Surveillance of Radiation Embrittlement of Reactor Pressure Vessels on the Basis of KTA-Safety Standards

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Summary

In contrast to the pressure vessels used in the conventional area, e.g. the chemical industry, reactor pressure vessels are subjected to neutron radiation, especially in the beltline area. The neutron radiation causes a change in the mechanical properties of ferritic steels and leads to an alteration of the transition temperature, which can be considered as a measure of the embrittlement of the steel used.

The Nuclear Safety Standards Commission (KTA) in the Federal Republic of Germany has recently developed the Safety Standard KTA 3203 "Monitoring the Radiation Embrittlement of Materials of the Reactor Pressure Vessel of Light Water Reactors", which is the topic of this paper.

With the requirements of limiting the neutron fluence at EOL to $1x10^{19}$ cm^{-2} ($E \geq 1$ MeV) and optimizing the chemical analysis of the steel used, the degree of embrittlement can be kept within pre-determined limits. Subsequently, specimens for tensile and impact testing are required to be taken from the original materials and subjected to irradiation.

The correlation between the time of withdrawal of specimen and the choice of lead factor will be examined and the surveillance program is also discussed.

The effect of irradiation on the steel used for the manufacture of the reactor pressure vessel will be shown by comparing the material characteristics before and after irradiation and the transition temperature shift as a function of the neutron fluence. The design of the reactor pressure vessel can be considered to be conservative, if it can be demonstrated that the transition temperature shift as determined is below the design trend curve.

Basically, there are only a few differences between the surveillance programs required by ASTM-E 185 (1979) and KTA 3203 (1983).

In contrast to steam boilers or to pressure vessels used in the chemical industry, reactor pressure vessels of light water reactors are subjected to neutron radiation, especially in the beltline area. This neutron radiation causes a change in the mechanical properties of ferritic steels. Neutron fluence values higher than $1x10^{18}$ cm^{-2} ($E > 1$ MeV) lead to an increase of the values of yield strength R_{eH} or $R_{p0,2}$, tensile strength R_m and hardness, and a decrease in the values of impact toughness, fracture constriction or reduction of area and elongation at fracture. In addition, the transition temperature, which can be considered as a measure of the embrittlement of a steel, is also shifted.

As a result of the property changes in the materials due to irradiation, it is necessary

1) to consider the degree of embrittlement in the design of the reactor pressure vessel and, in addition,

2) to experimentally verify the assumptions made for the design.

The experimental verification is done by irradiating samples removed from the original material with a lead factor. This irradiation program has so far been conducted for all nuclear power plants in the Federal Republic of Germany.

The Nuclear Safety Standards Commission (KTA), which has the task of setting up nuclear safety standards and promoting their use, assigned a working group with the job of developing a standard on surveillance programs for monitoring the embrittlement of pressure vessels in the beltline area. The principles for the establishment of nuclear safety standards and the steps involved are described in paragraph 7 of the Official Publication regarding the Establishment of a Nuclear Safety Standards Commission. Recently, the safety standard KTA 3203 "Monitoring the Radiation Embrittlement of Materials of the Reactor Pressure Vessel of Light Water Reactors" was completed.

Requirements for the Surveillance Programs

Based on the tendency of reactor steels to embrittlement, it can be assumed that starting from neutron fluence values of $1x10^{17}$ cm^{-2} ($E > 1$ MeV) the embrittlement has to be considered in the design of the reactor pressure vessel. In limiting the neutron fluence at the end of life (EOL, to $1x10^{19}$ cm^{-2} ($E > 1$ MeV), in reducing the contents, especially of copper, phosphorous and sulphur, in the chemical composition of the steel used and by adhering to a toughness concept, a high standard has been attained in solving the embrittlement problems. This can be seen as an explanation for a reduced testing program, when the predicted neutron fluence at the end of life has a value between 1×10^{17} cm^{-2} and 1×10^{18} cm^{-2} ($E > 1$ MeV). This is in contrast to the requirements of ASTM-E 185, which does not provide for a reduction in the surveillance program; only the number of surveillance capsules can change depending on the predicted transition temperature shift.

The number of specimens required can be represented as a function of the value of the neutron fluence at the end of life. In both programs, the same number of Charpy V-Notch specimens and tensile test specimens are required. For the reduced program, however, the specimens for the second base material and for the heat-affected-zones are not considered necessary.

The specimens to be chosen for the surveillance program must be removed from the original materials of those welded connections of the pressure vessel in the beltline area, which are expected to have the highest embrittlement at the end of life. The pre-determination of the embrittlement is based on design curves, which have been demonstrated to be conservative for reactor pressure vessels. These values have been calculated from data on the steels 20 MnMoNi 5 5, 22 NiMoCr 3 7 or ASTM A 533b, ASTM A 508 cl3 and ASTM A 508 cl2.

Tensile and impact test specimens can be removed either from an adequately long piece intended for the welders qualification test (work specimen) or from a weld specimen manufactured specially for the surveillance program. In either case, transverse test specimens shall be removed from the base materials I and II, which are to be welded together. In the case of Charpy V-Notch impact test specimens, the axis of the notch shall be perpendicular to the plane of the transverse and longitudinal directions or to the cylindrical surface.

The requirement to include the second base material, which can be from a different melting bath in the surveillance program, is a deviation from what has been practiced so far; in the past, only that base material had to be considered, whose calculated transition temperature shift at the end of life was more pessimistic. However, in the case of heated-affected-zones, it is adequate to examine that heat-affected-zone, which directly neighbours the base material with the more pessimistic value.

In addition, it must be ensured that sufficient material be available over and above that necessary for the regular tests; from this, specimens can then be removed for additional tests like fracture mechanics tests.

Fracture mechanics specimens or fatigue cracked impact test specimens are not part of the surveillance program. However, they can be of use in individual cases, especially if it can be seen from the evaluation of the first set of irradiated specimens that the design assumptions will be exceeded.

A minimum of 12 Charpy V-Notch impact test specimens and 3 tensile test specimens are required. However, further specimens including that of a correlation material can be considered in the program.

At least two sets of specimens for irradiation and one set to show the non-irradiated conditions are necessary.

The irradiated sets of specimen should only then be withdrawn, when the value of the neutron fluence for the specimens has reached 50% and about 100% respectively of the value, on which the design of the pressure vessel is based. The number of sets to be irradiated depends primarily on the lead factor chosen.

With a lead factor of 2 and the requirement that 50% or 100% of the final fluence value must be reached, the first set of specimens can be withdrawn after 10 years. This would correspond to the irradiation characteristics of 20 years of operation of the pressure vessel. The second set must be withdrawn after 20 years of irradiation and this corresponds to 40 years of operation of the pressure vessel.

If the results of the first set of specimens is required prior to the first recurrent pressure test of the reactor pressure vessel, i.e., after 8 years of operation, then either the sets of specimen or the lead factor must be increased. A reduction of the neutron fluence for the irradiation under 30% of the neutron fluence at the end of life is not recommended. The KTA-Safety Standard stipulates a lead factor of at least 3, but does not lay down an upper limit. This opens the possibility of withdrawing the first set of specimens prior to the first recurrent pressure test of the reactor pressure vessel.

In the light water reactors in operation today, there are channels or attachments within the core of the reactor pressure vessel, into which the capsules containing the specimens to be irradiated are inserted.

To determine the neutron fluence and temperatures on the specimen, neutron dosimeters and temperature monitors are also inserted into the capsules. The choice of the neutron dosimeter is left to the plant vendor, but he is obliged to use an acceptable method to determine the fluence value. In all the surveillance programs to date, iron-dosimeters have been used; this, however, has a threshold energy of only 4 MeV and a half life of 0.86 years.

Niobium-dosimeters are best suited, since they have a threshold energy of 0.4 MeV and a half life of 13 years.

One disadvantage of dosimeters with a radioactive source is that they have to be handled within the controlled access area.

There is the further possibility of removing samples directly from the inside wall of the reactor pressure vessel. The experience so far with the use of such samples has been very promising and this method could be a useful tool in future in determining the neutron fluence.

The irradiation temperature of the specimen has two components - the coolant temperature and the increase of the temperature through gamma absorption. However, it can be assumed that the gamma absorption component does not contribute more than 5 K and can therefore be neglected. Nevertheless, temperature monitors are used to measure the upper temperature limit; for this, alloys with known melting points are used. It is quite possible today, to use such monitors to measure temperature differences of about 5 K.

Application of the Results

The effect of irradiation on the embrittlement of the ferritic steels in the beltline is shown by comparing the material characteristics and the transition temperature before and after irradiation. The transition temperature is determined from average curves of impact testing with an energy corresponding to 41 Joule on non-irradiated and irradiated specimen. If it can be demonstrated that the transition temperature as determined is below the design trend curve, then the design of the reactor pressure vessel can be considered to be conservative. If, however, the values of the design curve have been exceeded, then additional evidence of safety and especially fracture mechanics tests are required. However, for the plants presently under construction, it can be assumed that these values will not be exceeded because of the strict requirements made. For reactor pressure vessels of the first generation, it may be necessary to carry out additional tests and have extended programs, if the embrittlement is of a higher degree than originally anticipated.

The evaluation of the results of each irradiated set of specimen must be presented to the authorized expert according to Para. 20 of the German Atomic Energy Act (AtG). In addition, this authorized expert also checks the surveillance plan with respect, e.g., to choice of materials and their history (origin and analyses), calculated neutron fluence at the inside wall of the reactor pressure vessel, lead factor and number of capsules.

All documents, tests and evaluations must be included in the documentation. The documentation must be complete and contain all details starting from the manufacture of the specimen to the results of the tests after irradiation.

It can be stated that the surveillance program is a redundant measure for the design of the reactor pressure vessel. By limiting the fluence at the end of life to 1×10^{19} cm^{-2} ($E > 1$ MeV) and by optimizing the chemical composition of the reactor steels, the embrittlement is kept within predetermined limits. For anticipated neutron fluence values of 1×10^{18} cm^{-2} ($E \geq 1$ MeV), it is therefore sufficient to irradiate tensile test and Charpy V-Notch impact test specimens from both base materials, one heat-affected-zone and the weld material and to compare these with the values used in the design.

It is hoped that KTA 3203 will thus help to increase the safety of nuclear power plants and to promote their acceptance.