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Through-Thickness Notch Ductility and Tension Properties as a Function of Neutron Exposure to a Simulated Pressure Vessel Wall of A302-B Steel

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ABSTRACT

Notch ductility and tension-property measurements have been made using specimens irradiated within a large steel test assembly simulating the pressure-vessel wall of a light-water-moderated power reactor. The A302-B steel specimens, spaced at intervals through the 6-in. thickness of the assembly, showed the greatest embrittlement and tensile property changes from irradiation locations nearest the fuel core, and correspondingly smaller changes farther from the core. Measured neutron fluxes of energies greater than 1 MeV, based upon an assumed fission spectrum, compared well with calculated spectrum neutron fluxes of energies greater than 1 MeV for all test assembly locations, thus providing the basis for future estimates of property changes through the thickness of heavy-walled reactor pressure vessels.

PROBLEM STATUS

This completes one phase of the problem; other phases of this research effort are continuing.

AUTHORIZATION

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THROUGH-THICKNESS NOTCH DUCTILITY AND TENSION PROPERTIES AS A FUNCTION OF NEUTRON EXPOSURE TO A SIMULATED PRESSURE VESSEL WALL OF A302-B STEEL

INTRODUCTION

Pressure vessels for the present generation of light-water-moderated central-station nuclear power plants are commonly constructed from low-alloy or carbon steel, since these materials are economical and are capable of being formed in the required large sizes and heavy section thicknesses. The exposure of these steels to high-energy neutrons, however, causes a rise in the ductile-to-brittle transition or nil-ductility transition (NDT) temperature (1), which is of particular concern in reactor design and operations. Detrimental radiation effects are also evidenced by the loss of tensile ductility and uniform elongation and by the more rapid elevation of the yield strength relative to the tensile strength (2). It is important, then, to determine the mechanical property changes, as a function of physical location and neutron flux, across the thickness of a heavy-sectioned reactor pressure vessel, since all of the steel volume determines the aggregate vessel strength and reliability. Furthermore, evaluation of such data will reveal the ability of test reactor experiments to accurately represent and define the condition of an operating reactor-vessel wall, and thus provide designers and operators with more knowledge concerning through-thickness properties changes for better prediction of pressure-vessel wall performance.

The concepts developed at NRL by Pellini and Puzak, which have been well validated for steels in the unirradiated condition, present an integrated approach to a basic understanding of fracture in steel structures (3). The use of the concepts and fracture analysis diagram procedures, for heavy-section reactor pressure vessels subjected to radiation effects, ideally requires testing the irradiated steel removed from the reactor component to establish its mechanical properties at the given point in service. Since this is obviously impractical, a second less direct approach is to test pressure-vessel steel from decommissioned reactors. The study made on the SL-1 pressure vessel gave the first opportunity to compare experimental irradiation data to exact data on the service-embrittled condition of an operating reactor vessel (4). While the vessel wall thickness of just 3/4 in. allowed the removal of test specimens representative of only the center portion of the wall, the satisfactory correlation of data between accelerated irradiation tests and the operating case was quite significant. The extremely limited availability of other such material is demonstrated by the fact that steel from the 1-in. thick Organic Moderated Reactor Experiment and the 2-3/8-in.-thick PM-2A pressure vessels present only the second and third opportunities, to date, to further test this correlation.

In the absence of a significant body of data from dismantled reactor pressure vessels, experiments have been performed to provide for the irradiation of Charpy V-notch ductility and tension test specimens appropriately positioned through the thickness of a 6-in.-thick steel test assembly simulating a pressure-vessel wall. Measurements with these specimens were designed to reveal embrittlement variations and tensile property changes to be expected through the thickness of such a vessel wall. This report presents the results of two experimental irradiations to maximum fluences of 2.6 and 5.0×10^{18} n/cm^2 (> 1 MeV) respectively, and draws conclusions concerning the through-thickness properties of heavy-walled pressure-vessel plates.

EXPERIMENTAL PROCEDURE

The experimental test assembly, designed to simulate a 6-in.-thick pressure-vessel wall of ASTM A302-B steel, has been described (5). To review briefly, an 8-in.-square test block, large enough to minimize the effect of neutrons impinging from the sides, top, and bottom, was made with accommodations through its thickness for five equally-spaced test-specimen capsules (Fig. 1). Thus, the specimens within the capsules, indicated by the shaded areas in Fig. 1, simulated vessel material at those locations. Since solid-steel plugs were attached to the tops of the capsules, which in turn were located within the tight-fitting pockets in the test block, the entire experimental test assembly presented, for all practical purposes, a volume of solid steel to the neutron flux. The narrow layer of water present on either side of each specimen capsule was considered to be too small to have any meaningful consequence as a moderating medium.

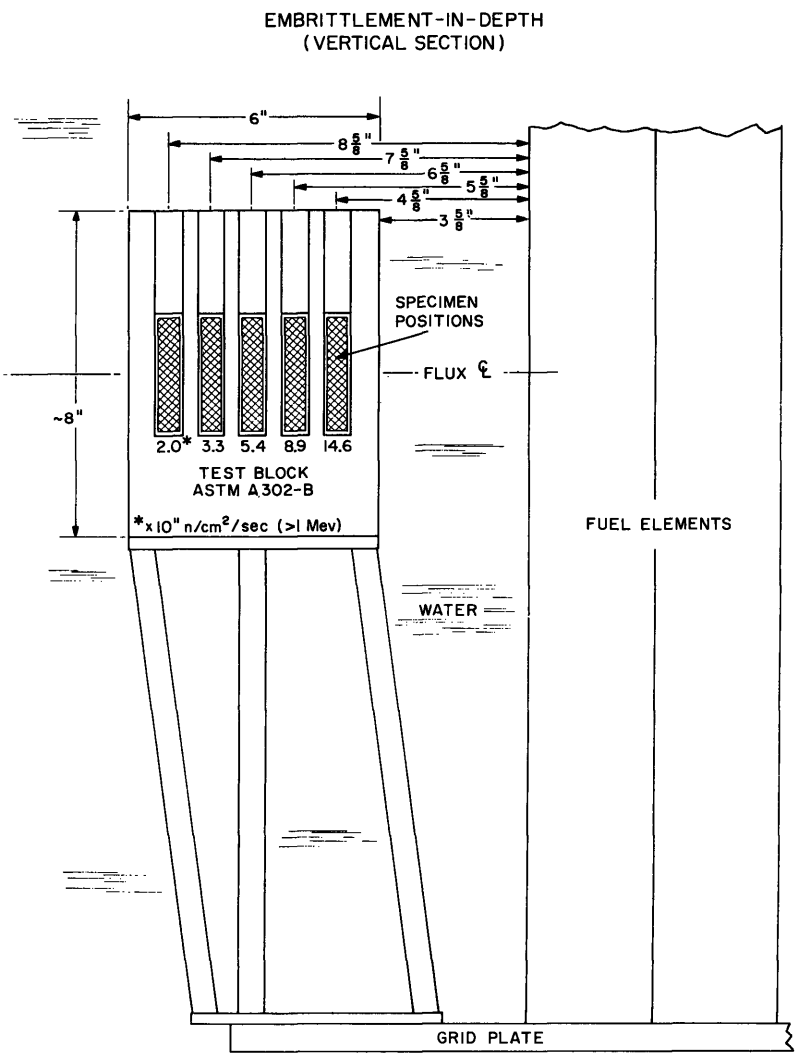

Fig. 1 - Schematic representation of the five specimen capsule locations within a steel test assembly positioned at the face of the Industrial Reactor Laboratories' pool reactor

The components within a typical specimen capsule for the higher fluence experiment (5.0×10^{18} n/cm^2 > 1 MeV, maximum) are shown schematically in Fig. 2. The specimen capsules of the lower fluence experiment (2.6×10^{18} n/cm^2 > 1 MeV, maximum) were quite similar in design but contained only Charpy specimens, threshold neutron flux, and temperature monitors. Sealed tubes containing U-238 and Np-237 fission flux monitors were positioned in the upper-right-hand corner of each assembly (Fig. 2). These were withdrawn after 20 hr of irradiation for counting and analysis of La-140 activity. Another sealed tube, containing only U-238 fission flux monitor material, was placed in the upper-left-hand corner of each capsule for the entire irradiation period. The fission-monitor data thus generated have been reported elsewhere (6) and will not be reviewed in detail here.

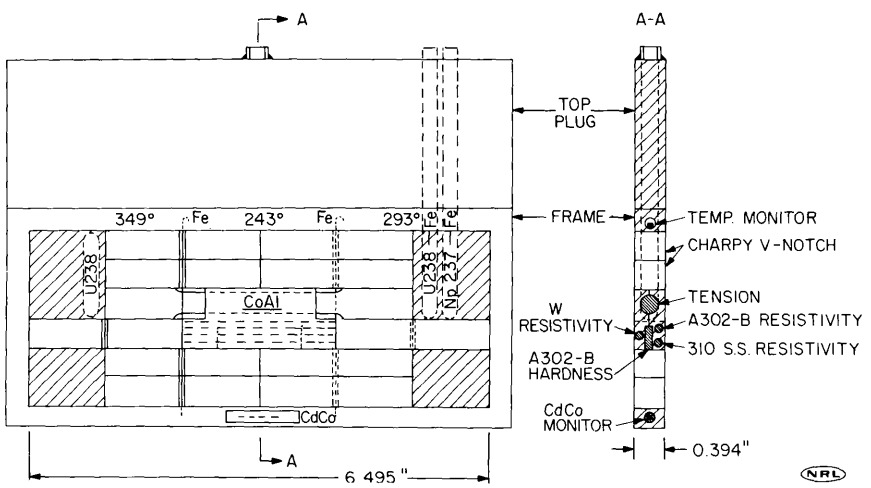

Fig. 2 - Schematic diagram showing the location of metallurgical test specimens and neutron flux monitors in a capsule designed for irradiation, in a test assembly simulating a reactor pressure-vessel wall

Resistivity specimens of tungsten, AISI 310 stainless steel, and A302-B steel, as well as a hardness test specimen of A302-B steel, were located just below the centerline of each capsule. These specimens were supplied by the General Electric Company, Nuclear Materials and Propulsion Operation (GE-NMPO) for irradiation in the NRL experiment under the cognizance of the USAEC Fuels and Materials Branch. The GE-NMPO material data will be reported separately by that organization.

As indicated in Fig. 2, one tension specimen was located just above the center of each capsule. Ten Charpy V-notch (C_v) specimens were located in the remainder of the capsules, as shown. Note that all but two of the C_v specimens were located at least 1 in. away from the edge of the capsule, thus placing the specimen notches at least 2 in. away from the capsule edge. With this arrangement it is believed that the shielding afforded by the steel test assembly block, plus the centralization of the specimens within the capsules, effectively minimized the number of neutrons entering from the sides of the block.

The capsules were covered with a 0.010-in.-thick stainless steel sheath welded to the capsule framework and compressed at 1700 psi to assure good contact between the sheath and the specimens, in the interest of high heat transfer. Postirradiation inspection of the low-melting-point-alloy temperature monitors located in the edges of the

capsule framework (243°F, 293°F, and 349°F) revealed that exposure temperatures did not exceed 243°F. Iron wires (Fe) and Al-0.1% Co wire, both unshielded (CoAl) and shielded with 0.020-in.-thick cadmium (CdCo), were placed in appropriate locations for neutron flux monitoring, and the results of those determinations will be discussed in a later section of this report.

The experimental test assembly was irradiated in the Industrial Reactor Laboratory, Inc. (IRL) pool reactor in a cooperative program with the U.S. Rubber Company, one of the co-owners of the IRL facility. The assembly, centered on the horizontal peak flux plane of the reactor fuel core, was exposed for 678.5 hr in the lower fluence experiment and for 1363 hr in the higher fluence experiment at average power levels of 4.3 and 4.8 MW(t) respectively.

EXPERIMENTAL RESULTS

Preirradiation Properties

The composition and heat treatment of the 6-in.-thick ASTM A302-B steel plate used in this investigation are given in Table 1. Because of a time interval between experiments, the Charpy V-notch specimens for the two experimental irradiations were taken from separate sections of the main plate. The NDT and C_v 30 ft-lb transition temperatures* of the first plate section (higher fluence experiment) were 10°F and 15°F, while the C_v 30 ft-lb transition temperature of the second plate section (the lower fluence experiment) was 30°F. The yield (0.2% offset) and tensile strengths of the particular section used for tensile property determinations were 70.7 ksi and 93.2 ksi.

Table 1
Composition (Weight-Percent) and Heat Treatment* of Reference Heat of ASTM A302-B Steel

C	Mn	Si	P	S	Ni	Cr	Mo
0.20	1.31	0.25	0.012	0.023	0.20	0.17	0.47

*Austenitized at 1650°F for 2 hr, water quenched; tempered at 1200°F for 6 hr, furnace-cooled to below 600°F.

Postirradiation Properties

Notch Ductility Properties – Charpy-V specimens were tested in the NRL High Level Radiation Laboratory using equipment and techniques which have been described (7,8). The radiation-induced transition temperature increases for the five sets of Charpy specimens from the low and high fluence experiments are shown in Figs. 3 and 4 respectively.

*A differentiation is to be made between a nil-ductility transition (NDT) temperature increase and a transition temperature increase. An NDT temperature increase implies that drop-weight specimens of a material have been tested to establish the nil-ductility transition temperature directly. Charpy V-notch results for the same materials may then be indexed to this temperature by selecting the corresponding energy value ("fix point" on the Charpy curve) which best represents the NDT temperature. This level can then be used for the determination of the neutron-induced NDT increase. Transition temperature increase refers to the method of simply selecting an arbitrary energy level (preferably based upon an average correlation point representative of the NDT for that class of steels) and translating along that energy level to the curve for the irradiated condition, thus determining the increase in temperature of the reference value.

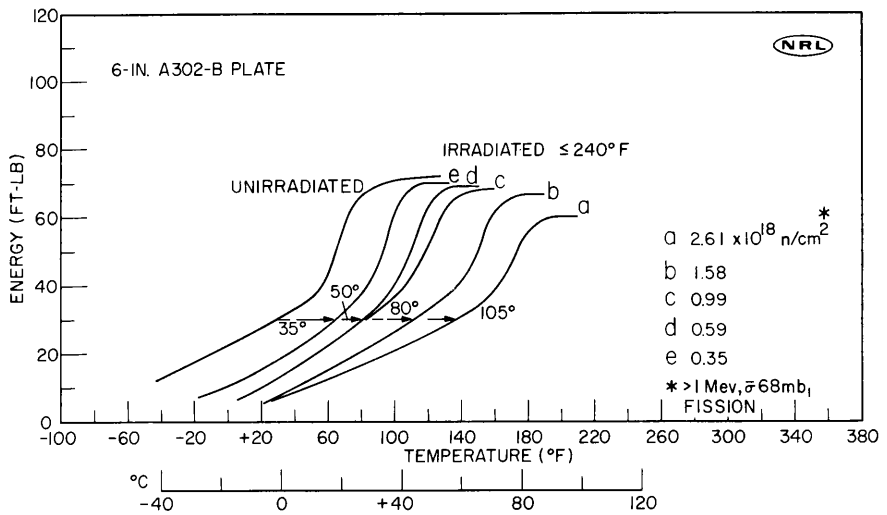

Fig. 3 - Charpy-V 30 ft-lb transition temperature increases for five sets of specimens irradiated to a maximum fluence of 2.6×10^{18} n/cm^2 > 1 MeV, in a pressure-vessel-simulating test assembly located at the face of a pool reactor. The curves (a) to (e) represent specimens located progressively farther from the core face of the reactor.

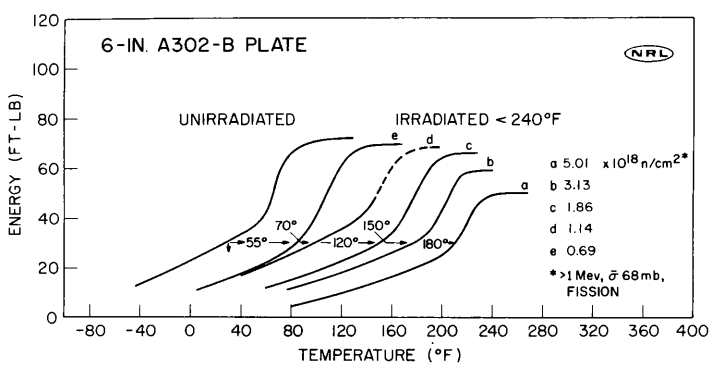

Fig. 4 - Charpy-V 30 ft-lb transition temperature increases for five sets of specimens irradiated to a maximum fluence of 5.0×10^{18} n/cm^2 > 1 MeV, in a pressure-vessel-simulating test assembly located at the face of a pool reactor. The curves (a) to (e) represent specimens located progressively farther from the core face of the reactor.

Pertinent data from these experiments are given in Table 2. In both cases curve (a) identifies steel in the test assembly location nearest to the reactor core and thus depicts the highest transition temperature increase of each experiment. Correspondingly, curves (b) through (e) were developed from specimens located progressively farther from the core face. The relatively small differences between curves, coupled with the small number of specimens in each capsule allotted for the definition of each, precluded the development of a full transition curve for each group of specimens in some cases. Nevertheless, the 30 ft-lb transition temperatures for all of the curves are considered to be accurate to within ± 10°F.

Table 2
Notch Ductility Properties of A302-B Steel Before and After Irradiation Within a 6-In.-Thick Steel Test Assembly Located at the Face of a 5-MW Pool Reactor

Location*	30 ft-lb Transition Temp. Increase (ΔNDT) (°F)	Full Shear Energy Absorption		Fluence (Fission Spectrum) (n/cm^2 > 1 MeV)
		Initial	Irrad.	
4-5/8	105	72	59	2.6×10^{18}
5-5/8	80	72	66	1.6
6-5/8	50	72	67	0.99
7-5/8	50	72	68	0.59
8-5/8	35	72	68	0.35
4-5/8	180	72	51	5.0
5-5/8	150	72	59	3.13
6-5/8	120	72	64	1.86
7-5/8	70	72	67	1.14
8-5/8	55	72	69	0.69

*Distance in inches of specimens from the face of the reactor fuel core.

The C_v 30 ft-lb transition temperature increases from Figs. 3 and 4 have been plotted versus neutron fluence values based upon an assumed fission spectrum in Fig. 5, where the data are referenced to the NRL trend band for steel irradiations at temperatures less than 450°F. While a few of the data points are somewhat displaced from the left boundary of the trend band, the scatter of the points about the boundary line is quite small. It is thus felt that the C_v data in Fig. 5 are in acceptable conformance with the large body of data which initially formed the trend band depicting steel response to irradiation at less than 450°F. This conformance provides one basis for extrapolations of embrittlement trends within heavy-walled pressure vessels constructed from a low-alloy steel such as ASTM A302-B.

Tensile Properties — Tensile specimens contained in the second or higher fluence experiment were tested at ambient temperatures which have been described (9). The nominal stress versus natural strain curves for these specimens are shown in Fig. 6 together with a reference data band for the unirradiated condition. Tensile properties data are listed separately in Table 3. Although this study was performed with only one specimen per radiation exposure level because of space limitations in the specimen capsule, it nevertheless became possible to establish a trend for tensile properties which could be expected at successive locations through the thickness of a heavy-walled pressure vessel. Specifically, the curves in Fig. 6 and the data in Table 3 demonstrate a consistent trend in properties change as a function of neutron fluence in a manner paralleling that shown by Charpy V-notch specimens. That is, the specimen placed nearest to the fuel core face, representing that portion of the steel at the inner surface of a pressure-vessel wall, had the highest neutron fluence and the greatest change in tensile properties. On the other hand, the specimen placed farthest from the fuel core face, representing the lowest neutron fluence, indicated a commensurately smaller change in tensile properties.

An overall comparison of the yield (0.2% offset) and tensile strengths for the several conditions shown in Table 3 reveals that the two measurements are rapidly approaching a common value after low-temperature (< 240°F) irradiation, despite the low neutron

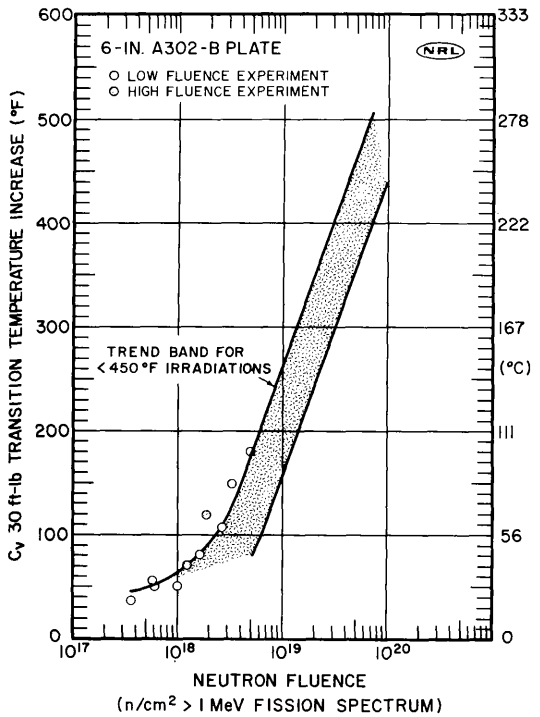

Fig. 5 - Charpy-V 30 ft-lb transition temperature increases versus neutron fluence values based upon the fission spectrum as given by two sets of experimental data depicting locations within a pressure-vessel-simulating test assembly. Open data points represent the lower fluence (2.6×10^{18} n/cm^2 > 1 MeV) experiment, and the shaded data points represent the higher fluence (5.0×10^{18} n/cm^2 > 1 MeV) experiment.

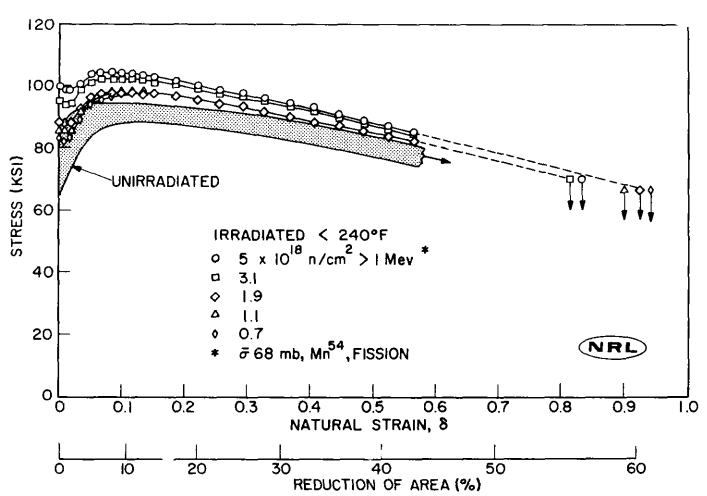

Fig. 6 - Nominal stress versus natural strain and reduction of area curves for A302-B steel in the unirradiated condition and after irradiation in a test assembly simulating a thick pressure-vessel wall of a light-water-moderated reactor

fluence level of 5×10^{18} n/cm^2 > 1 MeV. It is significant, however, that uniform elongation has not been drastically reduced by the relatively low neutron fluence. Thus, the data trend suggests that a small but, nevertheless, effective potential for stress redistribution in flaw-free areas may exist for this lower range of neutron fluences in light-water-moderated reactor vessels of A302-B steel. This suggestion is based upon the condition that the low-temperature irradiation data are applicable to service irradiations at the higher operating temperatures of pressurized and boiling-water reactors.

Table 3
Tensile Properties of 6-In. A302-B Steel Before and After Irradiation within a Steel Block at the Face of a 5-MV Pool Reactor

Reactor Location (in.)*	Fluence (n/cm^2 > 1 MeV)†	Upper Yield Point (ksi)	0.2% Yield Point (ksi)	Tensile Strength (ksi)	Reduction of Area (%)	Elongation in 1-in. (%)	Hardness (Rockwell-B)
Not Irradiated	—	—	70.7	93.2	64.9	26.9	92
4-5/8	5.0×10^{18}	100.0	100.0	104.5	56.6	—‡	96
5-5/8	3.1×10^{18}	95.0	94.2	102.2	55.6	—	95
6-5/8	1.9×10^{18}	83.5	88.4	97.6	60.2	—	94
7-5/8	1.1×10^{18}	87.0	85.5	98.0	59.2	—	93.4
8-5/8	0.7×10^{18}	83.2	83.2	97.5	60.7	—	—‡

*Distance of specimens from face of reactor fuel core.
†$\bar{\sigma}$ 68 mb, $Fe^{54}(n,p)Mn^{54}$, fission spectrum.
‡Not determined for this experiment.

Neutron Dosimetry — Measured values of fast (> 1 MeV) and thermal (2200 m/sec) neutron fluxes, as given for each capsule of both experiments, are based upon an assumed fission spectrum at all irradiation locations (Table 4). The cross section averaged over a fission spectrum for the $Fe^{54}(n,p)Mn^{54}$ reaction was 68 mb, while that for the $Co^{59}(n,\gamma)Co^{60}$ reaction was 36.3 barns. The actual neutron spectra for each irradiation location within the test assembly (10) were calculated by Dahl and Yoshikawa (11). The results of the experimental irradiations are considered in the light of these theoretical neutron spectra in the following section.

NEUTRON SPECTRAL CONSIDERATIONS

The theoretical neutron spectra at the five locations in the irradiation test assembly have been calculated (Fig. 7) by R. E. Dahl and H. H. Yoshikawa of the Battelle-Northwest Laboratories in Richland, Washington, using transport theory, S_n reactor physics code (12). The calculations were performed for two locations within each of the five test assembly capsule positions. The locations were at the center of the V-notches of each assembly (Fig. 2). They are further depicted by the inserts on each of the spectral plots in Fig. 7. For simplicity, the spectra of Fig. 7 represent only one of the two locations for each capsule (indicated by the X); the designation of the location being referred to is the extension of "Row 5" of the reactor core.

Similar transport theory calculations for a number of irradiation locations in a wide variety of reactor types have also been performed by Dahl and Yoshikawa. Consideration of the effect of all the neutrons in each reactor spectrum has led these investigators to propose that mechanical properties data can be best interrelated if neutrons of energies greater than 0.5 MeV in the calculated spectra are reported (13).

An extensive evaluation of this thesis has been reported which used all the irradiation data developed by NRL from irradiations of A302-B steel at temperatures less than

Table 4
Summary of Neutron Dosimetry Results Based upon $Fe^{54}(n,p)Mn^{54}$ and $Co^{59}(n,\gamma)Co^{60}$ Reactions for Five Locations in a Steel Test Assembly

Location*	Thermal Flux (n/cm^2-sec 2200 m/sec)	Fast Flux (n/cm^2-sec > 1 MeV)	Fluence (Fission Spectrum) (n/cm^2 > 1 MeV)	Fluence (Calculated Spectrum) (n/cm^2 > 0.5 MeV)
4-5/8	—	10.65×10^{11}	2.6×10^{18}	3.3×10^{18}
5-5/8	—	6.40×10^{11}	1.6×10^{18}	2.7×10^{18}
6-5/8	—	3.95×10^{11}	0.99×10^{18}	2.2×10^{18}
7-5/8	—	2.35×10^{11}	0.59×10^{18}	1.6×10^{18}
8-5/8	—	1.42×10^{11}	0.35×10^{18}	1.3×10^{18}
4-5/8	8.63×10^{11}	10.21×10^{11}	5.0×10^{18}	6.3×10^{18}
5-5/8	3.23×10^{11}	6.35×10^{11}	3.13×10^{18}	5.2×10^{18}
6-5/8	1.76×10^{11}	3.79×10^{11}	1.86×10^{18}	4.1×10^{18}
7-5/8	1.22×10^{11}	2.33×10^{11}	1.14×10^{18}	2.9×10^{18}
8-5/8	1.08×10^{11}	1.46×10^{11}	0.69×10^{18}	2.3×10^{18}

*Distance in inches of specimens from the face of the reactor core.

450°F (10). The results of this evaluation are presented in Fig. 8, where the data from the through-thickness embrittlement experiments are represented by the points at the low fluence portion of both curves. This figure shows the data plotted versus (a) n/cm^2 > 1 MeV based upon a fission spectrum (open and lightly shaded points) and (b) n/cm^2 > 0.5 MeV based upon calculated spectra (closed and darkly shaded points).

Reporting neutron fluence based upon calculated spectra greater than 0.5 MeV neutrons can be shown, on a theoretical basis, to best correlate data from a wide variety of reactor environments and to best accommodate the critical evaluations of embrittlement in specific reactor vessels. However, the scatter of data points from different heats of steel irradiated in widely different environmental irradiations overshadows most of the normalization which can be affected by this fluence-reporting technique (10). The calculations of theoretical neutron populations for different energy groups have, nonetheless, proven to be of considerable value in the application of the measured fluxes and embrittlement data to the general case of through-thickness embrittlement of heavy-walled pressure vessels.

Theoretical fluxes of neutrons having energies greater than 1 MeV have been summed by energy group for the five specimen capsule locations in Fig. 9 for comparison with the measured neutron fluxes of energies greater than 1 MeV determined for the same physical locations. In both parts of Fig. 9 the curves connecting the data points have been extrapolated to the inner and outer edges of the test assembly and up to the face of the reactor itself. This procedure is considered reasonable and valid, since the extrapolated value at the face of the reactor compares favorably with values measured at that location by the reactor operating staff (14). On this basis the decrease in flux intensity (Fig. 9) from the front to the rear of the test assembly is 94.8%, or approximately 15.8% per in. of thickness according to the measured flux curve, and is 95.2%, or approximately 15.8% per in. of thickness from the calculated flux curve.

The very close agreement between the measured and calculated flux decrease through the thickness of the test assembly provides a sound and meaningful foundation for future predictions of embrittlement in heavy-sectioned pressure vessels.

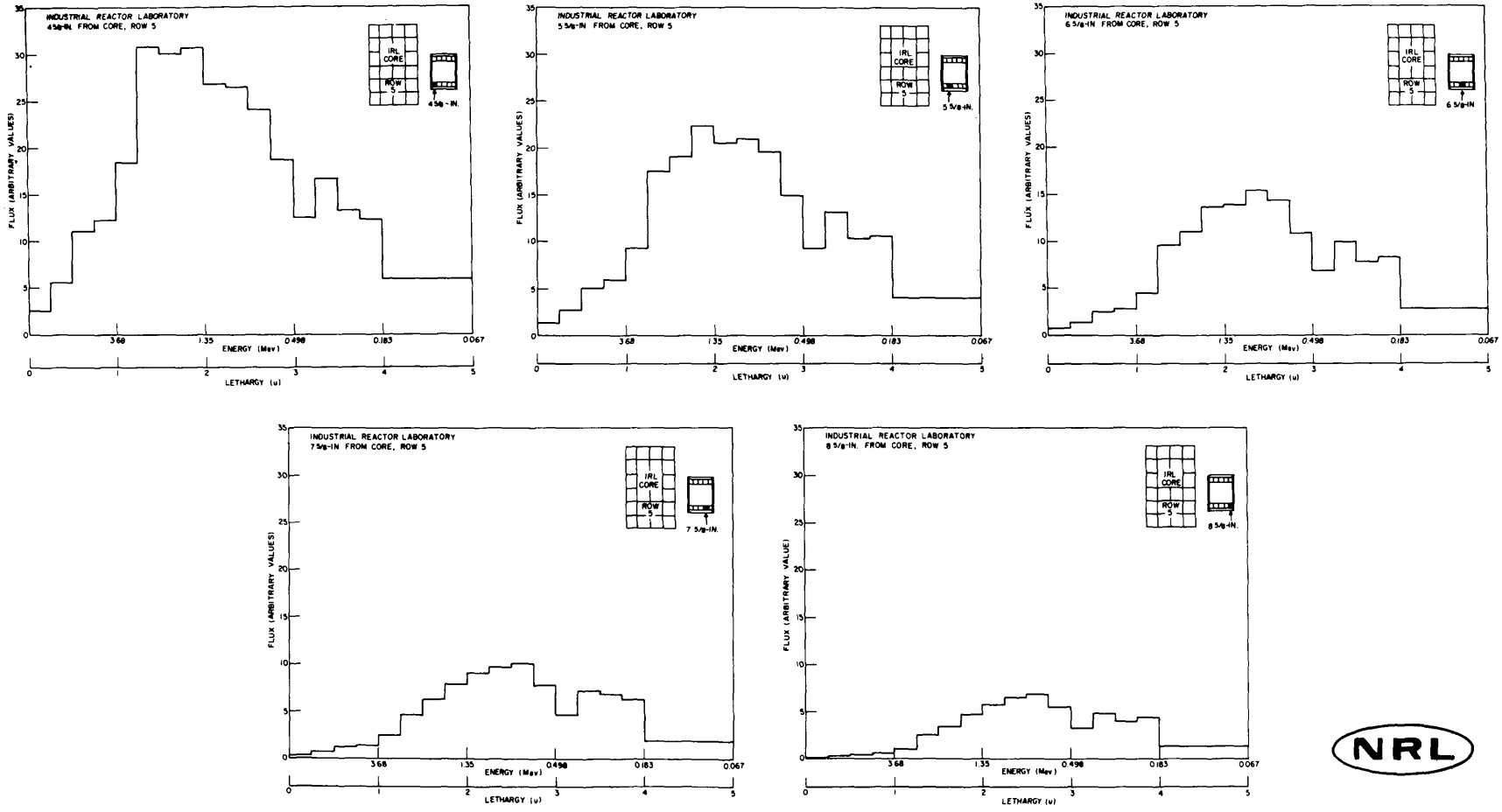

Fig. 7 - Graphical representation of the neutron spectra in the five locations of the test assembly designed to simulate a pressure-vessel wall for irradiation at the face of the light-water-moderated Industrial Reactor Laboratories' pool reactor. Two series of spectra were calculated for the test assembly at positions indicated by the two specimen rows shown by the individual inserts. Depicted in this figure are the spectra for the five locations on the line extension of "Row 5" of the reactor fuel core.

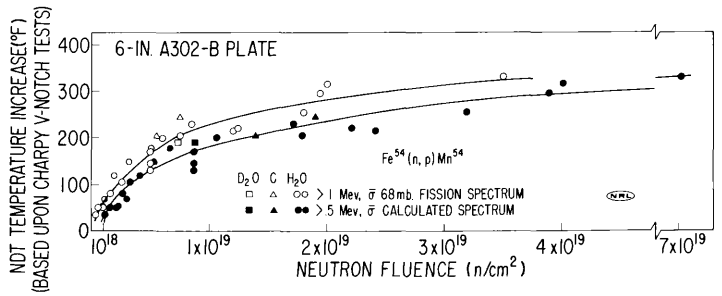

Fig. 8 -Charpy-V 30 ft-lb transition temperature increase data from numerous experiments including that from two experiments in a pressure-vessel-simulating test assembly. these points form the lower portion of two curves which represent, respectively, neutron fluences > 1 MeV based upon a fission spectrum (open and lightly-shaded circles) and neutron fluences >0.5 MeV based upon calculated spectra (closed and heavily-shaded circles)

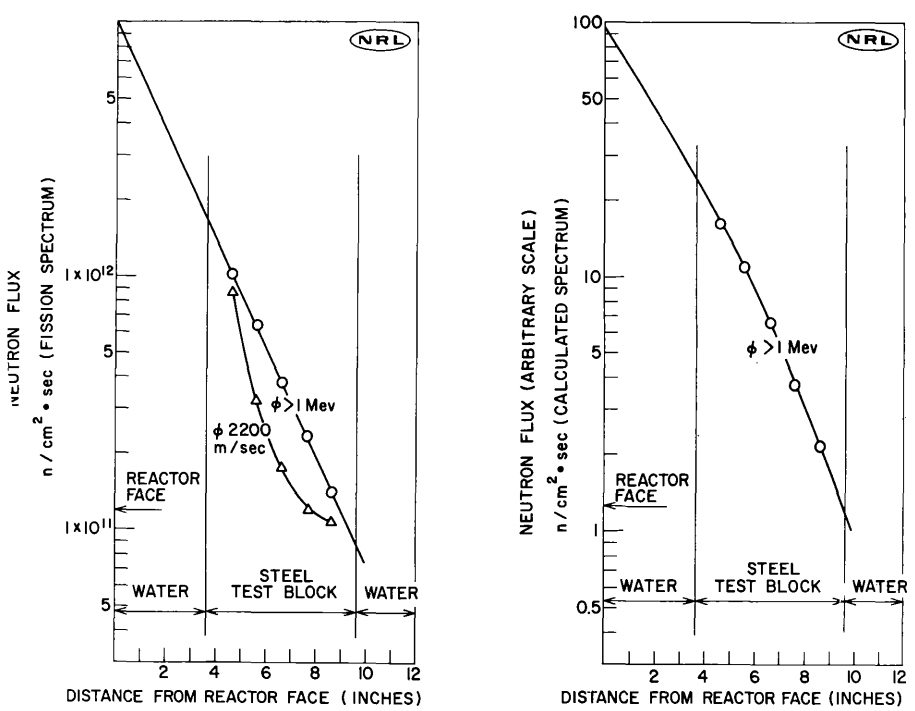

Fig. 9 - The decrease in neutron flux as determined by measurements based upon a fission spectrum (left side) and by theoretical calculations of the neutron spectra (right side) within a 6-in.-thick steel test assembly designed to simulate the pressure-vessel wall of a light-water-moderated power reactor. The flux as determined from front-to-rear measurements in the steel test assembly was almost 95 percent, while the flux decrease as determined from spectral calculations was just over 95 percent.

SUMMARY

Two experimental irradiations have been performed wherein Charpy V-notch and tension specimens of a single plate of ASTM A302-B reference steel were placed inside a specially prepared, 6-in.-thick test assembly, located at the face of a light-water-moderated pool reactor, to simulate a heavy-section reactor pressure-vessel wall. Subsequent testing of the C_v specimens, at the end of the irradiation period, revealed trends in embrittlement which closely match trends noted previously by NRL for the notch ductility behavior of this A302-B heat under low-temperature irradiation.

The tension-specimen behavior was shown to follow a trend of increasing yield and tensile strengths for the higher levels of neutron fluence; but, quite interestingly, the decreases in ductility as measured by uniform elongation and percent reduction of area were less than that expected. The relatively substantial amount of uniform elongation retained by this steel, when coupled with the well-defined lower yield point behavior, suggests a potential for redistribution of stresses about localized (but flaw-free) regions of strain. However, the experimental irradiation exposures were at the low end of the fluence range expected for pressure-vessel service lifetimes.

Validation of the trends in mechanical property changes permits the employment of neutron flux data for drawing conclusions concerning the decrease in flux through the thickness of the test assembly. This neutron flux decrease, deduced from experimental measurements, was shown to compare favorably with the rate of decrease determined by a method of calculation involving summation by discrete energy groups of the fluxes in calculated spectra of the same experimental test-assembly locations. Observation of this correlation provides a firm basis for future engineering estimates of mechanical property changes and neutron fluence trends in heavy-walled reactor pressure vessels.

CONCLUSIONS

1. Notch ductility and tensile properties of steel from locations throughout the thickness of heavy-sectioned reactor pressure vessels have been shown, on a simulated experimental basis, to follow previously observed trends in mechanical property change as a function of neutron fluence.

2. A well-defined lower yield point, developed in ambient-temperature testing coupled with a relatively substantial amount of uniform elongation, suggests a potential of the steel for stress redistribution about localized, flaw-free regions of stress.

3. The decrease of neutron flux of energies greater than 1 MeV through a 6-in.-thick test assembly has been shown to be about 95 percent or almost two orders of magnitude, regardless of whether the reported flux is from measured values based upon an assumed fission spectrum or from the summation of fluxes by energy group in the calculated spectra for the same irradiation locations.

4. The applicability of information, from accelerated irradiations to operating reactors, has been enhanced by the results of these experiments, since the data compare very favorably with prior research and form a bridge, in terms of neutron flux level, between pressure vessels of operating reactors and accelerated experimental irradiation facilities of test reactors.

5. The experimental results of this study contribute a basis for the estimation of mechanical property changes as a function of neutron flux level (and fluence) for future engineering applications, with a much higher degree of confidence than was previously possible.

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Notch ductility and tension-property measurements have been made using specimens irradiated within a large steel test assembly simulating the pressure-vessel wall of a light-water-moderated power reactor. The A302-B steel specimens, spaced at intervals through the 6-in. thickness of the assembly, showed the greatest embrittlement and tensile property changes from irradiation locations nearest the fuel core, and correspondingly smaller changes farther from the core. Measured neutron fluxes of energies greater than 1 MeV, based upon an assumed fission spectrum, compared well with calculated spectrum neutron fluxes of energies greater than 1 MeV for all test assembly locations, thus providing the basis for future estimates of property changes through the thickness of heavy-walled reactor pressure vessels.

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