Chapter 10
Research on the Behavior of Components Important to the Safety of NPPs and More Particularly the Aging of Such Components

This chapter provides an overview of the most significant R&D efforts on metallic components, civil engineering structures, and other components that play a role in ensuring the safe operation of pressurized water reactors in France's NPPs and which, in particular, play a confinement role ("barrier").

The R&D efforts in which IRSN is particularly involved are related to:

– the behavior of components during accident conditions (earthquake, core melt, etc.);

– aging in a broad sense, i.e. the effects of various mechanisms of damage – or pathologies – that may adversely affect components (metallic structures, civil engineering structures, etc.) over time and subsequent to their use (under normal operating conditions).

A gradual damaging of components can be generated by operating loads and their surrounding environment, such as pressure, temperature, thermal transients, vibrations, irradiation, and the chemical composition of the surrounding medium. Operating experience, and the results of periodic tests and inspections in particular, shows that crippling degradations can occur even though mechanisms of damage—at least those that are known—are taken into account in the design, sizing, manufacturing, and operation of components. Cracks on vessel head adapters and tube bundles of steam generators are just some examples of such degradations. In other cases, the kinetics of degradations were faster than expected (thermal fatigue of pipes).

In the case of components studied in terms of loading during accident conditions, the compliance of "end-of-life" components with safety requirements must of course be verified by factoring in the effects of aging during normal operation—or their possible replacement by new ones.

Research on the behavior of metallic components subjected to loading during accident conditions focuses on understanding the behavior of complex structures such as overhead cranes. Regarding civil engineering structures, this research focuses on developing laws and models of thermomechanical behavior for each component element of these structures (concrete, rebar, etc.). These laws and models are then integrated into simulation codes such as Cast3M, developed by the CEA and which is used by IRSN as parts of its research in support of assessments or before key deadlines—such as the ten-yearly safety reviews of France's reactors and the possibility of operating these reactors beyond their 40-year design lives ("operating life" project [DDF]).

Aging management is based on two key principles—**proactiveness** and **monitoring**. It relates particularly to the components involved in the second and third confinement "barriers"—the reactor-coolant pressure boundary and the containment itself. Indeed, it is vital that these components retain their intended design characteristics (compliance) throughout their service lives and up to and including their disassembly during facility dismantling operations. It is worth reminding that a fundamental objective of metallic components that play a containment barrier role is to maintain a sufficient degree of ductility throughout their service lives. This requirement particularly relates to the PWR vessels, which gradually weaken under the effects of irradiation (increase in the ductile-to-brittle transition temperature).

The purpose of R&D on aging is to improve knowledge on the mechanisms of damage, or pathologies, that can adversely affect components (Figure 10.1). In France, this research took on a new dimension after EDF announced its intention to continue operating its NPPs well beyond their 40-year design lives although some of their components are either irreplaceable or difficult to replace (vessel, containment).

The R&D efforts discussed below—specifically those led by IRSN—thus relate to:

- the metallic components of the reactor coolant system,
- the concrete containments,
- the polymers used as insulation around electrical cables, the inner linings of some containments, and seals.

Generally speaking, IRSN's research is intended to highlight phenomena not taken into account by licensees and improve its understanding of phenomena that are not well documented but nevertheless are important to safety. Therefore, and along with R&D efforts led by other bodies (either alone or jointly), it contributes to the development of standard practices in terms of the design, fabrication, and in-service monitoring of

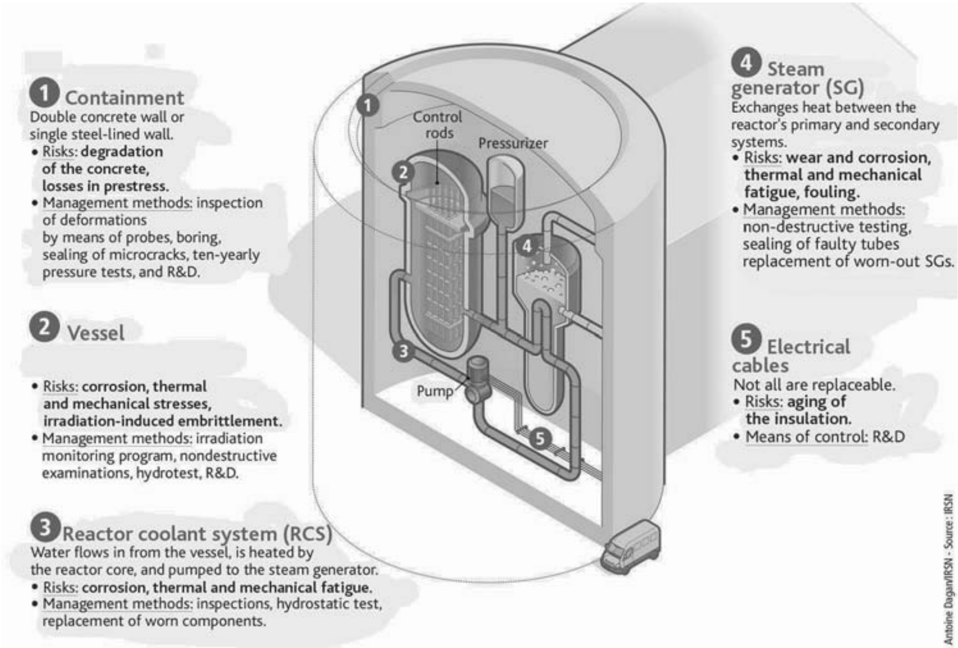

Figure 10.1 Areas of aging in PWRs. Areas 1 and 2 are highly critical, for they contain non-replaceable equipment.

metallic or civil engineering structures, such as those of the RCC-M[180] which applies to the metallic structures in France's NPPs.

10.1. R&D on metallic components

R&D efforts on metallic components involving the key players in France's nuclear power industry primarily relate to the following phenomena:

- irradiation-induced embrittlement of vessel steel. As stated above, this embrittlement alters the mechanical properties of the steel (and increases the ductile-to-brittle transition temperature in particular) and adversely affects its behavior in the case of thermal shocks. This embrittlement can be assessed using empirical prediction formulas adjusted to data derived from analysis of test specimens from the irradiation monitoring program[181] and supplemented by irradiation programs

180. Rules on the Design and Construction of Metallic Equipment (design and construction rules derived and adapted, in the 1970s, from those issued by the ASME [American Society of Mechanical Engineers]).
181. The irradiation monitoring program (PSI) involves testing, for each reactor vessels in France's fleet, samples representative of the vessel steel placed inside capsules along the circumference of the reactor core. These capsules also contain dosimeters to measure the neutron flux density received by the samples. Due to their locations, the capsules are exposed to a higher neutron flux than that received by the vessel walls. This makes it possible to foresee the behavior of the materials after 10, 20, 30, and 40 or more years of operation.

conducted in research reactors. The European PERFORM[182]-60 (2009–2013) project led by EDF and associating AREVA and CEA made it possible to develop a first generation of tools for simulating the microstructural effects of steel irradiation. Research in this field is continuing;

– thermal embrittlement of cast austenitic-ferritic steels used to fabricate a number of components (such as the elbows of the RCS in some French reactors currently in operation). In the early 1980s, it was found that these cast items were susceptible to thermal aging when the RCS was kept at its service temperature for extended periods. The resulting degradation of their mechanical properties (hardening and gradual embrittlement, loss of toughness) was attributed to unmixing of the Fe-Cr-based solid solution into Fe-rich α and Cr-rich α' domains *via* a spinodal mechanisms and precipitation in the ferrite of a nickel and silicon-rich intermetallic phase. The potential risk considered was that of a change in the failure mode of the components, i.e. from a ductile mode to a brittle one requiring little energy, the suddenness of which was particularly worrisome for pressure vessels. With CEA, IPSN studied and tested model materials to (i) understand the structural changes that take place when they are maintained at high temperatures for extended periods and assess the consequences of such changes, (ii) study the scale effects (tests on large and small specimens), and (iii) identify the influence of the ferrite content and the aging duration. These studies and tests made it possible to identify that although the failure of aged cast items is macroscopically brittle, it is actually caused by a ductile failure mechanism that is less dangerous because it requires an input of energy. Numerical simulation tools have been developed to predict the aging of cast elbows on RCSs throughout their entire service life and assess their behavior in the presence of fabrication defects (casting defects, such as shrinkage cavities) or taken into consideration in safety studies (cracks). EDF is also conducting R&D on this issue to justify whether cast elbows deemed sensitive should be left in place or replaced;

– stress corrosion of stainless steels. Conditions that aggravate this damage mechanism (water chemistry, choice of materials and manufacturing conditions) have been identified. Experimental studies have been conducted on this phenomenon as part of international projects, such as the OECD/NEA Halden reactor project (1995–2008) and the EPRI CIR[183] project (1995–2009). Both of these projects yielded significant findings about the initiation and propagation of cracks;

– irradiation-assisted stress corrosion of stainless steels (affecting the vessel internals), which significantly decreases their ductility and leads to the initiation of cracks. This phenomenon has been studied as part of international projects, such as the European PERFECT project (2004–2008), which made use of a multi-scale approach to predicting damage;

182. Prediction of the Effects of Radiation FOR Pressure Vessel and in-core Materials using multi-scale Modelling – 60 years foreseen plant lifetime.
183. Cooperative Irradiation-assisted stress corrosion cracking Research.

- irradiation-induced swelling of stainless steels, which can adversely affect their operability. This phenomenon is being studied by the GONDOLE program (2006–2016) led by CEA at the OSIRIS reactor in Saclay, France, as part of an international partnership (EDF, AREVA, SUEZ-GDF[184], EPRI);
- stress corrosion of nickel-based alloys used to fabricate components such as steam generator tubes and bimetal bonds such as in the vessel head adapters and vessel bottom head penetrations. The effects associated with the chemical conditions of the water in the primary and secondary systems studied as part of the CIRCE and CIRCE-2 programs implemented by EDF and EPRI. IRSN has undertaken research in this field (thesis) as well;
- thermal fatigue, particularly following the 1998 incident at Civaux. This subject will be discussed hereafter;
- wear of steam generator tubes due to their supports. This phenomenon is the main mechanism of damage affecting tube bundles on the latest generation of steam generators in France's NPPs. IRSN, working with CEA, has set up an experimental program to study this wear and determine changes in the coefficient of friction of these tubes with their surrounding conditions (chemistry, steam, etc.).

The implementation of probabilistic approaches to better take the uncertainties and variabilities of the parameters involved into account, particularly in terms of assessing vessel behavior, is also being considered in relation to the possibility of operating France's NPPs beyond their 40-year design lives. EDF is conducting development efforts in collaboration with CEA on this subject. However, IRSN has reservations about the possibility and relevance of lending such approaches to safety demonstrations.

IRSN, working with CEA, is also leading research on the seismic behavior of metallic components such as overhead cranes and precast floors.

Research is also being conducted in the field of non-destructive testing of metallic components.

Both of these subjects will also be discussed further hereafter.

In 2008, EDF created the Materials Ageing Institute (MAI). Led by EDF, the institute is co-financed by NPP licensees including EPRI (Electric Power Research Institute, which represents all nuclear reactor licensees in the United Sates) and KEPCO (Kansai Electric Power Company, in Japan). The MAI brings together the skills of these organizations to predict aging in NPPs and increase the durability of the materials, components, and structures used in them. Knowledge about this phenomenon will be shared with IRSN.

Some of the aforementioned research subjects are discussed more in detail hereafter.

184. French gas utility.

10.1.1. Research on thermal fatigue

Thermal fatigue is a mechanism of damage that EDF, among others, was forced to deal with when, in May 1998, a 30 m^3/h water leak occurred on an RHRS[185] pipe in reactor 1 at the Civaux NPP (N4 series) while the reactor had been shut down for maintenance. This leak was caused by a crazing-like pattern of cracks—known as elephant skin fracture—emerging at the extrados of a pipe elbow (Figure 10.2).

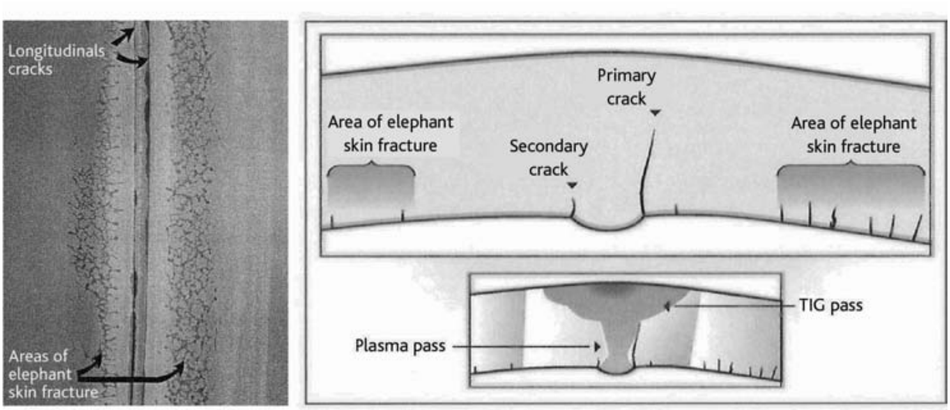

Figure 10.2 The inside of the pipe elbow that caused the leak that occurred at the Civaux NPP in 1998. Crazing-like fractures are visible on either side of the weld bead. © EDF (left) and IRSN–Source: EDF (right).

Thermal fatigue can be summed up as follows: materials expand or contract when subjected to temperature variations. But stresses occur if they cannot expand or contract so freely. Damage from thermal fatigue can result if these variations are too frequent[186].

The crazing-like fracture area of the RHRS pipe (Figure 10.3) of Civaux reactor 1, which was made of austenitic stainless steel, was subjected to significant temperature fluctuations downstream of a mixture of streams (one at 180 °C and the other at 20 °C) at a pressure of 27 bar. The leak occurred after only 1500 hours of operation.

The phenomenon of thermal fatigue in the mixing areas had not been foreseen during the design of the PWRs. The incident that occurred in 1998 could neither be foreseen nor explained by conventional methods and criteria of mechanical fatigue analysis, such as those encoded in the RCC-M and based on the assessment of fatigue "usage factor[187]." It is worth noting that water leaks induced by thermal fatigue had occurred on the RCS before the incident at Civaux. However, they were caused by an altogether different

185. Residual Heat Removal System.
186. Known as high-cycle fatigue, as opposed to oligocyclic fatigue.
187. This usage factor corresponds to the ratio between the number of loads applied to a given component and the maximum number of loads indicated by the mechanical fatigue curve of the component's material.

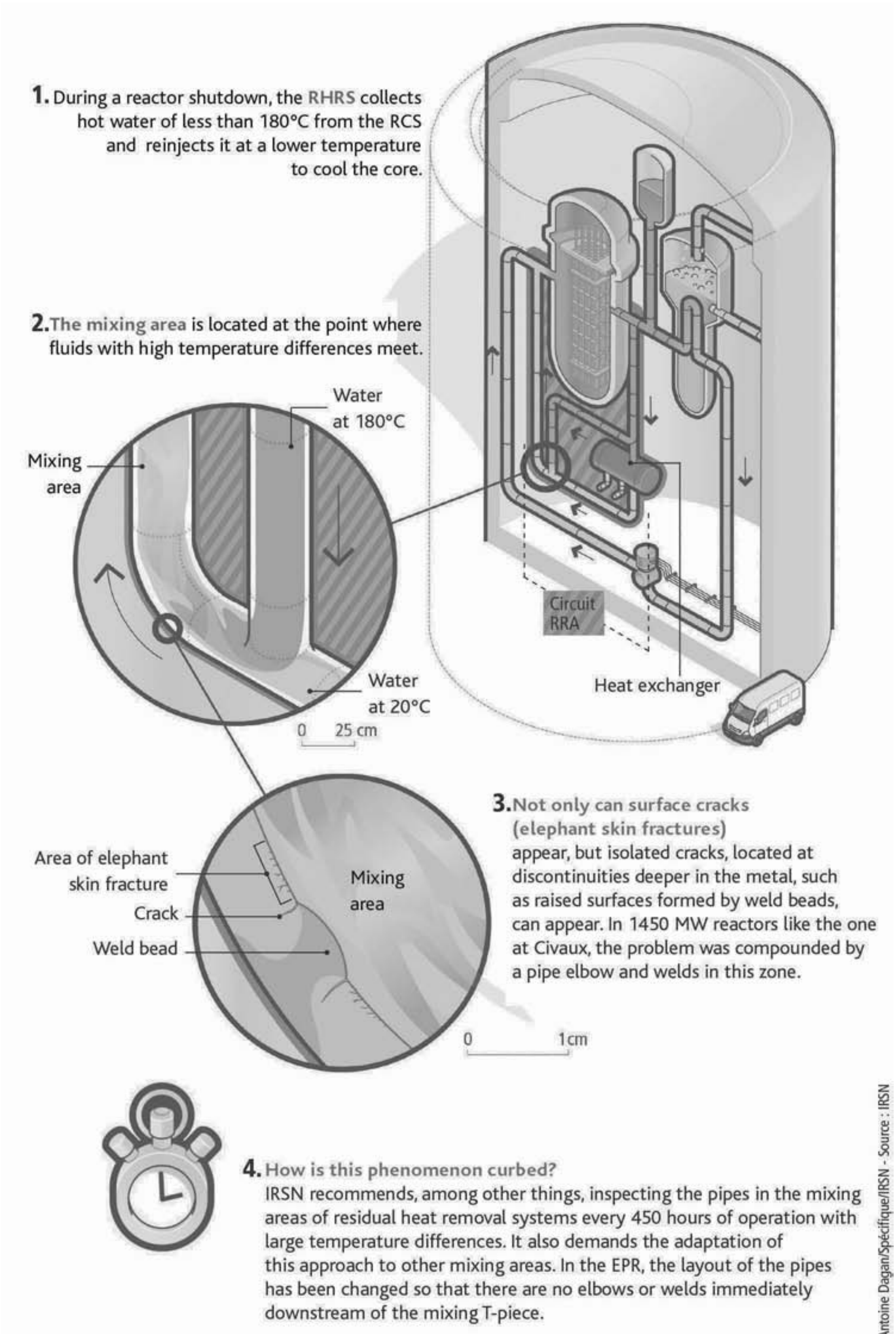

Figure 10.3 Crazing-like fracture on a pipe elbow of the RHRS at Civaux in 1998. © Antoine Dagan/Spécifique/IRSN–Source: IRSN (above) and IRSN–Source: EDF (below).

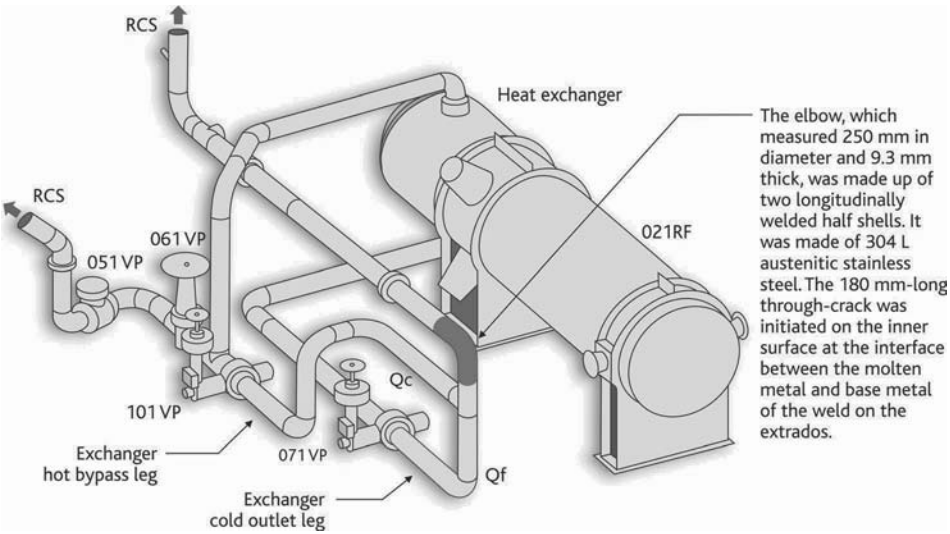

Figure 10.3 (*afterpart*).

phenomenon: dead leg (Farley 2 in 1987, Tihange 1 in 1988, Dampierre 2 in 1992, and Dampierre 1 in 1996).

Starting in 1999, ultrasonic examinations were conducted at every NPP in France. These inspections revealed that the problem was generic (not specific to Civaux) and that all of the examined pipes showed signs of cracking. This prompted EDF to replace the mixing areas of the RHRS circuits in the entire fleet and make improvements to reduce the susceptibility to thermal fatigue (particularly by leveling out the welds).

This finding marked the starting point of more than 10 years of studies and research to understand the cause of the phenomenon and find adequate solutions. These efforts were conducted by EDF and AREVA, and also by IRSN with CEA in particular. Academic laboratories were associated.

EDF's work focused on identifying and assessing the risks of fatigue. Systematic analysis of dismantled pipework, supplemented by mock-up tests (BVS, DUPLEX, FATHER, etc.) and modeling, revealed key factors in the onset of cracks. The primary cause is temperature differences greater than 50 °C between hot fluids and cold fluids and extended periods of repeated pipe loading with this high temperature difference.

IRSN, working with CEA and academic laboratories, implemented a two-pronged research and study program:

- thermal-hydraulic simulations of the flow and mixing of water streams at different temperatures in pipes to better understand the mechanisms at work and key parameters that affect mechanical loads applied to pipes. These simulations were carried out using the Cast3M code in particular;

– mock-up tests (FABIME, SPLASH and FAT3D) to understand the initiation and propagation conditions of thermal fatigue cracks. The FAT3D testing system is illustrated in Figure 10.4 below. Cold water is periodically injected along the inner surface of the tube (test specimen). This water describes a parabola on the inner surface of the tube such as to obtain a temperature gradient in three directions.

A few key teachings were derived from these studies and tests:

– temperature fluctuations at low frequencies (approx. 1 Hz) are responsible for the rapid propagation of cracks through the thickness of components;

– they are caused by turbulence and the specific flow geometry, which are influenced by the configuration of the circuit upstream of the stream mixing area (elbows, straight sections, etc.);

– although thermal fatigue damage is accelerated by welded joints, it can also occur along continuous weld beads (FAT3D test 6). This important teaching invalidated the position previously defended by EDF, prompting it to expand inspections of its facilities to areas not comprising welds;

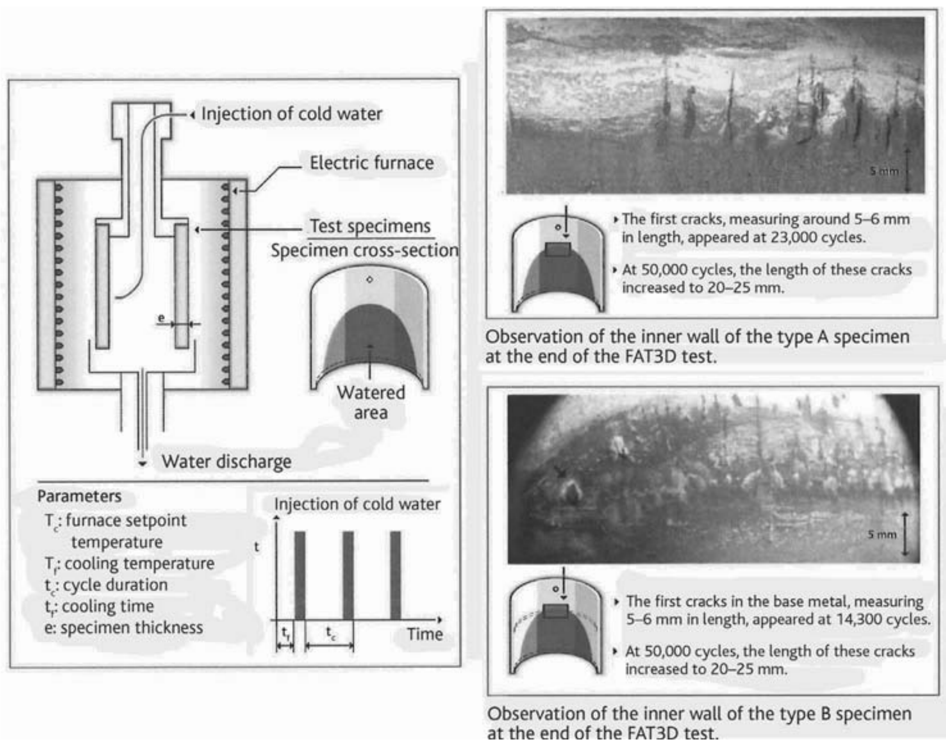

Figure 10.4 The FAT3D test system (left) and the results obtained from tests on specimens (A) without a weld bead (top right) and with (B) a weld bead (bottom right). © IRSN.

- the biaxial nature of thermo-mechanical loads and "environmental" effects may account for the deficiency of conventional methods and criteria used to design and size materials subject to fatigue. These aspects are still being studied (see further on);
- studies conducted on the nozzle area of the chemical and volume control system (CVCS) of the RCS — an area subjected to temperature differences much greater than those occurring in the elbows of the RHRS system (and reaching as high as 280 °C) — led researchers to consider that *a priori* the role of the flow type is more important than that of the temperature difference. Indeed, investigations have shown that the CVCS nozzles in the fleet suffered very little damage from thermal fatigue. This finding has not been invalidated by in-service inspections conducted since.

Based on the results of thermal fatigue studies and research, EDF defined a policy for the operation, in-service monitoring and replacement of mixing areas within all of its reactors. Starting in 2000, the areas of the RHRS circuits subjected to high temperature differences were ultrasonically examined after every 450 hours of operation (following, in this regard, a recommendation of IRSN), and maximum periods of operation at high temperature differences were defined for all susceptible areas.

Nonetheless, research on fatigue is continuing, for much work remains ahead to understand the mechanisms involved and the conditions that cause damage to appear. In 2013, IRSN embarked on the EVA[188] project with INSA in Lyon. The aim of the project is to specifically explore the aforementioned environment effects under the conditions typically encountered in PWRs (pressure, temperature, water chemistry) on the fatigue life of austenitic (and even austeno-ferritic) steels. These effects may explain the lack of conservatism of the in-air fatigue curves of the ASME and transposed in the RCC-M. The testbed will consist of a fatigue loading machine and an autoclave. IRSN has also been a partner of the European INCEFA[189] project, which explores the same subject, since 2014.

10.1.2. R&D on non-destructive testing

Inspections performed during the fabrication of components intended to be used in the construction and operation (in-service inspections) of nuclear facilities are a vital part of defense in depth. For the French fleet of NPPs, these practices are codified in a rule book on in-service inspection for mechanical components (RSE-M[190]). Nonetheless, a number of questions and difficulties arise, such as the adequacy of the type and performance levels of inspection equipment used by manufacturers and licensees (or their contractors) to areas to be inspected that have complex geometries (see Figures 10.5 and 10.6) or materials with specific metallurgical structures. This prompted IPSN, in the early 1990s, to begin R&D efforts to develop sensor prototypes able to adjust to these complex shapes. This research in the field of non-destructive testing (or non-destructive examination [NDE]) deals with ultrasonic waves and eddy currents as well as

188. Study of the Ageing of Steel.
189. INcreasing Safety in NPPs by Covering gaps in Environmental Fatigue Assessment.
190. Rules for the Monitoring of Mechanical Equipment in Operation.

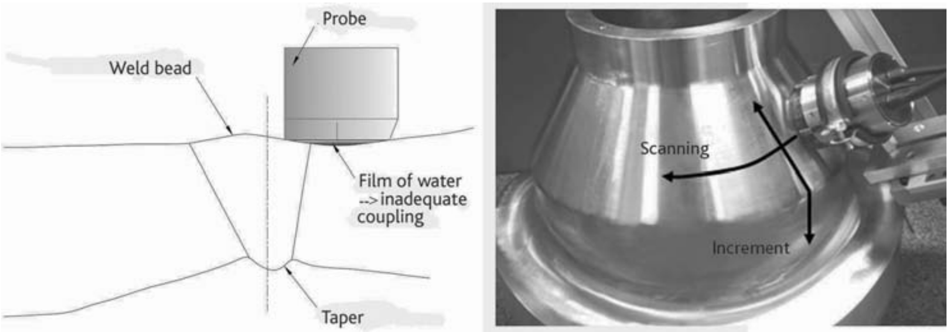

Figure 10.5 Examples of complex geometries: manual ultrasonic examination of a weld bead and ultrasonic examination of an intricately shaped nozzle. © IRSN.

simulations of radiographic examinations. Its aim is to motivate licensees to look for and use the best possible techniques for inspecting the components in their facilities.

Research in this field, conducted primarily by IRSN and CEA, pertains to all nuclear facilities that may be affected by aging. As mentioned above, it deals with the development of innovative prototypes of transducers or probes and examination simulation models. The transducers or probes developed through this research are

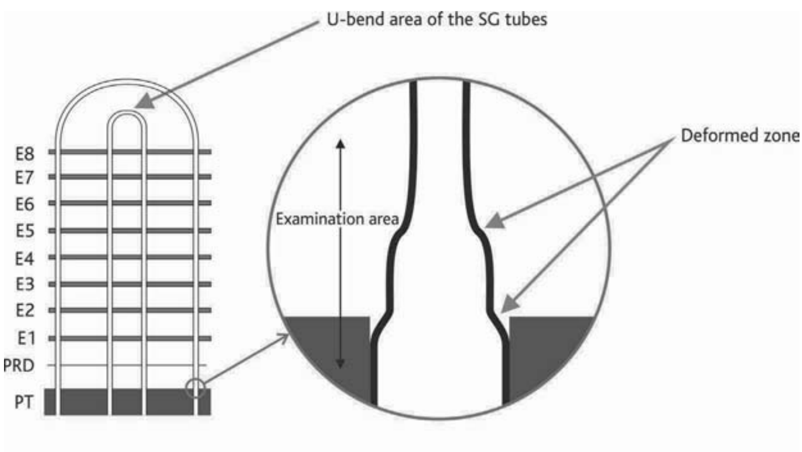

Figure 10.6 Another example of complex geometry: the transition region of the roll-expanded portion of the steam generator tubes and the[191] U-bend areas of the tubes. © IRSN.

191. The purpose of roll expansion of a tube into a plate is to create a radially expansion by mechanical mean or by internal hydraulic overpressure and cause the tube to plastically deform. The residual stresses cause contact pressure between the tube and the plate, resulting in a strong mechanical bond between the tube and the plate. The tools used must not create sharp sloped marks on the metal, tear off any metal, or initiate cracks (Figure 10.6).

intended to be used to check for damage, such as cracks, that may adversely affect safety. These innovative devices, which are usually patented, relate especially to components important to the safe operation of PWRs, some of which are considered failsafe[192] in the safety analysis but for which degradation mechanisms are nevertheless postulated. Significant examination difficulties still remain for some components, such a reactor-coolant pipes made of coarse-grained heterogeneous materials, in which the ultrasonic waves propagate with difficulty, or components with complex surfaces, or the U-bend portions of steam-generator tubes (Figure 10.6). The difficulties encountered with these components should be overcome by the higher performance obtained with these adaptive sensors, which make it easier for ultrasonic waves and signals to penetrate into materials. Signal processing techniques are also being considered for coarse-grained materials.

Simulation models have been developed for the most commonly used inspection techniques, i.e. as ultrasonic examination, radiographic examination, and eddy current examination. These models have been integrated into the CIVA simulation platform developed by CEA and are now accessible to all the relevant users. They are used routinely by IRSN and have become essential to assessing the performance of examinations conducted by manufacturers, licensees or their contractors, in order to substantiate its technical opinions to safety authorities. Most of these developments have been engineered in collaboration with CEA. More specific studies on the examination of coarse-grained materials are being conducted with the U.S.NRC in association with PNNL, and studies on the examination of steam generator tubes are being conducted with ANL.

A) Development of transducers and probes

► Development of a "conformable" ultrasonic transducer for parts with complex geometries

A conformable ultrasonic transducer—i.e. one that conforms to the shapes of parts—was developed by IPSN in collaboration with CEA and patented in 2003.

The main objective of this transducer (Figure 10.7) is to enable or improve detection and geometric characterization of defects in components with complex geometries (small elbows, small nozzles, non-uniform surfaces, etc.) on which acoustic coupling of conventional rigid sensors can no longer be suitably accomplished. It was necessary to improve these examinations, particularly for components most important to safety and for which cracking damage is a possibility. The aforementioned multi-element flexible transducer adjusts to the shape of parts.

192. Components for which design, fabrication, and in-service inspection provisions make it possible to consider that their failure is highly unlikely. Their failure is not addressed by specific consequence-mitigation provisions.

Figure 10.7 Prototype of a conformable transducer – IRSN/CEA patent. © IRSN/CEA.

As seen in Figure 10.8, which shows a study conducted by Laborelec and CEA for the examination of welds on large nozzles, this type of transducer is now industrially applicable.

Figure 10.8 Examination of a nozzle with a conformable transducer (2014 Cofrend Conference – Strategy for robotized examination of nuclear components: from examination design to experimental results). © CEA.

► Development of eddy current sensors for examining steam generator tubes

Examination of steam generator tubes is an important safety concern due to the risk of tube failure and release into the environment of radioactive substances (reactor containment bypass). A steam generator contains anywhere between 3000 and 5000 tubes.

These tubes are examined using eddy current sensors that are generally rigid and well suited to detecting defects in uniform surfaces, such as the straight sections of the tubes. However, conventional rigid sensors can quickly reach their limitations with complex shapes encountered in the transition regions of the roll-expanded portions of steam generators or in small tube bends that may contain deformations (ovality, crushing, etc.). IRSN and CEA jointly developed a flexible sensor to overcome such difficulties and adequately conduct examinations in these areas potentially subject to damage.

Flexible sensor technology (Figures 10.9 and 10.10) is based on the use of winding etched on a Kapton substrate[193] and giant magnetoresistance (GMR) receivers that are highly sensitive to magnetic fields. This technology improves the detection of circumferential defects in areas that are hard to examine due to the presence of longitudinal cracks. CEA and IRSN filed a patent for this sensor in 2009.

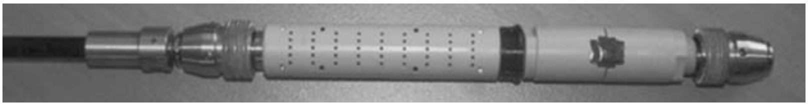

Figure 10.9 Flexible eddy current sensor for examining SG tubes. © DR.

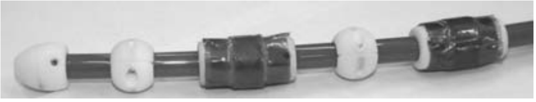

Figure 10.10 Prototype of a flexible eddy current sensor for examining SG tube bends. © DR.

► Development of sensors for castings and heterogeneous structures

Prototype ultrasonic transducers and a prototype of a low-frequency eddy current sensor have also been developed to detect and size planar defects in castings, which have a very coarse grain structure.

Surface-breaking defects along the inner walls of components (pipework, etc.) are generally adequately detected by ultrasonic waves, for the echoes result from an ultrasonic concentration of energy obtained by a "corner effect[194]." However, the size of the planar defects is obtained by measuring the diffraction echoes obtained at the tops of the defects. These echoes actually have very low amplitudes and are

193. Kapton is in insulating polyamide film with mechanical properties (flexibility, thinness, temperature resistance, film etching, etc.) that allow for many uses in electrical and electronic applications.

194. The corner effect is a phenomenon of overintensity of the ultrasonic echo due to multiple reflections on the corner formed at the intersection of the inner surface of the material with planar crack, whether surface-breaking or near-surface, directed generally perpendicular to the surface.

accompanied by a more or less loud noise due to the coarse-grained metallurgical structure that leads to a low signal-to-noise ratio. The progress is significant since extension defects at depths of 10 to 15 mm are sized (characterized) in structures known to be difficult. That said, more progress remains to be made to further improve and confirm performance through tests on a wider range of materials and by improving signal processing and transducer technology.

IRSN and CEA are jointly developing low-frequency eddy current sensors to detect defects located below the surface of thick components. Encouraging results have been obtained with the low-frequency sensor, which also includes a GMR receiver. The prototype detects planar defects that vertically extend between 5 and 15 mm below the surface and produce signals having an amplitude related to the vertical extension of the defect. This eddy current technology should be able to supplement and, in some cases, even replace ultrasonic examinations where measuring the vertical depth of planar defects is not possible, particularly in components made of coarse-grained stainless steels and other materials that strongly disrupt the propagation of ultrasonic waves. Eddy current technology nevertheless has the drawback of requiring access to the inner surface of the piping or component to be examined.

Research to improve the detectability of defects for both ultrasonic examination and eddy current examination will be conducted through a partnership between IRSN and the U.S.NRC.

B) Development of simulation models

► Simulation of ultrasonic examinations of materials with a homogeneous structure

In the late 1990s, IPSN and CEA began developing ultrasonic examination simulation models to provide support tools for the expertises. These models were integrated in the CIVA platform. The studies made it possible to address the case of homogeneous materials in which ultrasonic waves propagate without any particular difficulty for parts of simple or complex shape. These models can be applied to the ultrasonic contact sensors (i.e. in direct contact with components) and contactless immersion sensors[195] on the in-service inspection equipment used to examine welds on the reactor vessels used in France's NPPs. This research was subsequently continued to take into account defects of more complex shape that may be disoriented, in order to be able to adequately simulate defects most similar to those actually encountered in the components. An example of a study is shown in Figure 10.11.

Engineered for homogeneous materials, these models are operational and frequently used in expertises conducted by IRSN for thenuclear safety authorities.

195. Other models for simulating examinations using the time of flight diffraction (TOFD) technique have been developed by CEA and EDF and are available in CIVA.

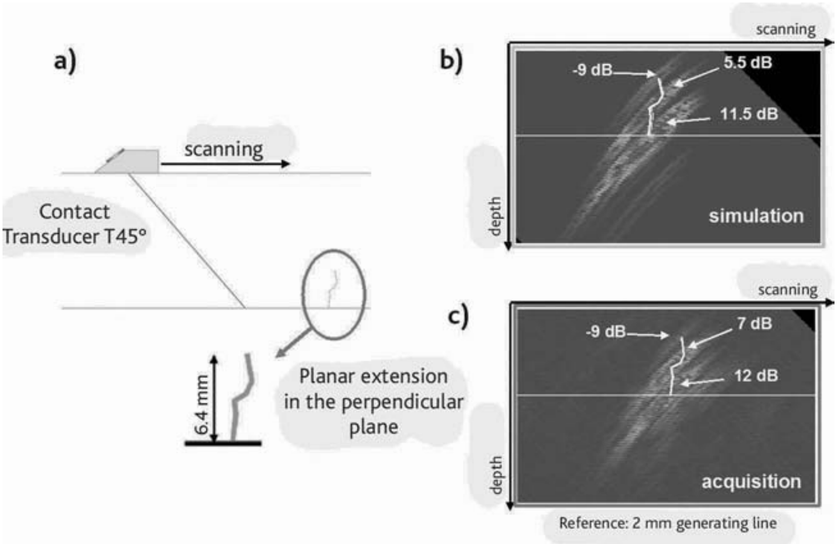

Figure 10.11 Ultrasonic response of a multifaceted defect during contact examination using 45° transverse waves. a) examination configuration; b) "simulated" defect image; c) "experimental" defect image. © DR.

► Simulation of eddy current examinations

CIVA's functionalities for eddy current examinations make it possible to simulate examinations of steam generator tubes with an axial probe (Figure 10.12), rotating probes (STL[196], STT[197], +Point), and multi-element probes. The first phase of development of these models was intentded for a simple configuration with a tube and an isolated defect. Now models make it possible to take into consideration a more realistic geometry of tubes and their environment (geometric transition region between the roll-expanded portion of tubed in tubes plates and their straight portions, areas located under the spacer plates, U-bend areas). It is also possible to model complex defects or a network of defects. Validation of CIVA is accomplished by comparing it with experimental data and by conducting test cases (benchmarks). The resulting confidence in the use of simulation tools makes it possible to use them to assess examination techniques.

Furthermore, CEA has developed CIVA functionalities that simulate the X-Probe, an eddy current multi-element probe most commonly used outside France, particularly the United States. The functionalities will enable IRSN to better understand the performance and limits of this probe, which now numbers among the examination methods usable by EDF.

196. Long Rotating Probe.
197. Transverse Rotating Probe.

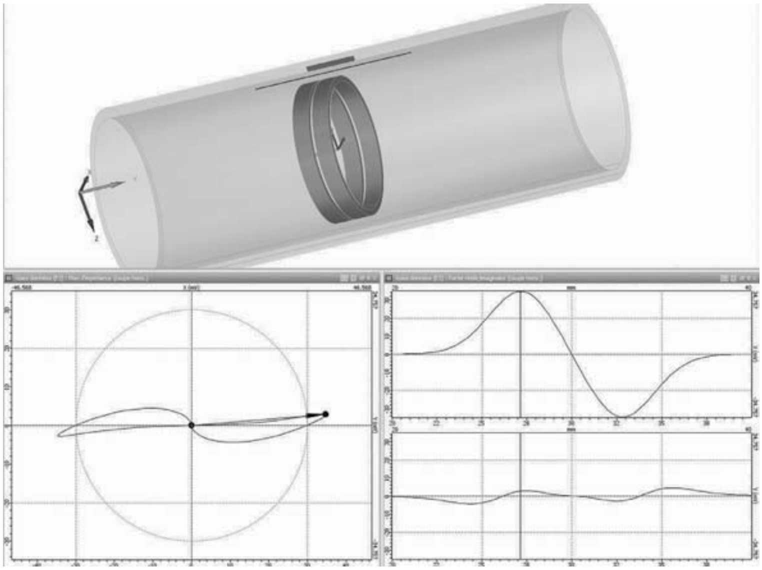

Figure 10.12 Simulation of the response of the axial eddy current probe for an outer notch. © IRSN.

► Simulation of radiographic examinations

The development by IRSN, in collaboration with CEA, of tools for simulating radiographic examinations began much later than research on ultrasonic and eddy current examinations. As a first step, the configurations of the radiographs most commonly used for the components of nuclear facilities—which can have various dimensions, thicknesses, shapes, access conditions—were identified. This made it possible to develop models that reproduced the most common operating conditions and took into account iridium and cobalt radiation sources and silver-image films in use in the facilities. Attention was then turned to the case of welds of dissimilar materials (e.g. stainless steel coatings deposited on ferritic steel, bimetallic welds used to bond stainless steel pipes to ferritic steel components). This was accomplished, for example, by using computer-aided design (CAD) software to describe the part to be radiographed and the various areas of material on the part and to which the properties of each material in question is assigned. A phased approach was also applied to defects for which detection is to be simulated; these defects shifted from very simple shapes at the start of the studies to much more complex shapes that can be described using 3D CAD software.

Starting in 2010, emphasis was especially placed on experimental validation of simulation models using mock-ups representative of components and with calibrated defects. Such validation was conducted on a few of the radiograph configurations studied previously. They made it possible to compare the results obtained through simulation with those obtained during the experimental radiographs. These model validations are being continued to the works to address the most common operating conditions[198] of radiographic examination and are the subject of publications or benchmarks.

Since 2013, research is being continued to take into account such aspects as other radiation sources like selenium (less exposure to operators, but penetrate less through structures), X-ray for large thicknesses, and digital films.

Here, too, the models developed and integrated into CIVA are used regularly at IRSN for expertises and parameter studies.

► Simulation of ultrasonic examinations of coarse-grained materials with a heterogeneous structure and improvement in these examinations

After developing simulation models suited to ultrasonic examinations of components fabricated from homogeneous materials or homogeneous welds—which generally do not prevent the propagation of ultrasonic waves and make it possible to assess with a high degree of confidence the performance of many ultrasonic examinations—IRSN considered that it was necessary to continue its studies in order to develop simulation models more specifically suited to ultrasonic examination of heterogeneous materials.

The heterogeneous materials in the RCS of PWRs—found, for example, in cast portions or welds used to bond stainless steel pipes to large RCS components made of ferritic steel (bimetal bonds)—can significantly hinder the propagation of ultrasonic waves. This is due to the special metallurgical structure of the materials, which contain coarse grains (both castings and bimetal bonds) that vary in direction and size according to the depth. This causes the speed of the waves to change and leads to various dispersion or attenuation phenomena at the interfaces with the grains. These alterations impair the detection or sizing of defects. In either case, simulated and real-life examination of these specific areas of materials remains a major focus of R&D and there are many difficulties to overcome. Simulating these examinations requires being able to describe the complex structure of the material (base metal and weld) in the 2D or 3D model and to calculate the acoustic field transmitted through the materials while not forgetting that complex structures will produce more noise than homogeneous structures. Lastly, it requires being able to calculate the various interactions in the disrupted field during penetration into the various structures with the encountered defects.

Despite the limited number (Figure 10.13) of mock-ups representative of this type of materials, it was possible to develop models that predict the propagation of ultrasonic waves. This was accomplished by macrographic examinations, conducted directly on these mock-ups, and which made it possible describe simplified structures that yield the contours and directions of grains. The results of these models were very promising, but, due to the limited number of cases studied, tests on other mock-ups had to be conducted to confirm and improve the predictions.

198. Panoramic radiograph with the source centered in the piping and the films located outside, radiographs with the source located outside in contact with the piping, etc.

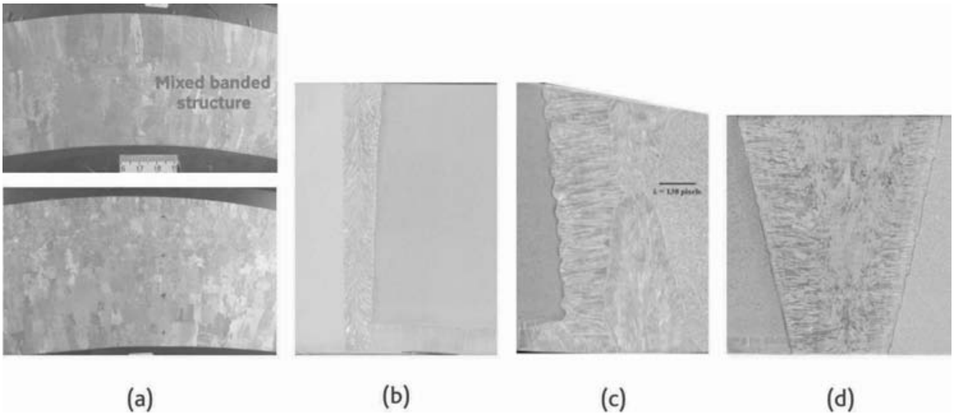

Figure 10.13 Coarse-grained metallurgical structures observed on IRSN mock-ups representative of: (a) a cast component fabricated by Manoir; (b) a narrow groove weld; (c) stainless steel bimetal bond fabricated by Cockerill; (d) "N4" vessel bimetal bond. © IRSN.

Another major difficulty in the simulation of ultrasonic examination of these complex structures is that the description of the structure of the materials must be known in order to make a simplified description of the structure and its grains. This can be achieved via shape recognition (Figure 10.14) from macrographic examinations such as those shown in Figure 10.13 for mock-ups used to develop the models.

Unfortunately, there are no representative samples of such components for most of the heterogeneous structures and/or welds encountered in France's NPPs or in those of other countries. Other methods of description than those using macrographic examination must therefore be investigated.

Studies are thus continuing. Firstly, to explore non-destructive methods that will make it possible to obtain sufficient descriptions of the structures of materials and their

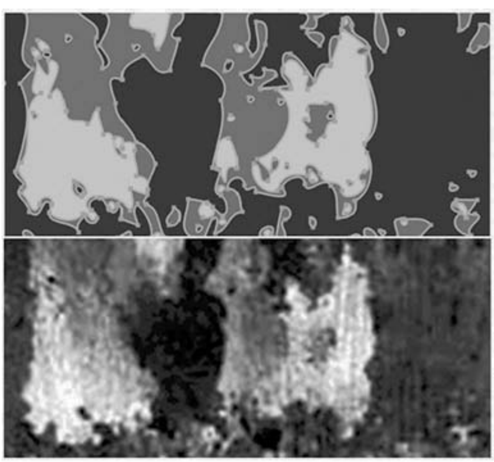

Figure 10.14 Example of grain-shape recognition for cast steel fabricated by Manoir. © IRSN.

grains and correctly predict the performance of examinations, and secondly to assess and even develop codes that simulate solidification from fabrication parameters for cast and welded components for bimetal bonds. It would then be necessary, for example, to be able to simulate examinations on the components in the fleet, to have access to fabrication or welding data that could be described by category. Such access could be made possible by contacting the manufacturers or builders.

Models developed for bimetal bonds and cast coarse-grained materials require further improvement, for the differences between simulated and experimental results remain too significant and depend on the reliability of the methods used for determining the characteristics of the materials. IRSN thus considers that simulations for these components are not sufficiently operational. Research being conducted by IRSN and the U.S.NRC, in association with CEA and PNNL, will make it possible for these bodies to share their data- and mock-ups representative of components. This collaboration is expected to continue until 2017 and will also include research on developing prototypes of transducers and eddy current probes better suited to detecting and sizing defects in these difficult-to-inspect materials.

10.1.3. Studies and research on the seismic behavior of overhead cranes

IRSN, in collaboration with CEA, began conducting studies and research on the dynamic behavior, under seismic loads, of overhead cranes made up of welded elements[199]. The behavior under seismic loading of such structures is highly complex, for phenomena such as multiple impacts (in case of blocking of a wheel) or trolley slippage may occur.

IRSN investigated this issue during assessments of documents relating to seismic reviews of various nuclear facilities, for the demonstrations by the licensees of the earthquake resistance of overhead cranes and absence of load drops involved methods having a robustness that called for confirmation (modal analysis method associated with the use of "reduced spectra" or "reduced weights").

It first appeared necessary to improve the understanding of the seismic behavior of overhead cranes by deriving the maximum possible benefit from the experimental (shake tables) and simulation (finite-element calculation codes) means available.

This research (2005–2013 [1]) consisted of a theoretical part of numerical simulations, and an experimental part with tests performed on the AZALEE shake table at CEA's Saclay research center[200] (Figure 10.15). The tests were conducted on a reduced-scale

199. I.e. assemblies with welded connections, as opposed to assemblies with bolted connections, which have gaps.

200. The Seismic Mechanic Studies Laboratory (EMSI) operated by CEA's Saclay research center has been conducting earthquake-resistance engineering research for more than 40 years. EMSI's aim is to use its numerical and experimental tools to further understand the behavior of structures, equipment, and components under seismic excitation. The laboratory's experimental equipment is housed within the TAMARIS (Tables and Means of Analysing Seismic Risks) facility. AZALEE is currently the largest three-axis shake table in Europe.

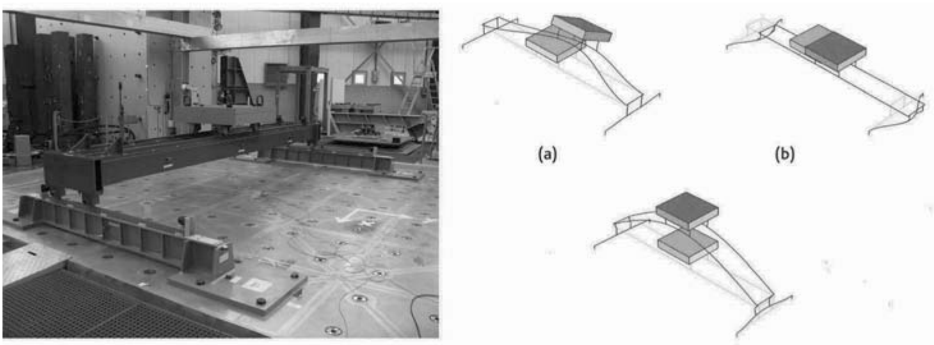

Figure 10.15 Overhead crane tested on the AZALEE table (left) and the three main modes of deformation obtained by numerical simulation. © CEA (left).

mock-up (1/5). Seismic loading was simulated by controlled jacks that moved the structure in both horizontal directions of the table. Firstly, bibliographic researches made it possible to identify the parameters that affect the behavior of overhead cranes. The mock-up's geometry and scale were defined using finite-element calculations and the objective was to find, in the mock-up, the main eigenmodes of the crane's actual structure[201]. More than 100 configurations were achieved by varying a number of influencing parameters, in particular the trolley position, the type of contact at the wheels (blocked wheels or freely rotating wheels), the "added weight" (load suspended from the trolley) and its position. A simplified numerical model able to reproduce the dynamic behavior of this structure with reduced calculation times was implemented. This model was validated for several configurations representative of the operating conditions of the overhead cranes. It may be used by IRSN for seismic risk studies, in particular to understand the behavior of overhead cranes in case of earthquake beyond the design basis earthquake (DBE).

The lessons derived from this research may subsequently enable to IRSN to propose tracks for ultimately establishing rules on analyzing the dynamic interactions between overhead cranes and civil engineering structures.

10.2. R&D on civil engineering structures

R&D work on the civil engineering structures of PWRs relates to primarily to the containment. The containment is essential, for it is the last "barrier" between the reactor and the natural environment.

We would like to remind the reader that three types of containment are used at the PWRs in France's NPPs:

- the first type (900 MWe reactors) consists of a building having a single wall of prestressed reinforced concrete lined on the inside with steel plate, referred to as a static containment;

201. The reference overhead crane is a generic crane used in nuclear facilities.

– the second type (1300 and 1450 MWe reactors) are double-wall buildings comprising an inner wall of prestressed concrete and an outer wall of reinforced concrete. Dynamic containment is ensured by ventilation and filtration of the annulus space between the two walls, thus supplementing the static containment achieved by the inner wall. Composite (resin) liners were applied to the intrados of the inner walls to improve their leaktightness;

– the third type (EPR) is a combination of the previous two: leaktightness is achieved by a metallic liner on the intrados of the inner wall and supplemented by the dynamic containment associated with double-wall containments.

The behavior of these containments under design-basis conditions and during core melt accidents is described in the IRSN document on the state of knowledge on core melt accidents in power reactors[202].

Furthermore, an important aspect regarding pre-EPR reactors in France's NPPs is that the thermo-mechanical loads on the containments during core melt accidents is more severe than the design-basis loads (around 5 bar absolute[203]), since that the "U5" device was subsequently fitted to decompress the containments should the need arise and vent filtered releases to the environment (sand-bed filter).

One of the aims of research on concrete civil engineering structures is to study the behavior of containments during accident conditions and taking aging effects into account. Its purpose is to allow the assessment of the mechanical behavior and leaktightness of the walls of the containments, which can be altered by the combined mechanisms of shrinkage, creep, and "pathologies." Some of these "pathologies," such as alkali-aggregate reactions (AAR) and delayed ettringite formation (DEF), are caused by chemical reactions that occur a few decades following construction. Research is being conducted to analyze these complex phenomena and control them in order to ensure that the safety requirements for these civil engineering structures are met.

Before discussing the R&D efforts being conducted more specifically by IRSN on nuclear civil engineering structures, three major research projects that EDF has led or is currently leading on such structures must be mentioned:

– MAI (mentioned in Section 10.1);

– the nationwide CEOS.fr project on the behavior and assessment of cracking and shrinkage defects in special structures [2008–2014] and for which the French Institute for Applied Research and Experimentation in Civil Engineering (IREX) provided administrative and logistical support. The aims of this project—which was supported by 41 partners, including IRSN—were to study cracking induced by various loads (monotonic or cyclical static loading) in walls and massive components made of reinforced concrete and the early-age behavior—extending from

202. See Chapters 6.2 and 6.3 of "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.

203. This value corresponds to the design pressure that encompasses LOCA, and it is used to conduct containment integrity tests.

10 days following casting—to the maturity under the various aforementioned loads, with special attention given to cracks formed in these structures. Recommendations for controlling these cracking phenomena were compiled in a document published in 2015 [2];

– the VERCORS project (2013–2021), which we will discuss further on in this document.

10.2.1. Development of constitutive equations for civil engineering structures

IRSN is leading research on the behavior of containments under seismic loads or and during core melt accident conditions. It uses the Cast3M numerical simulation code, which was developed by CEA with IRSN's contribution (for example, it provided CEA with a number of rheological models derived from IRSN-funded PhD work). An important point that must be emphasized is that these simulations are checked and validated by comparing them to the containment monitoring activities conducted continuously by EDF and during testing.

Since the 1980s, IRSN has been working with CEA's specialist laboratories that produce the developments necessary to establish constitutive equations for the component materials of the containments (concrete, steel), ranging from the linear behavior of the structures to their non-linear behavior and taking account of concrete cracking and structural damage. These equations are used in research on the behavior of the containments under complex loads. The objective is to be able to simulate the behavior of the containments from their construction to a potential accident.

The development of these equations is based on tests conducted on specimens that have been verified on structures (slabs, beams, gantries) and validated with the results of tests conducted on large-scale mock-ups (a particular example is the tests conducted on the RCCV[204] and PCCV[205] mock-ups at Sandia National Laboratories in the United States)[206].

These R&D efforts make it possible for IRSN to assess, using Cast3M and as part of physics research conducted in support of the development of level-2 probabilistic safety assessments (PSA), the behavior of the containments during core melt accidents. These studies began with the 900 MWe reactor containments in the 1990s. A multi-scale approach developed by IRSN, and based on the computing resources available at the time, was used to determine with a sufficient degree of accuracy, the susceptible areas of these containments (Figure 10.16). These studies showed that the equipment hatch—and more particularly the closure flanges—was a weak spot in the event of a core melt

204. Reinforced Concrete Containment Vessel.
205. Prestressed Concrete Containment Vessel.
206. Test conducted as part of a collaboration between Sandia National Laboratories (SNL), the U.S. NRC, and the Nuclear Power Engineering Center of Japan (NUPEC).

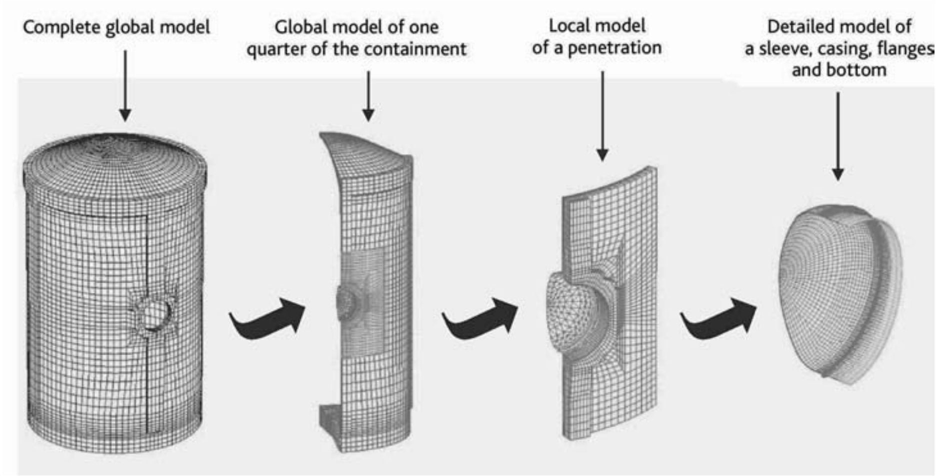

Figure 10.16 Modeling of the containments and the equipment hatch. © Georges Nahas/IRSN.

accident. This prompted EDF to schedule the closure system to be changed during the third ten-yearly outages of these reactors[207]. The system now comprises bolts that are larger in diameter and made of a different grade of steel.

These simulation tools thus offer the possibility of exploring the behavior of containments subjected to other loads during accident conditions (collisions, earthquake, etc.) and even beyond-design-basis combinations of loads during accident conditions.

10.2.2. R&D on the behavior of civil engineering structures under seismic loads

As mentioned above, the development of constitutive equations for the various components of the civil engineering structures and of rheological models makes it possible, with the Cast3M simulation code, to study the non-linear dynamic behavior of structures under seismic loads up to and beyond the design-basis earthquake.

The experimental programs were conducted at the request of IPSN (then IRSN) on simple and complex structures such as gantries and walls using the shake tables at CEA's Saclay research center (dynamic loads). At the same time, other tests on identical structures and under alternate static and increasing loads up to failure were conducted at CEBTP[208]. All these tests made it possible to acquire new knowledge about the dynamic behavior of reinforced concrete structures during earthquakes.

207. See Section 4.4.2.2 of "Nuclear power reactor core melt accidents – State of knowledge" – Science and Technology Series – IRSN/EDP Sciences – 2013.
208. French Construction and Public Works Test Center.

R&D efforts are being conducted in other areas, in particular:

- soil-structure interaction,
- evaluations of motions transferred from floors to equipment installed on different floors,
- the dynamic behavior of reinforced concrete structures.

Another focus of research conducted in collaboration with CEA pertains to seismic isolation systems for the structures, in particular regarding their application to new facilities (Jules Horowitz reactor (JHR), International Thermonuclear Experimental Reactor (ITER), both at Cadarache center).

These areas are discussed in more detail below.

A) Soil-structure interaction

Knowledge of the effects of the soil-structure interaction (SSI) is key to assessing the dynamic response, under seismic loads, of a building such as a PWR, which weighs around 50,000 tonnes. This interaction can be studied using different methods. Two were explored by IRSN—one developed at the École Centrale de Paris (integral equations method), and one developed by researchers at CEA's Saclay research center (absorbing boundary method[209] to limit the size of the field representing the soil). IRSN initiated a collaboration with the École Centrale de Paris to use its method with the MISS3D[210] code. The use of the integral equations method in the finite-element calculations of the structures was achieved by sequentially chaining the MISS3D and Cast3M codes. Both methods were validated during IRSN's participation, in collaboration with CEA, in the international KARISMA[211] benchmark set up by the IAEA following the earthquake (magnitude of 6.8 on the Richter scale) that affected the Kashiwasaki Kariwa NPP and its seven boiling water reactors on July 16, 2007. Preference was given to the method developed by CEA following numerical simulations conducted on one of the Kashiwasaki Kariwa BWRs. However, although the comparison of the calculation results obtained with the on-site measurements confirmed the ability of the model achieved with this method to reproduce the horizontal displacements, differences due to the influence of detachment of the basemats during the earthquake were observed for the vertical displacements. Additional research is being conducted to take into account, in the model, the influence of the non-linearity introduced by detachment.

B) Assessment of displacements transferred to equipment

Advances in simulation capabilities also serve to improve the prediction quality of displacements transferred by the buildings to equipment by taking into account the

209. This method makes it possible to conduct seismic soil-structure interaction calculations via finite-element analysis by limiting the size of the field representing the soil. The resolution is made in the time field for the entire soil-structure meshing and makes it possible to model the detachment and slippage of the basemat (Cast3M code manual).
210. 3D Modelling of Ground-Structure Interaction.
211. KAshiwazaki-Kariwa Research Initiative for Seismic Margin Assessment.

non-linear behavior of the structures (cracking effect of the concrete) in, for example, assessments of existing margins beyond design-basis loads for the structures and equipment.

Seismic margin assessment is a priority that emerged particularly in the United States in the 1980s. The aim was to understand the possible contribution to the overall risk of core melt accident by beyond-design-basis earthquakes that could affect NPPs located in the eastern[212] part of the nation. This resulted in different approaches[213] referred collectively as Seismic Margins Assessment (SMA).

Furthermore, after the earthquake that affected the Kashiwasaki Kariwa NPP on July 16, 2007, *in situ* observations showed that some equipment items had withstood loading beyond their design basis. More recently, an earthquake affected the North Anna NPP in Mineral, Virginia, in August 2011. The ground accelerations at the two reactors were assessed to be 0.2 g and 0.3 g, higher than the design-basis earthquake, which was 0.12 g and 0.18 g, respectively[214]. The cumulative absolute velocity (CAV) threshold, of 0.16 g.s, was exceeded during the quake, causing the reactors to automatically shut down. The generators started up following the resulting electric network failure. The reactors were returned to operation in December 2011 after investigations (including inspections by the U.S.NRC) showed that the earthquake had not caused significant damage to the NPP's safety-important equipment.

Needless to say, the issue of seismic margins returned to the forefront following the Fukushima Daiichi nuclear accident in the case of complementary safety evaluation (ECS) conducted in France, where the important concept of robustness—favoring design and construction solutions that are stablest under multiple loads with margins to cover unexplored fields—emerged.

It should be said that, in the case of Generation III reactors, France's "Technical Guidelines for the Design and Construction of the Next Generation of PWR Reactors" (2004) stipulate that *"the designer must also specify how it intends to prove the existence of sufficient design margins that are consistent with the general safety targets (...). Margin assessments shall be conducted to demonstrate that no cliff-edge effect in terms of radiological consequences could occur by assuming acceleration values greater than those specific to the site. The corresponding method shall take into account the actual behavior of representative equipment and of possible simultaneous equipment failures"*.

The aim of current R&D efforts that IRSN is participating in is to develop simplified numerical models able to simulate the non-linear behavior of structures and determine the seismic forces that act on equipment (transferred spectra). A number of tests

212. The high seismic area of the California coastline has been excluded indeed from the scope of the study.

213. See the SMA method described in U.S.NRC NUREG/CR-4334 (1985) as well as NUREG/CR-4482 and NUREG/CR-5076; the probabilistic PRA-based SMA method, which is also described in an U.S. NRC document; and the EPRI SMA method described in document EPRI NP-6041-SL Revision 1 (1991).

214. These reactors had been seismically reassessed for an earthquake with a PGA (Peak Ground Acceleration) of 0.3 g in the 1990s.

conducted with the AZALEE shake table have thus been used to validate the ability of these models to reproduce the measurement results and observations.

Among these tests are those recently conducted as part of the ENISTAT[215] program [3], which aimed to understand the robustness of structures built according to the Eurocode 8 standard[216] and fitted with thermal insulation systems between their vertical walls and floors (thermal break elements, which create discontinuity between the vertical walls and the floors).

The half-scale, asymmetrical reinforced-concrete mock-up with a total weight of around 40 tonnes was designed for a zero-frequency acceleration at 0.3 g.

The mock-up was subjected to horizontal seismic loading ranging from 0.1 g to 0.8 g. Although a wall failed at 0.8 g, cracks appeared at low loading. The test results made it possible to improve the simplified models to better take into account the displacements observed at the centers of the floors.

An important point emerged during the tests. When three-directional seismic loads (horizontal and vertical simultaneously) were applied, the mock-up's stability was affected by the displacements even at a low acceleration of 0.07 g. The tests were therefore continued beyond 0.07 g without any vertical quake. This observation led to IRSN to plan research on the adverse effects of the vertical component of earthquakes.

The simplified models will be validated in particular by making the most of the results of new tests conducted as part of the SMART program (Figure 10.17). These new tests have been conducted on a 1/4-scale mock-up designed in accordance with ASN Guide 2/01 for the structures in nuclear facilities.

C) Dynamic behavior of reinforced concrete structures

The Fukushima Daiichi accident brought back to the fore the issue of load combinations, in particular that of a core melt accident following an earthquake or of an aftershock following a core melt accident triggered by the main quake. Determining the total effects of both transient loads of different type and duration is complex, more particularly if the reinforced concrete structure enters the non-linear field or even becomes damaged. IRSN is developing and implementing various methods to meet this need. In particular, it is implementing simplified models—such as the lumped-mass stick model (Figure 10.18)—that are able to reproduce the overall behavior of the structures. The local behavior is subsequently determined for the critical times with a finer model of the structures by applying the load sets derived from the simplified analyses. Other innovative techniques are being developed with CEA and ENS Cachan[217].

215. Experimental and Numerical Investigation of Shear wall reinforced concrete buildings under Torsional effects using Advanced Techniques, conducted as part of the European SERIES program (2009–2013), led by Middle East Technical University (METU), in Turkey.
216. "Design and dimensioning of structures for earthquake resistance".
217. A prestigious French higher National School.

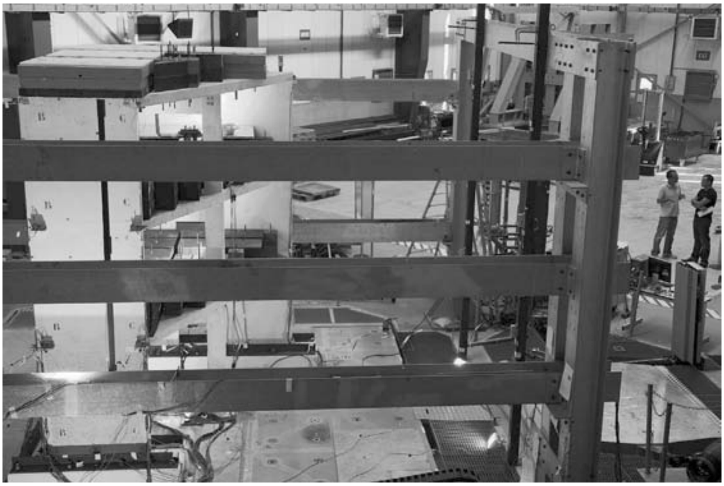

Figure 10.17 The SMART2011 instrumented mock-up (used in support of a joint research program between CEA and EDF) is representative of a 1/4-scale "nuclear" building. Weighing around 47 tonnes (including 36 tonnes of additional weights), SMART2011 was instrumented with more than 200 measurement channels (acceleration, displacement, deformation) and was subjected to a series of seven earthquakes of increasing magnitude. © P. Stroppa/CEA.

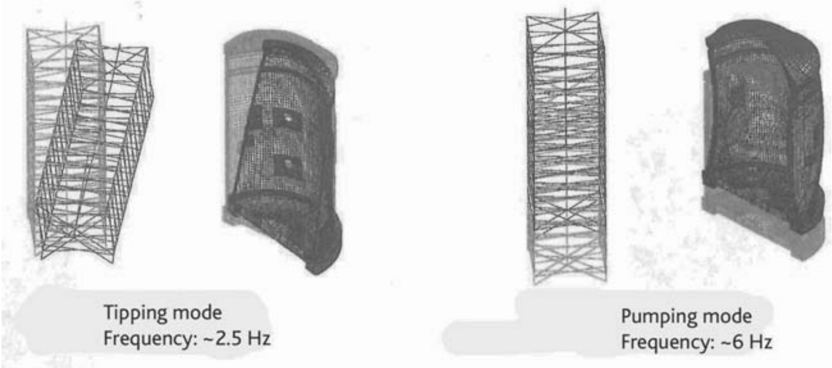

Figure 10.18 Simplified lumped-mass stick model of a containment to simulate its behavior under seismic loads: re-creation of the fundamental modes. @ IRSN.

D) Other Areas of R&D in Seismic Loading

IRSN is exploring other areas:

– seismic isolation systems. A number of safety issues have been raised during studies on the seismic isolation systems adopted for the JHR and ITER facilities (damping pads arranged underneath the buildings). Other mixed systems that combine passive isolation with active (or semi-active) control, or three-directional isolation, which is being explored by the Japanese, could also prove useful for France's NPPs. In 2013, R&D work was begun with CEA to provide knowledge with an application part of value for the expertise[218];

– the seismic behavior of nuclear structures without earthquake-resistant provisions, more particularly the floors made up of preslabs. Such floors are found in particular in the fuel cycle facilities (laboratories and plants) built between 1960 and 1990. IRSN, working with CEA, has embarked on an experimental program on this issue to study the ultimate behavior of these floors and identify their modes of collapse as a function of loading. This program assesses the relevance of checks proposed by technical regulations based on loads obtained by an analysis conducted by assuming the linear behavior. The full-scale mock-ups were thus developed by making sure replicate, as closely as possible, the building practices—in terms of concrete characteristics and implementation methods—that were in use when the facilities were built (1960–1990). These mock-ups were subjected to static tests by CEBTP and dynamic tests by CEA on the AZALEE shake table. The experimental program ended in 2013, and the results have been analyzed [4, 5]. A finite-element numerical model simulating the mechanical behavior of this assembly was used to analyze the behavior of these complex structures. The comparison of the test results with those of the simulations with finite-element models, which presupposes the monolithism and mechanical continuity of the structures, confirmed the monolithism of the behavior this complex, for the seismic loads explored, despite the absence of seismic protection provisions.

10.2.3. R&D on the behavior of civil engineering structures in cases of collision

Improving methods for assessing the impact vulnerability of nuclear civil engineering structures (following a load drop or due to a projectile) has been the focus of R&D since the beginning of France's NPPs program. The risk taken into consideration was an accidental crash by a Cessna business jet. The aircraft was assimilated with a rigid projectile colliding with the wall of a nuclear building such as a containment. This R&D work was extended to military aircraft in the late 1980s. Since the attacks of September 11, 2001, this research was further extended to assess the risks in case of a commercial wide-body aircraft impact.

These efforts can currently be split among three areas of R&D:

– an experimental area, with medium-speed collision tests (commercial aircraft) on reinforced concrete slabs as part of the IMPACT experimental program conducted at VTT in Finland. This program, which began in 2005, is being carried out in collaboration with the U.S.NRC, HSE, STUK[219], ENSI[220] and GRS;

218. A PhD thesis was started in 2014.
219. Finnish Radiation and Nuclear Safety Authority.
220. Swiss Federal Nuclear Safety Inspectorate (Eidgenössisches Nuklearsicherheitsinspektorat).

– numerical simulations conducted with fast-dynamics codes (LS-DYNA and RADIOSS). The international benchmark IRIS 2010[221] was launched and led by IRSN, under the leadership of the OECD. Its aim was to assess the ability of various codes and teams to predict the behavior of structures impacted by hard and soft projectiles. This benchmark brought together 28 teams from 11 countries. Following the findings and recommendations from this first exercise, two new benchmarks have been scheduled with OECD's partners:
  - the first took place in 2012 to calibrate the first simulations and present simplified models—by providing the partners with the test results and test conditions (actual boundary conditions), as well as the characteristics of the materials used,
  - the other began in 2014 and is being conducted to study vibration propagation in a structure. A mock-up is being built to form the experimental basis of this benchmark.

  The knowledge obtained from this area of R&D has already enabled IRSN to draw up an initial draft of recommendations for the fast-dynamics analyses conducted using numerical simulations, documented in a report by the OECD;

– characterization of impacted material, modeling, and validation. The aim of this area is to identify influencing parameters and improve the rheological model of the behavior of concrete subjected to impacts. This aspect was researched[222] with Joseph-Fourier University and CNRS and culminated, in 2013, in the confirmation of the influence of the moisture content of concrete and the rate of deformation on the dynamic behavior of impacted structures.

10.2.4. R&D on the behavior of containments during core melt accidents

A) Assessment of air and steam leaks through an idealized crack

In 1989, IPSN commissioned the Mechanical and Thermal Engineering Department (DMT) at CEA's Saclay research center to conduct an R&D program on local cracking in order to quantify leakage through a containment wall during emergency situations (air and steam) compared to dry-air leaks quantified during in-service testing. Known as SIMIBE, this program comprised tests conducted on two glass plates simulating local cracks. It resulted in the development of a numerical model integrated in the Cast3M code and able to simulate the two-phase behavior of the air-steam mixture in such cracks. SIMIBE was used to interpret the MAEVA tests described below and, in 2000, allowed IPSN to consolidate its position in relation to the air and air-steam transposition factor, i.e. use a conservative transposition factor of 1. The transposition of glass to concrete required additional research to take into account the permeability of cracked concrete and the communication between local cracks. This additional research, based on the double-porosity theory whereby exchanges occur between the two media (local cracks and the porous medium), was implemented in the ECOBA[223] project described below.

221. Improving Robustness assessment of structures Impacted by missileS.
222. Doctoral thesis.
223. Study of Reinforced Concrete Containment Structures.

B) Assessment of air and steam leaks through a crack under conditions representative of a containment

The assessment of air and steam leaks through a crack under conditions representative of a containment during emergency situations required using large-scale tests representative of such conditions. It has been conducted as part of experimental programs since the early 1980s, which include:

- the RCCV tests conducted in 1984 at Sandia National Laboratories in the United States (tests on a complete containment—made of non-prestressed concrete—of a Westinghouse 900 MWe reactor, at 1/6 scale, with a metallic liner;
- the MAEVA tests conducted in France between 1994 and 2002;
- the PCCV tests conducted at Sandia National Laboratories in 2000 on a mock-up of a Japanese PWR containment, at 1/4 scale (prestressed concrete, with steel liner);
- the VK2/2 tests conducted in 2001 at the University of Karlruhe in Germany (tests conducted on a 1.2 meter-thick pre-cracked slab extended by jacks);
- the tests in the ECOBA program (ANR project);
- the nationwide CEOS.fr project mentioned above;
- the VERCORS, also implemented by EDF.

These experimental programs were accompanied by benchmarks and workshops.

As a reminder, in single-wall containments, confinement is ensured by the steel liner (case of the RCCV and PCCV mock-ups). The goal of research programs on double-wall containments (MAEVA, VK2/2 and ECOBA) is to assess the leakage of the inner wall of prestressed reinforced concrete.

The key difference between these two types of mock-ups is the damage mechanism—a failure mode for the first and leak mode for the other.

Some of the aforementioned tests are discussed more in detail below.

► Tests conducted on the MAEVA mock-up

In 1994, EDF decided to build a containment mock-up in order to study its mechanical strength and better assess the leakage of the double-wall containments of 1300 MWe reactors. EDF's aim was to conduct an experimental study of the thermomechanical behavior of the inner prestressed concrete wall for design-basis and beyond design-basis situations (core melt) by subjecting the mock-up to sequences of increased pressure and temperature. These aims were as follows:

- assess the air and steam leakage rates under accidental conditions compared to those measured with dry air during in-service testing;
- study the behavior of the composite liners placed on the intrados of the inner wall for the in-service tests and the various accidental scenarios as well as validate their conditions of industrial implementation.

IPSN participated in the MAEVA tests, in particular by helping to define two sequences: one involving an increase in air pressure and steam (to simulate a core melt accident) up to the design-basis pressure initially intended for the EPR containment without steel liner (6.5 bar absolute), the other up to a greater pressure. IPSN also made a number of measurements (helium tracing of leakage rates, quantification of steam titers in the water-steam mixture in the core-melt simulation sequence).

The MAEVA[224] mock-up (steam and air containment mock-up) represented a straight portion of the inner wall of the containment, at 1/3 scale for the diameter and 1/1 scale for the wall thickness. The annulus space between the two walls was also represented, but the outer concrete wall on the mock-up was replaced by a steel wall (Figure 10.19). The concrete wall thus had a diameter of 16 m, a thickness of 1.2 m, and a height of 5 m. The mock-up was built at the Civaux NPP using concrete of the same characteristics as those of the concrete used for reactor No. 2 of Civaux's NPP (high-performance concrete). The top slab was supported by four prestressed concrete columns arranged at each quadrant of the surface area. The inner wall was divided into quadrants, two of which were lined with a composite material similar to that used for repairing reactors in operation.

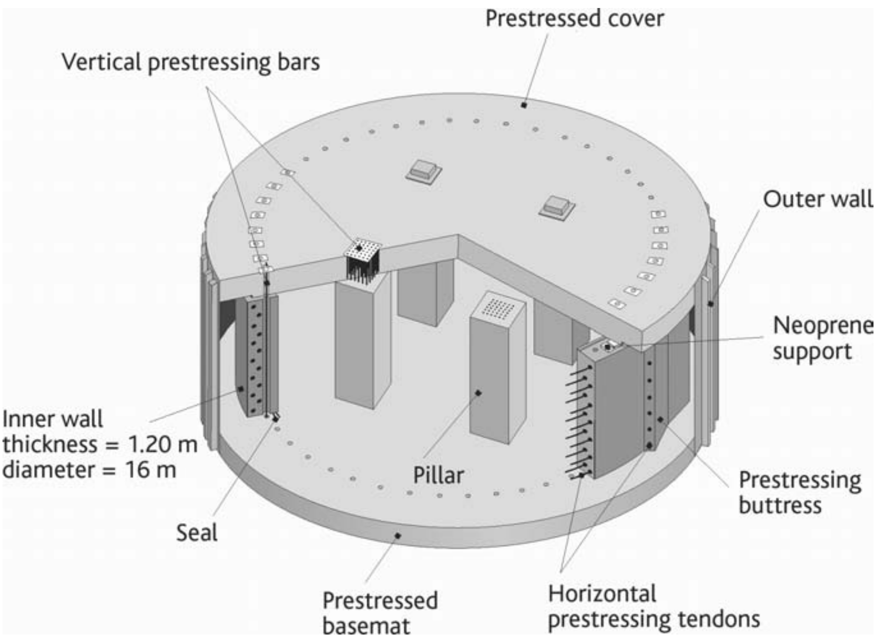

Figure 10.19 Diagram of the MAEVA mock-up. © Georges Nahas/IRSN.

224. Steam Air Exchange Model.

The prestress had been calculated to obtain, as for the containments of the 1300 MWe reactors in operation, a mean residual compression of 1 MPa for an internal pressure of 6.5 bar absolute.

Seven test sequences using air and an air-steam mixture were conducted with the mock-up. For the sequence simulating a core melt accident with air and steam, the pressure was increased in three steps (2.6 bar absolute, 5.3 bar absolute, and 6.5 bar absolute). For the air sequence conducted to study the behavior of the containments beyond their design-basis pressure, the pressure was raised to 9.8 bar absolute.

During each test, measurements were taken to determine the leaks in the annulus space—divided into four sealed quadrants (each quadrant being referred to as a chamber—for dry air and an air-steam mixture—, various temperatures and pressures, and displacements of the inner wall of the mock-up. During the air and steam tests, the extrados of the mock-up's wall was heated to 60 °C in the chambers to help vaporize the water exiting the cracks and reduce the temperature gradient in the thickness. The cracks were identified, the composite liners were auscultated visually, and pull-off strength tests of the liners were conducted. The pull-off strength test made it possible to quantify the adhesion of the liners to the concrete wall after the test sequences.

The analysis of results also made it possible to determine:

- the transposition factor to be used between the leakage rate for dry air and the leakage rate an air-steam mixture (factor of 1). This factor is then used to determine the leakage rates of the actual containments during accidental situations from the leakage rates measured during the tests;
- changes in cracking and measured leaks for a number of accidental scenarios.

These last results on the changes in cracking and measured leaks were applied to validate the leakage quantification method (nationwide CEOS.fr project and the ECOBA project).

Furthermore, the measures taken during the air and steam sequences showed that the leakage rates collected in the chambers were quite high and greater than predicted by EDF, which believed that the steam would condense in the cracks and remain in the reinforced concrete wall. This assumption was therefore invalidated.

These observations accounted for much in considerations on the robustness of the EPR. Realizing that composite liners would not ensure prolonged integrity during core melt conditions—a design basis of the EPR—EDF decided to install a steel liner on the intrados of the containment of the Flamanville 3 reactor.

► ECOBA Project

The ECOBA research project to study the confinement properties of reinforced concrete structures (2010–2014) was conducted by the French National Research Agency (ANR) as part of its "white" program (interdisciplinary projects in all fields of research). ECOBA brought together IRSN and three academic laboratories with complementary skills: GeM[225], the ENS Cachan, and the University of Pau and Pays de l'Adour (UPPA). Using a full-scale mock-up representative of an inner containment wall of a 1300 MWe (P4 series) reactor, its aim was to study the various mechanisms of concrete cracking in order to establish relations between damage, cracking, and permeability. It was based on tests conducted at the GeM laboratory of the École Centrale de Nantes on two mock-ups representative of a straight portion, referred to as the "effective area" (1.50 m wide, 1.20 m high and 0.90 m thick) of an inner wall of the containment. Each mock-up, which measured 3.90 m wide and 2.40 m high (overall dimensions) and weighed 20 tonnes, was built at the experimental laboratory of the École Centrale de Nantes. Both were instrumented with vibrating-wire strain gauges identical to those used in the actual containments and supplemented by various measurement devices to validate the results and obtain a robustness of the measurements. The mock-ups were subjected to direct tensile stress —four jacks driven load and displacement—to create cracks representative of those that could appear in a containment during accidental loading. A metallic box for injecting either air or air and steam was fitted upstream of the areas of cracking. The leakage rates through the cracks were measured using a gas tracer dilution technique developed by IRSN and which made it possible to identify the surface openings of the cracks.

The first round of tests made it possible to observe the development of cracks in the effective area—which were similar to those seen on actual structures subjected to these types of loads—and quantify the leakage rates in the sequences with air injection. The first set of results confirmed the project's relevance to progress in characterizing the parameters that affect the integrity of NPP containments and to consolidate the assessments conducted on the confinement capability of the containments in the event of core melt accidents.

► VERCORS Project

In 2013, EDF launched a far-reaching project of studies and R&D on reactor containments. Known as VERCORS (for *Vérification réaliste du confinement des réacteurs*), its aim is to acquire sufficient knowledge in order to demonstrate that NPPs can be safely operated for a period of 60 years. VERCORS is expected to continue until 2021. Its goals are as follows:

- experimentally demonstrate the strength of the containments in core melt situations (under simultaneous and maintained pressure and temperature loads);
- predict by experience changes in containment leaktightness;
- improve knowledge about leakage and the models used to predict their changes;
- find new tools of leak detection and quantification.

A number of tests will be conducted on a large-scale and thoroughly instrumented containment mock-up. In light of the feedback obtained from tests conducted on large-scale mock-ups, such as MAEVA or in Sandia National Laboratories, EDF decided to build

225. French Civil and Mechanical Engineering Research Institute. A joint research unit brings together the French National Center for Scientific Research (CNRS), the École Centrale de Nantes, and the University of Nantes.

a mock-up of a double-wall containment (P'4 series 1300 MWe reactors) at the same scale for the large dimensions and thickness (1/3). This mock-up will be built on hard soil and feature the same discontinuities (such as the ledge used for the crane runway), penetrations (in particular the equipment hatch), prestressing cables and rebars. The 1/3 scale will make it possible to accelerate some physical phenomena inherent in concrete, in particular:

- shrinkage/curing of a factor of 9,
- creep of a factor of 3,

this gives an equivalent aging factor of 7 on average.

At this scale, around nine years will suffice to simulate 60 years for the actual containments.

The ten-yearly tests will thus be conducted on the mock-up once every 14 months. The core melt accident will be simulated after the first nine years.

Because the delayed strains (shrinkage and creep) in the concrete that will be used are moderate, the extrapolation of the results will have to be examined closely in the case of the susceptible containments, for which delayed strains are greater.

C) Assessment of non-localized leaks in a concrete wall

The results of the experimental programs on the large-scale mock-ups together with the observations and the measurements of the leakage rates during the ten-yearly tests of the containments reveal the existence of non-local leaks—referred to as non-localized leaks—that contribute significantly to total leaks.

Each PWR containment in the France's NPPs has an inner surface measuring around 10,000 m^2. As a result, non-localized leaks through these containments as a function of the applied load are likely to become significant. Research, which ended in 2011, was conducted with the École Centrale de Nantes on the gas (nitrogen) and steam permeability of concrete under compressive mechanical loads. The experimental setup made it possible to measure the air-steam mixture permeability of 11 × 22 cm hollow concrete specimens. The concrete formulation was representative of that of the concrete of the containments themselves. The results revealed a difference in flow between steam and nitrogen and made it possible to identify a difference in the flow times of the gas and steam. Nitrogen flows through the accessible porous network in a matter of minutes, whereas steam takes much longer (around 10 to 25 hours depending on the injection pressure).

This observation partially explains the time difference between the air and air-steam leaks observed during the MAEVA tests.

D) Study on the early-age behavior of concrete

Early-age behavior of concrete is an important topic. During this complex phase, which spans from the setting of the concrete to around one month thereafter, exothermic physicochemical changes followed by shrinkage and creep phenomena create weak areas in the concrete's structure. Once concrete shrinks (Le Châtelier effect), the combined changes in the temperature gradients throughout the structure and the mechanical characteristics of the concrete (Young's modulus, tensile strength limit, compressive strength limit, etc.) cause the concrete to crack. The extent of these cracks depends on parameters such as the type of formwork, the outdoor temperature, the geometry of the structure, and the amount of rebars.

These cracks will affect both the durability and mechanical behavior of the structure in the event of accidental loading with propagation of these cracks. In addition, the physical and chemical changes may, under certain conditions, lead to subsequent pathologies such as delayed ettringite formation (DEF – see below).

IRSN began researching this aspect in 2007 in order to study the behavior of civil engineering structures from their construction (concrete pouring) onward. Concrete hydration reactions, with the resulting temperature gradients, water exchanges and the incompatibility of stresses of the constituents as well as the restraining caused by resumption of concreting, which causes concrete to crack, were identified as areas of research to be developed.

The first area looked at was the effect of early-age cracking of concrete on its air and gas permeability. This effect was analyzed by taking into account the rate of cooling, the rebars, and resumption of concreting. This research was begun in 2007 (PhD thesis) with the ENS Cachan. The experimental part of this thesis comprised tests on a 10 × 10 cm concrete ring representative of that in the containments and heated to a temperature representative of that measured at the core of the early-age concrete. It yielded important data on the influence of construction methods on the creation of the first structural cracks. IRSN used the results obtained to assess the method used by EDF to resume concreting of the Flamanville EPR containment.

This research was followed by another thesis (2011–2014) on the modeling of the aforementioned phenomena supplemented by shrinkage and creep, which also occur during the early-age phase. This research is currently being applied in the numerical simulation benchmark of the VERCORS mock-up.

In addition, the quantification of early-age exothermic phenomena also makes it possible to assess the temperatures reached during pouring of fresh concrete and the risk of onset of pathologies, in particular DEF.

E) Development of mesoscopic simulation models

Analysis of the results of the observations and measurements obtained from the experimental programs and ten-yearly tests of the containments show the importance of characterizing the geometry of local cracks and porosity (corresponding to non-localized leaks) as a prerequisite to realistic assessments of leakage rates. This characterization requires using mesoscopic models—i.e. models at the scale of cracking—to be able to simulate crack openings of less than 1 mm in concrete. High-performance simulation codes are required to model at the mesoscopic scale. Research conducted by IRSN, working with the University of Pau and Pays de l'Adour, have made it possible to identify important parameters, in particular initial structural stresses and their effect on the creation of cracks in concrete during accident conditions, communication between microcracks and local cracks, and residual opening of cracks after unloading. These data may be applied to the macroscopic models used by IRSN for its studies in support of the level-2 PSAs on the behavior of double-wall containments of France's NPPs, a particularly important point for the safety reviews associated with the third ten-yearly outages of 1300 MWe reactors.

10.2.5. R&D on containment aging

Aging is the process of changes over time to reinforced concrete structures such as reactor containments. Aging is caused by two phenomena: a natural phenomenon driven by delayed strains in concrete (shrinkage and creep) and a phenomenon caused by the onset of pathologies.

A) Delayed strains in concrete

In the case of containments, this aging process decreases structural strains, causing the prestressing cables to lose tension and thus decreasing the compression in the containment walls. Because this compression makes it possible ensure the necessary level of confinement required for these walls, its changes must be controlled throughout the life of the structure.

R&D on this topic are thus conducted to assess the delayed behavior of the reactors containments and its consequences on the ability to confine radioactive material in the event of an accident. This research comprises theoretical parts and experimental parts, with numerical simulations. Tests are performed on concrete specimens representative of the concrete of the containments. The main purpose of this work is to obtain the scientific knowledge and skills needed to make decisions on the risks caused by changes to delayed strains in concrete. Research is primarily conducted on the following topics:

- **Tensile creep of concrete and its effects on containment leaktightness**: research was started in 2005 with IFSTTAR and the ENS Cachan. Tests were performed on cylindrical hollow specimens to study the effects of creep on concrete leaktightness. The tensile-loading time was limited to 36 hours (in-service testing time of the containments). A numerical model was developed to simulate the mechanical behavior of the containments. The data acquired during this research made it possible to better understand the changes in cracks observed during the ten-yearly tests and the progression of cracking during consecutive pressure-hold steps.
- **Multiaxial creep of concrete and its effect on leaktightness**: the containments are biaxially prestressed structures. To take into account the effects of delayed strain (concrete creep and shrinkage), these structures were designed in accordance with the RCC-G[226] and using formulae in French standards and codes. However, feedback has shown that these formulae underestimate the effect of these strains for multiaxial loads, a fact borne out by numerical simulations

226. Rules on the Design and Construction of Civil Engineering Structures.

conducted by IRSN. The reason is that these formulae were established from observations and tests on primarily uniaxially loaded structures. Furthermore, monitoring activities conducted by EDF on the containments of its NPPs are not readily usable in the calculation models because the measured strains are due to a combination of phenomena (basic creep, drying creep, drying shrinkage, autogenous shrinkage). Research to quantify these phenomena was begun in October 2011 with the ENS Cachan. This quantification will make it possible to better simulate the delayed behavior of the containments, assess the solutions proposed by EDF to reduce the kinetic of these strains (such as wetting of the containment facings), and predict containment behavior to extend the operation life of reactors. Once data has been acquired from the other research, the final application will be to simulate a containment from its construction lift by lift with the prestressing phase, delayed behavior, ten-yearly tests, and end of operation. This model will make it possible to simulate the changes over time in the confinement capability of the containment, a decisive factor in decisions on whether to extend the service life of reactors.

B) Pathologies of reinforced concrete structures

In order to take into account all the phenomena that may alter the safety functions of nuclear facilities, and of containments in particular, IRSN deemed it essential to initiate research programs to investigate concrete and rebar pathologies. One reason in particular for this research is that EDF is seeking to extend the service life of France's NPPs to 60 years.

The development of pathologies in reinforced concrete structures can lead to damage that could affect the mechanical properties of the concrete and even undermine the confinement capability required from some structures. Swelling reactions—such as alkali-aggregate reaction (AAR) and delayed ettringite formation (DEF) in particular—are two examples of potentially harmful pathologies.

EDF conducted research on AAR (a chemical reaction between some types of aggregate and the cement matrix) and its change over time into an expansive gel and pop-outs around reactive aggregate. EDF has classified all of the containments in its fleet into five categories (from 0 to 4 and ranging from zero risk to very high risk) according to the characteristics of the concrete used and the environment of the structures.

DEF—which is related to the type of concrete, its environment, and its core temperature during concreting—is identified as a risk for nuclear structures. DEF can be triggered by temperatures above 65 °C during construction of structures as well as by certain moisture conditions. In 2009, IRSN alerted EDF (with ASN) about the importance of such a risk of DEF. This alert was based on feedback from research programs conducted by reconstituted concrete and led by IRSN in collaboration with IFSTTAR (Figure 10.20).

The temperature of concrete play an even greater role in the development of DEF, in particular in the case of structures which may, during their operation, be taken to temperatures of around 80 °C followed by a cooling cycle. Recent research conducted by

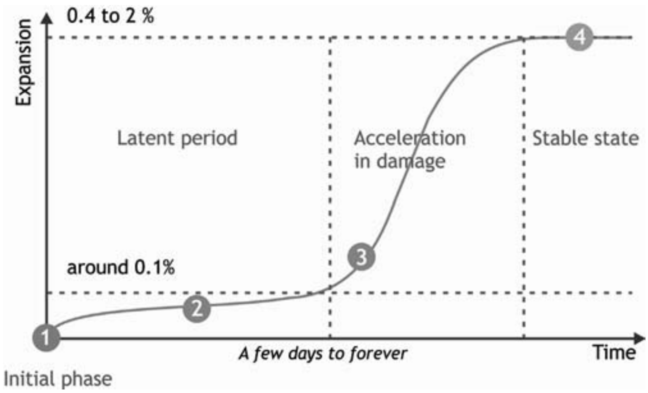

Figure 10.20 S-curve of the change in DEF proposed by X. Bruneteaud in 2005 [7]. © Georges Goué/ IRSN–Source: Xavier Bruneteaud.

the scientific community and verified by IRSN (EDF-funded tests performed on specimens at IFSTTAR [6]) have confirmed that delayed DEF can occur under some conditions even though the concrete has not been subjected to temperatures above 65 °C when it was made. Such a configuration could occur in the containments at the reactor vessel and the steam-tube penetrations, some fuel-assembly storage facilities, or after a controlled fire. IRSN is assessing the opportunity of conducting a research program on this subject.

Another parameter, related to the type of cement, affecting the development of DEF is the use of limestone filler[227] as additives. Research started in 2009 (PhD thesis) in collaboration with IFSTTAR and the ENS Cachan made it possible to study the effects of these additives as well as other parameters that influence the development of this pathology, such as the moisture content of the outdoor environment, concrete permeability, and their effects on reaction kinetics.

EDF has classified all of the containments in its fleet according to the characteristics of the concrete used and the environment of the structures. The ranking of these five categories, which range from zero DEF risk to very high DEF risk, is similar to that used for the AAR.

An analysis of both maps of the potential risk of containment pathology, derived from the classifications, reveals cases where this risk is high for both pathologies. These cases mean that R&D work must be conducted until feedback or experimental results are available. In 2014, IRSN—working with IFSTTAR and the ENS Cachan—began research (PhD thesis) on the accelerated aging of concrete affected by both pathologies together and separately.

In the current state of knowledge, there are no curative solutions for these swelling pathologies. The only recommendations are to monitor structures and, in the case of structures with high and very high risks (see, for example [8] and [9]), limit the ingress of water from external sources.

227. Dust from the cutting of limestone that is added to concrete and other building materials.

C) Predicting the effects of aging: the ODOBA project

In light of the importance pathologies have on the longevity of reinforced concrete structures and the plans to extend the service life of France's NPPs, IRSN decided the time was right to create an observatory on the longevity of nuclear civil engineering structures. This observatory will be shared with the scientific community and will yield data on the aging of these structures and the defects that could affect them.

IRSN therefore created the Observatory of the Durability of Reinforced Concrete Structures (ODOBA[228]) in 2014. ODOBA's purpose is to study pathologies found in nuclear civil engineering structures, such as rebar corrosion (due to chloride ions or carbonation), concrete swelling (DEF, AAR) and leaching, and their effects on the safety requirements for these structures.

Knowledge will be acquired concomitantly with the assessment of plans to extend the service life of France's NPPs beyond 40 years. The experimental part comprises the construction, at IRSN's Cadarache research center, of concrete structures representative of the concrete of the containments at France's NPPs. These structures—60 blocks measuring 1 m thick and several meters in height and width—will be subjected to either an accelerated aging process or the natural aging process in order to determine the equivalent durations of accelerated aging. A first difficult task was finding the quarries and cement used to build a few containments classified by EDF as having potentially high or very high risks of swelling reactions. This work was conducted with the assistance of IFSTTAR and consisted in investigating the archives of the French Central Laboratory for Bridges and Roads (LCPC). This investigation made it possible to locate nine quarries and their cement for 13 susceptible NPPs. The ODOBA project will use equivalent generic concrete for the other susceptible NPPs.

The scientific part of ODOBA is led by a scientific committee comprising IRSN, the ENS Cachan, IFSTTAR, the Construction Materials and Durability Laboratory (LMDC) in Toulouse and the Mechanics and Acoustics Laboratory (LMA) in Aix-en-Provence. The purpose of this committee is to monitor the project's scientific progress and verify the scientific relevance of the choices made as part of it.

ODOBA is divided into a number of stages that will make it possible to obtain intermediate results aligned with IRSN's assessment needs, such as analysis of the ten-year safety reviews conducted on France's NPPs.

In 2015, IRSN signed a cooperation agreement with the U.S.NRC to share the results of experiments on the pathologies of the concrete used in nuclear reactor containments. Other organizations are interested in ODOBA and discussions are under way to define the conditions of their involvement.

The first concrete blocks have been cast in 2016.

228. Monitoring Center for the Durability of Reinforced Concrete Structures.

D) Non-destructive testing of structures

Non-destructive methods of testing for defects in structures are useful tools for assessing the health of susceptible structures such as reactor containments and structures were sampling is very limited and controlled. IPSN then IRSN conducted R&D efforts with CEA's Saclay research center to study the possibilities offered by ultrasonic waves. These inspection methods have proven their effectiveness for metallic structures and make it possible to detect defects. A forward-looking development project conducted with CEA's Saclay research center consisted in adapting these methods to concrete by studying the various types of probes and the signal frequency to be emitted, and by adapting the parameters of the CIVA simulation platform and its simulation code for predicting noise and attenuation phenomena. Currently, the operational monitoring capability of this method is limited to a thickness of 40 cm and a very low amount of rebars. The results obtained by this feasibility study have shown that this method cannot be used on the containments.

In the frame of the ODOBA project, the need for non-destructive testing methods meant that other techniques and methods had to be found. These include the use of fiber optics and novel non-destructive testing techniques studied by the LMA, the LMDC and IFSTTAR. These new techniques are being developed to detect the onset of defects and pathologies in the concrete of structures. They will be validated by collecting core samples from the blocks.

10.3. Research on polymers

The main purpose of research on polymer aging (polymers are used for electrical wires, seals, and coatings) is to study the effects of radiation and temperature on the degradation of polymer properties.

For licensees, electrical wiring, which is located inside the reactor building and provides a safety function, is not easily replaceable. It must therefore perform its function throughout the life of a NPP, including in the event of an accident at the end of the facility's service life.

The insulation and jackets of wiring are made of polymers. Temperatures and irradiation can, depending on their intensity and synergies, break up the polymer chains or lead to crosslinking (creation of bonds between the chains), oxidation, or even a loss of plasticizers due to hydrochloric acid migration. Understanding these mechanisms is necessary in order to assess the relevance of the conditions of accelerated aging used during qualification or during accelerated aging simulations. This is because the electrical properties directly related to the functionality of the wires and cables generally do not vary much before significant degradation of the polymers occurs. Indicators of aging must therefore be looked for among the mechanical and physical and chemical properties of polymers.

The service life of wires and cables can then be predicted using experimental data and extrapolation models.

The purpose of a first study initiated by IRSN was to assess the aging resistance of EVA polymer cables used in the N4 series reactors. Sections of cable were aged thermally then by irradiation at dose rates between 3 and 1000 Gy/h. They were then subjected to the accident conditions that would result from rupture of a RCS pipe. The first results showed that EVA is highly stable, virtually immune to temperatures and the dose rate provided that the antioxidants in the material have not been completely consumed.

Furthermore, in order to determine the representativeness of the conditions of accelerated aging, new PVC and EPR/Hypalon[229] cables were laboratory aged using an accelerated method. To define these tests, IRSN assessed the acceleration of aging so as not to alter the aging mechanism foreseeable under actual conditions; this assessment was conducted using a similitude of the experimentally determined activation energies of the mechanisms of degradation of the materials. The mechanical properties of these aged cables were compared to those of identical cables collected from the Cruas site after seven years of operation. This comparison revealed that the properties of the cables collected from Cruas and the laboratory-aged cables were similar.

IRSN thus demonstrated that it is possible, after a study of the degradation phenomena of materials, to establish accelerated aging conditions that are representative of actual aging.

Since 2014, IRSN has been conducting a research program to study the impact of an accident on aged silicone seals of the equipment hatch. The IRMA irradiator and the EPICUR facility in particular are being used for this research. Experiments conducted on mock-ups to assess the temperature (40 °C to 150 °C) and irradiation (25 kGy to 75 kGy) effects show that the properties of the silicone are greatly affected by these exposure conditions. All the tests under LOCA conditions, after ageing, have been conducted to date; the results will be synthetized in 2017.

Accelerated aging tests of 3–4 mm coatings made of epoxy-matrix composites were conducted by CIS-bio international (CEA–Labra[230]) for IRSN between 2003 and 2008. Two epoxy resins (manufacturers: Max Perles and Chrysor) similar to those used by EDF to line the intrados of the containments of its 1300 MWe and 1450 MWe reactors were used for these tests. A variety of dose rates (1 to 100 Gy/h), doses (40 and 320 kGy) and temperatures (40 °C to 70 °C) were applied. The results showed that the epoxy resins were highly susceptible to oxidation. The adhesion of all of the coatings dropped significantly after the four postulated LOCA tests. A critical analysis of this program in relation to the tests presented by EDF in 2014 is under way. A new R&D program will be proposed in 2017 to deepen understanding on the degradation mechanisms identified and their impact on integrity.

New research on polymer aging (cable insulation, paint, resins, and anti-seismic damping pads) is being scheduled.

229. Insulation made of cross-linked ethylene/propylene with a Hypalon jacket.
230. Laboratory for Applied Radiation.

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