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The aging of nuclear power plants

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Nuclear power plant equipment is expressly designed, built and operated to ensure that its expected aging process – damage gradually sustained over time due to operating loads (pressure, heat transients, vibrations, etc.) and the service environment (temperature, irradiation, NSSS chemistry) – will not impair compliance with safety criteria, even in accident situations. Operating experience, and specifically the results of periodic tests and inspections, have nonetheless revealed unforeseen forms of degradation, among them cracks in vessel head adapters and the steam generator tube bundle, embrittlement of austenitic-ferritic steel castings used in the reactor coolant system and changes in mechanical properties of elastomers. In still other cases, aging damage such as RPV irradiation embrittlement, thermal fatigue of piping and prestressing losses in containment concrete has been shown to occur faster than anticipated.

What's more, with the exception of isolated cases like vessel embrittlement surveillance, which uses actual specimens of irradiated metal, in-service monitoring cannot provide a true picture of material degradation. Predictions are therefore needed, and these must also be validated to enable allowance for aging in evaluating the conditions of materials and their behavior under postulated accident loadings.

It is the operator's duty to carry out whatever programs are required to anticipate aging effects likely to jeopardize plant safety or cause difficult-to-manage generic repair problems in France's standardized NPP population. IRSN itself is conducting aging studies, some of which are described in this article, to support its expert assessment activity and corroborate and perfect results obtained by the operator. In some instances (direct, *in situ* aging measurement), possible new processes or methods are explored to test their possibilities.

To determine its R&D focus, IRSN must first investigate the possible impact of aging on components and systems. This investigation then serves to identify developments necessary both to enhance understanding of aging phenomena and to make available suitable means for detecting them and, if appropriate, repairing concomitant damage. The second step is to consider any aging effects that underlie abnormal conditions detected in NPPs worldwide. A "cross-cutting"

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approach to operating feedback – is premature aging of one component likely to repeat itself in another made of similar material? – means improved forecasts and better targeting of exploratory research.

Finally, two points are especially vital to defining IRSN's R&D programs. They are

- specific features of French PWR facilities in comparison with the world's other NPPs:
  - materials (e.g. thermal aging of austenitic ferritic steel components with molybdenum adjuncts);
  - design characteristics (leaktightness of double-wall concrete containments);
  - loadings (wear and fatigue phenomena resulting from special operating modes such as load follow), and
- the advent of new technologies already applicable or potentially suited to repairs or upgrades and requiring new safety assessment know-how and tools (e.g. the digital I & C now replacing outdated electromechanical systems).

After describing the work conducted by IRSN to enhance crucial testing/inspection expertise for aging detection and in-service surveillance, the following pages also present research aimed at better control of aging-induced degradation (in steels, concrete, electrical cables) and assessment of new technologies (digital I & C). This is concluded by an overview of international partnerships and the future orientations of aging studies.

Nondestructive inspections

IRSN considers it essential for in-service inspection to progress in two directions: better appraisal of characterization methods for aging effects such as cracking, corrosion and, in general, the internal condition of a structure; and devising new inspection methods for configurations not accessible to current tools. Existing inspection methods are very often inaccurate or inoperative for particular component profiles (bends, conical parts, uneven surfaces or weld beads) or defects (whose orientations may not be conducive to correct detection and identification).

In the field of ultrasonic inspection, IRSN has initiated the development of new transducer technologies in order to demonstrate that there is room for progress. Related actions already completed or still underway with CEA/STA in Saclay center mainly on systems with "multielement" ultrasonic transducers, and specifically on:

- smart contact transducers – essentially for enhanced inspection of components with complex shapes – whereby the ultrasonic field adapts itself to such shapes (**figure 1**);
- tools that simulate the ultrasonic field and can optimize emitter control and predict inspection performance, while avoiding systematic use of time-consuming, expensive mockups (**figure 2**);
- the FAUST ("focusing adaptive ultrasonic tomography") system, comprising a prototype acquisition channel for focusing the ultrasonic fields of immersion-type phased array transducers.

Figure 1

Prototype of an adaptive contact transducer.

In the area of "eddy current technology", IRSN has concentrated its efforts on enhancing the efficiency of and otherwise improving SG tube bundle inspections.
Eddy current inspection is being used more and more to detect the increasing numbers of circumferential cracks initiated on tube outer surfaces.
The significance of such degradation in terms of safety (risk of guillotine break) justifies an especially fine evaluation of the nondestructive inspection tool used.
To achieve this, IRSN, in conjunction with various CEA departments and outside suppliers, has initiated a crack generation and inspection study aimed at assessing the performance of eddy current probes currently deployed at NPPs in France and other countries. The same applies to the subtle energy transducer (SET) developed on its behalf by CEA (**figure 3**).

Figure 2

Comparison of transducer responses during tests/simulations.

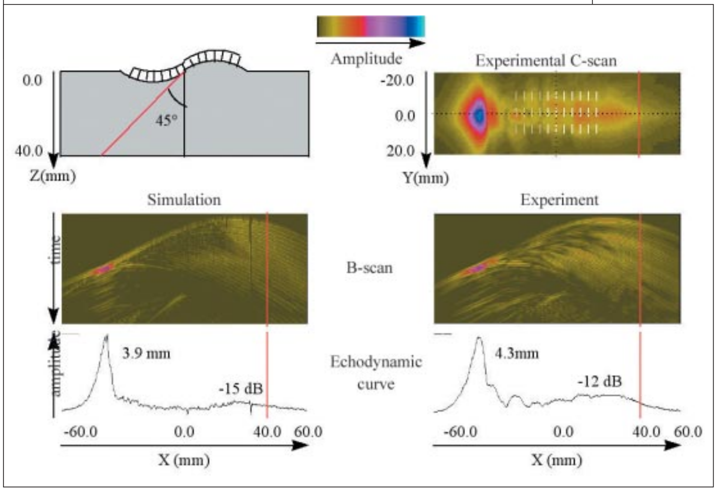
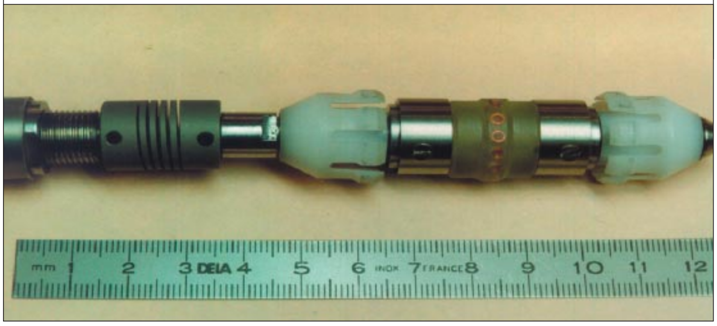


Figure 3

SET developed by CEA for IRSN.



Embrittlement of reactor coolant piping

In French PWRs, some parts of the reactor coolant system (RCS) are made of cast austenitic-ferritic stainless steel. In the early 80s, thermal aging was observed in such castings after operation for long periods at RCS service pressure.
The resulting deterioration in the steel's mechanical properties (gradual hardening and embrittlement) was attributed to demixing of chromium by precipitation or by spinodal decomposition and precipitation in ferrite of a nickel- and silicon-rich intermetallic phase.
Under such conditions, there is risk of change in the mode of component fracture from ductile to brittle, that is, in this case, to "cleavage fracture", a low-energy phenomenon that, because it occurs suddenly, is seen as a particular threat to pressure vessels. IRSN has performed tests and studies on model materials to explore structural changes induced by extended service at temperature and to assess the impact of the changes.
This work has enhanced understanding of fracture mechanisms in the aged metals. The ferrite phase that is hardened by chromium demixing undergoes cleavage at the onset of plastic deformation, thus causing cavities to form in the metal. Sensitivity to cleavage is determined by ferrite and austenite orientation with respect to loading. Austenite fractures result from shear or tearing between cavities. IRSN research has thus established that the fracture mechanism here, which is macroscopically perceived as a brittle one, is in fact a ductile behavior that is less dangerous because it requires higher energy input.
These studies also showed that damage is not uniform and that the response of a material depends on the volume and zone being challenged. There is thus a scale effect that merits close investigation, since tests are usually performed on small piping specimens whose size is limited by the excess metal available for *in situ* sampling. IRSN thus decided to test larger, CT20 specimens as well as various other types of

Damage mechanisms have been characterized as a function of ferrite content and duration of aging.

notched specimens. The ensuing results confirmed those already obtained with the smaller series.

The Institute has also developed and applied numerical models for assessing behavior of aged metals with as-fabricated (e.g. casting) defects or defects postulated for safety analysis (cracking).

Damage mechanisms have been characterized as a function of ferrite content and duration of aging. This entailed observing tensile test specimens as they underwent deformation, both shortly before and after rupture. A fracture mechanics model based on the "local approach" was then developed and applied to calculation of crack initiation and growth in the specimens. Such numerical simulations suitably reflect the toughness dispersion of the material as well as its crack growth resistance.

Finally, to eliminate uncertainties due to inadequate laboratory data when predicting the extended equipment service scenarios required by currently anticipated plant lifetimes (200,000 to 300,000 hours), IRSN has investigated suitable nondestructive means for *in situ* measurement of the aging process. This has meant evaluating techniques that measure aging-induced changes in the electrical or magnetic properties of materials. A device based on measurement of thermoelectrical power was ultimately selected for industrial-scale production and was then installed by the PWR operator.

Irradiation-assisted stress corrosion cracking of reactor vessel internals

The complex processes involved in the *Irradiation-Assisted Stress Corrosion Cracking* (IASCC) of internal structures of pressurized and boiling water reactors are still poorly understood. IASCC has been the subject in different countries of numerous studies geared to improving knowledge of irradiation impact on:

- material properties and how they vary locally at grain boundaries;
- chemistry and local corrosiveness of fluids;
- composition of more or less protective oxide films, notably at grain boundaries;
- thermal stresses and those related to swelling of structures.

Component degradation caused by IASCC is a perfect example of the type of aging process that requires advance consideration, since it can raise end-of-life problems in PWRs. To optimize use of IRSN assessment expertise in an area requiring both complex action and large capital outlays, IRSN decided, on the strength of its own experience in appraising damaged components from the Chooz A PWR plant, to participate actively in the CIR program coordinated by a group of international experts, including representatives of other safety agencies such as the NRC (US Nuclear Regulatory Commission) and SKI (Swedish safety authority). The main orientations of its current actions are:

- detailed crack front studies (grain boundary segregations, characterization of oxide compositions and status of grain boundary oxidation) for bolts and other components removed from the plants (Tihange 1, Oyster Creek, etc.);
- testing and parametric evaluations of specimen materials irradiated with proton beams;
- validating the representativeness of proton beam irradiation compared to neutron flux-induced degradation, particularly at high fluence levels;
- in-depth evaluation of passivation under irradiation and characteristics of products generated by oxidation or corrosion of materials and grain boundaries.

Loss of containment leaktightness

1300 and 1450 MWe plant units have a double-wall concrete containment with a common basemat and a system for collecting and filtering leakage from the annulus between the walls. This design, in which "third barrier" leaktightness is afforded by the double containment without need for a metal liner, is unique to French plants.

The inner containment wall, built of prestressed concrete, is designed to withstand the complete, instantaneous severance of a large-diameter reactor coolant pipe. Such a break would cause pressure and temperature increase in the containment; the latter is therefore prestressed at the time of construction to avoid tensioning of the concrete at maximum accident pressure.

However, the cumulative effects of shrinkage and creep (slow, gradual deformation of the structure subjected to the permanent prestressing force) have been shown to reduce the level of concrete compression faster than expected by designers, thus jeopardizing containment leaktightness under accident conditions.

As part of its studies on this subject, IRSN has developed models for predicting leak rates through cracks in the containment walls. These models are adjusted to credit leakage test results (**figure 4**) and feedback from field measurements made during periodic containment tests at 1300 and 1450 MWe plant units. The objective of this action is to ultimately develop a complete method for prediction of containment leakage behavior, based on deferred deformation and its associated cracking.

IRSN also participates in mockup tests at the MAEVA facility (**figure 5**) built by EDF at the Civaux plant site. This mockup permits large-scale simulation of double-wall containment behavior in the event of a LOCA (loss of coolant accident). Through the MAEVA test series, IRSN has also been able to develop and test an improved method for containment leakage distribution measurement, using a tracer gas.

Figure 4

Simibe validation tests for modeling leakage through a crack.

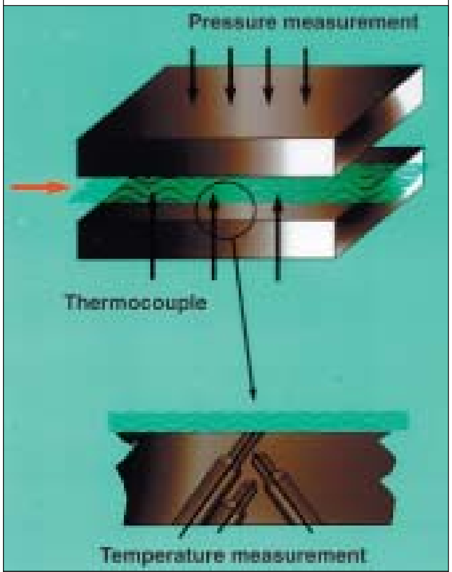


Figure 5

MAEVA mockup.

Electrical cable aging

From the operator's standpoint, electrical cables performing a safety function inside the reactor building are too hard to replace. Such cables must therefore be able to perform their function throughout the life of the plant, and this also includes end-of-life accident situations.

Insulators and cable sheathing are made of polymers. Temperature and irradiation may, at certain levels, with certain synergy effects, cause breaks in polymer chains, along with crosslinking (formation of chemical links between chains), oxidation or even loss of plasticizers by hydrochloric acid migration. Understanding these mechanisms is a prerequisite for determining the suitability of accelerated aging conditions used for cable qualification or aging simulation tests. This is because there is little change in the electrical properties directly responsible for cable functionality unless polymers have deteriorated significantly. Aging indicators must therefore be found among the mechanical and physico-chemical properties of these materials. Once these factors have been identified, cable life predictions can be made on the basis of experimental data and extrapolation by mathematical models.

A first study to this effect was initiated by IRSN to assess aging behavior of EVA polymer cables used in N4 model PWR units.

Sections of cable were thermally aged, then irradiated at dose rates ranging from 3 to 1000 Gy/h; they were subsequently exposed to accident conditions illustrative of a reactor coolant system pipe break (**figures 6 and 7**). Initial results showed EVA to be a well-stabilized material exhibiting little sensitivity to temperature or dose rate as long as its antioxidants had not been fully destroyed.

To test the representativeness of aging acceleration conditions, a set of new PVC cables and EPR(XLEP)/Hypalon cables (insulated with crosslinked ethylene/propylene and jacketed with Hypalon) were subjected to accelerated aging in a laboratory. In planning these tests, IRSN had evaluated the acceleration process to avoid modifying the aging mechanism likely to prevail under actual plant conditions. This evaluation was based on similarities between experimentally determined activation energies for the degradation mechanisms occurring in the materials.

The mechanical properties of cables aged in this way were compared to those of identical cables sampled from the Cruas plant after seven years of operation. Comparisons revealed that the properties of laboratory-aged cables were similar to those that had aged in service.

IRSN thus demonstrated that, subject to careful prior study of material degradation phenomena, it is possible to simulate accelerated aging con-

Figure 6

Irradiation aging of cables in a temperature-controlled chamber.

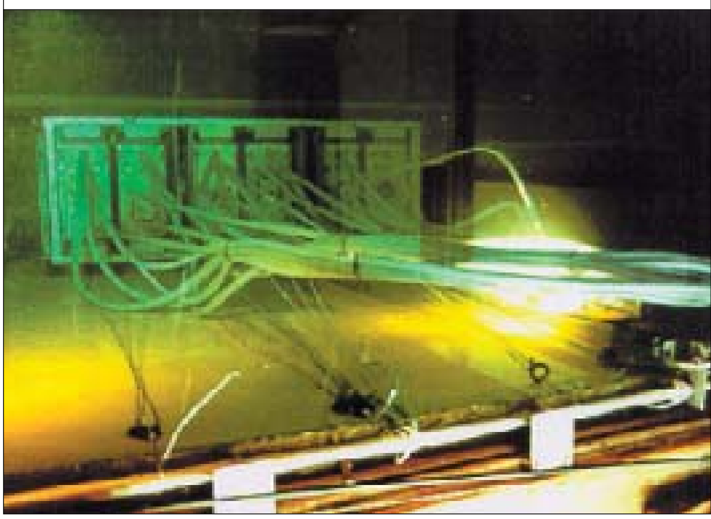


Figure 7

Accident simulation device used in cable testing.

ditions illustrative of real life aging. Finally, IRSN programs investigate the correlation between the mechanical properties and the oxidation of EPR/Hypalon-type polymers and their halogen-free (e.g. EVA) counterparts. However, use of these results to develop a cable aging inspection method is still hindered by problems such as their sensitivity to the type of material and complexities inherent in oxidation measurement.

Developments in control system software safety analysis

The replacement of electrical or electromechanical equipment made obsolete by technology advances is also a consequence – albeit indirect – of aging. One example of this is the upgrading of electromechanical instrumentation and control systems to programmed I & C.

For several years now, IRSN has been developing methods and computerized aids to analyze software quality and reliability for safety-important systems, so that its own safety assessment expertise can be tailored to the programs used in new I & C technologies. This has led, for example, to creation of the ATLAS workshop (for analysis and testing of safety-related controller software). The ATLAS package comprises various computer tools enabling:

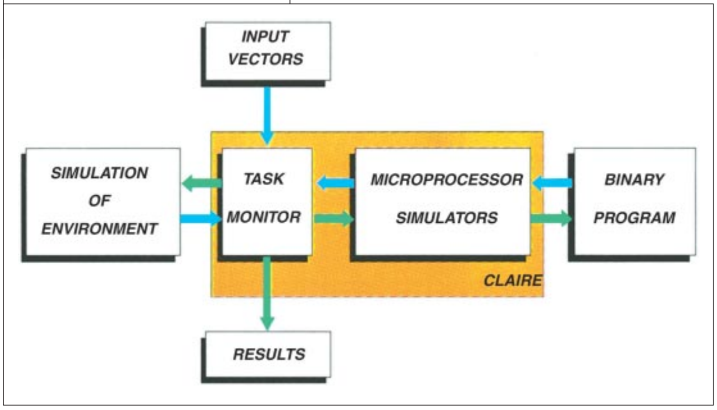
- evaluation of development and maintenance methods used by software manufacturers, based on compliance with industrial standards and the current state of the art;
- verification that functions assigned to the software are in fact being carried out, to the exclusion of all others, and that the software can continue to perform its functions despite equipment failures or development errors.

The ATLAS workshop includes both “static analysis” tools for evaluating all documentation – from the specifications to the “source” program – and “dynamic analysis” tools, for assessment of executable performances under normal or abnormal conditions. These tools have been used to evaluate the safety of I & C programs for the French N4 plant series.

To keep abreast of rapid changes in I & C technologies, the ATLAS approach has been extended by development of a tool for determining component functional paths (mathematical relationships between a component output and the set of inputs required to generate it), based on source program analysis. This tool verifies that the selected test scenarios explore all of the possible functional paths programmed in the software.

Figure 8

The CLAIRE simulation tool.



Another IRSN-developed tool, CLAIRE (**figure 8**) can now simulate software operation without need for the hardware (CPU board, peripheral cards, etc.) used in the plant. CLAIRE employs models to replace the actual circuits and their environment (clock, communication circuitry, memories, etc.).

A last development underway is that of Chronoscope, a package dedicated to static analysis of “real time multitasking software”, since, despite the many conventional, sequential-

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Figure 9

"Mini SG" developed for IRSN by CEA.

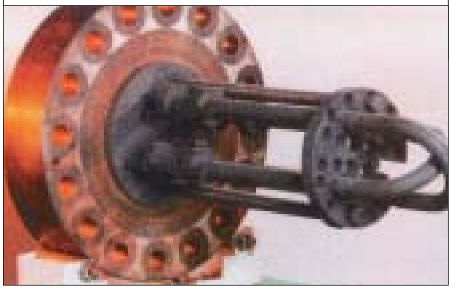

Figure 10

SCC tests being performed in an autoclave.

type analysis and test tools on the market, there is currently no customized product available for "real time multitasking" programming.

Steam generator corrosion

In a PWR plant, the predominant type of SG tube damage is caused by corrosion cracking. A good knowledge of this phenomenon is essential to ensuring performance of the SG's main two safety functions:

- core heat removal;
- containment of the reactor coolant and the radionuclides it carries in an apparatus combining both second and third barrier functions.

In certain confined zones of the steam generator (rolling zone, tube-to-tube support plate intersections, etc.), buildup of corrosion product and solid particle deposits can lead to certain types of cracking, in some cases the through-wall type and, more importantly, to significant decrease in tube mechanical resistance, whose possible outcome is rupture. This is of serious concern to safety. Over the past few years, IRSN has been engaged in R&D aimed at improving detection and understanding of stress corrosion cracking (SCC) that develops locally on the outer walls of SG tubes wetted by the secondary fluid. Such efforts focus on estimating tube outer diameter crack growth rates, and on developing and assessing methods for crack size determination, primarily nondestructive inspection.

Corrosion phenomena have been investigated at the LETC (technological corrosion testing laboratory) in La Hague, by development and construction of "mini steam generators" (**figure 9**) which simulate the thermal flux conditions prevailing in an actual PWR SG. Environments seemingly conducive to OD crack initiation/growth have been tested in micro-autoclaves, using "U-bend" specimens. Preliminary results show that cracking occurs in acid sulfate, degraded resin and lead-containing environments. A second stage of testing performed in an autoclave concerns rate of stress corrosion crack growth and involves use of fracture mechanics specimens (**figure 10**).

Information obtained about crack growth kinetics should lay the groundwork for building physical models of this phenomenon.

These tests likewise serve to develop and validate a method for fine acoustic emission detection that will enable monitoring of the various phases in corrosion crack growth. Information thus obtained about crack growth kinetics should then lay the groundwork for building physical models of this phenomenon.

International partnerships and future programs

Paralleling IRSN's own research on PWR component aging, is its participation in international working groups (particularly in OECD/NEA/CSNI initiatives) and certain programs in other countries – large-scale testing of prestressed concrete containment behavior at the Sandia laboratories, the CIR program on irradiation-induced corrosion, evaluations of nondestructive inspection technologies. In the area of safety assessment tools and methods for I & C software, IRSN has initiated a partnership with the NRC by seconding to the United States one of its own specialists.

New research programs at the Institute deal with the aging of elastomer and composite materials that help ensure leaktightness of NPP second and third containment barriers. Finally, IRSN is developing nondestructive methods for direct *in situ* measurement of aging-affected parameters, to supplement justifications provided theoretically and experimentally. Such a method is being readied for assessing embrittlement of castings used in the PWR reactor coolant system. A new, similar type of application, involving ultrasonic measurement of prestressed concrete containment aging, is now under evaluation. Its objective is to permit direct measurement of characteristic aging parameters such as stress states in cables and concrete, crack status and concrete porosity.