DETERMINATION OF RADIATION DAMAGE TO STRUCTURAL COMPONENTS OUTSIDE THE CORE OF PWRs.

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ABSTRACT

With regard to the determination of the influence of neutron irradiation on reactor components an analysis was performed on the accuracies of neutron flux calculations and the spectral dependence of the displacement cross sections. The connection between mechanical property changes and neutron irradiation is discussed too. A comparison between calculated and measured azimuthal neutron flux distributions showed good agreement to within 11% or less. In the case of a 1300 MWe PWR the azimuthal factor for the atomic displacement rate decreases from about 5.5 at the core barrel to about 3.5 at the pressure vessel. Sensitivity studies of radiation influence in the pressure vessel of PWRs showed that the cross sections of H, O and Fe are most important. At the inner surface of the pressure vessel of the 1300 MWe PWR benchmark problem about 80% of the total displacements are produced by neutrons in the MeV region. The atomic displacements cause a decrease of the toughness of steel during irradiation. The property change as a function of fast neutron fluence is derived conservatively from experimental data. It is shown that the operational range of a reactor vessel in every case lies far outside the zone of crack instability which is determined from the fluence dependent toughness properties. One important point within the "Structural Integrity of Components" German research project is to quantify the relation between neutron fluence, neutron spectra, and the mechanical properties of steel alloys.

INTRODUCTION

The prediction of radiation influence on steel structures of reactor internals outside the core such as core barrel, grid plates or shroud and in the pressure vessel is an important task of reactor shielding. These structures are essential for safe reactor operation and directly associated with the life time of a power plant. Therefore, safety analyses must consider in advance all changes of material properties during operation. To avoid highly conservative results the energy dependence of both the neutron fluence and the appropriate damage functions must be taken into account.

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EXPERIMENTAL CHECK OF TWO-DIMENSIONAL NEUTRON FLUX CALCULATIONS

Besides the maximum fluence within a steel component, the radial, axial and azimuthal gradients are required as well. Because of the offset structure of the core and the complicated geometry of the steel components, the use of two-dimensional computer codes is indispensable.
What accuracies are found on the results of two-dimensional neutron flux calculations? To deliver a contribution to answer this very important question, an experimental check was performed of the azimuthal neutron flux distributions at the bounderies of the core-barrel and of the thermal shield for a 350 MWe PWR which had already been in operation for several years. During the refuelling period in 1976 a lot of very small steel samples were taken from the outer surface of the core-barrel and the inner surface of the thermal shield by a special machine. The sampling positions were distributed within an octal of the cylindrical shields. Preparation of the steel samples and measurement of their 54Mn-activity was performed by the KWU radio-chemical laboratory.
The calculation of the neutron flux density above 1 MeV was performed with regard to the shut down periods and the changing load and power density factors within the operation periods. The neutron spectra used for the calculation of the 54Fe(n,p)54Mn reaction cross sections were calculated by DOT-3 (ref.1).

Fig. 1 shows the geometry of the core and the neutron flux densities E > 1 MeV along the outer surface of the barrel which were calculated from the 54Mn-activity measurements. The curve in Fig. 1 shows the azimuthal neutron flux density distribution calculated by DOT-3. The comparison between the theoretical flux density values and those obtained by 54Mn-activity measurements shows at all sample positions good agreement within 11% or less. At all positions the discrepancies are smaller than the error of 20% in the measured flux values which is essentially caused by the error in the 54Fe(n,p) 54Mn reaction cross section. The experiment showed that the calculation of azimuthal neutron flux distributions with the use of the DOT-3 code is justified.

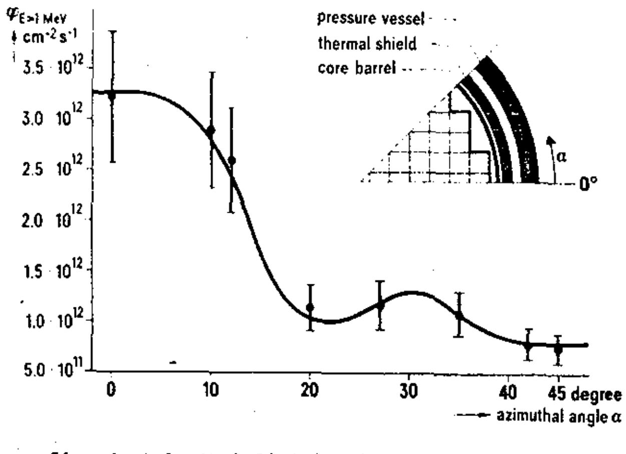

Fig. 1. Azimuthal Distribution of the Fast Neutron Flux Density Along the Outer Surface of the Core Barrel. Comparison of Measurement and Calculation.

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1300 MWe PWR-BENCHMARK, AZIMUTHAL DISTRIBUTIONS OF THE ATOMIC DISPLACEMENT RATES

In addition to the one dimensional benchmark problem[2] concerning a 1300 MWe PWR, some two-dimensional calculations in (r, θ) geometry were performed to obtain the relative azimuthal distribution of the radiation damage along the inner surface of the core-barrel and the pressure vessel.

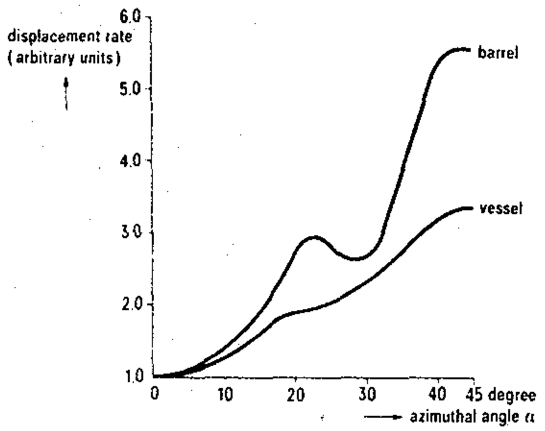

Fig. 2. 1300 MWe PWR-Benchmark, Rel. Azimuthal Distribution of the Atomic Displacement Rate

The results are shown in Fig. 2. The curves show the relative azimuthal distribution of the atomic displacement rate along the inner surface of the core-barrel and the pressure vessel. The strong azimuthal dependence of the radiation damage expressed in Fig. 2 is caused once again by the offset structure of the reactor core. The variable thickness of the first water shield zone results in an azimuthal dependence of the neutron spectrum as well. That means, that the contribution to the whole radiation damage caused by neutrons with energy above a certain energy boundery, i.e. 1 MeV, is a function of the azimuthal angle. The azimuthal dependence of not only the absolute displacement rate but also of the neutron spectrum decreases because of the great thickness of the second water gap.

CROSS SECTION SENSITIVITY STUDIES FOR NEUTRON DAMAGE

One-dimensional sensitivity studies of radiation damage in the pressure vessel to neutron cross sections have been performed for a 660 MWe PWR and a 1300 MWe PWR. In both cases the nuclides can be divided according to their importance into three categories. The most important cross sections for damage are those of H, O, and Fe. The nuclides of Cr, U-38, Ni, and Zr belong to the category of second importance, whereas the rest like Mn and U-35 are unimportant for damage calculations.

The sensitivity profiles calculated for the elastic and inelastic cross sections of the different nuclides show the importance of a narrow energy range between 5 and 7 MeV and a second smaller maximum at 3 MeV. To avoid an underestimation of the calculated neutron fluence the group cross sections must have a relatively fine structure within the important energy range, which has to be regarded in two-diemensional calculations especially.

CORRELATION OF RADIATION EFFECT BETWEEN SURVEILLANCE SAMPLES AND PRESSURE VESSEL

Atomic displacement cross sections have been determined for pressure vessel steel and stainless steel from the latest ENDF/B-4 neutron data.[3] The results are shown in Fig. 3 for the EURLIB-structure in 100 neutron groups. The two values for the thermal neutron group represent a soft spectrum in water and a hard spectrum in iron. The correlation analysis between different surveillance sample positions and the pressure vessel wall were done with the production of atomic displacements in the material.

Fig. 4 gives the displacement production in function of the neutron energy at several positions along the main radial axis of a 1300 MWe PWR. In the pressure vessel about 80% of the total damage is produced by neutrons in the MeV-region and 20% by keV-neutrons. At the position of the surveillance samples there is a small shift to the contribution of keV-neutrons as can be seen from Fig. 4.

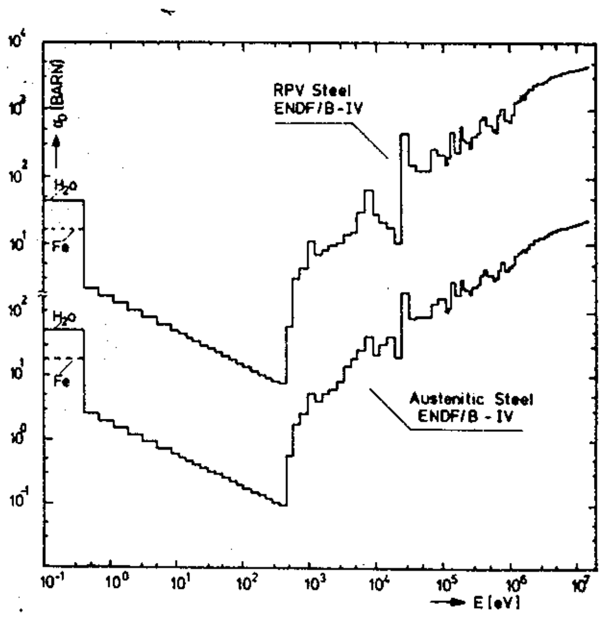

Fig. 3. Atomic Displacement Cross Section in RPV Steel and Austenitic Steel

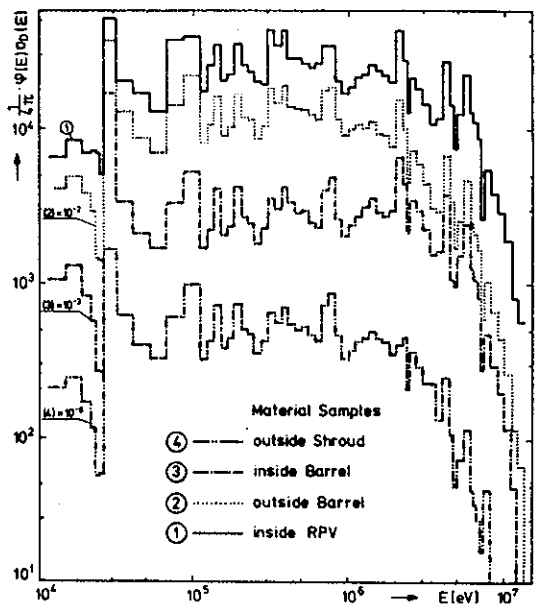

Fig. 4. Spectral Displacement Production Rate in Different Positions along the Core Main Axis

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In the range of operating temperatures of water-cooled reactors the high-temperature irradiation embrittlement caused by helium from (n, α)-reactions, and the void swelling are not relevant. Here we are concerned with hardening and embrittlement by displacements of atoms ("Low-temperature irradiation embrittlement").

The advantageous effect that the yield and ultimate strength of metals are increased by irradiation is well known from experiments. In the case of iron and low alloy steels, brittle fracture can occur at low temperature at stresses below yield strength. The probability of such an event, however, is very low.

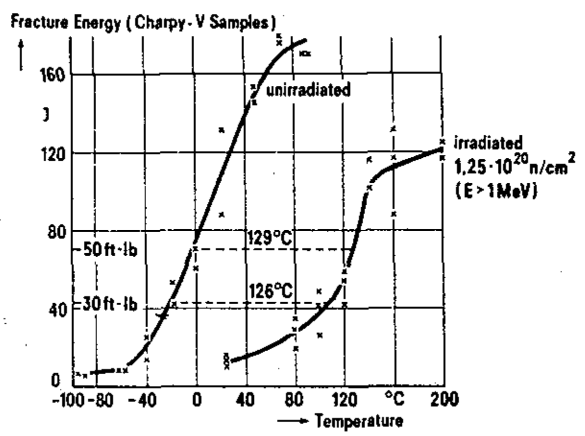

Fig. 5. Charpy-V Impact Transition Curves of RPV Steel 22NiMoCr37 (A 508 cl.2) (ref. 4)

Fig. 5 shows the fracture energy vs. temperature curves of charpy V-notch impact samples of a usual RPV-steel. It demonstrates the transition from lower toughness at lower temperatures to higher toughness at higher temperatures and the shift of the transition region by irradiation. For a more quantitative measure of the "nil ductility transition" (NDT) temperature the PELLINI drop-weight test is used. Also the fracture toughness, a measure for stability conditions of cracks, is evaluated.

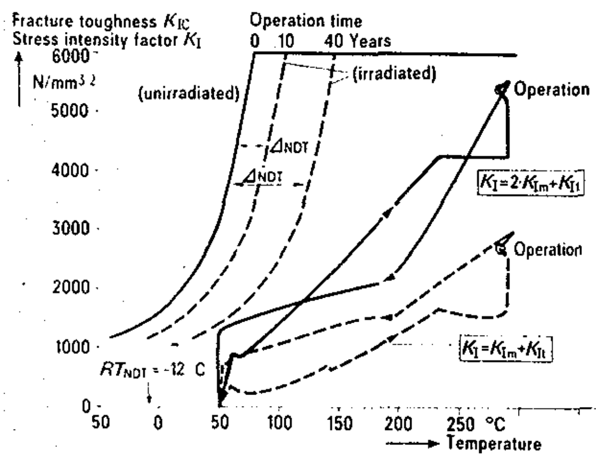

Fig. 6. Fracture Mechanics Diagram According to ASME-Code, Section III (ref. 5)

For RPV design it is demonstrated (Fig. 6) that the stress intensity (proportional to stress and defect size) caused by operating stresses on a hypothetical crack is always outside the zone of crack instability terminated by the fracture toughness vs. temperature curve. In addition to irradiation results of drop-weight and fracture toughness tests, the prediction of the irradiation-induced shift of that curve is based on charpy-V impact data. According to research results known at present this approach is conservative.

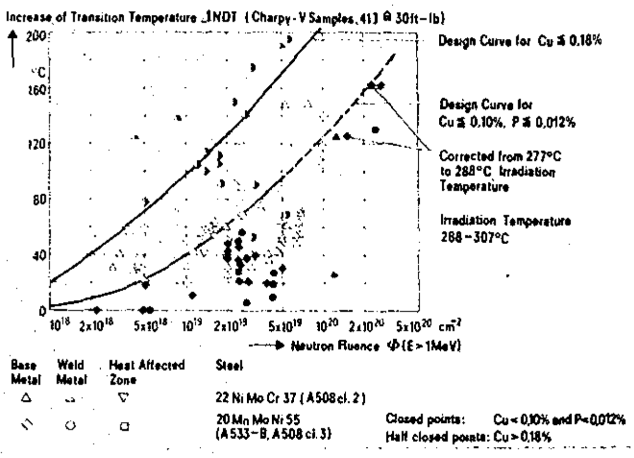

Fig. 7. Literature Data and Design Curves of Irradiation Embrittlement of RPV Steels (for detailed) references see ref. 6)

Fig. 7 shows most of the literature data on charpy-V transition temperature shifts of the two most common RPV-steels and related weldments. Two curves are drawn in this figure. It is shown that all data superceeding the upper curve correspond to materials with more than 0.18% copper content. The copper content, known to support irradiation embrittlement of steel, was specified to a maximum amount of 0.18% for RPVs of KWU in the past. Therefor this curve is a conservative design curve for those RPVs. The other curve is used for 0.10% max. Cu and 0.012% max. P content material. All data corresponding to this specification (valid for more recent RPVs) are below this curve.

The diagram is based on fast neutron fluence (E > 1MeV). But as far as spectrum informations were available, the data were corrected by damage function calculations for the spectrum at a PWR pressure-vessel. The result showed that the curves are conservative upper boundaries also for the corrected data.

RESEARCH PROGRAM ON RPV-STEEL IRRADIATION EMBRITTLEMENT

In order to quantify the safety margin of the reactor pressure vessel against brittle fracture there must be known, on the one hand, the loading inclusive emergency conditions, and on the other hand, the loading capacity of the material with regard to the quality degrading influences (Fig. 8).

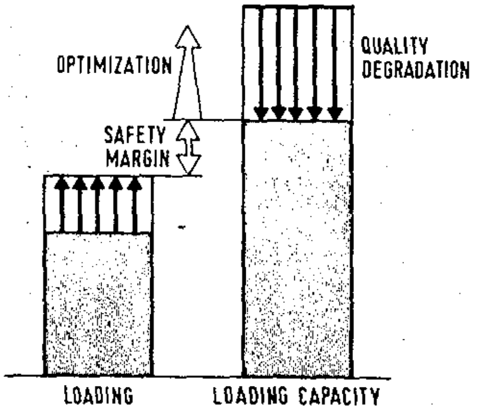

Fig. 8. Safety Margin

Quality degrading influences from fabrication and operation (irradiation) can cause reduced ductility and/or, eventually, crack formations. In addition to the considerable tests carried out in this field up to now, the transfer of results derived from the programs of research and surveillance on to the real component should be more quantified. This is among others due to variations in neutron spectra and time of irradiation (Fig. 9)

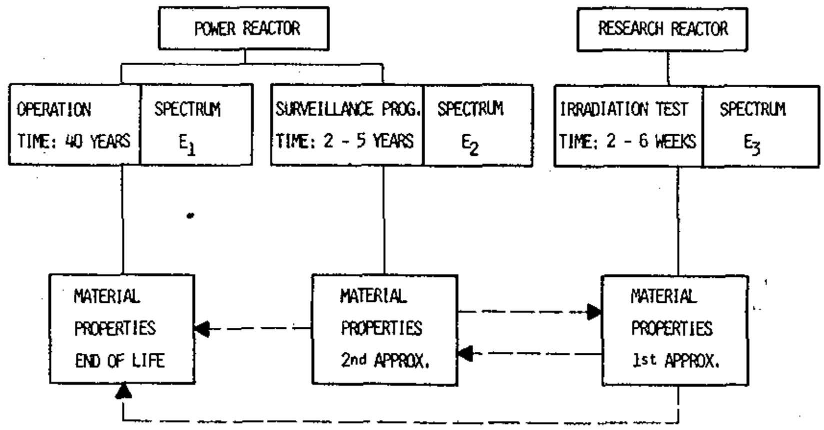

Fig. 9. Transferability of the Materials Properties after Irradiation under Different Neutron Fluxes and Spectra

In the research project "Structural Integrity of Components" getting now under way in Germany it is among other aims planned to examine the question more closely concerning the transferability of results derived from different conditions of irradiation. It is intended to develop the damage function in each case with defined material conditions. For this purpose, fracture mechanics specimens of definite material condition are, on the one hand, irradiated in the power reactor, and, on the other hand, on different positions in the research reactor in order to analyse a wide band of neutron spectra as much as possible (Fig. 10). With the aid of these results it will be possible to refer the results of surveillance and research irradiation programs more quantitatively to the behaviour of pressure vessels. Moreover, further steps within the scope of the research program are aimed at approximating test parameters to the real process in the component (base material, weld material, and heat affected zone (HAZ)) by including the influence of the coolant* and the mechanical loading. The investigation of the concurrent presence of the main influential parameters will contribute to approach the quantitative assessment of the safety margin more closely. Therefore, it is intended to irradiate fracture mechanics specimens of different material conditions and predamaged conditions under definite mechanical stress with and without protecting capsules in a power reactor, or in a corrosion loop in a research reactor.

* for theoretical consideration, since all low alloy steel components are cladded against the coolant.

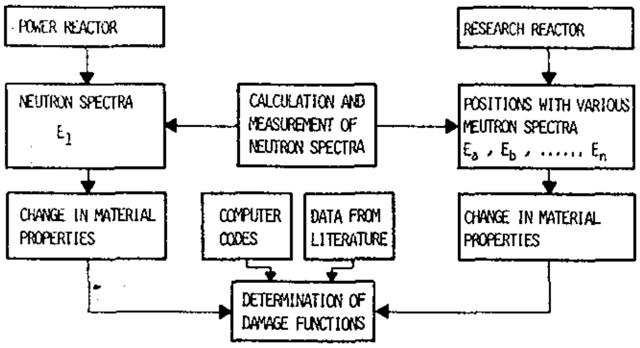

Fig. 10. Determination of Damage Functions According to Spectral Effects.

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