AVN's experience as TSO in safety assessments of steam-generator-replacement and power-uprate projects

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Abstract:

Much experience has been gained in Belgium on steam-generator replacements (SGR) and power uprates (PU) of nuclear power plants. Indeed, steam generators were replaced in all but one of the seven Belgian Nuclear Power Plants (NPP) in operation. The replacement was accompanied by a PU for four of them.

The purpose of the paper is twofold. It first presents an overview of the steam-generator replacements occurred in Belgium, summarizing the main steps and providing a historical picture. Then, it describes several important aspects in the licensing studies related to such plant modifications, as seen by AVN, taking for the sake of illustration the example of the recent PUSGR performed at the Doel 2 NPP.

1 INTRODUCTION

In Belgium, AVN plays the role of technical support (TSO) to the competent Safety Authorities, the Belgian Federal Agency for Nuclear Control (FANC), and its main duty is to make sure that an adequate level of safety is maintained during the course of operation of the Belgian NPP.

In particular, AVN is involved in different steps of all the SGR and PU performed at the Belgian NPP. These steps include the review of the safety studies (verification of completeness, auditing of new proposed calculation codes, assessment of methodology reports, review of the results of the safety studies), the following of the requalification and commissioning tests by the AVN inspectors on site, and the updating of documents (Safety Analysis Report, Technical Specifications, plant procedures and in-service inspection programme). This paper focusses on the treatment of the licensing studies.

The initial license of each NPP has been given on the basis of a deterministically defined list of accidents to be considered in the design. Given the decision of the Belgian Nuclear Safety Commission to use the USNRC rules and regulations (with the exception of some particular requirements which are specific to Belgium), that list was the one given in the Regulatory Guide 1.70 and the licensing review took account of the Standard Review Plan (NUREG-800) available at the time of licensing. The postulated initiating events have been classified into 4 "categories" (Conditions I to IV) according to the ANSI standard N18.2 of 1973. Some site-specific external hazards were added to the list.
The acceptance criteria were those given in the USNRC regulatory documents, with the exception of the radiological consequences. Given the high population density in Belgium, stricter limits were imposed.

A brief historical overview of the SGR and PU occurred in Belgian NPP is given in section 2. Section 3 describes several important aspects in the licensing studies related to such plant modifications, as seen by AVN, taking for the sake of illustration the example of the recent PUSGR peformed at the Doel 2 NPP.

2 SGR AND PU PERFORMED IN BELGIUM

Table 1 presents the milestones in the life of the seven Belgian NPP, starting from their initial design. All plants have three loops, except Doel 1 and Doel 2, which are twin two-loop plants. Except for Doel 1, all plants underwent important modifications, either a SGR alone (2 plants), either a PU alone (in the single case of Tihange 2, for which the PU occurred in two steps, the second one with a SGR), or a PUSGR (4 plants).

In Table 1, the modifications (SGR, PU or PUSGR) are specified in chronological order for each plant, with the related year, reactor nominal thermal power (in MWth) and, in case of PU, the rate of the power increase (in %), with respect to the initial-design power.

The latest modification occurred at the Doel 2 NPP (PUSGR). A detailed description is given in the next section.

NPP	***What***	***When***	***Power (MWth)***	***PU***
Doel 1	Initial Design	1974	1192	-
Doel 2	Initial Design	1974	1192	-
Doel 2	PUSGR	2004	1310	10%
Doel 3	Initial Design	1982	2785	-
Doel 3	PUSGR	1993	3064	10%
Doel 4	Initial Design	1984	2988	-
Doel 4	SGR	1996	2988	-
Tihange 1	Initial Design	1974	2660	-
Tihange 1	PUSGR	1995	2873	8%
Tihange 2	Initial Design	1982	2785	-
Tihange 2	PU	1992	2905	4.3%
Tihange 2	PUSGR	2001	3064	10% (+5.7%)
Tihange 3	Initial Design	1984	2988	-
Tihange 3	SGR	1998	2988	-

Table 1 - Milestones in the life of Belgian NPP

3 A CASE STUDY – PUSGR AT THE DOEL 2 NPP

3.1 Input for the safety analyses

A set of new safety analyses had to be performed because of:

- the modified characteristics of the new steam generators provided by Mitsubishi Heavy Industries (increased heat exchange area, increase of the primary free volume, almost unchanged secondary free volume, decrease of the total secondary mass in the steam generators at full power, reduced outside diameter of the steam generator tubes);

- the increase by 10% of the nominal thermal power, from 1,192 MWth to 1,311.2 MWth;
- the higher residual heat to be removed.

The definition of these new characteristics of the nuclear power plant resulted from a feasibility study.

In addition a new reference core had to be defined. The characteristics of this reference core determine the nuclear core initial and boundary conditions for the safety analyses to be performed.

The power uprating and validation of the considered reference core characteristics was believed to be made possible by the improved efficiency of the new steam generators, and by the application of new calculation codes and methodologies which would enable a better identification and use of the available safety margins.

3.2 Choice of the operating domain

The new operating point, after SG replacement, depends on the turbine and SG characteristics. However, the primary average temperature program was left unmodified: the primary average temperature keeps its values of 299.6°C (nominal power) and 283.4°C (zero power).

The operating domain takes also into account a stretch-out at the end of the fuel cycle. The associated decrease of the average primary temperature leads to a secondary pressure decrease, which brings additional limits to the operating domain (turbine limit).

These new nominal operating point and operating domain had to be confirmed by the analysis of the most limiting accidents with regard to the performance of the reactor core, by the verification of the capacity of the safeguard/safety and emergency/auxiliary systems, and by the verification of the mechanical integrity of the primary components. Indeed, plant operation under stretch-out conditions is foreseen to occur during a rather long time (with a maximum of three months). Plant transients need therefore to be analysed assuming the whole operating domain for defining the initial plant conditions.

The values of the primary flow rate were calculated with due account of the new plant conditions, and in particular of the new steam generators. The minimum and maximum values of this flow rate are of particular importance for the safety studies.

The thermalhydraulic design flow rate (minimum value) was kept at its value before the PUSGR (32,700 m^3/h), providing some margin as:

1. the maximum allowed SG tube plugging was reduced from 25% to 5%, reducing the global flow resistance of the primary circuit;
2. to a smaller degree, the pressure losses at the primary side of the new steam generators are smaller.

The mechanical design flow rate (maximum value) was fixed at 40,000 m^3/h, about 10% above the best-estimate flow rate value after SG replacement.

3.3 Reference core and nuclear design

A reference core was defined. It had to be bounding as much as possible with respect to the equilibrium cycle of the project (main characteristics: fuel cycle length up to 12 months, plus a stretch-out period of up to 3 months for the sake of conservatism in the studies; reloads of

36 UO2 fuel assemblies of the Framatome-ANP HTP 14x14-6-1 type; enrichment of 4.5% in U235). A set of parameters, called “key parameters”, was calculated for this reference core. After adding the related uncertainties and a provision for the above-mentioned bounding character and for the variability in the real loading patterns, the values of these key parameters (hot-spot factor FQ = 2.30, hot channel factor FDH = 1.65, ...) were then validated by the project accident analyses.

The reference core is supposed bounding for the core reloads subsequent to the PUSGR. However, the real subsequent cores are never identical to the project reference core: more or less important differences can be observed, for example depending on the fuel type used or the way the core is managed.

In order to verify the conformity of a reload without repeating all the safety studies, the reload key parameters are compared to the project ones.

3.4 Thermalhydraulic core design and safety limits

The thermalhydraulic core design was performed using a statistical methodology (Siemens Statistical Thermal Design Procedure of Framatome-ANP) and a critical-heat-flux correlation (HTP) compatible with the reference fuel. The thermalhydraulic core design results in DNBR design limits and in safety limits for the core protection (OTDT and OPDT setpoints).

The Siemens Statistical Thermal Design Procedure was then used in studies of DNB accidents by Framatome-ANP, or, equivalently, the Statistical Thermal Design Procedure was used in studies of DNB accidents by Tractebel Engineering.

3.5 Accident analyses (design basis accidents)

In the frame of the PUSGR project, the accidents presented in the Safety Analysis Report had to be reanalysed.

Some accidents are in fact not affected by the change in operating parameters resulting from the PUSGR, or are enveloped by other accidents. Justifications for not reanalysing such accidents have been produced. These justifications also cover the stretch-out conditions.

The accident analyses validated the setpoints of the reactor protection system and safety systems. Some setpoint modifications were needed as a result of the PUSGR (examples for the scram signal: OTDT, OPDT, FW/MS mass flow mismatch; examples for the SI signal: high steam flow, low steam pressure).

Accident studies were performed by Framatome-ANP, by Tractebel Engineering or by Westinghouse, mainly with the use of:

- the RELAP5 Mod2.5, LOFTRAN, WCOBRA/TRAC or NOTRUMP codes, for system-thermalhydraulic calculations;
- the COBRA3-CP code, for core-thermalhydraulic and DNBR calculations;
- the PANBOX3 code, for 3D-neutronkinetics calculations.

Most used codes were accepted in the frame of previous applications in Belgium. Otherwise, AVN performs an audit, to examine the documentation of the code, its validation, its utilisation and the associated Quality Assurance program.

Below is a selection of noteworthy studies.

3.5.1 Steam Line Break at hot zero power

The reactor is protected against the Steam Line Break (SLB) accident at hot zero power by the low advanced steam pressure signal in one steam line, and by the very low pressurizer pressure signal. Both signals lead to the start of the safety injection system and to the normal feedwater isolation. Steam isolation then occurs on coincidence of the safety injection signal and on the high steam flow signal in one steam line. As the SLB accident is a cooldown accident, the start of the auxiliary feedwater system has an adverse effect and needs to be accounted for in the study.

The accident was studied by Tractebel Engineering with respect to the DNBR criterion, using the RELAP5 Mod.2.5 code (for the system-thermalhydraulic calculations) and the PANTHER code (for the 3D-core neutronic and thermalhydraulic calculations). Both of these codes are coupled using the TALINK code. Then, the COBRA3-CP code is used for calculating the DNBR, but in a deterministic, not statistical way, and with the W3 critical-heat-flux correlation. Both the PANTHER (kinetic aspects) and TALINK codes required an audit.

When using codes in a decoupled way, it is a common practice in licensing to penalize each code separately. Although acceptable for the licensing purpose, it is recognized that such a practice builds overconservatisms because of the overabundance of unphysical behaviours. For instance in the present accident study, conservative temperatures (with reference to the loop mixing models used to calculate the temperatures at the core inlet) combined with conservative neutronic retroaction assumptions would lead to very unrealistical high power excursions. Protection systems would then be designed against situations that are physically impossible. Using coupled codes has obviously the advantage of evaluating more realistically the incidence of neutronic parameters upon thermalhydraulic behaviour and vice versa.

The study of the SLB accident at hot zero power is the first application of a coupled-codes methodology for licensing in Belgium. It is noteworthy that traditional assumptions accepted as conservative might ask for reconsideration. For instance, a minimum initial primary flow rate was a conservative assumption when decoupled codes were used, but it had to be checked when using coupled codes, which allow to take into consideration the incidence of the primary flow rate on the axial power profile.

The base case of the study considers a double-ended guillotine break as initiator and conservative assumptions; in particular, the most reactive rod is supposed completely blocked out of the core. For the licensing case, obtained after performing sensitivity studies among others on the break size and on the loss of offsite power, the study shows that the DNBR criterion is respected.

3.5.2 Loss of Coolant Accident

The reactor is protected against the Loss of Coolant Accident (LOCA) by the low pressurizer pressure signal, giving rise to reactor trip, by the low-low pressurizer pressure signal, which starts the safety injection system (high and low pressure, as well as containment spray), and by the accumulators, injecting additional water into the primary system. On the secondary side, the auxiliary feedwater system provides additional cooling.

The accident was studied by Westinghouse, as well for the small-size break (SBLOCA) as for the large break (LBLOCA). In the short term, the acceptance criteria are as follows:

- The calculated peak fuel element cladding temperature (PCT) is below 1204°C;
- The maximum localized cladding oxidation remains below 17%, w.r.t. the cladding thickness before oxidation;

- The amount of hydrogen generated by fuel element cladding that reacts chemically with water or steam does not exceed an amount corresponding to interaction of 1% of the total amount of zircalloy in the reactor.

3.5.2.1 SBLOCA

The SBLOCA accident was studied with the help of the NOTRUMP code (for the thermalhydraulic transient calculations) and the LOCTA IV code (for calculating the heating of the cladding). The study was performed according to a methodology, which received acceptance in the frame of a previous project (Upper Plenum Injection) related to the same plant, with a single exception: the COSI model, introduced in order to better simulate the condensation phenomena occurring at the safety injection into the cold legs, was evaluated in the frame of the PUSGR project.

For the licensing case, obtained after performing sensitivity studies on the break size and on the time of Reactor Coolant Pump trip, the study shows that the above-mentioned criteria are fulfilled.

3.5.2.2 LBLOCA

The LBLOCA accident was studied using the "Superbounded" methodology, with the help of the WCOBRA/TRAC code (for the reactor vessel and system thermalhydraulic transient calculations), the COCO code (for the containment analysis), the PAD code (for the calculations related to the new reference fuel in begin-of-life conditions), the THRIVE code (for calculating the reactor vessel steady-state hydraulics), and the TREQ code (for calculating differences between end-of-life and begin-of-life fuel temperatures).

The used codes and methodology were accepted in the frame of a previous project (Upper Plenum Injection) related to the same plant, with a few exceptions leading to specific evaluations.

- In order to account for fuel thermal conductivity degradation with burnup, the analysis performed for end-of-life conditions took a hot-spot factor reduction into account. This assumed reduction is kept sufficiently small so that specific verifications during core reloads are not mandatory. The TREQ code used for the end-of-life analysis required an audit.
- As limiting PCTs are sufficiently high to rise a significant metal-water reaction, contrary to those obtained in previous projects (Upper Plenum Injection for example), an analysis based on Appendix-K models (already accepted by AVN) was needed to justify these limiting cases. The main contributor to the differences in the results was shown to be the cladding swell, modelled in the Appendix-K approach but conservatively not in the "Superbounded" one.

The study shows that the above-mentioned criteria are fulfilled with a hot-spot factor FQ of 2.30 and a hot channel factor FDH of 1.65.

3.6 Radiological consequences of accidents

As the power uprating of the plant leads to an increase of the fission product inventory of the fuel present in the core, Tractebel Engineering provided justifications for the radiological consequences of accidents and, when judged necessary, performed recalculations of these consequences. However, the maximal activity values allowed in the Technical Specifications for the primary circuit limited the scope of reanalyses to the radiological consequences of three accidents (Loss Of Coolant Accident, Feedwater Line Break Accident and Fuel Handling Accident).

For the Loss of Coolant Accident, the study concluded that the admissible leak rates for the containment penetrations and for the recirculation circuit loop had to be reduced. Leak criteria had to be adapted in the Technical Specifications.

3.7 Studies of the reactor core

Tractebel Engineering performed core power capability studies for the reference core of the project, in order to verify that several design criteria related to the power peaking factors are satisfied. For normal operation, it has been verified that the hot-spot factor limit for the Loss Of Coolant Accident is never exceeded, and that the reference axial power profile used in the analyses of accidents limited by DNB is bounding. For Condition-II transients, the studies fixed the limit of the linear power to 656 W/cm, and determined the penalization functions of the OPDT and OTDT protections. Core power capability verifications cover stretch-out conditions.

Framatome-ANP verified the thermomechanical behaviour of the HTP fuel rods (UO2 rods with PCA-2b cladding) in post-PUSGR conditions with the reference loading pattern of the project. By design, the integrity of the rods must be guaranteed during normal operation and Condition-II transients.

3.8 Verification of the safeguard and emergency systems

The power uprate and the SG replacement also lead to verify in detail whether the capacity of safety related systems and their support systems still meets all applicable safety requirements. This verification usually focuses on system-related safety criteria defined either during design or in a later stage.

This group of studies includes more specifically:

- verifications of the system capacities to provide adequate protection against primary overpressure, secondary overpressure and containment overpressure, in case of design basis accidents;
- verifications of the safeguard and emergency systems capacity for aspects not covered by the accident studies, i.e. mostly long(er)-term issues resulting from the increased residual heat.

The studies of primary overpressure (hot and cold conditions) and secondary overpressure are needed because of the nominal core power increase. For each of these studies, the most penalizing transient is considered and a conservative modelling of the mechanical behaviour of the safety valves (pressurizer SEBIM valves or secondary safety valves) is applied. A setpoint modification of the SG safety valves was needed as a result of the PUSGR.

The integrity of the primary containment against the pressure and temperature transient is verified for LOCA and SLB, as the PUSGR may lead to an increase of mass and energy release into the containment atmosphere. In comparison with previous studies performed during the first 10-year safety reassessment of the Doel 1 and 2 plants, a more elaborate methodology was followed, covering the whole power and break size spectrum and accounting in more detail for several plant specifics. As a result, a setpoint modification was eventually needed for the SG isolation signal on high containment pressure.

Moreover, verifications of the structural integrity of containment compartments (floor slabs, walls), in case of short-term differential overpressures after LOCA, resulted in some structural reinforcements of the concrete floor slabs of the pressurizer and surge line compartments.

For the safeguard and emergency systems, some analyses were needed for mid or long term criteria. These safety criteria, which are not verified in the accident analyses (commonly focussing on short term criteria), are often imposed by design limits of specific system components (tank capacity, maximum allowable temperature, etc.). For the PUSGR at the Doel 2 plant, typical examples are: the tank capacities of the auxiliary feedwater system and its make-up system, the LPSI flow rate during recirculation (delivered by the RHRS pumps), the boration system capacity needed to reach cold shutdown, the thermal loads and temperatures of the component cooling and raw water cooling systems, the spent fuel pool thermal loads and temperature.

These system capacity verifications are often confronted with an additional complexity due to the twin character of the Doel 1 and 2 plants. Indeed, some systems are common to both units (e.g., the component cooling system, the spent fuel pool cooling system) and this requires to carefully select the plant operating states for both units simultaneously, in order to determine the most penalizing system configuration. Moreover, scarce information was found available about the original design evaluation, but AVN could take advantage of the experience gained in his assessments performed in the frame of previous projects (e.g. SGR).

3.9 Mechanical analyses of primary circuit components

Besides the demonstration of the structural integrity of the replacement steam generators, structural and mechanical analyses are required to demonstrate that the other components of the Reactor Coolant System are in compliance with the licensing criteria and requirements for the PUSGR conditions. Basically, there are two main reasons for requiring those re-evaluations. Firstly, the PUSGR lead to modify some of the Nuclear Steam Supply System (NSSS) parameters, also referenced to as the plant operating parameters (see paragraph 3.2 above), which are the fundamental parameters used as primary input in the NSSS analyses. Secondly, the design transients that are used for the analyses of the cyclic behaviour of the NSSS components are also modified following the PUSGR.

The re-evaluation of the Reactor Coolant System components under the PUSGR conditions requires as a necessary first step (1) to re-perform the Reactor Coolant Loop analysis the objective of which is to calculate the loads applied to the piping, piping nozzles, component nozzles, supports and restraints as a result of the different loadings acting in the revised level A, B, C and D service conditions and (2) to re-determine the LOCA hydraulic forcing functions for the revised initial plant conditions.

For all the plants, Leak-Before-Break analysis of the Reactor Coolant Loop piping has been performed before SGR. The demonstration of the Leak-Before-Break following the evaluation procedure of the draft Standard Review Plan 3.6.3 requires, among others, to determine the critical crack size margin and the applied loads margin. Both normal operating loads and the faulted loads are used as inputs to this evaluation. As the re-analysis of the Reactor Coolant Loop required by PUSGR leads to re-evaluate the operating loads as well as the faulted loads, the Leak-Before-Break margins previously calculated are not expected to remain unchanged. Then, a re-evaluation of the Leak-Before-Break analysis of the reactor coolant loop is deemed necessary.

The re-evaluation of the Reactor Coolant System components under the PUSGR conditions also considers the following. As permitted by the application of the Leak-Before-Break to the Reactor Coolant piping, the Utility took the opportunity of the SGR for modifying the steam generators supports originally installed to cope with the breaks postulated in the Reactor Coolant piping (e.g., removal of some snubbers or tie-rods at the lower support). As a consequence thereof, the dynamic re-analyses of the Reactor Coolant Loop account for the modification brought to the Reactor Coolant System support.

In addition, the Utility also took the opportunity of the re-analyses required by the PUSGR to finalize some chapters of the demonstration of the structural adequacy of the Reactor Coolant System components under stretch-out conditions. Plant operation under stretch-out conditions leads to define specific design transients. However, plant operation under stretch-out conditions may also be considered to occur during a sufficiently short time when compared to the operating time under normal operating condition that only the few most probable (level A and B) transients are taken into account. Hence, the revised set of design transients required by the PUSGR includes specific stretch-out transients.

Consideration of plant operation under stretch-out conditions has also a consequence on the determination of the LOCA hydraulic forcing functions. Indeed, in the second stage of the LOCA event when the primary coolant depressurizes down to the saturation pressure, the amplitude of the hydraulic forces is roughly proportional to the difference between the initial internal pressure and the saturation pressure. Then, considering plant operation under stretch-out conditions leads to redefine the initial plant conditions to be assumed for calculating the LOCA hydraulic forces since the most penalizing effects are associated to the lowest initial reactor coolant temperatures.

Classically, a detailed re-analysis of the Reactor Coolant System components is not carried-out but the re-evaluation for the effects of the revised design conditions is only performed on the most limiting locations of each component with regard to the stress values, i.e., the primary stress intensities, the ranges of stress intensities and the fatigue cumulative usage factors. As a prerequisite, the selection of those most limiting locations requires that the results of the original analyses are sufficiently documented in the original stress reports for allowing to estimate the effects of the revised design conditions on the original stress values. As the re-evaluation is usually performed in accordance with a later edition of the ASME Boiler and Pressure Vessel (B & PV) Code than the one used in the original design, the changes in the Code are also taken into consideration.

The specific application of that re-evaluation procedure to the Doel 2 case raised however some difficulties as outlined in the following.

The original design basis of the Doel 2 plant did not require any structure, system or component to withstand the effects of earthquake. Indeed, seismic and geologic considerations had led to conclude that the maximum seismic ground motion due to earthquake was sufficiently low at the Doel site for not including seismic criteria in the design basis. Moreover, at the time when the design basis of the Doel 2 plant was defined, the US NRC rules (Appendix A to 10 CFR 100) did not request the maximum vibratory acceleration of the Safe Shutdown Earthquake (SSE) at the foundation of any plant to be at least 0.1g. However, in the frame of the first 10-year safety reassessment of the Doel 2 plant, a Safe Shutdown Earthquake has been defined for which the maximum vibratory acceleration at the foundation was set to 0.058g and the Reactor Coolant Loop piping was verified to have the necessary structural resistance to withstand the SSE using the Level D stress criteria of the ASME B & PV Code, Section III.

The Reactor Coolant Loop piping was originally not designed to the ASME B & PV Code but well in accordance with the USAS B31.1 Code. According to the USAS B31.1 Code, only the stresses due to pressure and thermal expansion in normal operating conditions are compared to the allowable stresses. However the reactor coolant loop piping was also verified to the ANSI B 31.7 Code. This evaluation included the assessment of the primary stresses (design conditions), secondary stresses and peak stresses, and the fatigue analysis for the specified normal and upset transients.

The reactor pressure vessel internals of Doel 2 were designed prior to the introduction of Subsection NG of the Boiler and Pressure Vessel Code Section III and, as a consequence thereof, a plant specific stress report was not required. Moreover scarce information was found available about the original design evaluation. For the SGR at Doel 2, AVN took advantage of the experience gained in his assessment of the re-evaluations performed for previous SGR to allocate most of his efforts to the re-evaluation of the reactor pressure vessel internals. More specifically the re-evaluation of the internals under LOCA condition raised concerns. Less usual analysis procedures such as the collapse load analysis using the LS DYNA computer code were deemed necessary to demonstrate the structural adequacy of some components of the internals. Moreover, it was found necessary to revisit the concept of the arbitrary pipe break classically required by AVN to be postulated in the design basis of the internals after successful application of the Leak-Before-Break.

4 CONCLUSION

The licensing of the SGR or the PU of a NPP is a long and difficult process. Moreover, the present Belgian legislation introduces imperative constraints for the reception of the studies and the implementation of the project (fixed delays, no option of follow-up). Considering merely the aspect of licensing studies, keystones for the success of that process are:

- adequate organization and project management involving all concerned organizations (Utility, Designers, TSO);
- planning of the consecutive phases of the licensing process; this planning should take account of the complexity of the licensing tasks, the degree of innovation of the proposed methodologies and the available manpower within each organization;
- early approval of not yet licensed methodologies;
- timely execution of the safety studies in compliance with the planning;
- good communication between the involved organizations all along the licensing process.

Any shortcoming in one of them can endanger the timely acquisition of the license or the subsequent permission to start up the nuclear plant. In order to follow the planning and to improve the organization and the communication, a project coordination has been set up, with an objective of pragmatism and the power to take decisions.