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In-Depth Embrittlement of a Simulated Pressure Vessel Wall of A302-B Steel

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ABSTRACT

A problem of major concern to the operators of nuclear reactors is the radiation-induced increase in the ductile-brittle transition temperature of the steel used for the primary pressure containment vessel of the reactors. Because of the self shielding and attenuation properties of the vessel material, a nuclear reactor pressure vessel will have a neutron flux and spectrum variation across its thickness. As a result of this variation, a pressure vessel should show various degrees of neutron-induced embrittlement throughout its thickness, and it is postulated that the embrittlement will be greatest at the inner wall and least at the outer wall. This phenomenon has been investigated by the irradiation of a large block of A302-B steel at the core face of a pool reactor in a position simulating the location of an actual pressure vessel. The steel block, 6 in. thick, was made to accommodate five equally spaced assemblies of Charpy V-notch specimens which, in turn, represented the vessel material at comparable positions.

The notch ductility test results of the irradiated specimens demonstrate a significant degree of embrittlement as well as a significant decrease in the degree of embrittlement through the simulated pressure vessel wall. However, the observed decrease is small when related to the respective variation in neutron dosage.

A correlation of the notch ductility data developed in this study to that between test reactor experiments and in-service power reactor conditions is indicated. Neutron dosage values in terms of n/cm^2 (>1 Mev) determined for positions in the test block as well as similar positions in water alone form the basis for this correlation. Thus, the values obtained enhance the validity of the >1 Mev criterion for reporting neutron dose in radiation embrittlement studies.

PROBLEM STATUS

This completes one phase of the problem; work on other phases is continuing.

AUTHORIZATION

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IN-DEPTH EMBRITTLEMENT TO A SIMULATED PRESSURE VESSEL WALL OF A302-B STEEL

INTRODUCTION

A problem of major concern to the operators of nuclear reactors, as well as to the agencies responsible for the safety and regulation of reactor pressure vessels, is the radiation-induced increase in the ductile-brittle transition temperature of the steel used for the primary pressure containment vessels of the reactors (1). Research conducted at NRL to determine the extent of radiation-induced embrittlement to these steels (2) is an extension of work related to the nil-ductility transition (NDT) temperature concept developed by Puzak and Pellini (3). The NDT concept relating the results of drop-weight and Charpy V-notch tests to service failure conditions has permitted the establishment of fracture safe operating conditions for nonnuclear pressure vessels (4).

Radiation-induced embrittlement test results obtained primarily from accelerated experiments conducted within or near the fuel core of various research and test reactors may not represent the true picture of embrittlement to reactor pressure vessels, since embrittlement to these vessels is not sustained under similar conditions of neutron spectrum and dose rate. A significant aspect of the embrittlement research, therefore, involves the development of the relationship of the experimental embrittlement data generated by notch ductility specimen testing to the case of damage sustained throughout the thickness of a heavy-walled reactor pressure vessel.

A nuclear reactor pressure vessel will have a neutron flux and spectrum variation across the thickness because of the self shielding and attenuation properties of the vessel material. As a result of this variation, a pressure vessel should show progressive degrees of neutron-induced embrittlement throughout its thickness; the embrittlement being greatest at the inner wall (nearest the fuel) and least at the outer wall. A pressure vessel which is significantly brittle only a short distance into its thickness may not present the potential hazard of one which is significantly brittle throughout its thickness.

The study made on the SL-1 pressure vessel provided the first opportunity to compare experimental data to the embrittled condition of an operating reactor vessel (5). While the SL-1 had a wall thickness of only 3/4 in. and provided test specimens representative of only the center portion of the wall, the satisfactory correlation of data between accelerated tests and the operating case was quite significant.

The extension of the correlation effort begun with the SL-1 must be made on other reactor vessels as they are taken out of service. Two such reactors, the Organic Moderated Reactor Experiment and the Army PM-2A, may be evaluated to provide pressure vessel materials for further correlation. Several theoretical studies are also being made in order to aid in this correlation. Comprehensive computer programs have been performed by Beeler (6), Shure (7), and Pawlicki (8) to predict the behavior of damage-producing neutrons in a variety of reactor vessel/thermal shield/water configurations. This report describes a study performed to experimentally measure the variation in degree of radiation-produced embrittlement throughout the thickness of a simulated steel pressure vessel wall by the irradiation and testing of appropriately positioned and shielded notch ductility (Charpy V-notch) specimens, and presents the results of that study.

Corollary to the problem of developing the relationship between test reactor data and actual power reactor vessel conditions is the problem of validating the neutron dose criterion used as one of the major parameters for assessing radiation-induced embrittlement. The presently accepted criterion for defining damage-producing neutron dose involves reporting only that portion of the neutron energy spectrum, incident upon the material being tested, having energies greater than 1 million electron volts (>1 Mev). The contribution of thermal neutrons to steel embrittlement is thought to be insignificant. Since most neutron flux computations are based on activation reactions having effective thresholds of 3 Mev or more, an extrapolation to 1 Mev based on the fission neutron spectrum is necessary. The inaccuracies of such a method are not lessened by the assumptions made in the definition of the neutron spectrum. Practically, however, there is little other reasonably accurate choice of spectra which may be broadly used as a basis for reactor flux calculations. The fission spectrum obviously is not the actual environment in every portion of a reactor neutron flux field. However, the neutron flux and dose data, which have been measured in various locations of several reactors in accelerated experiments using the fission neutron spectrum basis, appear consistent and correlate well with other radiation damage parameters.

Supplementary information regarding certain neutron dosimetry aspects of this study is also reported. A comparison is made of the thermal and fast neutron fluxes measured in water alone, and in the water-steel environment of a simulated pressure vessel wall. Data from this phase of the experiment also add to the validity of the >1 Mev criterion for reporting neutron exposure.

EXPERIMENTAL APPROACH

The experiment was designed to simulate a portion of a 6-in.-thick pressure vessel wall of a light-water-moderated reactor. The experimental test piece, a steel block large enough to minimize the effect of neutrons impinging from the sides, top, and bottom, was made with accommodations for five equally spaced assemblies of Charpy V-notch specimens throughout its thickness (Fig. 1). These specimens simulated vessel material at comparable locations. The assemblies, spaced 1 in. apart through the block, fit their individual "pockets" very closely. Plugs were used to fill the gaps between the assemblies and the top of the block (Fig. 2) so that, in effect, the entire test piece presented a solid, continuous piece of steel to the neutron environment. All assemblies were made to remain at a low temperature ($\leq$240°F) during the irradiation so as to obviate the temperature variable. These assemblies were designed for ease in withdrawal after irradiation as well as for ease in the separation of specimens from the assembly framework. In this way, specimens of material identical to the block itself, and accurately representing the condition of the material at the specific locations through the thickness of the block, were readily available for testing.

The 8-in.-square test block and the test specimens were made of ASTM type A302-B carbon steel, the most commonly used steel for reactor pressure vessels. The block was centered on the peak flux plane, 3-5/8 in. from the face of a pool reactor, as shown in Fig. 3. In this manner, the belt line of a light-water-moderated, 6-in.-thick, reactor pressure vessel was reasonably simulated. The irradiation was performed in the Industrial Reactor Laboratory (IRL), 5-megawatt pool reactor at Plainsboro, N.J., in a cooperative program with the U.S. Rubber Company, one of the owners of the IRL facility.

The experiment was performed in three separate irradiations. First, a series of thermal and fast neutron flux threshold monitors were irradiated for a short period of time at full power on the peak flux plane in water alone. The area monitored included that which was to be occupied by the test block but also extended beyond that on each side. This provided thermal and fast neutron flux values for the unperturbed water environment condition. Second, the steel test block simulating the pressure vessel wall was positioned at the face of the core as shown in Fig. 3, and a second series of thermal and fast neutron flux monitors were irradiated for a short period of time at full power. This time

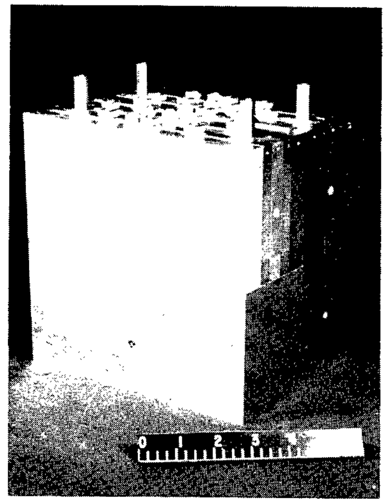

Fig. 1 - Test block simulating a reactor pressure vessel wall

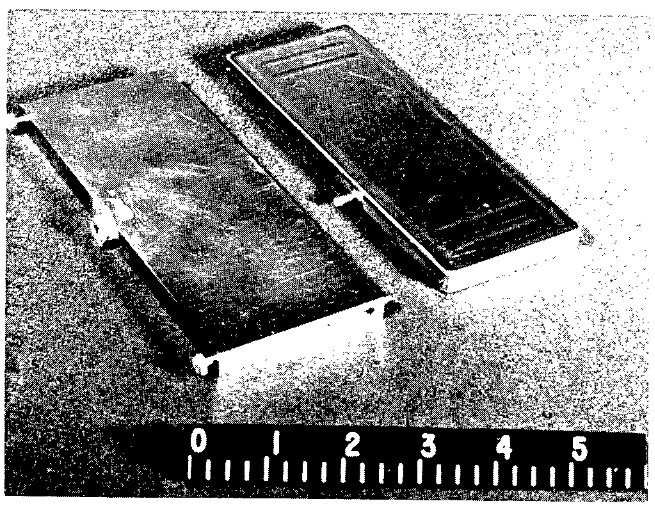

Fig. 2 - Charpy V-notch assembly (right) and top plug (left)

the monitors occupied positions in special plugs placed in the "pockets" of the block. One of these monitor-containing pocket plugs is shown in Fig. 4. This established the flux values for the perturbed (water-steel-simulated pressure vessel) condition. Third, with the monitor plugs removed, the block pockets were loaded with the Charpy V-notch assemblies, which also contained neutron flux monitors, and the entire experimental unit was irradiated to doses sufficient to produce a significant loss of notch ductility in the steel specimens. After irradiation, the assemblies were removed from the block and the specimens were taken from the assembly framework and tested to develop the transition temperature curves for each of the five locations.

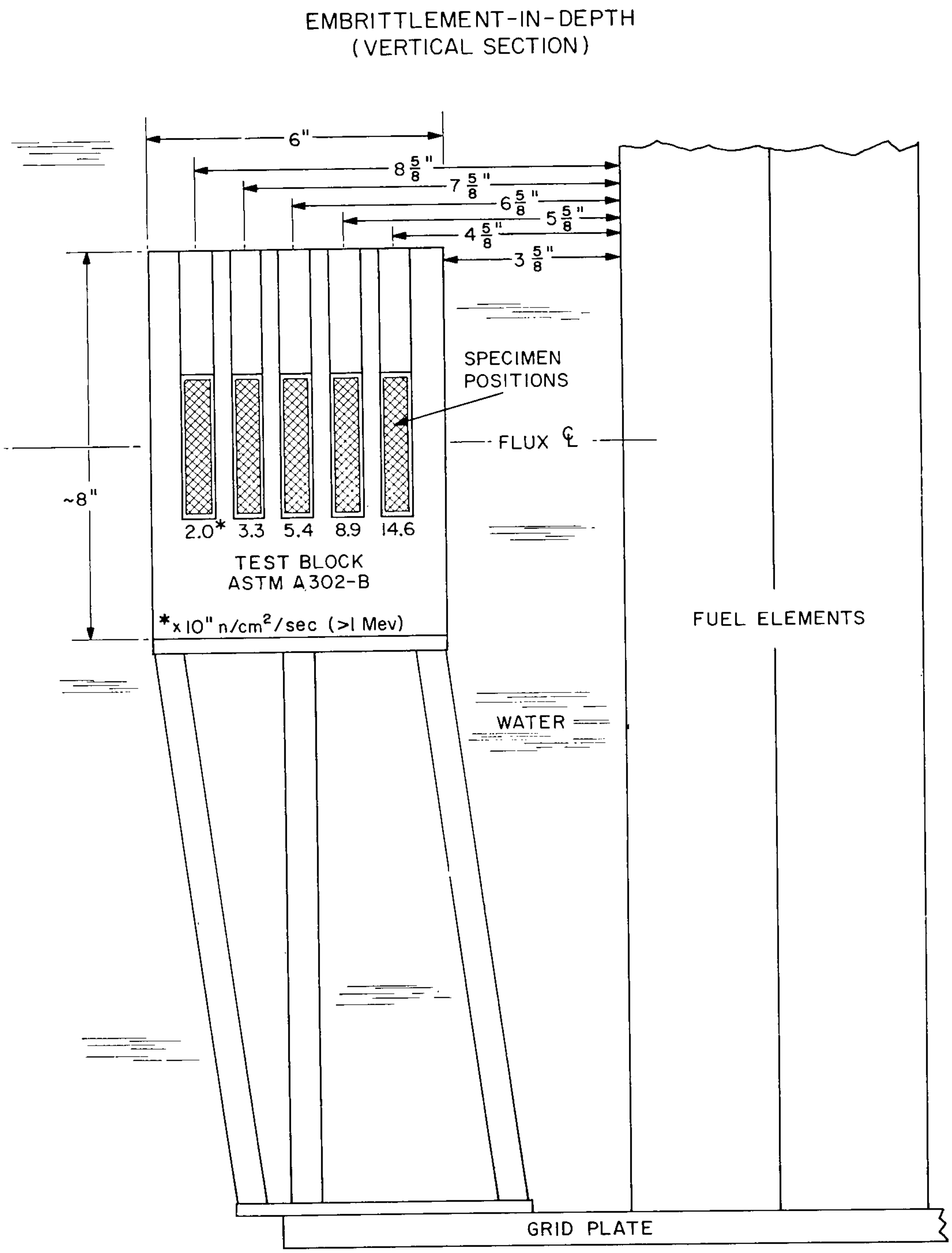

Fig. 3 - Schematic view of test block at face of pool reactor showing instantaneous flux values at specimen positions

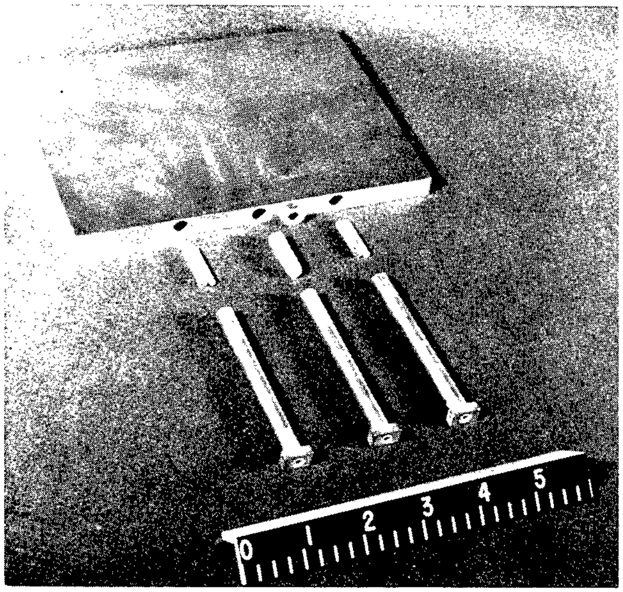

Fig. 4 - Exploded view of "pocket" plug and flux monitors

EXPERIMENTAL RESULTS

The radiation-induced transition temperature increases (increases in NDT), developed from the five Charpy V-notch specimen assemblies are shown in Fig. 5. Curve a represents the material closest to the fuel and thus shows the greatest increase. Curves b-e represent material located progressively farther from the fuel and show reduced embrittlement. Curves c and d show a separation above the 30 ft-lb energy level (commonly accepted NDT reference point), but are practically identical at that point and below.

The fast flux measured at the face of the test block corresponding to the inner wall of the pressure vessel was about 1.3×10^{12} n/cm^2/sec, so that the fast neutron accumulation at this point was about 3.71×10^{18} n/cm^2 (>1 Mev). This exposure would suggest a transition temperature increase of 145°F in the trend line of Fig. 6. This compares with an integrated neutron dosage of 2.61×10^{18} n/cm^2 for the position 1 in. within the simulated pressure vessel on the core side and a transition temperature increase of 120°F.

The relationship of the series of transition temperature increases developed in this study to those developed for the same material under the same temperature conditions but in accelerated in-core experiments is shown in Fig. 6. This figure indicates a steady buildup of embrittlement at low neutron exposures and a transition of the series of low exposure points into the upper side of the trend band developed for material in accelerated irradiations. It has been recognized that the trend band must tail off at very low exposure levels, but only limited data (such as that provided by data from the SL-1 reactor pressure vessel) have previously been available to define this effect.

When the series of transition temperature increases of this study is plotted on a linear scale, together with the points developed in the accelerated in-core experiments (Fig. 7), the true nature of these points is demonstrated. Significantly, the ratio of neutron dose to the transition temperature increase in the range covered by this experiment, about 7:2, is the same ratio exhibited by the data developed in the accelerated in-core experiments for the lower exposures below the knee of the curve.

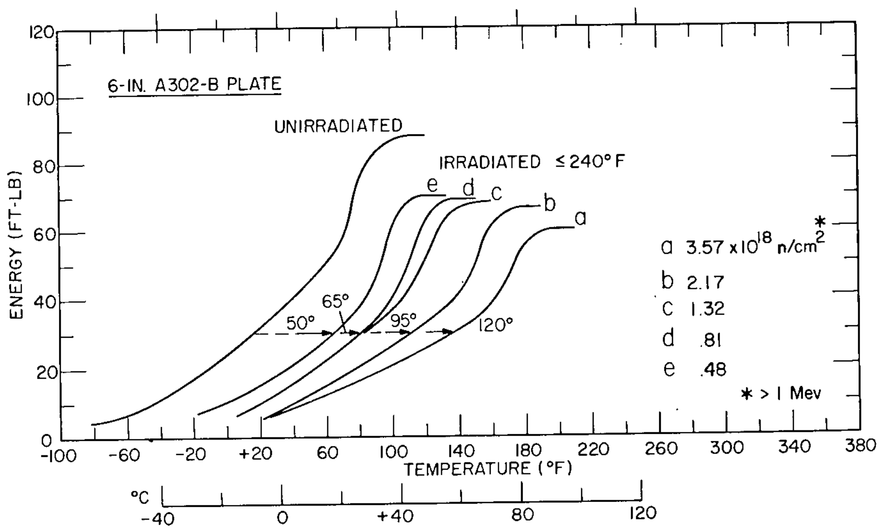

Fig. 5 - Notch ductility properties of A302-B steel at five locations inside test block

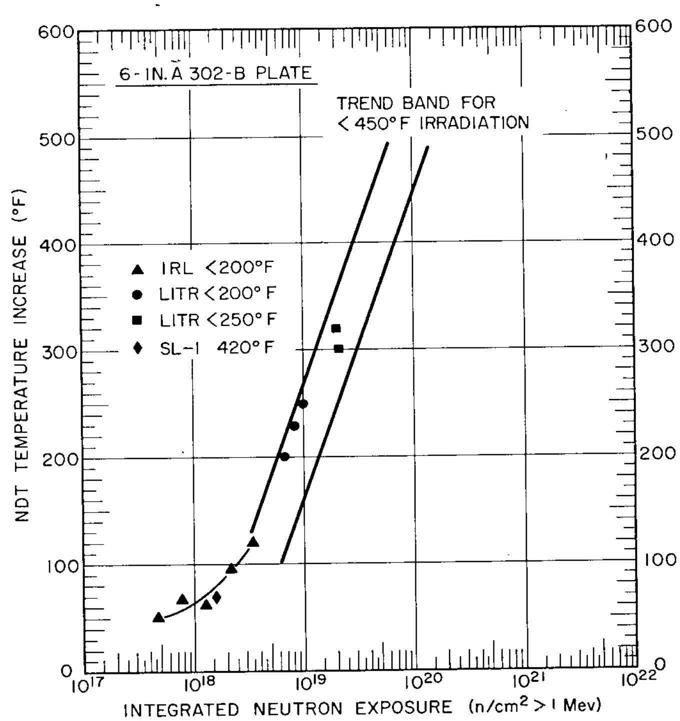

Fig. 6 - Increase in NDT temperature for A302-B pressure vessel steel for less than 450°F irradiations. Data point for SL-1 is for A212-B steel

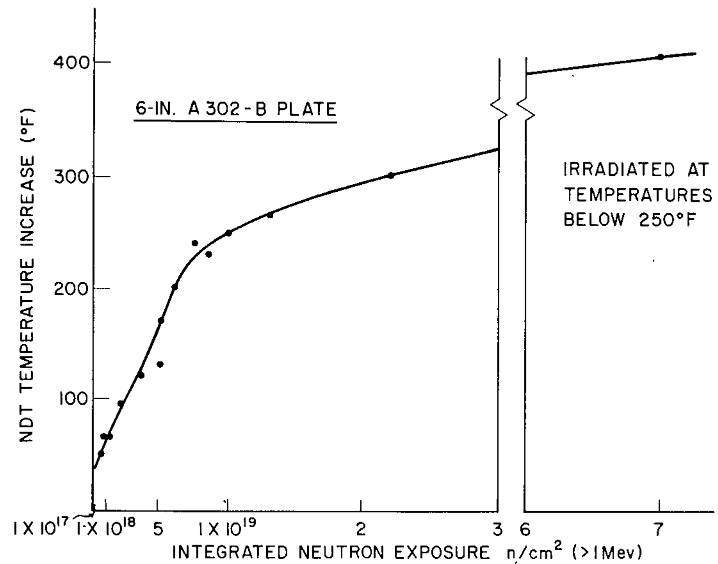

Fig. 7 - Increase in the NDT temperature of A302-B steel irradiated to various integrated neutron exposures at temperatures below 250°F

A comparison of the thermal and fast neutron flux measured in water alone and in the water-steel experimental environment is presented in Fig. 8. Clearly shown is the similarity of the magnitude and slope of the plots of the fast flux in both the water-only and water-steel media, versus the considerable disparity of the magnitude and slope of the plots of the thermal flux under the identical conditions. Both thermal and fast neutron flux monitors were grouped together at each of the several positions monitored in the two environments.

It is of interest to note that the thermal flux measured in the water-steel environment was practically identical to the fast flux (>1 Mev) at the specimen position in the test block nearest the fuel, and that it dropped off considerably through the block but rose again at the rear of the block to near equality with the fast flux. This effect is very similar to that described by Nielson and Ellis, Eds. (9), and by Taylor (10) in experimental and theoretical shielding studies in which the thermal flux is shown to drop off through an iron slab, but to rise rapidly above the fast flux to a peak at the rear of the slab. No measurements were made at the rear of the steel test block in the present study, but the trend of the data gathered in the study from within the block agrees very closely with the referenced works.

DISCUSSION

The NDT temperature increases developed from the specimens irradiated in the test block show that the loss of notch ductility is greatest nearest the fuel and is least for specimens irradiated farthest from the fuel. From the good continuity of data points shown in Fig. 7, which essentially places into agreement the accelerated in-core experiments and the case of the simulated pressure vessel, it can be said that a fair relationship between test reactor data and actual or simulated power reactors has been established. Therefore, the condition of material within the vessel wall of a light-water-moderated reactor can be estimated with a reasonable degree of certainty if the neutron exposure to the operating pressure vessel is known.

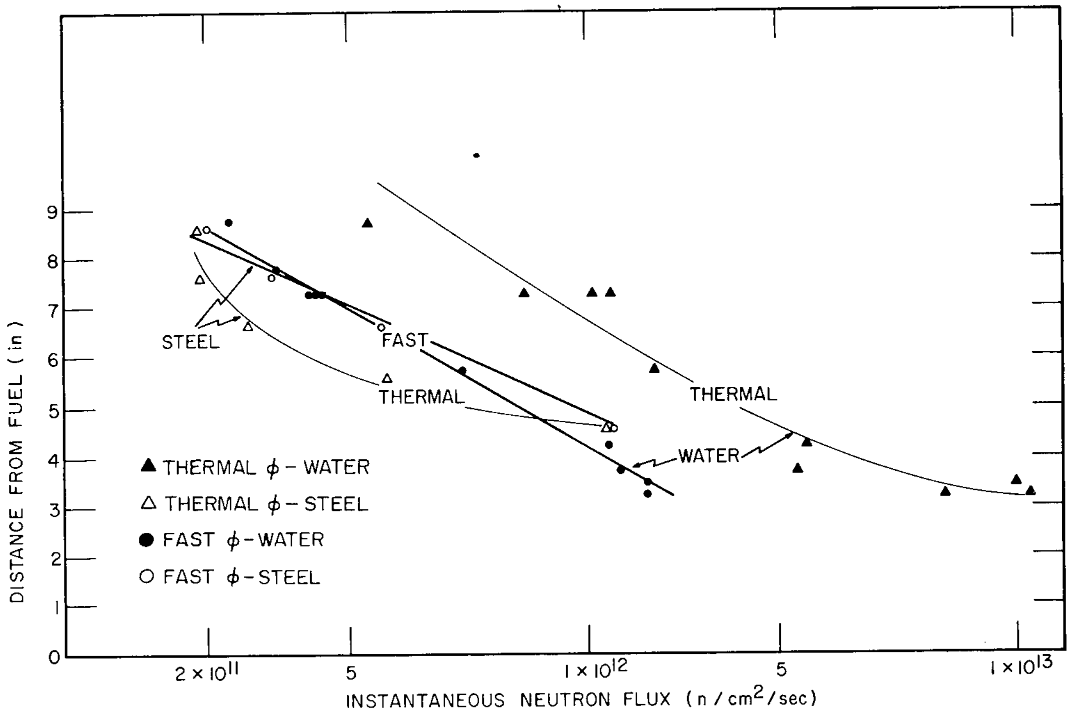

Fig. 8 - Comparison of fast and thermal neutron fluxes measured in water and water-steel test block media. Comparison of fast (>1 Mev) and thermal (2200 m/sec) neutron fluxes measured in water and water-steel test block media.

The transition temperature increase of the SL-1 vessel material is included in Fig. 6 for comparison of materials exposed to a relatively low dosage under operating power reactor conditions. Although the present test results were made on A302-B steel, and the SL-1 vessel was of A212-B steel, the data show very good agreement. The SL-1 point being slightly to the right of the curve developed in this study may reflect the fact that a thermal shield and two water layers were located between the nuclear fuel and the pressure vessel wall. This also strengthens the thesis that the data gathered in accelerated tests can be fairly correlated to the condition of a reactor pressure vessel wall.

The validity of the >1 Mev criterion for reporting neutron exposures has been enhanced by the thermal versus fast neutron flux comparison presented in Fig. 8. Since the fast flux was virtually the same in both the water alone and the water-steel media, it is clear that even at considerable distances from the active fuel region of a reactor, the high energy neutron flux does not widely fluctuate nor rapidly degenerate. This is obviously not the case for thermal neutrons. The water-only moderating media shows a relatively consistent spread between thermal and fast neutrons of about one half to one full order of magnitude – the thermal neutrons being more numerous. On the other hand, in the water-steel moderating media the thermal flux varies widely between near equality with, and considerable disparity to, the fast flux. The variable nature of the thermal flux within the steel block does not correspond to the rather consistent drop-off in degree of embrittlement exhibited by the measured transition temperature increases of the material. The fast flux, however, does appear to be consistent with the embrittlement trend of the material in this study and indeed to the data gathered in the accelerated experiments. In fact, with the exception of the thermal flux dependence reported by Roberts and Harries (11) on niobium-stabilized austenitic steels, there has been no comprehensive correlation between radiation damage in steels and thermal neutron exposure. In the study by

Roberts and Harries, the thermal flux dependence is thought to be a function of transmutation effects rather than atomic lattice disruption which is normally associated with radiation embrittlement of carbon steels in the operating temperature range of current power reactors.

It is recognized that the neutron energy spectrum measured in this experiment is not comprehensive. In a forthcoming experiment, however, it is hoped that a better definition of the neutron spectrum incident upon the test block will be made. This will involve the irradiation of the fission monitors neptunium-237 (threshold ~0.7 Mev) and uranium-238 (threshold ~1.5 Mev), along with thermal and fast threshold-type neutron monitors, in an irradiation similar to the one performed previously to better define the full neutron spectrum in the water-steel medium. The fission threshold monitors were not available for the original experiment but are now in an advanced stage of development.

CONCLUSIONS

Fast neutron induced radiation embrittlement has been shown to accumulate gradually and continuously throughout the thickness of a simulated heavy-walled reactor pressure vessel. Furthermore, it appears that the reduction in embrittlement through the 6-in. simulated pressure vessel wall is not large enough to be significant for minimizing the potential hazard of brittle fracture in a neutron embrittled reactor pressure vessel, but can only serve to add a margin of safety against the occurrence of such a fracture.

The ratio of neutron dose to transition temperature increase has been shown to compare very favorably to that exhibited by steels in accelerated irradiation experiments conducted in positions near nuclear fuel. The direct applicability of data from test reactor experiments to operating power reactor vessels has thus been indicated.

The validity of the >1 Mev criterion for reporting neutron dose as a major parameter in radiation embrittlement studies in light-water-moderated reactors is further enhanced by the findings of this study.

ACKNOWLEDGMENTS

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