Neutron fluence at the reactor pressure vessel wall –

a comparison of French and German procedures and strategies in PWRs

U. Jendrich (GRS), N. Tricot (IRSN)

Abstract: While the neutrons within the core may take part in the chain reaction, those neutrons emitted from the core are basically lost for the energy production. This “neutron leakage” represents a loss of fuel efficiency and causes neutron embrittlement of the reactor pressure vessel (RPV) wall. The latter raises safety concerns, needs to be monitored closely and may necessitate mitigating measures. There are different strategies to deal with these two undesirable effects: The neutron emission may be reduced to some extent all around the core or just at the “hot spots” of RPV embrittlement by tailored core loading patterns. A higher absorption rate of neutrons may also be achieved by a larger water gap between the core and the RPV. In this paper the inter-relations between the distribution of neutron flux, core geometry, core loading strategy, RPV embrittlement and its surveillance are discussed at first. Then the different strategies followed by the German and French operators are described. Finally the conclusions will highlight the communalities and differences between these strategies as different approaches to the same problem of safety as well as economy.

1. INTRODUCTION

1.1 Definition of flux, flux density and fluence

There are three terms to be distinguished: Neutron flux in [n/s)], which defines the number of neutrons (n) passing a predefined area per unit of time, neutron flux density in [n/(cm² s)], which is flux per unit area, and neutron fluence in [n/cm²], the number of neutrons accumulated during an irradiation period within a unit area. For the determination of fluence at the vessel wall, only those neutrons with energy beyond some defined threshold value are counted. This energy threshold is conventionally set to E = 1 MeV in Western countries and E = 0.5 MeV in Russia and Eastern European countries. All fluence values given in this paper will correspond to neutrons with an energy E > 1 MeV. The order of magnitude of the maximum fluence at the vessel wall after 40 years of operation is 10^{19} n/cm² [1,2].

1.2 Neutron embrittlement and fuel efficiency

Neutron embrittlement describes the degradation of the mechanical properties of the ferritic steel of the RPV wall by neutrons. This effect is visualized in figure 1, which shows the “brittle-ductile transition”. This is typical for all ferritic steels, the most important type of material for all heavy steel constructions. The figure illustrates that fast, brittle fracture of the material with low energy consumption is possible under severe loading at low temperatures. For the RPV this implies a possible failure of the vessel wall under conditions as a pressurized thermal shock. At high temperatures any fracture will only proceed by ductile crack growth with a high energy consumption needing much higher loads. This implies that failure of the vessel is practically impossible under any operational or accidental loading in this temperature range. In between those two regimes there is the brittle-ductile transition, usually indexed by a transition temperature TT, where the material behaviour is in between those two extremes and where mixed mode fracture might occur. In this picture, neutron embrittlement means a shift of the transition curve to higher temperatures and eventually a lowering of its upper shelf. This extends the temperature range, where brittle failure is possible. So the margin to failure under a pressurized thermal shock will become smaller with increasing fluence, eventually limiting the lifetime of the vessel for safety reasons. [3]

The RPV are all made thick-walled forging rings (or sometimes plates) of low-alloyed ferritic steel, welded together by similar welding materials. Typical values for the TT of these materials in the unirradiated state are around 0°C for materials produced in the 60ies. TT subsequently decreased due to technological progress to values usually below -20°C, sometimes even below -40°C. The major contributors to embrittlement in the material (i.e. copper and phosphorous impurities as well as high nickel content) were revealed in the early seventies. Thus some of the vessel materials made before these findings and irradiated to a “high” fluence level exhibited irradiation induced shifts of the TT in the range of 150°C. Later steel production could benefit from the research findings and typical shift values were mostly well below 100°C, further decreasing with technological progress, aiming at increasingly “cleaner” steels of “optimized” composition. This shall be born in mind, when RPV of different vintage are compared.

Fuel efficiency is also related to neutron flux outside the core since neutrons leaving the core are lost for the fission process in the core. This neutron “leakage” cannot be totally avoided because this required boundary conditions impossible to achieve in reality: either a core of infinite size or an effective reflector for neutrons installed all around the core. However, this leakage can be decreased with respect to the “standard” core loading patterns by special core loading schemes which are called “low leakage cores” (LLC). Naturally these LLC will also reduce neutron flux at the vessel wall. Therefore similar core loading patterns have been introduced to mitigate neutron embrittlement. So both objectives, mitigation of neutron embrittlement of the RPV and fuel efficiency require similar measures. However, due to plant specific priorities different approaches may be taken.

1.3 Determination of RPV fluence

In principal the neutron flux density distribution within the RPV is calculated with different codes. This is compared and validated by experimental results from dosimeters placed at some points within the vessel [1, 4, 5]. Two types of codes are used for this:

- The “source code” describes the fission processes in the core, creating neutrons with a certain spectrum of energy typical for each fissile isotope. Thus, this code has to take into account the amount of each type of fissile isotope in the core which is changing continuously with burnup and the spatial distribution of power history within the core.
- A “transport code” describes the transport of the neutrons from their source to any point of interest. The created neutrons interact with other atomic nuclei, i.e. they may be absorbed or subject to elastic and inelastic scattering processes within the core and on the way to and through the vessel wall. For this purpose the material and the geometry of the vessel internals have to be represented in detail for this calculation.

This sounds rather straightforward, however, the determination of neutron flux or fluence, necessary to create and validate these codes, is a complicated matter, since there is no direct method to count neutrons or to measure their kinetic energy. Thus only indirect measures can be used from dosimeters placed somewhere close to the vessel wall. They contain some amount of an isotope, which is transformed into a radioactive isotope by interaction with neutrons. The produced radioactivity of the dosimeter during a defined irradiation period can then be measured. One widespread isotope is ^{54}Fe, an isotope of iron, which transforms into ^{54}Mn, a radioactive isotope emitting gamma rays. The structural materials within the vessel also contain ^{54}Fe. Thus, this isotope can also be used for a so called “auto-dosimetry”, by scraping off samples from the structural materials.

However, to calculate from this radioactivity the neutron fluence accumulated in the dosimeter, one needs to know the probability of the nuclear reaction of the initial isotope with neutrons, i.e. the “cross section” for this reaction. Unfortunately this cross section is a function of the energy of the neutrons. But where to know their energy from, if there is no direct measure ? Just take some dosimeters containing materials of known cross sections !? In fact a long iterative process has been necessary: many experiments with different neutron sources and different dosimeter materials had to be evaluated by trying to make their results consistent with each other. In principle this process allows to approximate the “true” cross sections and “true” neutron spectra of the sources step by step with higher accuracy. In fact, this is a continuous and still ongoing process.

The resulting data for the fission spectra of the fissile isotopes and the cross sections of all kinds of isotopes of interest have been stored in large data banks called libraries. Although these libraries are evaluated on an international level, the nuclear data are still the major source of uncertainties for the resulting fluence. In spite of the vast amount of data and benchmarks accumulated today, these uncertainties are still considered to be in the range of 20% for a comparison of a single measurement with a calculation. In the nineties major “corrections” to some of the data, in particular the cross section of iron have been made. These corrections have generally led to an increase of the calculated fluence values for the RPV wall of PWRs. In the context of a major re-evaluation of the design fluence for the French reactors an increase of 40% has been reported [6]. Much smaller corrections have been made for the German plants.

1.4 Surveillance of neutron embrittlement of the RPV

The majority of the vessels in Western countries has an individual irradiation surveillance programme for each vessel. This includes specimens made from the original or “representative” materials which are irradiated within the vessel close to its wall. These specimens receive a neutron flux density which is higher than the flux density at the vessel wall itself by a “lead factor” > 1. So the specimens accumulate the same fluence earlier than the RPV wall. Thus their state of embrittlement is “in advance” with respect to the RPV wall. Several sets of different specimens are included in such a programme. These sets are withdrawn during plant outages, when predefined fluence levels have been reached, which should be well distributed between 0 and the expected maximum fluence level at the end of life of the vessel. The sets are usually composed of specimens for mechanical tests of strength and toughness of the material as well as dosimeters, all together encapsulated for corrosion protection. These capsules are attached at the outside of the core barrel or thermal shield in most cases. [7, 8]

The large uncertainties of the absolute fluence values do not apply to the evaluation of RPV toughness, since this is mainly based on the surveillance specimens, thus on a relative measurement of fluence. This suffers much smaller uncertainties, since similar systematic errors from the nuclear data enter the calculation of vessel fluence and of specimen fluence. This is a major advantage of surveillance programmes in each individual RPV with specimens close to the vessel wall. The full uncertainty of the fluence values has to be taken into account, if results from different reactors with different geometries or neutron spectra are compared.

2. THE INFLUENCE OF RPV AND CORE GEOMETRY ON FLUX DENSITY

2.1 Core geometries and the resulting flux distribution

The ideal geometry of the core to minimize the loss of neutrons at its boundary would be a sphere. However, due to technical reasons, all light water reactors have nearly cylindrical core geometries. This geometry is approximated by a parallel, vertical arrangement of bar shaped fuel assemblies (FA). For the French and German PWRs they have square cross sections of 20 to 23 cm side length. To increase the output of successive generations of plants from 360 to 1450 MW_{el}, the size of the core had to be enlarged. Therefore the length of the FA has been increased stepwise from about 3 to 4.50 m and the number of the FA has been increased from 121 to 157, 177, 193 and 205. The latter has been realized at the French N4 plants only. With respect to the core with 193 FA, it has just 3 additional FA at the end of each principal axes, i.e. at 0, 90, 180, 270 and 360°. The arrangement of the 4 smaller cores is sketched in figure 2. [2, 9]

The space between the core and the wall is filled with water and some sheets of steel (core baffle, core barrel and eventually thermal shield), which act as strong absorbers of neutrons. Besides water also moderates neutrons, i.e. it decreases their mean energy. Thus the width of this space governs the level of neutron flux density onto the vessel wall. In most PWRs there is roughly half a meter of water and steel between the core and the vessel wall decreasing the neutron flux density by about 3 orders of magnitude.

Since the shape of the core configurations realized in the plants is not perfectly round, the distance between core and vessel wall varies. The azimuthal distribution of the flux density at the vessel wall has maxima and minima, as is illustrated in figure 2. The figure shows a ratio of maximum to minimum flux density of 2.5 to 5 for most cores geometries, while only the core with 177 FA has a low ratio of about 1.2. For most of the core geometries the “hot spot”, where the flux density is maximal coincides with the principal core axis, conventionally defined as 0°. However, there are exceptions with a hot spot at 45° or 10°. All this is true only for “standard” core loading patterns and may change drastically with loading scheme as described in the next chapter. [10]

2.2 RPV fluence levels of French and German PWRs

The maximum fluence at the vessel wall after 40 years of operation for the French 900 and 1300 MW_{el} plants is about 7 and 4 to 5 • 10^{19} n/cm² respectively. This corresponds to the level of RPV designed by Westinghouse. This level has been reduced slightly for the latest 1450 MW_{el} units of the N4 type.

In Germany a decision has been taken to reduce the neutron fluence at the vessel wall to a level below 10^{19} n/cm². The reason for this decision were considerable uncertainties about the extent of and the contributing factors on neutron embrittlement which still existed in the early 70ies. This decision has finally been laid down in the RSK Guidelines for PWRs in 1974. To respect this upper limit, Siemens designed RPV with a wider water gap for all their large PWRs with more than 1000 MW_{el}. This additional layer of 20 to 25 cm water reduced the maximum fluence after 40 years of operation another order of magnitude: from about 4 to 7 • 10^{19} n/cm² for the first plants in Germany to a design level as low as 0.5 • 10^{19} n/cm². However, the extreme large diameter of the RPV for the new generation of PWRs required major improvements of the steel making and forging technology for the huge forging rings of high quality. [2, 11]

3. THE INFLUENCE OF CORE LOADING ON FLUX DENSITY

3.1 Some basic trends

A few facts can already explain qualitatively the differences between the loading patterns [1, 12]:

- Neutrons within the core have a free path length of the order of the width of a FA. So a major part of the neutron flux outside the core, i.e. the “neutron leakage”, comes from the peripheral FA.
- The neutron emission rate of a FA or a fuel pin is proportional to its power output. In a given environment this depends on the density of fissionable nuclei in the fuel: the neutron emission rate increases with ^{235}U enrichment of the fresh fuel and decreases with increasing burnup. The different starting levels have to be taken into account when comparing FA at a certain burnup with different initial enrichment.
- There are some advantages for larger cores: Due to their larger volume to surface ratio they suffer relatively smaller neutron leakage. Therefore they can become critical at a lower density of fissionable nuclei, i.e. they need a somewhat lower initial enrichment of the fuel and can reach higher burnup and longer cycles for the same initial enrichment. Besides they offer more flexibility for the core loading.

3.2 “Out-in” and “in-out” core loading patterns

In standard or “out-in” loading patterns fresh FA are placed at the periphery of the core. This configuration produces maximum neutron leakage with more than 80% of this neutron leakage stemming from the fresh peripheral FA. At the hot spots of the vessel wall, up to about 40% of the neutrons may come from the one single FA closest to it. [12, 13]

In contrast to this, low leakage cores (LLC) correspond to “in-out” patterns, where fresh FA are placed inside the core, while those FA placed at the periphery have already undergone one or more cycles of operation; see figures 3c and 4b as examples. These loading schemes may decrease the flux density versus the vessel wall by 50%. Therefore changing to a LLC pattern looks like a “win-win” situation: It mitigates irradiation embrittlement and saves fuel cost. LLC core loading strategies have been adopted at many plants already 10 to 20 years ago. However, there are restrictions to be observed, explaining why many operators did not jump on it immediately and why the out-in pattern is still applied in many plants.

For a better understanding we shall have a look at the radial power distribution of different core loading schemes:

If hypothetically the entire core was loaded with identical, fresh FA with the same enrichment level, this would create a strong maximum of local reactivity and power in the centre of the core. In order to achieve a more homogeneous power distribution within the core, FA with a lower density of fissionable nuclei are placed in the centre, either by using fuel of a lower enrichment level, as it is done for first cycle core loadings of new plants or by using FA with some advanced level of burnup, according to the out-in loading scheme.

Even in this loading scheme most peripheral FA do not reach more than 60 to 80% of the mean power level in the core, because they are not fully embedded within other FA. In fact there is always a steep decrease of power output from one fuel pin to the next towards the core edge. The lower output of the peripheral FA is counterbalanced by other FA in the centre producing more than 100% power. The FA with the maximum power in the core, called the "hot channel", may produce more than 120%. [14]

If a LLC core loading is realized, the total power of the core usually remains the same, however, the horizontal gradients are amplified. The exact figures depend on several factors: the initial enrichment of the fuel and the burnup level of the peripheral FA, the position of the fresh FA and the size of the core. While the power level of some peripheral FA might become as low as about 30%, the hot channel might reach about 150%.

The increase of the allowed maximum power of a FA has partially been possible due to some technological developments in the past, e.g. an optimization of coolant mixing within the FA, thinner and hence more numerous fuel rods per FA or an increase of the coolant flow rate through the core. Besides best estimate modelling resulted in a "more realistic" and mostly less conservative definition of the safety margins based on statistics. Nevertheless, the high power level of the hot channel in a core with very low leakage but high total power output may not always be achievable under the given technical and regulatory boundary conditions, while it is close to today's technological and regulatory limits in other cases. [15]

3.3 Fluence reduction to mitigate vessel embrittlement

As described in chapter 2 and shown in figure 2, the azimuthal distribution of the vessel flux density has sharp maxima for most core geometries. Therefore plants which primarily aim at the mitigation of vessel embrittlement may just place FA with an advanced burnup at the periphery of the core close to these hot spots to reduce the flux density at these spots. Thus the resulting approach is different from the in-out loading pattern, see figures 3 and 4 as examples for cores with 157 and 193 FA. Reductions of the flux density at the hot spot in the order of 20 to 40% (157 FA) and 30% (193 FA) have been achieved by these measures. The exact reductions mainly depend on the remaining power output of these peripheral FA close to the hot spot. Since the 3 FA at the end of the main axes of a 157 FA core contribute more than 80% to the flux density at the hot spots, the reductions could even be somewhat higher for this core type, if 3rd and 4th cycle FA were placed at these positions. [12, 16]

3.4 The trend towards higher enrichment and discharge burnup

Minimizing the costs of the fuel cycle was incentive to reduce the number of FA to be replaced each cycle. For this purpose burnup has been increased by raising the enrichment level of the fresh FA. These add more reactivity to the core and can operate more cycles. Thus less replacement FA are needed per cycle.

In the seventies most PWRs started with annual cycles, out-in loading schemes and rather low enrichment in the range of 3 to 3.5%. About 1/3 of the FA in the core had to be replaced each cycle. The average discharge burnup of FA was around 25 MWd/kg U in the seventies and was raised rather continuously since 1980. In Western Europe it was in the range of 30 to 35 at 1990 and at about 43 MWd/kg U in 2000. If annual cycles are maintained, ¼ of the FA or less has to be replaced at this level. Alternatively, some plants introduced longer operating cycles of 18 months and still replaced 1/3 of the core each cycle. With today's average enrichment levels in the range of 4%, average discharge burnups of about 50 MWd/kg U can be achieved. [17]

If the out-in core loading pattern is maintained and fresh FA are placed at the edge of the core a higher enrichment will also induce a significantly higher neutron flux density at the vessel wall. However, since fuel efficiency is the main objective, some kind of in-out core loading strategy has often been adopted at the same time. Besides, the reactivity of the fresh FA containing higher enrichment fuel is larger and may necessitate to embed the fresh FA by other FA with high burnup to limit power peaking. Thus, in these "advanced" core loading schemes a few fresh FA have to be distributed evenly within the interior of the core, see figure 5 [14].

Another way to limit the reactivity of fresh FA with relatively high enrichment is the use of burnable reactivity poisons, i.e. the inclusion of neutron absorbing elements within the fuel matrix. This absorption takes place by a nuclear reaction of these elements, thereby using up or "burning" the absorbing nuclei. The most common element is gadolinium, boron or others may also be used. Since longer operating cycles need a larger surplus of reactivity of the fuel at the beginning of the cycle, burnable poisons are especially useful to counterbalance this large surplus for longer than annual cycles.

3.5 The use of MOX fuel

MOX fuel usually contains a mixture of different plutonium isotopes issued from recycling of burnt fuel embedded in a matrix of UO_2, which may be made from depleted uranium. This MOX fuel emits more neutrons per fission than UO_2 fuel. This may have to be taken into account with regard to vessel fluence, if the MOX FA are placed at the periphery. However, in most cases the MOX FA amount only to some 30% of the total core and they are not placed at the periphery of the core. In these cases the impact on vessel fluence is not significant. [12]

4. COMPARISON OF STRATEGIES AND PROCEDURES IN FRANCE AND GERMANY

4.1 Core loading strategies

4.1.1 The situation in Germany

There are 13 PWRs in operation in Germany, where the small core sizes with 121, 157, 177 FA are represented by one unit each, while the majority of the large PWRs with 1200 to 1400 MW_{el} all have a core with 193 FA. In principle there is no common core loading strategy among the plants, since they are operated by several operators. Nevertheless, the technical, economical and regulatory boundary conditions are similar leading to similar strategies in many cases. Some of these common conditions are [17, 18]:

- Nuclear power plants account for about 30% of the electricity in Germany and mainly provide base load, although some load following is done by some units.
- Back-end costs are very high and are calculated essentially on the basis of the number of disposed FA, regardless of burnup. In contrast to this, burnup is a major factor of these costs taken into account in other countries.
- Furthermore, there is no regulatory limit on burnup in Germany. Instead there are limits to the oxide thickness of the fuel rod cladding (17% ECR criterion) and the failure rate in case of a loss-of-coolant-accident (10% of the fuel rods). The latter basically restricts the internal pressure in the fuel rods, which increases with burnup. The compliance with these criteria has to be shown for each individual reload.

Under these conditions most German PWRs aim at very high burnup and have adopted LLC loading strategies for economic reasons. However, even for an individual plant, there is no such thing as a “standard core loading pattern” or an “equilibrium core”. Core loading scheme and number of replaced FA may therefore change from cycle to cycle, see figure 5 for examples. German units generally operate in annual cycles with some stretch-out at the end. However, sometimes extra short cycles of 6 months are introduced in order to optimize fuel efficiency: Some FA which have already achieved a high burnup can be driven to even higher burnup, if a reshuffling of the FA is done after 6 months.

Some units introduced LLC right from the beginning; others had a long transition period with a slow and careful increase of the maximum power of the hot channel. The fuel savings amount to about 3 to 5 FA per cycle and flux density reductions are about 50% with respect to an “out-in” core loading scheme. Thus, for the large German PWRs, the extrapolated maximum fluence at the vessel wall of at the end of 40 years of operation is in the range of $0.25 \cdot 10^{19}$ n/cm², i.e. about a factor 20 lower than for most plants with Westinghouse design.

To achieve these large flux density reductions FA at the periphery may be in their fifth, sixth or even higher cycle, depending on initial enrichment and individual power history of the FA. Enrichment levels are in the range of 3.5% to 4.4% ^{235}U, where the maximum level of 4.4% has just been introduced recently in some plants. Sometimes fresh FA of two or more different levels of enrichment are loaded, some of them with gadolinium as burnable poison. Today average discharge burnup levels in the range of 50 MWd/kg U have been attained with annual cycles. Short cycles of 6 months allow to increase this number by some 10 to 15% [14].

The situation is different for the oldest plants in Germany, which still had high vessel fluence levels and introduced a LLC as a strategy to limit vessel embrittlement. The oldest unit at Obrigheim illustrates an extraordinary and extreme case of flux reduction: it has a small core of just 121 FA, and successively introduced a LLC, then placed 12 dummy assemblies at the end of the main axes (analogous to figure 3b), and finally placed 24 dummy assemblies at the periphery all around the core. These dummy assemblies contain rods of pure stainless steel instead of fuel and helped to reduce the maximum flux density at the vessel wall to less than 10% of the original value. Besides, the hot spot has been shifted from the main axis towards some 20°. Of course the total fluence at the vessel is still appreciable due to its accumulation during the first cycles, when no reduction measures had been taken. However, a reduction of the vessel fluence of about a factor of 3 has been achieved, which ensures a sufficient margin of the vessel with respect to brittle failure, even if extrapolated to 40 years of operation. [5,19,10]

4.1.2 The situation in France

Compared to Germany the situation is quite different: one single operator for large series of nearly identical 900 and 1300 MW_{el} units and four blocks of the latest N4 type. Since three quarters of the electricity is produced by nuclear power, most plants operate in the load following mode. For each type of unit there are clearly defined loading patterns, which have, however, been modified several times to allow for the use of MOX fuel, to increase burnup and finally to mitigate vessel fluence. The different loading schemes and their influence on vessel fluence are described in the following. [12,16]

- 900 MW_{el} units

All the 34 units started with an out-in loading, annual cycles, and an enrichment of 3.25%. 1/3 of the core (52 FA) had to be replaced each cycle in this loading scheme called "standard", which also defines the reference level for any modification of the vessel fluence. The target mean batch discharge burnup was 33 MWd/kg U.

In 1987 a higher enrichment of 3.7% ^{235}U has been introduced, where only ¼ of the FA has to be recharged and the batch burnup is increased to 43 MWd/kg U. Since an out-in loading strategy has been maintained at first, the higher reactivity of the peripheral fresh fuel increased vessel fluence by about 10%. Since 1992, however, two steps of flux reduction measures have been taken by placing irradiated FA at the ends of the main axes. This reduced the flux density at the hot spot of the vessel by about 20% for the configuration illustrated in fig. 3a and about 40% for the configuration illustrated in fig. 3b.

In 1987 the use of MOX fuel from recycling started for some of the units of this series. With an enrichment of the MOX fuel equivalent to 3.25% ^{235}U these FA operate 3 annual cycles. The part of the MOX FA is always restricted to a maximum of 30% of the core, the rest being UO_2 fuel. At first these MOX FA have been introduced in the framework of a 3.25% - 1/3 of the core loading scheme. Since 1994 the same type of MOX FA has also been mixed with 3.7% ^{235}U FA operating 4 cycles. In both cases the MOX FA have been placed in the second line next to the core edge, where their influence on the vessel fluence is not significant.

In 2001 a further increase of the enrichment to 4.2% ^{235}U fuel has been introduced for the 6 oldest units, called the "CP0" sub-series, still operating with UO_2 only. The increase of the target batch burnup to 45 MWd/kg U is only marginal, however, the higher enrichment of these FA allows for longer fuel cycles of 16 to 18 months, increasing availability of these units due to a lower number of planned outages. More than half of the fresh FA contain some fuel rods with Gadolinium to absorb the large surplus of reactivity at the beginning of the cycle. These are distributed in the interior of the core. The core loading strategy applied foresees to place three 2nd and 3rd cycle FA at the ends of the main axes (see figure 3b). The predicted reduction of the flux density at the hot spot is about 40% with respect to the reference level.

- 1300 MW_{el} units

As their smaller predecessors, these 20 units started with out-in loading, annual cycles and a replacement of 1/3 of the FA per cycle and a target mean batch discharge burnup of 33 MWd/kg U. Due to their larger core size, the necessary enrichment was only 3.1% compared to 3.25% for the 900 MW_{el} units.

In 1996 an increase of the enrichment to 4% was introduced, which allowed batch burnups of 45 MWd/kg U and an elongation of the cycle to 18 months. The calculated effect on vessel fluence for an out-in loading scheme was a plus of 10% at the hot spots. Two irradiated FA have been placed at the end of the diagonal as illustrated in figure 4a, which reduced the flux density by about 30%. More than half of the fresh FA contains some fuel rods with Gadolinium and is distributed in the interior of the core or the second line next to the core edge.

- 1450 MW_{el} units

The four N4 plants will progressively reach their equilibrium cycle. The equilibrium loading pattern is associated to an annual cycle and a replacement of ¼ of the FA per cycle. The enrichment is 3.4%. The target mean batch burn up is 43 MWd/kg U. Nevertheless, in the near future (2007), this fuel management might evolve towards a new one, associated to an increase of the cycle length from one year to 20 months and a replacement of 1/3 of the FA per cycle. Proposed target enrichment levels are in the range of 4.2% to 4.5%. This new fuel management scheme might also be taken as an opportunity for the utility to reduce RPV fluence.

4.2 Determination of vessel fluence and surveillance of neutron embrittlement

4.2.1 The situation in Germany

In Germany Framatome ANP GmbH as the successor of Siemens performs practically all the fluence calculations and the evaluations of the surveillance programmes for the power reactors, including the mechanical tests of the surveillance specimens.

Neutronic calculations are performed with commercial codes developed in the USA, using custom-made libraries of nuclear data. The source term is calculated in three-dimensions, fuel pin by fuel pin to evaluate the power distribution within the core and the resulting burnup of each loading cycle. In contrary to this, the standard transport calculations are done with DORT, basically a two-dimensional code. To represent 3-dimensional distributions, the results of 2-dimensional calculations in horizontal planes are multiplied with the results of two-dimensional calculations in vertical planes and divided by the results of one-dimensional calculations in radial direction. This is a widespread procedure delivering reasonable good results for the flux density distribution close to the horizontal mid-plane of the core, where axial gradients are small. This is the most important area, where the flux density attains its maximum and the surveillance specimens are situated. [5]

In general, 4 or more sets of those surveillance specimens are installed in the “older” reactors with higher fluence levels, while the standard surveillance programme for the plants of the “pre-Konvoi” and “Konvoi” type with their low fluence level only comprises two sets, to be withdrawn at about 50 and 100% of end of life fluence. The standard capsules contain ^{54}Fe and ^{93}Nb as dosimeter materials. These complement each other in the sense that the half lives of their activation products are largely different (312.5 days for ^{54}Mn and 16.1 years for ^{93m}Nb). Thus, together they can cover both, long and short term irradiation history of the capsule.

Practically all the surveillance sets of the standard programmes have already been withdrawn by now. Except for the two oldest PWRs they have confirmed a low level of embrittlement with projected maximum transition temperatures at end of life below 40°C. This level has recently been introduced as “design level” for fluence values up to 10^{19} n/cm² in the German code KTA. The surveillance programmes just have to show that real values are below this limit. This approach replaces the former one where the actual irradiation induced shift of the transition temperature was compared with a code curve. This code curve was supposed to be conservative, however, the measured shifts were sometimes larger than predicted due to high nickel content of some weld materials or due to relatively large scatter. In any case the absolute values of the transition temperature were low and these results above the code curve had no real safety significance. [8, 20]

The two oldest plants have both reduced vessel fluence more or less drastically. Besides they have scraped off samples from vessel internals or the cladding to improve dosimetry and carried out special irradiation programmes to better characterize the irradiation behaviour of their core welds. These are the most critical materials of the vessel with respect to embrittlement. With these measures it could be shown that the transition temperatures of these welds will remain below some 100°C. Additionally they have taken mitigating measures to reduce possible accident loading (hot leg injection and preheating of the emergency core cooling water) and to improve non-destructive testing of the beltline region of the vessel. [10, 19, 21, 22]

4.2.2 The situation in France

In France the tasks are distributed among different parties: Framatome as the manufacturer has performed the fluence calculations for the design of the vessels and their surveillance programmes. Their approach is similar to the one of Siemens (now Framatome ANP GmbH), as far as the codes and the libraries are concerned. On behalf of the operator EDF, the French research organisation CEA has developed their own source code as well as a three-dimensional transport code TRIPOLI. The latter is based on the Monte Carlo method, which principally allows for higher precision modelling. Refined reference calculations are performed with TRIPOLI for each reactor type and each standard loading pattern. A simplified, one-dimensional code named EFLUVE has been developed by EDF for the fluence calculations for each individual cycle of each unit. It has the advantage of being a simple and fast tool, able to do quick calculations and parametric studies. This code is “validated” by comparison with results obtained with TRIPOLI. [4, 16]

While 8 sets of surveillance specimens have been installed in the 6 oldest reactors of the CP0 series, the standard programme for the later units comprises 4 sets plus 2 in reserve. The 4 sets are to be withdrawn at about 25, 50, 75 and 100% of end of life fluence. Each of them contains as many as 8 or 9 different dosimeter materials, 2 of them fissile materials. Their evaluation is commonly performed by CEA and EDF. [6, 23]

In accordance with international practice, the irradiation induced shift of the transition temperature of the vessel materials is predicted by an empirical formula based on the chemical composition of the materials. Based on the database of the French vessel production, custom-made formulas have been developed in France. They initially predicted maximum values for the transition temperature for the oldest 900 MW reactors just below 100°C for end of life fluence levels. By now most of the surveillance specimens of the whole 900 MW series have been tested and the results basically confirm the predictions or show lower than predicted values of transition temperature shift. Maximum absolute values for the transition temperature are now evaluated to be in the range of 60°C. This reduction compared to the first predictions is due to favourable results of the surveillance programmes as well as the fluence reduction measures. This moderate level of embrittlement also reflects the benefit of the “late start” of the main French PWR programme, taking advantage of major progress in materials research and steel manufacturing technology in the 60ies and early 70ies. [7, 23]

5. CONCLUSIONS

As far as neutron fluence and vessel embrittlement are concerned, two different approaches have been followed in France and Germany:

French units have largely been constructed according to Westinghouse design with a relatively high fluence level and an extensive surveillance programme. This has been accompanied by the development of sophisticated methods for fluence calculations. The large series of identical units helped to limit the efforts per unit and to improve the statistical basis of the results. In contrast to this, in all but the three oldest German plants the vessel fluence level has been reduced by design to such a level, that the surveillance programmes could be limited, just to ascertain that design level embrittlement is not exceeded. For the fluence calculations commercially available codes are applied in accordance with international practice. For the oldest plants, special measures have been taken to decrease uncertainties on vessel fluence and material properties, reduce neutron flux onto the vessel and eventually to mitigate accident loading of the vessel.

It shall be pointed out, that all the vessels of French and German units will attain only low to moderate levels of embrittlement compared to some older units of American or Russian vintage, which attain transition temperatures in the range of 150°C.

As far as fuel strategies are concerned operators in both countries follow the general trend to higher enrichment and burnup. Up to 30% MOX FA are loaded in many plants. While vessel fluence may increase with enrichment in traditional out-in loading schemes, MOX is used in such a way that this has little effect on vessel fluence. Increasing enrichment is often accompanied by the use of some FA with gadolinium as burnable poison.

Nevertheless, there are also some differences in the core loading strategies in both countries: While EDF defines standard core loading patterns for each series of units, there is no such thing for German plants, not even for an individual unit. Due to their particular cost structure and the absence of a direct regulatory limit, German operators generally tend to push the burnup to higher levels. Today a mean batch discharge burnup of about 50 MWd/kg U has been attained, compared to 43 to 45 MWd/kg U in France. While most German plants have adopted low leakage cores to increase fuel efficiency, core loading schemes applied in French plants are still basically of the out-in type. However, second or third cycle FA are placed close to the hot spots of the vessels to mitigate their embrittlement. At enrichment levels = 4% EDF takes advantage of the higher surplus of reactivity to elongate fuel cycles up to 18 months and reduce downtime for their CP0 and 1300 MW units, while most German plants still stick to annual cycles and simply do more cycles with the same FA. Some German plants even introduced short cycles of 6 months to further increase burnup and thereby fuel efficiency.

6. REFERENCES

1. D. Beretz:
   Spectres neutroniques, flux et doses d'irradiation
   Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled « Irradiation des matériaux : expérience et simulation », Paris, 3rd june 1999

2. C. Leitz, J. Koban :
   Development of Reactor Pressure Vessel Desing, Neutron Fluence Calculation, and Material Specification to Minimize Irradiation Effects
   Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An Intern. Review (Vol. 3), edited by L.E. Steele, American Society for Testing and Materials, Philadelphia, PA, 1989, ASTM STP 1011, p. 130-144

3. K. Kussmaul:
   Der Integritätsnachweis für strahlenversprödete Reaktordruckbehälter
   VGB Kraftwerkstechnik 62, p. 1060 - 1076, 1982

4. D. Beretz, S. Saillet, A. Bache: Dosimétrie du Programme de Surveillance
   Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled "Fluence Cuve", Paris, 8th of mars 2001

5. OECD/NEA, Nuclear Science Committee, Task Force on Computing Radiation Dose and Modelling of Radiation-induced Degradation of Reactor Components:
   Computing Radiation Dose to Reactor Pressure Vessel and Internals, State-of-the-art report OECD/NEA/NSC/DOC(96)5, OECD Paris 1997

6. C. Courteau : Organisation et méthodologie adoptée par EDF pour les calculs de fluence cuve
   Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled "Fluence Cuve", Paris, 8th of mars 2001

7. Brillaud, C., Hedin, F.:
   In-service evaluation of French pressurized water reactor vessel steel
   Proc. of the 15. Intern. Symp. on the Effects of Radiation on Materials, in Nashville, Tenn., 19 - 21 June, 1990, edited by R. E. Stoller, A.S. Kumar, D. S. Gelles,
   ASTM STP 1125, American Society for Testing and Materials, Philadelphia, PA, 1992

8. KTA 3203: Überwachung des Bestrahlungsverhaltens von Werkstoffen der Reaktordruckbehälter von Leichtwasserreaktoren, Edition 06/2001, Carl Heymanns Verlag, Köln

9. World Nuclear Industry Handbook 1991
   Special Nuclear Engineering Internation Publications
   Reed Business Publishing Group 1990

10. E. Polke:
    Fluence Surveillance by Scraping Samples from the Inner Surface of the Thermal Shield in the Nuclear Power Plant Obrigheim in Germany (KWO)
    Proc. of the 8th ASTM-Euroatom Symposium on Reactor Dosimetry, Vail, Colorado/USA, August 29 to September 3, 1993, edited by H. Farrar IV, E.P. Lippincott, J.G. Williams, D.W. Vehar
    ASTM STP 1228, American Society for Testing and Materials, Philadelphia, PA, 1994

11. RSK-Leitlinien für Druckwasserreaktoren einschließlich der Rahmenspezifikation Basissicherheit, Gesellschaft für Reaktorsicherheit (GRS) mbH,
    Geschäftsstelle der Reaktorsicherheitskommission (RSK), Köln,
    3rd edition 1981 (1st edition 1974)

12. J.C. Lefebvre, P. Leroy, C. Rieg, H. Schaeffer, J.C. Nimal, R. Lloret:
    Neutron Fluence Management to Optimize Pressure Vessel Lifetime
    IAEA specialists' meeting on Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An Intern. Review (Vol. 4), Balatonfüred, Hungary, Sept. 26-28, 1990, edited by L.E. Steele
    ASTM STP 1170, American Society for Testing and Materials, Philadelphia, PA, 1993

13. M. Chiron:
Fluence Cuve – Méthodes de calcul de reference
Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled "Fluence Cuve", Paris, 8th of mars 2001

14. Personal communications of U. Jendrich with Dr. Johann, 2000 and 2002

15. Fuel Safety Criteria Technical Review
Results of OECD/CSNI/PWG2 Task Force on Fuel Safety Criteria
NEA/CSNI/R(99)25, OLIS 20-Jul-2000

16. B. Roulier, F. Marcel:
Gestion de la fluence cuve en exploitation industrielle
Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled "Fluence Cuve", Paris, 8th of mars 2001

17. R. Güldner, C.P. Bathelmes:
How a Nuclear Vendor Can Contribute to Improving Nuclear Power Plant Competitiveness
VGB PowerTech Vol. 82, 5/2002, p. 76-81

18. Betriebserfahrung mit Kernkraftwerken
VGB PowerTech Vol. 82, 5/2002, p. 27-68

19. U. E. Kaun, F. K. A. Koehring:
Investigations and Measures to Guarantee the Safety of a Pressurized Water Reactor's Pressure Vessel Against Brittle Fracture
Radiation Embrittlement of Nuclear Reactor Pressure Vessel Steels: An Intern. Review (Vol. 2), ASTM STP 909, ed. by Steele, L.E., American Society for Testing and Materials, Philadelphia, PA, 1986

20. R. Langer, W. Backfisch, R. Bartsch:
Results of Evaluations of Irradiation Surveillance Programs of Light Water Reactors in Germany
Paper presented at the IAEA specialists' meeting on "Irradiation Embrittlement and Mitigation", Madrid, Spain, April 26 – 29, 1999

21. Nagel, G., J.-G. Blauel:
Evaluation of the standard master curve for fracture toughness determination
Nuclear Engineering and Design 190 (1999), p. 159-169

22. R. Langer, R. Bartsch, G. Nagel:
Irradiation Behaviour of Submerged Arc Welding Materials with Different Copper Content
Proc. of the 19. Intern. Symposium on the Effects of Radiation on Materials in Seattle, Washington, 16-18 June 1998, edited by M.L. Hamilton, A.S. Kumar, S.T. Rosinski und M.L. Rossbeck, ASTM STP 1366, American Society for Testing and Materials, West Consohocken, PA, 2000

23. H. Churier-Bossennec, S. Saillet, B. Houssin:
Présentation du Programme de Surveillance des effets de l'Irradiation des cuves (PSI)
Paper presented at the Technical Meeting of the French Society for Nuclear Energy (SFEN) entitled "Fluence Cuve", Paris, 8th of mars 2001