

FRENCH NUCLEAR PLANT LIFE MANAGEMENT STRATEGY APPLICATION ON REACTOR PRESSURE VESSELS AND STEAM GENERATORS LIFE MANAGEMENT

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ABSTRACT

The process applied by the French utility, concerning the **Reactor Pressure Vessel** integrity assessment, on 58 PWR NPPs 3-loop and 4-loop Reactors, involved the verification of the integrity of the component under the most severe conditions of situation and taking into account of the decreasing of the characteristics of the material under irradiation, was engaged several years ago and the result was the justification of the 900 and 1300 MWe RPV life management.

Since 2000, the studies was carried out on the PWR NPPs Reactors and the recent results obtained shows the demonstration of the integrity of the RPV, in the most severe conditions of loading in relation with **RTNDT** (Reference Nil Ductility Transition Temperature), and considering major parameters particularly the severity of the transient. This approach, is based on specific mechanical safety studies on the 1300 MWe RPV, to demonstrate the absence of risk of failure by brittle fracture. For these mechanical studies the major input data are necessary :

- 1 - the fluence distribution and the values of 4-loop RPV **RTNDT** during the lifetime in operation,
- 2 - the temperature distribution values in the downcomer and the PTS evaluation.

The main results must show significant margins against initiation of the brittle fracture. The flaws considered in this approach are shallow flaws beneath the cladding (subclad flaws) or in the first layer of cladding. The major tasks and expertises engaged by EDF are:

- better knowledge of the vessel material properties, including the effect of radiation,
- more precise assessment of the fluence and neutronic calculations,
- the NDE inspection program based on the inspection of the vessel wall, with a special NDE tool.

For the **Steam Generators** life management and Replacement policy, the approach used by the French utility, concerning the Aging Management system of the Steam Generators (SG), applied on PWR NPPs, involves to evaluate the degradation mode of the SG and the criteria associated to maintain the component in operation and after that to determine the Life Management of each to prepare long term life time in operation, taking into account the degradation of Alloy 600 material and the replacement of these materials by components made with Alloy 690.

The financial stakes associated with maintaining the lifetime of nuclear power stations are very high; thus, if their lifetime is not very evaluated and the anticipation of actions is not determined and decided, the cost will be high a large life management was decided to take the good decisions.

This paper shows the program to follow the aging evaluation with application of specific criteria for SG and RPV. The strategy of Steam Generators Replacement at the best period is developed and RPV integrity assessment programme of monitoring are detailed.

Keywords: French NPPs Plant Life Management, Reactor Integrity, Steam Generator Strategy

1 - INTRODUCTION

EDF's Pressurized Water Reactors, 54 units, currently in operation supplied 82 % of the domestic power generation in 1999 (in particular it should be noted that these nuclear power plants provide about important part of all electricity generation).

The first nuclear PWR reactor (900 MWe) built by EDF was put into service twenty years ago. The 900 and 1300 MWe units in operation have an average age of 17 years for the thirty-four 900 MWe units and 12 years for the twenty 1300 MWe units.

These production power plants are now almost 50 % amortized, and represent a technical and financial investment of overriding importance, both for EDF and for France.

Over the last twenty years, the choice of the nuclear program has formed the basis for a wide margin of competitiveness in electricity production, guaranteed energy independence and reduced releases of carbon dioxide into the atmosphere. Therefore control over existing nuclear power plants is of overriding importance and involves the following three objectives :

- maintain current operating performances (safety, availability, costs, security, environment) in the long term, and possibly improve on some aspects ;
- wherever possible, operate the units throughout their design lifetime, in other words 40 years, and even more if possible ;
- consolidate acceptance of nuclear energy, largely based on earning the public's trust.

The financial stakes associated with maintaining the lifetime of nuclear power stations are very high; thus, if their lifetime is shortened by about ten years, dismantling and renewal would be brought forward which would increase their costs by several tens of billions of Euros.

2- APPROACH APPLIED FOR RPV ASSESSMENT

The Reactor Pressure Vessel (RPV) of all of EDF's three and four loop units is an important component to determine the life management of NPPs, it is a major task to justify the RPV and components for all conditions of loading, particularly in emergency and faulted situations, and will remain in operation for lifetime beyond 40 years.

The nuclear boiler was designed based on an engineering lifetime of 40 years, as used in safety reports. However, from a regulatory point of view, French law does not specify any time limit on the operating lifetime of the installations, within the construction authorization decree.

Therefore, EDF should do everything in its power to justify to Safety Authorities and to the public that this lifetime can be reached and, if possible, extended in order to make maximum use of investments already made.

The lifetime of a nuclear power plant may be affected by three main factors :

- normal wear of its components and systems - sometimes called aging - that depends particularly on their age, operating conditions and maintenance conditions applied to them ;
- the safety level, that must consistently comply with the safety requirements applicable to the power stations at all times, and which could change as a function of new regulations ;
- competitiveness, which must remain satisfactory compared with the competitiveness of other production means.

Within this context, obtaining a lifetime for at least 40 years depends primarily on control of the safety level which must comply with safety requirements at all times, and secondly on all technical and industrial aspects in order to operate units with a good level of competitiveness and safety [1].

Technically, the objective is to derive an understanding of aging effects, and to define and then implement appropriate safeguards to maintain the performance level of units at their current level.

Considering these elements, the general strategy is based on the following two points :

- the ten-year safety reassessment process for each ten years period,
- the implementation of two structured programs in order to make sure that all technical and industrial actions necessary to achieve a lifetime of at least 40 years are actually implemented :

Obviously, there are ongoing discussions with the Safety Authorities about this process and these programs.

3 - PROGRAM ON REACTOR PRESSURE VESSEL

Since 1987, EDF has been setting up a "Lifetime" program in order to understand and anticipate aging problems.

This program reviews everything that can have an impact on the lifetime of installations, considering purely technical aspects related to equipment, set industrial, economic and regulatory aspects.

The "Lifetime" program also identifies progress necessary to improve knowledge about aging phenomena and to support Research and Development actions in order to make a better link between operating conditions and maintenance conditions for components and their lifetime.

This program makes a distinction between [2]:

- Two non-replaceable components : the reactor vessel and confinement containments,
- Other components replaceable included in exceptional maintenance strategies with the objective of reaching a lifetime of at least 40 years in the maintenance policy, primary circuit components for example.

One example of a presentation of new development for life evaluation and monitoring is presented for the RPV combined with the vessel in-service inspection on the core zone and other part of the vessel,

In fact what input data are necessary to introduce in thermal-hydraulic calculation and mechanical analysis.

4 - TOOLS AND METHODS FOR FLUENCE AND RTNDT ASSESSMENT COMBINED WITH RPV IN-SERVICE INSPECTION PROGRAM AND MECHANICAL ANALYSES

The methodology applied for RPV assessment is based on :

- 1st - The evaluation of RTNDT (Reference Nil Ductility Transition Temperature) at different period (Lifetime period),
- 2nd - The evaluation of RTNDT after mechanical analyse computation and thermohydraulic calculation.

The diagram (Fig 1) shows the methodology used for vessel by vessel evaluation.

Table1
METHODOLOGY OF SPECIFIC RT_{NDT} EVALUATION VESSEL by VESSEL

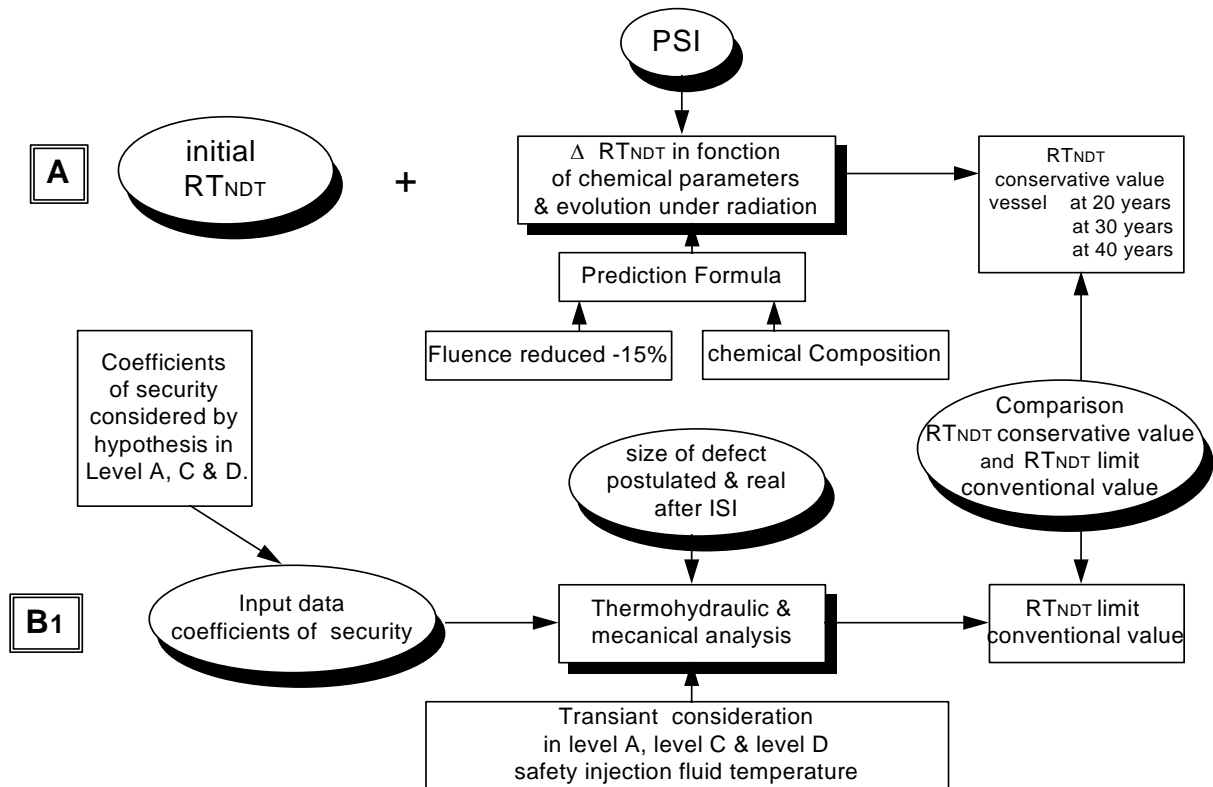


Fig 1 - Diagram to evaluate RT_{NDT} vessel by vessel

5 - EVALUATION OF FLUENCE AND RT_{NDT}

Neutron-induced embrittlement of vessel materials is measured by the increase in nil ductility transition reference temperature (RT_{NDT}). This is an internationally established practice, since RT_{NDT} is known to be closely linked to fracture toughness. A shift in RT_{NDT} also reflects the envelope shift observed in fracture toughness curves as a result of aging.

The basis for demonstrating an absence of brittle fracture risk is predicting maximum shift in end-of-life RT_{NDT} under radiation, using formulas that allow for the chemical composition of the affected materials.

The fluence value, evaluated at regular intervals, is required to follow and to determine the RT_{NDT} of the vessel during the lifetime. The diagram presents the relation between the fluence level and the ΔRT_{NDT} assessment in operation period, for each 10-year reassessment process.

In fact for each vessel with the RT_{NDT} value at different period of life is evaluated by the relation

$$\text{Initial } RT_{NDT} + \Delta RT_{NDT} = RT_{NDT} \text{ value (40-50 years)}$$

5.1 - Fluence Management of Reactor Vessels

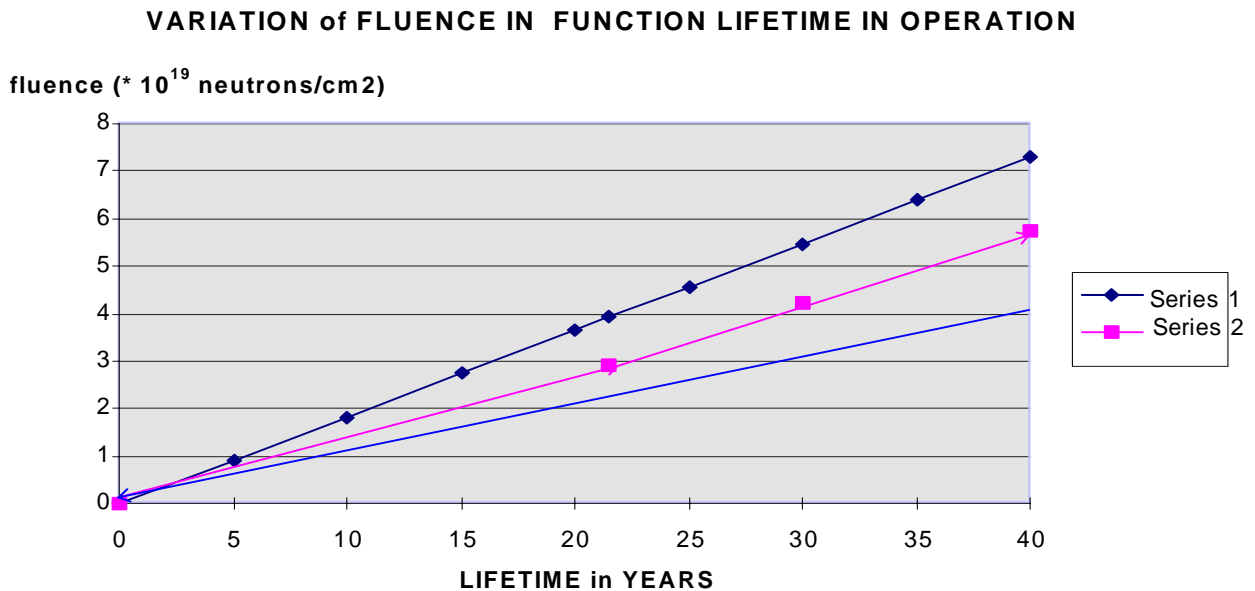
Regarding the diagram of the Figure 1, the first analysis allows the fluence and specific RT_{NDT} vessel by vessel obtained after radiation damage evaluation (methodology [A]).

The fluence management (reduction of fluence under -15%) is input data to determine ΔRT_{NDT} with prediction formulas and specific chemical composition of the vessel.

Use of core physics codes in conjunction with database of recently determined cross sections validated through extensive R & D, enables accurate assessment of the fluence experienced by all zones of the vessel. This information is vital to ensuring coherent calculation of RT_{NDT} shift.

Not to be content with such measures, EDF has now undertaken to optimise fuel loading patterns to reducing fluence, while allowing for two increasingly widespread trends: deployment of MOX fuel in hybrid management schemes (in CP1/CP2 900 MWe reactors series) and fuel cycle expansion to an average 18 months (in 1300 MWe units).

Parallel to these developments, fuel loading patterns are being optimised in terms of maximum fluence undergone by reactor vessels. Depending on the loading pattern and barring any operating contingencies, a decrease of 15 to 40 % in maximum flux can be achieved with respect to re-evaluated design values, for the standard fuel arrangement (Fig. 2). With a special neutronic calculation code " EFLUVE ", each year the fluence level is determined before the arrangement of the refuelling plan unit per unit for all of 3 loop and 4 loop reactors [3].



0 -> 7,3 Black curve = Basic design fluence curve
 0 -> 5,8 Pink curve = Fluence reduced - 15% curve
 0 -> 4, 3 Bluecurve = optimized fluence Fluence - 40 % curve

Fig. 2 Fluences curves

5.2 - Monitoring Radiation Impact on RT_{NDT} Shift

Right from the start of 900 and 1300 MWe plant operations, a comprehensive program of radiation surveillance was devised to collect data on RT_{NDT} evolution in each vessel (Table 2).

Table 2 - CP1 and CP2 REACTOR VESSEL SURVEILLANCE PROGRAM - Current Status

	capsule 1 U	capsule 2 V	capsule 3 Z	capsule 4 Y
Residence Time in Vessel (years)	4	7	9	14
Equivalent Vessel Wall Radiation Time (years)	11.2	19.5	28.1	39.1

This program is based on withdrawal, at predetermined outage intervals, of four radiation specimen coupons previously placed inside the vessel (Fig. 3), at positions exposed to greater neutron flux than the vessel wall itself. Subsequent analysis of the Charpy test samples taken from these capsules yields quantitative data on the impact of neutron-induced embrittlement and enables suitable anticipation of irradiated vessel mechanical "health" at the 40-years service milestone [3].

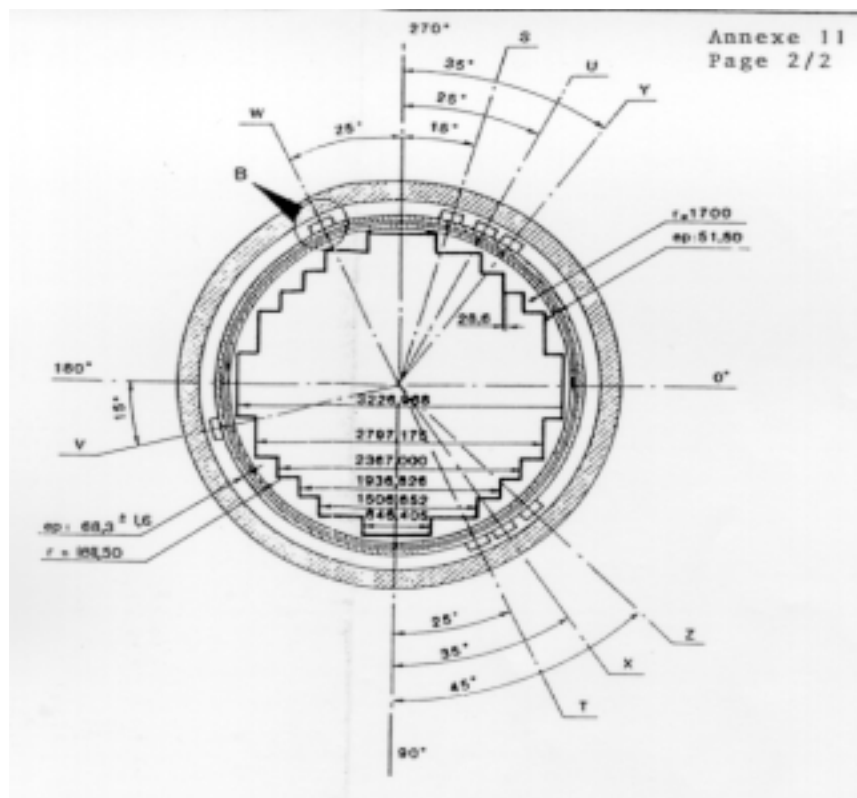


Fig 3 - Situation of radiation specimen capsule in RPV

With this radiation surveillance program, EDF decided to reintroduce two reserve radiation specimen capsules in the vessel. EDF has decided to begin taking measures to evaluate vessel fluence and RT_{NDT} beyond age 40 years (service life goal for its 900 MWe PWRs: 50 years).

This will entail putting back into vessels the two extra capsules supplied by FRAMATOME the constructor of the Reactors for the radiation surveillance program.

These "reserve" specimens, procured from the same sample ring as the original capsules, will be reinserted after removal of the latter, during upcoming outages. thirteen years of radiation will then be required to obtain a "picture" of vessel fluence for a 40-years duration.

5.3 - PRESENTATION ORGANIZATION OF THE SURVEILLANCE PROGRAM - RESULTS OF EXPERISES ON SPECIMEN CAPSULES

As it has already been the subject of several presentations[1] , we shall restrict ourselves here to the main characteristics of the organization of the French surveillance program (Table 2).

Largely inspired by American regulations, practice fulfills the requirements of the French order and its circular dated 26th February 1974. Each reactor vessel is the subject of monitoring of the materials in its core zone, defined as being susceptible to marked embrittlement towards the end of its design life, in this case 40 years. Generally, this core zone is made up of two shells and the associated weld for all 900, 1300 and 1450 MWe reactors.

On the basis that the risk of brittle fracture is highest at the end of design life, a base metal, a weld and a heat affected zone are selected for each vessel. These materials, installed in the reactor by means of capsules at locations characterized both in terms of temperature and neutronic conditions, allow the evolution of the mechanical characteristics of the component to be monitored.

Each capsule contains nuclear instrumentation, based on fissile and activation dosimeters, and thermal instrumentation, based on low melting point alloys, so as to determine accurately the conditions of its stay in the reactor and to optimize the use of the samples for mechanical tests.

Analysis is mainly based on impact strength tests; by hypothesis, the shift ΔT_{cv} , resulting from the comparison of the impact strength curves pre and post irradiation, is taken to be representative of that of the codified toughness curve for this type of material and the temperature scale of which is index linked to the RT_{NDT} of the material. This shift is measured at the 56 Joule level or at 0.9 mm of lateral expansion.

The main objective of this analysis is to verify the conservatism of the hypotheses adopted at the design stage as regards aging, which are based on the determination of the initial RT_{NDT} of each component and on an assessment of its evolution (ΔRT_{NDT}) by means of an empirical prediction formula.

Currently, we use a formula developed by FRAMATOME for the products used in France, known as **FIS** (embrittlement by higher irradiation) :

$$\Delta RT_{NDT} (^{\circ}C) = 8 + [24 + 1537 (P-0.008) + 238 (Cu-0.08) + 191 Ni^2Cu] [\varnothing/10^{19}]^{0.35}$$

- where :
- . \varnothing : fluence in $n.cm^{-2}$ ($E > 1$ Mev),
 - . $10^{18} < \varnothing < 6.10^{19}$
 - . P, Cu, Ni: content by weight - %
 - . $Cu - 0.08 = 0$ if $Cu < 0.08$
 - . $P - 0.008 = 0$ if $P < 0.008$
 - . $275^{\circ}C \leq T_{irradiation} \leq 300^{\circ}C$

Following the results, a new formula, **EDFs**, has been recently developed by EDF concerning the welds. This development took into account all the available data on irradiated welds in the chemical and neutronic ranges met in the French reactors.

This is :

$$\Delta RT_{NDT} (^{\circ}C) = 22 + [13 + 823 (P \geq 0.008) + 148 (Cu-0.08) + 157 Ni^2Cu] [\varnothing/10^{19}]^{0.45}$$

- where :
- . Φ : fluence in $n.cm^{-2}$ ($E > 1$ Mev),
 - . $3.10^{18} < \Phi < 8.10^{19}$
 - . P, Cu, Ni: content by weight - %
 - . $Cu - 0.08 = 0$ if $Cu < 0.08$
 - . $285^{\circ}C \leq T_{irradiation} \leq 290^{\circ}C$

Logically, in the light of experience feed-back, organization and practice have evolved over the course of time and different standardized plant series built in France, this mainly concerns the type of fracture mechanics specimen, associated with the impact strength test specimen in the capsules, and the nuclear instrumentation [4].

5.4 - Results of the Analysis of the Surveillance Program

To date, 130 capsules, removed from 51 power units, have been analyzed. The content of embrittling elements (copper, phosphorus and nickel) present in the materials concerned is relatively low, with the exception of the welds on the CP0 standardized plant series (6 first 900 MWe power units), as shown in table 2.

The results of mechanical characterization are associated with a neutron dose, obtained from the interpretation of the activity measured on the fissile and activation dosimeters. This metrological process, performed according to a well-established methodology, requires a certain coherence to be obtained between the activity measured, the operating diagram of the power unit and the migration calculations performed according to reactor geometry. This coherence is evaluated at less than 10% over all the results and indicates the quality of methodology used. Furthermore, these results reveal very homogenous operation of all the reactors.

The embrittlement of the various materials monitored in the surveillance program may be expressed in the form of shifts in the transition of the impact strength curves and is illustrated by figures 1 to 3. **It remains moderate, yet with relatively high dispersal of the results. As regards the base metal, the maximum value obtained is 82°C and, for the weld, 73°C for the base metal, correspond to a dose of $6.5 \times 10^{19} n.cm^{-2}$.**

For both the base metal and welds, the measured shifts reveal a scatter in the embrittlement results of up to 60°C for a given fluence, together with cases of “abnormal” embrittlement kinetics. We can notice that, in the case of HAZ, the scatter is significantly lower. This shows that irradiation is not only the phenomenon at the origin of this scatter.

6 - MECHANICAL ASSESSMENT TO DETERMINE RT_{NDT}

For thermal-hydraulic and mechanical evaluation the input data values necessary are:

- the different transient considerations in level A - in level C and in level D ; the most severe situation of loading in this case is the small-break LOCA considered in level C,
- the temperature of safety injection fluid (9°C),
- the coefficients of security considered by hypothesis in level A - level C - level D,

the size of the defect postulated at azimuthal situation and maximum fluence value and after in-service inspection the defect size and position in the vessel shell.

In fact the Relation B in the diagram (fig. 1) is the following relation

$$A(Cs) \cap B(D) \cap C(T) = RT_{NDT} \text{ limit value (1)}$$

A(Cs) is a function of safety coefficient ,

B(D) is a function of defect (in subcladding area),

C(T) is a function of transient.

6. 1 - In-Service Inspection and defect size to determine Fluence

In methodology [B], the inspection of the core zone and the results in terms of defect size in the undercladding area or not is input data introduced to determine the fluence and RT_{NDT} values.

In-service inspection of the vessel zone adjoining the core (Fig. 4) is intended to identify subcladding defects that are likely to aggravate end-of-life brittle fracture risk under incident or accident conditions.

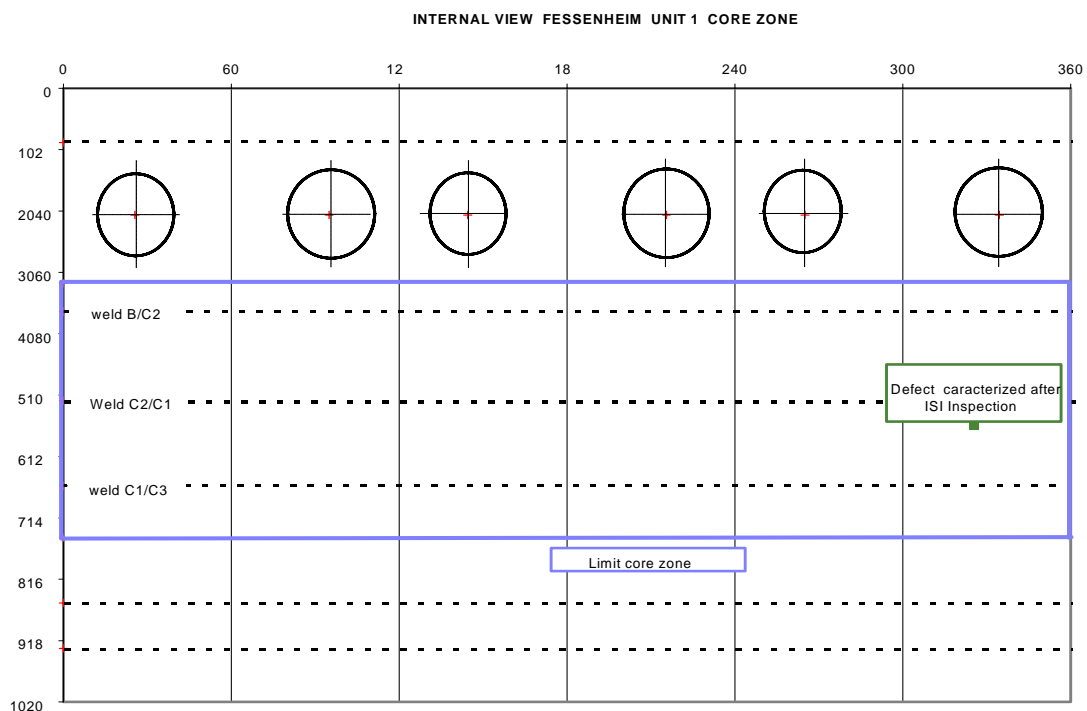


Fig 4 - all core zone inspected by ultrasonic Probe

A non-destructive, underwater testing method based on "focused ultrasonic probe" technology was thus developed since 1998 to cover the vessel subcladding zone and applied each 10-years outage NPPs.

As part of its in-service inspection program for French power plants, EDF has now included the systematic testing of all RPVs during outages scheduled every 10 years. On the basis of the minimum detection threshold, a "reference" defect, flaw 6 mm deep and 60 mm long, has been defined for all vessel integrity studies [5].

Mechanical tests carried out on all 900 MWe-class vessels show that such a defect remains acceptable, regardless of its location in the "sensitive" zone of the vessel, under all transient loadings, after allowance for the effects of irradiation aging.

7 - SYNTHESIS OF MAIN ELEMENTS INPUT DATA TO THERMAL-HYDRAULIC CALCULATIONS AND MECHANICAL ANALYSIS

To perform this demonstration, and more generally to optimise vessel operating life, it is thus necessary:

- to determine the fracture toughness of irradiated vessel materials. This entails upstream knowledge of initial vessel RT_{NDT} , fluence values at all points in the "sensitive" zone and shift in RT_{NDT} induced by radiation. It also means verifying the quality of the correlation between RT_{NDT} and toughness.
- to make suitable flaw size assumptions (based on in-service surveillance results), to both define the size of the minimum defect that will not "escape detection" and provide an exhaustive inventory of the significant defects recorded.
- to perform the computations required to ascertain true safety margins, by constructing an in-depth thermal-hydraulic model of the most severe transients, and associating it with precise thermo-mechanical calculations of loadings in the vessel.

This method and tools associated are applied on specific vessel by vessel assessment, and methodology [A] shows fluence and RT_{NDT} levels of the vessels and will compare with RT_{NDT} obtained with methodology [B] .

8 - STEAM GENERATORS AGING AND LIFE MANAGEMENT

An important result obtained after the Research and Development program was the better knowledge of the root cause of SG tubes under PWSCC. After large program of Engineering studies combined with research on materials under pressurized water chemistry condition, the replacement material for SG was ALLOY 690 [6].

The Criteria and the classification of SG degradation and plugging rate are function of wear and level of degradation. It will estimate the time in operation of all of Steam Generator, components by components by a specific program of monitoring. In parallel, when the replacement of SG was identified, the planning et industrial aspects were included future operations with the order to made SG for replacement.

The classification of all of SG is divided in three groups.

Group 1

- Units with SG affected by important degradation on ALLOY 600 tubes and the planning of SG replacement will be planed (date will be scheduled),

Group 2

- Units that SG lifetime will not exceed 30 years (SG tubes in ALLOY 600 without treatment)

Group 3

- Units that lifetime of SG is the lifetime of the plant (Fig. 5).

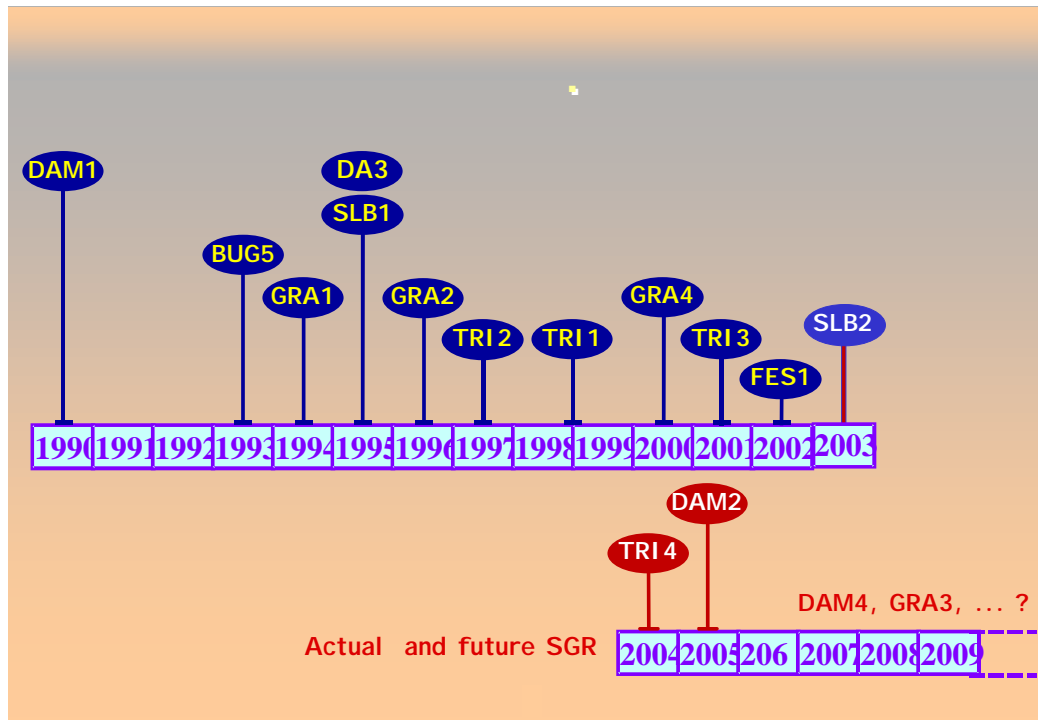


Fig 5 - Program of Steam Generators Replacement

8.1 - Steam Generator Lifetime criteria

The Steam Generator aging management criteria is considering the volume of plugging of ALLOY 600 tubes [6].

This criteria is the plugging rate. The technical end of life of SG tubes is the major parameter for monitoring.

The plugging rate Criteria is the maximum value of tubes plugged.

The criteria is Plugging rate between 12,5% and 15%.

8.2 - Industrial Aspects

For the Industrial point of view it is important to anticipate and to manage the future considering the spare parts of Steam Generators and to have strategic stock of SGR.

The key factors are a good balance between operations and long term planning for replacement coupled with industrial capability. Major parameters to take into account are :

- SG spare parts,
- Plugging rate,
- Management of interventions,
- Industrial aspects ; planning coupled SGR and Ten-Years outage,
- Exceptional program of replacement SG with Vessel Head or with inlet and outlet Steam Generators cast elbows.

The first Steam Generators replacement was carried out in 1990 on DAMPIERRE unit 1 plant (Fig. 6). The experience feedback of this first SGR was examined to take good solution and to reduce the outage duration and to optimise the planning of different operations.

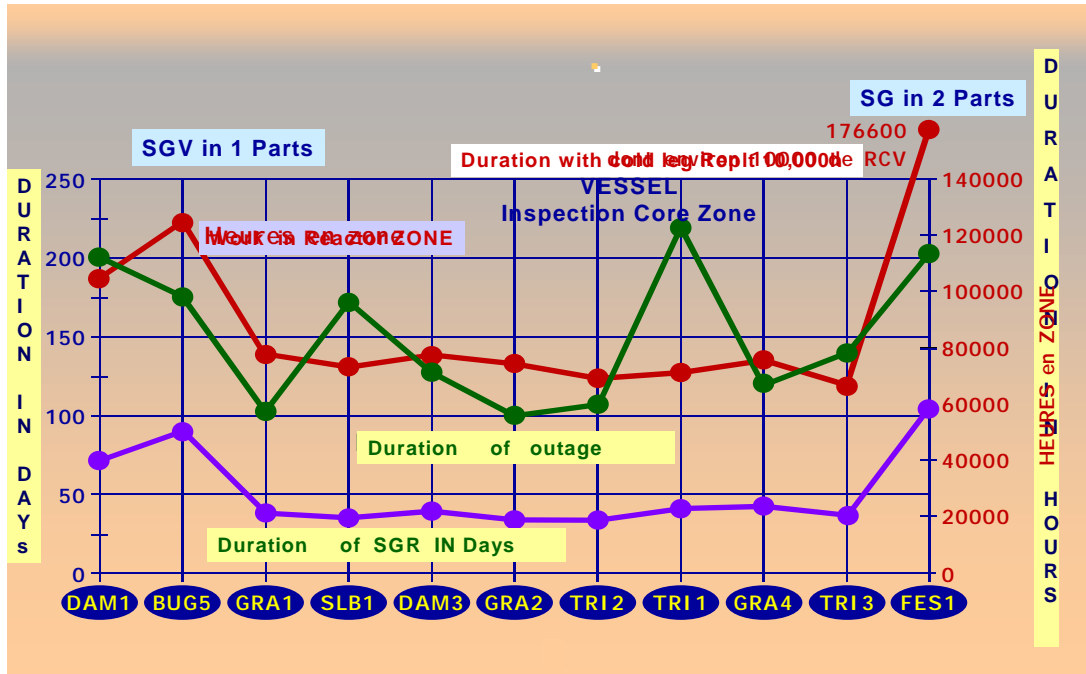


Fig. 6 - Planning of SGR

Since 1990, 12 Steam Generators Replacement were carried out. The actual SGR program is exclusively on 900 MWe plant. The lifetime of all of 4-loop 1300 MWe plant Steam Generators the is lifetime of the plant.

The management of maintenance program and SGR need a main condition a good balance between a pragmatic approach based on experience feedback and a theoretical approach based on predictive models.

9 - CONCLUSION IMPLEMENTATION OF THESE TOOLS AND METHODOLOGY TO FOLLOW UP THE RPV ASSESSMENT

The evolution during the lifetime in operation of all of EDF's 58 PWR units requires a good knowledge of the evolution of mechanical and metallurgical parameters of each **Reactor Pressure Vessel** and primary circuit components. The development of specific tools and methodology to follow up the evolution of RPV fluence level and the RT_{NDT} to verify the integrity assessment is a major objective for EDF.

The **Steam Generators** Life Management and maintenance program is based on large experience since 1990. The program of Replacement is carry out for:

- One piece Steam Generators (one-piece),
- Steam Generators made in two pieces for oldest 900 MWe plants on first series NNPs .

The Classification of Steam Generators in three families is a good parameter. Large monitoring actions on SG tubes inspections and plugging are applied periodically following the maintenance program, combined with the plugging rate criteria.

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