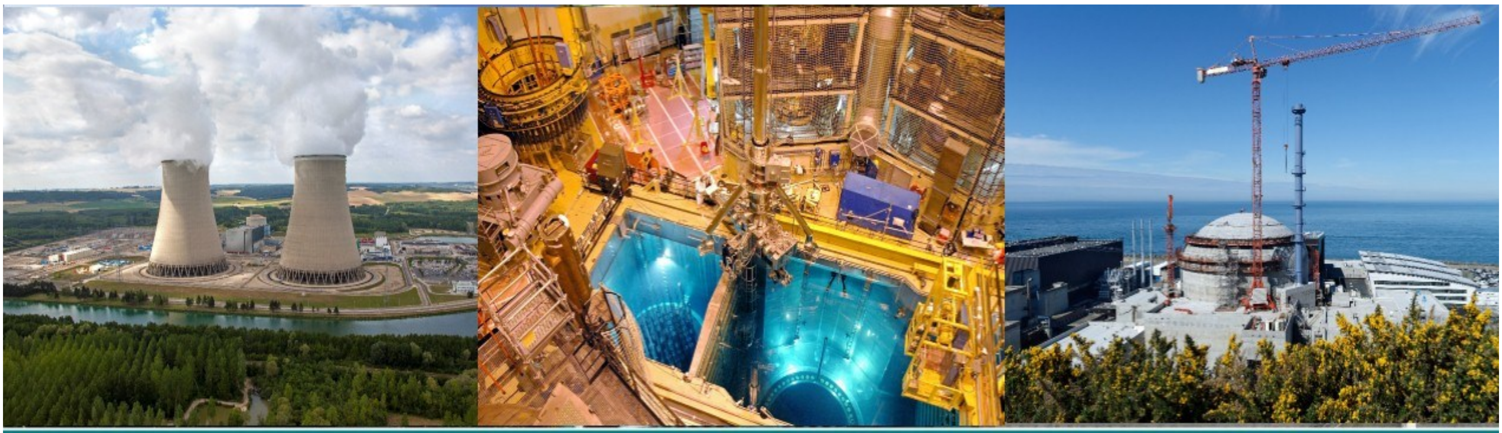

FRANCE

SAFETY ASSESSMENT OF NUCLEAR FACILITIES IN FRANCE

AGEING MANAGEMENT

NATIONAL REPORT

**PRODUCED PURSUANT TO ARTICLE 8e
OF COUNCIL DIRECTIVE 2014/87/EURATOM
DATED 8 JULY 2014**

December 2017

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RÉPUBLIQUE FRANÇAISE

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Summary:

The ageing management of nuclear facilities is incorporated into the French regulations, more specifically in:

1) The Environment Code, which includes the provisions relating to the periodic safety review process for all Basic Nuclear Installations,

2) Orders which comprise provisions concerning the regulation of the design/construction or in-service monitoring of nuclear pressure equipment (NPE). It should be pointed out that the regulations make provision for partial requalification of the NPE every 5 years, as of the third ten-yearly outage,

3) ASN requests concerning licensee management of the ageing of its nuclear facilities and their potential continued operation beyond 40 years, in particular with regard to the nuclear power reactors,

4) The provisions of ASN-IRSN guide n° 22 "Pressurised Water Reactor design" applicable to the search for improvements to be made to existing reactors, which extend to non-NPE components the provisions of the order of 30 December 2015, concerning NPE, on the need to take account of equipment ageing as of the design stage.

French regulations set no time limit on the operation of the facilities.

Nuclear power reactors

The nuclear power reactor fleet currently consists of 58 pressurised water reactors (PWR). These reactors are operated by EDF. They were built in successive standardised series (900 MWe, 1300 MWe and N4) with an average age of 36, 30 and 20 years respectively.

In 2009, EDF announced its intention to extend the operating life of its NPPs beyond 40 years.

As of 2001, EDF initiated the development and implementation of an ageing management programme within the framework of its reactor third ten-yearly outage inspections (VD3). This programme is primarily based on design, operating, in-service monitoring and routine maintenance measures, supplemented by exceptional maintenance.

EDF has adopted an approach to demonstrate the ageing management of the systems, structures and components (SSC) potentially affected. This approach is built around 4 steps:

• ***SSC selection process****: The SSCs potentially susceptible to ageing and whose failure can have an impact on safety are identified,*

• ***Individual analysis of ageing mechanisms****: the ageing mechanisms for each SSC are identified and analysed, in order to check management of their ageing with regard to the operations and maintenance provisions in force. This step also comprises an analysis of the actual repairability and/or replaceability of the SSCs. This analysis is contained in an Ageing Analysis Sheet (AAS);*

• ***Additional actions and studies****: when ageing management cannot be demonstrated by means of normal operating provisions, additional actions or studies are identified for ageing management. This step is contained in a component Detailed Ageing Analysis Report (DAAR),*

• ***Drafting of an ageing analysis report specific to the reactor (UAAR),*** *for each reactor nearing its VD3 and for the subsequent VDs, based on the generic AASs and component DAARs.*

ASN emphasises the fact that the EDF ageing management programme complies with the requirements of international standards. *Furthermore, it appropriately incorporates national and international operating experience feedback and is accompanied by a major research and development (R&D) programme.*

With respect to its request for the continued operation of the NPPs, EDF proposes reusing this approach for the fourth ten-yearly outage inspections (VD4). This approach will be extended to all SSCs important for the management not only of radiological risks, but also conventional risks.

With regard to ***the electrical cables,*** *the approach adopted by EDF for ageing management covers all the cables needed for operation of the reactors. The degradation mechanisms were studied on the basis of national and international operating experience feedback, as well as R&D on the behaviour of polymer materials.*

For the purposes of in-service monitoring, EDF identifies the cables subjected to particular environmental or operational stresses. As necessary, EDF implements specific checks to detect ageing symptoms (measurement of delta tangent and partial discharges for the MV cables, visual inspection of LV cables). ***ASN considers that the checks carried out are in compliance with the state of the art and are satisfactory.***

ASN also considers that the characterisation work carried out on the cables sampled from the EDF reactors, in conjunction with the conclusions of the cable predictive lifetime studies, give a high level of confidence with regard to their ability to retain their original functionality for the next 10 years.

With regard to ***concealed pipework****, EDF has defined a programme of inspections for piping zones identified as being susceptible, not only for buried piping but also for piping which cannot be inspected or which is inaccessible. This identification is based on a risk assessment, taking account of the consequences of the failure of the piping in terms of safety, radiation protection and the environment. In this context, inspections are under way, as at the date of publication of this report, on the Tricastin, Fessenheim and Bugey sites, with the aim of defining a generic programme of inspections and ensuring that, for the VD4, it is possible to conclude whether these pipes can be maintained in service or need to be refurbished. The examination is in progress and the conclusions are expected in 2018. The first results of EDF's approach will be presented to the ASN Advisory Committee for reactors (GPR) in 2018.*

With regard to the ***reactor pressure vessel,*** *the approach adopted by EDF for management of its ageing was the subject of numerous examinations between 1987 and 2015. This approach is based on monitoring and on mechanical studies. These studies enable the loadings to be identified, more specifically in the irradiated core zones, with selection of the most severe thermal-hydraulic transients.* ***ASN considers this approach to be satisfactory.***

The monitoring of the equipment of the reactor coolant system and thus of the reactor pressure vessel, is covered by the regulations specific to NPE defining the general requirements and requiring the implementation of:

• in-service monitoring measures demonstrating that the equipment is functioning in the same situations as provided for in the design (situations accounting),

• checks to detect flaws harmful to the integrity of the equipment; these checks take account of the susceptibility of a zone to a degradation mode (including the risk of fast fracture) and of national and international operating experience feedback,

• a programme to monitor the properties of the materials having an impact on the integrity demonstration: the irradiation-induced ageing mechanism is the subject of a specific programme to monitor changes in the properties of the reactor pressure vessel steel (mechanical tests on test pieces of the RPV material subjected to neutron radiation in the vessel).

As part of its RPV ageing management programme, EDF has identified the ageing mechanisms through national and international operating experience feedback, the results of inspections and tests performed on the reactors in operation and permanent measurements taken on the equipment. National and international operating experience feedback has already led to the implementation of preventive measures (reduction in the neutron flux at the reactor pressure vessel hot spot) and corrective measures (replacement of all the RPV closure heads equipped with Inconel 600 adapters following the discovery in 1991 of stress corrosion on an RPV closure head adapter in Bugey), or the implementation of appropriate checks or monitoring (for example, checks initiated on the French RPVs following the discovery of micro-cracks on the RPVs in the Belgian nuclear power reactors).

On the basis of the ageing analyses contained in the AASs, EDF drafted a Detailed Ageing Analysis Report (DAAR) for the reactor pressure vessel, which deals with two topics requiring additional actions or studies: radiation-induced ageing of the core zone and thermal ageing of the outlet nozzles.

*With regard to the **containments**, their ageing management programme applies to the concrete containments and to their liners and coatings: it benefits from EDF fleet operating experience feedback owing to their uniform design.*

As part of its containments ageing management programme, EDF identified the ageing mechanisms on the basis of their operation and the operating experience feedback from the inspections and tests carried out. These mechanisms concern the concrete (cracking, creep, shrinkage, carbonation and internal swelling pathologies), the metal liner (corrosion and blistering), the prestressing tendons (relaxation and corrosion), the passive rebars (corrosion), the containment instrumentation system (malfunction of certain instruments), as well as the paints and composite coatings (ageing and behaviour in accident situation, including a severe accident).

The containment tightness tests and the monitoring of mechanical behaviour using the containment instrumentation system, are a means of observing and anticipating the mechanical behaviour and tightness of the containments: this notably led to the implementation of both preventive measures (installation of inner coatings) and corrective measures (repair of outer facings). ASN considers that the implementation of these measures leads to satisfactory management of the containment ageing phenomena.

Research reactors

The research reactors in service, with a power output of 1 MWth or more, are the Cabri and Orphée reactors operated by CEA and the high-flux reactor (RHF) of the Max von Laue-Paul Langevin Institute (ILL). They were commissioned in 1963, 1980 and 1971 respectively.

*Ageing management of **research reactors** is tailored to each individual facility. It takes account of the diversity of operating conditions and in particular the replaceability of parts which are sometimes produced in very small quantities. One specific feature of research reactors must be highlighted: practically all the equipment, except for the containment, can be replaced.*

Ageing management of the research reactors is currently based on the maintenance programmes and on periodic checks and tests.

ASN considers that ageing management must be defined in a more formal manner by the research reactor licensees. In particular, the licensees must implement an approach for ensuring that the checks and tests performed are sufficient and, as necessary, define additional checks to ensure the ability of the equipment to perform its functions in the light of the ageing mechanisms that could affect it.

More specifically, ageing management of the ***electrical cables*** *consists mainly of specific measurements (resistance of measurement lines, isolation resistance of classified cables) and partial visual inspection of the condition of the insulation.* ***ASN considers that the ageing management programme is limited and needs to be enhanced, more particularly for classified cables subject to environmental or operational stresses, in order to provide a lasting guarantee that they are able to perform their functions****.*

With regard to ***concealed pipework****, the corresponding ageing management has no safety implications insofar as these pipes do not perform safety functions.*

With regard to the ***reactor pressure vessel,*** *the ageing problem is not comparable to that of the nuclear power reactors. The pressure vessel is a replaceable item and is not therefore a limiting factor in the lifetime of the facility. At each periodic safety review, the licensees reassess the lifetimes of the components according to the fluence received and replace them as and when necessary. In addition, checks are carried out to ensure that there is no corrosion: this corrosion may be caused by the quality of the water or the susceptibility to intergranular corrosion for high neutron flux levels.*

With regard to the ***containments,*** *their ageing is mainly monitored during periodic checks and tests. The compliance of the containments is verified during the periodic safety reviews.* ***ASN considers that the ageing management programme remains limited. This programme must be enhanced and, on the basis of their specific features and the results of tests and checks as well as the knowledge derived from national and international R&D programmes, the ageing mechanisms which could affect them must be identified.***

Conclusion

With regard to the nuclear power reactors, ASN highlights the fact that since 2001, EDF has succeeded in developing an ageing management programme which addresses the main nuclear safety and radiation protection issues. This programme is also reinforced with a view to continued operation beyond 40 years. ASN will rule on this programme in its generic position statement on the VD4.

With regard to research reactors, while admitting the specific nature of each facility, ASN considers that the ageing management programmes must be enhanced within the framework of the periodic safety reviews, for the Cabri and RHF reactors.

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1 GENERAL INFORMATION

1.1 IDENTIFICATION OF NUCLEAR FACILITIES ASSESSED

1.1.1 EDF NUCLEAR POWER REACTORS

The fleet of nuclear power reactors in service falling within the scope of this assessment comprises 58 pressured water reactors (PWR), built in successive standardised series, which were coupled to the grid between 1977 and 1999 and are all in service.

In 2016, EDF's PWR nuclear power reactors produced 384 TWh, or about 72% of all electricity produced in France (416.8 TWh and 76% in 2015, 415.9 TWh and 77% in 2014 respectively). They are grouped on 19 NPPs which each comprises two to six reactors of the same series. The 58 reactors were built by the same supplier, Framatome, which is today known as AREVA NP. There are three different plant series (900 MWe, 1300 MWe and N4[1]).

The 900 MWe series comprises:

- the CP0 series, comprising the 2 reactors at Fessenheim and the 4 reactors at Le Bugey;
- the CPY series, consisting of the other 28 reactors of 900 MWe (4 reactors at Dampierre, 6 reactors at Gravelines, 4 reactors at le Blayais, 4 reactors at Tricastin, 4 reactors at Chinon, 4 reactors at Cruas and 2 reactors at Saint-Laurent-des-Eaux).

The 1300 MWe series comprises:

- the P4 series, consisting of 4 reactors at Paluel, 2 reactors at Flamanville and 2 reactors at Saint-Alban;
- the P'4 series, consisting of 2 reactors at Belleville-sur-Loire, 4 reactors at Cattenom, 2 reactors at Golfech, 2 reactors at Nogent-sur-Seine and 2 reactors at Penly.

The N4 series comprises 4 reactors of N4 and consists of the 2 reactors at Chooz and the 2 reactors at Civaux.

In December 2017, the average age of the reactors, based on the first reactor criticality dates, stood as follows:

- 36 years for the thirty-four 900 MWe reactors;
- 30 years for the twenty 1300 MWe reactors;
- 20 years for the four N4 reactors.

During the course of the plant series design process, various changes were introduced:

- the design of the CPY series buildings differs slightly from those of the CP0 series: the dome of the CP0 reactors has an external tightness covering.
- significant changes in relation to the CPY plant series were made in the design of the core protection circuits and systems for the 1300 MWe reactors and in the buildings housing them. The power increase results in a reactor coolant system with four loops instead of three. Furthermore, the reactor containment comprises a double concrete wall instead of a single wall with a steel leaktightness liner. The reactors of the P'4 series differ slightly from those of the P4 series, particularly with regard to the fuel building and system configurations.

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[1] 1450 MWe.

- the N4 series differs from the previous series, notably with a more compact steam generator design, different reactor coolant pumps and the use of a computerised interface for reactor operations.

In 2009, EDF informed ASN that it wished to extend the operating life significantly beyond forty years and to maintain open the option of an operating life of 60 years for all reactors in service[2].

Construction of an EPR type reactor began on the Flamanville site in 2007.

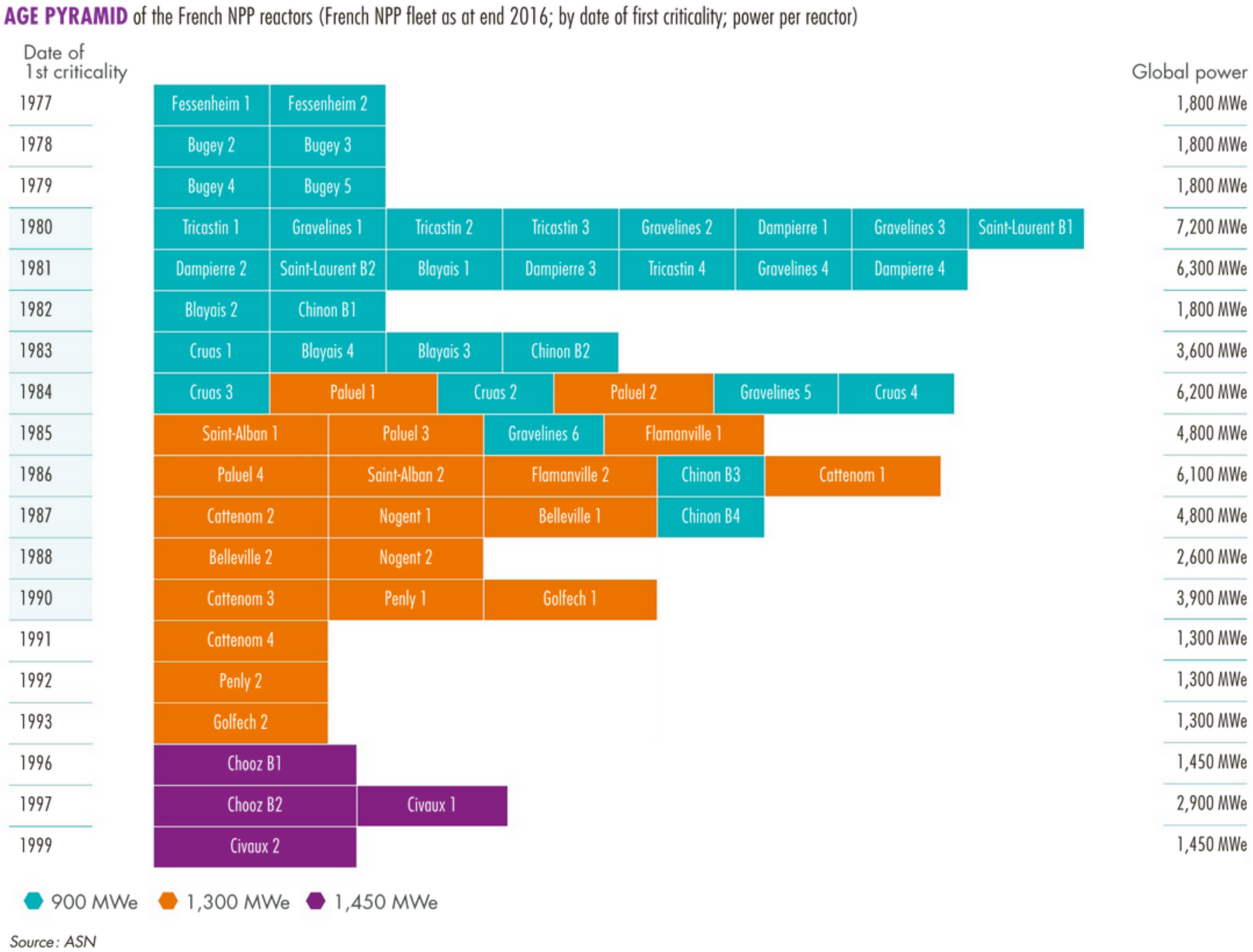

Figure 1 – Age pyramid of EDF nuclear power reactors

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[2] See EDF DPI letter of 29 January 2009 and EDF DIN letter of 17 March 2009.

1.1.2 LIST OF NUCLEAR POWER REACTORS

No. BNI	NAME AND LOCATION OF THE FACILITY	Licensee	Type of facility	Date of first criticality:
75	FESSENHEIM NPP (reactors 1 and 2) 68740 Fessenheim	EDF	2 PWR reactors CP0 900 MWe	1977
78	BUGEY NPP (reactors 2 and 3) 01980 Loyettes	EDF	2 PWR reactors CP0 900 MWe	1978
84	DAMPIERRE-EN-BURLY NPP (reactors 1 and 2) 45570 Ouzouer-sur-Loire	EDF	2 PWR reactors CP1 900 MWe	1980 1981
85	DAMPIERRE-EN-BURLY NPP (reactors 3 and 4) 45570 Ouzouer-sur-Loire	EDF	2 PWR reactors CP1 900 MWe	1981
86	LE BLAYAIS NPP (reactors 1 and 2) 33820 Saint-Ciers-sur-Gironde	EDF	2 PWR reactors CP1 900 MWe	1981 1982
87	TRICASTIN NPP (reactors 1 and 2) 26130 Saint-Paul-Trois-Châteaux	EDF	2 PWR reactors CP1 900 MWe	1980
88	TRICASTIN NPP (reactors 3 and 4) 26130 Saint-Paul-Trois-Châteaux	EDF	2 PWR reactors CP1 900 MWe	1980 1981
89	LE BUGEY NPP (reactors 4 and 5) 01980 Loyettes	EDF	2 PWR reactors CP1 900 MWe	1979
96	GRAVELINES NPP (reactors 1 and 2) 59820 Gravelines	EDF	2 PWR reactors CP1 900 MWe	1980
97	GRAVELINES NPP (reactors 3 and 4) 59820 Gravelines	EDF	2 PWR reactors CP1 900 MWe	1980 1981
100	ST-LAURENT-DES-EAUX NPP (reactors B1 and B2) 41220 La Ferté-St-Cyr	EDF	2 PWR reactors CP2 900 MWe	1980 1981
103	PALUEL NPP (reactor 1) 76450 Cany-Barville	EDF	1 PWR reactor P4 1300 MWe	1984
104	PALUEL NPP (reactor 2) 76450 Cany-Barville	EDF	1 PWR reactor P4 1300 MWe	1984
107	CHINON NPP (reactors B1 and B2) 37420 Avoine	EDF	2 PWR reactors CP2 900 MWe	1982 1983

No. BNI	NAME AND LOCATION OF THE FACILITY	Licensee	Type of facility	Date of first criticality:
108	FLAMANVILLE NPP (reactor 1) 50830 Flamanville	EDF	1 PWR reactor P4 1300 MWe	1985
109	FLAMANVILLE NPP (reactor 2) 50830 Flamanville	EDF	1 PWR reactor P4 1300 MWe	1986
110	LE BLAYAIS NPP (reactors 3 and 4) 33820 Saint-Ciers-sur-Gironde	EDF	2 PWR reactors CP1 900 MWe	1983
111	CRUAS NPP (reactors 1 and 2) 07350 Cruas	EDF	2 PWR reactors CP2 900 MWe	1983 1984
112	CRUAS NPP (reactors 3 and 4) 07350 Cruas	EDF	2 PWR reactors CP2 900 MWe	1984
114	PALUEL NPP (reactor 3) 76450 Cany - Barville	EDF	1 PWR reactor P4 1300 MWe	1985
115	PALUEL NPP (reactor 4) 76450 Cany - Barville	EDF	1 PWR reactor P4 1300 MWe	1986
119	SAINT-ALBAN NPP (reactor 1) 38550 Le Péage-de-Roussillon	EDF	1 PWR reactor P4 1300 MWe	1985
120	SAINT-ALBAN NPP (reactor 2) 38550 Le Péage-de-Roussillon	EDF	1 PWR reactor P4 1300 MWe	1986
122	GRAVELINES NPP (reactors 5 and 6) 59820 Gravelines	EDF	2 PWR reactors CP1 900 MWe	1984 1985
124	CATTENOM NPP (reactor 1) 57570 Cattenom	EDF	1 PWR reactor P'4 1300 MWe	1986
125	CATTENOM NPP (reactor 2) 57570 Cattenom	EDF	1 PWR reactor P'4 1300 MWe	1987
126	CATTENOM NPP (reactor 3) 57570 Cattenom	EDF	1 PWR reactor P'4 1300 MWe	1990
127	BELLEVILLE-SUR-LOIRE NPP (reactor 1) 18240 Léré	EDF	1 PWR reactor P'4 1300 MWe	1987
128	BELLEVILLE-SUR-LOIRE NPP (reactor 2) 18240 Léré	EDF	1 PWR reactor P'4 1300 MWe	1988

No. BNI	NAME AND LOCATION OF THE FACILITY	Licensee	Type of facility	Date of first criticality:
129	NOGENT-SUR-SEINE NPP (reactor 1) 10400 Nogent-sur-Seine	EDF	1 PWR reactor P'4 1300 MWe	1987
130	NOGENT-SUR-SEINE NPP (reactor 2) 10400 Nogent-sur-Seine	EDF	1 PWR reactor P'4 1300 MWe	1988
132	CHINON NPP (reactors B3 and B4) 37420 Avoine	EDF	2 PWR reactors CP2 900 MWe	1986 1987
135	GOLFECH NPP (reactor 1) 82400 Golfech	EDF	1 PWR reactor P'4 1300 MWe	1990
136	PENLY NPP (reactor 1) 76370 Neuville-lès-Dieppe	EDF	1 PWR reactor P'4 1300 MWe	1990
137	CATTENOM NPP (reactor 4) 57570 Cattenom	EDF	1 PWR reactor P'4 1300 MWe	1991
139	CHOOZ B NPP (reactor 1) 08600 Givet	EDF	1 PWR reactor N4 1450 MWe	1996
140	PENLY NPP (reactor 2) 76370 Neuville-lès-Dieppe	EDF	1 PWR reactor P'4 1300 MWe	1992
142	GOLFECH NPP (reactor 2) 82400 Golfech	EDF	1 PWR reactor P'4 1300 MWe	1993
144	CHOOZ B NPP (reactor 2) 08600 Givet	EDF	1 PWR reactor N4 1450 MWe	1997
158	CIVAUX NPP (reactor 1) BP 1 86320 Civaux	EDF	1 PWR reactor N4 1450 MWe	1997
159	CIVAUX NPP (reactor 2) BP 1 86320 Civaux	EDF	1 PWR reactor N4 1450 MWe	1999
167	FLAMANVILLE NPP (reactor 3) 50830 Flamanville	EDF	1 PWR reactor EPR 1600 MWe	Under construction

Table 1 – List of nuclear power reactors

1.1.3 LOCATION OF NUCLEAR POWER REACTORS

The 58 nuclear power reactors in operation are distributed around France as shown in the map below. The Flamanville EPR reactor is under construction.

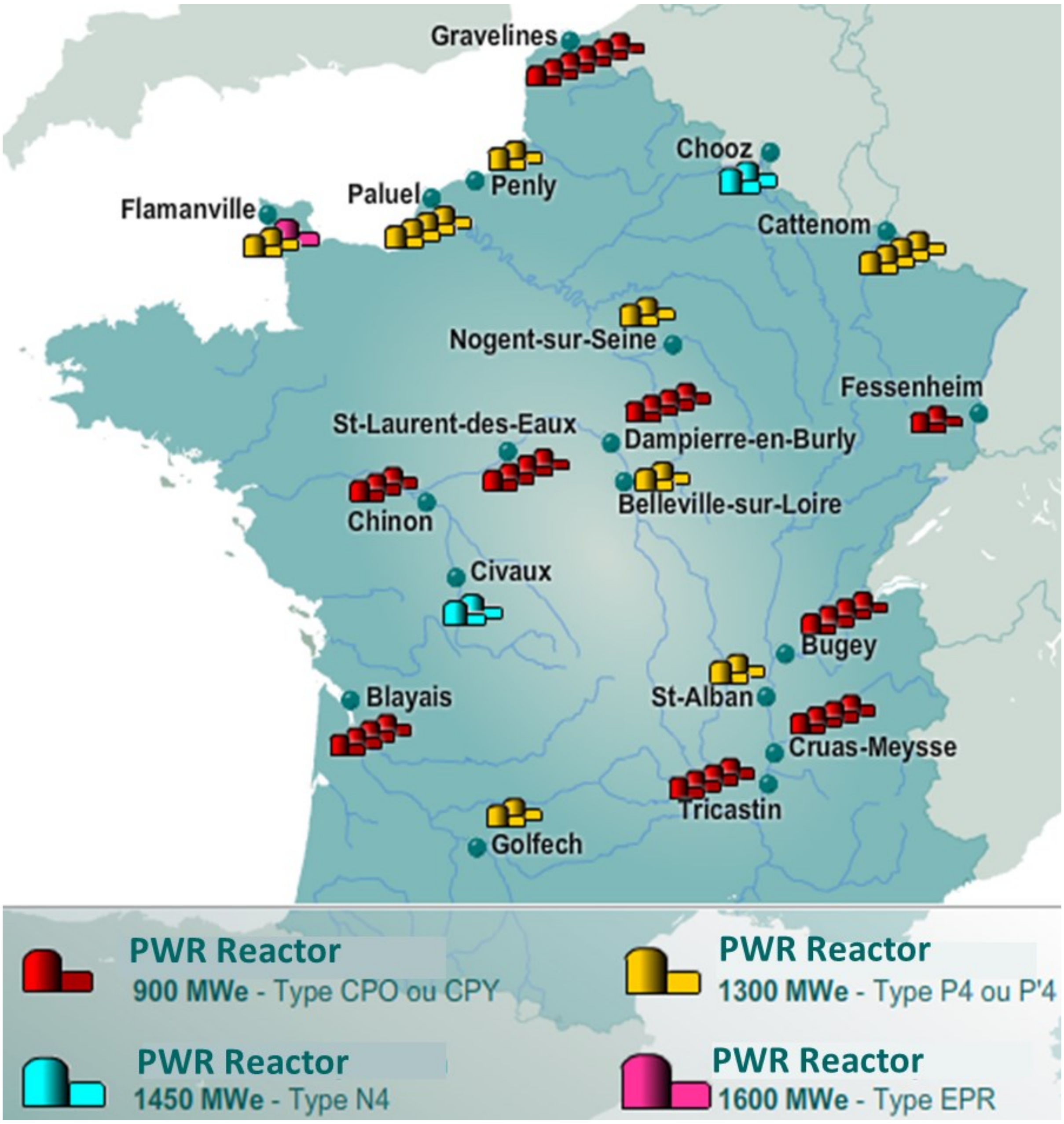

Figure 2 – Location of nuclear power reactors in service and under construction in France

1.1.4 RESEARCH REACTORS

Ten research reactors are subject to the regulatory conditions of a facility in service.

BNI N°	Reactor name	Date of commissioning	Power	Type of facility	Status
24	CABRI	1963	25 MWth + power pulse	reactor for studying accident and safety situations	in service
39	MASURCA	1966	0.005 MWth	critical mock-up	prolonged temporary outage
40	ISIS	1966	0.7 MWth	teaching and training reactor	in service
40	OSIRIS	1966	70 MWth	technological irradiation reactor	finally shutdown
42	ÉOLE	1965	0.0001 MWth	critical mock-up	in service
37	RHF	1971 1994[3]	58.3 MWth	fundamental research reactor	in service
92	PHÉBUS	1977	38 MWth	reactor for studying accident and safety situations	finally shutdown
95	MINERVE	1959 1977[4]	0.0001 MWth	critical mock-up	in service
101	ORPHEE	1980	14 MWth	fundamental research reactor	in service
172	RJH	-	100 MWth	technological irradiation reactor	under construction

Table 2 – List of research reactors in France

These facilities include:

- 4 critical mock-ups or teaching reactors: ÉOLE, MINERVE, MASURCA and ISIS.
- 6 research reactors: OSIRIS, PHÉBUS, CABRI, ORPHÈE, high-flux reactor – RHF and Jules Horowitz reactor – RJH.

These facilities are used for various purposes, such as the production of neutrons for research activities, the performance of tests, or materials testing. The RJH reactor is under construction on the Cadarache site: this new reactor will help cover research and development needs as well as produce artificial radionuclides for medical purposes in the light of the shutdown in the short to medium-term of the European irradiation reactors currently in service.

[3] New criticality after complete replacement of the reactor block (new authorisation decree).

[4] New criticality after relocation of the facility to the Cadarache platform.

Although considered to be in service in the administrative sense of the term, some of these facilities are shut down (pending decommissioning, or in prolonged outage):

- the two OSIRIS and PHÉBUS reactors are shut down with core unloaded pending decommissioning, with the decommissioning files to be submitted in 2018. The first operations in preparation for decommissioning (OPDEM) are in progress. In the short term (3 to 4 years), a change in administrative regime should thus take place in favour of the decommissioning regime. The OSIRIS and PHÉBUS reactors were not therefore included;
- the MASURCA critical mock-up is currently in prolonged outage. With a view to resuming operation of this facility, major modernisation[5] works are planned (construction of a new fuels store and reassessment of the safety of the existing structures in the light of current standards), it was not therefore included in this report;
- the ÉOLE, MINERVE, ISIS facilities, with a power output of less than 1MWth (between 0.001 and 0.7 MWth), were not included in this assessment. They will be shut down soon (end 2017 for the first two and mid-2019 for ISIS) in preparation for decommissioning.

The Orphée, Cabri, RHF and RJH facilities were thus included in this present assessment. A more detailed description of these 4 facilities is given in appendices 10.8 to 10.11.

1.1.5 LIST OF RESEARCH REACTORS SELECTED FOR THE ASSESSMENT

The reactors considered for this assessment are presented below, along with their main characteristics.

BNI	Name and location of the facility	Licensee	Thermal power	Date of 1st criticality	Comments
24	CABRI (Cadarache) 13115 Saint-Paul-lez-Durance	CEA	25 MWth	1963	Pool type reactor Test reactor for the study of reactivity accidents. Reactor shut down from 2006 to 2015: complete refurbishment of the building to incorporate a new experimental device (pressurised water loop). Cabri previously had a sodium loop. The Cabri facility is undergoing a periodic safety review (started in November 2017).

[5] This file is currently being examined by ASN.

BNI	Name and location of the facility	Licensee	Thermal power	Date of 1st criticality	Comments
67	HIGH-FLUX REACTOR (RHF) 38041 Grenoble Cedex	Max von Laue Paul Langevin Institute	58.3 MWth	1971	Pool type reactor with heavy water as moderator and reflector. Operates in cycles of about 50 d. Production of neutrons with different energy spectra, captured via 13 channels (thimbles). The neutrons produced are intended for fundamental research. The RHF reactor also takes part in the production of radionuclides for medical purposes. Facility temporary outage from 1991 to 1994 for replacement of the reactor block. The resumption of operations required a second authorisation decree (decree n° 94-1042 of 5 December 1994). Major reinforcement work following the Fukushima Daiichi accident (back-up systems, installation of a bunkerised emergency centre). The RHF facility is undergoing a periodic safety review (started in November 2017). A periodic safety review was conducted in 2002.
101	ORPHÉE (Saclay) 91191 Gif-sur-Yvette Cedex	CEA	14 MWth	1980	Pool type reactor with heavy water as moderator and reflector. Production of neutrons captured via channels (thimbles). The neutrons produced are intended for fundamental research. The Orphée reactor also takes part in the production of radionuclides for medical purposes. 2 periodic safety reviews conducted: 1997 and 2010. Significant work carried out following the 1st periodic safety review in 1997 more specifically with replacement of the core vessel and refurbishment of the I&C. Next periodic safety review in March 2019 Final shutdown of the facility planned before 2020.
172	JULES HOROWITZ (RJH) (Cadarache) 13115 Saint-Paul-lez Durance Cedex	CEA	100 MW	Under construction	Pool type reactor Materials testing reactor: testing and qualification of fuels and materials under irradiation. The RJH reactor will also take part in the production of radionuclides for medical purposes. Creation authorisation decree dated 12/10/2009. Commissioning scheduled for about 2021.

Table 3 – Detailed list of research reactors selected for this assessment

1.1.6 LOCATION OF RESEARCH REACTORS

The research reactors concerned by this assessment are located on the CEA sites at Saclay and Cadarache, as well as in Grenoble for the RHF (facility operated by the Laue-Langevin Institute).

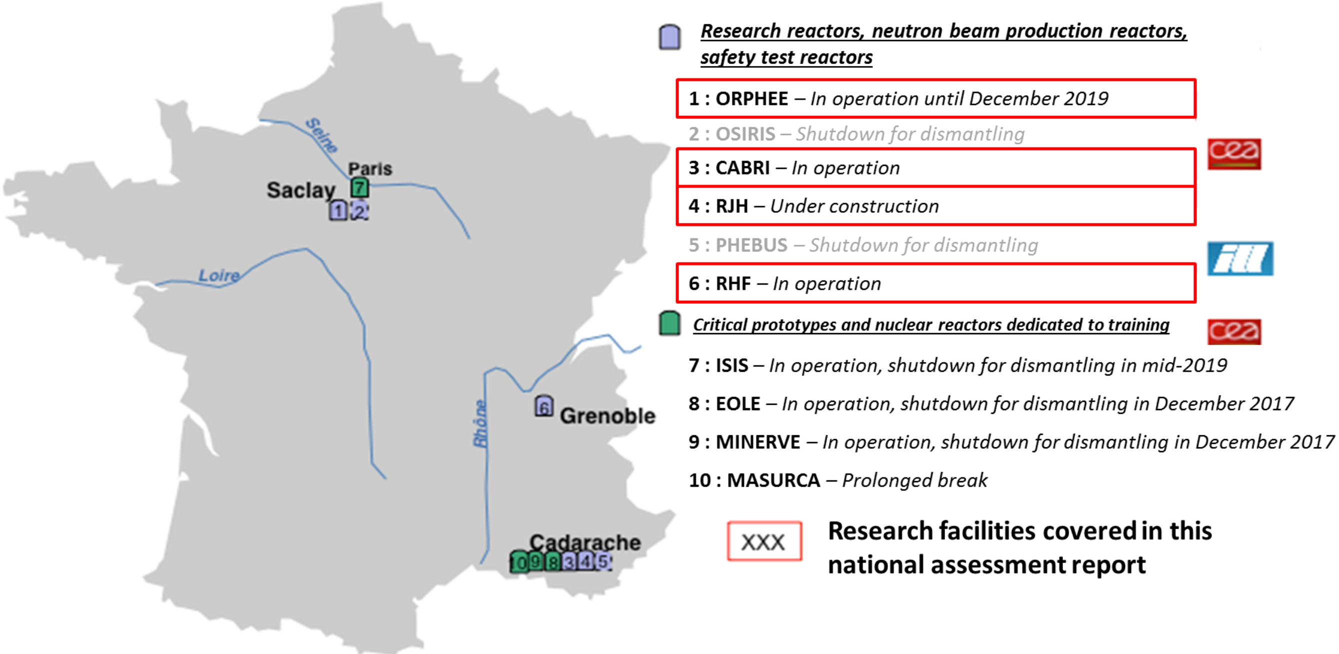

Figure 3 – Location of research reactors in service and under construction in France

1.2 PROCESS APPLIED FOR DRAFTING OF THE NATIONAL REPORT

This report was drawn up by the French nuclear safety regulator (ASN), which acted as coordinator, with contributions from the French Institute for Radiation Protection and Nuclear Safety (IRSN), the nuclear reactor licensees: Électricité de France (EDF), the French Alternative Energies and Atomic Energy commission (CEA) and the Laue-Langevin Institute (ILL).

The national report drafting process began in February 2017 when ASN sent the licensees (EDF, CEA, ILL) official notification of initiation of the assessment.

The contributions from the licensees were sent to ASN in mid-2017 and were reviewed by it over the summer of 2017. The draft report was validated by ASN as of November 2017.

The corresponding schedule for the exercise is schematically represented in the following figure:

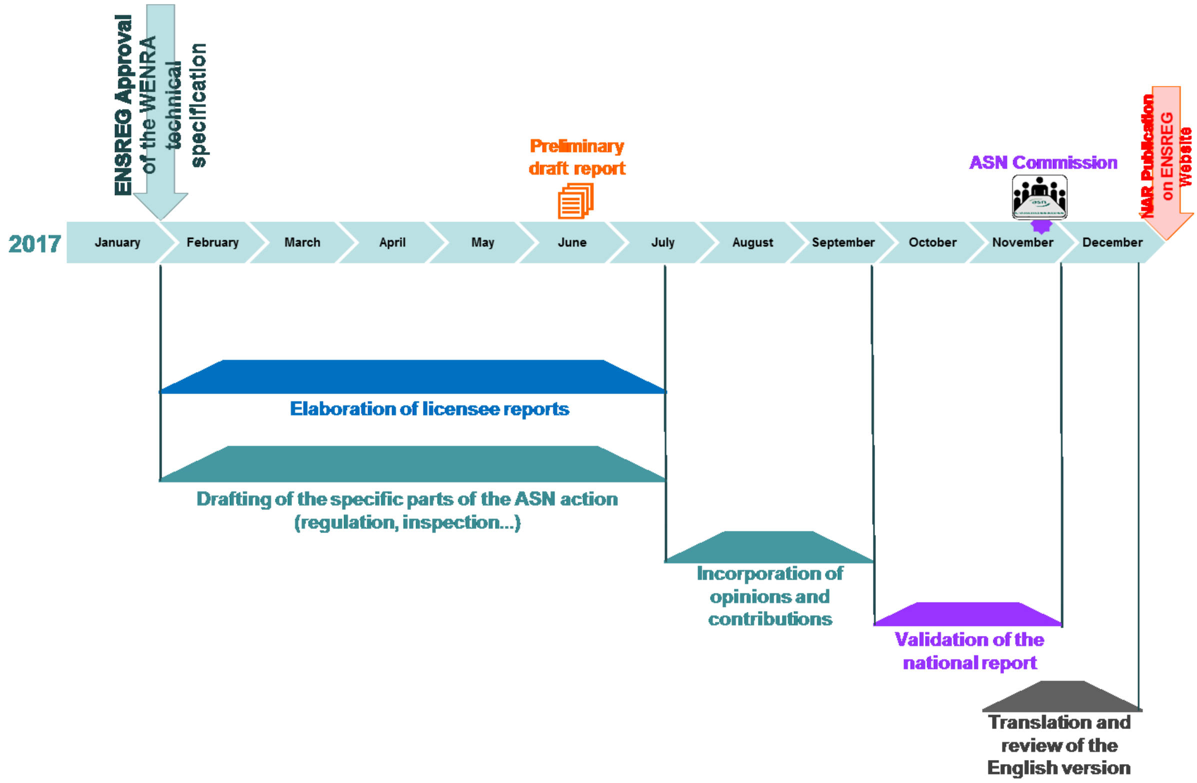

Figure 4 – Schedule for drafting of the national report

2 OVERALL AGEING MANAGEMENT PROGRAMME REQUIREMENTS AND IMPLEMENTATION

Summary:

The ageing management of nuclear facilities is incorporated into the French regulations, more specifically in:

1) the Environment Code, which includes the provisions relating to the periodic safety review process for all Basic Nuclear Installations,

2) orders which comprise provisions concerning the regulation of the design/construction or in-service monitoring of nuclear pressure equipment (NPE). It should be pointed out that the regulations make provision for partial requalification of the NPE every 5 years, as of the third ten-yearly outage,

3) ASN requests concerning licensee management of the ageing of its nuclear facilities and their potential continued operation beyond 40 years, in particular with regard to the nuclear power reactors,

4) The provisions of ASN-IRSN guide n° 22 "Pressurised Water Reactor design" applicable to the search for improvements to be made to existing reactors, which extend to non-NPE components the provisions of the order of 30 December 2015, concerning NPE, on the need to take account of equipment ageing as of the design stage.

French regulations set no time limit on the operation of the facilities.

As of 2001, EDF initiated the development and implementation of an ageing management programme within the framework of its reactor third ten-yearly outage inspections (VD3). This programme is primarily based on design, operating, in-service monitoring and routine maintenance measures, supplemented by exceptional maintenance.

EDF has adopted an approach to demonstrate the ageing management of the systems, structures and components (SSC) potentially affected. This approach is built around 4 steps:

• SSC selection process: The SSCs potentially susceptible to ageing and whose failure can have an impact on safety are identified,

• Individual analysis of ageing mechanisms: the ageing mechanisms for each SSC are identified and analysed, in order to check ageing management with regard to the operations and maintenance provisions in force. This step also comprises an analysis of the actual repairability and/or replaceability of the SSCs (contained in an Ageing Analysis Sheet (AAS)),

• Additional actions and studies: when ageing management cannot be demonstrated by means of normal operating provisions, additional actions or studies are identified for ageing management (contained in a component Detailed Ageing Analysis Report (DAAR),

• Drafting of an ageing analysis report specific to the reactor, known as the "UAAR" based on the generic AAS, and component DAARs, for each reactor nearing its VD3 and for the subsequent VDs.

ASN emphasises the fact that the EDF ageing management programme complies with the requirements of international standards. Even if it does not precisely follow the formal layout of the IAEA documents, this programme appropriately incorporates national and international operating experience feedback and is accompanied by a major R&D programme.

In the context of a continued operation of the NPPs, EDF proposes reusing this approach for the fourth ten-yearly outage inspections (VD4). This approach will be extended to all SSCs important for the management not only of radiological risks, but also conventional risks.

Ageing management of ***research reactors*** *is tailored to each individual facility. It takes account of the diversity of operating conditions and in particular the replaceability of parts which are sometimes produced in very small quantities. One specific feature of research reactors must be highlighted: practically all the equipment, except for the containment, can be replaced.*

Ageing management of the research reactors is currently based on the maintenance programmes and on periodic checks and tests. ASN considers that ageing management must be defined in a more formal manner by the research reactor licensees. In particular, the licensees must implement an approach for ensuring that the checks and tests performed are sufficient and, as necessary, define additional checks to ensure the ability of the equipment to perform its functions in the light of the ageing mechanisms that could affect it.

2.1 FRENCH REGULATORY FRAMEWORK FOR AGEING MANAGEMENT OF NUCLEAR FACILITIES

2.1.1 AGEING PHENOMENA

The phenomena linked to ageing must be taken into account in order to maintain a satisfactory level of safety for the entire operating life of the facilities.

Ageing management must be demonstrated, relying on operating experience feedback, the maintenance provisions and the possibility of either repairing or replacing the components. Quite apart from simply the time that has passed since its commissioning, other factors must be taken into account, in particular physical phenomena which can modify the characteristics of the equipment, depending on their function or their conditions of use. Consideration must therefore be given to the deterioration of replaceable items and the lifetime of non-replaceable items.

The periodic safety review is a particular opportunity for an in-depth examination of the effects of ageing on the equipment.

The regulatory requirements concerning ageing management are contained in the following texts:

- **the periodic review process contained in articles L. 593-18 and L. 593-19 of the Environment Code;**
- **article R. 557-14-2 of the Environment Code for nuclear pressure equipment (reactor vessels in particular);**
- **the provisions of the order of 30 December 2015 relative to nuclear pressure equipment ;**
- **article 2.5.1 of the order of 7 February 2012 setting the general rules for BNIs, requiring that *"design, construction, testing, inspection and maintenance provisions ensure that this qualification is maintained for as long as necessary";***
- **the provisions of the order of 10 November 1999 relative to the monitoring of operation of the main primary system and the main secondary systems of nuclear pressurized water reactors;**
- **ASN position statements (see below);**
- **the ageing management provisions of the *"PWR reactor design"* guide produced jointly by ASN and IRSN and published on 18 July 2017.**

Finally, for certain items, equipment or systems with particular ageing management implications, ASN may issue specific prescriptions. For example, specific technical prescriptions have been issued for the aseismic bearing pads on which the nuclear unit of the RJH research reactor is installed[6].

Equipment obsolescence management is also required to guarantee a satisfactory level of safety.

2.1.2 THE PERIODIC SAFETY REVIEWS

French regulations set no time limit on the operation of the facilities.

[6] ASN resolution 2011-DC-0226 of 17 May 2011

Ageing management is regulated by the process of periodic safety reviews carried out every ten years, pursuant to article L. 593-18 of the Environment Code.

Article L. 593-18 of the Environment Code

The licensee of a BNI periodically reviews its facility using the best international practices.

This review should allow an assessment of the facility's situation with regard to the rules that apply to it and an update of the assessment of the risks and detrimental effects of the facility for the interests mentioned in article L. 593-1, more specifically taking account of the status of the facility, the experience acquired during operation, changes in knowledge and in the rules applicable to similar facilities.

These reviews are held every ten years. However, the authorisation decree may set a different frequency if so warranted by the particularities of the facility. For facilities subject to Council Directive 2009/71/Euratom of 25 June 2009 establishing a Community framework for the nuclear safety of nuclear facilities, the interval between periodic safety reviews may not exceed ten years.

As applicable, the licensee may provide a separate report containing any information which it considers could compromise one of the interests mentioned in article L. 124-4 if divulged. With the exception of this reservation, the periodic safety review report may be communicated to any party pursuant to articles L. 125-10 and L. 125-11.

This arrangement is in response to the requirements of article 8c of European Council Directive 2014/87/Euratom of 8 July 2014 amending Directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear installations.

Article 8c – Initial assessment and periodic safety reviews – of Council directive 2014/87/EURATOM of 8 July 2014 modifying directive 2009/71/Euratom establishing a Community framework for the nuclear safety of nuclear facilities.

Member States shall ensure that the national framework requires that:

a) any grant of a licence to construct a nuclear installation or operate a nuclear installation is based upon an appropriate site and installation-specific assessment, comprising a nuclear safety demonstration with respect to national nuclear safety requirements based on the objective set in article 8a;

b) the licence holder under the regulatory control of the competent regulatory authority reassesses systematically and regularly – at least every ten years - the safety of the nuclear installation as laid down in article 6 c). That safety reassessment aims at ensuring compliance with the current design basis and identifies further safety improvements by taking into account ageing issues, operational experience, most recent research results and developments in international standards, using as a reference the objective set in article 8a.

The periodic safety review process is explained in article L. 593-19 of the Environment Code.

Article L. 593-19 of the Environment Code

The licensee sends ASN and the Minister responsible for nuclear safety a report containing the conclusions of the review set out in article L. 593-18 and, as applicable, the provisions it intends to take to remedy the anomalies observed or improve the protection of the interests mentioned in article L. 593-1.

After analysing the report, ASN may set new technical prescriptions. It notifies the Minister responsible for nuclear safety of its analysis of the report, along with the prescriptions it is using.

The provisions proposed by the licensee during the reviews beyond the thirty-fifth year of operation of a nuclear power reactor are, following a public inquiry, subject to ASN's authorisation procedure mentioned in article L. 593-15, without prejudice to the authorisation mentioned in II of article L. 593-14 in the event of a substantial modification. The ASN prescriptions comprise measures concerning the regular long-term monitoring of the condition of the equipment important for protection of the interests mentioned in article L. 593-1. Five years after submission of the review report, the licensee submits an interim report on the condition of this equipment, on the basis of which ASN may issue further prescriptions.

The periodic safety review is an opportunity to conduct an in-depth examination of the condition of the facilities, to check that they are in conformity with the applicable baseline safety requirements. Its aim is also to improve their level of safety.

Pursuant to article L. 593-18, the periodic safety review should be carried out taking account of best international practices, the condition of the facility, operational experience and developments in knowledge and in the rules applicable to similar facilities. The changes to rules applicable to ageing management are therefore taken into account.

The requirements applicable to the facility undergoing the periodic safety review are compared with those applicable to the most recent facilities and, more specifically for power reactors, with the significantly more demanding requirements of the EPR.

These requirements must also be reassessed in the light of operating experience feedback and research and development both in France and elsewhere, identifying areas for improvement, in particular for the prevention and management of severe accidents.

This position complies with the requirements of article 8c of the above-mentioned directive 2014/87/Euratom and is consistent with that expressed in 2014 by the WENRA[7] association concerning the safety of new NPPs, the objectives of which should be used as a reference for existing reactors.

[7] WENRA. Report – Safety of new NPP designs, 2013.

> **WENRA Report – Safety of new NPP designs**
>
> […]
>
> The safety objectives address new civil nuclear power plant projects. However, these objectives should also be used as a reference to help identify reasonably practicable safety improvements for "deferred plants" and existing plants during Periodic Safety Reviews.
>
> […]

The safety regulator members of WENRA issued a joint declaration in 2014[8] in which they made a commitment to ensuring that the new reference levels for existing reactors, published in 2014[9] and taking account of the lessons learned from the Fukushima Daiichi accident, are integrated into the national regulations.

The periodic safety review includes an examination of reactor conformity and ageing management and a safety reassessment.

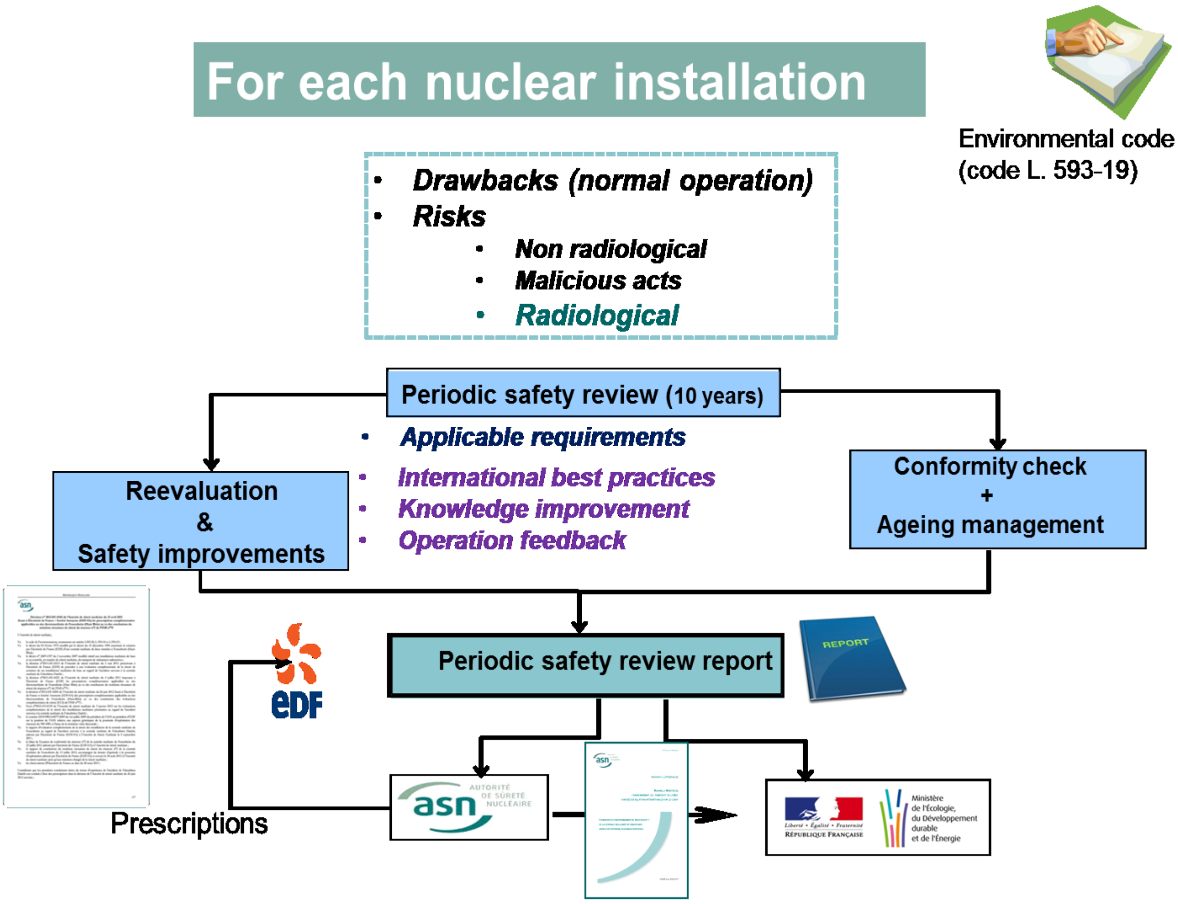

Figure 5 – The various subjects covered during a periodic safety review.

[8] WENRA. Statement regarding the revision of the SRLs for existing reactors taking into account the lessons learned from the TEPCO Fukushima Daiichi Nuclear Accident, 2014.

[9] WENRA, Report – Safety Reference Levels for Existing Reactors, 2014.

2.1.2.1 THE PERIODIC SAFETY REVIEWS OF EDF NUCLEAR POWER REACTORS

The ten-yearly inspections, which are lengthy outages, are ideal opportunities to implement the modifications resulting from the periodic safety review. To determine the calendar of reactor ten-yearly outage inspections, EDF takes account of the deadlines for performance of the containment tests and the pressure tests stipulated by the nuclear pressure equipment regulations and the ten-year frequency for the periodic safety reviews.

The EDF reactor periodic safety review process comprises a preliminary, or orientation phase, prior to the periodic safety review, during which EDF proposes the generic[10] programmes to check the status of the facility and of the safety reassessment, which will be the subject of an in-depth review. The generic programmes are the subject of an ASN position statement issued following consultation of the Advisory Committee for reactors (GPR) and possibly of the Advisory Committee for nuclear pressure equipment (GPESPN). On this basis, EDF carries out safety reassessment studies and defines modifications. Following its review and the consultations with the GPR/GPESPN, ASN gives its opinion on the results of these studies and on the modifications allowing the envisaged safety improvements.

Following the ten-yearly outage inspection of each reactor, the licensee sends ASN a periodic safety review conclusions report. In this report, the licensee states its position on the regulatory compliance of its facility as well as on the modifications made to remedy deviations observed or to improve the safety of the facility. ASN then informs the Minister responsible for nuclear safety of its analysis of the review conclusions report for each reactor, mentioned in Article L. 593-19 of the Environment Code and may issue new prescriptions regarding its continued operation.

The periodic safety review process, in conjunction with the ten-yearly outage inspections, is illustrated on the following figure:

[10] For a given type of reactor (900 MWe, 1300 MWe and N4) given the standardisation of the fleet of reactors in service.

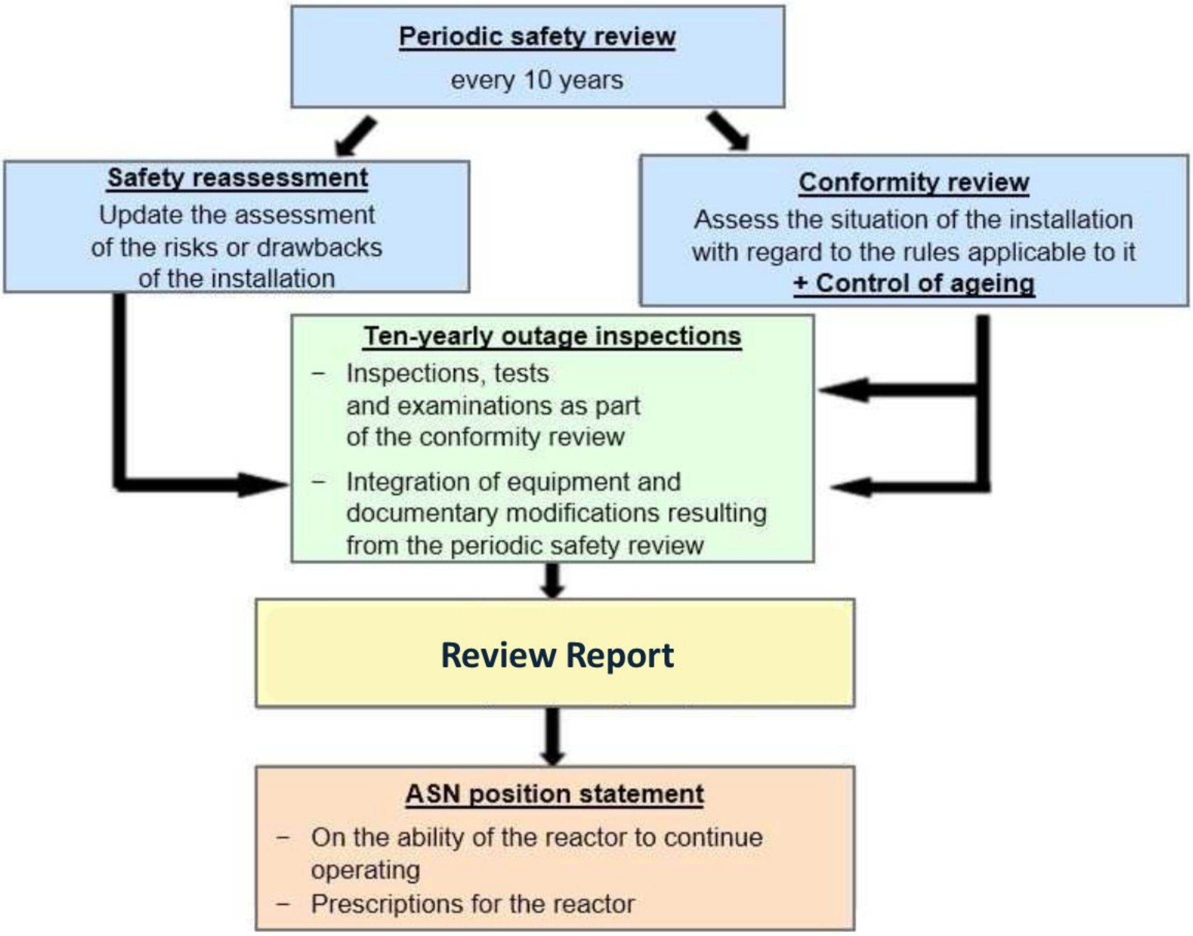

Figure 6 – Periodic safety review process for EDF nuclear reactors

The following table shows the progress of the 10-yearly outages (VD) for the reactors of the different standardised nuclear reactor plant series.

	VD1 10 years	VD2 20 years	VD3 30 years	VD4 40 years
900 MWe – 3 loops (34 units)	Done	Done	2009 to 2020	2019 to 2030
1300 MWe – 4 loops (20 units)	Done	Done	2015 to 2024	2025 to 2034
N4 – 4 loops (4 units)	Done	2019 to 2022	2029 to 2032	2039 to 2042

Table 4 – Ten-yearly outage inspections of the EDF nuclear reactor plant series

Over the 2013-2015 period, the main projects concerned the 900 MWe reactors (performance of the third 10-yearly outage inspections (VD3) and preparation of the fourth 10-yearly outage inspections (VD4)), the 1300 MWe reactors (performance of the VD2, preparation and first VD3) and the 4 reactors of the N4 series (preparation of the VD2).

As at the end of 2017, VD3 have been carried out on 30 of the 34 reactors of the 900 MWe series. VD2 have been carried out on the 20 reactors of the 1300 MWe series and the VD3 have started for the 1300 MWe reactors (4 out of 20). The VD1 have been carried out on all the N4 reactors.

2.1.2.2 PERIODIC SAFETY REVIEWS OF RESEARCH REACTORS

Since November 2007 (publication of decree 2007-1557 of 2 November 2007), the periodic safety review obligation has been extended to nuclear facilities other than nuclear power reactors. The deadline for performance of this review is set for no later than November 2017. However, certain facilities considered to have "special implications", had already undergone a periodic safety review before 2007; this is the case with the RHF, Cabri and Orphée reactors.

It should be pointed out that each research reactor is unique and has specific risks, so there is no generic file as is the case with the nuclear power reactors.

The periodic safety review is an important tool in research reactors ageing management. When assessing the conformity of the facility, the licensee may need to identify work to be performed to ensure compliance with its baseline requirements.

Some of these facilities, more particularly the test reactors, may undergo significant changes during operation, in order to incorporate new test devices[11]. These significant changes are to be considered in terms of ageing management.

BNI N°	Name	Reviews performed	Date of next review
24	CABRI	Partial review in 2004 (with examination of the substantial modification)	11/2017
67	RHF	2002	11/2017
101	Orphée	1997, 2010	31/03/2019
172	RJH	-	To be defined after commissioning, according to article 24 of the amended decree of 02/11/2007

Table 5 – Periodic safety reviews of research facilities concerned by this report

2.1.3 PROVISIONS CONCERNING AGEING MANAGEMENT OF NUCLEAR PRESSURE EQUIPMENT

2.1.3.1 THE PROVISIONS OF THE ORDER OF 30 DECEMBER 2015 RELATIVE TO NUCLEAR PRESSURE EQUIPMENT

Article R. 557-14-2 of the Environment Code notably requests that *"The equipment is maintained in good condition and checked as often as necessary"*. The order of 30 December 2015 concerning nuclear pressure equipment (NPE) details this objective by first of all making provision for ageing management measures as of the design of the equipment.

[11] For example, significant work has recently been carried out on the Cabri facility (2006-2015) to install a new test device (pressurised water loop).

Therefore, as early as the design stage, the NPE manufacturer must take account of alteration of the materials over time and of the ageing phenomenon, in particular irradiation-induced ageing.

Appendix 1 of the order of 30 December 2015
relative to nuclear pressure equipment

2. Design

The equipment is designed so as to minimise the risk of loss of integrity in the light of the foreseeable alterations to the materials.

The design is based on measures such as to mitigate the risk of failure and a calculation method aiming to verify that the design can indeed guarantee the required level of safety. These measures are taken in order to reduce the risks linked to:

- oligocyclic thermal fatigue or to a large number of cycles;
- the different thermal behaviours of materials welded together;
- vibration fatigue;
- local pressure peaks;
- creep;
- stress concentrations;
- corrosion phenomena;
- harmful local thermal-hydraulic phenomena;
- equipment drainage in the event of a pipe break.

The calculation method may be supplemented by an experimental design method.
The design takes account of irradiation-induced ageing.

This order then asks the NPE manufacturer to meet a level of manufacturing quality consistent with the importance of the NPE built, more specifically with regard to the production of the materials used in the various components and the methods of assembling these components. The purpose of these manufacturing provisions is to minimise the risk of flaws forming in service, as a result of mediocre manufacturing conditions.

This same order specifies that the NPE manufacturer must notify the licensee of all information allowing operation of the equipment in the conditions specified for its design.

Appendix 1 of the order of 30 December 2015
relative to nuclear pressure equipment

3.7. Service instructions

Pressure equipment is accompanied by an instruction manual.
The instruction manual gives the particular design characteristics that are decisive factors in the service lifetime of the equipment.

These characteristics comprise at least:

- for creep, the theoretical number of operating hours at specified temperatures;
- for fatigue, the theoretical number of cycles at specified stress levels;
- for corrosion phenomena, the thickness allowance or corrosion protection characteristics;
- for thermal ageing, the theoretical number of hours of operation at specified temperatures;
- for irradiation-induced ageing, the maximum theoretical flux at given irradiation temperatures.

This information must appear in an instruction manual specifying the phenomena taken into consideration at the design stage and which must not be exceeded during operation, along with the operating and ageing conditions.

Finally, this order defines the in-service monitoring conditions for the NPE with a relatively high risk, on the basis of the quantity and nature of the fluid contained (harmfulness, activity) and its pressure.

For equipment belonging to the main primary system and main secondary systems of nuclear power reactors, the order of 30 December 2015 relative to nuclear pressure equipment makes reference to the order of 10 November 1999 concerning the main primary system and main secondary systems of pressurised water reactors, detailed below.

For the NPE (excluding main primary system and main secondary systems) with a relatively high level of risk, the in-service monitoring procedures are determined by appendices 5 and 6 of the order of 30 December 2015 relative to nuclear pressure equipment. These procedures are based on the following three principles:

1. for each equipment item, the licensee must draft a programme of maintenance and monitoring operations, the aim of which is to manage the possible deterioration of the equipment considered in order to prevent it from failing; the licensee is required to keep this programme up to date;

> **Appendix 5 of the order of 30 December 2015**
> **relative to nuclear pressure equipment**
>
> 2.4 The licensee updates the programme of maintenance and monitoring operations whenever necessary, based on the actual usage of the equipment, possible changes to it during operation, in particular the properties of materials and flaws and degradations observed, as well as operating experience feedback and the result of periodic requalifications.

2. the equipment with the highest risk in terms of fluid contained and fluid pressure must systematically be periodically inspected by the licensee (every 40 months) with periodic requalification by an independent organisation (every ten years):
   a. the periodic inspections consist of an examination of the outer and inner walls of the pressurised compartments and verification of the operation of valves;
   b. the periodic requalifications consist of the same checks as the periodic inspections, plus containment pressure tests of the pressurised compartments;
3. equipment repairs and modifications are carried out in accordance with the rules in force for the design and manufacture of new equipment.

The regulations authorise certain adaptations of the in-service monitoring rules. It is sometimes difficult to carry out internal inspections and pressure tests on equipment not originally designed to meet relatively recent regulatory requirements (January 2006). In this case, the licensee takes compensatory measures and carries out more frequent and/or more precisely targeted monitoring.

2.1.3.2 THE PROVISIONS OF THE ORDER OF 10 NOVEMBER 1999 RELATIVE TO THE MAIN PRIMARY SYSTEM AND THE MAIN SECONDARY SYSTEMS OF NUCLEAR PRESSURIZED WATER REACTORS

The order of 10 November 1999 relative to the monitoring of operation of the main primary system and the main secondary systems of pressurized water nuclear reactors requires monitoring of the NPE making up these systems. The systems are called “equipment” in the order.

This monitoring comprises:

- **periodic equipment monitoring programmes to verify the absence of defects or, if manufacturing defects are indeed found, to check that they do not develop;**
- **a programme to monitor the degradation modes of the properties of the materials;**
- **a precise documentary system precisely identifying the actions to which the equipment has been subjected and indicating all the observations liable to affect its maintained integrity.**

In-service monitoring of the main primary system and the main secondary system is regulated by articles 14 and 15 of the order of 10 November 1999. The in-service monitoring provisions are stipulated in article 14.

Article 14 of the order of 10 November 1999

Without prejudice to the provisions of articles 12 and 13, the licensee uses in-service monitoring and appropriate verifications and maintenance to ensure that the equipment and accessories, in particular the regulating and letdown, overpressure and isolation protection devices, remain in good condition at all times and are able to perform their functions in normal and accident conditions.

The licensee has a periodic inspection of the equipment carried out, called the partial inspection, without the interval between two inspections exceeding two years after the first complete inspection for the main primary system and forty months for the main secondary system.

The licensee ensures that the equipment and accessories can be inspected in acceptable conditions of radiation protection and safety for the personnel concerned and, if not, shall define the necessary compensatory measures in good time.

It shall draw up a detailed report for each inspection mentioning the examination processes used, the observations made and more specifically the flaws detected and the steps taken as a result thereof. This report is kept at the disposal of the regional director for industry, research and the environment with local responsibility and a summary is sent to them before each equipment restart.

Insofar as they are required, the files mentioned in article 4 (II, d) and 4 (II, e) specify the monitoring conditions and the programme of partial inspections.

Operations regarding periodic requalification are regulated by article 15 of the order of 10 November 1999. They more specifically include a complete inspection and a pressure test. The maximum interval of ten years between two periodic inspections is consistent with the time interval between two periodic safety reviews.

As of the third ten-yearly outage inspection, the regulations require partial requalification comprising an in-depth inspection of the equipment five years after the ten-yearly outage inspection, this being applicable to all subsequent ten-yearly outage inspections.

Article 15 of the order of 10 November 1999

I. - The equipment is required to undergo periodic requalification. In this respect, at the diligence of the licensee, each equipment item undergoes complete requalification comprising a complete inspection performed under the supervision of the licensee, a hydro-test and examination of the safety devices performed under the supervision of the licensee. The first complete requalification of the reactor coolant system is performed no later than thirty months after the first fuel loading. The first complete requalification of each main secondary system is performed no later than ten years after the last pressure test of the corresponding steam generator. The maximum interval between two complete requalifications is set at ten years, without prejudice to application of article 16, barring a waiver granted by the Prefect of the département concerned in the light of conclusive data and not to exceed one year.

II. - The complete inspection is in principle performed during the reactor outage required for performance of the test, but some of the operations involved may, subject to the observations of the regional director for industry, research and the environment with local responsibility, be carried out during prior inspections if they do not predate the pressure test by more than two years. Insofar as it is required, the file mentioned in article 4 (II-e) specifies the procedures for the complete inspection. The licensee draws up a detailed complete inspection report mentioning the processes used, the observations made and in particular the flaws detected and the measures taken as a result thereof. This report is presented to the regional director for industry, research and the environment with local responsibility before the pressure test. With the approval of the regional director for industry, research and the environment with local responsibility, some inspections may however be carried out before the pressure test and before the equipment is returned to service.

III. - The pressure test of each equipment item takes place in the presence of a representative appointed by the regional director for industry, research and the environment with local responsibility. The test pressure is at least equal to 1.2 times the design pressure of the equipment item concerned. The test must be performed with no serious defect and no significant leak.

- In the event of an unacceptable risk for the personnel tasked with the inspection during the pressure test, alternatives to visual inspection shall be used after prior qualification in the conditions stipulated in article 8.
- An examination of the safety systems is carried out after the pressure test, under the supervision of the licensee, in order to ensure that they remain effective. The results are sent to the regional director for industry, research and the environment.
- The parts of the equipment items situated downstream of the component of the last isolation device which performed effective isolation may not be subjected to pressure during the pressure test.

IV. - Partial requalification, limited to an in-depth inspection under the supervision of the licensee and for which the programme is communicated beforehand to the regional director for industry, research and the environment with local responsibility, is carried out in the following cases:

- on the pressure-resistant replaced parts of the main primary system, no later than thirty months after this replacement;
- after the occurrence of an event which could correspond to a third-category situation, on the equipment item(s) affected;
- between four and six years after each complete requalification for equipment in service for more than thirty years, without prejudice to the revision of the content of the inspection pursuant to article 5 of this order . […/..]

Article 15 of the order of 10 November 1999 (cont.)

IV. […]

Partial requalification, comprising containment pressure tests but for which the inspection programme is limited with the consent of the regional director for industry, research and the environment with local responsibility, is performed:

- after a major intervention as defined in article 10-I;
- following application of article 16;
- no later than thirty months after the replacement of a part of the main primary system by a primary component with characteristics that are significantly new with regard to its utilisation in an NSSS of the type considered.

V. - If the results of a requalification comprising containment pressure tests are satisfactory, the regional director for industry, research and the environment draws up a requalification report in duplicate for the equipment item considered and submits one copy to the licensee. The report constitutes confirmation of requalification of the equipment item.

If the results of a requalification are not considered to be satisfactory, the procedure stipulated in the last section of article 16 may be initiated.

Articles 11 and 12 of the order of 10 November 1999 contain provisions for monitoring the chemical conditions which could have an impact on the equipment and provisions for monitoring degradation phenomena of the properties of the materials.

Article 11 of the order of 10 November 1999

I. – The licensee ensures that the following are adequate with regard to the corrosion risks:

- the composition of the reactor coolant fluid and secondary fluid;
- prior to implementation, the processes used for conditioning during outage, cleaning and any decontamination of equipment;
- the tools and fluid used during interventions,

also taking account of their impact on radiation protection.

II. – The licensee defines and updates the concentration limits of the necessary chemical species, in order to prevent and, failing which, to mitigate the corrosion damage.

Article 12 of the order of 10 November 1999

I. – The licensee implements the necessary means for understanding the in-service changes to the properties of the materials making up the equipment and having an impact on their maintained integrity.

It implements particular monitoring for each degradation mode of the properties of the materials, identified at the design stage and liable to significantly compromise the initial values of the properties of the materials included in the equipment strength demonstrations. This monitoring also concerns the degradation modes discovered during operation.

It sends the regional director for industry, research and the environment with local responsibility - with a copy to the director for the safety of nuclear installations - the main results of this monitoring and its conclusions with regard to the maintained integrity of the equipment and its serviceability for the next ten years.

II. – For those materials subjected to and susceptible to them, the degradation modes studied include irradiation-induced embrittlement, forms of thermal ageing and the main corrosion modes in the conditions consistent with the provisions of article 11.

Finally, article 7 provides for a documentary record of the findings made on the pressure equipment liable to affect their integrity and the actions to which they were subjected.

Article 7 of the order of 10 November 1999

I. – The licensee ensures that the operating conditions of the equipment item remain at all times compatible with the technical demonstrations provided concerning its strength. It performs tests and draws up the necessary instructions accordingly.

II. – The licensee has a documentary system enabling the observations liable to concern the maintained integrity of the equipment to be easily identified, along with their date, in particular:

- the observations made during the equipment pre-service inspection as stipulated in article 9;

- the observations made during the inspections stipulated in articles 14 and 15;

- the operating incidents, in particular activation of over-pressure protection devices and situations encountered that are potentially more severe than those of the second category;

- the major and notable interventions defined in article 10;

- the monitoring results defined in article 12;

- the accounting of the situations on the main primary system and in the zones of the main secondary system subjected to significant cyclic loadings.

The licensee shall take care to conserve the documents which could at a later date shed light on the actions carried out on the equipment items.

These documents are kept at the disposal of the regional director for industry, research and the environment with local responsibility.

The regional director for industry, research and the environment with local responsibility shall be directly informed of events liable to compromise the integrity of the equipment items.

2.1.4 ASN POSITION STATEMENTS

2.1.4.1 EDF METHODOLOGY FOR AGEING MANAGEMENT IN THE CONTEXT OF THE THIRD TEN-YEARLY OUTAGE INSPECTIONS (VD3) ON THE NUCLEAR POWER REACTORS

Until 2001, questions concerning the operating lifetime were covered by the "Lifetime" (PDV) project set up in 1985, and by the "replaceability" fleet issue.

This project was assessed by ASN following EDF's transmission of a progress report in 1991. This assessment led to ASN requests in 1993[12] subsequently supplemented in 1996[13].

In order to improve the overall consistency of the approaches adopted by EDF, ASN asked it - in letter DSIN-GRE/SD2/34-2001 of 19 February 2001 – to set up a structured and coherent ageing management programme in preparation for the first VD3 of the 900 MWe reactors, beginning in 2008.

In this letter, ASN specified the principles on which this approach was to be based:

- concerning the identification and processing of sensitive components:
  - updating of the list of SSCs whose ageing can affect reactor safety and determine its operating lifetime;
  - definition of the parameters associated with the ageing mechanisms identified for these SSCs and which, if exceeded, would require specific action (repair, replacement, modification, change in environmental or operating conditions);
  - production of the repair and replaceability file;
- concerning monitoring actions:
  - a detailed experience feedback analysis, based on the information resulting from the maintenance and monitoring of the facilities, as well as on the results of the VD2, should make it possible to deal with degradations linked to ageing mechanisms which were not anticipated owing to the complexity of the phenomena;
- concerning R&D:
  - studies must be carried out on the ageing phenomena and their development kinetics, taking account of the actual environmental and operating conditions.

ASN asked that this programme be implemented, within the framework of the VD3, with a continued operability file being produced for each reactor along with a detailed ageing management programme beyond the VD3.

In response to these ASN requests, EDF proposed an approach based on the creation of:

- ageing analysis sheets (AAS): the AAS are produced following the survey of the components important for safety for which an ageing mechanism could compromise the safety of the facilities. These AAS also take account of adaptation of the operating or maintenance provisions or difficulties with repair or replacement;
- Detailed Ageing Analysis Reports issued for the SSCs (component DAARs) for which at least one AAS demonstrated that they were susceptible to ageing: for example, the vessel internals, the K1

[12] Letter DSIN/Paris n° 1520/93 : "*Durée de Vie*" Project of PWR.

[13] ASN, Letter DSIN/GRE/SD2.5/n° 420/96.

electrical cables located in a “hot spots” atmosphere, containment electrical penetrations, I&C, or reactor containments are the subject of a component DAAR;

- Detailed Ageing Analysis Report specific to each reactor undergoing a VD3.

This approach (AAS, component DAAR, reactor UAAR) was examined during two meetings of the GPR, in 2003 and in 2006. ASN returned its opinion on this approach in its letter DEP-SD2-0424-2006.

2.1.4.2 OPERATING LIFE EXTENSION OF THE NUCLEAR POWER REACTORS BEYOND 40 YEARS

In 2009, EDF informed ASN that it wished to extend the operating life significantly beyond 40 years and to maintain open the option of an operating life of 60 years for all reactors. EDF sent ASN the proposed corresponding generic programme comprising the following aspects: the methodology proposed, the main safety objectives and the topics to be addressed. This programme will be implemented during the periodic safety reviews associated with the VD4 for the 900 and 1300 MWe series reactors.

More specifically with regard to ageing management, EDF wishes to reuse the approach implemented for the VD3 and which has already been examined by ASN (see above). With the support of IRSN, ASN examined this generic programme and consulted the GPR in 2012. For ASN, management of the ageing of the facilities, in particular of equipment whose integrity is essential for safety (such as the reactor vessel and containment) is essential for maintaining a satisfactory long-term level of safety. Following this examination, ASN issued its opinion in its letter CODEP-DCN-2013-013464 of 28 June 2013. ASN considered that the approach applied by EDF since the third ten-yearly outage inspections of the 900 MWe reactors could be reused. However, it more specifically asked EDF:

- to complete the identification of the ageing mechanisms in the light of national and international experience feedback and of the appropriate R&D programmes, taking account of the increased operating life beyond forty years being asked for by EDF;
- to provide a robust demonstration of the mechanical strength of the vessels beyond their fourth ten-yearly outage inspection;
- to identify the possible vulnerabilities in the components industrial replacement processes, including in the case of an unforeseen operational event on the reactors, and propose steps to improve the robustness of these processes;
- to reinforce EDF ability to verify compliance and, if necessary, restore it.

In preparation for the VD4 900 periodic safety review, EDF readopted the ageing management approach applied since the third periodic safety review of these reactors, while reinforcing its equipment renovation and replacement projects with a view to continued operation up to 60 years. EDF also supplemented its file with the answers to the above-mentioned ASN requests.

Following the IRSN review of EDF’s generic programme with a view to the VD4 900 periodic safety review and the consultation of the GPR in April 2015 and the GPESPN in June 2015, ASN issued its opinion in April 2016[14] regarding the guidelines and the additions to be made to its programme by EDF for the VD4 900 periodic safety reviews, with a view to its operational implementation, in particular with regard to management of the ageing and obsolescence of the SSCs of these reactors.

ASN thus noted that EDF had set up an organisation for identifying the various equipment degradation modes, adopting the relevant countermeasures and taking account of operating experience feedback.

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[14] ASN, Letter CODEP-DCN-2016-007286 of 20 April 2016.

However, with regard to the nuclear pressure equipment on the main primary system and the main secondary systems, it considers that the programme of work relating to ageing management of this equipment needs to be supplemented by additional studies and inspections, more particularly with regard to the mechanical strength of the vessels, the consideration of environmental effects on the mechanical fatigue phenomenon and changing material properties.

For the other PWR equipment, ASN considers that the steps EDF intends to take in the guidelines for the VD4-900 periodic safety review, in order to manage the ageing and obsolescence of the 900 MWe reactors up until the next periodic safety review, are on the whole satisfactory. However, it considers that additional steps are required, more particularly to evaluate the need for exceptional maintenance operations.

2.1.4.3 ASN GUIDE N° 22 ON THE DESIGN OF PRESSURISED WATER REACTORS

Guide n° 22 "Design of pressurised water reactors" published on 18 July 2017, was produced jointly by ASN and IRSN. It presents the ASN and IRSN recommendations for the design of pressurised water reactors and is intended for future PWR licensees in France, who are responsible for managing the risks and drawbacks of their facility in accordance with article L.593-6 of the Environment Code.

Its article 1.3 states that *"As the recommendations of this guide apply primarily to the design of new PWRs, they may also be used, for reference, to seek improvements to existing reactors, for example on the occasion of their periodic safety reviews, in accordance with article L. 593-18 of the Environment Code and articles 8a and 8c introduced by the European Directive of 8 July 2014."*

Part IV.2.5 specifically deals with taking account of industrial practices, maintenance, in-service monitoring and the constraints relative to their ageing in the design of EIPs. Article 4.2.5.3 states that *"measures must be taken at the design stage to facilitate monitoring of the planned ageing mechanisms and to detect any deterioration or unexpected behaviour that could arise during operation of the BNI."*

2.2 INTERNATIONAL STANDARDS

2.2.1 WENRA REFERENCE LEVELS

France is an active participant within WENRA and its working group, the RHWG[15]. It thus contributed to the drafting of the reference levels for existing reactors, the latest update of which was in September 2014[16].

In a joint declaration in 2014[17], the safety regulator members of WENRA made a commitment to ensuring that the reference levels for existing reactors – modified or added in the 2014 version – are incorporated into the national regulations before 2017. A peer review exercise was carried out within the WENRA RHWG to assess the action plans put into place by the various States in order to meet this objective.

All 2014 WENRA reference levels for existing reactors will therefore eventually be considered during reviews or inspections by means of their transposition into national regulations.

Topic I of the WENRA reference levels is devoted to ageing management. The topic I reference levels are explained below.

The reference levels prescribe the existence of an ageing management programme based on:

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[15] RHWG: Reactor Harmonization Working Group.

[16] WENRA. Report – Safety Reference Levels for existing Reactors, 2014.

[17] WENRA. Statement regarding the revision of the SRLs for existing reactors taking into account the lessons learned from the TEPCO Fukushima Daiichi Nuclear Accident, 2014.

- **identification of the ageing mechanisms to which the SSC are subjected that is as exhaustive as possible;**
- **monitoring and inspection activities;**
- **and regular reassessment of this programme in the light of new information that has become available.**

I1.1	The operating organisation shall have an Ageing Management Programme[18] (AMP) to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs.
I2.1	The licensee shall assess structures, systems and components important to safety taking into account relevant ageing and wear-out mechanisms and potential age-related degradations in order to ensure the capability of the plant to perform the necessary safety functions throughout its planned life, under design basis conditions.
I2.2	The licensee shall provide monitoring, testing, sampling and inspection activities to assess ageing effects to identify unexpected behaviour or degradation during service.
I2.3	The Periodic Safety Reviews shall be used to confirm whether ageing and wear-out mechanisms have been correctly taken into account and to detect unexpected issues.
I2.4	In its AMP, the licensee shall take account of environmental conditions, process conditions, duty cycles, maintenance schedules, service life, testing schedules and replacement strategy.
I2.5	The AMP shall be reviewed and updated as a minimum with the PSR, in order to incorporate new information as it becomes available, to address new issues as they arise, to use more sophisticated tools and methods as they become accessible and to assess the performance of maintenance practices considered over the life of the plant.
I3.1	Ageing management of the reactor pressure vessel[19] and its welds shall take all relevant factors including embrittlement, thermal ageing, and fatigue into account to compare their performance with prediction, throughout plant life.
I3.2	Surveillance of major structures and components shall be carried out to timely detect the inception of ageing effects and to allow for preventive and remedial actions.

Table 6 – WENRA reference levels for ageing management

To date, level I3.1, devoted to ageing management of the vessel and its welds, is transposed into the regulations by the order of 10 November 1999 relative to the monitoring of operation of the main primary system (CPP) and the main secondary systems (CSP) of nuclear pressurized water reactors.

Level I2.3 is taken into account by article L593.18 of the Environment Code. With regard to the other levels, they are taken into account "semi-officially" through the various ASN position statements presented in section 2.1.4.

[18] Ageing is considered as a process by which the physical characteristics of a structure, system or component (SSC) change with time (ageing) or use (wear-out). An Ageing Management Programme (AMP) should be understood as an integrated approach to identifying, analysing, monitoring and taking corrective actions and document the ageing degradation of structures, systems and components.

[19] Or its functional equivalent in other designs.

2.2.2 THE IAEA STANDARDS

Within the IAEA, ASN actively participates in the work of the Commission of Safety Standards (CSS) which draws up international standards, notably for the safety of nuclear facilities. In this capacity, it is a member of the NUSSC[20] committee which drafted the following documents dealing with ageing management:

For NPP reactors:

- **IAEA Safety Guide NSG-2.12 (DS485, 7 July 2015).** *Ageing Management and Development of a Programme for LTO of Nuclear Power Plants. 2015.*
- **IAEA Safety Reports Series n°82.** *Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL). 2015.*
- **IAEA Safety Standards SSR-2/1.** *Safety of Nuclear Power Plants: Design. 2012.*
- **IAEA Safety Standards SSR-2/2.** *Safety of Nuclear Power Plants: Commissioning and Operation. 2011.*
- **IAEA Safety Standards SSG-25.** *Periodic Safety Review for Nuclear Power Plants. 2013.*
- **IAEA Safety Standards.** *Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants. 2006.*

In conjunction with the document "*Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL)"*, mentioned above, the IAEA launched the IGALL programme in 2010 and it today comprises 120 experts from 23 countries. France plays an active role in it. The goal of this programme is to develop and maintain a documentary base concerning ageing management of the SSC important for safety. The IGALL database at present comprises 76 AMPs, including the following[21]:

- **IAEA IGALL AMP201.** *Insulation Materials for Electrical Cables and Connections not subject to Environmental Qualification Requirement.*
- **IAEA IGALL AMP202.** *Insulation Materials for Electrical Cables and Connections not subject to Environmental Qualification Requirement Used in Instrumentation Circuits.*
- **IAEA IGALL AMP210.** *Condition Monitoring of Electrical and I&C Cables subject to Environmental Qualification Requirements.*
- **IAEA IGALL AMP209.** *Ongoing Qualification of Electrical and I&C Components Relevant to an Environmental Qualification.*
- **IAEA IGALL AMP125.** *Buried and underground piping and tanks.*
- **IAEA IGALL AMP118.** *Reactor vessel surveillance.*

For research reactors:

- **IAEA Safety Standards SSG-10.** *Ageing Management for research reactors.*

2.3 DESCRIPTION OF EDF OVERALL AGEING MANAGEMENT PROGRAMME

2.3.1 SCOPE OF THE OVERALL AGEING MANAGEMENT PROGRAMME

[20] "Nuclear Safety Standards Committee".

[21] This list corresponds to the IGALL of the SCCs which are dealt with in this thematic assessment.

The technical reference baselines in the field of ageing are first of all drawn up per plant series and per status (VD2, VD3…), and are then implemented for each reactor.

The tasks relating to the ageing management process are distributed among the national design and operation engineering centres and the local engineering centres, with the support of EDF R&D:

- UNIE: operations engineering
- SEPTEN: design engineering (doctrine, technical baseline)
- DIPDE: nuclear island design engineering
- CNEPE: conventional island design engineering
- CEIDRE: materials appraisal, manufacturing, NDI/NDT (non-destructive inspection/non-destructive testing), chemistry
- UTO: spare parts, exceptional operational and anticipated maintenance (assurance files)
- NPP: local reactor engineering

The EDF process for Ageing Management of systems, structures and components (SSC) is presented in a specific process notice based on an "Ageing management of PWR reactors" methodology guide.

The purpose of the Ageing Management process is as follows:

- to demonstrate ageing management of the Systems, Structures and Components (SSC) whose ageing can affect reactor safety and thus determine their operating lifetime;
- to identify the additional steps to be taken to ensure SSC ageing management.

Ageing management is guaranteed through design, operating, in-service monitoring and routine maintenance measures, supplemented by exceptional maintenance, which allow:

- design and operation of reactors while mitigating, delaying or eliminating the consequences of ageing mechanisms;
- the prediction or sufficiently early detection that an SSC can be degraded to the extent that it can no longer perform its function, thus helping to ensure the overall safety of the facility;
- definition of countermeasures to prevent the appearance of these degradations and if necessary take corrective measures (including repair or replacement) to ensure the level of safety.

2.3.1.1 STRUCTURE OF THE PROCESS

To demonstrate ageing management of the Systems, Structures and Components (SSC) whose ageing can affect reactor safety and thus determine their operating lifetime, a 4-step approach has been adopted:

- selection of the SSC potentially susceptible to ageing and whose failure can have an impact on safety;
- drafting of the list of pertinent SSC/ageing mechanism combinations. Each pertinent SSC/ageing mechanism combination is analysed as shown by an AAS (Ageing Analysis Sheet) the aim of which is to verify the degree of ageing management in the light of the operating and maintenance provisions in force, along with the repairability and replaceability conditions;
- for each component or structure potentially susceptible to ageing, the failure of which could have an impact on safety and for which ageing management cannot in principle be demonstrated by routine operating provisions, the production of a component Detailed Ageing Analysis Report (DAAR), comprising the analysis of the ongoing or scheduled actions, designed to manage ageing and define the additional actions or studies to be carried out to this end;

- for each reactor undergoing VD3 (third ten-yearly outage inspection) and the subsequent ten-yearly outage inspection (VD), drafting of an ageing analysis report specific to the reactor, known as the UAAR, based on the AASs and component DAARs. This UAAR comprises the Local Ageing Management Programme which is to be implemented during the ten-year period following the VD.

These 4 steps correspond to the 4 sub-processes of the ageing management process:

- SP1: inventory of the SSC to be considered, per plant series;
- SP2: drafting and review of the AASs, per plant series;
- SP3: drafting and review of the component DAARs, per plant series;
- SP4: drafting of the UAARs, drafting and monitoring of the local ageing management programme, per reactor.

The list of SSC/ageing mechanism combinations, the AASs and the component DAARs were initially issued for the 900 MWe plant series and then for the 1300 MWe plant series, by pairs of experts from the design and operations engineering centres, on the basis of knowledge of the behaviour of the equipment and structures and the ageing mechanisms which could concern them.

The demonstration of Ageing Management beyond the VD3 is based on a periodic review of these documents to verify their pertinence and to take account of the new elements resulting from the studies connected to the process, the R&D programmes on ageing management, operating and maintenance experience feedback as well as the comments made by the various units involved in the process, more specifically the NPPs when they draft their UAARs.

With regard to operating and maintenance experience feedback, it comprises elements of the experience feedback process itself, including international OEF, lessons learned from the AP913 (INPO Advanced Process) equipment review for the equipment covered by this process, as well as the results of the supplementary investigations programme (PIC).

This review can lead to a change in the list of sensitive equipment, with modifications to existing AASs, the creation of new AASs, changes to the component DAARs, or the creation of new component DAARs.

The time-frames for the creation and updating of these documents are as follows:

- Prior to each VD3 of a plant series, the entire generic file (list of AASs, collection of AASs, component DAARs) is created;
- The AASs are reviewed annually so as to incorporate operating and maintenance experience feedback and changes in knowledge;
- The component DAARs are updated every 5 years (give or take 1 year) to build on the results of the work done and to incorporate the reference baselines of each new VD for the plant series concerned.

The various components of the process presented above are described in detail in the following sections.

2.3.1.2 CONTROL OF THE PROCESS

Sub-processes SP1 and SP2

Oversight of sub-processes SP1 and SP2 is entrusted to the SEPTEN.

This oversight more specifically comprises the following actions:

- organisation of the annual AAS review, which includes:

    - drafting of a scoping note by the SEPTEN with the support of the UNIE, which identifies the new elements to be taken into account, operating experience feedback in particular, the comments from the contributing entities, the updates to the maintenance documents and the supporting R&D programmes, as well as the AASs issued locally by the NPPs;
    - the organisation of thematic meetings with the participation of the experts from the design and operations engineering centres in charge of the equipment concerned, under the oversight of the SEPTEN with the support of the UNIE;
    - monitoring of the production of documents concerning updating of the AASs concerned;
    - transmission of the output products to ASN: annual collection of AASs and summary notes;
- updating of the methodology guide as required.

Sub-process SP3

Oversight of sub-processes SP3 is entrusted to the SEPTEN.

It comprises updating of the component DAARs every 5 years (give or take 1 year) to incorporate changes to the baseline requirements, on the basis of the DRR (Regulation Reference File) for the components of the main primary system and the main secondary systems.

The component DAARs are documents produced by the SEPTEN with the support of the UNIE and the other engineering entities concerned (depending on the component). One or more UNIE experts are appointed for each component DAAR and their role is to:

- provide experience feedback data;
- provide elements taken from doctrines, strategies and, more generally, from the maintenance and operating documentation;
- provide industrial context data;
- identify any needs in terms of the assurance file (anticipated exceptional maintenance file ready for implementation);
- check the operational feasibility and the on-site impacts of the actions proposed in the component DAAR.

The component DAARs are distributed in advance:

- to the engineering centres concerned;
- to the first-off NPPs of the plant series;
- to the NPPs concerned by special measures or in which the reactors are identified as being particularly susceptible to the ageing phenomena studied.

The component DAARs are validated by an assets review board chaired by the senior management of the DPN (Nuclear Power Generation Division) which must decide on the adequacy and the engagement of the ageing management actions (studies, R&D, assurance files, etc.) with a view to clearing the equipment for continued operation. Performance of these actions is also examined by the assets review board of the DPN, every year.

Sub-process SP4

Oversight of sub-process SP4 is entrusted to the UNIE.

The UAARs are documents drawn up by the NPPs for the purposes of the VD3 and the subsequent VDs.

The UAAR working group, overseen by the UNIE, brings together the NPPs to share best practices and propose improvements to methods or organisation. It meets twice a year, ahead of the process reviews.

The UNIE ensures that the NPPs include the production of these DAARs in preparations for their VD and assists the NPPs in this process. Each NPP appoints a manager, who sets up a multidisciplinary team for drafting of this DAAR, with the organisation being described in a local memo. Progress of drafting of this DAAR is monitored by the NPP's senior management. This reactor DAAR is validated by the NPP senior management and then transmitted to ASN.

2.3.1.3 DRAFTING AND CONTROL OF PROCESS DOCUMENTS

The production of each document leads to the appointment of an author and a checker by the entity responsible for its production (see table below). Similarly, the contributing entities appoint the persons responsible for contributing the various elements.

As part of their duties, the pairs of experts from the design and operations engineering centres in charge of the SSC:

- incorporate new knowledge of ageing mechanisms, operating and maintenance experience feedback;
- analyse their consequences on ageing management;
- under the supervision of the SEPTEN, update the products concerned (AAS list, AASs and possibly component DAARs).

The experts concerned belong to the DIPNN design engineering centres (SEPTEN, CNEPE, CEIDRE) and DIPDE as well as the DPN operations engineering centres (UNIE, UTO).

The following table specifies the frequency and the unit responsible for updating the various documents.

Document	Creation/Update	In charge
AAS list	Creation before the VD3 Annual review and update if necessary	SEPTEN (with contribution by UNIE, DIPDE, CNEPE, CEIDRE, UTO)
AAS	Creation before the VD3 Annual review according to OEF and changing knowledge which can lead to the creation of new AASs and updating of certain AASs.	SEPTEN (with contribution by UNIE, DIPDE, CNEPE, CEIDRE, UTO)
Component DAAR	Creation before the VD3 Update every 5 years (give or take 1 year according to VD baseline requirements)	SEPTEN (with contribution by UNIE, DIPDE, CNEPE, CEIDRE, UTO)
List of UAARs	Annual	UNIE
UAAR	Creation 12 months before the beginning of the VD (DAAR index 0) Update 6 months after return to criticality following the VD (DAAR index 1)	NPP (with contribution by UNIE, SEPTEN DIPDE, CNEPE, CEIDRE, UTO)

Ageing mechanisms summary report	Drafting and updating according to changing knowledge and OEF.	R&D
CapCoV knowledge base	According to updates of mechanism reports.	SEPTEN (with contribution by EDF R&D, UNIE and CEIDRE)

Table 7 – Frequency and unit responsible for updating the various documents

2.3.1.4 INVENTORY OF THE SSC TO BE CONSIDERED (SP1)

The purpose of sub-process SP1 is, from among all the SSCs of a reactor, to identify those components for which an ageing phenomenon or time-dependent degradation mode can lead to difficulty with carrying out a safety function:

- SSCs important for safety (EIPS),
- non-EIPS SSCs, for which ageing could lead to failures liable to compromise the design hypotheses adopted in the safety case;
- non-EIPS SSCs which, with respect to the PSA (Probabilistic Safety Assessments) make a significant contribution to limiting the core melt risk.

With regard to preparation for VD4-900 and the subsequent VDs, the scope of the SSCs is expanded:

- to all the SSC which are elements important for the protection of interests (EIP);
- to the other SSCs considered for the seismic, fire and internal flooding hazard PSAs.

A list of the SSCs considered is drawn up. It lists the SSCs, the particular zones identified or the groups produced per SSC family.

<u>Grouping of ageing analysis reports</u>

In order to limit the number of similar AASs and carry out coherent processing of SSC groups, grouping is possible on the basis of various criteria such as:

- same type of SSC;
- same function;
- same safety class;
- same design;
- same material;
- same supplier;
- same ageing monitoring programme;
- same environment (internal or external fluid).

To date, the number of AASs is about 600 for the 900 MWe plant series and 500 for the 1300 MWe plant series.

2.3.1.5 PROCESS QUALITY ASSURANCE

From the quality assurance viewpoint, the ageing management process is a basic operations engineering process linked to the FMGPI macro-process (Equipment Reliability and Industrial Assets Management).

Results indicators:

- AASs updated: compliance with principles of use of the methodology (methodological guide).

- Time-frames for production of reactor DAARs: compliance with deadlines for transmission of revision indices 0 and 1.

Control indicators:

- Monitoring of numbers:
  - total number of AASs
  - AASs of status 1 and 2
  - new AASs (and analysis of associated statuses)
  - AASs with a status upgrade
  - AASs for which the degradation mechanism has gone from “potential” to “confirmed”
- Monitoring of number of national operating experience feedback reports linked to SSC ageing
- Monitoring of number of assurance files concerning ageing management (available and in progress)
- Monitoring reactor DAAR medium-term production plan.

2.3.2 ASSESSMENT OF AGEING

2.3.2.1 DRAFTING AND REVIEW OF AAS (SP2)

For each of the three large families of materials making up the SSC of PWR reactors (metal, mineral and organic materials), a list of ageing mechanisms to be considered is drawn up. These ageing mechanisms may occur during outage situations or normal operation, during testing or maintenance operations (opening/closure of components, etc.).

The list of ageing mechanisms considered is based on the list in appendix 3 of NSG-2-12 and the US-GALL. This list is supplemented by the EDF process with the identification of new ageing mechanisms via national and international operating experience feedback (for example: leaching of concretes added during review of the AASs at end of 2016). In order to assist the authors of the AASs and DAARs, the ageing mechanisms are characterised by R&D (see § 2.3.2.4 and 2.3.2.5).

During the step consisting in analysing ageing management of an SSC or group of SSCs, mechanisms other than those of the list established may be taken into consideration if felt to be necessary. An overall analysis of ageing could also be made according to this principle, when mechanisms said to be “diffuse” lead to a functional degradation and there would appear to be no point in trying to specifically identify these mechanisms but rather to focus on their effects.

The list of SSCs considered is compared with the pertinent ageing mechanisms for this equipment. This comparison leads to a selection table defining the list of SSC/pertinent ageing mechanism combinations.

Each line on this table corresponds to the analysis of the management of an ageing mechanism/SSC combination.

An AAS is drafted for each line in the table. It gives a referenced overview of the analysis performed, aiming to justify and record the filling out of the table and the choices made. It is used to verify the degree of ageing management in the light of the operating and maintenance provisions in force, along with the repairability and replaceability conditions.

These AAS are open-ended documents with revision index, which are periodically reviewed and, if necessary, updated.

Furthermore, lines associated with new AASs can be added to the table. This table and the AASs are simultaneously updated.

Each AAS is the subject of a “status” classification, used to judge the ability of the existing provisions to ensure long-term ageing management. It shall in particular consider:

- the potential or confirmed nature of the ageing mechanism;
- the suitability of the operating and maintenance actions currently implemented: monitoring, testing, inspection and maintenance;
- the difficulty of repairing and replacing the SSC.

The classification is made on the basis of these criteria families, according to the following table.

STATUS	Confirmed ageing mechanism			Potential ageing mechanism		
Operating and maintenance provisions	**Appropriate**	**Adaptable**	**Adaptable with difficulty**	**Appropriate**	**Adaptable**	**Adaptable with difficulty**
“High” difficulty of repair **and** replacement	**2**	**2**	**2**	**0**	**1**	**2**
“Medium” or “Low” difficulty of repair **or** replacement	**0**	**1**	**2**	**0**	**1**	**1**

- *status 0: the operating and maintenance provisions are appropriate;*
- *status 1: additional examination required to guarantee that ageing is under control (standby status);*
- *status 2: need to create a DAAR for the component concerned, comprising the analysis of the actions in progress or scheduled, in order to manage ageing and the definition of the additional actions or studies required.*

Table 8 – Criteria used for AAS classification

The AASs are annually reviewed in order to incorporate new events. The input data for this review, which are written up in a scoping document, are as follows:

- changes to the maintenance baseline, processing of obsolescence;
- analysis of events from national and international operating experience feedback: data taken from the national experience feedback reports marked “ageing”, results of the PIC, AP913 equipment reviews, etc.
- feedback from UAAR proofreading committees (analysis of local AASs);
- process of R&D work, incorporating experience feedback from collaboration and international exchanges in the field;
- progress of actions concerning the AASs with status 1 and 2, incorporating the results of in-service inspections;
- updating of the DRR, ensuring consistency between plant series, ASN requests, hazard approach, etc.

This review can lead to a change in the list of sensitive equipment with modifications to existing AASs, the creation of new AASs, changes to the component DAARs, or the creation of new component DAARs.

It can also identify additional requirements in terms of R&D, updating of the maintenance baseline, or other actions necessary for ageing management.

The local AASs, produced by the NPPs within the framework of the UAARs, are also examined at the national level and are incorporated into the general collection as soon as they concern at least two sites of the same plant series.

An example of an AAS template is appended.

2.3.2.2 DRAFTING AND REVIEW OF COMPONENT DAAR (SP3)

The Detailed Ageing Analysis Reports concern all components or structures for which at least one zone was the subject of an AAS classified with status 2 for an ageing mechanism or a damage mode.

Each DAAR precisely indicates the scope covered and its limits:

- a component or a structure;
- a group of components or structures; certain groups broader than those of the selection table can be envisaged at this level;
- the operating lifetime considered.

The zones considered and the ageing mechanisms said to be sensitive are recalled.

The component DAAR analyses ageing risks management for the component or structure, or the group of components or structures, with a view to demonstrating their serviceability. It describes the corresponding ageing management programme, including the in-service monitoring, routine and exceptional maintenance, operating conditions, possible modifications and R&D aspects.

The component DAARs are updated every 5 years in order to build on the results of the work done and to incorporate the reference baselines of each new VD, give or take one year, so as to adjust the date of the component DAAR revision to the VD schedule.

There are currently 12 component DAARs for the 900 MWe plant series and 9 for the 1300 MWe plant series.

The component DAARs are validated in the operations engineering decision-making body before distribution to ASN, the NPPs and the participating engineering centres.

<u>Continued operability criteria</u>

The component DAARs comprise an “operability” section, devoted to defining component continued operability criteria for the identified ageing mechanism. The operability criterion corresponds to the maximum acceptable value for the safety-related effects of the ageing mechanism (for example, for the corrosion ageing mechanism, a maximum allowable thickness loss) and can be translated into an anticipated lifespan.

For the components which are not part of a component DAAR, a field concerning the continued operability criteria is directly incorporated into the corresponding AASs.

2.3.2.3 DRAFTING OF REACTOR UAAR, DEFINITION AND MONITORING OF THE LOCAL AGEING MANAGEMENT PROGRAMME (SP4)

In order to supplement the analyses carried out by the national engineering centres via the AASs and component DAARs, the NPPs produce an Ageing Analysis Report for each reactor (UAAR).

The drafting of the UAARs is part of the periodic safety review. This analysis is carried out for each reactor in preparation for each VD as of the third VD. The UAARs are sent to ASN and their conclusions are incorporated into the RCRP (periodic safety review conclusion reports).

The composition of a UAAR is based on the following approach:

1) assimilation by the NPP of each component DAAR and, during this assimilation, identification of any design, manufacturing, production or operating particularities of the reactor or site with respect to the plant series;
2) assimilation by the NPPs of the AAS (Ageing Analysis Sheets) concerning assemblies of equipment / components / structures not covered by the generic component DAARs and, during this assimilation, identification of any reactor or site particularities with respect to the plant series;
3) joint analysis with the national level (UNIE, SEPTEN) of the particularities with respect to the plant series identified by the site in the assimilation phase and definition of the follow-up measures to be taken together with the national level (as applicable modification / addition to the national collection of AASs, or decision to revise or create a component DAAR);
4) drafting of the UAAR by the NPP;
5) validation before transmission to ASN.

In its UAAR, the NPP takes account of the actual condition of the facilities and local ageing OEF.

If a site specificity concerning ageing is detected, the NPP may be required to create a local AAS. This will be the case if:

- an SSC specific to the site and included within the perimeter of the process, is subject to an ageing phenomenon not covered by the national AASs;
- or if a local ageing phenomenon, not covered by the national AASs, is identified by the NPP.

An initial version (revision index 0) of the UAAR is drawn up in preparation for the reactor ten-yearly outage. Subsequent to this ten-yearly outage (VD), this version is supplemented (revision index 1) to take account of:

- the results of the checks and inspections performed during the VD; the summary of modifications and renovations performed during the VD and with an impact on ageing management;
- the analysis of fleet ageing OEF since the previous UAAR revision, based on the new AASs issued and the AASs whose status has been upgraded.

The actions to be scheduled to manage the ageing of SSCs for the ten-year period following the VD and which are identified by the NPP when drawing up the UAAR, constitute the Local Ageing Management Plan (PLMV). This programme supplements the ageing management measures decided on at the national level.

Each NPP appoints a manager to supervise this programme and defines the organisation ensuring that it is deployed after distribution of the UAARs and until the next ten-yearly outage inspection.

2.3.2.4 AGEING MECHANISMS KNOWLEDGE BASE

The ageing mechanisms knowledge base, CapCoV, was developed by EDF R&D to address a two-fold objective:

- build on currently available knowledge concerning ageing mechanisms and their effects, in the form of summary notes with a structured state;
- offer a knowledge consultation tool for all Ageing Management stakeholders (in particular the mechanisms and/or equipment experts in the design and operations engineering centres) on the basis of various search criteria.

The notes, drawn up by EDF R&D, describe the main ageing mechanisms (corrosion, fatigue, thermal ageing, radiation-related phenomena, wear, etc.) and their effects, affecting equipment with major safety implications for reactors in service, comprising various grades of steel, but also non-metal materials such as concretes, composites and polymers.

As for the IT tool, it enables the experts in charge of the equipment or structures:

- to find out the state of knowledge regarding ageing mechanisms;
- to access reference documents;
- to identify the EDF experts in the ageing mechanisms dealt with.

The role of EDF R&D is:

- to draft ageing mechanism reports and validate them with the support of the design and operations engineering centres concerned;
- as required, to update the existing reports and if necessary draft new ones, depending on changes in knowledge and on the basis of OEF.

The role of the SEPTEN is:

- to provide strategic oversight of EDF's R&D activities;
- to ensure the acceptance (compliance with requirement) and integration of the reports into the centre's document baseline;
- to define and carry out validation, hosting, software maintenance of the knowledge base and management of access to it.

2.3.2.5 R&D PROGRAMME FOR CONTINUED OPERATION

The goals of the R&D programme to support the SSC ageing management process are:

- to understand and model the ageing mechanisms of materials in order to predict the ageing of components;
- to determine the characteristics of the materials at the age of 60 years (metal, mineral and organic materials);
- to ensure long-term availability of knowledge on ageing mechanisms and place it at the disposal of the experts from the other EDF Centres;
- to develop new engineering methods or practices, in particular for fatigue and fast fracture analyses in the DRR and the RPV in-service strength file;
- to advance equipment monitoring and inspection means (NDT/NDI), looking for the best available technologies;
- to contribute to the development of repair or mitigation processes.

In addition, for the continuation of reactor operations, R&D supports:

- the maintenance process through its ability to appraise removed components and the provision of equipment monitoring and diagnosis/prognosis tools;
- innovation, by assessing new materials or new technologies and contributing to their qualification.

2.3.3 MONITORING, TESTING, SAMPLING AND INSPECTION ACTIVITIES

The monitoring, testing, sampling and inspection activities are more specifically contained in the Basic Preventive Maintenance (PBMP see § 2.3.4.2), Non-Destructive Testing (NDT) Programmes, the Supplementary Investigation Programmes (PIC) and in-service monitoring.

2.3.3.1 PRODUCTION OF INSPECTION PROGRAMMES

For the equipment monitored, in particular that covered by the pressure equipment regulations, maintenance doctrines (or equivalent) summarise the design data, the failure risks analysis, the operating conditions and feedback, enabling an objective assessment to be made of the occurrence of failure mechanisms. If a risk is identified (fatigue, fast fracture, corrosion, etc.) or if an area is the subject of particular attention in terms of defence in depth, then an inspection is proposed, with definition of its objective and its frequency (nature of damage looked for, minimum flaw dimensions, location, etc.), along with the NDT processes used. If the equipment is subject to article 8 of 10 November 1999 (see §2.3.3.2) the type of qualification is specified.

These inspections (visual, NDT, etc.) and their frequency are prescribed by the PBMP (see §2.3.4.2). Experience feedback from the results and the implementation of these inspections enables the performance of the inspection processes to be periodically assessed and modified as necessary.

2.3.3.2 NDT QUALIFICATION

Pursuant to article 8 of the order of 10 November 1999 concerning the operation of main primary and main secondary systems of PWRs, the NDT processes must be qualified prior to implementation on the site by an independent, recognised organisation. As a type B internal inspection organisation as defined by standard NF EN ISO / CEI 17020, EDF NDT Qualification Commission has been accredited by COFRAC (French Accreditation Committee) since 2002. The role of this Commission is to provide completely independent confirmation of the compliance of the NDT processes to be qualified with the Licensee's functional requirements baseline. This compliance review results in the issue of a certificate of compliance in the form of an attestation of qualification.

2.3.3.3 SUPPLEMENTARY INVESTIGATIONS PROGRAMME (PIC)

A Supplementary Investigations Programme (PIC) was implemented by EDF for VD2 900 and VD3 900, as well as for VD2 1300. It is being defined for VD3 1300 and VD4 900. The PIC is implemented in addition to the maintenance strategies and programmes. The aim is to corroborate the hypotheses regarding the absence of significant in-service degradation in the zones not covered by the Basic Preventive Maintenance Programmes or by special maintenance programmes.

2.3.3.4 IN-SERVICE MONITORING

In-service monitoring is carried out in the control room, but also throughout the facility, in the technical areas and on the equipment itself.

"Local" monitoring at the very least is carried out systematically by each shift, with a clearly defined frequency and scope. This is known as field inspections. They ensure that there are no leaks (water, oil, coolant, air, gas) or other operating anomalies (release of heat, noise, vibration, friction, etc.). Some of these inspections lead to the in-situ input of a certain number of operating parameters (pressure, temperature, flowrate, levels, etc.) into a portable data terminal. This collection is followed by computer processing to detect anomalies, notably slow drifts and latent defects. Each of these defects is then dealt with appropriately or justified as-is.

Local monitoring is not however limited simply to the daily field inspections: for example, the start-up of a large equipment item (ASG pump, CRF pump, turbine) entails a prior check as close as possible to the item and specific monitoring for early detection of any malfunction so that the operator can be alerted.

Moreover, the MEEI management project "maintaining the facilities in exemplary condition" has been in place for several years, in line with international practices. It makes provision for periodic verification of the conformity and cleanness of all premises, equipment and systems. The observable requirements concern the road networks and condition of the buildings, cleanness, tidiness, worksites, the prevention of migrating parts, packaging, marking/display/labelling/tag-out, fire-fighting equipment, leak management, cable trays, lighting, the condition of equipment (supports, anti-seismic measures, heat insulation and apparent condition), security and radiation protection equipment.

Moreover, pursuant to Chapter IX of the General Operating Rules, periodic tests are performed to guarantee management of the safety functions and the availability of the elements important for safety.

Periodic tests are technical inspections performed on equipment and systems qualified as being available and constituting a line of defence to ensure:

- the absence of any unfavourable trend with respect to the design baseline requirements,
- compliance with the hypotheses of the accident studies,
- monitoring of the availability criteria for equipment and the associated fluids constituting the required safety functions,
- monitoring of the operability of the incident and accident operating procedures.

Trend monitoring by comparison with the criteria of the general operating rules is implemented to analyse the measurements and conditions of the periodic tests, in order to be able to rule on the long-term availability of an equipment item and the implementation of corrective measures as necessary.

In addition to the inspections, field inspections and periodic tests, the operating conditions are monitored, more specifically through:

- monitoring of the chemical and radiochemical parameters (chemical conditioning and impurities in the reactor coolant, chemical conditioning of the reactor coolant system auxiliary systems, as well as the secondary systems, impurities concentrated in the SGs, etc.),
- accounting of the MPS and MSS equipment situations in order to check compliance with the inventory of design transients used for the design (accounting reports are periodically produced and are analysed in order to limit the occurrence of sensitive situations)[22],
- monitoring of the behaviour parameters for certain equipment items, such as the pumps equipped with permanent instrumentation giving access to vibration levels, bearing temperatures, flow, pressure and other parameters, on the basis of which inspections can be scheduled,
- instrumentation of certain lines and pipes to measure the occurrence of operating situations and verify their ability to deal with the risk of fatigue (e.g. local thermal-hydraulic phenomena such as in the "MPS dead legs"),
- continuous monitoring of ambient environment parameters in certain areas (temperature, dosimetry, etc.) in order to verify compliance with the operating technical specifications.

2.3.3.5 EQUIPMENT QUALIFIED FOR ACCIDENT CONDITIONS (MQCA)

For equipment qualified for accident conditions, maintaining the corresponding requirements is more specifically ensured by the following:

- prescriptions in the work procedures,
- periodic activities identified in the maintenance programmes as being necessary for maintaining qualification,
- compliance with the procurement routes for the spare parts used.

A gradual qualification approach was implemented by EDF to maintain the qualification of the MQCA beyond VD4. This approach involves:

- specific analyses or studies;

[22] For certain sensitive areas, such as the fluid mixing zones, the total operating duration beyond a temperature threshold is counted on the MPS or the MSS (RRA or ASG system) and can trigger monitoring and/or specific maintenance.

- campaigns to monitor the MQCA operating conditions (temperature, radiation, vibration);
- assessments and tests on MQCA sampled from the site after more than 30 years of operation;
- replacements and renovations.

2.3.3.6 POTENTIAL DEVELOPMENTS IN NON-DESTRUCTIVE TESTING (NDT)

The NDT development programme is based on dealing with identified threats and risks for the resources and on the foreseeable technological intelligence options.

It comprises four main orientations:

- contribution to management of risks linked to gamma radiography by:
    - replacing existing gamma radiography methods with possible alternatives (selenium source or X-ray accelerators when the thicknesses so allow, digital media when the technology is able to meet the functional needs, TOFD ultrasounds technique and Phased Array),
    - a survey of the areas where weld surface grinding is possible, which will allow easier deployment of techniques other than gamma radiography,
    - codification and its updating to facilitate the deployment of innovative technologies,
- active industrial intelligence to initiate development and pre-industrial testing of processes which could meet the needs of the nuclear fleet in areas where performance is inadequate or non-existent,
- good visibility given to contractors to help them prepare and reinforce their skills,
- storage and archiving of data resulting from the inspections.

2.3.4 PREVENTIVE AND CORRECTIVE MEASURES

2.3.4.1 MAINTENANCE POLICY

The purpose of the maintenance organisation at EDF is to guarantee the operation of equipment in accordance with the safety requirements and the best production conditions.

The maintenance policy is structured in such a way as to guarantee the required reliability level for equipment and systems, by anticipating the maintenance of equipment with a view to extending the operating lifetime of the reactor fleet up to VD4+20 years.

When the Fleet was originally created, the preventive maintenance programmes were mainly derived from operating experience feedback from the manufacturers and the conventional thermal power plant fleet. As time went by, they were enhanced with nuclear reactor OEF from EDF and licensees in other countries, as well as with the incorporation of new maintenance methods: RCM (Reliability-Centred Maintenance), condition-based maintenance, pilot equipment based maintenance, then by gradual implementation of AP913 as of 2009.

To guarantee that the reactors of the 900 MW plant series have an operating life of VD4+20 years, the equipment maintenance strategies were supplemented when necessary by larger-scale exceptional maintenance operations. These operations are scheduled for the period running from the VD3 to the VD4 and even beyond, by means of an approach that is resolutely forward-looking.

2.3.4.2 ROUTINE MAINTENANCE

Preventive maintenance programmes have been constantly added to through analysis of operating experience feedback (OEF) concerning the behaviour of structures and components, from:

- observations made during equipment inspections or disassembly operations;
- analysis of events affecting the EDF nuclear fleet and the international fleet.

For a certain number of components, the preventive maintenance activities were redefined on the basis of RCM studies. This approach, inspired by the aeronautical industry, uses a rigorous methodology developed within the company to incorporate probabilistic safety assessments, functional systems analysis, failure modes, effects and criticality analyses and operating experience feedback. This RCM approach was gradually implemented as of 1995.

Condition-based maintenance is built around the definition of parameters which are monitored to be able to accurately anticipate the unavailability of the equipment and thus determine the preventive maintenance to be scheduled before it actually fails. This condition-based maintenance, built around the measurement of parameters representative of equipment operation, relies on the most recent technological monitoring developments (sensors, data collectors, data transmission and analysis, etc.). It also relies on a variety of techniques: modal monitoring of vibrations, acoustic monitoring, infrared thermography, analysis of oils and dissolved gases, etc. Its implementation enables the condition of numerous equipment items to be diagnosed: servomotors, valves, turning machinery, electrical devices, etc.

EDF has developed a policy of “pilot” items, for monitoring of samples representative of the overall behaviour of a fleet of comparable machines operating in similar or less harsh operating conditions. These items are the subject of assessment reviews and reports.

Whether from EDF fleet internal operating experience feedback, from that of the manufacturers of the various equipment items, or from that of the international fleet, numerous operational observations are input into an iterative process to fine-tune the pertinence of the maintenance definition or frequency choices. Over and above this qualitative OEF, objective reliability data or monitoring of the equipment unavailability specified in the General Operating Rules (duration and number), permanently inform the licensee EDF of the compliance of its maintenance policy with the requirements relating to the protection of interests.

As part of a continuous improvement approach, EDF decided to implement the INPO AP913 method. Following on from RCM, the first aspect of this method is to classify the components according to their criticality, in terms of both safety and availability. It allows a more pragmatic and systematic approach to be developed to the maintenance of components and systems, by combining monitoring and maintenance of these items. Consequently, it makes it possible to define maintenance and monitoring programmes depending on the functional importance of the equipment.

The preventive maintenance programmes are permanently updated:

- by the observations made during equipment inspections or disassembly operations;
- by the analysis of events affecting the EDF nuclear fleet and the international fleet.

This approach is supplemented by a forward-looking approach which consists in analysing the potential degradations which could occur according to the equipment and its operating mode. For the equipment which falls within the scope of AP913, this approach takes the form of system and/or equipment reviews, which are drawn up and analysed both locally and nationally.

This permanent, iterative review process leads to the routine maintenance programmes being either supplemented or, on the contrary, reduced:

- Basic Preventive Maintenance Programmes;
- Local Preventive Maintenance Programmes;
- Additional reviews resulting from various projects or contracts.

2.3.4.3 EXCEPTIONAL MAINTENANCE

Exceptional maintenance refers to maintenance operations differing from conventional preventive maintenance by one or more of the following aspects:

- occasional (maintenance operations which only occur from one to a few times during the service life of the reactors) and possibly systematic nature, in other words linked to a deadline set in advance (e.g. ten-yearly outage inspection);
- technical difficulty (implementation of the operation, significant impacts on the equipment present at the functional and/or geographical interfaces);
- significant financial investment;
- lengthy design or manufacturing times;
- significant impacts on the outage duration necessary for performance of the operation.

This type of exceptional maintenance operation may consist of replacements, overhauls, or repairs.

2.3.4.4 INCORPORATING AGEING MANAGEMENT INTO THE DESIGN

In the case of the Flamanville 3 EPR, design measures to mitigate the effects of ageing were taken with the goal of achieving a reactor service life of 60 years. They apply in particular to non-replaceable components such as the reactor pressure vessel and the containment, as well as to the MQCA for which the initial qualification, via the NSQ (Qualification Summary Report) stipulates a qualified service life of 60 years as a general rule (or less, in multiples of ten years). The replacement components subject to significant cyclic loadings are also the subject of a fatigue failure mode analysis. The inventory of design transients (DDS) describes these loadings for a service life of 60 years and the situations during operation are counted, as with the previous plant series.

2.4 ASSESSMENT AND UPDATING OF EDF OVERALL AGEING MANAGEMENT PROGRAMME

2.4.1 EXTERNAL ASSESSMENT OF THE PROCESS

EDF Ageing Management process is subject to regular external assessments by:

- ASN, IRSN and the GPR (Advisory Committee for Reactors, comprising experts from the field of nuclear reactors);
- the IAEA, via its OSART audits and LTO modules.

The results of these external audits and inspections are incorporated into the AAS review input data (see § 2.3.2.1) and into the process reviews (see § 2.4.2).

Examples of external assessments:

The Ageing Management process has undergone a number of assessments by the GPR (Advisory Committee for Reactors):

- Ageing GP of 2003 (overall process assessment, AAS, NDT programme, R&D programme, maintenance programmes, etc.);
- Ageing GP of 2006, (component DAAR, etc.);
- VD3 900 MWe GP of 2008 (UAAR for Tricastin 1, etc.);

- DDF[23] programme guidelines GP of 2012 (operation after VD4);
- Guidelines GP for VD4-900 of April 2015.

EDF was inspected by the IAEA during the Corporate OSART from 23 November to 9 December 2014. Prior to the inspection, EDF carried out a self-assessment using IAEA Safety Guide NS-G-2.12 of January 2009.

2.4.2 INTERNAL REVIEW AND UPDATING OF PROCESS

EDF reviews these processes. The Ageing Management process review of June 2015 thus responded to two requests:

1. the FMGPI review (Increasing Equipment Reliability and Managing Industrial Assets) of 18 April 2014 which, although it found no weaknesses in the Ageing Management process, asked for further progress: "Promote greater involvement by the local engineering centres in the drafting of the UAARs, within the framework of the UAAR pilots network".
2. preparation of the VD4 900 MWe Guidelines Advisory Committee, within which EDF made the following undertaking: "After ten years of implementation, the SSC ageing management process will in October 2015 undergo a process review to assess its ability to anticipate the identification and processing of ageing mechanisms; its implementation by the NPPs; its overall working, notably at the interfaces between the various local and national stakeholders involved".

The process review becomes periodic as of 2017. The process review input data are taken primarily from:

- the external assessments performed by ASN, IRSN, the GPR or IAEA (see § 2.4.1),
- the AAS reviews (see § 2.3.2.1), the component DAARs (see § 2.3.2.2) and the UAARs (see § 2.3.2.3);
- the comparison between the ageing management process and the international standards set out by the IAEA concerning ageing management (see § 2.4.4).

Measuring the effectiveness of ageing management:

The effectiveness of ageing management is assessed by the number of national OEF reports attributable to ageing. This indicator is defined in the process note.

2.4.3 REVISION OF THE METHODOLOGY GUIDE

The purpose of the latest revision of the methodology guide at the end of 2016 is:

- to include the 1300 MWe plant series;
- in preparation for the VD4-900, expansion to include the SSC EIP (previously only applicable to the SSC EIPS);
- expansion of the hazards approach: incorporation of the contributor SSCs with regard to the seismic, internal flooding and fire risks PSAs;
- integration of international developments; IGALL, NS-G-2.12 (DS485);
- implementation of the process beyond VD4 (VD4 + 20 years) with updating of the AAS template.

[23] Operating service life.

2.4.4 CORRESPONDENCE OF THE EDF PROCESS WITH INTERNATIONAL STANDARDS

The EDF ageing management process complies with the requirements of all the international standards. However, the documentary structure put into place by EDF differs from that of the IAEA documents. More specifically, the elements specified in the three IAEA documents (AMR, AMP and TLAA) are implemented within the EDF process via several documents with a different documentary architecture, which notably results from the management of the EDF NPP fleet by means of documentary series.

EDF has initiated a number of measures to establish a link between its ageing management process and the international standards drawn up by IAEA. At the last update of its methodology guide for ageing management of PWR reactors, EDF introduced an appendix explaining how its process complied with these international standards.

AMR (Ageing Management Report):

The AMR is the equivalent of the AAS list. The AASs issued for the SSC/pertinent mechanism combinations in terms of ageing management, contain most of the items given in the AMR. The engineering studies, the DRR studies, the documents concerning equipment qualification for accident conditions and the documents concerning SSC maintenance (maintenance doctrine, PBMP) are more specifically mentioned.

AMP (Ageing Management Programme):

For the EDF fleet in service, the activities relating to understanding of the ageing mechanisms are:

- R&D described in reports produced for each mechanism, on which the AAS and component DAARs are based;
- analyses, assessments or tests carried out for maintained qualification after the VD4 described in the NSQP (gradual qualification strategy reports) for equipment qualified for accident conditions.

The other actions are part of reactor operation and maintenance. They concern the following fields:

- chemical conditioning of the systems;
- the implementation of low flux fuel loading plans;
- certain modifications;
- situations accounting;
- operating provisions to avoid or mitigate the situations;
- NDT;
- in-service monitoring (field inspections, periodic testing, MEEI evaluations, etc.);
- assessments during intrusive or condition-based maintenance work;
- preventive, remedial and exceptional maintenance;
- assurance files;
- operating experience feedback.

These actions are mentioned in the component DAARs, the AASs and the UAARs. They are generally produced within routine and exceptional maintenance prescriptions.

TLAA (Time-Limited Ageing Analysis):

In the EDF ageing management process, the IGALL TLAAs correspond to the following documents:

- For the equipment of the main primary system and the main secondary systems: Regulation Reference File (inventory of design transients, behaviour analysis files, fast fracture files, materials files)
- For equipment qualified for accident conditions: qualification summary reports
- For the other equipment: specific engineering reports.

2.4.5 INTERNATIONAL TECHNOLOGICAL INTELLIGENCE

The EDF ageing management process evolves according to operating experience feedback, both national and international. EDF is involved in various respects (expert, observer, contributor, coordinator, etc.) in various international working groups, whether for mechanical components or for civil engineering structures, I&C and electrical components.

The objectives are the following:

- to stay abreast of the recommendations, guides or documents issued by the Safety Regulators;
- to take part in topics that are important for EDF (ageing, integrity, safety margins and consequences);
- to ensure that the conclusions and recommendations issued by the main groups are consistent with the positions adopted by EDF;
- to compare the approaches adopted by the main nuclear countries on reactor ageing management and the corresponding justifications;
- to utilise international frameworks to share R&D measures in support of ageing management;
- to consolidate the French approaches, service life methodology, integrity analysis practices, margin assessment, break preclusion, Codes and Standards (RCCM, RSEM), other methodologies, etc.

EDF collaborates notably with the IAEA, OECD, EPRI, ASME, the NUGENIA association, the PWROG and FROG licensee groups.

2.5 EDF EXPERIENCE WITH APPLICATION OF ITS OVERALL AGEING MANAGEMENT PROGRAMME

The EDF ageing management process evolves in line with requests and operating experience feedback:

- 2001: DSIN letter of 19/02/01 “ageing: operation of reactors beyond their VD3”
- 2002: Launch of Project entitled “Ageing management of components and structure of 900 MW reactors”, the aims of which were as follows:
  - Draw up an inventory of knowledge about ageing mechanisms,
  - Define a methodology,
  - Draw up an exhaustive list of equipment susceptible to ageing for the PWR 900,
  - Issue the first component DAARs.
- 2003: definition of the EDF organisation in preparation for the ageing GP:
  - Drafting of the list of components susceptible to ageing,
  - For each of these susceptible components, determine the additional actions required,
  - Creation of the AASs with status 0, 1 and 2,

    - Creation of the component DAARs.
- 2005: sustainability of the “Ageing management “process:
    - Joint decision DIN-DPN-R&D rev. 0 defining the EDF organisation,
    - Creation of the methodology guide.
- 2006: Ageing GP of 2006 – review of the component DAARs:
    - Requests for sampling of MQCA more than 20 years old,
    - Monitoring of failure rates of electrical and electronic equipment,
    - Additional information about the R&D programmes,
    - Revision of AASs every year and of component DAARs every 5 years.
- 2008: GP VD3 900 MWe of 2008: review of the Tricastin 1 UAAR:
    - No recommendation concerning Ageing Management.

Since 2006, the structure of the process has remained basically the same, with a few adjustments as a result of continuous improvement. The review of the collection of AASs has been carried out every year since 2006 for the 900 MWe plant series and since 2012 for the 1300 MWe plant series.

The EDF ageing management process is updated on the basis of internal and external operating experience feedback during process reviews and reference document updates (process report, methodology guide).

An example of such an update is the introduction of the continued operability criteria in all the 900 AASs which are not the subject of a component DAAR (the component DAARs already include this criterion). This introduction follows on from an ASN request, corresponding to the use of the IGALL for which the continued operability criterion is one of the attributes of the AMP.

2.6 RESEARCH REACTORS AGEING MANAGEMENT PROGRAMMES

2.6.1 CEA

2.6.1.1 SCOPE OF PROGRAMME

As a general rule, ageing management is based on periodic safety reviews (every ten years) and a periodic preventive maintenance programme.

This preventive maintenance is carried out periodically in the same way as the periodic inspections and tests, in accordance with validated procedures and accompanied by a risk assessment if the intervention is liable to have an impact on safety. It should be underlined that the requirements concerning the frequency of the inspections are proportionate to the implications and potential consequences.

Satisfactory performance of the tests during the periodic safety reviews and implementation of the periodic maintenance plans, in accordance with the specified frequencies, enables the items concerned to be declared available. The aim of systematic maintenance is to prevent failures of these items of equipment and to preserve their ability to fulfil their function with the required performance. It should also be noted that in practice, all equipment can be replaced except for the containment. Moreover, on the CABRI reactor, some equipment with a significant safety impact and reliability requirement (for example, the digital and analogue control assembly for the transient rods, ensuring the reactivity insertion pulse) are subject to particular monitoring (management of part temperature, recording of operating hours, periodic maintenance with replacement of parts).

Research reactors furthermore have a reduced radiological inventory. This notably means that the doses received by the equipment remain limited and do not generally lead to accelerated ageing, except for a few items which undergo particular monitoring.

2.6.1.2 ASSESSMENT OF AGEING

To date, CEA mechanical and electrical equipment does not benefit from a “fleet” effect and the aim is to look for bulk purchases with the goal of ensuring security of supply.

The purpose of the “Operations” process of the integrated management system in the Research Reactors Operations Division is notably to use a systematic analysis of problems detected and measures considered to be best practices, to implement a guide of best practices for equipment and systems ageing and obsolescence management.

Four priority levels are defined:

- Priority 1: obsolete equipment for which spares are no longer available and requiring complete replacement and major work, or a system with no equivalent nor possibility of upgrade → significant or penalising impact on the facility;
- Priority 2: obsolete equipment for which spares can still be bought, although with an average procurement lead-time, or a more widely commercially available system for which continued operation requires work or modifications → limited impact on the facility;
- Priority 3: obsolete equipment that is easily replaced by a recent and rapidly available equivalent → low impact on the facility;
- Priority 4: no obsolescence (equipment or parts still commercially available).

Depending on the priority level, an action plan is carried out for replacement of equipment or systems.

Analysis of operating experience feedback is a key aspect allowing detection of events linked to ageing and thus improvement of the monitoring programme for equipment liable to be affected by a failure with a generic origin.

The obsolescence management approach for the SIREX[24] I&C fitted to certain CEA research reactors (ÉOLE, MINERVE, CABRI, ISIS[25]) is controlled by the corresponding competence centre and based on a contract placed with the exclusive supplier (Rolls Royce). This arrangement guarantees the continued supply of spare parts. This service, which has been covered by contracts with the system designer company since 2005, is a means of ensuring:

- obsolescence monitoring of the hardware and software in order to maintain the ability to ensure corrective maintenance of the hardware, boards and subassemblies, by guaranteeing the availability of the various components;
- obsolescence processing through CEA proposals for preventive measures, which can more particularly be:
  - stock-piling of components by the contract-holder before they disappear from the market,
  - replacement by a compatible component,
  - “complex” processing if it becomes impossible to find an equivalent component;

([24]) The EIP classified SIREX system allows permanent monitoring of the reactor’s power and how it develops by measuring the doubling time, from the neutronic flux measurement. It participates in reactor protection through alarms and emergency shutdown commands when the neutronic flux value is too high or in the event of a flux variation.

[25] The approach should also be extended to ensure neutron control of the MASURCA reactor as part of its refurbishment.

- compilation and processing of operating experience feedback. Using all the data collected by the contract-holder (assessment report for equipment repaired in the factory, information sheets forwarded by the sites), the contract-holder is asked to create and update a database for:
  - monitoring the specific configurations of each facility,
  - ensuring the traceability of failures on each board and each rack specific to each facility,
  - compiling OEF and extracting summaries and recommendations,
  - ensuring a monitoring and alert function for problems which are occurring and which could require in-depth analysis;
- the provision of technical assistance to the manufacturer in looking for causes of malfunctions;
- the performance of occasional services such as corrective maintenance, preventive maintenance, specific studies and training.

With regard to other I&C and automation systems, various targeted measures are taken on certain products, based on local or central initiatives. Finally, the ageing management of the radiation monitoring panels (TCR) is covered by an overall approach including detectors and monitors, designed to accompany the facilities in the transition from one generation to the next.

2.6.1.3 MONITORING ACTIVITIES, PREVENTIVE MEASURES, PROGRAMME ASSESSMENT AND EXPERIENCE OF THE LICENSEES

Monitoring activities and preventive measures are carried out as part of the periodic safety reviews, implementation of the periodic preventive maintenance plans and periodic inspections and tests.

It should also be noted that practically all the equipment can be replaced except for the containment.

For example, on the CABRI reactor:

- the periodic tests represent 417 types of inspection per year, including regulation verifications. They are able to monitor the EIP or non-EIP equipment: 18 daily inspections, 5 weekly, 24 monthly, 10 every 3 months, 27 every 6 months, 144 every year, 12 every 2 years, 40 every 3 years or 40 months, 13 every 5 years, 56 every ten years and 68 inspections with no precise frequency;
- the preventive maintenance operations linked to the equipment and carried out by the maintenance contractors represent about 350 maintenance operations on the pumps, valves, motors, lighting, emergency lighting, chillers, heaters, batteries, etc.

All of these checks concern all areas of activity (mechanical, radiation protection, water chemistry, electricity distribution, fire, I&C, environment, etc.) and thus take part in ageing management.

The 155 EIP on CABRI are concerned. There are no general rules with regard to the frequency. This depends on the characteristics provided by the manufacturers, operating experience feedback from certain equipment, changes in physical parameters, in the regulations, etc.

2.6.2 THE ILL

2.6.2.1 SCOPE OF THE PROGRAMME AND AGEING ASSESSMENT

The ageing management programme for the reactor equipment is based on mechanical, electrical and electronic maintenance programmes, as well as on inspection and periodic testing programmes. The maintenance programmes aim at guaranteeing availability. The performance of the equipment is checked by means of periodic checks and tests (CEP).

The two essential safety requirements are:

- to maintain the water inventory around the fuel element because natural convection is sufficient to guarantee its cooling once the rods have dropped. This is based on the mechanical strength of the reactor pressure vessel components and redundant back-up systems guaranteeing that this water inventory is maintained;
- to maintain confinement which is based on the double containment (reinforced concrete + steel) and on the redundant back-up systems which allow the reactor containment negative pressure and filtration to be maintained.

Each year, about 600 periodic test procedures and 100 maintenance procedures are carried out. They are distributed in a similar manner between the mechanical equipment and the electrical/electronic equipment.

The mechanical and electrical/electronic maintenance plans cover all the EIP. Some maintenance is periodic, based on operating experience feedback or on known lifespans according to exposure to radiation or a number of cycles. Other maintenance operations are triggered after the repeated need for adjustment to return to the nominal range has been observed, even if remaining within the tolerance range. Some turning machines are equipped with a vibration analysis system, on which a change in measurements determines the scheduling of maintenance. Finally, some maintenance is exclusively remedial, when the level of redundancy so allows.

EIP performance is:

- mechanical (tightness of various confinement systems of the reactor coolant system and containments, absence of corrosion or flaws, drop time of safety rods, valve opening times, etc.);
- electrical (independence of continuous sources, emergency pick-up by diesel generators, etc.);
- electronic (performance of neutronic and thermal-hydraulic channels, safety systems, etc.).

The ten-yearly safety reviews are an opportunity to verify the conformity of the EIP. On the occasion of these reviews, the ILL carries out a conformity analysis to check that the manufacturer's files, the condition of the equipment and its performance are indeed in compliance with what is described in the baseline safety requirements and are not affected by ageing. For example, the ILL verifies the leak rate from leaktight cables, carries out non-destructive testing of metal structures (dye-penetrant, radiography, ultrasounds inspection, eddy current inspection). Then, on the basis of the safety analysis review, the ILL identifies any discrepancies between the new requirements in the updated standards or regulations (earthquake to be considered, new assessment rules) and the existing situation in order to define a conformity alignment programme including modifications to the facility and an additional programme of periodic test and maintenance.

The ILL transmitted the periodic safety review report in November 2017.

2.6.2.2 MONITORING ACTIVITIES, PREVENTIVE MEASURES, PROGRAMME ASSESSMENT AND EXPERIENCE OF THE LICENSEES

The preventive maintenance and periodic inspection and testing programmes define the required frequencies and performance and may concern one of the 2, 3 or 4 existing systems alternatively, in order to avoid the same error on redundant systems. The organisation makes provision for hold points at important verification steps, specifying what is expected of the person carrying out the verification. Finally, independent spot checks are also carried out to ensure correct application of the programmes but also to check the pertinence of these programmes.

The maintenance programmes are a means of preventing equipment performance from drifting. The deviations management process is a means of returning to the operating range defined within the safe operating range.

The management reviews generate feedback which can be put to good use to modify the maintenance programmes, for example by reducing or increasing the frequency of the periodic inspections and tests (CEP). The management reviews are held yearly. On the occasion of these management reviews, the following in particular were brought to light:

- an increase in the number of electrical connection faults which led the ILL to schedule periodic checks on the tightening of terminals on certain equipment;
- several faults due to the failure of pneumatic slide valves, leading the ILL to initiate a campaign to replace this type of component;
- repeated drift in the thermal measurement channels, which led the ILL to replace the electronics and relocate them to a less harsh radiological atmosphere.

The inspection programmes for example led to replacement of the reactor block with reflector vessel after 20 years of operation. The design of the new reactor block makes it relatively simple to replace the parts most exposed to neutron radiation, with an anticipated life of 50 years for the new vessel.

2.7 REGULATORY OVERSIGHT PROCESS

ASN employs many oversight methods. This oversight primarily consists of:

- inspections on-site – or in the departments of the licensees or their contractor for activities with a significant impact on safety – worksite inspections during maintenance outages of facilities and technical meetings on the site with the BNI licensees or the manufacturers of the equipment used in the facilities;
- technical examination of the substantiating files and documents;
- EDF NPP reactor outage oversight.

2.7.1 INSPECTIONS

Within the fields for which it has oversight responsibility, ASN periodically identifies those activities and topics with significant implications, on which it will focus its inspection means and exercise direct verification at a predetermined frequency. Each year therefore ASN draws up a forward-looking inspection programme. This programme identifies the targeted facilities, activities and topic. Those responsible for the nuclear activities are not made aware of it.

To carry out these inspections, ASN has inspectors chosen for their professional experience and their legal and technical expertise. These inspectors are qualified further to a training course appropriate to their functions and are then appointed by an ASN resolution. They exercise their inspection activity under the authority of the Director-General of ASN. They take an oath and are bound by professional secrecy.

With regard to EDF NPP reactors, ASN carries out inspections in the NPPs on the topic of ageing management, to coincide with the progress of the VD3 900 and VD3 1300. These inspections more specifically constitute an opportunity for ASN to verify how the NPPs have assimilated the process defined by the EDF national engineering centres, notably with regard to taking account of any specific aspects of their facilities in their local ageing management programme. On average, 5 inspections per year are carried out on this topic. Furthermore, the inspections performed on other topics (systems, maintenance, etc.) can also represent an opportunity to check how EDF manages the ageing of the SSCs in the NPPs.

In its inspections, ASN can check that the licensees of research reactors have taken account of ageing issues relating to their equipment and materials. The inspections concerning periodic checks and testing and maintenance are in particular a means of checking the correct performance of tests (in the broad sense) and that the corresponding equipment complies with the requirements defined for it. The steps taken to deal with any non-conformities are also examined. For example, since 2013, ASN has carried out two inspections on this topic of periodic checks and tests for the RHF and one inspection on CABRI.

The ageing management aspects can also be the subject of checks during thematic inspections linked to monitoring of the operation of nuclear pressure equipment (NPE) (three inspections carried out since 2013 on the RHF) or to the containment. The tests, checks and maintenance on the containment are verified during inspections concerning this particular topic, along with any non-conformities and their processing.

More generally, since 2013, in accordance with a graded approach commensurate with the implications, ASN has carried out:

- 19 inspections on ORPHÉE,
- 20 inspections on the RJH (under construction),
- 16 inspections on CABRI,
- 48 inspections on the RHF.

2.7.2 EXAMINATIONS

The purpose of the files provided by the licensee is to demonstrate that the objectives set by the regulations, or those set by the licensee itself, are met. ASN checks the completeness of the file and the quality of the demonstration.

Examination of these files may lead ASN to ask for additional demonstrations, studies or even works to ensure that compliance is restored. ASN issues its requirements in the form of resolutions. ASN may thus request an opinion from its technical support organisations, the most important of which is IRSN, whenever it so considers necessary. For the more important issues, ASN requests the opinion of the competent advisory committee, to which it and its technical support organisation present the result of the assessments; for most other matters, the safety assessments are given in IRSN opinions requested directly by ASN.

The examinations carried out with regard to ageing management of EDF NPPs are detailed in sections 2.1.4 and 2.4.1.

2.7.3 EDF NPP REACTOR OUTAGE OVERSIGHT

Article 2.4.2 of ASN resolution 2014-DC-0444 of 15 July 2014 concerning NPP pressurised water reactor outages and restarts states that the licensee must send ASN an approval request for the approach to criticality and then criticality operations after an outage during which some or all of the fuel present in the vessel has been renewed. The reactor criticality approval request includes a demonstration by the licensee that the installation is capable of functioning over the forthcoming cycle in conditions that suitably protect the interests mentioned in Article L. 593-1 of the Environment Code and in compliance with the baseline requirements applicable to the installation.

The items accompanying the request more particularly include:

- a detailed statement of the activities carried out on the EIPs during the shutdown, and any differences with respect to the activities listed in the reactor shutdown presentation file;
- the list of deviations affecting the EIPs for which the licensee has not implemented all the remedial measures defined pursuant to Article 2.6.3 of the abovementioned Order of 7 February 2012 and a summary of the justification - with respect to the protection of the interests mentioned in Article L. 593-1 of the Environment Code, of why they have not been resolved, with the deadline also being specified for each deviation.

Before issuing its approval for reactor criticality, ASN may thus ensure that the ageing-related deviations (among others) are dealt with appropriately by the licensee.

2.8 ASN ASSESSMENT OF THE OVERALL AGEING MANAGEMENT PROGRAMME AND CONCLUSIONS

2.8.1 NUCLEAR POWER REACTORS

For the fourth periodic safety reviews of the 900 MWe reactors, EDF proposed once again using the ageing management approach applied since the third periodic safety reviews of these reactors, while reinforcing the projects to refurbish and replace equipment with a view to ensuring continued operation up to 60 years. As previously explained, this approach is based both on a generic analysis performed at the national level, more specifically through generic ageing analysis sheets (AAS) and "component" detailed ageing analysis reports (DAAR), and on a local assessment specific to each reactor (UAAR).

For ASN, ensuring lasting compliance of the 900 MWe reactors with the baseline requirements for the protection of interests through ageing and obsolescence management of their systems, structures and components, implies a particularity with regard to the periodic safety review associated with their fourth ten-yearly outage (VD4 900). This is because some SSCs would be required to function beyond their initial design hypotheses. This is more specifically the case with unreplaceable components such as the reactor pressure vessel and the containment.

Therefore, after 10 years of application by EDF of its ageing management approach implemented on the occasion of the third ten-yearly outage inspection of the 900 and 1300 MWe reactors, both generic and local, and with a view to the possible continued operation of the 900 MWe reactors beyond their VD4, the entire approach is currently being examined by ASN and IRSN so that, in early 2018, the opinions of and any recommendations by the GPR and the GPESPN can be collected regarding this ageing and obsolescence management approach, in order to address the following question: are the steps taken and/or planned by EDF sufficient to ensure ageing and obsolescence management of the SSCs and thus maintain the compliance of the 900 MWe reactors with their baseline requirements for the protection of interests beyond their VD4 and up to their next periodic safety review, in the light of changing knowledge, operating experience feedback and best practices internationally?

With regard to the definition of the ageing management process by EDF**,** ASN has at this stage no particular comments with regard to the ageing mechanisms and the SSCs considered. ASN more particularly notes that in reply to the request made in its letter[26], EDF intends as of the VD4 to expand its approach to include SSCs classified as important for the protection of interests with regard to hazards not only related to radiological accidents, but also conventional hazards and detrimental effects.

In its process, EDF updates the generic AAS annually and updates the component DAARs every five years: this is in line with ASN requirements, because this should enable EDF to incorporate overall OEF from the French NPP fleet, along with any international OEF.

Moreover, in reply to an ASN request made in its letter[27], EDF confirms the introduction of a paragraph concerning the **extended operability criteria** into the generic DAARs, along with a field concerning these criteria in the AARs, on the occasion of their annual reviews from 2015 to 2017 for components which are not the subject of a generic DAAR : these operability criteria are consistent with the maximum acceptable limit for the consequences of an ageing mechanism on a structure, system or component (SSC), and can

[26] ASN, letter CODEP-DCN-2016-007286 of 20 April 2016.

[27] ASN, letter CODEP-DCN-2016-007286 of 20 April 2016.

be translated into an anticipated lifetime. Such criteria are essential decision-making aids when ruling on continued operation beyond the fourth ten-yearly outage inspection.

ASN notes that EDF has set up a ***research and development (R&D) programme*** to support its ageing management process in order to progress and build on knowledge of ageing mechanisms and of the properties of the materials after 60 years of operation. This programme is used to advance engineering practices and the means of monitoring and inspection and also contributes to the development of processes to repair or mitigate the consequences of ageing, as well as to the assessment of removed equipment items. EDF sent the elements to be examined by the ageing Advisory Committee (GP) scheduled for 2018, in reply to the ASN request[28] of 28 June 2013 concerning the identification of ageing phenomena, in particular on the basis of appropriate Research and Development (R&D) programmes.

Given the importance of **non-destructive testing (NDT)** in the ageing management approach, ASN is continuing its efforts to acquire efficient NDT processes for the duration of the service life of its reactors. The main guidelines of the EDF programme with regard to NDT developments, covering the service life extension, are being examined within the context of the preparations for the VD4 900 and ASN has no particular comments at this stage.

The supplementary investigations programme (PIC) implemented in addition to the maintenance strategies and programmes since the VD2-900, is in response to a request from ASN. It aims to confirm the hypotheses adopted concerning the absence of damage occurring during operation in zones not covered by maintenance programmes. In the PIC VD4-900, EDF makes provision for spot checks on zones comprising elements important for protection to take account of ASN's request in its above-mentioned letter of 28 June 2013 concerning the tightening up of the inspection programme. It confirms that the analyses associated with the PIC VD4-900 will be extended to the SSC contributing to the management of risks linked to storage in the spent fuel pit, environmental discharges and those for which a failure could have an impact on the operation of systems performing a post-accident function. As part of the examination for the VD4 900 ageing GP scheduled for 2018, ASN and its technical support organisation IRSN will rule on this PIC VD4-900.

Preventive and corrective measures

EDF's maintenance policy must aim to guarantee that the level of reliability required for equipment and systems is maintained over time, while anticipating certain actions such as to enable an operating life extension up to 20 years beyond the VD4. It covers both "routine" preventive maintenance and "exceptional" maintenance.

The equipment maintenance strategies were therefore supplemented when necessary by exceptional maintenance operations. These operations are scheduled for the period running from the VD3 to the VD4, or even beyond.

Routine maintenance

The preventive maintenance programmes used at start-up of the reactors have changed, taking account of national and international operating experience feedback concerning the behaviour of structures and components, as well as the conclusions of the analyses and those of the findings obtained during inspections or disassembly of equipment and events which occurred during operation.

The pertinence of the EDF routine maintenance strategies was specifically examined. In particular, following EDF's 2010 announcement of its intention to deploy a new maintenance methodology, called

[28] ASN, opinion CODEP-DCN-2013-013464 of 28 June 2013.

AP-913 (Advanced Process 913), drawn up in 2001 by the Institute of Nuclear Power Operations (INPO) with American licensees, the various steps of this methodology and the organisational conditions of its deployment in the NPPs were examined by ASN with the technical support of IRSN. ASN considers that the components classification methodology and the production of maintenance templates and programmes stipulated in the organisation and methodology notes associated with AP-913 is acceptable. However, ASN considers that proactive steps must be taken with the NPPs to allow correct implementation of this new method and ensure that it is effective.

Exceptional maintenance

In addition to the routine maintenance activities, EDF is making provision for exceptional maintenance measures. ASN considers that this approach is in principle satisfactory. Within the framework of the VD4 900 ageing GP, ASN will examine how EDF includes this type of maintenance (repair or replacement) in the AASs of equipment which are not the subject of a generic DAAR and for which a risk of exceeding the serviceability criterion has been identified.

Review and updating of the ageing management process

ASN notes that the ageing management approach for the EDF reactors is based on three long-term operational processes:

- the process to manage the ageing of components implemented as of VD3 and continued in VD4;
- the in-service inspection and maintenance process which takes account of the hypothesis of continued operation of the reactors up until VD4+20 years;
- the process to deal with the obsolescence of equipment and spares.

As described in 2.1.4, ASN and its support organisation IRSN have several times in the past reviewed EDF's ageing management approach. The process reviews and revisions described (see § 2.4) are currently being examined by the VD4 900 ageing GP.

However, at this stage, ASN considers that the specific aspects of the site and of each reactor could be better taken into account in the local ageing management programme (PLMV) and the UAAR.

At the design stage, the Flamanville 3 EPR reactor benefited from knowledge acquired through operating experience feedback from reactors in service and through R&D.

2.8.2 RESEARCH REACTORS

The ageing management of research reactors is elaborated at the scale of each facility. It is currently based mainly on maintenance programmes and periodic inspections and tests. The ASN considers that the management of ageing should be more formalised by research reactor licensees.

The ASN therefore considers that the licensees of research reactors should implement an approach to ensure the sufficiency of the checks and tests carried out and, where appropriate, define the additional checks to ensure that they are able to perform their functions with regard to the ageing mechanisms that could affect the EIPs.

The ASN points out that the RJH has benefited from feedback from other research reactors; significant improvements in the control of ageing equipment have been made.

3 ELECTRICAL CABLES

Summary:

With regard to the electrical cables, the approach adopted by EDF for ageing management covers all the cables needed for operation of the reactors. The degradation mechanisms were studied on the basis of national and international operating experience feedback, as well as R&D on the behaviour of polymer materials.

For the purposes of in-service monitoring, EDF identifies the cables subjected to particular environmental or operational stresses. As necessary, EDF implements specific checks to detect ageing symptoms (measurement of delta tangent and partial discharges for the MV cables, visual inspection of LV cables). ASN considers that the checks carried out are in compliance with the state of the art and are satisfactory.

ASN also considers that the characterisation work carried out on the cables sampled from the EDF reactors, in conjunction with the conclusions of the cable predictive lifetime studies, give a high level of confidence with regard to their ability to retain their original functionality for the next 10 years. This conclusion is consistent with the results of predictive life expectancy studies carried out by EDF R&D, which conclude that cables can maintain their functionality beyond 40 years of operation.

With regard to research reactors, the monitoring of ageing of electrical cables consists mainly of measurements (resistance of measurement lines, insulation resistance of classified cables) and partial checks of the state of insulation by visual inspection. The ASN considers that the ageing management programme for these reactors remains limited and should be completed, particularly for classified cables which are subject to environmental or operational constraints in order to ensure that they are qualified to perform their functions over time.

3.1 DESCRIPTION OF THE EDF ELECTRIC CABLE AGEING MANAGEMENT PROGRAMME

3.1.1 SCOPE OF THE ELECTRIC CABLE AGEING MANAGEMENT PROGRAMME

All the electric cables assigned directly to production on the EDF nuclear sites are subject to monitoring with respect to ageing. Only the electric cables that have no direct link with the production facilities (lighting, telephony, etc.) are not monitored.

Among the electric cables associated with the production facilities, safety-classified and non-classified (NC) cables are treated in exactly the same manner. The reason for this is that as the composition of NC cables is identical to that of safety-classified cables, they have been integrated in the electric cable ageing management programme. Integration of the NC cables therefore allows a wider sampling range and the consideration of cables which are sometimes subjected to more stringent operating conditions.

There are four main groups of electric cables covered by an ageing management programme:

- Medium-voltage (MV) power cables;
- Low-voltage (LV) power cables;
- Measuring cables (LV) other than the specific cables mentioned below;
- Instrumentation and control cables (LV).

The types of cables associated with these groups are described in appendix 10.3.

Specific cables on certain instrumentation connections such as the coaxial electric cables used on the neutron flux measuring channels (RPN) do not fit into the groups described above. These cables are monitored under the ageing management programme of the system with which they are associated.

All these cables were procured during the construction of the EDF nuclear fleet applying generic technical specifications, including severe qualification test sequences. This allowed the installation of cables of very much higher quality than the cable manufacturers' "catalogue" products.

3.1.1.1 METHODS AND CRITERIA USED TO SELECT THE ELECTRIC CABLES SUBJECT TO AGEING MANAGEMENT

No method or criterion is used to select cables: as indicated in the preceding paragraph, all the electric cables associated with the production facilities (classified and NC) are concerned by the ageing management programme.

An initial visual inspection is carried out when the cable ageing management programmes are put in place. It was carried out between 2012 and 2015 for the MV cables and will be carried out over the 2017-2019 period for the LV cables. This initial visual inspection serves to identify the highly-stressed cables. These cables will then be subject to specific monitoring (see § 3.1.1.2.1).

3.1.1.2 METHODS OF IDENTIFYING ELECTRIC CABLE AGEING MECHANISMS

There are several methods with varying levels of maturity for characterising the ageing of an electric cable at a given moment in time. These complementary methods are appropriate for identifying and understanding cable ageing mechanisms and for estimating cable service life.

The cable ageing mechanisms are described in § 3.1.1.2.2.

As indicated in chapter 2, as part of the procedure put in place by EDF to demonstrate its ability to operate the nuclear fleet reactors beyond 40 years, the electric cables form the subject of "Ageing Analysis Sheets" (AAS) and "Detailed Ageing Analysis Reports (DAAR).

3.1.1.2.1 Main diagnostic methods usable on site

Visual inspection

Visual examination is a very simple means of diagnosis that allows detection of the first signs of cable ageing and severe environmental conditions that could speed up the cable ageing process. Visual inspection also enables the state of the cable raceways to be checked (for corrosion).

The technique consists in visually examining the cable (with the naked eye, if necessary using a lamp and/or a magnifying glass). The visual inspection can be supplemented by a tactile inspection intended to assess the flexibility of the cable sheathing materials (sometimes the insulation materials).

The main ageing symptoms looked for during visual inspections are:

- discolouration of the cable or a notable change with respect to the original colour, either generalised or located in a zone with high environmental stresses,
- a change in the external appearance (shiny/matt), a shiny appearance - in comparison with other similar cables - possibly being due to the release of an oily constituent from the sheath,
- the presence of cracks or crazing,
- the presence of injuries or other defects, etc.

This inspection by nature most often involves checking the condition of the outer sheath of a cable. The sheath usually consists of a polymer which is technically less elaborate than the polymer used in the insulant, and the sheath it is more exposed than the insulant. The sheath therefore ages faster than the insulant. If there is no sign of ageing of the sheath, it can generally be concluded that the insulant is not degraded.

Visual inspection is inexpensive and simple to implement. It is therefore a good method for LV cables, given the very large number of cables to check and the absence of high dielectric stresses.

The visual inspection of MV cables is good for detecting severe service conditions (or even accidental injuries) that could speed up the ageing of the MV cables. It is generally accompanied by an "electrical" diagnosis.

Insulation resistance measurements

The insulation resistance measurement consists in measuring the current passing through the insulated when a continuous voltage is applied.

A linear resistance value of 30 Megohm.km is the internationally accepted threshold below which investigations must be carried out to identify the cause of the drop in insulation resistance

The insulation resistance measurement alone cannot constitute a cable operability criterion. The following cases can be mentioned as examples:

- A cable's insulation can have suffered serious degradation without this resulting - in a dry environment - in a drop in insulation resistance.
- Conversely, certain reductions in insulation resistance linked to internal migrations of the insulant components do not affect the dielectric strength or service life of the cable.

There is no established rule between the drop in cable insulation resistance and a drop in its dielectric strength that could lead to a short circuit: the physical phenomena involved are different.

A drop in insulation resistance indicates a possible change in the mechanical and electrical characteristics of a cable, but cannot in itself constitute a criterion for determining cable operability. Moreover, there is no internationally recognised "replacement threshold" adopted by the operators (contrary to the "investigation threshold" of 30 Megohms.km).

The detection of a drop in insulation resistance on a cable must therefore be supplemented by measurements to refine the diagnosis.

Delta tangent measurements

The delta tangent (dissipation factor) value, reserved for MV cables, is the ratio between the resistive current I_r and the capacitive current I_c crossing the insulant. If the insulant is degraded, the I_r/I_c ratio is higher and increases with the voltage applied.

The delta tangent value and how it changes with voltage are used to establish a diagnosis of the overall condition (dielectric losses) of MV cable insulants.

There is an international consensus on the delta tangent measurement. It is used to monitor the degradation of MV cables in the North American nuclear power plants. A 2011 report[29] of the EPRI (Electric Power Research Institute), collating the results of the delta tangent measurements taken on more than 37 North American sites shows that this measurement is suitable for detecting a potential degradation in the insulation of MV cables. All the EDF sites have been equipped with delta tangent measurement test sets since 2013.

The procedures applied (low-frequency alternating current measurements on voltage levels varying from 0.5 to $2U_n$[30]) and the associated criteria have been defined by EDF R&D according to the type of insulant. The two main criteria to monitor are the delta tangent value and the change in this value between two voltage levels. The risk of insulation breakdown on an MV cable is assessed by combining these two criteria (delta tangent value and change in value).

The delta tangent measurement provides an overall assessment of the cable condition. It does not however enable a fault to be located, or to distinguish generalised ageing from a serious isolated fault. Do this the delta tangent measurement must be combined with partial discharge measurements, as detailed below.

Partial discharge measurements

The measurement of partial discharges is also reserved for MV cables. It supplements the delta tangent measurement in order to locate one or more isolated defects along the length of a cable.

Partial discharges are micro-breakdowns occurring in the insulant under the effect of the electric field around certain defects such as microvoids or treeing (associated with the presence of moisture). The principle that permits the locating of partial discharges is comparable to that used in reflectometry.

As with delta tangent measurement, partial discharges are measured using low-frequency alternating current at different voltage levels (commonly 0.5 – 0.75 – 1 – 1.25 and 1.5 U_n). The energy of the micro-breakdowns is measured. It depends on the nature and geometry of the cables.

It is relatively difficult to make a diagnosis of the risk of breakdown of a cable solely on the basis of partial discharges being present. It is also difficult to reveal a trend enabling the residual service life of a cable to be estimated. The measurement of partial discharges will therefore complement the delta tangent measurement to consolidate a diagnosis and above all to locate the zones at risk.

[29] EPRI 1022968: Plant engineering: cable Aging Management program Implementation Guidance, 2011.

[30] U_n =6.6 kV for the cables of the fleet in service – U_n=10 kV for the cables of the EPR series.

For a number of years now the measurement of partial discharges has also become a reference test in the monitoring of MV cables (as has the delta tangent measurement). All the EDF sites have been equipped with partial discharge measurement test sets since 2015.

Time domain reflectometry

The principle of time domain reflectometry (TDR) is to generate an electrical pulse at the end of a cable and to observe the signal reflected from that pulse: the electrical pulse is propagated along the entire length of the cable and reflected when it reaches the other end, but also if it encounters a defect in the insulation or a variation in the impedance of the cable. Knowing the speed of propagation of the pulse, it is possible to determine the location of the defects along the length of the cable.

This technique, which can be used on LV cables, is similar to the principle of partial discharge measurement used on MV cables. It does not enable the state of degradation of a cable to be assessed directly, but is used as a complement to other diagnostic methods.

The measurement is taken using a reflectometer. It is particularly effective on shielded and coaxial connections, and is used by EDF on all the reactors in service for the inspection of the coaxial cables used on the RPN measuring channels, for example.

3.1.1.2.2 Predictive cable ageing models

The predictive modelling of cable service life is an important factor in electric cable ageing management. It is used to obtain an initial estimate of the service life of the cables qualified according to normal or accident environmental conditions (particularly for the EPR) and to assess more precisely the state of degradation of the cables in operation.

Semi-empirical model

Predictive studies of the service life of nuclear power plant cables by EDF R&D began in the 1990's. A first step involved developing a "semi-empirical" model that could provide an overview of all the degradation mechanisms that could affect polymer materials.

The semi-empirical model developed by EDF is based on a mathematical law (evolution law) which describes the evolution (monotonic) over time of a property sensitive to ageing as a function of temperature and dose rate. The various experimental studies performed on cable ageing show that the mechanical properties of the cable constituent polymers, and more particularly elongation at rupture, enable the state of degradation of a cable to be represented with a certain margin.

The evolution law parameters are determined from experimental results obtained during accelerated ageing tests comprising different temperature and dose rate conditions.

The change in elongation at rupture as a function of time and the chosen environmental conditions enables the service life of the cable to be estimated (time after which temps the end-of-life criterion of 50% absolute elongation at rupture is reached - see § 3.1.1.2.4). The curve obtained for the EPR[31] insulant used on the K1 cables is shown below.

[31] EPR insulant: Ethylene Propylene Rubber insulant.

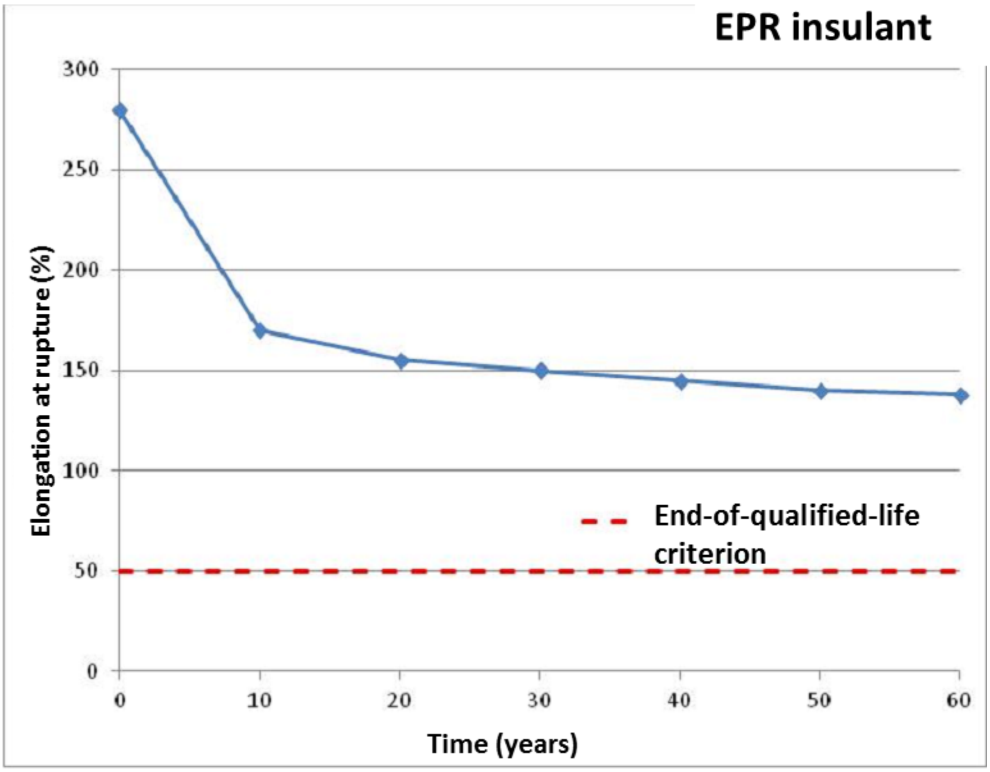

Figure 7 – Change in elongation at rupture obtained with the semi-empirical model for the EPR insulating material and the environmental conditions in the reactor building (50°C, 0.1 Gy/h)

The conservative nature of the semi-empirical model has been confirmed several times by examinations of cable samples taken on site (see § 3.1.1.2.3) after different durations of operational service (up to 35 years).

Multi-scale model

EDF has been contributing to the development of a new analytical model since the 2000's: the "multi-scale" model based on an in-depth understanding of the various polymer degradation mechanisms.

The aim is to simulate the changes in the material on the most appropriate scale for predicting the consequences of the changes on the targeted macroscopic property. Figure 9 illustrates the principle of this multi-scale approach.

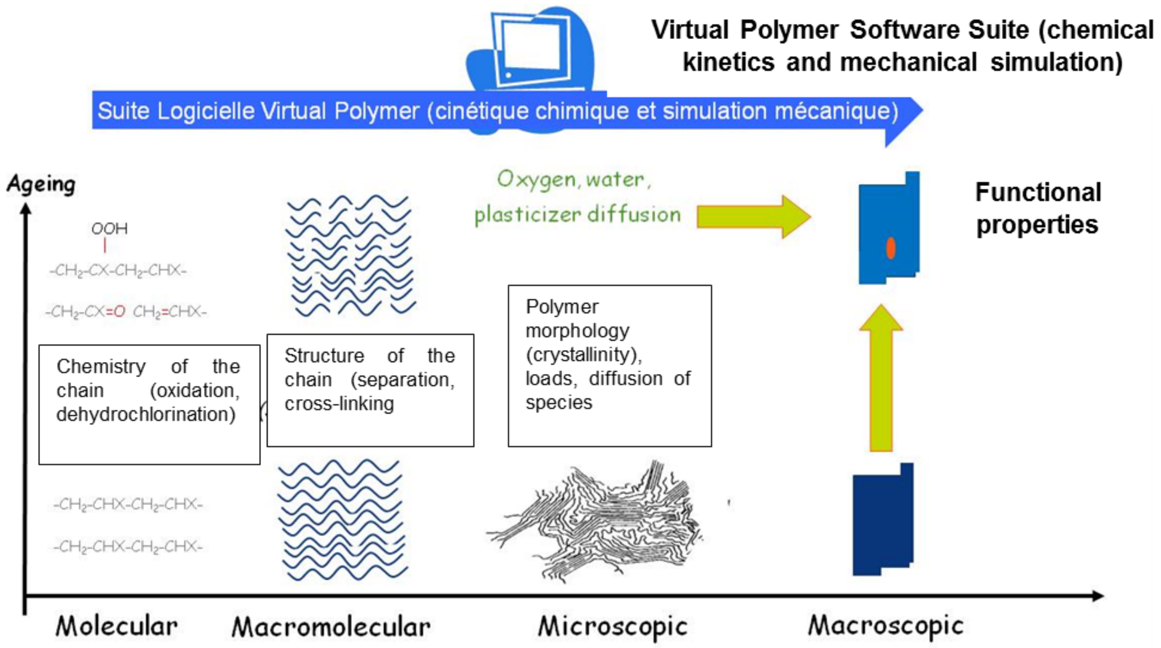

Figure 8 – Schematic diagram of the multi-scale approach

The first step consists in identifying the most appropriate scale (molecular, macromolecular or microscopic) to explain the changes in the macroscopic properties of the studied material. This depends on the type of material, the macroscopic property in question and the type of ageing.

The second step consists in simulating the changes in the material on the scale identified as the most appropriate. The kinetic parameters of all the reactions/processes of interest are determined on the basis of analyses of samples aged at different temperatures or dose rates. These data are obtained from the literature or the EDF laboratories.

The last step consists in using the changes in the material on the molecular, macromolecular or microscopic scale to predict the changes in the macroscopic property that interests us.

At present, models are available for Polyethylene and PVC[32]-based polymers. The work has brought significant advances in the understanding of the ageing phenomena affecting cable polymers. These advances have been made thanks to the work of several theses conducted in collaboration with internationally-renowned laboratories in the field of polymer ageing (Arts et Métiers ParisTech, ESPCI ParisTech, ICMPE CNRS in Thiais, CIRIMAT Carnot Institute, MATEIS in Lyon).

Over the 2017-2022 period, EDF R&D is joining 12 partners including cable manufacturers, licensees, laboratories and universities, in the European "TeaM CABLES" project with two main objectives: the development of a "multi-scale" ageing model that can improve the predictive studies of the service life of halogen-free (HF) cables, and the development of non-destructive testing (NDT) methods for in-situ diagnosis of LV cables. The European Commission approved the funding of this project in early 2017.

3.1.1.2.3 On-site cable sampling

Both the semi-empirical model and the multi-scale model require the predictive model results to be compared with the results of examinations of cables that have aged naturally on site.

[32] PVC: Polyvinyl Chloride.

EDF has been taking cable samples on site for examination purposes for several years. The samples and examinations have improved the understanding of cable ageing mechanisms, confirmed the soundness of the predictive approach and validated cable operability beyond 40 years of service.

EDF has chosen to take samples of cables in service and not to install extra cables (control cables) on site.

The sampled cables are selected such that they are representative of the entire targeted population. For example, the sampled MV cables are cables which:

- pass through the turbine building and/or the exterior;
- have a high operating rate and if possible operate under high loads.

The MV cables are sampled over their entire length, ends included. The LV cables are sampled in sections. The location of each sample taken is precisely identified in order to record the operating conditions (temperature, humidity constraints, etc.) and to identify the cable routing and the zones of stresses (passage through openings for example).

The sampled cables are examined using the diagnostic methods presented in § 3.1.1.2.1 and those described below.

3.1.1.2.4 Laboratory diagnostic methods

In order to have a precise diagnosis of the condition of a cable, the in-situ measurements are supplemented by far more complex laboratory measurements: the electrical measurements are supplemented by physical-chemical and mechanical measurements. The main measurements are described below. Combining these techniques enables the important material parameters to be determined and the condition of the cable insulants to be monitored. At present, all these techniques require on-site cable sampling.

The results of these characterisations are used to validate the understanding of the ageing phenomena specific to each material and the cable service life prediction models.

Mechanical property characterisations

Elongation at rupture (often associated with rupture stress) is the reference indicator of cable insulant degradation. It is obtained by standardised tensile tests (in compliance with standard NF EN 60811-501 for France).

The appropriateness of this mechanical criterion is internationally recognised and the lower limit value of elongation at rupture for the EDF fleet cables is 50% absolute[33]. This value corresponds to the insulated conductor being wound over two times its outside diameter, a configuration that is far more severe than the cable's installation and service configuration (cables are installed with a bend radius of more than 10 times their outside diameter, resulting in induced insulant elongations of less than 10%). These conditions ensure a sufficient margin before electrical faults appear, and also cover other aspects such as potential inhomogeneities in the materials over the length of the cable, or installation imperfections (bending radii).

As measuring elongation at rupture is a destructive technique, the different physical-chemical methods presented below enable the state of ageing of a cable to be characterised more precisely.

[33] AIEA, Assessing and managing cable ageing in Nuclear Power Plants – N° NP-T-3-6, 2012.

Infrared analyses

Infrared (IR) spectroscopy enables polymer ageing to be evidenced through the detection of chemical groupings (such as oxidation products) resulting from the mechanisms of degradation of the cable polymer constituents. IR spectroscopy is performed in attenuated total reflectance (ATR) mode on small samples of polymer materials.

The IR spectra absorption bands for the EPR and XLPE[34] insulants are usually concentrated in the following ranges:

- 1730-1740 cm^{-1}: phenolic antioxidants marker;
- 1715 cm^{-1}: signature of the carbonyl functions produced during oxidation of the polymer;
- 3200-3600 cm^{-1}: signature of the hydroxyl functions (often associated with the presence of hydroperoxides).

The IR spectra absorption bands for PVC materials are concentrated in the following ranges:

- 1190 cm^{-1}: signature of the ester functions of phosphate-based plasticisers;
- 1710-1720 cm-1: signature of the ester functions of phthalate-based plasticisers;
- 1530-1590 cm^{-1} (doublet): signature of the stabilising functions of stearates (carboxylates) introduced into the PVC as thermal stabilisers;
- 1625-1635 cm^{-1}: signature of the stearic acids produced during the chemical consumption of the stabilising functions.

The polymer material samples are also subjected to infrared analyses at different depths using an IR microscope in order to identify composition gradients in the depth of the material and to have a complete mapping of the analysed material.

Oxidation Induction Time (OIT)

One of the commonest causes of chemical ageing of polymer materials is oxidation. It is therefore important to characterise the stability of the materials with respect to the oxidation mechanism.

By using an experimental technique such as DSC[35], it is possible to reveal the heat flux released by a sample during an oxidation reaction. The analysed sample (small sample of polymer) is initially maintained at a controlled temperature and swept with a neutral gas (nitrogen). The oxidation induction time (OIT) is defined as the time lapse between the introduction of oxygen into the furnace containing the sample (neutral gas sweeping interrupted) and the appearance of the exothermic oxidation reaction (schematic diagram below). The OIT is therefore an indicator of the oxidation resistance of the tested material. The OIT measurements are taken in accordance with standard ISO 11357-6.

[34] XLPE: Chemically cross-linked polyethylene.

[35] DSC: Differential Scanning Calorimetry - Technique that can be used to study the thermal transitions that a polymer undergoes when it is subjected to a specified temperature profile.

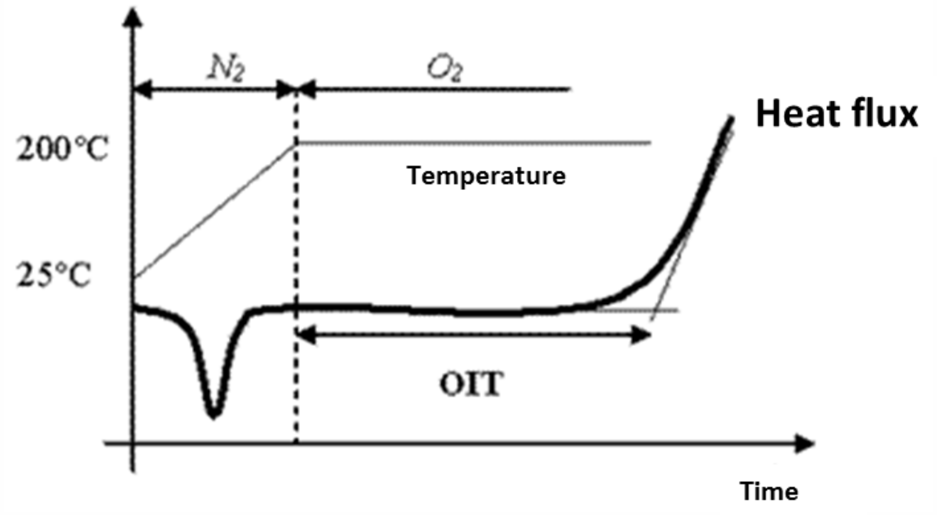

Figure 9 – Schematic diagram of the measurement of oxidation induction time

Relationships can be established between the OIT value and the residual presence of antioxidants in the polymer materials without it being possible to establish the link between the OIT value and the concentration of antioxidants.

The OIT values are highly dependent on the formulation of the materials. No direct relationship has been established between the OIT value and elongation at rupture to date. It is also known that OIT measurements are not suitable for quantifying certain amine-type antioxidants: this is because the OIT is not affected by the concentration of amine-type antioxidants (particularly the HALS[36]).

The OIT is widely recognised internationally as a very early indicator of ageing (that is to say of the first phase of ageing, corresponding to the loss of the antioxidants), i.e. a first early warning sign.

Dehydrochlorination time

Measurements of the thermal stability of PVCs are also carried out on PVC insulants and sheaths. These are measurements of the dehydrochlorination times. The sample tested (0.5 g) is heated to 200°C in an oven swept with nitrogen. The hydrochloric acid (HCl) released by the PVC is diluted by bubbling in water situated in an adjacent compartment (50 ml distilled water) in which the conductivity of the aqueous solution is measured continuously. The dehydrochlorination time corresponds to the time between heating to temperature and the moment a change in conductivity of 50 µS/cm is attained.

The dehydrochlorination times measured on the PVCs of the cables are then compared with the value obtained for a pure PVC (with a thermal stabiliser).

Other properties of polymer materials can be also be measured to track changes in the polymers, such as:

- The melting and glass transition temperatures,
- The degree of crystallinity,
- The soluble fraction,
- The level of plasticisers, etc.

[36] *HALS: Hindered Amine Light Stabilisers*

These analysis results as a whole enable the state of degradation of polymers to be characterised and used as inputs for the models developed by EDF R&D to obtain an estimation of the cable service life.

3.1.1.3 GROUPING CRITERION FOR AGEING MANAGEMENT

All the electric cables on the EDF sites in operation which are directly assigned to the production facilities (classified and non-classified) are monitored with respect to ageing. The criteria used to group these cables originate from the main factors that can affect the ageing of these cables:

- The type and formulation (specific to each manufacturer) of the polymer materials constituting the insulants and the outer sheaths,
- The conditions of operation (mainly temperature, irradiation and humidity).

These grouping criteria satisfy the international recommendations in this area (IAEA[37], EPRI[38]).

The MV and LV cables are treated separately because MV cables are considered more "sensitive" as their insulants are subject to higher dielectric stresses which generate a risk of breakdown.

3.1.1.3.1 MV cables

Management of the ageing of the MV cables installed by EDF takes up the different steps of the cable ageing management programmes proposed by the various international authorities, particularly the EPRI, which come down to three essential steps:

1. Establish the database of the monitored cables;
2. Characterise their service environment and the stress and ageing factors;
3. Choose the appropriate monitoring mode for the cables and the environmental conditions.

The first step consists in identifying all the MV cables installed on the sites and recording their characteristics: identifier, from-end, to-end, length, cross-sectional area, manufacturer, part number, etc.

This was carried out on the EDF fleet between 2012 and 2015 by taking in-situ recordings which inventoried the characteristics of 14,500 MV cables. All the information concerning the MV cables on the national scale is recorded in a database baptised "CABLES HTA" (MV CABLES).

The MV cables are grouped by "batch". A batch is a set of cables grouping a uniform population of cables displaying common intrinsic characteristics. Batching is determined on the basis of the manufacturer, the part number, the type of insulant (PVC, XLPE, etc.), the cable geometry (single-core, three-core), etc.

Certain similar cable types may be grouped together (type 1x400mm^2 and 1x630mm^2), but cables with different architectures (typically single-core and three-core) that could lead to different ageing mechanisms must not.

The year of manufacture is also a criterion, particularly if a range of cables has been manufactured over a long period of time (the formulation of the polymer materials might have changed over the years).

This approach leads to the adoption of about 10 to 15 cable batches per site.

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[37] AIEA, Assessing and managing cable ageing in Nuclear Power Plants – N° NP-T-3-6, 2012.

[38] EPRI 3002000557 (Plant Engineering, Aging management Program Guidance for Medium-Voltage Cable Systems for Nuclear Power Plants, Revision 1, 2013); EPRI 1020804 (Plant support engineering: Aging Management development Guidance for ac and dc low-Voltage power cable systems for Nuclear Power Plants, 2010); EPRI 1021629 (Plant support engineering: Aging Management Program development Guidance for Instrument and Control Cable systems for Nuclear Power Plants, 2010).

As all the cables in a given batch have similar characteristics, the ageing of that batch can be considered uniform. It is therefore possible to estimate the ageing of a batch by conducting sampling inspections.

The second step consists in identifying the operating stresses that could speed up the ageing of MV cables. The stresses in question are:

- Environmental stresses (temperature, humidity, dosimetry, etc.).To be exhaustive, the identification of these stresses necessitates a visual inspection of all the MV cables over their entire length, identifying the areas of actual or potential risks.
- The service stresses (overload, short circuit, etc.).The main service stress concerns the possible overloading of a cable due to its sizing, leading to abnormal heating of the cable which speeds up its ageing.
- The various exceptional stresses suffered by the cables, such as accidental overvoltages or high current transients, are also looked for.

The MV cables are then classified in two categories:

- "Non-stressed" cables: the conditions of installation, operation and the history of these cables are in conformity with the design assumptions. Their behaviour and their qualification are not called into question. For these cables, the preventive maintenance consists primarily in periodically checking the conformity with the design conditions.
- "Stressed" cables: cables in this category are subject to a more severe environment (temperature, irradiation, humidity), cables having undergone repair work, cables installed with too tight a bend radius, overloaded cables, etc. The MV K1[39] cables are placed in this category by default due to the associated implications for safety. These cables undergo preventive maintenance inspections to guarantee their operability (see § 3.1.3.1). They serve as "control cables" for each batch of cables.

3.1.1.3.2 LV cables

There is no short or medium-term risk concerning the operability of LV cables. At present, no significant faults are observed on the reactor fleet in operation, which confirms the findings of the licensees in other countries. Only LV cables subjected to severe operating conditions (temperature, irradiation, humidity) could display a shortened service life.

Due to the very large number of LV cables (more than 20,000 per reactor, which represents over 1,350,000 LV cables for the entire fleet), recording all the cable part numbers is out of the question.

The adopted approach therefore consisted in identifying the severe operating conditions that could affect the cable service life, working on individual room scale, without seeking to identify each cable individually.

An initial visual inspection in all the LV cable rooms, with a subsequent periodic visual inspection in the zones identified as "risk" zones" is prescribed (see § 3.1.3.2).

The cables in the inspected zones are then classified in three categories:

- "Sound": The cables are installed in compliance with the installation specifications and display no visible signs of degradation.
- "Degraded": The cables display installation anomalies, degradations or signs of ageing that do not call into question the availability of the equipment they supply with power.

[39] K1: Reactor Building inside, required in accident conditions

- "Noncompliant": The cables display installation anomalies, degradations or signs of ageing that could call into question the availability of the equipment they supply with power, in normal or accident situations.

These categories are used to define the most severely stressed cables which are then specifically monitored. The level of risk is assessed for a ten-year period (the time between inspections) taking into account the possible aggravation of the observed faults.

Infrared inspections of the LV power connections are also prescribed in order to identify potentially overloaded connections, even if to date there has been no negative feedback concerning overloaded LV cables.

A mapping of the installed LV cables has been produced, recording the part numbers of the LV cables for each group (power, instrumentation & control and measurement) by sampling. A sampling programme for on-site LV cable appraisal has been defined on the basis of this mapping.

3.1.2 ASSESSMENT OF ELECTRIC CABLE AGEING

The following paragraphs detail the assessment of the ageing of the cables installed on the EDF fleet in operation for the following three electric cable categories[40]:

- MV cables,
- LV cables,
- Coaxial cables used on the neutron flux measuring channels (RPN).

The predominant ageing mechanisms affecting electric cables are associated with ageing of the organic materials (polymers) that make up the insulating envelopes and the outer sheaths.

3.1.2.1 MV CABLES

Operating experience feedback and the studies carried out by EDF R&D reveal the following principal ageing mechanisms that can affect the MV cables of the fleet in operation:

1. *Migration of additives*

Some of the additives present in polymers, such as the plasticisers used in the PVC formulations, can migrate over time by exudation or evaporation. A typical example is the exudation of the phthalate or phosphate plasticisers used in PVC.

Loss of the additives results in the loss of the properties they induced. With regard to PVC, migration of the plasticisers leads to a modification of the mechanical characteristics and, in some cases, reductions in insulation.

The various studies carried out to date, along with the results of examinations of cables sampled on site, have shown that the service life adopted for the qualification of the cables concerned is not called into question.

2. *Treeing phenomena associated with the presence of humidity*

For MV cables with Polyethylene-based insulant (XLPE insulant), the humidity/electric field combination causes electrochemical reactions that lead first to the appearance of microfissures, then gradually to the formation of water trees in the insulants. These treeing phenomena can ultimately lead to electrical breakdown.

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[40] It was decided not to restrict the assessment to MV and MV cables operated in severe conditions since the ageing management in place on the EDF sites concerns all the cables assigned directly to the production facilities.

International experience feedback relating to this phenomenon reports a few failures of MV cables subjected to a humid environment in the last ten years. EDF has decided not to bury the cables, thereby preserving them from water. Nevertheless, MV cables passing through galleries or trenches can occasionally be subjected to damp conditions. The measures put in place through the ageing management programmes enable such conditions to be detected and the monitoring of the cables in question to be increased.

3. Dehydrochlorination

This is the main mode of degradation of PVC. It is initiated by temperature under aerobic or anaerobic (absence of oxygen) conditions, and therefore also affects confined materials such as cable insulants.

The dehydrochlorination phenomenon consists in the formation of Carbon-Carbon double bonds and hydrochloric acid (HCl) further to the removal of Chlorine and Hydrogen atoms present in the PVC.

This phenomenon is the main cause of the drops in insulation resistance observed on several sites on MV K3/NC single-core cables with PVC insulation.

The first studies carried out by EDF R&D on MV cables with PVC insulation displaying drops in insulation resistance began in the 1990°'s. These studies have shown that despite the confirmed drop in insulation resistance, the residual dielectric strength of the MV cables remains satisfactory and guarantees their in-service integrity.

As from 2008-2009, studies have been carried out on MV cables aged artificially under thermal and electrical stresses to analyse how the drop in insulation resistance evolves over time. No change in the insulation resistance of the tested cables has been observed.

To further understand these phenomena, a preventive sampling programme for investigations has been launched on a series of MV cables displaying these reductions in insulation.

All the studies carried out confirm that the observed reductions in insulation resistance are effectively linked to the dehydrochlorination phenomenon and that these reductions do not affect the ability of these MV cables to ensure their electrical function, nor do they affect their service life.

4. Oxidation

For cables comprising Polyethylene-based polymers (XLPE, EPR and HF insulants), the predominant mode of degradation is oxidation. The oxidation mechanism occurs at molecular level. This results in the production of compounds called hydroperoxides whose decomposition, due to their instability, degrades the characteristics of the material.

Temperature (thermo-oxidation) and irradiation (radio-oxidation) are factors that influence the oxidation phenomenon, and its kinetics in particular.

To inhibit, or at least slow down oxidation-related degradation, stabilisers (antioxidants) are incorporated in the polymer material formulations.

The chemical reactions of polymer oxidation have consequences on the "macroscopic" characteristics of the material (insulation, dielectric, elongation at rupture, etc.).

To date, the results of the examinations of cables sampled on site show that all the cables, after more than 30 years of operation, are still in the first phase of ageing with respect to oxidation: consumption of the antioxidants. No impact on the macroscopic properties of the polymer materials (elongation at rupture) or the electrical properties has been evidenced.

5. Ageing of the MV cable ends

An increase in the connection resistance can appear on power cables and cause localised heating of the end of the cable. This can result from, among other things, an increase in the contact resistance between the aluminium core of an MV cable and the lug crimped onto this core, due to the formation of a layer of alumina. The localised heating induces premature ageing of the insulant, which becomes embrittled. It can then become fissured during manipulations such as the MV motor disconnection-reconnection operations.

This phenomenon has caused a few electrical breakdowns observed since 2010 on MV NC cables of the fleet in service (mainly the 900 MWe plant series). Inspections have been organised to prevent further electrical breakdowns (see § 3.1.3.1).

6. Conclusion

The operating experience feedback (OEF) concerning the MV cables of the reactor fleet in operation is highly positive. The few minor events mentioned above, plus international OEF, led EDF in 2011 to define a monitoring programme for MV electric cables aiming at:

- Tightening electric cable monitoring in order to ascertain that the cables are installed and operated under optimum conditions which can guarantee their service life with a view to continuing operation to 60 years.
- Improving the diagnosis of cable condition in order to detect any potential ageing problem and undertake preventive replacements if necessary

The actions undertaken comply with international recommendations, and with the EPRI guides published since 2010 in particular. They are detailed in § 3.1.3.1.

3.1.2.2 LV CABLES

LV cables are potentially subject to the same ageing mechanisms as the MV cables, with the exception of the treeing phenomena because the dielectric stresses involved are too low to induce this type of ageing.

At present, no significant faults are observed on the LV cables on the reactor fleet in operation, which confirms the findings of the licensees in other countries. There is no short or medium-term risk concerning the LV cables. Only cables subjected to very severe environmental stresses (humidity, temperature, irradiation) are likely to have a reduced service life.

Consequently, LV cable monitoring was put in place in a second phase (MV cables were considered to take priority) and comes down essentially to visual inspections for detecting severe operating conditions and the first visible signs of cable ageing (see § 3.1.3.2).

3.1.2.3 RPN COAXIAL CABLES

Two types of coaxial cables are used on the neutron flux measuring channels (see figure 10):

- Mineral coaxial cables

These cables comprise metallic and mineral constituents which by nature are not particularly sensitive to ageing, and an outer overcladding in a polymer material (SiXLPE[41] or scotch 27) ensuring "ground/shielding" insulation.

[41] Silane cross-linked polyethylene.

They are used inside the reactor building (RB), between the connector board and the reactor pit outlet on the CPY series and between the connector board and the containment electrical penetration on the N4 series.

- Organic coaxial cables

Two broad types of K2-qualified organic coaxial cables are used:

- The "CP" coaxial cables comprising a polyethylene-based insulant and a PVC sheath;
- The "CZ" coaxial cables comprising a polyethylene-based insulant and a CSPE[42] (Hypalon) sheath;

They are used outside the RB on all the reactors of the EDF fleet. They are also used inside the RB but outside the reactor pit on the 900 and 1300 MWe series, and between the connector board and the reactor pit outlet on the CP0 and 1300 MWe series.

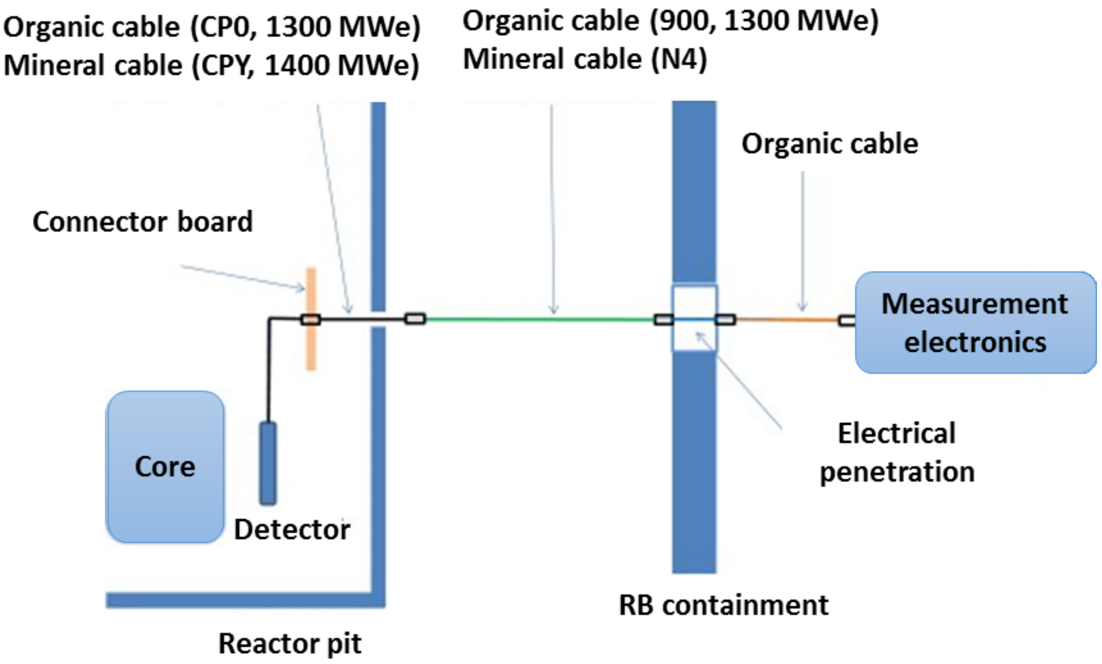

Figure 10 – Utilisation of mineral and organic coaxial cables on the RPN measuring channels of the different plant series

The examinations and measurements carried out as preventive maintenance (see § 3.1.3.3) allow the detection of faults induced by premature ageing of the coaxial cables.

For the mineral coaxial cables, only one confirmed ageing mechanism has been detected in operation. It concerns the outer sheath (SiXLPE or Scotch 27) of the mineral coaxial cables used inside the reactor pit. The outer sheath becomes degraded in the long term under the effect of irradiation and temperature. This degradation leads to noncompliance with the insulation criterion between the cable shielding and ground.

In the event of degradation of the outer sheath of mineral cables (Scotch 27 or SiXLPE) causing a ground/shielding insulation fault, the cable can be repaired by roller-pressing with Scotch 27 or using Raychem WCSF tape. The mineral connections can also be replaced if repair with Scotch 27 is difficult (operations involving exposure to radiation).

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[42] CSPE: Chlorosulfonated polyethylene.

For the organic coaxial cables used outside the RB and inside the RB but outside the reactor pit, operating experience feedback is highly positive since insulation faults are very rare.

As the organic coaxial cables are made up of polymer materials similar [43] to those of "conventional" LV cables, the R&D studies performed on materials such as Polyethylene, Hypalon and PVC can be used. These R&D studies have shown the ability of these materials to conserve their operability beyond 40 years of operation under environmental conditions complying with RCC-E[44]. To consolidate this demonstration, organic coaxial cables were sampled for examination in 2016 and 2017. The results of these examinations aiming at confirming the satisfactory behaviour of organic coaxial cables will be available in the course of 2018.

With regard to the organic cables operated inside the reactor pit, ageing of the outer sheath is observed (hardening, sometimes with the appearance of cracks) after about 20 years of operation under the effect of temperature and radiation, leading to ground/shielding insulation faults. They will be replaced every 20 years or so, depending on results of the inspections (see § 3.1.3.3).

3.1.3 MONITORING, TESTING, SAMPLING AND INSPECTION ACTIVITIES

3.1.3.1 MV CABLES

The monitoring programme for MV cables implemented since 2011 in accordance with international recommendations is based on the following activities:

- Constituting a database inventorying the 14,500 MV cables installed on the Fleet.
- Performing a initial visual inspection of all the MV cables with the aim of detecting zones where the routings display risks.
- Supplying the sites with diagnostic test sets for taking "Delta Tangent" and "Partial Discharge" measurements.
- Applying a preventive maintenance programme for the MV cables, prescribing the periodic sampling inspection of the MV cables using the diagnostic test sets.

The MV cable sampling inspection procedure is as follows:

- The cables of a reactor are listed exhaustively. The list of cables and their characteristics is recorded in a database called "CABLES HTA" (MV CABLES).
- The cables are grouped in "batches" according to their characteristics (see § 3.1.1.3.1).
- "Stressed" cables are sampled from each batch.
- Diagnostic measurements are carried out periodically on each cable in the sample.
- The measurement results enable a risk level to be defined for each batch of cables. The "low risk"[45] or "moderate risk" batches are specifically monitored. The "high risk" cable batches are subject to corrective measures (repair or replacement if necessary).
- The database is updated periodically, re-assessing the risk levels.

[43] The manufacturer of the coaxial cables of the 900 and 1300 MWe plant series (PRECICABLE) also provided the Control and Measuring cables for these plant series, cables made in PVC qualified K2 at the standard dose of 250 kGy.

[44] RCC-E : Rules for the Design and Construction of Electrical Equipment for Nuclear Units.

[45] The levels of risk are defined in accordance with the criteria adopted for the delta tangent and partial discharge measurements (§ 3.1.1.2.1).

A verification of the sizing of the MV cables was also carried out. No significant overload of the MV cables was detected on the 900 MWe series. The same procedure has been undertaken on the 1300 MWe and N4 series.

To confirm the absence of risk on the MV cables operated under conditions complying with RCC-E and to validate the results of the predictive service life studies, a programme for sampling MV cables for appraisal was initiated in 2011. This programme aims at covering all the ranges of cables with PVC, XLPE and EPR (K1) insulation. Only the HF cables, which are more recent (N4 and EPR series), are not yet included in this programme but ultimately they will be.

The results of the examinations carried out to date confirm the excellent behaviour of the cables (see § 3.2.1).

With regard to the ageing of the MV cable ends, several types of inspection were put in place on site during the MV motor disconnection/reconnection operations:

- Visual inspection of the cable ends,
- Delta tangent measurements,
- Verification of the temperature of the cable ends.

The cable temperature verification is carried out by systematic installation of temperature strips on the MV motor cable ends. Inspection of the cable end (and repair if necessary) is requested if there is a significant rise in the temperature recorded by these strips.

Further to the implementation of delta tangent measurements in 2015, several electrical breakdowns have been avoided, confirming the value of these inspections.

3.1.3.2 LV CABLES

There is no easy- to-apply and unanimously-approved diagnostic method for monitoring the ageing of LV cables (despite a recent IAEA[46] working group on this subject in which EDF R&D participated). As LV cables do not present a risk as far as reactor service life extension is concerned, except in cases of very severe stresses, the monitoring of LV cables essentially comes down to visual checks to detect these stresses and the first visible signs of cable ageing (discolouration, cracks, etc.).

The preventive maintenance programme put in place on the EDF Fleet prescribes:

- The initial visual inspection of all the premises with the aim of identifying the zones of stresses and the condition of the cables and cable raceways;
- A periodic ten-yearly visual inspection of the zones of stresses identified by the initial inspection;
- Thermographic checks with reactor under power to detect any overloaded power cables[47].

The visual inspection serves to detect the main symptoms of ageing or degradation visible on the outer sheaths of the cable (discoloration, change of appearance, cracks, crazing, etc.)

The visual inspection also concerns the cable raceways, on which the absence of corrosion of the metal structures and their attaching parts, the condition of the earthing connections and cable attaching parts.

[46] IAEA CRP (Coordinated Research Project) started in 2013. As part of this CRP, the document entitled "Condition monitoring, and management of ageing of low voltage cables in nuclear power plant life management" was published in November 2017.

[47] It is impossible to consider carrying out individual dimensional verification of more than 3000 LV power cables on each reactor. The chosen solution is therefore to carry out thermographic inspections under the power switchboards to detect any overloaded cables.

The visual inspection allows the identification of stressful environmental conditions that could speed up cable ageing: high ambient temperature, closeness of a high-temperature pipe, humidity (even flooding), high dosimetry, etc.

A procedure indicates the exact nature of the inspections to carry out (and appended to that procedure is an inspection form to fill out for each room or each stress zone).

The cables are classified in three categories: "sound", "degraded" or "noncompliant" (see § 3.1.1.3.2), in order to determine the cables that shall be subject to specific monitoring.

Each site carries out a periodic visual inspection:

- every 4 refuelling shutdowns, of the cables situated in the immediate vicinity of the main steam isolation valves;
- every ten years, of the stressed zones identified during the initial inspection and all the cables classified as degraded (or noncompliant and not yet corrected);
- every ten years, of the overloaded cables.

To confirm the absence of risk on the LV cables operated under conditions complying with the initial qualification, a sampling programme for LV K3/NC cables has been developed on the basis of the mapping of LV cables produced by sampling. The aim is still to cover the different ranges (manufacturers) of LV cables present on site and to distinguish the three families of LV cables (power, instrumentation & control and measurement).

Given that the I&C and measuring cables are not subject to thermal stresses associated with the load, cables that can be removed with the reactor under power, or even disused cables, are favoured.

For the LV K1 cables (which represent less than 5% of the LV cables installed but necessitate specific monitoring due to the associated safety risk), it was decided not to produce a detailed mapping of the installed cables because there are only a few qualified cable manufacturers. For the 900 and 1300 MWe series, only 3 cable product lines have been qualified: a sampling programme specific to the K1 LV cables aims at covering these three cable product lines.

3.1.3.3 RPN COAXIAL CABLES

The programme for monitoring coaxial cables for measuring the neutron flux is based on the following examinations and measurements, carried out as part of the preventive maintenance of the RPN measuring channels:

- visual inspection of the condition of the cables in the reactor pits;
- measurement of the core/shielding insulation;
- measurement of the shielding/ground insulation;
- inspection of cable condition by reflectometry;
- continuity measurements.

These checks are carried out every four refuelling shutdowns at least. They allow the detection of defects resulting from premature ageing of the coaxial cables. Repair or replacement solutions are identified where applicable.

As with the LV electric cables, organic coaxial cables were sampled on site outside the reactor pits for examination in order to confirm their good behaviour and their operability beyond 40 years of service.

3.1.4 PREVENTIVE AND CORRECTIVE ACTIONS FOR ELECTRIC CABLES

3.1.4.1 MV CABLES

No large-scale replacement of MV cables is envisaged as a preventive measure: only MV cables subjected to severe operating conditions (humidity, temperature and irradiation) would be likely to be replaced over the period of operation of the reactors.

In-service monitoring of the cables allows the identification of MV cables that are subject to such conditions and verification of their electrical characteristics (delta tangent and partial discharge measurements). In the event of confirmed ageing of a cable, it is replaced over its entire length or in sections.

3.1.4.2 LV CABLES

As with the MV cables, no large-scale replacement of LV cables is envisaged as a preventive measure: only LV cables subjected to severe operating conditions would be likely to be replaced over the period of operation of the reactors. The in-service monitoring of cables enables the LV cables that are subject to such conditions to be identified. In the event of confirmed ageing of a cable, it is replaced over its entire length or in sections.

To give an example, a highly specific case of initial under-sizing has been found on the cables supplying the pressuriser heaters. This concerns about a hundred cables which are now undergoing preventive replacement. Some of these cables display significant visible signs of degradation. Nevertheless, no failure has been observed on these cables despite a theoretical overload of up to 150%, confirming the substantial margins that exist on the LV cables produced to stringent technical specifications.

3.1.4.3 RPN COAXIAL CABLES

The ability of the mineral coaxial cables to fulfil their function is not called into question. As the repair of the organic overcladding (XiXLPE or Scotch 27) of the mineral coaxial cables is a costly operation[48] involving exposure to radiation, and due to obsolescence affecting the SiXLPE, entirely mineral triaxial cables were qualified by EDF in 2013. Further to this qualification, the cables will gradually be replaced as from 2017:

- On the 900 and 1300 MWe series, the replacement of the intermediate neutron flux measuring channel cables is planned over the period from 2017 to 2026,
- On the N4 series, replacement of all the neutron flux measuring channel cables is planned over the period from 2017 to 2029,

The organic coaxial RPN cables used outside the reactor pit are not subject to preventive or corrective actions to date due to the positive OEF from these cables and the results of predictive service life studies relative to their constituent polymer materials.

The organic coaxial cables used inside the reactor pit (CP0 and 1300 MWe series), which are subject to severe stresses (temperature, irradiation), are replaced every 20 years approximately (observed frequency), depending on the results of the monitoring (see § 3.1.3.3).

[48] In some cases this repair necessitates the opening of the cable raceways over their entire length, including the opening of the fire protections.

3.2 EDF EXPERIENCE IN THE APPLICATION OF ITS ELECTRIC CABLE AGEING MANAGEMENT PROGRAMME

3.2.1 MV CABLES

The OEF further to implementation of the MV cable ageing management programme on the EDF sites is positive to date.

Only two faults on MV cables associated with severe environmental conditions have been identified:

- In February 2012, an electrical breakdown occurred on a 1300 MWe reactor on a connection cable between the auxiliary transformer and the auxiliary boiler in a damp zone of a cable trench.
- In November 2015, the Delta Tangent measurements taken on MV cables supplying the auxiliary boilers of a 900 MWe reactor gave values very much higher than the permissible criteria. The premature ageing of these cables is associated with a high overload and the thermal stresses of the boilers.

As far as the ageing of the MV cables ends is concerned, the measures implemented have prevented several electrical breakdowns, confirming the value of the inspections performed.

The predictive studies carried out by EDF R&D on the various insulating materials used in the MV cables (PVC, XLPE, EPR) conclude that operating these cables beyond 40 years under environmental conditions complying with RCC-E poses no particular problem.

The examinations performed on the MV cables sampled on site have confirmed the good overall condition of the cables. A total of 12 MV K3/NC cables and 2 MV K1 cables (RRA)[49] have been sampled and examined to date on the 900 and 1300 MWe series.

The analyses of all these cable samples show no significant change in their properties after more than 30 to 35 years of operation on site. The mechanical characteristics of the polymer materials remain within the required range of values when new and are therefore very much higher than the end-of-qualified-condition for the cables (50% elongation at rupture).

All the insulants characterised (EPR for the K1 cables, PVC or XLPE for the K3 cables) still contain - after 30 to 35 years of operation - stabilising species which protect them against oxidation (EPR and XLPE insulants) and dehydrochlorination (PVC insulant).

At this stage of ageing, microscopic analyses (infrared measurements in the depth of the materials) are the only means capable of revealing an onset of loss of these stabilising species, an "expected" phenomenon after 30 to 35 years of operation. They show that polymer material constituents of the cables are still in the first phase of ageing, that is to say losing the stabilising species. The macroscopic properties (elongation at rupture) and electrical properties are not affected. These results confirm that the MV electric cables operated under normal conditions of temperature, humidity and irradiation display good behaviour. These results consolidate the results of the R&D studies which conclude that the MV cables can conserve their operability beyond 40 years of operation.

3.2.2 LV CABLES

The visual inspections prescribed for LV cables shall be carried out over the 2017-2019 period.

As with the MV cables, the predictive service life studies carried out by EDF R&D conclude that operating LV cables beyond 40 years under environmental conditions complying with RCC-E poses no particular problem.

[49] K3: Reactor Building inside, required in accident conditions / NC: not classified.

Two K1 LV cables have been sampled (from the RRA[50] system) and examined to date, one of which has been in operation for more than 30 years and subjected to a cumulated ageing radiation dose of 58kGy.

The results of the analyses are identical to those obtained for the MV cables:

- After more than 30 years of operation on site, there is no significant change in the properties of the polymer materials. The mechanical characteristics are close to those required when new and still very much higher than the end-of-qualified-condition criterion for the cables;
- The stabilising species are still present in the EPR insulants, protecting them from oxidation;
- At this stage of ageing, the microscopic analyses (infrared measurements in the depth of the materials) are the only means capable of revealing an onset of loss of the antioxidants. They show that the polymer materials are still in the first phase of ageing: loss of the antioxidants. The macroscopic properties (elongation at rupture) and electrical properties are not affected.

The cumulated radiation dose received by the K1 LV cable had no significant identified impact on the characteristics of its polymer material constituents.

These results confirm the excellent behaviour of the LV cables used under normal conditions of temperature, humidity and irradiation. These cables can conserve their operability beyond 40 years of operation.

3.2.3 RPN COAXIAL CABLES

The OEF concerning mineral coaxial cables is positive. These cables shall gradually be replaced by triaxial mineral cables as from 2017 (on the intermediate neutron measurement systems for the 900-1300 MWe series, and all the neutron measurement systems for the N4 series) in order to avoid repairing the overcladding by roller-pressing with Scotch 27 (a costly operation that exposes workers to radiation).

The OEF for the organic coaxial cables used outside the reactor pit is highly positive (insulation defects very rare).

Only the organic coaxial cables used inside the reactor pit display signs of ageing of the outer sheath under the effect of temperature and radiation after some twenty years of operation. These cables are replaced every 20 years approximately, depending on the monitoring results.

3.3 AGEING MANAGEMENT OF RESEARCH REACTOR CABLES

3.3.1 CEA

3.3.1.1 SCOPE OF PROGRAMME AND AGEING ASSESSMENT

The electric cables of the research reactors operated by the CEA are monitored by checking the insulation resistance of the neutron measurement cables of the neutron safety monitoring system and the failed fuel element detection systems. The insulation resistance measurement (change in leakage current) allows ageing of the cable and its connections to be monitored.

The cables are not subjected to mechanical stresses or to splashing with chemical products or to external hazards that could cause degradation. The low voltage network is monitored by permanent insulation monitoring systems and differential circuit breakers. These systems detect general insulation faults on the network and are not assigned solely to the monitoring of cable ageing.

[50] RRA: Residual heat removal system.

On the Jules Horowitz Reactor (JHR), the cables that are potentially subjected to radiation are specified by making conservative use of the qualifications performed by EDF which are described in the "Technical specifications book (CST) for electric cables for nuclear power plants".

On the CABRI reactor, the cables are largely inaccessible insofar as they are routed inside troughs under several layers of cables or in conduits. No cables are routed directly on the ground.

3.3.1.2 MONITORING, PREVENTIVE MEASURES AND LICENSEES' EXPERIENCE

The verification of the insulation of these detector / conditioning rack measuring cables is carried out during the annual maintenance performed by the licensee. The insulation resistance measurement (change in leakage current) allows the ageing of the cable and its connections to be monitored.

Partial verifications of the LV power supply and instrumentation & control cables (visual condition of the insulant) are carried out each year during the regulatory inspections, the defects are then signalled to the licensee in the work reports.

3.3.2 LAUE LANGEVIN INSITUTE (ILL)

3.3.2.1 SCOPE OF PROGRAMME AND AGEING ASSESSMENT

The cable performance and qualities must satisfy the requirements defined in the baseline safety requirements. The cables are poorly accessible for inspection but the pull boxes enable them to be withdrawn from their conduit if necessary.

The medium voltage cables and the measuring cables (about 100 cables covering 100 m for each of the two categories) are routed through buried conduits with pull boxes enabling cables to be pulled again if necessary.

The incoming high voltage cables (20 kV) are routed through trenches and buried conduits. They are not classified as EIPs (elements important for protection) and were recently replaced to increse their rating 15 kV to 20 kV.

All the cables in the reactor must be fire-retardant. If any original cables do not satisfy this requirement, they are replaced when work is carried out.

The performance factors verified are the line resistances, the insulation and the sealing of containment penetration cables.

3.3.2.2 MONITORING, PREVENTIVE MEASURES AND LICENSEES' EXPERIENCE

The EIP cables are checked systematically. The cables necessary for the availability of the facility are checked on a sampling basis.

The preventive actions consist essentially in replacing the oldest cables and those having suffered degradation (irradiation, mechanical stresses, etc.). The curative actions consist in repairing or replacing cables that do not satisfy the prescribed performance requirements.

The inspections have revealed some cables whose cladding material has become embrittled through ageing after about forty years, through exposure to gamma radiation in the case of the neutron detector splice connectors. The number of cables having had to be replaced is very small.

3.4 ASN ASSESSMENT OF THE ELECTRIC CABLE AGEING MANAGEMENT PROGRAMME

3.4.1 THE NUCLEAR POWER REACTORS

3.4.1.1 AGEING MANAGEMENT PERIMETER

The cable ageing management perimeter adopted by EDF covers all the cables that play a functional role in the facility: the elements provided go beyond the perimeter of the specification. The consideration of cables that are not safety-classified increases the size of the sample of monitored cables and also enables more severe operating conditions to be taken into account.

The proposed groups of cables are consistent with those proposed in the WENRA specification (the LV power cables are to be likened to the medium-voltage cables of that specification). To conclude, ASN underlines the extended perimeter of the cables covered by the ageing management programme with regard to the requirement (classified cables).

3.4.1.2 CABLES WITH POLYMER INSULANT

3.4.1.2.1 Ageing assessment

The ageing management programme for cables with polymer insulants is more detailed for the cables of the 900 MWe and 1300 MWe plant series than for the more recent cables (cables installed on the N4 series or the Flamanville 3 EPR).

The ageing management approach is similar: the differences in the polymer materials are found chiefly in the monitoring of different physical-chemical markers (antioxidants, oxidation products, carbon-carbon double bonds) and by different modellings of the physical-chemical degradation of these polymers (chemical reactions and associated kinetic constants). Identification of the predominant ageing phenomena has enabled the manufacturers to improve cable behaviour (improvement in the materials). Application of the ageing management programme will enable this improvement to be confirmed.

The modes of degradation considered by EDF raise no comments on the part of ASN. These degradation modes have been identified through:

- R&D actions aiming at studying the physical-chemical degradations of the polymer materials used in the French nuclear power plants (EPR, EP, EPDM, CSPE or Hypalon, EVA, PVC and XLPE),
- national and international OEF (exudation of PVC cables from certain supply sources, electrical breakdown on MV connectors) having led to the furthering of prospective R&D on these subjects and reinforcing of the monitoring programmes.

EDF has been particularly attentive in its consideration of the oxidation of K1 cable insulants (EPR, EVA and EPDM insulants in particular), as R&D work has revealed a phenomenon of autocatalysis of the oxidation on polyethylene-based materials beyond a given oxidation product concentration. The dehydrochlorination phenomenon has been considered since the beginning of the 1990s, reductions in electrical insulation having been observed on certain highly temperature-stressed electric cables of the reactor fleet in operation.

Alongside this, EDF has also carried out substantial modelling work with the aim of better assessing the kinetics of the degradation phenomena and evaluating the margins with respect to the feared phenomena. **ASN emphasises the good practice of EDF which, as a complement to the utilisation of the empirical elongation at rupture criterion, has developed an approach aiming at acquiring early indicators of ageing.**

3.4.1.2.2 Verification, testing, sampling and inspection activities

EDF has developed physical-chemical characterisation methods for polymer materials (insulants and sheaths) and models aiming at better characterising the oxidation phenomena. The majority of these analyses (FTIR[51], oxidation induction time and dehydrochlorination time in particular) are performed on cables sampled in service, which necessitated identifying the most stressed cables (temperature and radiation) and the different cable supply sources (these phenomena depend on the chemical formulation of the insulant).

Measurements of the mechanical characteristics of the insulant (elongation and stresses at rupture) are carried out to check there is no significant deterioration in these characteristics and that the material elasticity remains sufficient.

The ageing phenomena of the connector contact areas are monitored in service, with the recommendation that measures be taken to monitor the temperature of the highly-stressed power connections (thermography or heat-sensitive strips). **Although this is an indirect measurement, ASN considers that the methods used comply with the state of the art and are satisfactory.**

With regard to degradations of low-voltage cable insulant associated with particular operating conditions, EDF is currently orienting its R&D work towards the development of non-intrusive electrical diagnostic methods that can be used in service. The difficulty lies in detecting and locating the faults before the key electrical characteristics of the cable become degraded (insulation resistance and continuity in particular). The different monitoring methods have been analysed in the European project ADVANCE and developments on this subject are currently in progress. The main characteristics of the cables (continuity, insulation resistance) are moreover verified through periodic tests of the instrumentation channels, the instrumentation & control cables and the electric actuator control and supply channels.

Developing and deploying new on-site diagnostic methods for the "low voltage" cables are considered satisfactory by ASN.

ASN has no other particular comments concerning taking account of the modes of degradation in terms of verification, testing, sampling and inspection actions.

3.4.1.2.3 Preventive and corrective actions

<u>900 MWe and 1300 MWe PWR</u>

At present, ASN notes that the characterisations carried out on the cables sampled on site give a high level of confidence in their capacity to conserve their operability for the next 10 years.

ASN notes that alongside these characterisation measures, EDF is continuing to develop methods aiming at detecting singular points on the low-voltage electric cables (reflectometry for example). These methods should in the future be used as a complement to the abovementioned method for locating degradations in poorly accessible areas which are subject to particular stresses (bending radii, heating).

<u>K1 and K2 cables (EPR/Hypalon)</u>

With regard to EPR/Hypalon cables, in view of the results of the physical-chemical characterisations performed on cables sampled on site, ASN considers that no large-scale replacement of these cables is necessary.

ASN considers however that the loss of antioxidants in the insulant and the long-term behaviour of the K1 cables (in the reactor building) under accident conditions (after several days) require consolidation.

[51] FTIR: Fourier Transform InfraRed spectroscopy

Specific samples shall be taken from the most highly stressed low-voltage cables beyond the 4th ten-yearly outage with this in mind.

Coaxial cables with polymer insulant (extension of the neutron flux measuring channels)

The RPN cables (coaxial cables) undergo insulation resistance measurements and verification by reflectometry to assess the attenuation of the pulsed signal transmitted over the link. The results of the physical-chemical characterisations (OIT, FTIR spectrometry, density) and mechanical characterisation (elongation at rupture/ultimate tensile strength) of the measuring cables sampled on site (results planned for 2018) will allow a decision regarding their operability beyond 40 years of service. This raises no comments from ASN.

K3 and K2 cables (PVC/PVC and XLPE/PVC)

With regard to the K3 cables and the K2 MV cables (PVC or XLPE insulant and PVC sheath), EDF's proposed monitoring programme for MV cables "at risk" (high load and damp environment) is based on electrical measurements (delta tangent and partial discharge measurements). The decision criteria adopted by EDF are more severe than those recommended by the EPRI. ASN considers that the measures adopted by EDF for MV cables will enable the cables sensitive to the electrical breakdown risk to be identified and regularly monitored. It moreover notes that additional checks are currently prescribed to verify that MV cable end heating remains low. The cable samples taken aim at covering the different supply sources and the most stressed cables, and they are considered to be appropriate.

Cables of the N4 series

With regard to the K1/K2 and K3 cables with halogen-free insulant installed on the N4 series, EDF plans for characterisations on cables sampled as part of the N4 series 2nd ten-yearly outage (20 years of operation). The characterisation methods will be appropriate for the type of insulant used (EVA-EPDM). ASN has no comments to make at this stage.

3.4.1.3 MINERAL INSULANT CABLES (RPN)

The degradations observed on the mineral insulant (alumina) cables are associated primarily with:

- inappropriate maintenance procedures (tightening of connectors in particular) which tend to deteriorate the condition of the connectors and allow the ingress of humidity into the alumina,
- the ageing of the polymer materials used to isolate the metallic cable sheath from the cable raceway (limitation of electromagnetic interference).

In view of the observed degradations, EDF has deployed:

- new tools which limit mechanical stresses during connector tightening operations and recommends systematically replacing seals (with new ones) following connection/disconnection operations. **ASN underlines this good approach**;
- new mineral links (triaxial cables) aiming at overcoming the ageing problems with the polymer materials used to isolate the metal cable sheath from the cable raceway. **ASN considers this approach to be satisfactory.**

3.4.1.4 REGULATORY OVERSIGHT EXPERIENCE

As a complement to the thematic ageing management inspections (see paragraph 2.7.1.1), ASN also checks that EDF takes measures to ensure the ageing management of electric cables during its on-site inspections devoted to other topics such as electrical and instrumentation & control systems (about 10 per year) or maintenance (about 10 per year) and its work site visits during reactor outages.

Furthermore, the oversight of reactor outages (see paragraph 2.7.3) also gives ASN the opportunity to verify the maintenance activities and the correction of deviations relating to electric cables, particularly with respect to degradations associated with ageing mechanisms.

3.4.2 RESEARCH REACTORS

The monitoring of electric cable ageing consists primarily in monitoring the resistance of the measuring lines, the insulation resistance of classified cables and performing partial checks of cables by visual inspection of the condition of the insulants (if in doubt about their state of degradation [observed brittleness or exudation for example], the cables are replaced).

ASN considers that the ageing management programme for these reactors remains limited and should - for the classified cables - be supplemented by characterisation of their environmental stresses (humidity, temperature, radiation) and their operating conditions (overload, short circuit, etc.) in order to identify the cables that could suffer accelerated ageing and, where applicable, the controls necessary to verify their capacity to fulfil their duties. More specifically, in view of the lessons learned from the degradations observed on certain cables, the following measures should be implemented:

- a diagnosis to assess the propensity to electrical trees associated with the presence of humidity ("water tree" phenomenon) on power cables situated in "damp" zones (trench for example);
- identification of the areas subject to severe temperature stresses (heating by joule effect) and, if necessary, take measurements to characterise the change in the mechanical characteristics of the insulant/sheath in these areas;
- monitoring of the temperature near the highly stressed power connectors (frequent connection/disconnection in operation) to ensure there is no heating.

4 CONCEALED PIPEWORK

Summary:

With a view to continuing reactor operation beyond 40 years, EDF has implemented a work programme to reinforce the ageing management of underground or poorly accessible piping. The first results will be presented to the Advisory Committee of Experts in March 2018. The procedure implemented by EDF is based on the North-American programme detailed in Guideline NEI 09-14 and an experiment conducted on a pilot site before being deployed on the other NPPs in the fleet.

With regard to research reactors, concealed pipes are not classified as important for safety. These are mainly pipes for industrial effluents or water. Some of them are subject to periodic inspections and verifications (checking of seals in particular). Managing the ageing of these pipes is therefore not a safety issue.

4.1 DESCRIPTION OF EDF AGEING MANAGEMENT PROGRAMME FOR CONCEALED PIPES

4.1.1 SCOPE OF THE AGEING MANAGEMENT PROGRAMME FOR CONCEALED PIPES

The "Buried and Poorly Accessible Pipes" programme comprises an initial stock-taking phase which aims at taking the inventory of pipes buried in the ground or placed in poorly accessible or inaccessible trenches (the pipes on nuclear sites are assigned to a basic functional system). The pipes can be gravity flow or pressurised pipes.

For each piping system, input data are collected in order to gather the information available on the pipes that are buried or installed in poorly accessible or inaccessible trenches. For example, the collected data concern the pipe location, material, nominal thickness or diameter. The data relative to the pipe environment, such as the physical-chemical soil data, can also be collected.

All the data are then entered into the BPWORKS software developed by the EPRI so that the pipes can be placed in hierarchical order according to the risk they represent, distinguishing the probability of occurrence of a failure and the consequence associated with a failure. The ageing mechanisms concerned are the degradations initiated via the inside or the outside of the pipe (leak, fracture or blocking).

4.1.2 ASSESSMENT OF THE AGEING OF CONCEALED PIPES

4.1.2.1 MATERIALS USED

As a general rule, pipes that are buried or installed in trenches are made of steel (stainless steel, carbon steel) or ductile cast iron.

4.1.2.2 MODE OF AGEING

The main mode of ageing of buried or entrenched pipes is corrosion from the inside and/or the outside.

4.1.2.2.1 Overall corrosion rate

The overall corrosion rate is the sum of the internal and external corrosions. The rate of corrosion is determined on the basis of ultrasonic measurements by comparing the thickness on a given date with the initial thickness.

4.1.2.2.2 Corrosion from the outside

Corrosion from the outside is due to:

- the atmosphere in the case of pipes in trenches;
- contact with the soil in the case of buried pipes.

The corrosion can be widespread or localised. Corrosion rates can be found in standards or EPRI documents:

- standards ISO 9223 and 9224 for atmospheric corrosion;
- EPRI corrosion tables for soils.

4.1.2.2.3 Corrosion from the inside

Corrosion from the inside is due to the fluid transported.

4.1.2.3 AGEING ANALYSIS PROCEDURE

4.1.2.3.1 Description

The overall ageing analysis procedure is based on determining the corrosion rates by taking in-situ measurements in order to judge the acceptability of continued operation of the pipes.

The risk is analysed using the BPWorks software developed by the EPRI.

The software divides the pipe into discrete sections on the basis of its characteristics and its environment.

The software evaluates a probability of failure for each pipe section according to the characteristics of each pipe (fluid transported, operating conditions, surrounding soil data, level of the water table, etc.).

In a second phase, BPWorks places the consequences of a failure of each section of the pipe in order of importance, particularly in terms of safety, security, loss of production, downtime, the existence of an emergency route, degradation history, repair cost, etc. If some data are not available, BPWorks by default considers a conservative value for the analysis.

The software then correlates the failure probabilities with the failure consequences in a risk matrix and thus provides a risk analysis that enables the pipe lines and their sections to be placed in order of sensitivity:

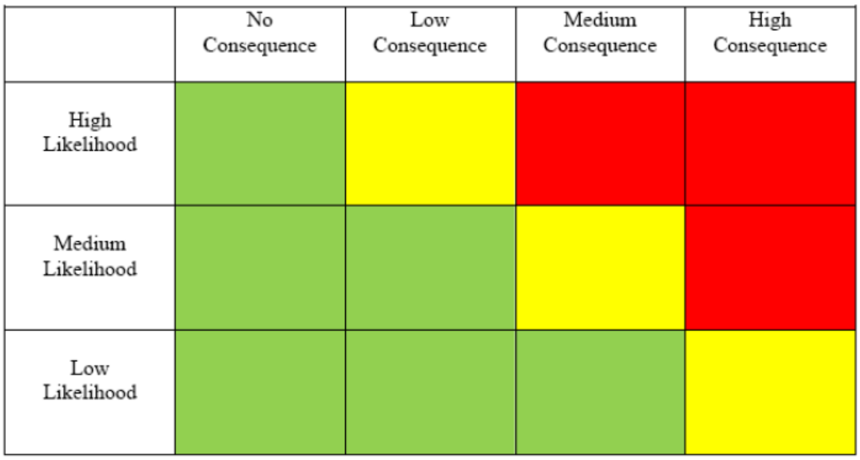

Figure 11 – BPWorks matrix of failure probabilities

The software uses a points system to estimate the risks of failure: points are assigned to each consequence of the failure (safety, security, loss of production, etc.). The risk level of a failure is assessed according to the total number of points.

Number of points N	Risk
$N \leq 20$	Low
$20 < N \leq 40$	Moderate
$40 < N$	High
$N = -999$	N/A Not Applicable

The pipes baptised "TRICE" (toxic, radioactive, inflammable, corrosive, explosive) undergo an additional analysis to ensure that the risk level is appropriate. The user then uses the results of the risk analysis and the site OEF (integrating the confirmed risks) to locate the areas in which the inspections must be carried out.

EDF uses the results provided by BPWorks as a basis for defining an on-site inspection programme. The most critical sections are visually inspected on the basis of this programme, then ultrasonic measurements are taken.

According to the sections inspected and any ageing discovered, a Fitness for Service (FFS) analysis is carried out to judge whether the pipe is fit for service for a period going beyond the 4th ten-yearly outage up to 60 years (i.e. fitness for continued operation criterion). This methodology is taken from the ASME (American Society of Mechanical Engineers) Code Case N-806. It is also used in the oil industry and is referenced in API 579.

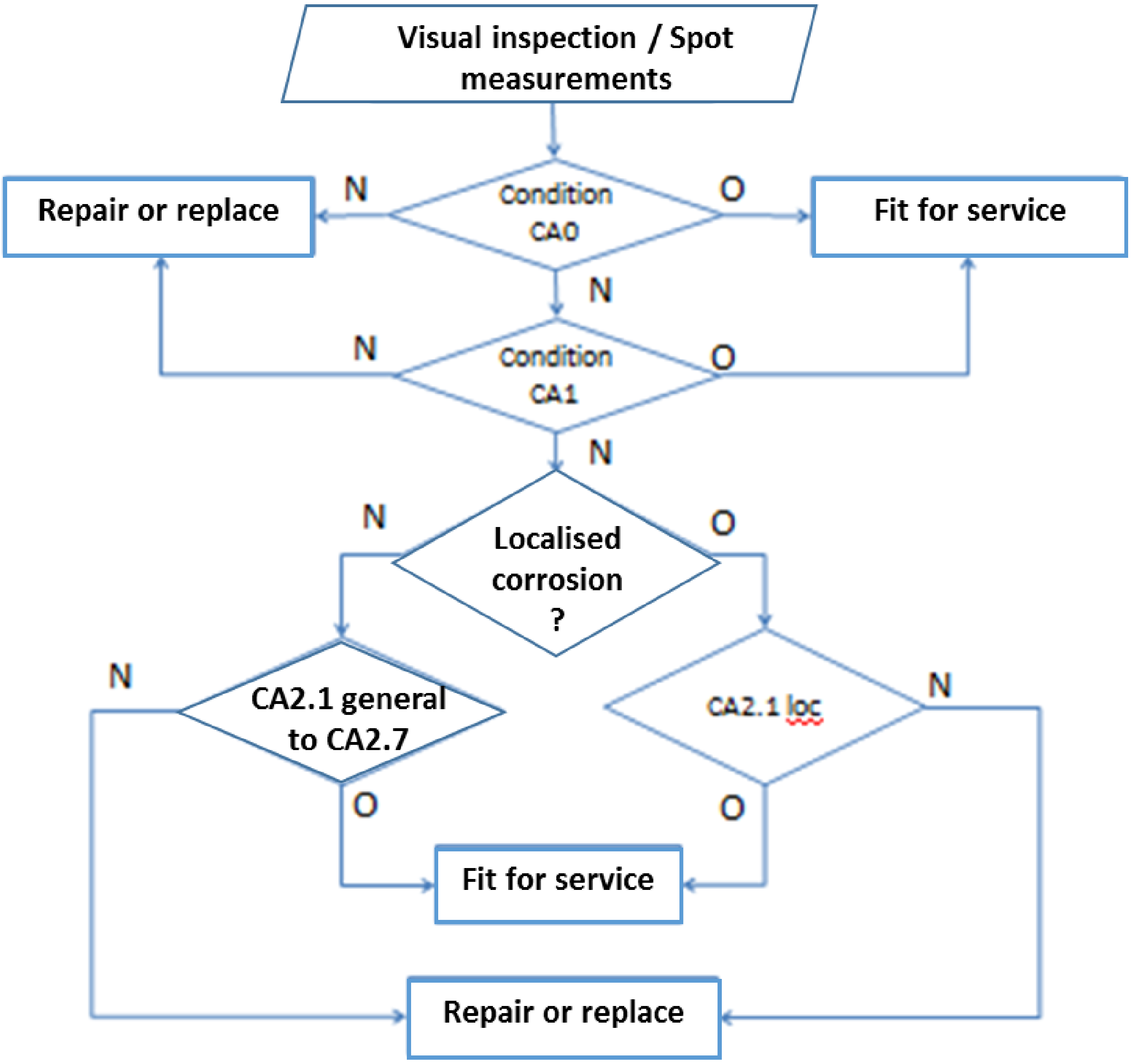

Figure 12 – Fitness for service assessment methodology

Proceeding to the next study condition (CA.X) or replacement or repair of the pipe is associated with a project decision. Expressed briefly, the conditions CA.X shown in the Figure 12 flow chart are validated taking into account the acceptance criterion defined by the FFS method. They are summarised below:

- Condition CA0: the thickness measurements are greater than or equal to the nominal thickness of the pipes. It can therefore be concluded that no corrosion has occurred over a long period of operation and the risk can be excluded for a longer period of operation under similar operating conditions.
- Condition CA1: the thickness measurements are less than the nominal thickness, meaning that corrosion is suspected (measurements within the manufacturing tolerances) or confirmed (measurements below the lower limit of the manufacturing tolerances), and condition CA0 is therefore not validated. A corrosion rate is determined from the measurements made and compared with EPRI data. The mechanical strength of the piping is checked on date t and on a later date t+Δt on the basis of the minimum measured thickness and the predicted thickness. The minimum values of the measurements and predictions at t+Δt are adopted for this analysis. Two situations are possible:
  - the predictions at t+Δt are higher than the manufacturing tolerances (usually +/-12.5%): it can be concluded that the in-service behaviour of the pipes will be satisfactory for the additional period Δt considered. The predictions at t+Δt are lower than the manufacturing tolerances but remain higher than the minimum thickness required under the loading conditions taking into account the methodology of the ASME Code Case N-806: it can be concluded that the in-service behaviour of the pipes will be satisfactory for additional period Δt considered.
- Condition CA2: the thickness measurements are less than the minimum thickness, therefore condition CA1 is not validated. The methodology of the ASME Code Case N-806 is applied and two situations are then examined according to the nature of corrosion:
  - in the case of localised corrosion, corresponding to the "localised" CA2 condition, the behaviour of the piping is checked against the pressure (CA2.1 "loc"),
  - in the case of widespread corrosion, the behaviour of the piping is checked for several load cases. For this "generalized" CA2 condition, in addition to the pressure analysis, additional checks (CA2.2 to CA2.7) are carried out depending on their applicability according to the pipe configuration (buried or trench-type installation, load case, surface loads, depth, ovalisation, etc.). The ASME Code Case N-806 methodology enables the analysis to be carried out on the basis of representative averages obtained from more precise measurements taken in the corroded zone.

4.1.2.3.2 Operating experience feedback

This methodology is in the deployment phase on the EDF nuclear fleet. The EPRI Buried Pipe Integrity Group meetings in which EDF participates enable the licensees to interchange on practices relating to buried pipes; EDF has thus seen that KEPCO, the South-Korean licensee, also uses the American approach:

- Risk analysis using BPWorks;
- Service life analysis using the Fitness For Service method.

4.1.3 MONITORING; TESTING, SAMPLING AND INSPECTION OF CONCEALED PIPEWORK

Inspections on nuclear sites are carried out on a sampling basis; the inspection areas are defined according to the results of the risks analysis study and the pipe accessibility limitations. The inspections are most often carried out by making excavations in the ground or openings in the trenches in order to gain access to the pipe to inspect. The inspections consist in visual (or televisual via the inside) examinations of the visible part of the pipes and pipe thickness evaluations (thickness measurements by ultrasounds for example).

4.1.4 PREVENTIVE AND CORRECTIVE ACTIONS FOR CONCEALED PIPEWORK

The preventive and corrective actions are not defined at this stage of programme progress.

4.2 EDF EXPERIENCE IN THE APPLICATION OF ITS AGEING MANAGEMENT PROGRAMME FOR CONCEALED PIPES

For each of the 19 EDF sites, the buried pipe inspection programmes are established site by site and begin with the 900MWe series. The programmes for the first sites to reach their 4th ten-yearly outage (VD4) (i.e. Tricastin, Bugey, Fessenheim) are available.

The inspections are in progress on the Tricastin, Fessenheim and Bugey sites, with the aim of defining a generic programme of verifications and being able to decide during VD4 whether the buried pipes are fit for continued service or the need to be renovated.

4.3 AGEING MANAGEMENT OF CONCEALED PIPES OF RESEARCH REACTORS

4.3.1 CEA

The principle applied at the design stage is to avoid embedding the pipes in concrete. Moreover, appropriate measures are taken to facilitate in-service tests and inspection.

There are buried secondary cooling lines which are not important for safety insofar as cooling can be ensured by natural convection.

<u>On the JHR reactor</u>, the only buried pipes are those of the tertiary system and the potable water supply system, systems which are not important for safety.

<u>On the CABRI reactor</u>, the concealed pipework buried in the ground do not concern safety (municipal water supply, sanitary effluents drainage system, stormwater drainage system). Only the industrial effluents drainage system undergoes an inspection every 3 years (televisual inspection + water tightness test) with regard to environmental protection.

4.3.2 LAUE LANGEVIN INSITUTE (ILL)

The objective of the ageing management programme is to check that the mechanical behaviour and pressure resistance are guaranteed in all the situations defined in the safety baseline requirements.

The pipes are all accessible apart from the river water supply and return lines which cool down the installations and are buried over a length of about one hundred metres. These pipes are not important for protection.

The sealing of the unions, clamps actuators and sensors are verified periodically. The sealing is also verified indirectly through measurement of the impurities present in the system fluids.

4.4 ASN ASSESSMENT OF THE AGEING MANAGEMENT PROGRAMME FOR CONCEALED PIPES

4.4.1 THE NUCLEAR POWER REACTORS

The degradation of buried pipes, which by nature are poorly accessible, could cause a failure of certain reactor safety functions. It is therefore necessary to be well informed of their state of ageing.

With a view to continued reactor operation beyond 40 years, EDF has undertaken an ageing management programme for buried or concealed pipework in addition to its monitoring provisions. Its objective is to define a programme of tightened verifications and, if necessary, rehabilitation for the first of the 4th ten-yearly outages of the 900 MWe reactors.

The methodology adopted by EDF is based on a risk analysis approach to these pipes, taking into account the consequences of their failure, particularly on safety, radiation protection and the environment. It is based on the North-American programme for which the broad lines are defined in guide NEI 09-14, namely:

- The analysis and hierarchical ordering of the risk;
- the defining of an inspection programme (thickness measurements by ultrasounds, corrosion rate measurements, visual and televisual inspections, hydrostatic tests and new inspection methods currently under development are envisaged);
- the defining of a rehabilitation programme.

The perimeter of the approach includes the buried pipes (including those embedded in civil engineering structures), the uninspectable and poorly inspectable pipes; the inaccessible and concealed pipework and the uninspectable pipes in trenches.

The pipes considered in this work programme include more specifically the conduits transporting TRICE (toxic, radioactive, inflammable, corrosive or explosive) fluids, the fire-extinguishing water conduits and the raw water conduits.

ASN notes that the EDF procedure is based on a generic approach carried out on the Bugey pilot site which, once validated and adapted, will then be applied site by site taking into account their particularities. Based on a method developed by the North American licensees, the method used by EDF takes French and international OEF into account.

Its application is currently being examined by ASN and IRSN, its technical support organisation: the first results of the approach adopted by EDF will be presented to the Advisory Committee of Experts for Reactors (GPR) in March 2018.

4.4.2 RESEARCH REACTORS

Concealed pipework are not classified as important for safety. These are mainly pipes for industrial effluents or water. Some of them are subject to periodic inspections and verifications (checking of seals in particular). Managing the ageing of these pipes is therefore not a safety issue.

5 REACTOR PRESSURE VESSELS

Summary:

The approach adopted by EDF for management of its ageing was the subject of numerous examinations between 1987 and 2015. This approach is based on monitoring and on mechanical studies. These enable the loadings to be identified, more specifically in the irradiated core zones, with selection of the most severe thermal-hydraulic transients. ASN considers this approach to be satisfactory.

National and international operating experience feedback has already led to the implementation of preventive measures (reduction in the neutron flux at the reactor pressure vessel hot spot) and corrective measures (replacement of all the RPV closure heads equipped with Inconel 600 adapters following the discovery in 1991 of stress corrosion on an RPV closure head adapter in Bugey), or the implementation of appropriate checks or monitoring (for example, inspections carried out on the French RPVs following the discovery of microcracks on RPVs of Belgian nuclear power reactors.

As part of its ageing management programme, EDF has identified the ageing mechanisms through national and international operating experience feedback, the results of inspections and tests performed on the reactors in operation and permanent measurements taken on the equipment. On the basis of the ageing analyses contained in the AASs, EDF drafted a Detailed Ageing Analysis Report (DAAR) for the reactor pressure vessel on the 900 MWe and 1300 MWe stages, which deals with radiation-induced ageing of the core zone and thermal ageing of the outlet nozzles.

The monitoring of the equipment of the reactor coolant system and thus of the reactor pressure vessel, is covered by specific regulations (NPE) defining the general requirements and requiring the implementation of:

• in-service monitoring measures demonstrating that the equipment is functioning in the same situations as provided for in the design (situations accounting),

• checks to detect flaws harmful to the integrity of the equipment; these checks take account of the susceptibility of a zone to a degradation mode (including the risk of fast fracture) and of national and international operating experience feedback,

• a programme to monitor the properties of the materials having an impact on the integrity demonstration: the irradiation-induced ageing mechanism is the subject of a specific programme to monitor changes in the properties of the reactor pressure vessel steel (mechanical tests on test pieces of the RPV material subjected to neutron radiation in the vessel).

Within the context of the NPP operating life extension, EDF transmitted a file demonstrating the possibility of prolonging the operating lifetime of the reactor pressure vessels for a period of 10 years following the VD4. This file is currently being reviewed and the conclusions are expected in 2018. ASN will issue a ruling in its generic position statement on the VD4.

With regard to the research reactors, the ageing problem is not comparable to that of the nuclear power reactors. The pressure vessel is a replaceable item and is not therefore a limiting factor in the lifetime of the facility. At each periodic safety review, the licensees reassess the lifetimes of the components according to the fluence received and replace them as and when necessary.

In addition, checks are carried out to ensure that there is no corrosion: this corrosion can stem from the quality of the water or the susceptibility to intergranular corrosion for high neutron flux levels.

5.1 DESCRIPTION OF THE REACTOR PRESSURE VESSEL AGEING MANAGEMENT PROGRAMMES

5.1.1 SCOPE OF THE REACTOR PRESSURE VESSEL AGEING MANAGEMENT PROGRAMME

The reactor pressure vessel (RPV) is a pressure equipment item that is part of the Main Primary Cooling System. It accommodates the RPV internal equipment and the fuel assemblies. It contributes to the containment of the radioactive primary coolant within the second safety barrier. It also contributes to maintaining the internal equipment in position and the circulation of primary coolant cooling the core and controlling its reactivity.

The RPV comprises two main components: the RPV body and the RPV closure head.

To this can be added the parts of the closing system (studs); body/closure head sealing is ensured by two concentric silver-coated metal seals.

With regard to the RPV closure heads, it should be noted that they have all been replaced in service on the 900 MWe and 1300 MWe series (see § 5.1.2.7).

The following paragraphs summarise the main descriptive elements concerning the RPVs of the EDF fleet and the scope of the corresponding ageing monitoring programme.

5.1.1.1 DESIGN / MANUFACTURING PROVISIONS

The design / manufacturing provisions applicable to the procurement of components have evolved between the launching of the nuclear programme in the 1970's and the construction of the EPR plant series. They are summarised in the table below:

<table>
<tr><th rowspan="2">Plant series</th><th rowspan="2">Sub-series</th><th colspan="2">RPV body</th><th colspan="2">RPV closure head</th></tr>
<tr><th>Design</th><th>Manufacture</th><th>Design</th><th>Manufacture</th></tr>
<tr><td rowspan="2">Reactors</td><td>CP0</td><td rowspan="2">ASME</td><td rowspan="2">CPFC</td><td colspan="2" rowspan="4">RCC-M (replacement closure heads)</td></tr>
<tr><td>CPY</td></tr>
<tr><td rowspan="2">1300 MWe</td><td>P4</td><td>ASME</td><td>CPFC</td></tr>
<tr><td>P’4</td><td colspan="2">RCC-M</td></tr>
<tr><td colspan="2">1450 MWe - N4</td><td colspan="4">RCC-M</td></tr>
<tr><td colspan="2">1650 MWe – EPR</td><td colspan="4">RCC-M</td></tr>
</table>

Table 9 – Applicable design / manufacturing provisions

5.1.1.2 CONSTITUENTS AND MATERIALS

The general structure of the RPV and its constituents are on the whole similar for the different plant series. The schematic diagrams of the RPVs with the usual part identifications are shown in appendix 10.4.

The types of materials used for the RPVs of the various plant series are similar; the details per series are shown in appendix 10.5:

- low allow steel grade nuance 16MND5 (equivalent SA508 cl.3) for the flanges, shell rings, domes and nozzles;
- stainless steel for the nozzle safe ends, the vent tube and the leak monitoring tube;

- nickel-base alloy for the closure head adapters, the radial guides, the closure head vent pipe and the bottom-mounted instrumentation penetrations;
- high tensile steel for the closure components (studs).

5.1.1.3 GENERAL MANUFACTURING INFORMATION

For all the plant series, the parts in low alloy steel are forged or made from rolled sheet metal. The RPV features no rolled or welded parts, therefore there are no longitudinal assembly welds. All the low alloy steel components are made from forged ingots or, in the case of certain parts (lower domes and upper domes), from rolled sheets. The following particularities can be noted:

- 4 replacement RPV closure heads of the 900 MWe plant series are monobloc parts, i.e. the flange / dome assembly has been forged directly from a solid ingot. This does away with the flange / dome weld;
- Some shells have been forged from a hollow ingot, which limits the presence of residual carbon-rich segregate zones on the surface to be cladded, thereby reducing the risk of cold fissuring flaws forming beneath the cladding. The shells involved are the core shell (N4 series RPVs + 2 RPVs in the 1300 MWe series) and the nozzle support rings (N4 series RPVs);
- The upper section of the EPR pressure vessel corresponding to the reactor vessel flange A / nozzle support ring B assembly was forged from a single solid ingot, which means there is no need for a welded joint in the thickest part of the RPV body and enables the set-in weld of the nozzles (saddle weld) to be replaced by a set-on weld (circular weld), which is easier to inspect.

The manufacture of the RPV involves the production of several types of welds:

- Homogeneous strength welds: these are welds joining low alloy steel parts. They are produced by an automatic process with a filler metal of the same nature as the base metal;
- Heterogeneous strength welds: these are welds joining stainless steel or nickel-base alloy parts to ferritic steel parts. There are two main types of such welds: welds joining nickel alloy-base parts to ferritic parts, produced with a filler metal of a similar grade to that of the joined part (i.e. Inconel 600 or 690 alloy equivalent);
    - Welds joining stainless steel safe ends to the ferritic nozzles (henceforth called dissimilar metal welds). These welds can involve a stainless steel filler metal (900 MWe and 1300 MWe RPVs and one N4 RPV) or a nickel-base alloy filler metal (three N4 RPVs and the EPR RPV);
- Coating deposition: all the surfaces of the ferritic sections are coated with a stainless steel deposit[52], applied in two layers by an automatic flux/strip-electrode process or manually by coated electrode welding in areas with more complex geometry. Local repairs have been carried out manually with a stainless steel filler metal or, more rarely, with a nickel-base alloy filler metal.

With regard to the welds, the following particularities can be noted:

- For the 900 MWe, 1300 MWe and N4 plant series, the nozzles are in a set-in configuration, whereas for the EPR series they are in a set-on configuration,
- On 4 replacement RPV closure heads of the 900 MWe series there is no connection weld between the closure head and the upper dome (monobloc forging),

[52] With the exception of the Inconel-coated transition ring of the EPR RPV, to which the radial guides are welded.

- On the EPR RPV, there is no connection weld between the RPV flange and the nozzle support ring (monobloc forging),
- On the EPR RPV, there are no bottom-mounted penetrations for the in-core neutron instrumentation, as this is introduced via penetrations welded to the closure head.

The main dimensional characteristics of each plant series are summarised in the table of appendix 10.6.

5.1.1.4 OPERATING CONDITIONS

The operating pressures and temperatures of the various plant series RPVs are relatively similar (see table in appendix 10.6):

- Nominal operating pressure of 154 bars rms.
- Cold leg temperature at 100% Pn: 286°C to 292°C
- Hot leg temperature at 100% Pn: 324°C to 330°C

There are nevertheless particularities regarding the conditions of operation of the upper dome (closure head):

- The 900 MWe, 1300 MWe and N4 reactors function in "cold dome" mode, that is to say that dome sweeping flow comes primarily from the fluid of the cold legs. It is considered that the closure heads of these reactors function at the cold leg temperature.

  The 900/CP0 and EPR reactors function in "hot dome" mode, that is to say that dome sweeping flow comes primarily from the fluid exiting the core. It is considered conservatively that the closure heads of these reactors function at the hot leg temperature.

5.1.1.5 EQUIPMENT AT THE SYSTEMS, STRUCTURES AND COMPONENTS (SSC) INTERFACE AND PERIMETER FOR THE AGEING MONITORING PROGRAMME

All the constituent parts of the RPV body and closure head presented above, and their joining welds, lie within the perimeter of the RPV SSC for the ageing monitoring programme.

The RPV is moreover interfaced with other items of equipment or civil engineering structures which are summarised below. Unless otherwise specified, the interface zone is not included in the perimeter of the RPV ageing monitoring programme.

The 900 MWe / 1300 MWe / N4 plant series (whose interfaces are similar) are distinguished from the EPR series, which displays some significant particularities in the instrumentation interfaces.

	Location of the interface	
Interface equipment	**900 MWe / 1300 MWe / N4 plant series**	**EPR series**
Primary coolant pipes	Homogeneous connection weld between primary coolant pipe and nozzle end-piece	
Vent pipe	Homogeneous connection weld on closure head vent tube	
Drip recovery pipe	Homogeneous connection weld between recovery pipe and RPV flange drip recovery tube	

In-core neutron instrumentation	Weld joining the instrumentation guide-tubes to the bottom-mounted instrumentation penetrations	Screwed connection on RPV closure head adapter flange. The thin-lipped seal weld (CANOPY seal) lies within the RPV SSC perimeter
Thermocouple instrumentation tube	Not applicable	Screwed connection on guide-tube for the thermocouples connected to the branch pipe welded to the RPV closure head.
Thermocouple columns	Screwed connection on RPV closure head adapter flange. The thin-lipped seal weld (CANOPY seal) lies within the RPV SSC perimeter	Not applicable
Control rod drive mechanisms	Screwed connection on RPV closure head adapter flange. The thin-lipped seal weld (CANOPY seal) is part of the RPV SSC perimeter	Bolted connection on RPV closure head adapter flange (Conoseal gasket)
RPV support ring	The RPV rests on the support ring anchored to the civil engineering structure via tubes equipped with support pads forged from the same stock as the parts	
RPV-to-pool seal ring	Weld between the RPV seal ledge (not subjected to pressure) and the seal ring (welded to the pool skin)	
RPV closure head lifting system	Lifting lugs welded to RPV closure head: the lugs and their welds lie within the RPV SSC perimeter	

Table 10 – Interface equipment and locations

The ageing management programme put in place through the Ageing Analysis Sheets (AAS) and the Detailed Ageing Analysis Reports (DAAR) has been carried out on the 900 MWe and 1300 MWe plant series Nevertheless, for all the plant series the ageing phenomena are taken into account in the maintenance doctrines for the preparation of the inspection programmes and the substantiation studies (taking account of the effects of ageing on material properties).

The main ageing mechanisms identified on the RPV and analysed under the ageing management programme are summarised in the table below on the basis of the list of AAS's for the 900 MWe / 1300 MWe plant series In view of the similarities regarding the general design and the conditions of operation of the RPVs in the EDF fleet, this list of mechanisms is also relevant for the N4 and EPR RPVs.

Each zone / ageing mechanism pair is analysed in an AAS and given a classification (0, 1 or 2), depending on whether the mechanism is confirmed or not, the maintenance measures applied and the difficulty of repair or replacement. An AAS of status 2 leads to the creation of a DAAR for the component, which is the case with the RPV for which two AAS's of status 2 are identified (irradiation ageing of the core zone and thermal ageing of the outlet nozzles).

5.1.2 ASSESSMENT OF REACTOR PRESSURE VESSEL AGEING

The reactor pressure vessel ageing mechanisms are identified on the basis of knowledge of their conditions of operation, French and international operating experience feedback, and the results of inspections performed on the reactors in service.

Mechanism	Component(s) analysed
Irradiation ageing	Core zone shells and welded joint
Thermal ageing	Parts in low alloy steel (RPV body flange and closure head, nozzle support rings, lower and upper domes, nozzles, transition flange, welded joints) Dissimilar metal welds with dilution zone anomalies on the outlet nozzles
Fatigue	Shells and welded joint in core zone, RPV and closure head flange, tappings, nozzles, lower and upper domes, studs, thin-lipped CANOPY seals
Boric acid corrosion	Parts in low alloy steel (RPV closure head and wall, RPV lower head)
Atmospheric corrosion	Dissimilar metal welds
Stress corrosion in primary system environment	Zones in nickel-base alloy: radial guides, bottom-mounted instrumentation (BMI) penetrations, closure head adapters and vent, nozzle repairs in Inconel Zones in stainless steel: thin-lipped seals of adapters

Table 11 – Analysed ageing mechanisms and components

These various mechanisms are taken up below for the different zones of the pressurised metal casing.

The assigned classification of the zone / ageing mechanism pair is indicated. The particularities of a plant series with regard to each ageing mechanisms are specified where applicable.

5.1.2.1 IRRADIATION AGEING OF THE CORE ZONE (SHELLS AND WELDED JOINT)

The phenomenon of radiation embrittlement of low alloy steels with a ferritic structure is characterised by a displacement of the brittle-ductile transition RT_{NDT} towards increasing temperatures and can be accompanied by a reduction in the energy at break in the ductile range. This embrittlement therefore affects the strength of the material with respect to the fast fracture risk, which is the design-basis damage mode in the core zone (core shells and their joining weld).

The main parameters influencing this embrittlement are:

- the neutron dose,
- the radiation temperature,
- the chemical composition of the material.

Irradiation ageing is assessed using an embrittlement formula ("FFI" formula) developed from numerous results obtained under the "PSI" programme (programme monitoring radiation effects) implemented on the French RPVs (see § 5.1.3.1).

This formula enables the shift in the ductile-brittle transition temperature to be expressed from the chemical composition of the material and the fluence received at the targeted analysis term.

The shift in transition temperature calculated in this way is included in the input data for the fast fracture studies of the core zone, which are updated for each plant series every ten years at least. Each RPV's fitness for service with respect to this ageing mechanism is established when the regulatory fast fracture resistance criteria are confirmed.

The monitoring provisions for this ageing mechanism are described in § 5.1.3.1.

The AAS is classified status 2 for the shells and the irradiated welded joint.

5.1.2.2 THERMAL AGEING OF LOW ALLOY STEELS

The thermal ageing mechanism which can over the long term affect the low alloy steels of type 16MND5 constituting the RPV at PWR operating temperatures is the segregation of embrittling impurities at grain boundaries

The embrittlement induced by this mechanism results in an increase in the ductile-brittle transition temperature RT_{NDT} accompanied by a change in fracture mode in the brittle range: the mode changes from cleavage fracture to intergranular fracture. This embrittlement affects the resistance of the material with respect to the fast fracture risk.

The effect of this ageing mechanism on the ductile-brittle transition temperature is established on the basis of the MacLean model and depends on the temperature, operating time and phosphorous content of the component. The conservative nature of the predictions established using this model is consolidated by R&D programmes focusing on experimental ageing that are representative of 60 years of operation at least.

This mechanism is not effective at temperatures below 300°C. In view of the operating conditions of the PWR reactors in the EDF fleet, only the zones functioning at the hot leg temperature might be concerned, namely:

- The outlet nozzles,
- The RPV closure heads functioning in hot dome mode (900 MWe / CP0 and EPR plant series), although their temperature nevertheless remains below that of the hot leg.

The following points can be noted:

- The EPR plant series has more stringent chemical specifications that those of the older reactors, particularly with regard to the phosphorous content, thereby limiting the effects of thermal ageing,
- The replacement RPV closure heads of the 900 MWe / CP0 plant series have very low phosphorous contents, which makes them relatively insensitive to thermal ageing.

The shift in transition temperature due to thermal ageing is included in the input data for the RPV fast fracture studies. It is to be noted that a minimum shift of 15°C in the transition temperature is always taken into account to cover any effects of strain ageing (a phenomenon which appears early in life and does not subsequently evolve). This effect cannot be summed with thermal ageing per se.

The fast fracture studies are updated for each plant series every ten years at least; the fitness for service of the zones concerned with respect to this ageing mechanism is established when the regulatory fast fracture resistance criteria are confirmed.

The monitoring provisions for this ageing mechanism are described in section 5.1.3.2.

The AAS is classified status 2 for the outlet nozzles.

The AAS is classified status 0 for the RPV closure heads function in hot dome mode.

The AAS is classified "not applicable" for all the other zones functioning at the cold leg temperature.

5.1.2.3 THERMAL AGEING OF THE DISSIMILAR METAL WELDS OF OUTLET NOZZLES WITH A DILUTION ZONE ANOMALY

Dilution anomalies in the first buttering layer were discovered on a nozzle of the Koeberg RPV in 1980; complementary inspections revealed that this type of anomaly also affects some of the RPVs in the 900 MWe plant series of the EDF fleet. The dissimilar metal welds (between low alloy steel and austenitic stainless steel) with a dilution zone anomaly display lower toughness properties than dissimilar metal welds with a normal dilution zone. Subjecting these dissimilar metal welds to continuous operating temperatures of between 300°C and 350°C is likely to lead to ageing of their microstructural components. Experimental tests carried out in the 1990's showed that dissimilar metal welds with dilution zone anomalies did not display any particular sensitivity to thermal ageing. A complementary experimental programme is currently being carried out to consolidate these conclusions.

The toughness properties of dissimilar metal welds with dilution zone anomalies have been established on the basis of experimental tests and are taken into consideration in the fast fracture resistance analyses of these zones.

The monitoring provisions for this ageing mechanism are described in section 5.1.3.3.

The AAS is classified status 0 for the dissimilar metal welds in question on the 900 MWe plant series.

5.1.2.4 FATIGUE OF LOW ALLOY STEEL COMPONENTS

Thermal or mechanical transients can cause cracks to appear in structures that are initially free of defects. Such cracks could then propagate in service, leading to a reduction in the fracture resistance of the structure.

The RPV is a level-1 component for which fatigue damage (initiation of a crack) is evaluated using the rules of the RCC-M and results in the calculation of a usage factor (f_u). These studies are updated every ten years at least. The usage factor is an input data item for the preparation of inspection programmes. The studies do not identify any zones on the RPV that could be concerned by a fatigue risk (i.e. f_u >1).

The particular cases such as local thermal-hydraulic phenomena or high cycle fatigue do not concern the RPV.

As a general rule, the RPV is not particularly sensitive to fatigue and the operating and inspection experience feedback is satisfactory.

As a complement, for zones subject to fast fracture analysis or displaying confirmed defects, fatigue propagation studies are carried out and show that such propagation in operation remains limited. These findings are confirmed by the monitoring measures.

In the case of the RPV, fatigue damage is therefore considered to be potential.

The monitoring provisions for this ageing mechanism are described in section 5.1.3.4.

The AAS's relative to the fatigue phenomenon on the RPV are classified status 0.

5.1.2.5 BORIC ACID CORROSION OF LOW ALLOY STEEL PARTS

In the event of an external leak resulting in primary coolant coming into contact with the low alloy steel of the RPV, the concentration of boric acid or the development of damp boric acid deposits can lead to a significant loss of metal from the external surface. Without prior degradation of certain components or sealing devices, this risk is excluded.

The risks of external leaks have been significantly reduced following completion of replacement of the original closure heads of the 900 MWe and 1300 MWe series RPVs, equipped with adapters in 600 alloy (see § 5.1.2.7). Residual risks of leaks can remain at the connections of the control rod drive mechanisms and of the BMI penetrations, or in case of leaks from the RPV seals towards the exterior.

The monitoring provisions for this ageing mechanism are described in section 5.1.3.5.

The AAS's relative to the boric acid corrosion phenomenon on the RPV are classified status 0.

5.1.2.6 ATMOSPHERIC CORROSION OF DISSIMILAR METAL WELDS

The presents of a damp atmosphere, possibly corrosive or slightly corrosive, can cause localised aqueous corrosion. Austenitic stainless steels with a high carbon content and sensitive to intercrystalline corrosion can be affected by this type of mechanism, which results in the presence of intergranular decohesions (crack-type defects). For the RPV, these defects are situated in the first buttering layer on the melted austenitic metal side, and can be situated on the outer surface of certain dissimilar metal welds. Such defects have been found in service, more particularly on the RPVs of the 900 MWe series.

Particular measures have been taken with the thermal insulation to avoid water retention in order to prevent this mechanism from occurring. An in-service monitoring programme has also been implemented to detect them. The detected defects have been eliminated, and the zone is then subject to a tightened inspection programme for ten years. The absence of recurrence of this type of defect after repair has shown the effectiveness of the corrective and preventive measures implemented.

The dissimilar metal welds in nickel-based alloy on the N4 and EPR plant series are less sensitive to this type of degradation. They benefit moreover from the improvements made in the thermal insulation on the other plant series.

The monitoring provisions for this ageing mechanism are described in section 5.1.3.7.

The AAS relative to atmospheric corrosion of the dissimilar metal welds of the RPV is classified status 0.

5.1.2.7 STRESS CORROSION OF NICKEL-BASED ALLOY PARTS IN PRIMARY SYSTEM ENVIRONMENT

The joint action of a particular environment and mechanical tensile stresses can result in the appearance of stress corrosion cracks. This phenomenon has been evidenced in France and elsewhere in the world on nickel-based alloy parts of PWRs. This phenomenon has particularly concerned parts in 600 alloy, which is more sensitive to stress corrosion in the primary system environment. The presence of residual manufacturing stresses can increase sensitivity to the phenomenon.

Several zones of the RPV feature nickel-based alloys (and their welds):

- the BMI penetrations,
- the radial guides (or M supports) and the cladding of the transition ring of the EPR RPV on which the radial guide are mounted,
- the RPV closure head adapters,
- the RPV nozzle repair zones in 182 alloy,
- the dissimilar metal welds of certain RPVs of the N4 series and those of the Flamanville 3 EPR RPV.

With regard to the BMI penetrations:

- The RPVs of the 900 MWe and 1300 MWe plant series and two RPVs of the N4 series have BMI penetrations in 600 alloy; the welds are stress-relieved in the factory but some BMI penetrations may have undergone subsequent straightening;
- Two RPVs of the N4 series have BMI penetrations in 690 alloy which is not sensitive to stress corrosion;

- The EPR RPV is not equipped with BMI penetrations, as the in-core neutron instrumentation is introduced via the closure head.

  The AAS relative to the BMI penetrations is classified status 1 pending finalisation of the definitive repair process.

With regard to the radial guides:

- The EPR RPV radial guides and their welds are made from 690 alloy which is not sensitive to stress corrosion. The transition ring cladding is also produced with a filler metal equivalent to the 690 alloy.
- The radial guides of the RPVs of the fleet in service are made from 600 alloy. Their welds are stress relieved in production, and from the knowledge acquired on the materials and the stresses, the risk of incipient stress corrosion in the base metal and the deposited metal is unlikely.

  The AAS relative to the radial guides is classified status 0.

With regard to the RPV closure head adapters for the entire EDF fleet, all the nickel-based alloy sections in contact with the primary coolant are in 690 alloy which is not sensitive to stress corrosion in the primary system environment, as the original closure heads of the 900 MWe and 1300 MWe series (equipped with adapters in 600 alloy) have been replaced in service. This large-scale replacement was carried out further to the occurrence of a leak on a closure head of the 900 MWe series during a hydrostatic test in the early 1990s, and the discovery of signs of stress corrosion on several other closure heads in the EDF fleet during the subsequent inspections. For the N4 and EPR plant series, the adapters have been in 690 alloy from the outset.

The AAS relative to the RPV closure head adapters is classified not applicable, since all the closures heads of the EDF fleet are now equipped with adapters made from 690 alloy.

With regard to the RPV nozzle repair zones, the studies and tests performed reveal reduced stresses and favourable behaviour of the diluted zone of the repairs. The risk of incipient stress corrosion of the Inconel repairs of these nozzles is very low.

The AAS relative to the RPV nozzle repair zones is classified status 0.

With regard to the dissimilar metal welds in nickel-based alloy:

- For the EPR RPV they are produced with a filler metal equivalent to the 690 alloy which is insensitive to stress corrosion.
- For three RPVs of the N4 series they are produced with a filler metal equivalent to 600 alloy and are stress-relieved in production. The knowledge acquired on the materials and the stresses indicates that the risk of incipient stress corrosion in the dissimilar metal welds in nickel-based alloy is very low.

The AAS relative to the dissimilar metal welds is classified status 0.

5.1.3 MONITORING, TESTING, SAMPLING AND INSPECTION OF THE REACTOR PRESSURE VESSEL

The French regulations define the general requirements concerning monitoring of the Primary Cooling System equipment. It obliges the following to be put in place:

- in-service monitoring measures showing that the equipment items function under the design-basis conditions,

- verifications for detecting defects that are prejudicial to equipment integrity; these verifications take into account the sensitivity of a zone to a mode of degradation (including the fast fracture risk) and operating experience feedback,
- a programme for monitoring the properties of materials that have an impact on the demonstration of integrity.

The first point results more specifically in the inventorying of the situations as seen by the Primary Cooling System, to verify that the stresses experienced by the equipment items are in conformity with the design assumptions, both in the way they occur and in their frequency of occurrence. The file describing these stresses is updated if necessary to take operating experience feedback into account. The chemistry of the primary coolant is also monitored to ensure that the environment of the materials remains compatible with their conditions of use.

The second point results in the implementation, during reactor outages, of qualified non-destructive examinations of the parts of the RPV identified as being the most sensitive to the degradation modes that concern it. Periodic verifications are also carried out as part of defence in depth on zones that do not display any particular sensitivity to the identified ageing mechanisms (more specifically, all the welded joints on the RPV body are inspected at each ten-yearly outage, whether they are concerned by an ageing mechanism or not). Lastly, sampling inspections are also carried out on certain zones under the Complementary Investigations Programmes to confirm the absence of modes of degradation that had not been identified.

The third point results in the implementation of test programmes to monitor properties which will be difficult to monitor on the equipment in situ, or the implementation of a monitoring programme specific to each RPV for the irradiation ageing mechanism.

The monitoring measures associated with the ageing modes presented in section 5.1.2.1 are described in the following paragraphs.

5.1.3.1 IRRADIATION AGEING OF THE CORE ZONE (SHELLS AND WELDED JOINT)

5.1.3.1.1 Programmes for Monitoring the Effects of Irradiation

The irradiation ageing mechanism forms the subject of a specific programme for monitoring changes in the properties of the RPV steel, designated the "PSI" programme. At the manufacturing stage, test pieces from the material of each RPV are prepared and packaged in metal capsules which are introduced into the RPV when it starts operating. These capsules are withdrawn at regular intervals and mechanical tests are performed on the test pieces they contain to assess the properties of the material after being kept in the RPV exposed to neutron radiation.

The principles of this monitoring programme are similar for all the plant series in the EDF fleet, from 900 MWe series to the EPR. Each RPV starts out with at least six capsules, four being introduced at reactor start-up and the following ones being introduced in service (reserve capsules). The capsules are withdrawn at regular intervals in order to encompass the irradiation of the RPVs when the ten-yearly outages are reached.

The tests performed are impact tests, which enable the shift in the material transition temperature to be determined, and tensile tests. Toughness test pieces are also available so that complementary test programmes can be conducted if necessary.

The monitored materials are:

- The base metal of the core shell displaying the highest end-of-life transition temperature (selected at the design stage on the basis of a prediction of its embrittlement),
- The irradiated welded joint,

- The heat-affected zone of the monitored shell.
- A reference metal sampled from the same given sheet for all the 900 MWe, 1300 MWe and N4 plant series (EPR is not concerned).

The embrittlement prediction model was updated in 2007 on the basis more specifically of numerous results from the PSI programme acquired since the start-up of the EDF reactors (more than 350 PSI measuring points taken into consideration in the development of the model). This provides a model that is as close as possible to the behaviour of materials representative of the EDF fleet RPVs under actual irradiation conditions. When each new capsule is withdrawn, the test results are compared with this model to verify that the embrittlement model encompasses the behaviour of the materials.

5.1.3.1.2 Verification of the core zone

As indicated in section 5.1.2.1, irradiation ageing tends to reduce the toughness properties of the material and therefore its resistance with respect to the fast fracture risk. The analysis of this ageing mechanism with respect to RPV integrity must therefore take into account the potential or confirmed presence of a defect.

During the manufacture of the first RPVs, defects induced by a cold cracking phenomenon were detected on certain coated components of the RPVs (nozzles in particular). The risk of these defects appearing was subsequently significantly reduced by applying additional precautions during the stainless cladding application operations (on the last reactors of the 900 MWe series and the intermediate reactors of the 1300 MWe series).

For the reactors in service (900 MWe, 1300 MWe and N4 plant series), the RPV core zone is subject to automated ultrasonic inspection at each ten-yearly outage, covering the under-cladding zone of the shells and the irradiated welded joint.

For the RPV of the EPR, EDF has planned to inspect the same zone during the Complete Initial Inspection (base metal of the shells and welded joint).

5.1.3.2 THERMAL AGEING OF LOW ALLOY STEELS

The material properties of the zones subjected to thermal ageing are not covered by a specific monitoring programme as is the case for irradiation ageing. The change in properties is covered by the prediction model established on the basis of numerous tests and whose conservative nature is confirmed at the PWR operating temperatures.

As indicated in section 5.1.2.4, thermal ageing tends to reduce the toughness properties of the material, and therefore its resistance with respect to the fast fracture risk. The analysis of this ageing mechanism with respect to RPV integrity must therefore take into account the potential or confirmed presence of a defect.

The presence of defects induced by a cold cracking phenomenon (see § 5.1.3.1.2) potentially concerns some of the 900 MWe and 1300 MWe plant series RPV nozzles which were not subject to additional precautions during manufacture. For the 1300 MWe series, only the first two RPVs are potentially concerned.

With regard to the RPV closure heads, precautionary measures were taken during the manufacture of the original closure heads of the N4 and EPR series and the replacement closure heads of the 900 MWe and 1300 MWe series

The verifications implemented are:

- For the nozzles that were not subject to additional manufacturing precautions: automated in-service ultrasonic inspection of the under-cladding zone, and monitoring inspection at each ten-yearly outage of the nozzles on which significant defects have been identified.
- For the 900 MWe series nozzles that were subject to additional manufacturing precautions: "point zero" inspection of the under-cladding zone; no significant defect was detected on the inspected RPVs.
- For the RPV closure heads: "point zero" ultrasonic inspection of the flange/dome weld at commissioning, and monitoring inspection at each ten-yearly outage of the welds on which significant defects were identified.

Precautions with respect to cold cracking were taken during the manufacture of the EPR outlet nozzles. However, their "set-on" configuration means that the weld joining them to the nozzle support ring is effectively subjected to the hot leg temperature. As is the case for the other RPVs in the EDF fleet, an inspection of the EPR nozzle connection welds is planned at each ten-yearly outage.

5.1.3.3 THERMAL AGEING OF THE DISSIMILAR METAL WELDS OF THE OUTLET NOZZLES WITH A DILUTION ZONE ANOMALY

The material properties of the dissimilar-metal welds with a dilution zone anomaly have been determined from experimental tests, taking account of the ageing effects. These data are taken into consideration for the fast fracture analyses.

The dissimilar-metal welds of the RPV nozzles undergo a double volumetric inspection at each ten-yearly outage to check there are no defects, whether they are concerned by the dilution zone anomaly or not:

- Automated ultrasonic inspection: performed on the RPVs of all the plant series,
- Radiographic inspection: performed on the RPVs of the 900 MWe, 1300 MWe and N4 plant series. On the RPV of the EPR, this inspection is only planned during the Complete Initial Inspection (VCI).

5.1.3.4 FATIGUE OF LOW ALLOY STEEL COMPONENTS

The metallic casing of the RPV is not particularly concerned by the risk of fatigue cracking. This risk is assessed on the bases of the usage factor of the different zones of the RPV, calculated at the design stage and re-assessed in service if necessary.

The examinations carried out with respect to the fatigue risk concern the zones outside the low alloy steel casing, namely:

- the RPV studs undergo eddy current inspection at each ten-yearly outage (all plant series),
- the thin-lipped seals ensuring the sealing of certain adapter flanges (CANOPY seals, see § 5.1.1.1) undergo visual inspection at each reactor outage and televisual inspection at each ten-yearly outage after hydrostatic testing.

5.1.3.5 BORIC ACID CORROSION OF LOW ALLOY STEEL PARTS

The RPV metal casing would only be concerned by this damage mechanism in the event of external leakage of the primary coolant onto the low alloy steel. Monitoring for leaks is carried out:

- In operation:
    - Leak assessments of the primary system are carried out daily to detect any significant leak in the primary system.
    - A system for detecting leaks at the RPV closure head seals is installed on all the plant series to detect a leak of the RPV internal seal (precursor of a potential external leak).

- During reactor outages:
  - Visual inspection at each reactor outage for leaks or signs of boron in zones that could affect the ferritic steel, particularly the seals above the RPV closure head.

    For the EPR RPV, the control rod drive mechanisms are assembled to the adapters by a bolted connection; each assembly has a leak recovery sump which can be inspected during reactor outages.
  - Visual and televisual inspections and acoustic monitoring during hydrostatic tests, at the adapter welds and BMI penetrations.

5.1.3.6 ATMOSPHERIC CORROSION OF DISSIMILAR-METAL WELDS

This damage mechanism results in the appearance of surface defects on the dissimilar metal welds. Monitoring of the zone with respect to this mechanism comprises two parts and is similar for all the plant series:

- A dye penetrant inspection of the dissimilar-metal weld is performed at each ten-yearly outage as part of defence in depth, independently of the sensitivity of the zone,
- If the dye enetrant inspection reveals an intergranular decohesion defect (see § 5.1.2.6), a tightened monitoring programme is implemented after the defects have been removed: four dye penetrant inspections are performed in succession over a ten-year period. If these inspections confirm the non-recurrence of these defects, the dissimilar metal weld is declared non-sensitive and the ten-yearly inspection interval is restored.

Given the preventive measures implemented and the fact that the inspections to date have revealed no defects, no dissimilar metal welds on the EDF fleet RPVs are classified as sensitive.

5.1.3.7 STRESS CORROSION OF NICKEL-BASED ALLOY PARTS IN THE PRIMARY SYSTEM ENVIRONMENT

The risk associated with the stress corrosion phenomenon is the appearance of cracks which can propagate in operation. The nature of the inspections in the nickel-based alloy zones potentially concerned by the stress corrosion risk is defined from the risk identified in the zone, taking account of the sensitivity of its material and the loadings to which it is subjected (including the residual stresses).

<u>With regard to the RPV closure heads:</u> on all the EDF RPV closure heads, the zones in contact with the primary coolant are in 690 alloy which is not sensitive to stress corrosion. Inspections are carried out in application of the defence in depth principle:

- A check by acoustic monitoring is carried out at each ten-yearly hydrostatic test for leak detection.
- An eddy current inspection of the peripheral adapters of the three control closure heads of the 900 MWe plant series is carried out every ten years.

<u>With regard to the radial guides:</u>

- For all the plant series a televisual inspection is carried out at each ten-yearly outage to detect any anomalies, independently of the risk of stress corrosion of the guides and their welds,
- For the 900 MWe, 1300 MWe and N4 plant series, a complementary ultrasonic inspection is implemented as from the 3rd ten-yearly outage to detect any defects induced by stress corrosion,
- The EPR series is equipped with guides in 690 alloy which is not sensitive to stress corrosion. It is not subject to specific inspections with respect to this damage mechanism.

With regard to the RPV nozzle repair zones: all the repaired nozzles in 182 alloy undergo ultrasonic or eddy current inspection at each ten-yearly outage to detect stress corrosion cracks. These inspections only concern five RPVs of the 900 MWe series and two RPVs of the 1300 MWe series.

With regard to the bottom-mounted instrumentation penetrations:

- Since 2011, the RPVs equipped with BMI penetrations in 600 alloy (i.e. the 900 MWe and 1300 MWe series and two RPVs of the N4 series) undergo an ultrasonic inspection every ten years, whereas previously it was a sampling inspection.
- The RPVs equipped with BMI penetrations in 600 alloy undergo a televisual inspection at each ten-yearly outage, before and after the hydrostatic test. The RPVs which have not yet undergone their first ultrasonic inspection are subject, in the interim, to a televisual inspection at each outage.
- The two N4 reactors equipped with BMI penetrations in 690 alloy undergo a televisual inspection before and after the ten-yearly hydrostatic test.
- The EPR plant series does not use BMI penetrations.

5.1.4 PREVENTIVE AND CORRECTIVE ACTIONS FOR THE REACTOR PRESSURE VESSEL

5.1.4.1 IRRADIATION AGEING

The principal factors influencing irradiation ageing are the fluence and the chemical composition of the irradiated material.

The RPVs whose predicted end-of-life fluence is the highest are those of the 900 MWe series. In-service measures are taken to limit the fluence of these RPVs:

- Since the 1990's, the loading plans are optimised to reduce the fluence at the RPV hot spots. A reduction of some 40% in the neutron flux has been lastingly obtained.
- As from the 4th ten-yearly outages, absorbent hafnium RCCAs[53] shall be installed in the core opposite the four hot spots, bringing an anticipated additional 40 to 50% reduction in the neutron flux.

Ultimately, the neutron flux at the RPV hot spots will have been reduced by a factor of 3 since the 900 MWe reactors entered service.

The fluences of the 1300 MWe and N4 series RPVs are significantly lower (about 20% less for an equivalent operating duration). For the EPR series, the core and RPV design, a greater water layer width and the presence of a heavy reflector give a much lower end-of-life fluence for an equivalent operating duration (3 to 5 times lower compared with the 900 MWe series, depending on fuel management).

The chemical specifications applicable to the core shells have moreover been steadily reinforced since the first 900 MWe RPVs were manufactured.

5.1.4.2 MANUFACTURE

The ageing modes that lead to degradation of the toughness properties (irradiation ageing or thermal ageing) have to be considered taking into account the potential presence of defects in the component.

Further to the detection of cold cracking defects under the cladding of the first RPVs of the EDF fleet, the cladding installation conditions have been modified to avoid reproducing this phenomenon. Although

[53] Rod Control Cluster Assembly.

these measures do not influence the ageing of the material itself in any way, they reduce the probability of manufacturing defects and therefore they reduce the fast fracture risk.

These measures have been gradually implemented on the last RPVs of the 900 MWe series and on the 3rd and subsequent RPVs of the 1300 MWe series. The replacement closure heads of the 900 MWe and 1300 MWe RPVs, the N4 series RPVs and the EPR RPV have all fully benefited from these additional manufacturing precautions. In addition to the in-process inspections, the in-service inspections of the components concerned by these precautions have confirmed the effectiveness of these measures, as no defects have been detected.

5.1.4.3 DISSIMILAR METAL WELDS

The intergranular decohesion defects detected on the dissimilar metal welds during in-service inspections (see § 5.1.3.6) have been systematically removed by excavation, after which tightened monitoring has been implemented.

Modifications to the thermal insulation of the nozzles have been progressively introduced to prevent the risk of moisture retention at the welds. These modifications have been effective, as no dissimilar metal welds are classified today as being sensitive to this mechanism.

5.1.4.4 NICKEL-BASED ALLOY ZONES

Bottom-mounted instrumentation (BMI) penetrations

Further to the discovery of a stress corrosion crack on a BMI penetration of a 900 MWe reactor which was initiated from a manufacturing defect, a repair process that involves plugging the BMI penetration has been developed and implemented. A repair process enabling the BMI penetration to maintain its function is also being studied.

RPV closure heads

All the RPV closure heads of the 900 MWe and 1300 MWe series originally equipped with adapters in 600 alloy have been replaced in service by closure heads equipped with adapters in 690 alloy. This replacement programme began in 1994 and was completed in 2009.

Similar provisions have been adopted during manufacture or before commissioning for the closure heads of the N4 and EPR series:

- Direct fitting of 690 alloy adapters on the EPR RPV and the last RPV of the N4 series during manufacture;
- Replacement of the 600 alloy adapters initially installed on the first three RPVs of the N4 series by adapters in 690 alloy.

5.2 EDF EXPERIENCE IN THE APPLICATION OF ITS REACTOR PRESSURE VESSEL AGEING MANAGEMENT PROGRAMME

5.2.1 IRRADIATION AGEING / MONITORING OF THE CORE ZONE

The PSI programme allows the behaviour of the materials in the core zone to be monitored up to irradiation levels equivalent to at least 60 years of operation, given the number of irradiation capsules available. The capsules are withdrawn at regular intervals in order to encompass the irradiation of the RPVs when the ten-yearly outages are reached. This schedule is re-examined at regular intervals to take into account the results of the counts of fluence effectively received by the RPVs, or the improvements in fluence management (see § 5.1.4.1).

The gradual acquisition of the PSI programme results enabled the irradiation embrittlement model to be updated in 2007. This model was revised on the basis of the results of the irradiation capsules of the 900 MWe series (more than 350 measuring points), which covers the widest range of fluence and chemical compositions for the RPVs of the EDF fleet. The revised model is more representative of the behaviour of the material over long periods of time than the initial model which was based on a smaller number of data in the irradiated state, obtained from tests in experimental reactors. Application of this model to the results of the 1300 MWe and N4 series confirms that it provides a satisfactory representation of the behaviour of the materials.

The core zone undergoes a complete automated ultrasonic inspection at each ten-yearly outage, covering the entire under-cladding zone. The defect detection performance levels are guaranteed by the qualification of the process. The successive inspections performed since the RPVs entered service have produced the following principal results:

- Under-cladding cold cracking fabrication defects have been detected on eight RPVs (five on the 900 MWe series and three on the 1300 MWe series); these defects display no development in service,
- Welding fabrication defects have been detected on two RPVs of the 900 MWe series; these defects display no development in service.

The observed absence of development in service is consistent with the propagation studies carried out on the defects in the core zone. It also confirms that the chosen inspection frequency for the core zone is appropriate.

These detected defects are the subject of specific substantiation, applying a conservative deterministic approach demonstrating that the fast fracture resistance of the RPVs concerned in pressurised cold shock situations is ensured under all circumstances.

A more general demonstration of the fast fracture resistance of the RPVs in service is provided by a generic file applicable to all the RPVs of each plant series. Carried out applying the same conservative deterministic approach, this generic analysis considers a hypothetical defect whose dimensions are at the detection limit, given the performance of the inspection process used, and which is situated in the most highly stressed and irradiated part of the RPV of each plant series presenting the most severe mechanical properties of the fleet (the highest ductile-brittle transition temperature RT_{NDT}). This analysis is carried out for a selection of the most severe transients likely to stress the RPV, covering the entire design-basis envelope, from normal operation to highly improbable accident situations.

In view of the results obtained, the monitoring programme for the zone of RPVs subject to irradiation is considered to be perfectly appropriate, providing the information necessary to assess the fitness for service.

5.2.2 THERMAL AGEING / INSPECTION OF THE NOZZLES AND CLOSURE HEADS

The thermal ageing model for low alloy steels is based on the MacLean model, parameterised in the 1970's and 1980's on low alloy steels thermally aged for 20,000 hours at temperatures of between 300°C and 550°C. Complementary ageing programmes have been implemented in the laboratory on materials whose composition encompasses that of the EDF fleet components, to confirm the validity of the model for prolonged exposure durations, equivalent to 60 years of operation under reactor conditions. The results obtained have confirmed the validity of the model for the different metallurgical zones concerned (base metal, heat affected zone and deposited metal). This ageing model is therefore suitable for defining the properties of materials after 60 years of operation, particularly for the RPV outlet nozzles.

A complete "point zero" inspection was performed on the RPV nozzles of the 900 MWe plant series, whether they had been subject to manufacturing precautions or not.

The RPV nozzles which are subject to an inspection programme during the ten-yearly outage are those of the 900 MWe series on which significant defects have been detected and which are subject to ten-yearly monitoring.

As the effectiveness of the manufacturing precautions to prevent the formation of under-cladding defects has been demonstrated, the nozzles of the other RPVs that have been subject to these precautions do not undergo an under-cladding zone inspection.

All the RPV closure heads have been subject to the manufacturing precautions; on this account, only the flange/dome weld forms the subject of a "point zero" inspection and possible monitoring.

As regards the nozzles on which defects were detected, the inspections confirmed that they were manufacturing faults which did not evolve in service, which is consistent with the studies performed.

In view of the results obtained, the monitoring programme for the zones of the RPVs concerned by thermal ageing is considered to be perfectly appropriate, providing the information necessary to assess their fitness for service.

5.2.3 FATIGUE OF LOW ALLOY STEEL COMPONENTS

The reactor pressure vessel is not a component that is particularly subject to fatigue stress. The zones displaying the highest usage factors are the studs and the thin-lipped seals of the closure head adapters. These zones undergo in-service inspections:

- Ten-yearly eddy current inspection of the studs
- For the thin-lipped seals, visual inspection at each outage and televisual inspection during the ten-yearly hydrostatic test.

The inspections performed on the EDF fleet RPVs have not revealed any degradation associated with a fatigue phenomenon.

For the other RPV zones on which manufacturing defects have been detected, the inspections have not evidenced any propagation by fatigue.

Furthermore, the situations inventory does not reveal any atypical behaviour with regard to the situations that could stress the RPV. The analysis of operation of the EDF fleet RPVs shows that the number of occurrences predicted on reaching 60 years of operation are globally covered by the number of occurrences initially predicted for 40 years of operation. The available studies therefore enable the RPV fatigue risk to be satisfactorily assessed.

In view of these elements, the planned provisions for the RPVs with respect to the fatigue mechanisms are considered to be perfectly appropriate.

5.2.4 THERMAL AGEING OF THE DISSIMILAR METAL WELDS OF THE OUTLET NOZZLES WITH A DILUTION ZONE ANOMALY

The toughness properties of dissimilar metal welds with dilution zone anomalies have been established on the basis of experimental tests. Experimental tests carried out in the 1990's showed that the dissimilar metal welds with dilution zone anomalies did not display any particular sensitivity to thermal ageing compared with a normal dissimilar metal weld. A complementary experimental programme is underway to consolidate this hypothesis to cover an operating duration of 60 years. The results of these programmes will be taken into account in the fast fracture studies.

The dissimilar metal welds of the RPV nozzles undergo a volumetric inspection at each ten-yearly outage, whatever the plant series and whether a dilution zone anomaly is present or not. The inspections performed to date have revealed no defects on the dissimilar metal welds.

Furthermore, the dissimilar metal welds are not fatigue-sensitive zones. Consequently, there is no identified risk of a defect appearing in operation.

In view of these elements, and subject to the finalising of the complementary dissimilar metal weld ageing programmes, the planned provisions with respect to the thermal ageing mechanism of the dissimilar metal welds are considered to be perfectly appropriate.

5.2.5 BORIC ACID CORROSION OF LOW ALLOY STEEL PARTS

Boric acid corrosion of low alloy steel parts is a phenomenon that can occur in the event of an external leak. Control of the corrosion risk is therefore ensured by regular inspections of the areas that could be the cause of a leak, in order to detect any degradation at an early stage.

In-service monitoring is ensured through daily leak assessments, and at the RPV seals through the presence of a leak detection system.

Monitoring is ensured during reactor shutdowns by performing visual or televisual inspections of the welds and sealing components of the RPV closure head, or at the BMI penetration, in order to detect any indication of primary coolant leakage.

In view of the monitoring measures implemented, the provisions adopted with respect to the boric acid corrosion mechanism are considered to be perfectly appropriate.

5.2.6 ATMOSPHERIC CORROSION OF DISSIMILAR METAL WELDS

On-site inspections have revealed intergranular decohesion surface defects which have been removed by local excavation. To prevent the formation of such defects, modifications have been made to the thermal insulation to avoid any retention of moisture (see § 5.1.4.3). There has been no defect recurrence in the excavated areas, and today all the dissimilar metal welds on the EDF fleet RPVs are insensitive to this mechanism.

Dye penetrant inspection of the dissimilar metal welds at each ten-yearly outage is maintained for the zone monitoring on all the RPVs in the EDF fleet.

As the origin of these defects has been identified and the defects removed, and the preventive measures have demonstrated their effectiveness, the provisions with respect to the atmospheric corrosion mechanism for dissimilar metal welds are considered to be perfectly appropriate.

5.2.7 STRESS CORROSION OF NICKEL-BASED ALLOY PARTS IN THE PRIMARY SYSTEM ENVIRONMENT

<u>Bottom-mounted instrumentation (BMI) penetrations</u>

The initial inspection perimeter with respect to the stress corrosion risk for the BMI penetrations in 600 alloy was limited to the RPVs whose BMI penetrations could possibly display significant residual stresses given the fabrication operations (deficient stress-relief heat treatment, straightening of BMI penetration after heat treatment, etc.). These inspections are ultrasonic inspections performed every ten years. The RPVs which have not yet undergone their first ultrasonic inspection are subject to tightened televisual inspections, including on the BMI penetrations in 690 alloy.

The detection of an incipient stress corrosion crack in 2011 on a manufacturing defect in a BMI penetration of a 900 MWe RPV confirms the relevance of the inspection programme.

<u>RPV closure heads</u>

All the RPV closure heads in the EDF fleet are equipped with adapters in 690 alloy which is not sensitive to stress corrosion. In application of the defence in depth principle, three control closure heads from the

900 MWe series which were among the first replaced undergo an eddy current inspection every ten years. The inspections performed to date have not revealed any stress corrosion cracks.

A televisual inspection is carried on all the closure heads at each ten-yearly outage to check there are no disorders, along with acoustic monitoring during a hydrostatic test to check there are no leaks. These examination have not revealed any damage to the closure head adapters or their welds.

Radial guides

The radial guides of all the RPVs of the EDF fleet undergo a televisual inspection at each ten-yearly outage to check there are no anomalies.

These inspections have been supplemented since 2009 by ultrasonic inspections carried out during the ten-yearly outage to seek any stress corrosion indications in radial guides in 600 alloy and their welds.

This programme has been gradually applied to the 900 MWe and 1300 MWe RPVs since 2009 and to date has revealed no stress corrosion cracks.

The EPR RPV is equipped with guides in 690 alloy which is not sensitive to stress corrosion.

Zones with repaired Inconel nozzles

These inspections only concern five RPVs of the 900 MWe series and two RPVs of the 1300 MWe series.

All the repaired nozzles in 182 alloy undergo ultrasonic or eddy current inspections at each ten-yearly outage to look for stress corrosion cracks. The inspections performed to date have revealed no defects of this type.

In view of the operating experience feedback acquired from the inspections of zones in nickel-based alloy and their stresses (including residual stresses), the monitoring provisions adopted with respect to the mechanism of stress corrosion of zones in nickel-based alloy are considered to be perfectly appropriate.

5.3 AGEING MANAGEMENT OF RESEARCH REACTOR PRESSURE VESSELS

5.3.1 CEA

5.3.1.1 SCOPE OF PROGRAMME AND AGEING ASSESSMENT

The situation for the research reactors cannot be compared with that of pressurised water nuclear power reactors. This is because the reactor pressure vessel (or pile block) is removable and therefore does not constitute a limitation on the service life of the installation[54].

The Cabri reactor does not have a pressure vessel confining the core of fissile materials.

The Orphée reactor pile block is cooled by heavy water. This reactor vessel is itself immersed in a pool filled with light water. The thimbles are connected to the pile block. There is a sealing system between the reactor vessel and the thimbles.

The JHR reactor pile block is crossed by water at 16.1 bar and 80°C. Irradiation ageing has been taken into account in the design-basis studies through the significant irradiation analysis. All the welded joints are situated outside significant irradiation zones. An in-service monitoring programme based on representative test pieces is planned.

The service life of the components is calculated, depending on the material, according to the maximum fluence in conventional thermal or fast neutrons. Predictive calculations are performed using TRIPOLI 4

[54] The replacement of the Osiris reactor pile block in 2001 can, for example, be noted.

codes[55]. Adjustments are made with the results of measurements taken on samples. The service lives of the components are reassessed at each periodic safety review.

On the Orphée reactor, assessments focused on fast fracture analysis and the analysis of the notch sensitivity of the thimbles which are made from AG3NET. The objective is to avoid attaining low ductility which depends on the percentage of Silicon formed and the Mg_2Si precipitates, which tend to harden the aluminium alloy. The thimbles of channels 4F and 9F were replaced in 2008 and 2012 respectively further to inspections. The new thimbles were produced in accordance with the RCC-MX code taking into account detailed modelling, an analysis of all the operating situations and a level-1 criterion, taking the effect of irradiation into account for the maximum acceptable flux.

Corrosion phenomena, including galvanic corrosion and erosion corrosion, can alter certain components over time. In this case specific monitoring measures are put in place, such as monitoring the chemical properties of the fluids in contact with these components.

5.3.1.2 MONITORING, PREVENTIVE MEASURES AND LICENSEES' EXPERIENCE

The pile block pressure equipment items of the Orphée reactor pile block are the subject of design provisions and an in-service monitoring programme, consistent with the equipment risk analysis. As indicated above, the thimbles of channels 4F and 9F were replaced in 2008 and 2012 respectively further to inspections.

Visual and televisual or ultrasonic inspections are also envisaged on the JHR reactor, as is the monitoring of the actual usage factor of the threaded fasteners.

5.3.2 LAUE LANGEVIN INSITUTE (ILL)

5.3.2.1 SCOPE OF PROGRAMME AND AGEING ASSESSMENT

The pile block of the high flux reactor (RHF) consists of a vessel to which thimbles are affixed. It is cooled by heavy water. The pile block as a whole comprises 50 contained fluid compartments and varied thermodynamic characteristics. The list of component data figures in appendix 11.13.

The reactor vessel pressure is four bars with a temperature of 50°C. The vessel is made from 5754 aluminium alloy and is integrated in a water-filled pool.

Sealing between the vessel and the components connected to it (in particular the thimbles which allow extraction of the neutrons to supply the experimental areas) is ensured by double seals with leak detection between the two seals.

Operating experience feedback has shown that the seals have to be replaced at every second thimble replacement. Corrosion phenomena can affect the components. Operating measures and inspections are implemented.

5.3.2.2 MONITORING, PREVENTIVE MEASURES AND LICENSEES' EXPERIENCE

The objective of the inspections is essentially to ensure there is no corrosion. Corrosion can result from the quality of the water or the susceptibility to intergranular corrosion at very high irradiation levels ($> 10.10^{22}$ n/cm²). This is what led to the operating envelope being specified in terms of pH or water resistivity, and the duration of irradiation of the vessel and the vessel components being limited.

Televisual inspections are carried out each year under water and in air. The removed equipment items are regularly subjected to qualitative tests (bending tests in cell on samples taken from the equipment

[55] TRIPOLI 4 is a code for solving the Monte Carlo transport equation developed at the CEA. It can be used to perform the neutron and photon calculations for radiation protection, core physics and criticality studies. It is the French reference in this area for industrial studies and safety studies alike.

items) and quantitative tests (dimensional and thickness measurements). An eddy current inspection campaign was recently conducted on the vessel inlet and outlet manifolds to ensure there were no defects that could cause a failure after 15 years of operation.

The preventive measures consist essentially in maintaining water quality and ensuring the required quality of construction.

Only two anomalies have been recorded since 1971:

a. In the 1980's, loss of sealing of a thimble was observed. It was caused by an intergranular corrosion phenomenon occurring at an irradiation level exceeding 20×10^{22} n/cm²; the lessons learned from this mean that this fluence is no longer exceeded and the condition of the material has been modified.

b. The second event was observed in 2002: further to the instruction to pressurise the inside of certain thimbles with helium to reduce in-service stresses, a leak developed at the end of the reactor outage works with the vessel depressurised: one rectangular-section thimble thus ended up with an internal overpressure higher than the operating pressure and incompatible with the design; the geometry of the thimble was modified and its internal pressure dropped when the reactor was emptied.

5.4 ASN ASSESSMENT OF THE REACTOR VESSEL AGEING MANAGEMENT PROGRAMME

5.4.1 THE NUCLEAR POWER REACTORS

5.4.1.1 IRRADIATION AGEING / MONITORING OF THE CORE ZONE

ASN considers that regular inspection of the RPV condition is essential for two reasons:

- the RPV is a component whose replacement is not envisaged, owing to both technical feasibility and cost;
- rupture of this item is not considered in the safety studies. This is one of the reasons why all steps must be taken as of the design stage to guarantee its integrity for the entire duration of reactor operations, including in the event of an accident.

The question of the core zone of the EDF fleet RPVs has formed the subject of several examinations and presentations to the Advisory Committee of Experts for Nuclear Pressure Equipment.
The approach adopted by EDF consists in:

- evaluating the fluence received by the vessel and taking the most highly irradiated point called the "hot spot";
- evaluating the embrittlement of the vessel metal according to the fluence received and the chemical composition of the material;
- performing tests on test pieces that are representative of the vessel material in order to confirm the embrittlement predictions;
- evaluating the most penalising thermal-hydraulic transients in each situation category;
- calculating the mechanical margins with respect to the risk of an incipient defect according to the above elements;
- for all the vessels, the zone subjected to neutron irradiation undergoes non-destructive inspections every ten years to check that no new defects have appeared and that existing defects have not evolved.

ASN presented this method and the results at Advisory Committees of Experts meeting sessions in 1987, 1999, 2005, 2010 and 2015 for the 900 MWe and 1300 MWe series RPVs. ASN considers the approach to be appropriate. Furthermore, in order to guarantee sufficiently conservative conditions, the French regulations impose the use of safety coefficients (Article 13 of the Order of 10 November 1999) when performing these mechanical calculations. These coefficients, which must be applied to the loadings, depend on the studied situation categories, which are presented in the table below.

	INCIPIENT	INSTABILITY
2nd category	1.3	2
3rd category	1.1	1.6
4th category	-	1.2

EDF has submitted its substantiation file for the 900 MWe series RPVs. This file aims to demonstrate that there is no risk of incipient defects on any of these RPVs for a period of ten years following the fourth ten-year outages of these reactors. ASN, assisted by IRSN, its technical support agency, must assess this file which will be presented to the Advisory Committee of Experts for Nuclear Pressure Equipment (GP ESPN) in the course of 2018.

5.4.1.2 THERMAL AGEING / INSPECTION OF THE NOZZLES AND CLOSURE HEADS

EDF has implemented a thermal ageing programme for the base metal of the reactor pressure vessels. It has concluded that the extension of reactor operation beyond the fourth ten-yearly outage will not change the ageing of the material considering the data taken into account at the design stage. EDF bases this conclusion on the tests performed and the content of embrittling elements in the RPV constituent materials, in view of the thermal ageing.

ASN considers that the programme implemented by EDF relative to the thermal ageing of the RPV constituent base material enables the final characteristics of this material to be known. Some RPV nozzles display under-classing defects which originated during manufacture. The nozzles in question were manufactured before the precautions to prevent the risk of cold cracking occurring when depositing the cladding were applied as a matter of course.

The in-service inspections serve to ascertain that these defects do not evolve and that no new defects appear in service. ASN considers that these inspections must be maintained.

5.4.1.3 FATIGUE OF LOW ALLOY STEEL COMPONENTS

EDF specifies that the RPVs are not subject to fatigue, with the exception of the RPV studs and the thin-lipped seals of the adapters, which consequently undergo specific inspections. In the light of the inspections performed, ASN considers that the RPV is not an item subject to fatigue.

5.4.1.4 THERMAL AGEING OF THE DISSIMILAR METAL WELDS OF THE OUTLET NOZZLES WITH A DILUTION ZONE ANOMALY

The toughness properties of dissimilar metal welds with dilution zone anomalies have been established on the basis of experimental tests. In the context of continued operation beyond the fourth ten-yearly outages, EDF has undertaken an experimental programme to check there are no impacts on the thermal ageing of these welds, which is satisfactory. ASN will examine the conclusions of these tests.

5.4.1.5 BORIC ACID CORROSION OF LOW ALLOY STEEL PARTS

ASN considers that boric acid corrosion of the low alloy steel of the RPVs can only occur if there is contact between the primary coolant and the steel. Insofar as the RPV is clad with a layer of stainless steel, and in view of the design and the monitoring measures applied, this type of corrosion is not a feared phenomenon on the RPVs. The monitoring measures serve to ascertain that there is no degradation of the stainless steel cladding.

5.4.1.6 ATMOSPHERIC CORROSION OF DISSIMILAR METAL WELDS

The dissimilar metal welds were originally sensitive to atmospheric corrosion, which was attributed to the original design of the thermal insulation, which has since been modified. EDF maintains the periodic inspections of these zones, and ASN considers that the methods applied are appropriate.

5.4.1.7 STRESS CORROSION OF NICKEL-BASED ALLOY PARTS IN PRIMARY SYSTEM ENVIRONMENT

Bottom-mounted instrumentation (BMI) penetrations

In 2011, during an examination, EDF discovered the presence of an indication on a BMI penetration on the Gravelines reactor No. 1.

Since that date the BMI penetration has been withdrawn and the RPV has been repaired. The investigations carried out tend to reveal stress corrosion cracking initiated on a manufacturing defect.

This event led ASN to ask EDF to inspect all the BMI penetrations according to a schedule proposed by EDF, as the BMI penetration had not been subject to systematic examinations until then. The inspections performed revealed one case of circumferential cracking on the Cattenom reactor No. 3; no other defects were evidenced by any of the other examinations performed on the BMI penetrations.

RPV closure heads

Further to the discovery of stress corrosion on an RPV closure head adapter of the Bugey reactor No. 3 in 1991, EDF undertook the replacement of all the RPV closure heads equipped with adapters made from Inconel 600 alloy. The replacement closure heads are equipped with adapters made from Inconel 690 alloy.

EDF continues to monitor a number of closure heads. ASN considers that the monitoring is appropriate.

Radial guides

The EDF monitoring of the radial guides is appropriate.

Zones with repaired Inconel nozzles

The EDF monitoring of the radial guides is appropriate.

5.4.2 RESEARCH REACTORS

The pressurised vessel of the CEA's Orphée and Jules Horowitz reactors and of the ILL's high flux reactor corresponds to the pile block. It must be emphasised that, unlike the nuclear power reactors, these pile blocks are entirely immersed in pools. They are replaceable and are subjected to relatively low thermal-hydraulic pressures and temperatures. They are built in material grades that are on the whole compatible with the physical-chemical conditions of the surrounding fluids, and these conditions are optimised to increase this compatibility as far as the operating conditions permit. With regard to the Orphée and RHF reactors, the pile blocks and associated equipment feature different compartments which are subjected to variable pressure and temperature conditions. These compartments are listed in appendices 10.12 and 10.13.

The objective of the in-service monitoring is to ascertain that the nuclear pressure equipment, including the vessels, remain in a condition that allows them to be operated without consequences that can be prejudicial to the protection of the environment as set by the regulations.

With regard to the Orphée and RHF reactors, thimble-replacement criteria based on the maximum fluence to which they have been subjected have been introduced in recent years with the aim of preventing them from undergoing irradiation hardening (which reduces ductility). The thimbles are particularly subject to irradiation due to their location with respect to the reactor core. The drop in ductility of the aluminium alloys results from the formation of Mg_2Si precipitates due to the combination of magnesium present in the alloy with the silicon formed by the neutron bombardment of the aluminium.

The order of 30 December 2015 defines the inspections and curative actions for nuclear pressure equipment (appendices 5 and 6 of the order of 30 December 2015):

- performance of an annual video inspection;
- replacement of the aluminium thimbles according to the received fluence[56] ;
- control of the parameters of the internal and external fluids of the compartments (particularly purity, pH and heavy water resistivity);
- permanent monitoring of pressures.

The block pile of the RHF is a nuclear pressure equipment item which is subject to appendices 5 and 6 of the order of 30 December 2015. This comes from the fact that some of its compartments (sources, thimbles) are pressurised with gas, leading to an increase in the category of the equipment as a whole. The Orphée reactor, on the other hand, is not subject to these provisions. ASN has examined the sensitivity of the pile block constituent materials to the different potential modes of degradation: no mode of degradation that is not addressed in operation has been identified. Regular inspections focusing on nuclear pressure equipment - and therefore including the vessel - are carried out; they have given rise to no remarks relating to the in-service monitoring of the block.

[56] There is also a limit on the received fluence for the zircaloy-4 compartments, but it exceeds the life time of the facility.

6 CALANDRIA AND PRESSURE TUBES (CANDU)

Not applicable in France.

7 CONCRETE CONTAINMENT STRUCTURES

Summary:

For nuclear power reactors, the ageing management programme of the containments applies to the concrete containments and to their liners and coatings: it benefits from fleet operating experience feedback, owing to their uniform design.

As part of its containments ageing management programme, EDF identified the ageing mechanisms on the basis of their operation and the operating experience feedback from the inspections and tests carried out. These mechanisms concern the concrete (cracking, creep, shrinkage, carbonation and internal swelling pathologies), the metal liner (corrosion and blistering), the prestressing tendons (relaxation and corrosion), the passive rebars (corrosion), the containment instrumentation system (malfunction of certain instruments), as well as the paints and composite coatings (ageing and behaviour in accident situation, including a severe accident).

The containment tightness tests and the monitoring of mechanical behaviour using the containment instrumentation system, are a means of observing and anticipating the mechanical behaviour and tightness of the containments: this notably led to the implementation of both preventive measures (installation of inner coatings) and corrective measures (repair of outer facings). ASN considers that the implementation of these measures leads to satisfactory management of the containment ageing phenomena.

For research reactors, the ageing of the containments is mainly monitored during periodic checks and tests. The compliance of the containments is verified during the periodic safety reviews. ASN considers that the ageing management programme remains limited. This programme must be enhanced and, on the basis of their specific features and the results of tests and checks as well as the knowledge derived from national and international programmes, the ageing mechanisms which could affect them must be identified.

Reactor containments constitute the third and final barrier between the fuel and the environment. So that the containments can perform their safety functions during the reactor operating period, they must meet mechanical strength and leaktightness requirements. Compliance with these requirements must be continuously proven by the licensee, more particularly through implementation of an ageing management programme appropriate to the safety implications. This programme must take account of the design and construction characteristics of the containments, along with their potential or confirmed ageing phenomena.

7.1 DESCRIPTION OF AGEING MANAGEMENT PROGRAMME FOR EDF CONCRETE CONTAINMENTS

7.1.1 SCOPE OF AGEING MANAGEMENT PROGRAMME FOR CONCRETE CONTAINMENTS

The containment encloses the entire primary reactor coolant system and some of the auxiliary equipment. With its internal structures, it supports this equipment in both normal and accident situations. It protects them from the outside environment: weather, external hazards.

In the event of an accident affecting the NSSS, it protects the outside environment by providing a barrier between the radioactive products and the environment. It thus performs mechanical strength and leaktightness functions which must be guaranteed for the entire lifetime of the NPP. This is why it is incorporated into the ageing management programme.

The design of the containments differs according to the EDF plant series (see Appendix 10.7):

- The containments for the 900 MWe reactors are single-wall prestressed concrete containments with a metal liner. It is this liner which provides the confinement; it is said to be static.
- The containments of the 1300 MWe and N4 reactors are double-wall: the inner prestressed concrete wall without liner and an outer reinforced concrete wall. The annulus between the two containments is used to collect and filter any leaks from the inner containment. This is the principle of dynamic confinement, which supplements the static confinement performed by the prestressed concrete inner containment.
- The EPR containment consists of a prestressed concrete inner wall with liner and a reinforced concrete outer wall. The EPR thus has static confinement through the metal liner, plus dynamic confinement.

7.1.1.1 DESCRIPTION OF CONTAINMENTS OF 900 MWE PLANT SERIES

The containment comprises:

- a reinforced concrete basemat about 3.50 m thick in the standard area,
- a prestressed concrete cylindrical part (called the skirt) with a bracket on the inner face to support the pivot bridge (CPY plant series) and vertical ribs on the outer face to anchor the horizontal prestressing tendons,
- a prestressed concrete gusset allowing connection between the prestressed concrete skirt and the reinforced concrete basemat,
- a torospherical dome, connected to the skirt by a belt comprising the upper anchors of the vertical prestressing tendons and those of the dome prestressing,
- a metal liner covering the entire inner face of the containment.

The containment is prestressed by means of post-tensioning. The tendons may be diverted to bypass containment openings and penetrations. The anchorages are fitted either with metal covers injected with cement grout, or reinforced concreate seals.

7.1.1.2 DESCRIPTION OF CONTAINMENTS OF 1300 MWE AND N4 PLANT SERIES

The double-wall concept is based on the principle of dynamic confinement to supplement the static confinement of the prestressed concrete inner containment: the depressurisation of the annulus between the two containments is used to collect and filter any leaks from the inner containment. The outer containment thus participates in the confinement of any radioactive products that could be accidentally emitted inside the reactor building.

The double-wall containment of the 1300 and N4 plant series consists of the following:

- a reinforced concrete basemat about 3m thick constituting the foundation of the two walls of the containment,
- a prestressed concrete inner wall (or inner containment) comprising:
  - a gusset for the connection between the inner wall and the basemat,
  - a cylindrical part (called skirt) with a bracket on the inner face to support the pivot crane and diametrically vertical ribs on the outer face for anchoring the horizontal prestressing tendons,
  - a torospherical dome, connected to the skirt by a belt comprising the upper anchors of the vertical prestressing tendons and those of the dome prestressing. The underside of the dome comprises prefabricated items acting as lost formwork for the inner surface and which do not play a role in the strength of the structure,
- a reinforced concrete outer wall (or outer containment),
- an annulus between the containments.

The prestressing of the inner wall uses the same type of prestressing as on the 900 MW plant series, which is prestressing by post-tensioning using tendons, followed by injection of cement grout.

7.1.1.3 DESCRIPTION OF THE EPR CONTAINMENT

The EPR containment comprises:

- a prestressed concrete inner wall (or inner containment) comprising:
  - a gusset for the connection between the inner wall and the basemat,
  - a cylindrical part (called skirt) with a bracket on the inner face to support the pivot crane and diametrically opposed vertical ribs on the outer face for anchoring the horizontal prestressing tendons,
  - a torospherical dome, connected to the skirt by a belt comprising the upper anchors of the vertical prestressing tendons and those of the dome prestressing, The underside of the dome comprises prefabricated items acting as lost formwork for the inner surface and which do not play a role in the strength of the structure,
  - a metal liner covering the entire inner face of the inner containment,
- a reinforced concrete outer wall (or outer containment),
- an annulus between the containments.

In the annulus, the EDE system enables negative pressure to be created so that in the event of an accident, any leak from the inner containment can be collected.

One specific feature of the EPR is a reinforced concrete basemat common to all the buildings of the nuclear island, on which the reactor building rests.

The upper part of the outer containment is designed to withstand an airplane crash. Along with the structures protecting the buildings housing the fuel building and the safeguard buildings of two of the four safety trains, it constitutes the APC shell.

The prestressing of the inner wall uses the same type of prestressing as on the fleet in operation, that is prestressing by post-tensioning using tendons, followed by injection of cement grout.

7.1.1.4 CONTAINMENT INSTRUMENTATION SYSTEM

As of the construction stage, the containment is equipped with instrumentation, distributed around the basemat, the gusset, the cylindrical part and the dome.

Note: The outer containment of double-wall containments is not instrumented.

The instrumentation system comprises:

- vibrating wire strain gauge (acoustic strain gauge),
- thermocouples,
- pendulums,
- Invar wires,
- dynamometers on 4 pure grease-injected vertical tendons on the first reactor of each site,
- direct levelling references (placed in the prestressing galleries).

7.1.2 ASSESSMENT OF AGEING OF CONCRETE CONTAINMENTS

The ageing mechanisms applying to containments are identified on the basis of knowledge of their operation and on inspections and tests carried out since the beginning.

The following table presents the various ageing mechanisms identified and described in Ageing Analysis Sheets (AAS) and their applicability according to the type of containment concerned.

Mechanism	**Containment with liner (900 MWe series)**	**Double-wall containment without liner (P4, P'4 and N4 series)**	**EPR**
Cracking of the prestressed concrete which could lead to corrosion of the passive rebars or prestressing tendon ducts	X		
Cracking of the reinforced concrete of the outer containment which could lead to corrosion of the passive rebars		X	X
Loss of prestressing by creep coupled with concrete	X	X	X

shrinkage, relaxation of tendons, tendon corrosion			
Corrosion of prestressing covers, anchorages and tendons	X	X	X
Pitting corrosion of the metal liner	X		X
Blistering of the metal liner	X		X
Ageing of composite coatings to supplement the tightness of the inner surface of the inner containment		X	
Ageing of paint and non-reinforced coatings	X	X	X
Malfunction of containment instrumentation system	X	X	X
Risk of concrete internal swelling (alkali-aggregate reaction, delayed ettringite formation)	X	X	X
Concrete carbonation	X	X	X

Table 12 – Ageing mechanisms affecting containment and written up in AASs

These ageing mechanisms are monitored by the maintenance programmes.

7.1.2.1 CRACKING OF THE PRESTRESSED CONCRETE WHICH COULD LEAD TO CORROSION OF THE PASSIVE REBARS OR PRESTRESSING DUCTS OR THE PRESTRESSED REBARS (900 MWE SERIES)

Under the effect of various phenomena (creep, shrinkage, containment pressure tests, freeze-thaw cycle, internal swelling reaction, etc.), concretes can crack.

A deep crack emerging on the outer surface could lead to corrosion of the rebars or prestressing ducts, by facilitating the penetration of corrosive agents (water, CO_2, chlorides). The prestressing tendons are for their part protected by ducts and cement grout.

A review of the condition of the outer facing of the 900 MWe containments was carried out in the early 2000s and confirmed the satisfactory condition of the 900 MWe containments, enabling the analyses to be restricted to corrosion of the dome rebars. This review led to visual inspections of the dome, the results of which led to the following measures being taken:

- the deployment of a national campaign on all the reactors of the CPY series consisting of treatment of cracks (opening of 0.3 mm or more) by injection of resins. The campaign was completed in 2010;
- definition of a frequency of 5 years ± 15 months for visual inspection of coastal reactors and of ten years ± 15 months for the other facilities.

These analyses show that there is nothing liable to suddenly modify the mechanical behaviour of the 900 MWe containments during their lifetime up to 60 years. This AAS is classified as status 0 (see table n° 8).

7.1.2.2 CRACKING OF THE REINFORCED CONCRETE OF THE OUTER CONTAINMENT WHICH COULD LEAD TO CORROSION OF THE REBARS (1300 MWE AND N4 SERIES)

Under the effect of various phenomena (creep, shrinkage, containment pressure tests, freeze-thaw cycle, internal swelling reaction, etc.), concretes can crack.

A deep crack emerging to the outer surface could lead to corrosion of the outer containment rebars.

Visual inspections of the two facings of the outer containment at intervals of 5 years ± 15 months for coastal sites and ten years ± 15 months for the other sites, is a means of detecting this phenomenon and implementing corrective measures to deal with it as necessary.

This phenomenon is not liable to suddenly modify the mechanical strength of the containments during their lifetime. This AAS is classified as status 0 (see table n° 8).

7.1.2.3 LOSS OF PRESTRESSING (ALL SERIES)

The phenomena of concrete shrinkage and creep and prestressing tendon relaxation lead to a drop in the prestressing of the containment over time.

Concrete creep under a sustained load, when coupled with concrete shrinkage, depends on the composition of the concrete and cannot therefore be delayed or neutralised. Relaxation of the prestressing tendons cannot be delayed or neutralised either, as these tendons are injected into a cement grout.

There is no means of counteracting the loss of tendon prestressing. Monitoring is therefore necessary. This is achieved by periodically measuring the deformation of the concrete at a frequency tailored to the kinetics of the phenomenon, as well as by measuring the instantaneous deformation during containment tests.

As prestressing is injected with cement grout, the tendons can be neither repaired nor replaced. This AAS is classified as status 2 (see Table n° 8).

7.1.2.4 CORROSION OF PRESTRESSING COVERS, ANCHORAGES AND TENDONS (ALL SERIES)

Full-penetration corrosion of the protective covers on the ends of the tendons could then lead to corrosion of the tendons themselves. The standard part of the tendons is protected from corrosion by the cement grout which passivates the steel.

The corrosion mechanism affecting the tendon ends and anchorages has been identified on the 900 MWe plant series only, and more specifically for coastal sites, although the assessments carried out on the general conditions of the tendon anchorages for the 900 MWe plant series concluded that it was not necessary to define measures in addition to the inspection of the tendon anchorage protective covers during the ten-yearly outage inspections.

This mechanism has not been confirmed for the 1300 MWe plant series. The double-containment design of the reactor building of the 1300 MWe and N4 plant series protects the prestressing components from the potentially corrosive marine atmosphere.

The visual inspections performed during each ten-yearly outage inspection identify the condition of the covers and can detect this phenomenon if it occurs, with implementation of corrective measures to deal with it. This AAS is classified as status 0 (see table n° 8).

7.1.2.5 PITTING CORROSION OF THE METAL LINER (900 MWE SERIES)

Pitting corrosion can develop in humid, unconfined, unprotected areas of containment liners. The formation of numerous full-penetration pits could degrade the leaktightness of the containment.

In the early 90s, containment liner corrosion flaws on the 900 MWe plant series were observed at the cylinder-gusset junction and the liner pressurisation channels at the bottom of the reactor building, but had no impact on the degree of leakage from the containments, as verified during the containment pressure tests. Following these observations, in-depth investigations were carried out on all the reactors of the 900 MWe series and work has been undertaken to stop this corrosion. In addition, R&D studies were carried out to assess the kinetics of this phenomenon according to the ambient conditions. They concluded that the risk of perforation of the liner at the end of operation were very limited.

The liner is inspected during each ten-yearly outage and an ultrasonic thickness measurement is taken on the areas considered to be sensitive (blistered areas in particular).

During the course of the VD3, additional thickness measurements were taken around the periphery of the penetrations as well as on the liner of the dome on certain reactors representative of the fleet. The examinations carried out revealed no in-service degradation which could call into question the maintenance programme and the continued operation of the containments.

It is possible to repair the liner, but the difficulty involved varies with the location of the flaw. Areas buried under the internal structures basemat are hard, or even impossible to repair.

As the liner is not replaceable and in the light of OEF observed on the 900 MWe plant series, this AAS is classified status 2 (see table n° 8).

7.1.2.6 BLISTERING OF THE METAL LINER (900 MWE SERIES)

Blistering of the liner is caused by deformation of the containment. This phenomenon has been observed on several 900 MWe reactors, but has no impact on the leak rate measured during the containment pressure tests.

Periodic examination of the liner to monitor this phenomenon is carried out during the containment pressure test every ten years.

The reassessments of the mechanical behaviour of the prestressed concrete wall and metal liner showed that blistering leads to stresses in the metal liner which remain acceptable and that there is no risk of the liner tearing in an accident situation. This AAS is classified as status 0 (see table n° 8).

7.1.2.7 AGEING OF COMPOSITE COATINGS TO SUPPLEMENT THE TIGHTNESS OF THE INNER SURFACE OF THE INNER CONTAINMENT (1300 MWE AND N4 SERIES)

For certain double-wall containments, the leak rate has increased over time. To limit this development, composite tightness coatings are applied to certain areas of the structure.

The coatings selection took account of normal service irradiation until VD4.

The coatings are liable to degrade over time, either owing to physical ageing (range of surface phenomena such as flaking, cracking or transport phenomena such as the penetration of corrosive agents), or owing to chemical ageing (range of phenomena leading to a modification of the material at a molecular level). The properties of composite coatings can also degrade and, in the event of a LOCA, can lead to a loss of effectiveness or the release of debris liable to impair the RIS/EAS [SIS/CSS] recirculation function. Coating repair or replacement work is possible, but costly.

The principle of the periodic inspections is:

- visual checks and dynamic testing of the coatings to look for flaws, flaw development and an analysis of their harmfulness,
- adherence tests on control areas and on test pieces (steel support) to check the correct behaviour of the coating with respect to the specified alert criteria.

The periodic inspections and containment tests show no alteration of the properties of these coatings.

This AAS is classified as status 0 (see table n° 8).

7.1.2.8 AGEING OF CONTAINMENT PAINT AND NON-REINFORCED COATINGS (ALL PLANT SERIES)

In the event of a LOCA, the adherence of the paints and coatings applied inside the containment could be impaired, thus releasing debris liable to clog and disrupt the operation of the RIS[57] / EAS[58] [SIS/CSS] systems. The coating qualification requirements include normal service irradiation for a time up until VD4.

In addition, an analysis of the potential degradation of paints in the reactor building in a LOCA situation, concluded that:

- the paints are not harmful to RIS/EAS filtration, even if released entirely in the form of debris,
- no realistic irradiation has any impact on the release of debris,
- no immersion or prolonged running water has any impact on the release of debris,
- no micro-debris are released following a LOCA and immersion.

Further to this analysis, the non-release of debris in the event of a LOCA would no longer appear to be a functional requirement.

Moreover, the prescribed maintenance documents include periodic visual inspections allowing detection of any deterioration, with repair as necessary.

This AAS is classified as status 0 (see table n° 8).

7.1.2.9 MALFUNCTION OF CONTAINMENT INSTRUMENTATION SYSTEM (ALL PLANT SERIES)

The mechanical behaviour of the containment is regularly monitored in operation and during testing by means of an instrumentation system which measures overall deformation and displacement (see § 7.1.1.4).

Malfunctions of the instrumentation systems can be observed during the lifetime of the containments. This is why the containment instrumentation system is the subject of measures to increase the life of the basic instrumentation system through the deployment of an Optimum Instrumentation System (see § 7.2.1) and the deployment of remedial measures as necessary (facing strain gauges, replacement of instrumentation, etc.).

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57 Safety Injection System

58 Containment Spray System

This AAS is classified as status 0 (see table n° 8).

7.1.2.10 RISK OF CONCRETE INTERNAL SWELLING BY ALKALI-AGGREGATE REACTION, DELAYED ETTRINGITE FORMATION (ALL SERIES)

The alkali-aggregate reaction is a chemical reaction occurring in certain humid conditions between the alkalis in the cement paste of concretes, the amorphous silica contained in certain aggregates and the interstitial water. The gel that forms swells when it absorbs water, which can deform the structure and lead to micro-cracking or opening of existing cracks.

The delayed ettringite formation reaction leads to the delayed formation of ettringite, frequently as a result of heating of the concrete when young (massive parts poured in place or heat-treated concrete). This reaction involves the sulphates present in the interstitial concrete solution, the aluminates mainly from the cement and the water from the interstitial solution.

Delayed ettringite develops after hardening of the concrete and can be accompanied by deformation and cracking of the concrete. The delayed ettringite reaction can concern concretes subjected to temperatures of at least 65°C, reached during setting, or even subsequently. The risks are linked to the presence of water (external input), temperature, the SO_3, C_3A and alkali contents of the concrete.

These two pathologies are accompanied by the risk of internal swelling of the concrete.

All the containments of the fleet in operation, as well as all civil engineering structures that are Important for Safety, have undergone a potential risk assessment with regard to the alkali-aggregate reaction and the delayed ettringite formation reaction according to the components of the concrete used and the implementation conditions.

The instrumentation installed on the containments and the visual inspections are a means of detecting and monitoring this phenomenon on the containments.

These pathologies are irreversible and as no final treatment is available, they are the subject of special in-service monitoring, based on the risk assessment for each individual reactor.

The AAS concerning the alkali-aggregate reaction is classified status 2, while that concerning the delayed ettringite formation reaction is classified status 0 (see table n° 8).

7.1.2.11 CONCRETE CARBONATION (ALL SERIES)

Concrete carbonation is a chemical reaction between the free lime in the cement pate and the carbon dioxide gas which leads to the formation of calcite. This reaction brings down the pH of the concrete, which leads to depassivation of the steel of the rebars and encourages initiation of their corrosion.

The visual inspections carried out during each ten-yearly outage are able to detect the presence of calcite and any corrosion and enable any flaws to be treated.

Moreover, concrete core-samples are taken when installing the strain gauges on the facing (see § 7.2.1). Microstructural and physical-chemical analyses carried out on these samples show that there is no carbonation problem on the containments of the French fleet.

This AAS is classified as status 0 (see table n° 8).

7.1.2.12 CONCLUSION

Of the ageing mechanisms identified, only the following are the subject of Detailed Ageing Analysis Files (component DAAR):

- the loss of containment prestressing, for both types of containments,

- pitting corrosion of the liner for the containments with liner,
- the risk of concrete internal swelling owing to the alkali-aggregate reaction and delayed ettringite formation pathologies.

These DAARs demonstrate the fitness for service of the containments of the French fleet up to the end of their service life, taking account of the ageing phenomena which can affect them. This demonstration includes the continued fitness for service criterion for the containments defined by compliance with the pressure test leak rate criterion established on the basis of the overall leak rate allowable in an accident situation defined in the Creation Authorisation Decrees for Basic Nuclear Installations (BNI).

7.1.3 MONITORING, TESTING, SAMPLING AND INSPECTION ACTIVITIES FOR CONCRETE CONTAINMENTS

Maintenance is based on scheduled periodic monitoring (inspection, test) the aim of which is to determine the status of the items participating in the various functions and to analyse how they change between two tests or two inspections, in order to guarantee the durability of the structures for the duration of their service lifetimes.

Monitoring is carried out:

- during reactor operation;
- during reactor maintenance outages;
- during containment pressure tests.

7.1.3.1 SUMMARY OF THE MONITORING CARRIED OUT ON CONTAINMENTS

The following table is a summary of the monitoring carried out and how it applies to the containment functions:

In green: *applicable to single wall containments only.* ***In purple****: applicable to double-wall containments only.*

		In-service monitoring	**Monitoring during maintenance outage**	**Monitoring during containment pressure tests**
Mechanical behaviour	Single wall containment and inner wall of double-wall containments	- Every three months (every month for certain cases) Objective: monitoring containment deformation under the effect of prestressing and time (instrumentation) **- Inspection of outer wall** **Objective: search for traces of corrosion, or cracking (frequency every 5 years)**	No special measures (monitoring is the same as during operation if the normal date of measurement coincides with the outage period)	- Ten-year frequency Objectives: monitoring of wall deformation under the effect of pressure variation, performed during the overall tightness test
	Outer wall of double-wall containments	**- Inspection of outer wall** **Objective: search for traces of corrosion, or cracking (frequency every 5 years)**		

Tightness	Single wall containment and inner wall of double-wall containments	- Check on mechanical penetrations (continuous) Objective: reveal major penetration leaks during the operating cycle - Checks on electrical penetrations	- Penetration tightness tests Objective: check on integrity of penetrations	- Overall leak rate measurement - Partial leak rate tests at penetrations with search for and location of leaks
	Outer wall of double-wall containments	**Objective: monitoring of pressure variation in annulus** **Monitoring condition of sleeves (3 or 5 years)**		**- Measurement of leak rate from outer wall**
Other maintenance	Single wall containment and inner wall of double-wall containments		- Monitoring of galleries (no water), of tendon anchorage covers and plates, no grease leaks from prestressing ducts (frequency 5 years) - Check on condition of instrumentation system **- Check on tightness coatings on inner walls during outages, except for refuelling outages**	- Hatches operating test - U5 tests **- Inspection of liner** **- Inspection of inner walls tightness coatings**
	Outer wall of double-wall containments		**- Check on annulus access means**	

Table 13 – Summary of monitoring

7.1.3.2 IN-SERVICE MONITORING

Mechanical behaviour

7.1.3.2.1.1 Instrumentation

The mechanical behaviour of the containment is regularly monitored in operation and during containment pressure tests by means of an instrumentation system which measures overall deformation and displacement. As of the construction stage, the containment is equipped with instrumentation, distributed around the basemat, the gusset, the cylindrical part and the dome. The layout and distribution of the instrumentation are defined so as to be able to monitor:

- overall displacements:
  - settling and tilting of the structure, determined by the nature of the foundation soil, are defined using topographical levelling,
  - overall deformation of the cylindrical part is determined using pendulums (diameter) and vertical Invar wires (height),
  - differential displacements, in particular between the reactor building and fuel building are determined in order to verify the integrity of the transfer tube.
- local deformations:
  - local deformation of the concrete is provided by the vibrating wire strain gauges (acoustic indicators). During containment pressure tests, the results from the strain gauges located in

the straight part in the middle of the cylinder are used to calculate the instantaneous modulus of elasticity of the wall and its Poisson's ratio. Throughout the lifetime of the structure, the measurements taken with the strain gauges in the straight part are used to evaluate the shrinkage and creep of the concrete. In addition to the straight zones, these instruments are installed in the basemat, the gusset, the toroidal belt and the dome.

- the temperature of the concrete:
  - the temperature measurements are taken from the concrete of the wall using thermocouples. These sensors are used primarily to apply temperature variations to correct the raw values given by the strain gauges and to find out the thermal status of the concrete and thus define a thermal behaviour model.
- the tension of the prestressing tendons:
  - the tension of the prestressing tendons is determined using strain gauges installed on 4 vertical tendons injected with grease, on the first reactors of each site.
  - the integrity of prestressing is verified by calculations based on the gradual deformations measured.

Readings are taken from the instrumentation system by the NPPs every 3 months. Most of the sites have telemetry and the readings can be more frequent if necessary. The levelling measurements are taken every year, every 2 years, or every 5 years depending on the kinetics observed.

The purpose of these readings is to monitor the shrinkage and creep of the concrete of the wall and the structural settling phenomena.

The instrumentation system involves considerable equipment requiring monitoring and calibration of the measuring instruments. This point is covered by a specific maintenance programme.

A process for collection and analysis of the instrumentation measurements was set up in 2008. This process consists in defining the monitoring required and the alert emergence procedures.

Monitoring consists in:

- acquiring the readings taken by the Licensee,
- validating them and transforming them into raw results,
- correcting them with reversible effects (temperature, pressure, etc.),
- analysing them from a metrological viewpoint, comparing them with previous measurements as soon as they are received in order to validate them before integrating them into the database,
- asking for confirmation of readings or additional information from the Licensee as necessary,
- ensuring that they remain in a database at least for the service life of the structure, or even beyond, during the decommissioning and dismantling phase, if this proves necessary,
- periodically issuing instrumentation reports (in-service and during containment pressure tests) about this monitoring.

The alert consists in drawing the appropriate level of Licensee and Engineering attention to a particular point resulting from monitoring.

Alerts are given whenever it is felt to be necessary:

- at validation and correction of the measurements,
- during metrological inspection of the measuring device,
- on the occasion of the instrumentation reports issued during operation,
- or during containment pressure tests.

The following fields may be the subject of alerts:

- the availability status of the instrumentation system and its adequacy with respect to the optimum instrumentation system,
- abnormal development of a measured phenomenon by comparison with operating experience feedback acquired.

7.1.3.2.1.2 Containment ageing

In order to guarantee the integrity of the containment, a periodic visual inspection is carried out:

- for single-wall containments: visual inspection of the external facing (dome and parts of the skirt visible from the outside). The aim of this is to look for traces of corrosion of the rebars and concrete facing defects. The purpose of the visual inspection of the reactor domes is also to verify the integrity of the tightness coating, in the case of domes which are coated with a tightness compound (Bugey and Fessenheim).
- for double-wall containments: visual inspection of the inner surface and outer surface facings (for the outer surface facing, the parts visible from outside the reactor building and for the inner facing those visible from the annulus and opposite the former). The following are carried out:
    - either a partial inspection (search for traces of rebar corrosion),
    - or a complete inspection (search for traces of corrosion and survey of cracks).

The criteria applicable to ageing monitoring are defined in a special maintenance programme.

As the inspections can be carried out during operation, the chosen frequency is:

	Single-wall containments	**Double-wall containments**
Coastal sites	5 years ± 15 months	5 years ± 12 months for a complete inspection.
Other sites	10 years ± 15 months	5 years ± 12 months for a partial inspection, 10 years ± 12 months for a complete inspection.

Every ten years, the inspection coincides with that scheduled during the containment pressure test.

Tightness

During operation, the tightness check on the inner wall of the double-wall and single-wall containments is by means of a special device (SEXTEN, based on monitoring the pressure of the containment and the instrument air inputs).

In addition, for the double-wall containments, the tightness of the outer wall is continuously monitored by measuring the negative pressure within the annulus. An analysis of these recordings is a means of verifying the tightness of the outer wall.

For double-wall containments in which the inner side of the inner wall was been coated, regular checks are made to monitor the condition of the coating. The maintenance programme comprises the following checks:

- Visual check:
  The check comprises a visual examination of coating flaws identified during the previous outage. These checks are performed during refuelling outages and during partial maintenance outages.
- Visual check and dynamic testing:
  The checks consists of a visual examination and specific testing of the entire coating except for the areas requiring the relocation of equipment items. These checks are performed:
  - During the ten-yearly outage, before and after the containment pressure test,
  - During the second partial maintenance outage following the ten-yearly outage (that is 5 ± 1 years after the ten-yearly outage).
- Adherence check:
  The check consists in verifying any change in the adherence on the control zones and on the steel test specimens. It is performed during the ten-yearly outage, after the containment pressure test.
- Final check:

  In addition, at the end of each outage, a visual examination is carried out to check that the coating has not been damaged by handling or maintenance operations carried out nearby.

7.1.3.3 MONITORING DURING CONTAINMENT PRESSURE TESTS

The containment pressure test comprises actions and measures concerning its tightness and its mechanical behaviour. It is performed at the design-basis pressure.

These actions comprise:

In green: applicable to single wall containments only,

In purple: applicable to double-wall containments and EPR only

- The actions prior to the pressure test:
  - Measurement of leaks from the penetrations,
  - Inspection of the prestressing galleries,
  - **Inspection of the metal liner (also for EPR),**
  - **Visual examination of the outer facing surface of the containment (survey of the most significant cracks which are instrumented for monitoring during the pressure test)**,
  - **Complete inspection of the inner facing of the inner containment (condition of coatings, cracks, sensitive points, condition of penetration zones, etc.),**
  - **Detailed inspection of inner and outer facings of the inner wall in the equipment access hatch zone,**
  - **Performance of standard tests on the annulus depressurisation system,**
  - **Measurement of leak rate from outer wall,**

- Actions during containment pressure tests:
  - Tightness test by measurement of the overall leak rate from the containment wall,
  - **Measurement of leak rate from outer wall,**
  - Strength test by complete readings from the instrumentation system:
    - Measurement of deformation of the containment wall,
    - Survey of differential concrete/equipment hatch shell displacements,
  - **Visual examination of the surface of the containment outer facing (monitoring of cracks identified after containment pressure tests, survey of new cracks).**
  - **Location and quantification of leaks found on outer facing of inner containment.**
- Actions after containment pressure tests:
  - **Inspection of the metal liner (for EPR also),**
  - **Visual examination of the surface of the containment outer facing (monitoring of cracks previously identified).**

7.1.3.4 TYPES OF TIGHTNESS TESTS AND ASSOCIATED CRITERIA

The General Operating Rules define the various types of tests to be performed, along with the acceptability criteria for these tests.

Containment tightness tests

These are overall containment tests designed to measure the tightness of the inner containments, as well as the outer containments as applicable.

The frequency of these containment tests is ten years and they are carried out at the design-basis pressure during each ten-yearly outage.

The leak rate criterion during the pressure test is based on the maximum leak rate in accident conditions defined in the BNI Creation Authorisation Decrees. It constitutes a fitness for service criterion for the containments of the EDF NPP fleet.

7.1.3.4.1.1 Total leak rate (all series)

The total allowable leak rate in an accident situation (noted F_a) is defined for the containments of the fleet in service in the BNI Creation Authorisation Decrees. It is limited to:

- 0.3% per day of the total mass of gas contained in the containment in LOCA conditions for single-wall containments (900 MWe plant series),
- 1.5% per day of the mass of gas contained in the inner containment in LOCA conditions for double-wall containments (1300 MWe and N4 series).

For the EPR, the total allowable leak rate in accident conditions is identical to that of a single-wall containment, that is F_a = 0.3%/day of the total mass of gas contained in the inner containment in accident conditions.

The pressure test equivalent leak rate (F_e) is obtained by considering a transposition factor defined taking account of the difference between the test pressures and temperatures and those of accident conditions. To take account of the ageing of the materials making up the containment and cover a potential drop in the confinement capacity of the containment between two pressure tests, a factor k is considered to define the acceptable leak rate during the pressure test (F_e^{acc}).

As defined in the general operating rules (RGE), the criterion concerning the total leak rate from the containment is a group A criterion. The RGE A criterion classification is applied to the test criteria for which non-compliance compromises one or more safety objectives. If an RGE A criterion is not met, the safety function (equipment or systems) is declared unavailable. In this case, non-compliance with the leak rate during the ten-yearly outage precludes reactor restart after containment pressure tests.

There is also a group B criterion (RGE B) concerning the consumption of the margin between two pressure tests (according to the hypothesis of linear development of the leak over time). For a pressure test N and the previous test N-1, the leak rate measured Fm^N must comply with the equation:

$$F_m^N - F_m^{N-1} \leq 0.75(F_e^{acc} - F_m^{N-1})$$

In the event of non-compliance with the group B criterion, which aims to verify that the ageing rate is not liable to create a deviation from the EDF baseline safety requirements, EDF is required to carry out a new pressure test within five years ("five-yearly" pressure test).

7.1.3.4.1.2 Outer containment leaks (1300 MWe and N4 series and EPR)

What is important is to prevent any leak from the annulus to the outside, other than via the EDE annulus ventilation system. Negative pressure must therefore be maintained at all points in the annulus in order to compensate for wind-induced suction phenomena.

The outer containment is considered to be sufficiently leaktight if its leak rate with a negative pressure of 3 hPa is less than 1% per day of the air mass contained in the space bounded by the outer containment. This test is performed every ten years or during containment pressure tests. A test to verify the outer containment leak rate is also carried out every cycle (with the reactor in operation) via periodic tests of the EDE system.

The leak rate adopted for the EPR at construction is 1.2% per day of the volume of gas bounded by the outer containment. This rate is given for a negative pressure of 6.2 hPa.

7.1.3.4.1.3 Non-through-wall leaks (1300 MWe and N4 series)

In addition to the overall test, for double-wall containments, non-through-wall leaks are measured, in other words leaks which go directly to the outside (either into the peripheral buildings, or into the environment) without passing through the annulus. This measurement thus encompasses:

- direct leaks,
- leaks through the basemat which do not go to the annulus.

There is an acceptance criterion for non-through-wall leaks.

7.1.3.4.2 Penetration tightness tests

Partial tests (type B tests) are run on the penetrations with passive tightness devices. The following in particular are concerned:

- the equipment access hatch (TAM),
- the personnel airlock,
- the transfer tube,
- the pressurisation penetration,

- the penetrations fitted with blind flanges,
- the electrical penetrations.

These type B tests are mainly carried out during each maintenance outage.

There are also two tests (type C tests) on penetration active tightness devices (manual or automatic shutoff valves actuated in the event of an accident).

For the single-wall containments of the 900 MWe plant series and for the EPR, the maximum share of the overall leak rate attributable to the penetrations is considered to be 60%. This thus leads to an allowable fixed leak rate through the penetrations in an accident situation (F_{ad}) equal to 60% of F_a, and thus to a pressure test leak rate acceptability criterion for the penetrations during containment pressure tests.

For the 1300 MWe and N4 plant series, the maximum share attributable to the penetrations is determined by applying correction coefficients to the above-mentioned 60% factor for the 900 MWe reactors, based on the difference between the number of penetrations and the difference between the pressure peaks in an accident situation.

7.1.3.4.2.1 Summary

	Variable
Single wall containment and inner wall of double-wall containments	Free volume (m^3)
	$P_{pressure\ test} = P_{Dim}$ (MPa abs.)
	Accident temperature (°C)
Overall leak: Type A test	Allowable leak rate in accident: F_a (%/d)
	Allowable leak rate in pressure test: F_e (%/d)
	Containment acceptability criterion: F_e^{acc} (%/d)
	Acceptance leak rate: Q_e^{acc} (Nm^3/h)
Leaks through penetrations B and C	Allowable leak rate in accident: F_{ad} (%/d)
	Allowable leak rate in pressure test: F_{ed} (%/d)
	Penetration acceptability criterion (B+C): F_{ed}^{acc} (including uncertainty) (%/d)
	Acceptance leak rate: $Q_e^{acc\ BC}$ (Nm^3/h)
Non-through-wall leak	Acceptable rate (Nm^3/h)
Type B test	Electrical penetrations
	Equipment hatch seal (Nm^3/h)
	RB access airlock (Nm^3/h)
	Blind flanges (Nm^3/h)
	Total penetrations type B (Nm^3/h)
Type C test	Piping shutoff devices (Nm^3/h)
Outer containment	Allowable leak rate (%/d)

7.1.3.5 MECHANICAL BEHAVIOUR

There is no specific criterion concerning the deformation and displacement amplitudes measured.

The pressure test results are analysed as a whole, with reference to the anticipated values determined by calculation, to the results of the previous tests and to changes according to the pressure variation. The main aim is to verify the linearity and reversibility of the deformations.

7.1.3.6 INSPECTION OF THE METAL LINER

A metal liner inspection is prescribed before and after the pressure test in order to look for blistering and traces of corrosion and determine how they develop during the test. This inspection notably consists in:

- A remote survey of the blistering of the wall over the entire visible surface of the cylindrical part,
- A remote visual inspection of all plate welds, more specifically entailing a search for corrosion,
- A precise survey of blistering on 5 plates of the wall, chosen owing to the presence of blistering before the pressure test. These 5 plates are kept for the same inspections during the subsequent pressure tests,
- Ultrasonic thickness measurement on the most significant blisters of the 5 plates, marking the points at which the measurements are taken,
- An inspection of the metal liner in the areas of the main penetrations and the penetrations which could be included in the blistering.

7.1.4 PREVENTIVE AND CORRECTIVE MEASURES FOR THE CONCRETE CONTAINMENTS

7.1.4.1 REPAIR OF OUTER FACINGS – RESIN INJECTION INTO CRACKS

Following visual inspection of the facings, cracks with an opening of more than 0.3 mm are injected to ensure leaktightness. A national campaign was carried out on all the 900 MWe reactors consisting of injecting resin into the zones with cracks open by 0.3 mm or more. The campaign was completed in 2010.

7.1.4.2 INSTALLATION OF COATINGS ON THE INNER SURFACE OF DOUBLE-WALL CONTAINMENTS

Any increase in the leak rate from the inner containment is currently offset by additional tightness measures (composite coatings) applied to the inner surface of the containment outer wall, enabling the criterion in the test rule to be met during the containment pressure tests.

The phenomena responsible for any increase in the leak rate diminish over time. EDF therefore carried out an advance study of the repair solutions for restoring safe confinement for containments on which the inner-surface tightness consolidation works have reached their limit. Solutions to repair the outer surface of the containments most susceptible to the ageing phenomenon will be applied in the next few years and in any case before the ten-yearly outage inspections.

7.2 EDF EXPERIENCE WITH APPLICATION OF THEIR AGEING MANAGEMENT PROGRAMME FOR CONCRETE CONTAINMENTS

7.2.1 STUDIES TO REASSESS THE MECHANICAL BEHAVIOUR OF THE WALL

Analysis of the measurement results from the containment instrumentation system revealed differences between the design-basis behaviour and the behaviour observed on-site. These differences can be attributed to the loss of prestressing of the concrete wall.

Following this observation, to demonstrate the fitness for service of the containments until the end of their operating lives, reassessments of the mechanical behaviour of the prestressed concrete walls were carried out. These studies include the measurement data showing the loss of prestressing and the actual status of the stresses within the concrete wall. They use the results from the instrumentation system measurements, updated and extrapolated for potential operation of the reactors to 60 years. They allow:

- evaluation of the mechanical performance of the concrete wall. The status of stresses in the concrete, in the prestressing tendons and the passive rebars is analysed taking account of gradual deformation, extrapolated from the values measured by the instrumentation system,
- identification of the susceptible and singular sites and areas, in the light of the gradual deformation,
- calculation of the wall deformation status necessary for the behaviour study of the metal liner for the 900 MWe plant series containments.

The results of these studies obtained on the basis of the gradual deformation reassessed in relation to the design, confirm the structural strength of the concrete wall of the containments for all plant series for an operating life of up to 60 years. For the containments of the 900 MWe plant series, the stresses in the metal liner are dependent on the deformation of the concrete wall: the studies performed show that the mechanical behaviour of the liner ensures confinement by the containment in all service conditions (normal service or accident situations).

7.2.2 OPERATING EXPERIENCE FEEDBACK ON METAL LINER CORROSION

In the early 90s, corrosion flaws were observed at various low points on the metal liner of the reactor containments of the 900 MWe series:

- at the peripheral local expansion seals at the interface between the basemat of the internal structures and the containment gusset,
- at the metal liner pressurisation channels at the bottom of the RB.

Following these observations, in-depth investigations were carried out on all the reactors of the 900 MWe plant series, which concluded that the particular location of the flaws could be explained by the presence of “stagnant and aerated” water in contact with the metal liner, along with chlorinated pollutants owing to leaching of the seal initial filling material (Flexcell) which were conducive to corrosion phenomena.

The following repair work was initiated on all the containments of the 900 MWe plant series:

- in the case of the containment basemat-gusset seal:
    - removal of the initial filling material (Flexcell),
    - application of anti-corrosion paint on the metal liner at the upper part of the peripheral seal,
    - repair by welding pads to the full-penetration flaws and most significant pitting corrosion (depth of 5 mm),
    - filling of the seal with petroleum wax,

- installation of protection using silicone type sealant caulking,
- (before hardening of the sealant) installation of a metal cover strip to minimise any “voids”, protect the seal mechanically and deflect any splashing water.

- the pressurisation channels were filled with cement grout (with basic pH to minimise the corrosion kinetics) in order to stop the ingress of water and air.

Additional studies were also carried out to assess the corrosion kinetics in various configurations. EDF concluded that the risks of perforation of the liner at the end of operation were very limited.

The occurrence of a loss of containment tightness on Bugey reactor 5 could potentially call the conclusions of these studies into question. The origin of this event was corrosion of the liner at the bottom of the peripheral seal, the precise location of which could not be identified. Repairs were made, consisting in replacing the basemat-containment gusset seal to stop the corrosion mechanism and restore containment tightness at this seal.

7.2.3 CONCRETE PATHOLOGIES - ALKALI-AGGREGATE REACTION AND DELAYED ETTRINGITE FORMATION.

The instrumentation system installed on the containments and the visual inspections enable the appearance of local internal swelling of the concrete to be detected. Elongations were measured using strain gauges embedded in the concrete of the basemats of the Chooz B containments. After analysis, these elongations were attributed to delayed ettringite formation. Visual inspections carried out as part of the maintenance programmes revealed the appearance of an alkali-aggregate reaction in the concrete of the internal structures at Civaux.

The scale of these phenomena nonetheless remains very limited on the French NPP fleet.

7.2.4 INSTRUMENTATION FAILURE – OPTIMUM INSTRUMENTATION SYSTEM

When the fleet was designed, the containment instrumentation system was the response to a significant need for instrumentation to verify the design hypotheses and ensure satisfactory construction. Instrumentation is today required for ageing management. Measures are being taken to increase the life of the instrumentation system through the deployment of an Optimum Instrumentation System. This is a means of addressing this requirement. The principles of the optimum instrumentation system comply with the definition of containment prestressed wall instrumentation means that are necessary and sufficient for:

- monitoring the structure,
- being able to carry out the required demonstration studies throughout the forthcoming industrial operation period,
- guaranteeing compliance with the functional requirements applicable to the containment instrumentation system.

The optimum instrumentation system thus comprises:

- the original instrumentation with replacements if faulty (topographical references, pendulums and invar wires, thermocouples, some strain gauges),
- additional instrumentation (new vertical invar wires, facing strain gauges).

The optimum instrumentation system is being deployed to all containments of the fleet in operation.

During the containment pressure tests, the behaviour of the facing sensors in the optimum instrumentation system, in terms of the amplitudes recorded, the measurement linearity and their impact on the calculation of the modulus of elasticity, is comparable to that of the embedded sensors. In operation, after a period of about 6 months following their installation, the deformation rates recorded, when averaged with those of the embedded sensors, allow a coherent estimation to be made of the containment creep amplitude.

7.3 RESEARCH REACTORS

7.3.1 CEA

7.3.1.1 SCOPE OF THE PROGRAMME AND AGEING ASSESSMENT

Description of containments

- ORPHÉE reactor:

The containment comprises a cylindrical skirt (Ø = 28 m, h = 30.65 m), a dome and a basemat. The "controlled leakage" type containment is supplemented by a leak collection annulus outside the skirt and designed to collect leaks from around the penetrations.

The skirt is made of reinforced concrete 60 cm thick. For the dome, the thicknesses vary between 30 cm (at the keystone) and 60 cm (at the springing). There is no prestressing of the containment.

- CABRI Reactor:

The reactor building is a semi-buried rectangle of dimensions L= 19.08 m, w = 12.58 m, h = 12.30 m. The containment is of the "controlled leakage" type. The walls of the reactor building are made of reinforced concrete 20 cm thick. The containment is designed to be capable of withstanding an accident overpressure of 0.04 bar and the design-basis earthquake. There is no prestressing of the containment of the CABRI reactor.

- RJH Reactor:

The building of the RJH nuclear unit constitutes the containment (third barrier). The containment is cylindrical (Ø = 35 m, h = 39.7[59] m), made of concrete and capped by a torispherical dome.

The nuclear unit containment comprises the following:

- the concrete shell of the building,
- the mechanical penetrations,
- the fluids penetrations, up to the isolation device in the leak collection zone,
- the electrical penetrations,
- the nuclear unit basemat.

The shell is made of partially prestressed high-performance reinforced concrete 80 cm thick (70 cm for the dome), with a concrete basemat 120 cm thick and not prestressed.

The shell prestressing tendons are of three types:

- horizontal tendons over the height of the cylindrical part superstructure above level 0 corresponding to the service floor.
- bottom-tensioned pure vertical tendons from the inter-basemat space and anchored in the toroidal belt,
- gamma tendons, which are vertical tendons extended into the dome and tensioned at both ends.

All the penetrations are routed to the leak collection zone belonging to the nuclear auxiliaries building, adjacent to the nuclear unit.

[59] 42.74 m with the crypt.

A tightness test will be performed on the nuclear unit on acceptance of the containment and then every 10 years thereafter. Finally, a provision to take account of containment ageing (potential variation in direct leak rate at the beginning and end of its lifetime) has been included.

Assesment of ageing

For research reactors, containment ageing is monitored by:

- examining compliance during the periodic safety reviews every ten years,
- periodic tests and inspections,
- a crack monitoring programme for certain reactors.

The nature of the ageing management steps taken is similar for the four installations concerned (Cabri, Orphée, RJH and RHF) and covers:

- monitoring of containment cracks,
- verification of containment leak rates and penetration tightness.

The scope of these actions applies to each research reactor, taking account of the specific features of each facility. There are significant differences between the containments of the four facilities concerned (the design-basis accidents and loadings considered are different). For example, only the concrete structures of the RJH (upper part) and of the RHF (at the junction with the dome) are prestressed, the geometry and thicknesses of the concrete in the containments vary (CABRI is rectangular, while the others comprise a cylinder with dome) and the RHF has a double-wall with metal outer containment and concrete inner containment. The RJH, currently under construction, will offer additional features for managing and monitoring the ageing of its containment. For example, it will be possible to replace the prestressing tendons if necessary, monitoring of containment and retaining wall deformation will be implemented as will monitoring of the anti-seismic bearing pads supporting the nuclear unit. Some of these provisions are the subject of specific ASN prescriptions.

7.3.1.2 MONITORING ACTIVITIES, PREVENTIVE MEASURES AND EXPERIENCE OF THE LICENSEES

The following containment monitoring activities are carried out for the ORPHÉE reactor:

- annual monitoring of micro-cracks in the concrete of the containment and the internal structures;
- every three years, a tightness check on the containment penetrations (leak rate);
- every five years, a check on the overall leak rate of the containment.

The reactor building leak rate, measured every 5 years, must be less than the allowable overall leak rate of 200 m^3/h (or 24% of the total volume, with over-pressure of 135 mbar). This value can be broken down as follows:

- 6% by the concrete and the hatch (or 50 m^3/h),
- 18% by the penetrations to the chamber (or 150 m^3/h).

The following inspections are performed on the CABRI reactor:

- an annual check on the leak rate from the reactor building shell, penetrations and door seals;
- a ten-yearly check during the periodic safety review, which is both visual (direct visual examination of the condition of the civil engineering structures (inside and outside), roof leaktightness, retaining structures) and uses representative sampling (evaluation of concrete carbonation and condition of

rebars), in association with investigations using FERROSCAN type non-destructive measurements.

The maximum allowable leak rate for an over-pressure of 10 daPa is 300 m^3/h. The air leak rate from the reactor building is checked every year.

For the RJH reactor, a structural monitoring programme is planned, comprising:

- a check on the mechanical behaviour of the containment (monitoring of deformation resulting from concrete shrinkage/creep, containment displacements, settling of the structure) using a permanent instrumentation system;
- a containment pressure test every ten years, to measure the leak rate from the containment and the tightness of the penetrations;
- monitoring of the prestressing tendons (visual inspections, weighing and replacement of tendons if necessary);
- monitoring of the anti-seismic bearing pads (visual checks, analysis of measurement by the installed instrumentation system, testing of samples);
- monitoring of the condition of the drains, paintwork and coatings, stacks, roof tightness, façade cladding.

The maximum authorised leak rates have been set by ASN (resolution 2011-DC-00226 of 27 May 2011, prescription [BNI 172-46]) at 0.7 %vol./day for uncollected leaks and 5% vol./day for collected leaks with an overpressure of 110 mbar. As of the design stage, CEA must implement an optimisation approach to reduce the collected leak rate (prescription [BNI 172-47] of the above-mentioned resolution).

7.3.2 THE ILL

7.3.2.1 SCOPE OF THE PROGRAMME AND AGEING ASSESSMENT

The RHF containment takes the form of a cylindrical skirt (Ø = 60 m, h = 53 m), with a spherical dome at its top and a basemat. The containment comprises a double structure, with:

- a reinforced concrete inner containment 40 cm thick (30 cm at the dome),
- a steel outer containment 11 mm thick (7 mm at the dome). The buried part of this containment is made of concrete,
- a basemat,
- a few penetrations.

The annulus is pressurised to 135 mbar relative to the pressure inside the concrete containment. In normal operation, the concrete containment is designed to withstand an external over-pressure of 150.5 mbar and an internal over-pressure of 150 mbar. The metal containment is designed to withstand an internal over-pressure of 150.5 mbar.

During construction, the belt was prestressed with tendons. The containment is the subject of ageing monitoring. Following inspections, the containment was reinforced by the addition of external prestressing with tendons in the zone situated at the top of the cylindrical wall, just below the belt. The lower slab was also prestressed.

7.3.2.2 MONITORING ACTIVITIES, PREVENTIVE MEASURES AND EXPERIENCE OF THE LICENSEES

The reinforced concrete structures participating in confinement are protected from direct atmospheric hazards and are regularly inspected. The outer metal containment is also the subject of regular, extensive inspections.

The checks on the reinforced concrete structures include monitoring of crack development, geometrical checks on the containment to differentiate between the various pressure situations, as well as leaktightness inspections. The metal containment undergoes visual inspection, dye-penetrant inspection of the welds and leaktightness checks. Every 5 years, the leaktightness of the reinforced concrete containment/metal containment assembly (direct leaks) is checked.

For a pressure of 135 mbar in the annulus, the concrete containment's nominal leak rate from the annulus to the inner containment is 200 m^3/h. The leak rates for an internal annulus over-pressure of 135 mbar are:

- 400 m^3/h from annulus to inner concrete containment,
- 400 m^3/h from annulus to outer metal containment.

During the first years, cracks appeared in the concrete containment and the join between cylindrical part and dome, so the decision was taken to add extra prestressing tendons.

7.4 ASN ASSESSMENT OF THE AGEING MANAGEMENT PROGRAMME FOR CONCRETE CONTAINMENTS

7.4.1 NUCLEAR POWER REACTORS

7.4.1.1 DESIGN OF CONTAINMENTS FOR THE EDF FLEET IN SERVICE. CONSEQUENCES FOR AGEING MANAGEMENT

The design of the containments of the EDF fleet is relatively uniform. They are reinforced and prestressed concrete shells about one metre thick placed on a thick reinforced concrete basemat. These containments are mechanically separated both from the internal structures of the reactor building and from the structures of the peripheral buildings. The same type of prestressing, with corrosion protection by means of injection of cement grout, is present in all the containments.

The instrumentation system on these structures comprises numerous instruments to measure deformation and displacement, placed in all the standard and singular areas of the wall. This system is a means of observing the mechanical behaviour of the structure, throughout its lifetime.

This design uniformity makes it easier to monitor the containments, by enabling those on the same site or on different sites to be compared, along with containments of the same design but different ages. If a particular phenomenon is observed on a containment, this then constitutes an alert which can be incorporated into the monitoring programme for all the containments.

Within this fleet with a uniform structural design, the structures can be divided into two separate sub-groups: single-wall containments with metal liner, for which confinement is static, and double-wall containments, for which the static confinement of the inner wall is supplemented by dynamic confinement, consisting of drainage and filtration of the air between the two walls. Certain ageing mechanisms are therefore specific to one or other type of containment.

7.4.1.2 AGEING MANAGEMENT PROGRAMME

Scope of ageing management

The scope of the ageing management adopted by EDF (see § 7.1.1) covers the civil engineering structure constituting the containment (one or two walls of reinforced concrete, a prestressing system and a metal or composite liner). The seals are also included within the scope[60]. The scope adopted by EDF is satisfactory when compared with the ENSREG specifications for concrete containments, as well as for their tightness coatings.

ASN points out that in the scope, EDF includes the painting of the various containment internal structures, even though this has no role in the tightness of the containment, but its behaviour must not impair the filtration of the recirculation function. This point will be reviewed by ASN and IRSN during the examination of fitness for service beyond the VD4.

Assessment of ageing

To assess the ageing of the concrete containments, EDF has listed and detailed (see § 7.1.2) the ageing mechanisms it considers to be pertinent, based on its knowledge of the working of the containments and experience feedback from the inspections and tests performed.

These mechanisms concern the concrete (cracking, creep, shrinkage, carbonation and internal swelling pathologies), the metal liner (corrosion and blistering), the prestressing tendons (relaxation and corrosion), the passive rebars (corrosion), the instrumentation system (malfunction of certain instruments), as well as the paints and composite coatings (ageing and behaviour in accident situation, including a severe accident). This list is considered by ASN to be exhaustive.

ASN also underlines the completeness of the R&D programme developed by EDF.

The acceptance criteria for the above-mentioned mechanisms – not specified in this report – are indicated in the document specifying the containment maintenance rules, more particularly the basic preventive maintenance programmes.

7.4.1.3 MONITORING, TESTING, SAMPLING AND INSPECTION ACTIVITIES

Section 7.1.3 of this report presents the monitoring, testing and inspection activities. The containment pressure tests and monitoring of its mechanical behaviour by means of the instrumentation system are more specifically described.

On the basis of these data and its experience of reactor assessments, ASN considers that the checks and tests performed by EDF offer satisfactory observation and interpretation of the mechanical behaviour and tightness of the containments. The significant number of containments of the same type and the robust experience of the teams in charge of testing and measurement are valuable assets in the ageing management of these concrete structures.

It should also be noted that monitoring of the condition of the containment outer facings is based on the search for traces of corrosion or cracking, at 5-year intervals. On the occasion of the inspections carried out by ASN, the cleanness of the walls, in particular the domes of the reactor buildings was found to be not always satisfactory, which could affect the durability of these structures. In order to improve this aspect

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[60] *This only concerns the 900 MWe reactors. This does not concern the seals of the various access airlocks which are regularly replaced.*

of containment ageing management, the licensee was asked to make changes to the maintenance documents.

Preventive and corrective measures

For the 900 MWe reactors, EDF states that it has looked for and used resin injection to treat cracks with an opening of more than 0.3 mm, on the outer facings of all the containments, in a campaign which ended in 2010. This is a positive point for ageing management of these structures. The checks performed concerning containment tightness were however unable to prevent or precisely identify the origin of the metal liner corrosion which occurred in 2015 on Bugey reactor 5. The repairs made were nonetheless considered by ASN to be satisfactory. These repairs could however be better prepared in advance in order to improve treatment in the event of the feared ageing phenomenon occurring.

For double-wall containments, the gradual deployment of tightness coatings on the inner surface and the development of repair solutions for the outer surface of the double-wall containments, indicates satisfactory anticipation by the licensee of the future consequences of the ageing of these containments. Ageing management of these coatings is itself ensured by their incorporation into the ageing management programme.

The preventive and corrective measures carried out by EDF lead to management of these containment ageing phenomena that is on the whole satisfactory. These measures are more specifically based on pertinent use of the instrumentation system installed when these structures were built.

7.4.1.4 EXPERIENCE WITH APPLICATION OF THE AGEING MANAGEMENT PROGRAMME

EDF presents four aspects of operating experience feedback from application of its ageing management programmes: greater gradual deformation than was expected in the design of the containments, partial failure of the instrumentation system, corrosion of the metal liner and local concrete swelling phenomena.

7.4.1.5 EXPERIENCE OF REGULATORY OVERSIGHT

In addition to the thematic inspections of ageing management (see section 2.7.1.1), ASN also checks that EDF carries out measures to ensure ageing management of concrete containments during its on-site inspections devoted to other topics such as civil engineering (between 5 and 10 per year), or maintenance (about 10 per year) and its site visits during reactor outages.

In addition, the reactor outage inspections (see section 2.7.3) are also an opportunity for ASN to check maintenance work and the correction of deviations regarding the concrete containments, notably with regard to damage linked to ageing mechanisms.

7.4.2 RESEARCH REACTORS

Ageing management is based on scheduled periodic monitoring (visual inspections, testing) the aim of which is to determine the status of the items participating in the various functions and to analyse how they change between two tests or two inspections, in order to guarantee the durability of the structures for the duration of their service lifetimes.

Containment ageing is mainly monitored during periodic checks and tests. The compliance of the containments is verified during the periodic safety reviews. The nature of the steps taken during these inspections primarily concerns:

- monitoring of containment cracks,
- penetration tightness,

- checking of containment leak rates.

It should be pointed out that the RJH, currently under construction, will offer additional features for managing and monitoring the ageing of its containment. For example, it will be possible to replace the containment's prestressing tendons if necessary. Furthermore, monitoring of deformation and of the anti-seismic bearing pads supporting the nuclear unit will be implemented. Some of these provisions are the subject of specific ASN prescriptions.

ASN considers that the containment ageing management programme for the other research reactors remains limited. This programme should be supplemented with identification of the ageing mechanisms that could affect the containment of research reactors (concrete cracking, corrosion of rebars, tendons, metal components, etc.) on the basis of an understanding of their specific aspects and the results of tests and inspections, as well as data from national and international programmes. The aim is to ensure that the checks and tests used are sufficient and, if necessary, to define additional checks to ensure that the containment is able to perform its functions over the long-term.

In particular, on the basis of operating experience feedback concerning damage observed on the PWRs in operation (gradual behaviour, concrete pathologies and cracking), monitoring of the displacement and deformation of the structural elements contributing to the strength of the structure could be implemented in order to define corrective measures, if necessary, to anticipate the possible development of significant anomalies.

8 PRE-PRESSED CONCRETE PRESSURE VESSELS (AGR)

Not applicable in France.

9 OVERALL ASSESSMENT AND GENERAL CONCLUSIONS

9.1 AGEING MANAGEMENT ISSUES IDENTIFIED BY THE TOPICAL PEER REVIEW

The WENRA reference levels for existing nuclear power reactors indicate the ageing management requirements. They prescribe the existence/implementation of an ageing management programme based on:

- an identification of the ageing mechanisms to which the systems, structure and components (SSCs) are subjected which is as exhaustive as possible;
- monitoring and inspection activities;
- regularly reassessing this programme in the light of the new information available.

Ageing management includes the engineering, operational control and maintenance actions undertaken by the licensee to prevent or maintain within acceptable limits the degradations of the SSCs in its facility in order to ensure the in-service availability of the safety functions.

Ageing management of nuclear facilities therefore applies to a large number of SSCs.

The TPR focuses on the ageing management programme for the installations addressed by the review, with, for France, examples of its application to the electric cables, reactor pressure vessels, buried pipes and reactor containments.

9.2 AGEING MANAGEMENT ISSUES IDENTIFIED BY ASN

To improve the overall consistency of the ageing management procedures engaged by EDF, ASN requested in 2001 that a structured and consistent ageing management programme be set up in preparation for the first of the 3rd ten-yearly outages (VD3) of the 900 MWe plant series reactors, due to start in 2008. In this letter, ASN indicated the bases that should underpin this approach:

- with regard to the identification and handling of sensitive components:
  - updating of the list of SSCs whose ageing can affect reactor safety and govern its duration of operation;
  - defining the parameters associated with the ageing mechanisms identified for these SSCs which, if exceeded, will determine a particular action (repair, replacement, modification, change of environmental or operating conditions);
  - constituting repair and replaceability files;
- with regard to the monitoring actions, a detailed analysis of operating experience feedback allowing the treatment of degradations associated with ageing mechanisms which could not be foreseen due to the complexity of the phenomena;
- with regard to R&D, studies must be carried out on the ageing phenomena and their development kinetics, taking the actual environmental and operating conditions into account.

ASN asked that this programme be implemented, within the framework of the VD3, with a continued operability file being produced for each reactor along with a detailed ageing management programme beyond the VD3 outages.

Within the framework of the third periodic safety reviews, EDF proposed an ageing management approach based on generic ageing analysis sheets (AAS) and detailed ageing analysis reports (DAAR), which are updated periodically, and on an ageing analysis specific to each reactor (UAAR). Today this approach can claim some ten years of application experience feedback.

In 2009, EDF announced its intention to extend the operating life of its NPPs beyond 40 years. In this context, the ageing management approach for the EDF reactors is based on three long-term operational processes:

- the component ageing management process implemented as of VD3 and continued in VD4;
- the in-service inspection and maintenance process which takes account of the hypothesis of continued operation of the reactors until VD4+20 years;
- the process to address the obsolescence of equipment and spare parts.

This approach is currently being examined by ASN to determine whether the measures implemented and/or planned by EDF are sufficient to manage the obsolescence of the SSCs due to ageing and thus maintain the conformity of the 900 MWe reactors beyond VD4. The conclusions of this examination are expected in early 2018**.**

9.3 GOOD PRACTICES

9.3.1 REGULATORY OVERSIGHT

Ageing management in France is regulated by:

1) the provisions relating to the periodic safety review process for all basic nuclear installations, which are written into the Environment Code,
2) the provisions of the Orders, currently being codified, relating to the regulation of nuclear pressure equipment (NPE) and non-nuclear pressure equipment items with regard to their design/construction or in-service monitoring. As of the third ten-yearly outage, the regulations require partial requalification comprising an in-depth inspection of the equipment five years after the ten-yearly outage and after all the subsequent ten-yearly outages.
3) for nuclear power reactors, ASN's requests concerning EDF management of the ageing of its facilities and their continued operation beyond 40 years,
4) for nuclear power reactors, the provisions of ASN-IRSN Guide n° 22 "Pressurised Water Reactor Design" applicable to the search for improvements to be made to existing reactors, for example in the context of the periodic safety reviews; it extends to non-NPE components the provisions of the order of 30 December 2015 relative to nuclear pressure equipment on the need to take equipment ageing into account in their design.

Furthermore, the international standards (IAEA and WENRA) are taken into account in the ageing management regulations.

9.3.2 THE AGEING MANAGEMENT APPROACH

EDF has implemented a **structured and consistent approach** to demonstrate the ageing management of systems structures and components (SSC) potentially affected by ageing. It is based on 4 steps:

- selection of the SSCs potentially susceptible to ageing and whose failure can have an impact on safety,
- establishing and analysing the SSC/ageing mechanism combinations to verify the ageing management in the light of the operating and maintenance provisions in force, along with the reparability and replaceability conditions (materialised by an Ageing Analysis Sheet (AAS);
- defining, if applicable, the additional ageing management actions or studies to carry out (materialised by a component Detailed Ageing Analysis Report (DAAR)),

- drawing up an ageing analysis specific to each reactor, called an UAAR, using the generic AASs and the component DAARs for each reactor reaching its third ten-yearly outage (VD3) and the subsequent ten-yearly outages.

In the context of the requests to extend the operating life of the NPPs, EDF proposes continuing this approach for the fourth ten-yearly outages (VD4). **This approach will be extended to all the SSCs that are important for the management not only of radiological risks but also of conventional risks.**

ASN underlines that the EDF ageing management programme meets the requirements of the international standards. It takes French and international operating experience feedback appropriately into account, which has enabled preventive and corrective actions to be implemented. It is moreover accompanied by a consequential R&D programme. EDF has set up a **research and development (R&D) programme** to support its ageing management process in order to progress and build on knowledge of ageing mechanisms and of the properties of the materials after 60 years of operation. This programme, which is used to advance engineering practices and the means of monitoring and inspection, also contributes to the development of processes to repair or mitigate the consequences of ageing, as well as to the appraisal of removed equipment items.

Operability criteria, corresponding to the maximum acceptable values of the consequences of the degradation mechanism in view of safety (for example, for the corrosion degradation mechanism, a maximum permissible reduction in thickness), are essential decision aids for ruling on continued operation beyond the fourth ten-yearly outage.

The composition of the French fleet in operation is also an asset for monitoring ageing: it provides significant feedback on the behaviour of reactor components from the same series or from different series.

Electric cables

EDF conducts appraisals on **cables samples** taken on site which have enhanced the understanding of the ageing mechanisms and carries out the predictive methods resulting from R&D enabling the service life of cables to be estimated. These appraisals have identified the cables that require in-service monitoring (cables subjected to severe environmental or operating stresses). Furthermore, considered along with the results of the predictive service life studies, they provide a high level of confidence that cable operability will be maintained for the next 10 years.

Reactor pressure vessel

EDF implements numerous inspections:

- on the parts of the RPV identified as being the most susceptible to the identified modes of degradation,
- in **zones that do not display any particular susceptibility to the identified ageing mechanisms** (particularly applying the principle of defence in depth, the core zone undergoes a complete automated ultrasonic inspection at each ten-yearly outage),
- in certain zones (as part of the Complementary Investigations Programmes) to confirm the absence of modes of degradation that might not have been identified.

Furthermore, the irradiation monitoring programme ("PSI" programme) allows the behaviour of the materials in the core zone to be monitored up to irradiation levels equivalent to at least 60 years of operation, given the number of irradiation capsules available in the fleet as a whole. The gradual acquisition of the PSI programme results enabled the irradiation embrittlement model to be updated in 2007. This model was revised on the basis of the results of the irradiation capsules of the 900 MWe series (more than 350 measuring points), which covers the widest range of fluence and chemical compositions for the RPVs of the EDF fleet. The revised model is more representative of the behaviour of the material over long periods of time than the initial model, which was based on a smaller number of data in the irradiated state, obtained from tests in experimental reactors.

In addition, EDF is planning to insert Hafnium RCCAs in some assemblies of the core periphery, which have the effect of reducing the neutron flux received by the vessel.

Finally, the approach used to justify the vessel integrity is deterministic, which implies that conservatisms are taken at each stage and gives these studies a very high level of confidence.

Containments

The instrumentation system on the containment structures comprises numerous instruments to measure deformation and displacement, placed in all the standard and singular areas of the wall. This system, reinforced on the containment of the first reactor of each site, enables the mechanical behaviour and tightness of the containments to be observed, and early assessments to be made, throughout their lifetime. The design uniformity of the containments of the fleet in service is an advantage for their monitoring as it allows comparisons between containments on a given site or on different sites, or of the same design but of different ages.

9.4 AREAS FOR IMPROVEMENT

With a view to continued reactor operation beyond 40 years, EDF has undertaken an ageing management programme for concealed pipes in addition to its monitoring provisions. Under this programme EDF performs inspections on the Tricastin, Fessenheim and Bugey sites, with the aim of defining a generic programme of verifications and being able to decide at VD4 whether the buried pipes are fit for continued service or need to be renovated. The examination is in progress and the conclusions are expected in 2018.

Furthermore, ASN considers that the specific aspects of the site and of each reactor could be better taken into account in the local ageing management programme (PLMV) and the UAAR.

9.5 ACTIONS RESULTING FROM THIS ASSESSMENT

This review highlights the fact that the monitoring of **research reactor** ageing is currently based on maintenance programmes, inspections and periodic tests. ASN considers that the research reactor licensees need to adopt a more formalised approach to ageing management. More specifically, ASN considers that the research reactor licensees must apply an approach that allows the adequacy of the inspections and tests implemented to be ascertained. They must also define additional verifications to ascertain their fitness to fulfil their functions with regard to the ageing mechanisms that could affect the EIPs (elements important for protection).

9.6 CONCLUSION

ASN emphasises that since 2001 EDF has developed an ageing management programme that satisfies the major issues of nuclear safety and radiation protection. This programme has moreover been stepped up in the context of continued operation beyond 40 years. ASN will give its opinion on this programme in its generic position statement concerning the VD4 900 outages.

With regard to the research reactors, while noting the specific nature of each installation, ASN considers that the ageing management programmes of the Cabri and RHF reactors must be supplemented within the framework of the periodic safety reviews.

10 APPENDICES

10.1 GLOSSARY

AGR	Advanced Gas-cooled Reactor
AMP	Ageing Management programme
AMR	Ageing Managemet Review
APRP	Loss of Coolant Accident Primary Refrigerant
ASN	French Nuclear Safety Authority
CEA	The French Alternative Energies and Atomic Energy Commission
CEIDRE	Centre of Expertise and Inspection in the Realization and Operation Domains
CNEPE	National Electricity Generation Equipment Centre
CNPE	Nuclear Power Plant
CPP	Main primary system
CSP	Main secondary systems
CST	Technical specifications book
DDS	The inventory of design transients
DIPDE	Deconstruction and Environment Park Engineering Division within EDF
DPN	Nuclear Power Generation Division within EDF
DRR	Regulation Reference File
EAS	Containment Spray System
EDE	Annulus ventilation system
EDF	Electricité de France
EIP	Element Important for the Protection of interests
ENSREG	European Nuclear Safety Regulators Group
NPE	Nuclear pressure equipment
FFS	Fitness for Service
FMGPI	Increasing Equipment Reliability and Managing Industrial Assets
GPESPN	The ASN Advisory Committee for nuclear pressure equipment
GPR	The ASN Advisory Committee for reactors
IAEA	International Atomic Energy Agency
IGALL	International Generic Ageing Lessons Learned

ILL	Institut Laue-Langevin
INPO	Institute of Nuclear Power Operations
IRSN	French Institute for Radiation Protection and Nuclear Safety
LTO	Long term Operation
MQCA	Qualified Materials for Accident Conditions
NSQ	Qualification Summary Report
NSQP	Gradual qualification strategy report
NUSSC	Nuclear Safety Standards Commitee
OEF	Operational experience feedback
OPDEM	Operations in preparation for decommissioning
PBMP	Basic Preventive Maintenance Programme
PDV	« Lifetime » Project
PIC	Supplementary Investigations Programme
PLMV	Local Ageing Management Plan
PSI	Programme monitoring radiation effects
RCRP	Periodic safety review conclusion reports
R&D	Research and Development
RGE	General operating rules
RHWG	Reactor Harmonisation Working Group
RHF	High-Flux reactor
RIS	Emergency core cooling system
RJH	Jules Horowitz reactor
RPN	Neutron flux measuring channels
SEPTEN	Design engineering centres within EDF
SSC	Systems, Structures and Components
TAM	Equipment access hatch
TOFD	Time Of Flight Diffraction
TLAA	Time Limited Ageing Analyses
UNIE	Operations engineering unit
UTO	Operational Technical Unit within EDF
VD	Ten-yearly outage inspection

10.2 AGEING ANALYSIS SHEETS (AAS) SPECIMEN

AGEING ANALYSIS SHEET							Sheet N°		
							Revision:		
							Date:		
							Knowledge base reference		
DIN	Written by		Unit		Checked by			Unit	
DPN	Written by		Unit		Checked by			Unit	
Plant series / Reactor(s)									
Component / structure									
Element / zone									
Mechanism	Acronym			Mechanism					

Tick if there is a change of methodology

Changes introduced at the last three revisions Tick here if there has been a change in the upstream data

Rev.	date	Reason for revision	Changes introduced		

Analysis grid	Reply / Justification / Comment	References
Safety classification		
Description of the mechanism considered / associated damage		
Regulatory design service life hypothesis		

Analysis grid	Reply / Justification / Comment	References
Confirmed mechanism / OEF / confirmed damage		
Adaptation or adaptability of the standard operational management or maintenance provisions (including consideration of obsolescence)		
Repair difficulties (including those linked to obsolescence)		
Replacement difficulties (including those linked to obsolescence)		
Status	Justification	
Follow-up decision		

OPERATION BEYOND 4TH TEN-YEARLY OUTAGE (VD4)		
Analysis grid	**Reply / Justification / Comment**	**References**
Study elements supporting continued operation		
Operating period covered		
Adaptation or adaptability of the standard operational management or maintenance provisions (including consideration of obsolescence)		
Status	Justification	
Follow-up decision		

10.3 GROUPS OF ELECTRIC CABLE CONSIDERED FOR AGEING MANAGEMENT

All the electric cables relating to the production facility (classified or non-classified) are covered by an ageing management programme. There are 4 main cable groups:

- Medium-voltage (MV) power cables;
- Low-voltage (LV) power cables;
- Measuring cables (LV),
- Instrumentation and control cables (LV).

They comprise the following main components:

- A copper or aluminium core (aluminium for cross-sectional areas (c.s.a.) of 50 mm² and more),
- An extruded insulating casing made from a mixture of organic materials,
- An intermediate sheath (possibly) extruded from a mixture of organic materials,
- Protective shielding against electromagnetic interference, consisting of copper braid in the measuring cables,
- Shielding for the MV cables,
- Metallic armour providing mechanical protection,
- An outer sheath extruded from a mixture of organic materials.

<u>Medium-voltage (MV) power cables;</u>

The MV power cables interconnect the various electrical auxiliaries of the 6.6 kV (900 MWe, 1300 MWe and N4 plant series) and 10 kV (EPR series) networks.

There are three types of cable in service:

- Single-core and three-core radial field cables with rated voltages of 6, 10 (12) kV, for the 900 MWe, 1300 MWe and N4 plant series,
- Three-core non-radial-field cables (called "belted" cables, the metal shielding is common to the three phases), with rated voltage of 6 / 6 (7.2) kV, for the 900 MWe, 1300 MWe and N4 plant series.
- Single-core and three-core radial-field cables with rated voltages of 8.7 / 15 (17.5) kV, for the EPR plant series,

Note: the rated voltage U_0 / U (Um) of a cable is defined as follows:

- $U_{0:}$ nominal voltage of the cable at 50 Hz between each conductor and ground,
- U: nominal voltage of the cable at 50 Hz between two conductors,
- Um: maximum voltage at 50 Hz between two conductors that can be withstood at any time under normal operating conditions.

<u>Low-voltage (LV) power cables;</u>

The LV power cables ensure the connections between the various electrical auxiliaries of the AC and DC LV networks.

These cables are generally single-core or three-core (as neutral is not usually distributed on the three-phase networks). They comprise insulated conductors (c.s.a. greater than 2.5mm²), possibly a filler, an armour-clad sheath, a metallic armour layer and an outer protective sheath.

These cables usually have a rated voltage of 0.6 / 1 (1.2) kV, whatever the network voltage level.

Measuring cables

The measuring cables are used to transmit the analogue measurements taken by the sensors.

The currents are low (a few milliamperes if not microamperes) with voltage levels of just a few tens of volts. The cross-sectional area of the conductor is usually 1 or 1.5 m².

These cables are generally shielded and the conductors are grouped in twisted pairs. They can also be coaxial cables. In this case they comprise an insulated core enclosed in braid which acts as the shielding and return conductor.

These cables have a rated voltage of 0.3 / 0.5 (0.6) kV or 0.6 / 1 (1.2) kV.

Instrumentation and control cables

The instrumentation and control cables transmit on-off signals between the various LV components.

These cables usually comprise between 2 and 15 conductors of 1 to 1.5 mm² csa. They have a rated voltage of 0.3 / 0.5 (0.6) kV or 0.6 / 1 (1.2) kV.

Different types of polymer materials are used to make the insulants and sheaths of electric cables. These materials have a complex chemical formulation, integrating a base polymer (which commonly gives its name to the material) and numerous additives designed to improve its mechanical and electrical properties. The following polymer materials are used in the EDF fleet:

- Cable with ethylene-propylene rubber (EPR) based insulant and chlorosulfonated polyethylene (CSPE) based sheath – trade name "Hypalon". These cables have a qualification level K1.
- Cable with Polyvinyl chloride (PVC) based insulant and outer sheath. These cables have a qualification level K3/NC, and also K2[61] for certain cable manufacturers.
- Cable with chemically cross-linked polyvinyl chloride (XLPE) based insulant and polyvinyl chloride (PVC) based outer sheath. These cables are single-core MW cables. Their qualification level is K3/NC.
- Cables with insulation and outer sheath based on "Halogen-Free" (HF) material comprising mixtures of ethylene-vinyl acetate (EVA) and cross-linked polyethylene (XLPE) or EPR polymers. These cables are present as of the N4 plant series in the reactor building and on the EPR series. These cables display improved reaction-to-fire performance.

The table below summarises the main materials used for the insulants and sheaths according to the plant series.

[61] K2 : inside reactor building, not required under degraded ambient conditions

	900 MWe series	1300 MWe series	N4 series	EPR series
K1 - MV cables	EPR/Hypalon[62]		HF	
K1 - LV cables	EPR/Hypalon		HF	
K3 - MV cables	PVC/PVC	PVC/PVC (three-core cables) XLPE/PVC (single-core cables)	XLPE/PVC	HF
K3 - LV cables	PVC/PVC			HF

Table 14 – Insulation and sheath materials used on the different plant series

[62] In designation formats such as EPR/Hypalon, the first term (EPR) designates the insulant and the second term ((Hypalon) designates the sheath.

10.4 GENERAL DRAWINGS OF REACTOR PRESSURE VESSELS

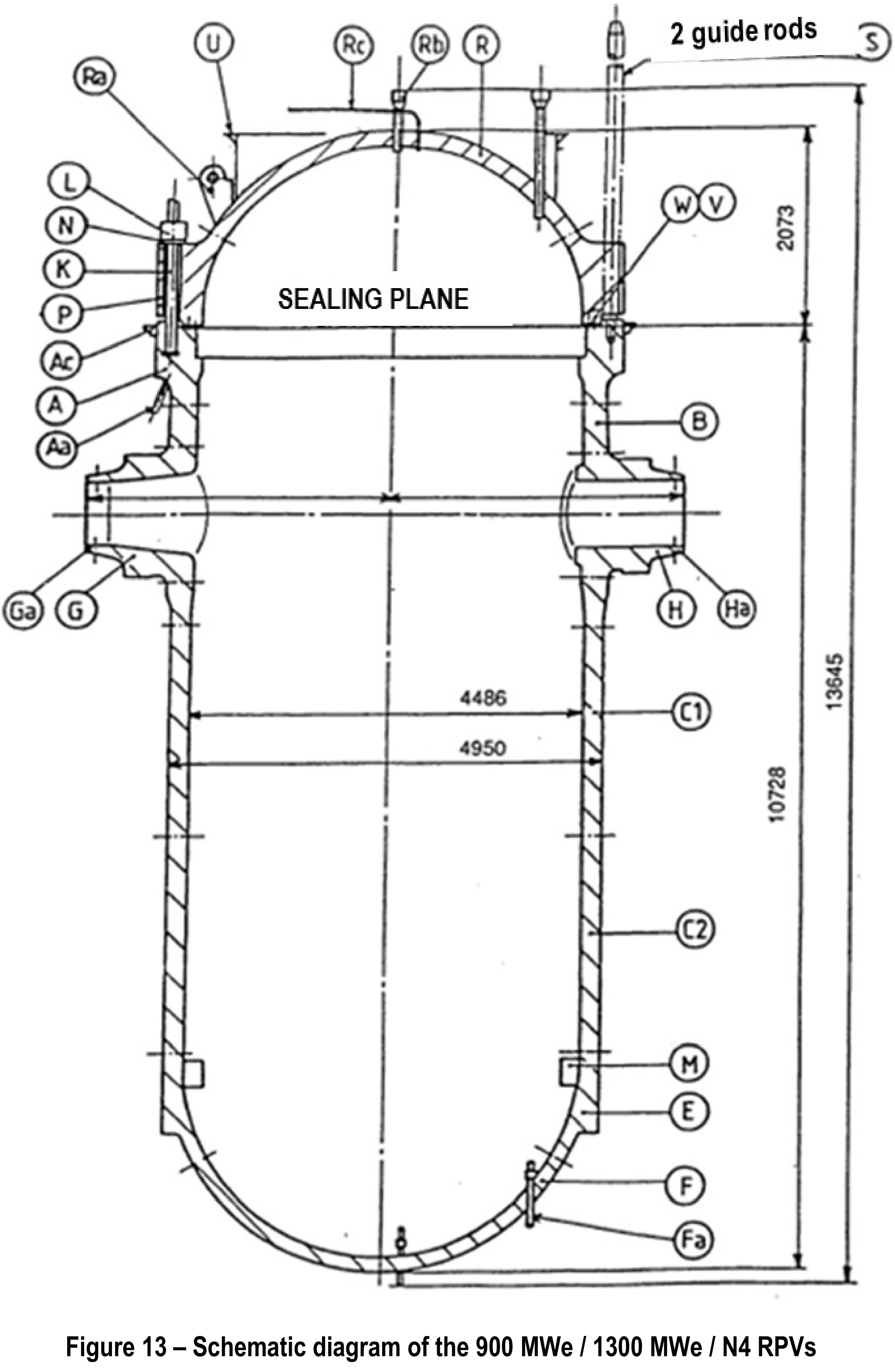


Figure 13 – Schematic diagram of the 900 MWe / 1300 MWe / N4 RPVs

Naming of the principal parts in accordance with the parts list for a typical N4 RPVS (these identifiers can vary from one plant series to another, but the main constituents are similar):

RPV body:

- RPV flange: item A
- Nozzle support ring: item B
- Core shells: items C1 and C2 (plus one shell item C3 on Fessenheim 1)
- Transition zone: item E,
- Lower hemispherical head or dome: item F,
- Inlet nozzles and their safe end: items G and Ga,
- Outlet nozzles and their safe end: items H and Ha,
- Bottom-mounted instrumentation penetrations: item Fa,
- Radial guides: item M,
- Seal leak monitoring tube: item Aa,
- Sealing flange (for connection between RPV sealing ring and pool, item L, not shown): item Ac,

RPV closure head:

- RPV closure head flange: item P,
- Upper hemispherical head: item R,
- Control mechanism adapters: item Rb,
- Vent tube: item Rc,
- RPV lifting lugs: item Ra,

Other parts:

- Closing part (studs, nuts and washers): items K, L and N,
- Body / closure head seals: items V and W,
- RPV guide rods for opening / closing operations: item S,

Identification of welds:

The welds are identified according to the item numbers of the components they join together, for example:

- Weld C1/C2 for the weld joining the core shells C1 and C2,
- Weld B/H1 for the weld joining the outlet nozzle H1 to the nozzle support ring B.

10.5 MAIN RPV CONSTITUENT MATERIALS

Plant series	900 MWe CP0 / CPY	1300 MWe P4 / P'4	N4	1650 MWe EPR FA3
RPV body				
RPV flange	Low alloy steel			Low alloy steel
Nozzle support ring	Low alloy steel			
Inlet / outlet nozzles	Low alloy steel			
Inlet / outlet nozzle safe end	Austenitic stainless steel			
Core shells	Low alloy steel			
Transition ring,	Low alloy steel			
Lower dome	Low alloy steel			
Radial guides	Nickel-based alloy			
BMI penetrations	Nickel-based alloy			Not applicable
Inter-seal leak monitoring tube	Austenitic stainless steel			
RPV closure head				
RPV closure head flange:	Low alloy steel			
Upper dome	Low alloy steel			
RPV closure head adapters (CRDM[63]/ instrumentation)	Nickel-based alloy			
Adapter flanges (CRDM / instrumentation)	Austenitic stainless steel			
Vent branch connection	Nickel-based alloy			
Vent tube	Austenitic stainless steel			
Closing parts				
Studs, nuts, washers	High tensile steel			

[63] CRDM: Control Rod Drive Mechanism

10.6 PRINCIPAL DIMENSIONAL AND FUNCTIONAL CHARACTERISTICS

	900 MWe	1300 MWe	N4	EPR
Inside diameter on cladding in core zone	≈ 4000 mm	≈ 4400 mm	≈ 4500 mm	≈ 4900 mm
Thickness of main parts in low alloy steel	130 to 200 mm	140 to 220 mm	145 to 225 mm	145 to 400mm
RPV body empty weight	≈ 260 t	≈ 330 t	≈ 350 t	≈ 400 t
Number of closure head penetrations for CRDM and core instrumentation	65 to 77			106
Number of BMI penetrations	50 to 60			0
Nominal operating pressure	154 bars rms			
Design-basis pressure	171 bars rms			175 bars rms
Design-basis temperature	343°C			351°C
Cold leg temperature in nominal operation	286°C	289°C	292°C	290°C
Hot leg temperature in nominal operation	325°C	324°C	329°C	330°C

10.7 GEOMETRICAL CHARACTERISTICS OF THE EDF REACTOR FLEET CONTAINMENTS

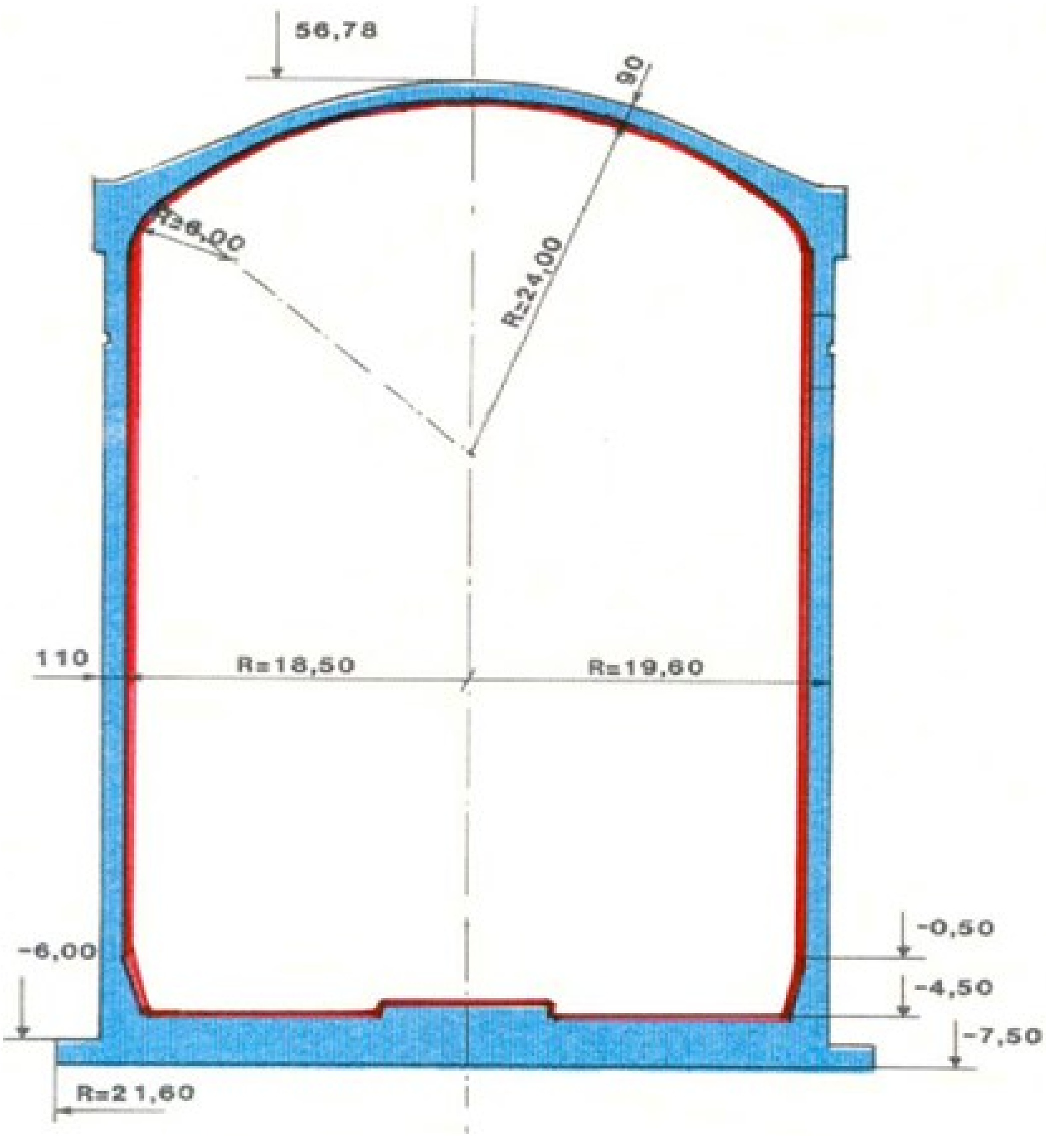


Figure 14 – Diagram of a single-wall containment with liner (CP0, CPY plant series)

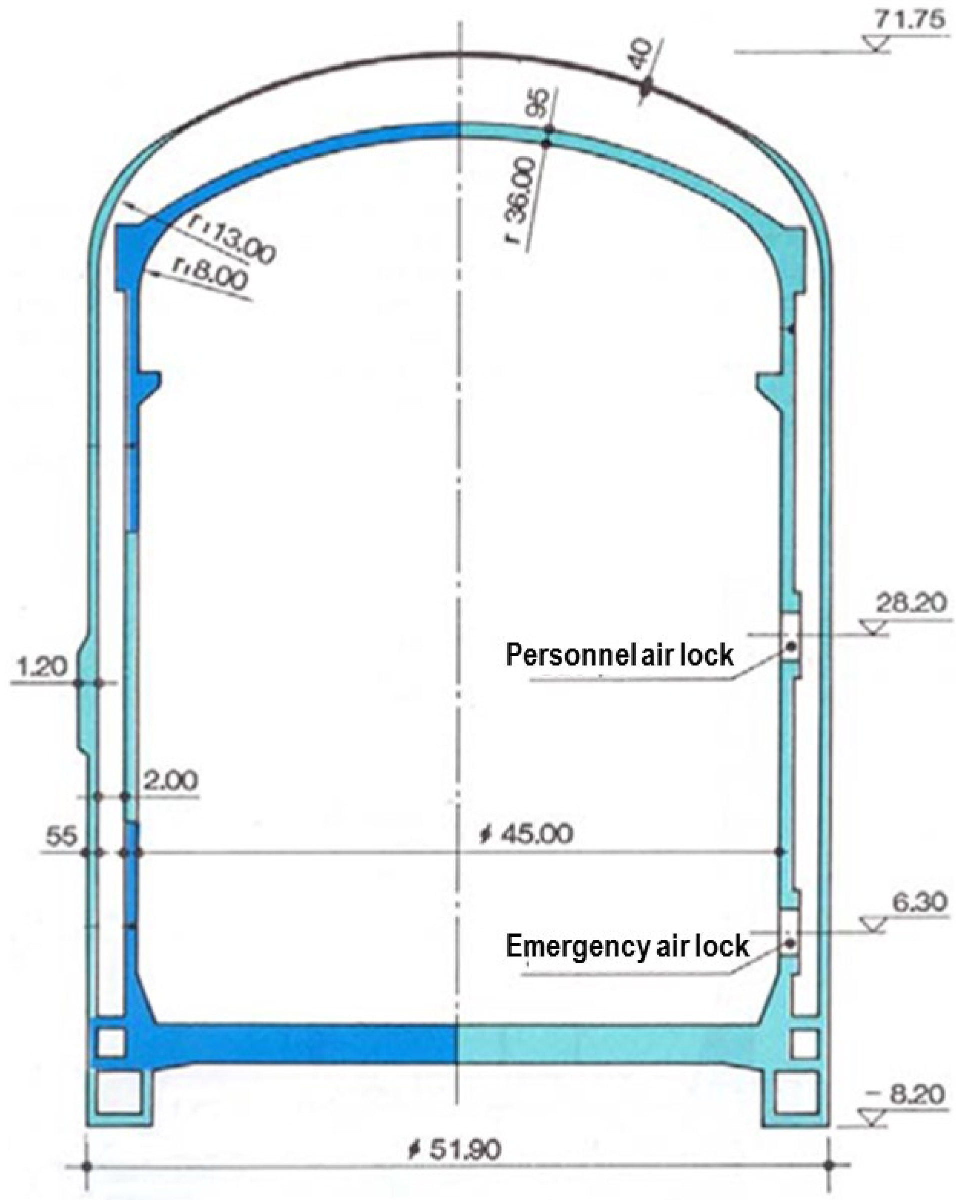


Figure 15 – Diagram of a double-wall containment without liner (P4, P'4, N4 plant series)

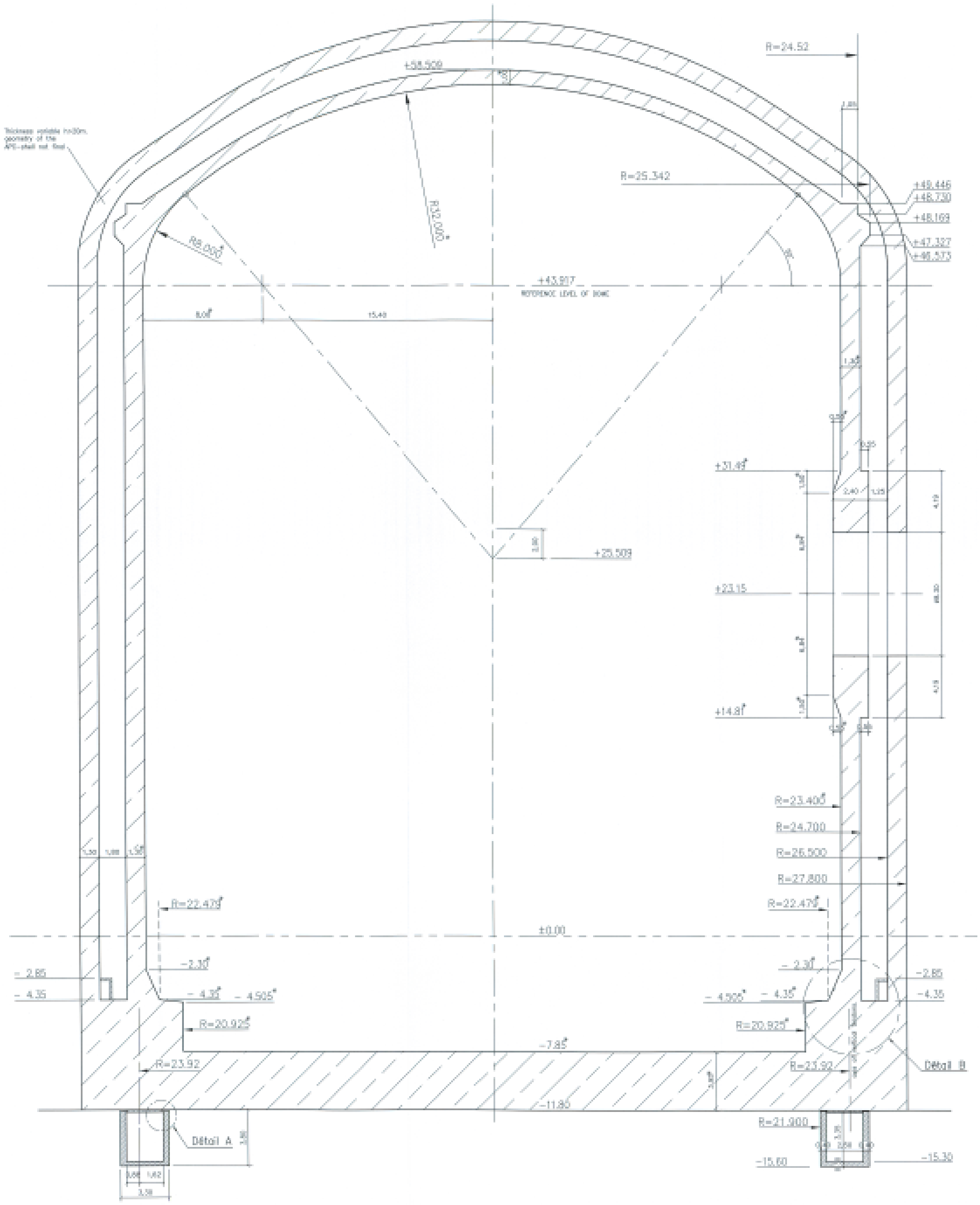


Figure 16 – Diagram of the EPR containment (double wall with liner)

10.8 DESCRIPTION OF THE ORPHEE RESEARCH REACTOR

ORPHEE is a research reactor operated by the CEA and situated in the CEA Saclay research centre. It is intended essentially to provide neutron beams for the purpose of fundamental research. The reactor is the work instrument of the Léon Brillouin Laboratory (LLB), which manages the experiments using the neutrons produced.

View of the ORPHÉE reactor

ORPHEE is a pool-type reactor with a nominal power of 14 MWth and a maximum thermal flux of 3.10^{14} $n.cm^{-2}s^{-1}$. It is used by researchers working in diverse disciplines (chemistry, metallurgy, physics, etc.) to carry out their experiments. It is also used for neutron radiography, analysis by activation and the irradiation of diverse samples. The reactor diverged for the first time in December 1980.

The reactor has 9 horizontal channels (or thimbles) (supplying 20 neutron beams) and 9 vertical channels (or thimbles) with 4 pneumatic channels for analysis by activation, 5 holes for irradiation in the pool).

The Orphée reactor pile block consists of a structure comprising several fluid compartments with different thermodynamic characteristics. The parts of the pile block containing water are at a pressure of 2.25 bar and temperature of 70°C at the most; the smaller parts containing gas are at a pressure of 7 bar and at the most and a temperature of up to 840°C. The list of component data figures in appendix 11.9.

1. Core
2. Heavy water vessel
3. Transfer canal
4. Primary system
5. Heat source
6. Heat sink
7. Pool
8. Heat exchanger
9. Pumps
10. Neutron guide
11. Spectrometer

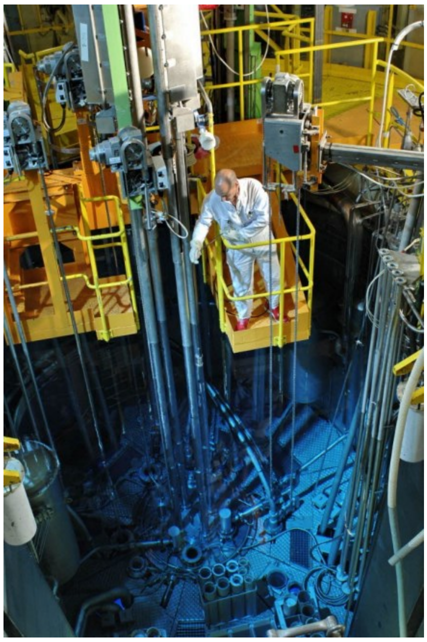
Figure 17 – Simplified cross-sectional view of the ORPHEE reactor

The reactor core, situated within a heavy water vessel, is extremely compact. This heavy water vessel is itself immersed in a pool filled with demineralised light water. This provides protection against radiation and facilitates handling operations above the pool. The reactor is also equipped with three local moderators: one heat source and two heat sinks, used to provide neutrons of varying energy levels. 26 experimental areas are situated around the reactor, either in the reactor building or in the guide area.

A transfer canal situated beside reactor pool is dedicated to the storage of the cooled spent fuel assemblies, beryllium assemblies or spent fuel or waste shipping casks.

The transfer canal and the reactor are separated by a removable gate.

Final shutdown of the reactor is scheduled for the end of 2019. The final shutdown notification from the CEA is therefore expected before the end of 2017.



View of the reactor pool, above the pile block

10.9 LIST OF NUCLEAR PRESSURE COMPONENTS OF THE ORPHEE PILE BLOCK

System name	Equipment name	PS (bars rel.)	TS (°C)	Fluids contained
Pile block				
ED	Upper tube sheet core	2.25	70	Heavy water
ED	Core containment structure		70	Heavy water
EL	Light water vessel	0.95	60	Light water
SC	Internal compartment	3	840	Gas (He)
SC	Gas gap compartment	7	400	Gas (He/N2)
Experimental devices				
Thimble	DdG 1T	0.2		Gas (He)
Thimble	DdG 2T	0.2		Gas (He)
Thimble	DdG 3T	0.2		Gas (He)
Thimble	DdG 4T	0.2		Gas (He)
Thimble	DdG 5C	0.2		Gas (He)
Thimble	DdG 6T	0.2		Gas (He)
Thimble	DdG 7C	0.2		Gas (He)
Thimble	DdG 8F	0.2		Gas (He)
Thimble	DdG 9F	0.2		Gas (He)
Heat sink thimble	Chaussette SF1	<0		Gas (He/N2)
Heat sink thimble	Chaussette SF1	<0		Gas (He/N2)
REA	Tube I1	0		Heavy water
REA	Tube I2	0		Heavy water
REA	Tube R1	0		Heavy water
REA	Tube R2	0		Heavy water
REA	Tube R3	0		Heavy water
Pneumatic channels	Pneumatic tube P1	0		Gas (air)
Pneumatic channels	Pneumatic tube P1	0		Gas (air)
Pneumatic channels	Pneumatic tube P1	0		Gas (air)
Pneumatic channels	Pneumatic tube P1	0		Gas (air)

10.10 DESCRIPTION OF THE CABRI RESEARCH REACTOR

The Cabri reactor (BNI 24), created on 27 May 1964, was designed for conducting experimental programmes aiming to achieve a better understanding of the behaviour of nuclear fuel in the event of a reactivity accident. The reactor is operated by CEA. The facility underwent substantial modifications between 2006 and 20154 so that it could run new research programmes. The sodium loop of the reactor was replaced by a pressurised water loop. This new arrangement enables the effects of an accident situation ("RIA" excursions under thermo-hydraulic conditions that are representative of pressurised water reactors) to be reproduced on the nuclear fuel rods. These tests are determined in the context of international research programmes on safety for reactors of this type. The CABRI reactor allows reactivity transients to be created on a fuel rod placed in its centre in an experimental arrangement.

View of the CABRI facility

The facility comprises several buildings, and in particular the building housing the reactor hall which contains the pool and its driver core, the auxiliary fuel storage tank of the driver tank and the test device storage tank.

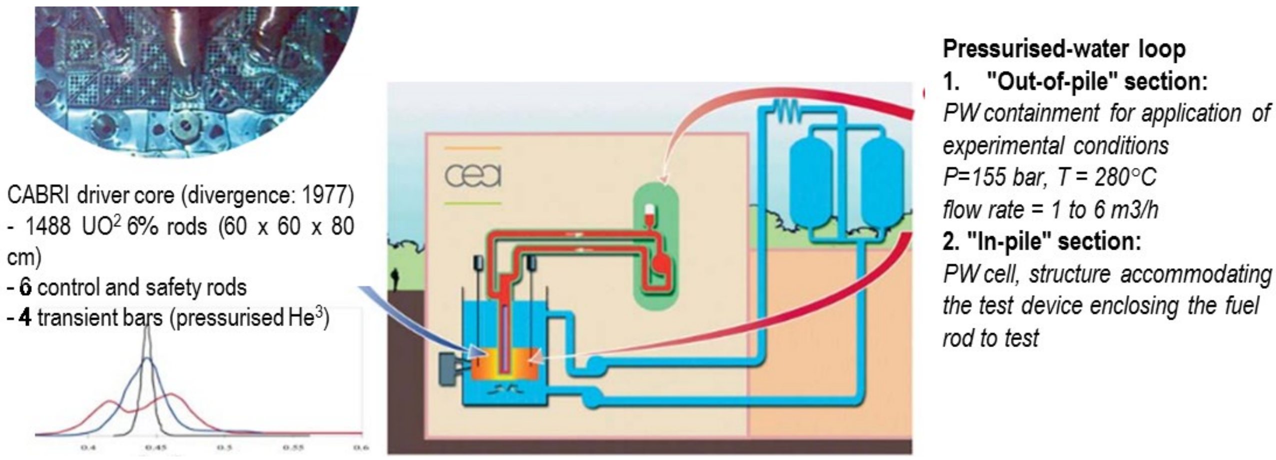


Figure 18 – Schematic diagram of the CABRI reactor

The reactor comprises:

- a driver core with maximum power of 25 MW in steady state operating conditions,
- a test loop containing the fuel rod to study,
- a system of transient bars equipped with pneumatic valves enabling a neutron-absorbing gas to be released in order to perform the reactivity injection.

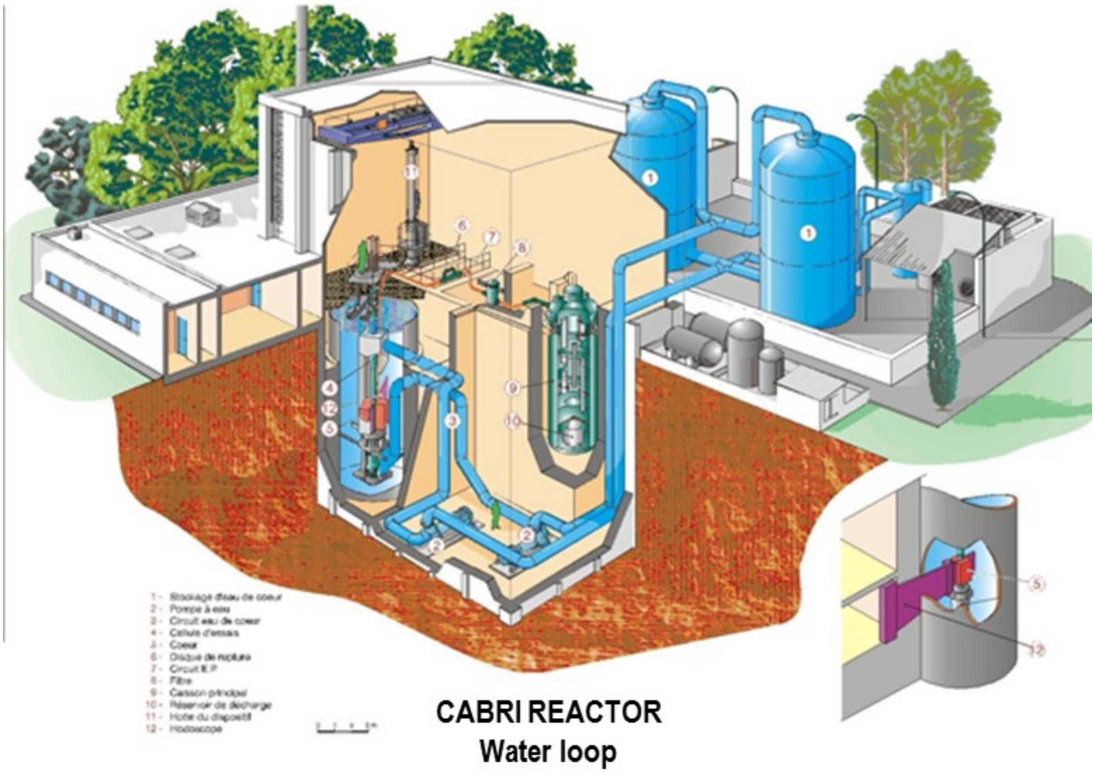


Figure 19 – Simplified Cross-sectional view of the CABRI reactor

10.11 DESCRIPTION OF THE JHR RESEARCH REACTOR

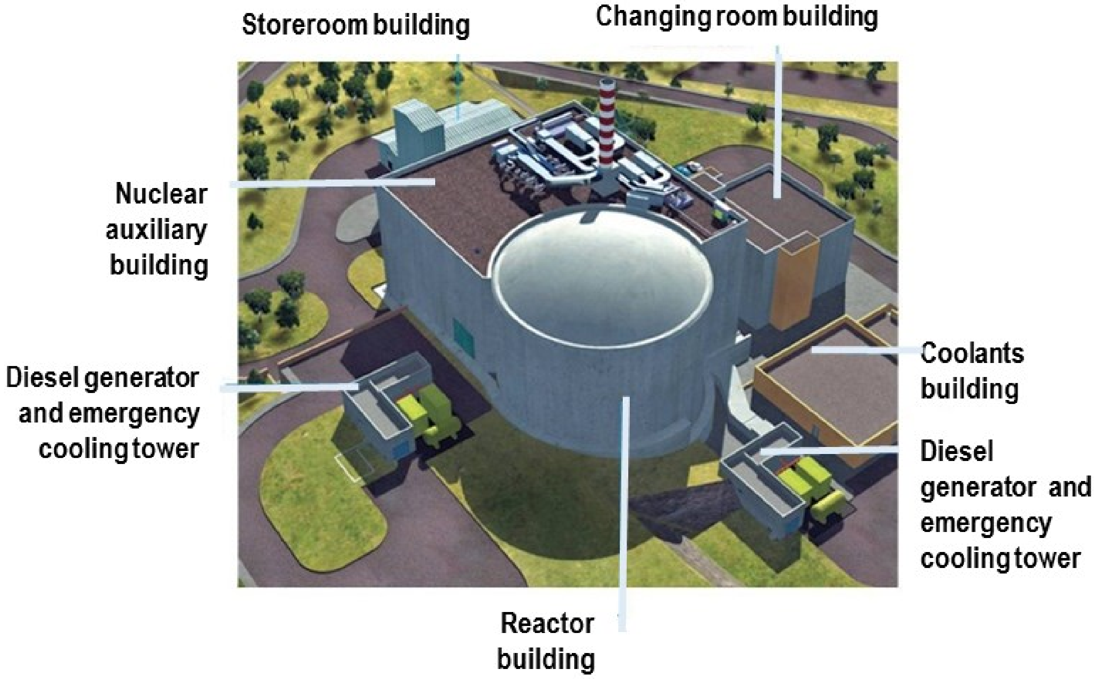


Figure 20 – Project view of the JHR after construction (projected state)

The Jules Horowitz Reactor (JHR), basic nuclear installation No. 172, is under construction on the Cadarache site situated in the municipality of Saint-Paul-lez-Durance (Bouches du Rhône département). Its creation authorisation decree was published on 12 October 2009.

Figure 21 – State of construction in May 2017

The JHR is a reactor that will allow high neutron flux irradiations to be carried out in order in particular:

- to perform experiments with the aim of improving or qualifying the materials and fuels of current and future reactors,

- produce a significant quantity of radioisotopes for medical use, thereby responding to public health issues.

Consequently, locations (within the core or on the periphery, in the reflector) shall be provided to allow the introduction of experimental devices and devices for producing radioisotopes for medical uses.

The JHR shall comprise two main buildings (a reactor building and a nuclear auxiliary building) and a series of support buildings constituting BNI No. 172.

Its first divergence is planned for September 2021.

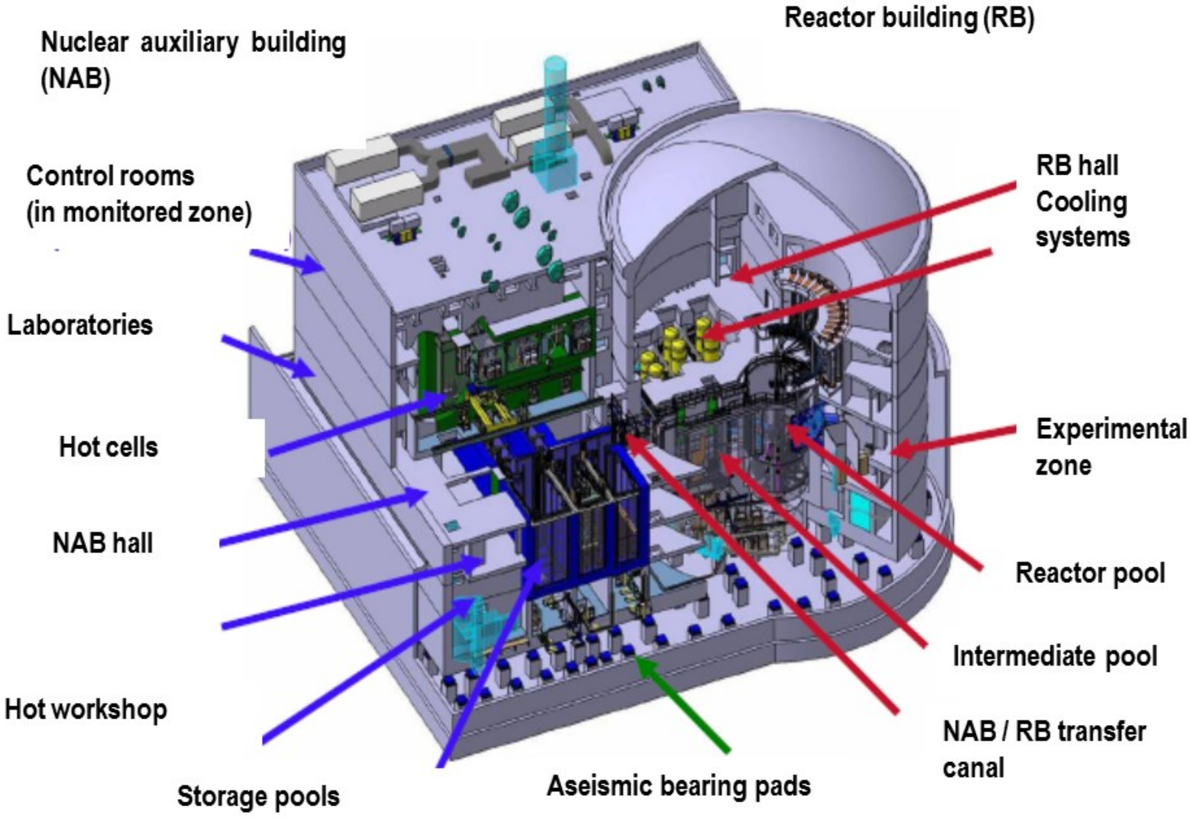


Figure 22 – Diagram of the JHR

The JHR is a pool-type reactor, moderated and cooled by water, and its nominal nuclear power is limited to 100 MW.

The reactor core in the reference configuration comprises 34 cylindrical fuel elements of about 10 cm diameter and 60 cm fissile height. The centre of each fuel element can receive either an experimental device, or a control absorber or an aluminium mandrel.

The fuel elements have a fuel core of enriched uranium made up of U_3Si_2 particles dispersed in an aluminium matrix, and an aluminium alloy cladding, type AlFeNi. The possibility of using a UMo-Al fuel (with a uranium-molybdenum core dispersed in an aluminium matrix) is also considered.

The core is contained in a containment structure connected to the primary system, immersed in a pool with the upper section blanked off by a removable cover supporting the experimental devices irradiated in the core. A beryllium reflector situated on the edge of the containment structure can also accommodate experimental devices.

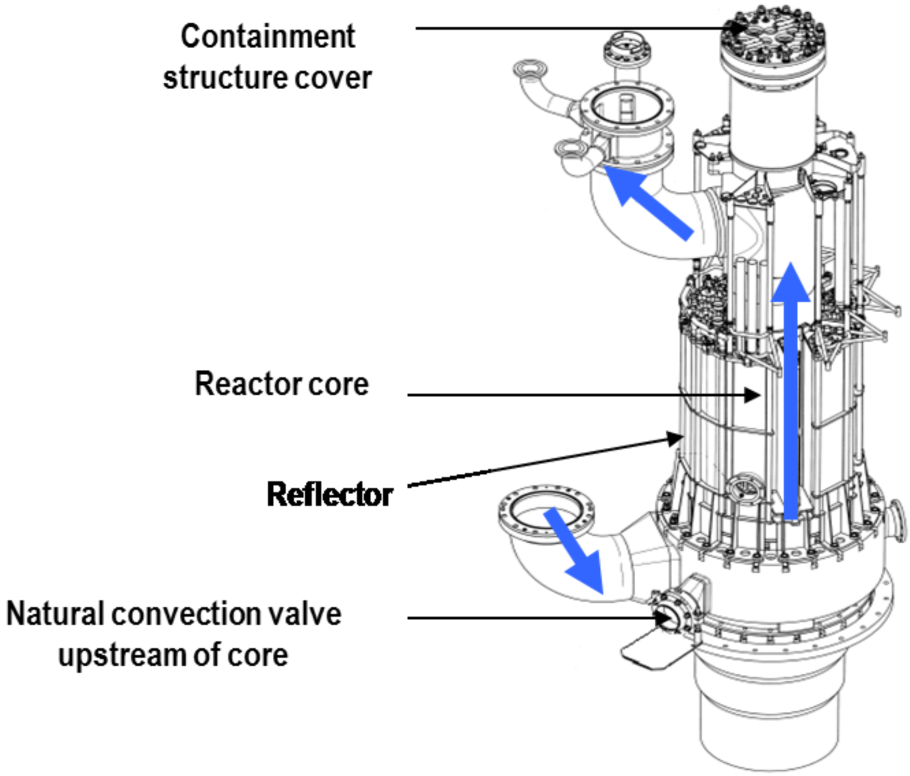


Figure 23 – Diagram of the JHR pile block

The cooling water will circulate:

- from bottom to top in the core,
- from top to bottom in the reflector.

10.12 DESCRIPTION OF THE HFR RESEARCH REACTOR

The ILL's High Flux Reactor (HFR) provides an intense source of thermal neutrons for experiments. The first divergence occurred on 31 August 1971.

ILL and HFR reactor view

The HFR operates continuously over 50-day cycles. Its core is made up of a unique heavy water-cooled, highly enriched uranium fuel element and produces the world's most intense neutron flux of 1.5 x 10^{15} neutrons per second per cm^2. The 58 MW thermal power is discharged by a secondary circuit fed by Drac water (located close to the installation).

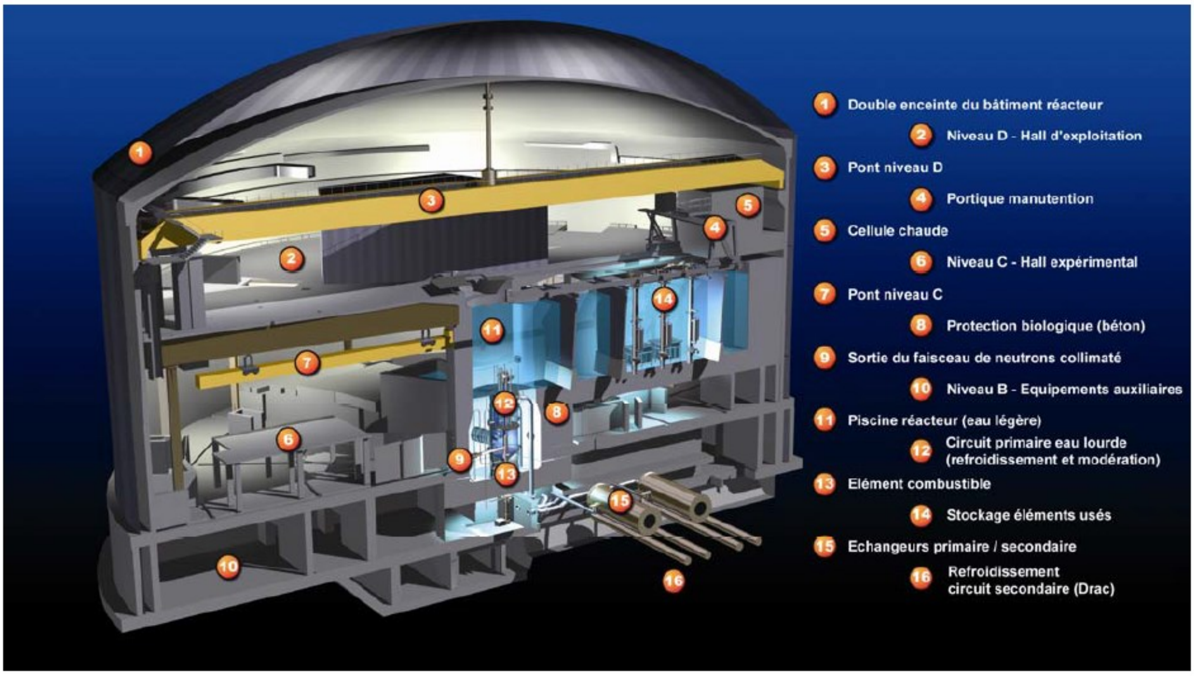


Figure 24 – Simplified cross-section view of the HFR reactor

The heavy water vessel containing the core is located in a pool filled with demineralized water that provides protection against neutron and gamma radiation produced by the core. The reactor is operated using a neutron-absorbing rod that is gradually extracted as uranium is consumed. In addition, it has 5 neutron-absorbing rods, whose function is the shutdown of the reactor.

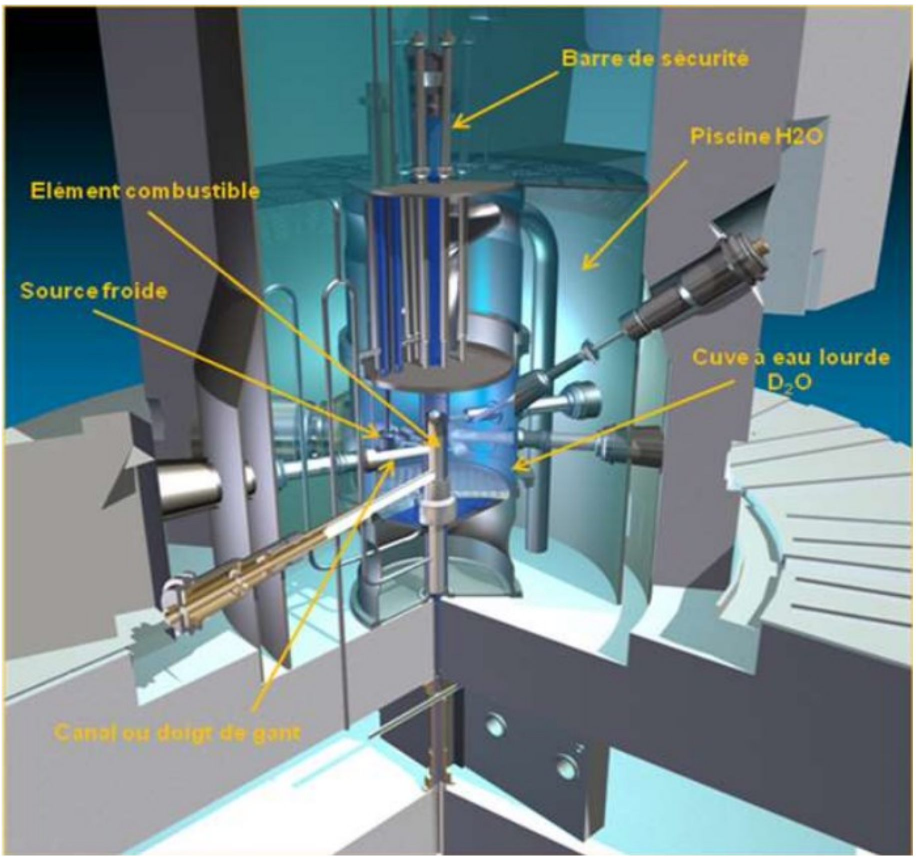


Figure 25 – Simplified view of the battery unit and reactor pool

The neutrons produced in the reactor by the fission reaction have very high energy (speed: 20,000 km/s). They are slowed down by heavy water in order to generate new fissions for maintaining the chain reaction (thermal neutrons with a speed of 2.2 km/s) and to support the scientists' experimental devices.

Three devices located in close distance to the core are neutron sources: they also produce hot neutrons (10 km/s) and cold and ultra-cold neutrons (700 m/s and 10 m/s).

HFR battery pack during charging

The neutrons are then collected inside the vessel by about twenty channels (glove fingers), some of which point to one of the hot or cold sources. These channels, extended by neutron guides, then feed some 40 experimental areas equipped with state-of-the-art instrumentation up to 100m from the reactor.

10.13 LIST OF NUCLEAR PRESSURE COMPONENTS IN THE HFR BATTERY PACK

Nom du compartiment	PS (bars rel)	TS (°C)	Fluides contenus
bloc pile HP	15,45	56,5	eau lourde
enceinte externe source chaude	9,2	35/300	hélium
enceinte interne source chaude	4	35/300	hélium
enceinte double enveloppe cellule SFV3	4	100	vide/D2 tritié
Cellule SFV3	3,5	100	D2 tritié
Tube TGV SFV3	2,3	100	Vide+hélium D2 tritié
Doigt de gant V4	1	56,5	hélium
Tube porte source V4	colonne 11 m d'eau	56,5	H2O
Doigt de gant V7	1	56,5	hélium
Tube porte source V7	colonne 11 m d'eau	56,5	H2O
Bloc pile BP	4,85	56,5	eau lourde
Soufflet manchette H1H2	1	56,5	hélium
Soufflet manchette H3	1	56,5	hélium
Soufflet manchette H4	1	56,5	hélium
Soufflet manchette H5	1	56,5	hélium
Soufflet manchette H6	1	56,5	hélium
Soufflet manchette H7	1	56,5	hélium
Soufflet manchette H8	1	56,5	hélium
Soufflet manchette H9	1	56,5	hélium
Soufflet manchette H10	1	56,5	hélium
Soufflet manchette H11	1	56,5	hélium
Soufflet manchette H12	1	56,5	hélium
Soufflet manchette H13	1	56,5	hélium
Soufflet manchette IH1	1	56,5	hélium
Soufflet manchette IH2	1	56,5	hélium
Soufflet manchette IH3	1	56,5	hélium
Soufflet manchette IH4	1	56,5	hélium
Doigt de gant H1H2, intérieur volume C	0,5	56,5	hélium/vide
Doigt de gant H1H2 nez H1, volume B	1,2	56,5	hélium/vide
Doigt de gant H1H2 nez H2, volume B	1,2	56,5	hélium/vide
Doigt de gant H1H2 obturateur liquide H1, volume D	0,5	56,5	hélium/vide/eau
Doigt de gant H1H2 obturateur liquide H2, volume D	0,5	56,5	hélium/vide/eau
Doigt de gant H1H2 fourrure H1H2, volume E	0,5	56,5	eau
Doigt de gant H1H2 plaque de guide H1, volume F	0,5	56,5	vide/hélium
Doigt de gant H1H2 plaque de guide H2, volume F	0,5	56,5	vide/hélium
Doigt de gant H3	1,2	56,5	hélium
Doigt de gant H4	1,2	56,5	hélium
Doigt de gant H6H7 (intérieur)	0,5	56,5	vide/hélium
Doigt de gant H6H7 (soufflets)	1	56,5	hélium
Doigt de gant H8	1,2	56,5	hélium
Doigt de gant H9	0,5	56,5	hélium
Doigt de gant H10	1,2	56,5	hélium
Doigt de gant H11	1,2	56,5	hélium
Doigt de gant H12	1,2	56,5	hélium
Doigt de gant H13	1,2	56,5	hélium
Doigt de gant IH13	1,2	56,5	hélium