Technical paper:

Aging of materials during plant operation:

Preventive measures in the design of the EPR™ reactor

The design of the EPR™ reactor takes benefit of significant advances derived from R&D performed on mechanisms of materials ageing, field experience and improvements gained during repair or replacement of components of units in operation. This brings confidence to the resistance to in-service ageing of materials of the EPR™ reactor for the full lifetime of 60 years.

Key words: Maintenance & repair, engineering & design, material selection

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The selection of materials must take into account some constraints that are specific to Pressurized Water Reactors (PWR):

- large size components, such as the reactor vessel, the primary pump or the steam generators,
- use of various manufacturing processes for these components: forging, welding, cladding,
- need to achieve good corrosion resistance for stainless steels, as related to strict requirements put on (primary and secondary) water coolant and effluents chemistries,
- need for strong knowledge of materials in-service behaviour, to fulfill regulatory requirements.

All these considerations lead to the choice of commercial materials grades which are well known and easy to manufacture, rather than newer "high performance materials".

In order to reach the required quality level, nuclear construction codes (e.g. RCC-M, ASME, etc require:

- reduced chemical composition ranges for main components,
- strict control of residual elements and inclusion content,
- severe non destructive examinations at various stages of manufacturing,
- additional testing for the first manufactured component (technical qualification),
- definition of the main parameters controlling material properties, in technical manufacturing programmes of suppliers.

Brittle fracture

One of the most significant regulatory requirements is the prevention of the risk of brittle fracture, defined as a sudden fracture without any preliminary significant overall deformation. As far as the material is concerned, brittle fracture prevention is controlled through the determination of notch impact toughness (Charpy tests) at several temperatures, at least down to 0°C, and ductility (i.e. elongation at rupture and reduction of area) through tensile tests. To prevent any risk of brittle fracture, strict requirements are imposed on materials at the procurement stage. As an example, for heavy components, additional margins have been introduced in the equipment specification by imposing a maximum RT_{NDT} value equal to -20°C at the procurement stage. Nevertheless, this is not sufficient in cases where ageing may reduce the resistance to brittle fracture through either an increase of the brittle-ductile transition temperature or an intrinsic reduction of fracture toughness. Further precautions have been taken to limit the effects of ageing, as described in the following paragraphs.

Corrosion

Materials corrosion is also an important issue not only from the view point of damage to the component, but also from that of generation of corrosion products.

Austenitic stainless steel grades selected for the EPR reactor are "very low carbon" grades or stabilized (i.e. containing titanium or niobium) grades to prevent sensitization during manufacturing operations.

The materials must provide a high degree of protection against general corrosion as well as local corrosion such as stress corrosion cracking, pitting or crevice corrosion.

Undesirable consequences of migration of corrosion products might also be significant. Deposition of such products on fuel elements or on heat exchanger surfaces might lead to ineffective heat transfer or formation of radioactive isotopes after activation inside the reactor core.

Strong efforts have been applied to the limitation of the cobalt content of materials so as to reduce reactor coolant irradiation source term and hence radiation exposure of maintenance personnel, especially for steam generator tubes for which an upper limit of 0.015% Co has been imposed. In addition, for main primary components surfaces directly in contact with the primary coolant, an upper limit of 0.060% Co has been set in stainless steels.

Potential cobalt release has also led to efforts to develop alternative materials to replace cobalt base alloys for hardfacing. These alternative materials are now available and can be used, particularly in valves.

Neutron irradiation

Neutron irradiation of materials induces point defect formation caused by the displacement of atoms in the crystal structure, as a result of collisions with neutrons. Irradiation temperature is an essential parameter, since at temperatures below 250°C, the possibility of healing by a recombination mechanism of point defects is reduced. At higher temperatures, other phenomena, such as void formation inducing swelling, may occur. At PWR operation temperatures (around 300°C), materials behaviour is mostly affected by the generation of point defects.

For the reactor pressure vessel (RPV – Figure 1) material, copper and phosphorus present as residual elements in the steel interact with point defects and are the cause of embrittlement. Beside the materials and its microstructure, other important parameters are the neutron flux and temperature. The consequence of irradiation on materials submitted to a high neutron flux (core of the RPV) is an increase of yield strength that shifts the brittle-ductile transition toward higher temperatures, therefore increasing the risk of brittle fracture.

Embrittling elements (copper, phosphorus, and possibly nickel which may have a synergistic effect) have been strictly limited in equipment specification since long ago.

For the EPR reactor, because of the large pressure vessel diameter, the neutron fluence for the RPV is reduced to low values, similar to these of the German Konvoi reactors. In addition, chemical limitations on copper and phosphorus contents in the steels have been made tighter. Thus, the ductile-brittle transition temperature RT_{NDT} expected after 60 years of operation remains below 30°C.

Figure 1. The reactor pressure vessel.

Internals

Material for the internals surrounding the core of the reactor is an austenitic stainless steel. These internals are cooled by the primary coolant but, as a consequence of γ irradiation which induces heating, their temperature lies between 300 and 400°C. Some areas are submitted to very high irradiation doses. The microstructure of austenitic stainless steels makes them more resistant to irradiation than the low alloy steel of the RPV. Meanwhile, the high level of irradiation may induce very significant changes in materials properties. The first noticeable effect is a hardening that increases as temperature decreases. Important changes in microstructure, especially near grain boundaries are caused by diffusion under irradiation of alloying elements, which may make the material susceptible to various kinds of corrosion. For the EPR a heavy reflector made of stainless steel was selected for the lower internals (Figure 2),

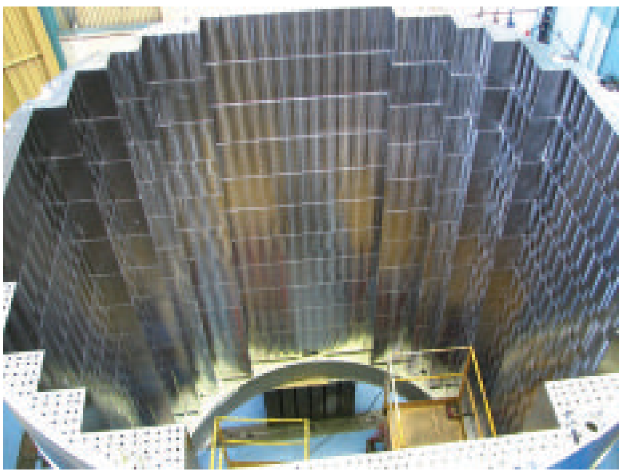

Figure 2. Lower internals for the EPR™ reactor (heavy reflector).

Figure 3. Example of a forged leg with integrated nozzle and elbow.

that reduces neutron "leaks" and reduces the neutron fluence for the reactor pressure vessel. A design based on the use of forged blocks, kept in place by tie rods kept away from the core, limits the risks of exposing bolts or welds to high neutron fluxes.

Thermal ageing

For some materials, diffusion and precipitation of some elements occur at operation temperatures. This so called "thermal ageing" induces a loss of ductility and a decrease in toughness. The two most significant instances are phosphorus precipitation at grain boundaries in ferritic steels and chromium demixtion in the ferritic phase of duplex austenitic-ferritic stainless steels.

Low alloy steels

Low alloy steels selected for the pressure boundary of heavy components (RPV, steam generators, pressurizer) may be affected by thermal ageing due to phosphorus segregation at grain boundaries, that takes places under some circumstances. Aggravating factors are embrittling residual element content (P, Sn, Sb, As), a higher temperature and coarse grains in HAZ's. Improvements in manufacturing (welding) processes were introduced several years ago to minimize this last effect.
For the EPR reactor, specifications on residual elements content have been made even more severe, which was made possible by significant progresses made in the steelmaking processes.

Cast austenitic-ferritic stainless steels

From an historical perspective, cast austenitic-ferritic stainless steels were mostly used for primary piping and the primary pump casing. These steels have a ferrite content ranging from about 10% to 25%, to avoid solidification cracking and obtain higher tensile properties.
Over time at service temperature, hardening of the ferrite phase takes place, inducing embrittlement. Cleavage can then be observed, even at elevated temperature (300°C).
Substantial research programs were undertaken in the 1980's to get a better understanding of the phenomenon and develop formulae for predicting fracture toughness of aged materials.
Significant changes took place in the past few decades. First, internal cleanliness of steels (inclusion content) has greatly improved and most of all, molybdenum containing grades are now forbidden for operation temperatures higher than 250°C.
For the EPR reactor, the molybdenum free grade (Z4 CN 20-09M), which is less susceptible to ageing, has been selected as the only material for the pump casing. The fracture toughness values that can now be guaranteed for this material, after extensive experimental programs, are now deemed satisfactory.
A most significant change is that cast austenitic-ferritic steels are no longer used for primary pipe branches; these were replaced by forged austenitic grades (Figure 3) which are not susceptible to thermal ageing.

Martensitic stainless steel

Martensitic stainless steels are mostly used for internals, bolting, and operating stems in valves. The consequences of thermal ageing has been observed in service, mostly valve stems ruptures. In some instances this phenomenon may occur very soon, sometimes after as little as 2 years in service, for the higher temperatures (350°C).
Therefore, martensitic stainless steels containing more than 13%Cr have been forbidden for some components submitted to loads at low temperature, but generally operating over 250°C.
For applications where higher mechanical properties are needed, a new grade was developed, which also benefits the EPR reactor.

Fatigue

Fatigue is another well known degradation mechanism. Cyclic loading may affect the lifetime of components. Cumulative straining induces slip bands on the surface and crack initiation.
Fatigue is a consequence of mechanical and thermal loading, but environmental effects also play an important part.
The prevention of crack initiation is obtained by using good design practice (avoidance of sharp transition geometries,…), fatigue analyses performed for stress concentration zones, taking into account all possible loading situations, and proper use of in service inspection.
Thus, fatigue cases met in service are, as a general rule, not related to transient loading of the reactor, but are mostly due to unexpected phenomena, such as pipe vibrations, thermal oscillation in mixing zones when two fluids at different temperatures meet, stratification and changing fluid levels.
Lessons learnt from field experience have been taken into account for the design of the EPRreactor. Among other things, further requirements for surface condition (roughness) of specific zones have been introduced to improve fatigue behaviour. New methods have also been developed for a better understanding

of phenomena occurring in mixing zones. They are now fully understood and controlled.

Stress corrosion cracking

Stress corrosion cracking may occur in service, with various features, when applied loads or residual stresses are high:

- strain induced corrosion of C-Mn steels or low alloy steels.
- chloride corrosion, with oxygen, for austenitic stainless steels,
- intergranular corrosion of nickel base alloys and austenitic stainless steel in caustic media,
- intergranular stress corrosion of nickel base alloy (alloy 600) in pure water at elevated temperature. This type of damage was first observed in steam generator tubes in transition zones, then on pressurizer instrumentation nozzles and on reactor pressure vessels control rod drive mechanism adapters.

After 15 years of accelerated tests in several laboratories, including those of CEA, EDF and AREVA, it was shown that alloy 690, as well as alloys 52 and 152 for welds, which have been used in substitution of alloy 600, are not susceptible to stress corrosion cracking during operation in the primary water.

For the EPR reactor, alloy 690, and alloys 52 and 152 for weld, have been selected for the steam generator tube bundle, pressure vessel penetrations and internal supports. Strict requirements for chemical compositions and heat treatments ensure a good reproducibility of properties and behaviour, close to the possible optimum.

Conclusions

Industrial constraints and safety requirements require that PWR components are manufactured with well known, easy to use materials. Possible changes are introduced with caution and are generally limited to chemical compositions and optimization of heat treatments.

Starting from the early choices made for the first generations of reactors, materials have been progressively improved, taking into account the field experience: chemical composition ranges have been reduced, as was made possible by progresses in steelmaking practices.

Although materials have not changed much, the knowledge of their properties has improved a lot, thanks to the research and development work performed these last 30 years. Suppliers have been associated to the work performed by AREVA, so as to identify manufacturing parameters that may influence in-service properties and behaviour.

Choices made for the generation 3+ EPR™ reactor are mostly derived from proved solutions already experienced in France and Germany. These choices bring confidence that a satisfactory resistance to various forms of ageing will be achieved for the 60 year intended lifetime.

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