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Yankee Reactor Pressure-Vessel Surveillance: Notch Ductility Performance of Vessel Steel and Maximum Service Fluence Determined from Exposure During Cores II, III, and IV

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CONTENTS

Abstract	iv
Problem Status	iv
Authorization	iv
INTRODUCTION	1
NOTCH DUCTILITY TEST RESULTS	2
In-Reactor Surveillance	2
EXPERIMENTAL SIMULATION OF REACTOR OPERATIONS	10
REACTOR VESSEL NEUTRON SPECTRUM AND FLUX	14
Neutron Spectrum	14
Reactor Vessel Neutron Flux	17
Discussion	20
EMBRITTLEMENT AS A FUNCTION OF NEUTRON FLUENCE	21
Comparison of Surveillance Versus Test Reactor Data	21
Evaluation of the Significance of Pressure Vessel Embrittlement	23
SUMMARY AND CONCLUSIONS	26
ACKNOWLEDGMENTS	27
REFERENCES	27

ABSTRACT

Charpy V-notch specimens, representative of one of the several heats of A302-B steel forming the Yankee reactor pressure vessel and irradiated as part of the Yankee surveillance program, have been tested by NRL. Specimens of this particular heat, irradiated in near-core (accelerated) as well as in vessel-wall locations, showed more embrittlement than did specimens of a reference steel heat cf the same nominal A302-B composition irradiated simultaneously in the same surveillance capsules.

Those specimens from both the Yankee vessel heat and the reference heat irradiated at the vessel-wall location depicted a higher damage rate than that for the accelerated location. The cause of this difference in embrittlement response could not be attributed to an effect of cyclic, service irradiation temperatures, but could be traced to a qualitative relationship of thermal to fast (>1 Mev) neutron fluxes. This ratio was in excess of about 9:1 at the vessel-wall location versus a ratio less than about 9:1 for the accelerated location.

The computation of a maximum service fluence of 1.46×10^{19} n/cm^2 (>0.5 Mev) was made possible by establishment of the neutron spectrum at the reactor vessel wall using computer calculations. The maximum fluence derived by this technique compared favorably with another value given by an independently-developed calculated neutron spectrum. The NRL computed service fluence in concert with the embrittlement data projects a maximum transition temperature increase of 265°F, which represents a level of embrittlement considered acceptable for the Yankee reactor vessel after thirty fuel cycles of operation at 600 thermal megawatts [Mw(t)].

PROBLEM STATUS

This is a final report on one phase of the problem; work on other phases of the problem is continuing.

AUTHORIZATION

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YANKEE REACTOR PRESSURE VESSEL SURVEILLANCE: NOTCH DUCTILITY PERFORMANCE OF VESSEL STEEL AND MAXIMUM SERVICE FLUENCE DETERMINED FROM EXPOSURE DURING CORES II, III, AND IV

INTRODUCTION

Pressurized, light-water moderated nuclear reactors are now being planned and designed in the United States which will generate upward of 1000 electrical megawatts [Mw(e)] and which will have pressure vessels of weights up to 1000 tons, thicknesses up to 14 in., diameters larger than 20 ft, and heights greater than 80 ft. Such huge vessels in order to contain design pressures up to 2000 psi must be quite strong; yet, because of their large size, must be constructed of economical material. Thus, reactor pressure vessels will continue to be made of conventional low-alloy or carbon steels. These steels, as a class, exhibit a nil-ductility transition (NDT) temperature below which fractures initiated from sharp flaws have brittle characteristics but above which fractures are propagated in a ductile mode. During service when the steel forming the pressure vessel is bombarded by high energy neutrons emanating from the nuclear fuel core, the steel's NDT temperature is known to rise appreciably. Further, if the reactor were to operate over a very long period of time, the rising NDT temperature might closely approach the operating temperature of the vessel itself (1). At this point in service, the presence of the proper combination of stress, vessel temperature, and a flaw of critical size (2) may trigger a brittle fracture of the vessel. In such a case, the required critical stress could be supplied by some unexpected but credible impact or pressure surge.

A sequence of events which may lead to the potential failure of a reactor pressure vessel has been noted as outlined above. In recognition of this problem, the in-reactor material surveillance programs are used in most power reactors. The purpose of such programs is to reveal the progressive changes in the vessel steel due to high energy neutron bombardment and to indicate when modifications in operating procedures, if any, are necessary to maintain safe operation of the plant. Specifically, reactor pressure-vessel surveillance programs are designed to yield basic information concerning the increase in NDT temperature of the vessel steel as a function of the incident fluence* of neutrons. With sufficient information, extrapolation of these data to a point in time corresponding to the planned service life of the reactor can reveal the magnitude of the allotted or built-in safety margin for the plant against a potential brittle failure.

This report presents and examines surveillance data from the Yankee reactor surveillance program. Metallurgical test specimens from this program have been evaluated by NRL to determine the response of the steel to radiation under the temperature and neutron environmental conditions of that plant. From an analysis of the resultant data, it has been possible to estimate the conditions of the Yankee vessel at the end of its service life as now planned.

*The term fluence is used in this report in place of such terms as exposure or dose. Fluence is taken to represent the time integrated, instantaneous neutron flux, in n/cm^2.

NOTCH DUCTILITY TEST RESULTS

In-Reactor Surveillance

The Yankee reactor pressure-vessel material surveillance program has already been described (3). Briefly, two elongated specimen capsules, designated as "vessel-wall capsules," were located between the thermal shield and pressure-vessel wall, while eight additional and physically similar capsules, designated "accelerated capsules," were located inside the thermal shield and adjacent to the shroud surrounding the fuel core (Fig. 1). Each surveillance capsule contained Charpy V-notch and tensile specimens from an A302-B steel fabrication test plate made from the runout of one plate for the upper, vessel shell course. Charpy V-notch specimens from a 6-in. A302-B plate of a reference heat (4) were also included in each capsule. The test plate is referred to in this report as "Yankee surveillance steel" (not as the Yankee vessel steel) since it was heat treated separately from the main, shell-course plates (Table 1) (5).

Ten material surveillance capsules were inserted in the Yankee reactor during loading of the second core. At the end of the second core life, the Yankee Atomic Electric Company withdrew four accelerated capsules from the reactor for Charpy V-notch testing at NRL. NRL test results have been reported previously (6) and are included in this

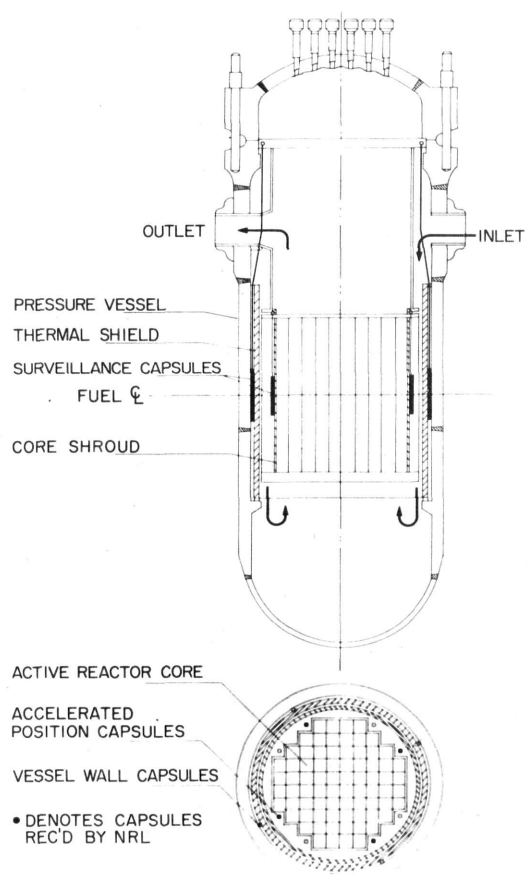


Fig. 1 - The Yankee Atomic Power Reactor showing surveillance positions and locations of capsules received by NRL (3)

Table 1
Chemical Composition and Heat Treatment* for 8-1/8-in.-Thick A302-B Yankee Surveillance Steel—Lukens Steel Melt No. 19281 (Mark 8)

Evaluator	Chemical Analysis (wt-%)								
	C	Mn	P	S	Si	Ni	Cr	Mo	Al
Lukens	0.20	1.27	0.020	0.028	0.21	—	—	0.48	—
NRL	0.23	1.16	0.012	0.025	0.22	0.18	0.08	0.50	0.04

*The heat treatment schedule followed by Babcock and Wilcox on the surveillance steel simulated the heat treatment given to the actual vessel-shell course plates (5). Heat treatment: Austenitized at 1750 to 1800°F, spray quenched to below 600°F; stress relieved eight times at 1100 to 1150°F; soaking times based upon 1 hr/in. of thickness.

report to present a complete notch ductility evaluation of the Yankee reactor in a single document. The tensile specimens from these capsules were to be tested by the reactor builder, Westinghouse Atomic Power Division (WAPD).*

Although the four accelerated capsules noted above were located nominally at the same physical positions within the reactor and were adjacent to fuel elements of the same nominal concentration, the neutron fluence determined for individual capsules was quite different. The capsule-to-capsule variations, which are postulated as having been caused by water-flow turbulence moving the capsules from their assigned locations, ranged from a low fluence of about 5×10^{19} n/cm^2 to a high fluence of about 9×10^{19} n/cm^2 (>1 Mev). These fluence calculations were based upon the assumption of a fission spectrum distribution of neutrons of energies >1 Mev at the exposure locations.

The Charpy-V specimens were tested as follows. All specimens of the Yankee surveillance steel from capsule 8 as well as the reference A302-B steel specimens from all four capsules were tested in the as-irradiated condition. The Yankee surveillance specimens from capsules 1, 2, and 6 were annealed for 168 hr at three different temperatures (640, 750, and 850°F) in order to ascertain the potential for restoration of the initial notch ductility properties of this steel.†

The notch ductility characteristics of the irradiated Yankee surveillance steel specimens from capsule 8 as well as estimates of the transition temperature behavior of specimens from capsules 1, 2, and 6 are presented in Fig. 2. The initial transition temperature of +10°F for the unirradiated condition was established from Charpy V-notch data provided by WAPD and subsequently reconfirmed by NRL. While the curve for capsule 8 was based upon actual tests of surveillance steel specimens, the curves for capsules 1, 2, and 6 are only approximations based upon the data obtained from reference steel specimens in these capsules. The one data point from a Yankee surveillance steel specimen tested in the as-irradiated condition from capsule 6 adds validity to the particular approximation for this capsule.

*As of this writing, tension test results and evaluations have not been published in the open literature by WAPD.

†One Yankee surveillance steel specimen from capsule 6 was tested in the as-irradiated condition. The data point is valuable in giving some indication of the transition temperature increase for the steel in that capsule.

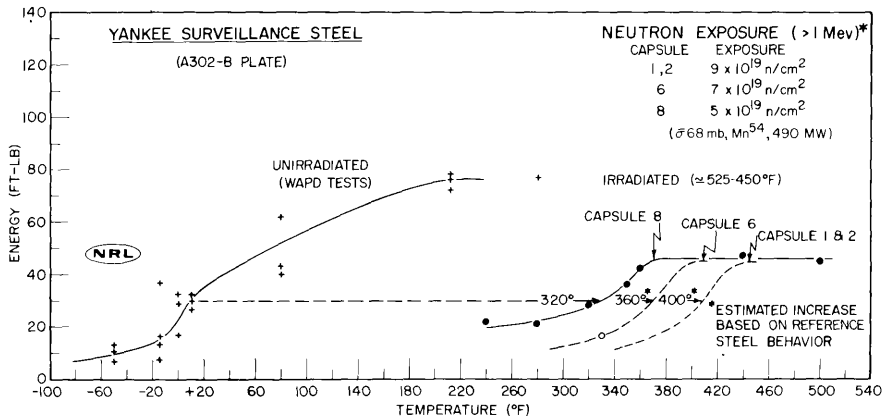


Fig. 2 - Charpy V-notch ductility characteristics of Yankee surveillance steel before irradiation and after exposure in accelerated surveillance locations in the Yankee Reactor during Core II

Charpy curves for the reference A302-B steel from the accelerated capsules are shown in Fig. 3. The initial NDT temperature* for the unirradiated condition (indicated by arrow) was determined by NRL directly through drop-weight tests of another section of the same reference plate (7). Although a very limited number of reference steel specimens were available for the development of these curves, the data give a fair approximation of the Charpy-V 30-ft-lb level for intercomparisons of as-irradiated NDT temperature increase. It is to be noted in Fig. 3 that the specimens from capsules 1 and 2 exhibited properties similar enough to be treated as representing a single exposure condition.

Results of postirradiation annealing at three different temperatures (640, 750, and 850°F) for 168 hr upon the remaining Yankee specimens are shown in Fig. 4. The data curves serve to indicate only the absolute transition temperature after annealing. The magnitude of notch ductility recovery for material contained in capsules 1, 2, and 6 cannot be determined since the reference curves for the as-irradiated condition are best estimates for these capsules. Test results for the as-irradiated and postirradiation annealed conditions as shown in Figs. 2–4 are summarized in Tables 2 and 3.

In contrast to the relatively high neutron flux incident upon the accelerated capsules, the capsules placed in vessel-wall locations were subjected to a much lower neutron flux. Thus, it was planned that the first of such capsules be withdrawn from the reactor only after three core lives (at the end of Core IV). The neutron fluence for this first capsule, based upon the same calculation criterion as used for the accelerated capsule determinations, was 2.2×10^{18} n/cm^2 (>1 Mev).

*A differentiation must 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 NDT 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" of the Charpy curve) which best represents the NDT temperature. This level can then be used for the determination of the neutron-induced NDT increase. A 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). Then, the increase in temperature of the reference value is determined by translating along that energy level to the curve for the irradiated condition.

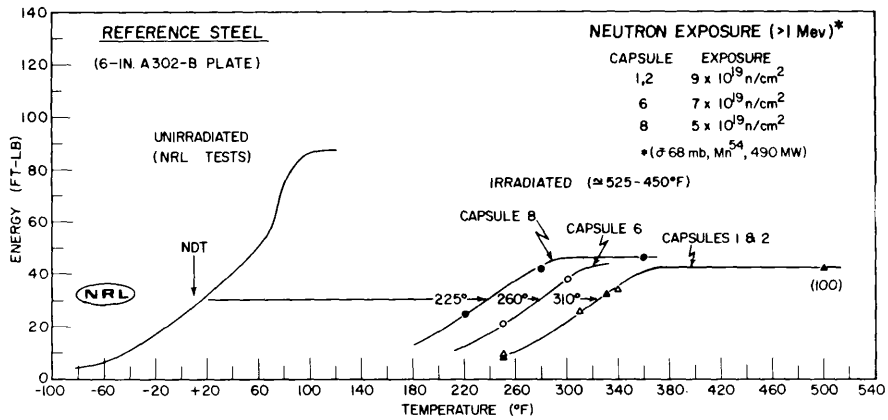


Fig. 3 - Charpy V-notch ductility characteristics of ASTM reference steel before irradiation and after exposure in accelerated surveillance locations in the Yankee Reactor during Core II

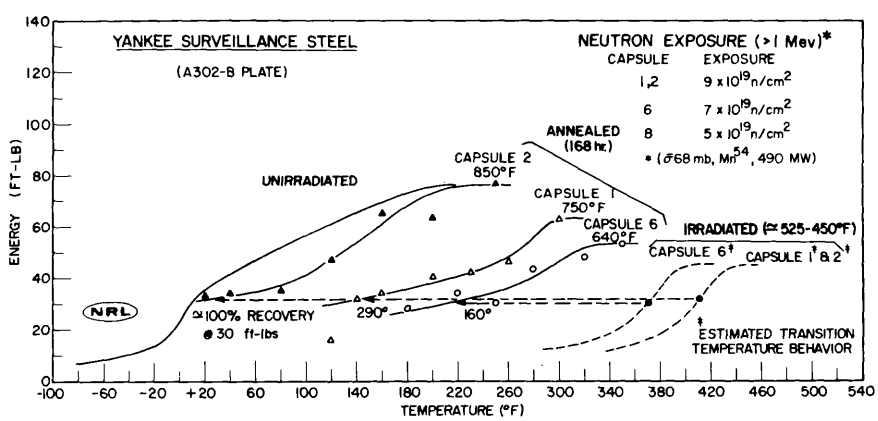


Fig. 4 - The effect of postirradiation heat treatment on the Charpy V-notch ductility characteristics of Yankee surveillance steel after irradiation in accelerated surveillance locations in the Yankee Reactor during Core II

Table 2
The Effect of Irradiation on the Notch Ductility Properties of Yankee Surveillance Specimens

Location and Capsule No.	Specimen Material	Neutron Fluence (n/cm^2) (fission spectrum)		Charpy-V 30-ft-lb Transition Temp. (°F)			Energy Absorption at Full Shear (ft-lb)		
		>1 Mev	2200 m/sec	Unirrad.	Irrad.	ΔT	Unirrad.	Irrad.	Δft-lb
Vessel wall	Yankee steel	2.2×10^{18}	–	10	120	~110	76	62	14
	Reference steel	2.2×10^{18}	–	35	120	85	87	62	25
Accelerated Capsule 1	Yankee steel	9×10^{19}	6.2×10^{20}	10	~410	~400	76	~45	~31
	Reference steel	9×10^{19}	6.2×10^{20}	15	325	310	87	42	45
Accelerated Capsule 2	Yankee steel	9×10^{19}	–	10	~410	~400	76	~45	~31
	Reference steel	9×10^{19}	–	15	325	310	87	42	45
Accelerated Capsule 6	Yankee steel	7×10^{19}	–	10	~370	~360	76	~45	~31
	Reference steel	7×10^{19}	–	15	275	260	87	44	43
Accelerated Capsule 8	Yankee steel	5×10^{19}	4.2×10^{20}	10	330	320	76	46	30
	Reference steel	5×10^{19}	4.2×10^{20}	15	240	225	87	46	41

Table 3
The Effect of Postirradiation Annealing on the Notch Ductility Properties of Yankee Reactor Surveillance Steel Specimens (approximated from limited data)

Capsule No.	Annealing Temperature (°F)	30-ft-lb Transition Temperature (°F)			Percent Recovery	Energy Absorption at Full Shear (ft-lb)		
		Irrad.	Annealed	ΔT (°F)		Irrad.	Annealed	Δft-lb Recovery
2	850	410	10	400	100	45	76	31
1	750	410	120	290	72	45	64	19
6	640	370	210	160	45	45	53	8

As shown in Fig. 5, several Yankee surveillance steel specimens (open circles) taken from a spare, unirradiated capsule, gave test results equivalent to those originally obtained (Fig. 2) in initial control tests by WAPD. Although the aggregate data give some indications of double transition behavior, the data scatter coupled with the limited test points in the range of 20 to 80°F do not permit its confirmation. More importantly, it is noted that either manner of drawing an average curve for transition behavior does not change the initial Charpy-V 30-ft-lb transition temperature of +10°F. Four unirradiated Charpy-V specimens of the A302-B reference steel from the same capsule were also tested by NRL to check the 30-ft-lb transition temperature of that material. These data, shown by the insert of Fig. 5, suggest an initial 30-ft-lb transition temperature of +35°F which is in good agreement with NRL Charpy test data for another plate section of the reference steel.

Referring next to the irradiated condition, the Charpy-V 30-ft-lb transition temperature increase for the Yankee surveillance steel was 110°F, while the corresponding increase for the reference steel was 85°F. Note in Fig. 5 that the data points for both

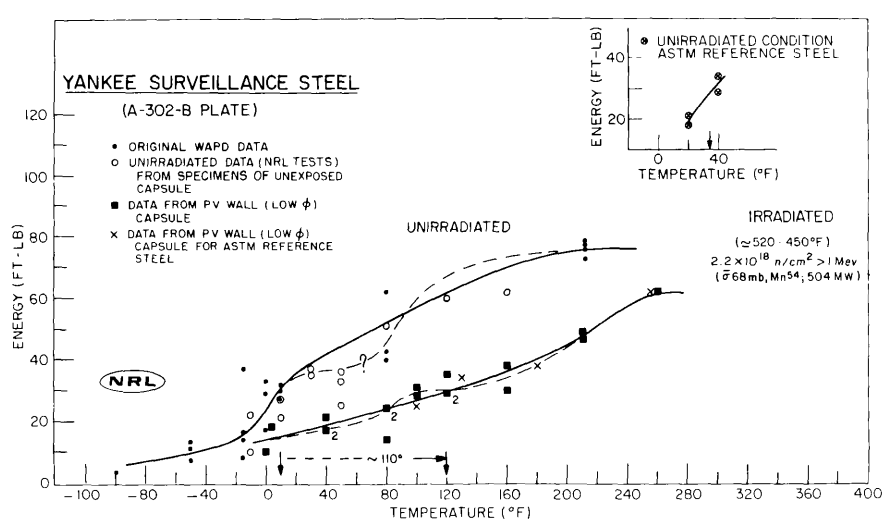


Fig. 5 - Charpy V-notch ductility characteristics of Yankee surveillance steel and ASTM reference steel before irradiation and after exposure at a vessel-wall surveillance location in the Yankee reactor during Cores II, III, and IV

steels in the as-irradiated condition denote a Charpy-V 30-ft-lb transition temperature of +120°F for this particular set of exposure conditions and fluence. While there is again some evidence of a double transition behavior in the case of the Yankee surveillance steel (dashed line), the as-irradiated transition temperature would not be altered by more than 5 or 10°F if this were assumed to be real. The notch ductility results for the vessel-wall location capsule have also been summarized in Table 2 for ready referral.

The surveillance capsule data from Figs. 2, 3, and 5 (Table 2) have been plotted in Fig. 6 versus neutron fluence >1 Mev based upon an assumed fission spectrum. On inspection, this figure has two interesting aspects. First, the Yankee surveillance steel specimens exhibit higher transition temperature increases than do the reference steel specimens contained within the same capsules for both the accelerated and vessel-wall locations. Second, the data points for transition temperature and NDT temperature increases from the vessel-wall location do not appear in the area wherein they might be expected for the recorded exposure temperature and measured (>1 Mev) neutron fluence conditions.

In considering the first of these features, it is pointed out that other observations of significant differences in sensitivity to radiation of the same nominal composition steel have been documented (8,9). Carpenter et al. (8) postulated that microstructural differences were the cause of the "sensitive" and "insensitive" heats of steel which they observed. In the study performed by Hawthorne et al. (9), positive evidence was produced showing that steel from a heavy section plate could be made "sensitive" or relatively "insensitive" to radiation simply by adjusting heat treatment procedures. Furthermore, these differences were confirmed as being directly related to the microstructure developed during such heat treatment. Since it will be remembered that the Yankee surveillance steel was heat treated separately from the main Yankee vessel plates as well as from the reference steel plate, differences in the heat treatment operations can be considered as one potential cause of the observed dissimilarities in radiation response of the two steels. For documentation, the microstructures of the two steels have been reproduced in Fig. 7. It is noted that the microstructure of the surveillance steel is quite coarse compared to that of the reference steel.

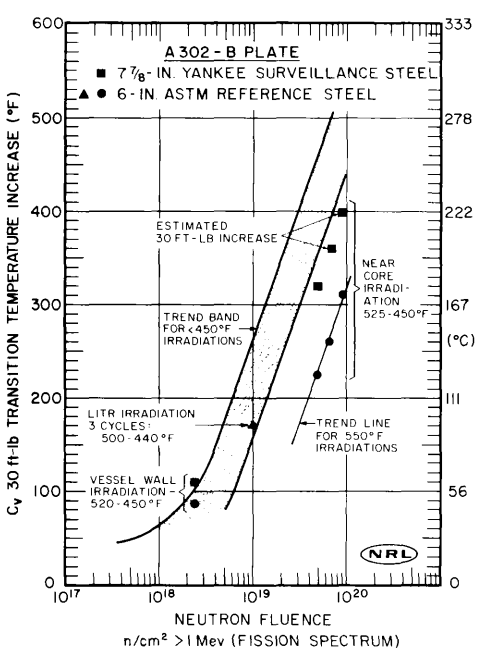
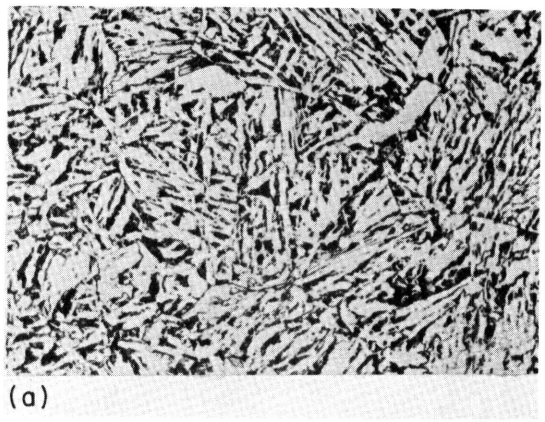
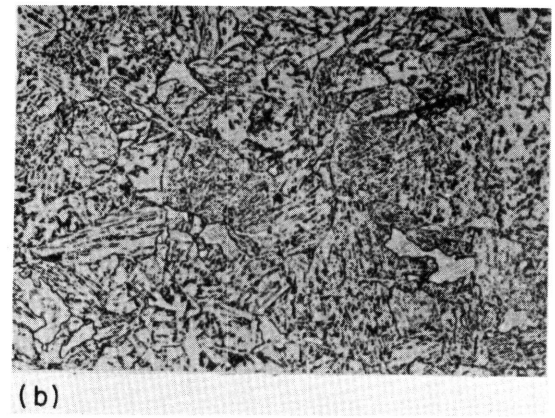


Fig. 6 - Transition temperature and NDT increase (indexed to Charpy-V 30-ft-lb level) versus neutron fluence [n/cm² (>1 Mev), $\bar{\sigma}$ = 68 mb, fission spectrum] for Yankee surveillance steel and ASTM A302-B reference steel. Specimens were irradiated at accelerated and vessel-wall surveillance locations in the Yankee Reactor during Cores II, and Cores II, III, and IV respectively.



(a) Yankee pressure-vessel surveillance steel; tempered upper Bainite



(b) ASTM A302-B reference steel; tempered upper Bainite

Fig. 7 - Microstructures of the Yankee surveillance steel and of the ASTM A302-B reference steel at 500X (nital-picral etchant)

The second unusual feature of Fig. 6 relates to the higher increase in NDT temperature for the vessel-wall surveillance data than would otherwise be expected for the measured damaging fluence. This anomalous effect may be due to either the neutron flux environment or the temperature of exposure.

If we consider the neutron flux environment to be responsible for this anomaly, then the specific cause of this anomaly is believed traceable to the measured ratio between thermal (2200 m/sec) neutron flux and "fast" (>1 Mev) neutron flux. NRL experimentation has revealed that irradiation facilities having thermal-to-fast neutron flux ratios below about 9:1 apparently are unable to produce this increased radiation damage; however, somewhere above this flux ratio, a threshold for increased radiation damage appears to exist. For example, the NDT temperature increases for two common pressure-vessel steels were significantly higher as a result of irradiation in an organic-moderated power reactor (10), an air-graphite moderated research reactor, and in two heavy-water-moderated reactors (11) whose thermal-to-fast neutron flux ratios were respectively 15:1, 22.5:1, 33:1, and 147:1. In comparison, NDT temperature increases were consistently low as a result of irradiation in or near the core of light-water-moderated test reactors wherein the thermal-to-fast neutron flux ratio ranged from about 1:1 up to as much as 8:1.

This increased radiation damage, characterized by an increased NDT temperature, is believed to occur when a thermal neutron interacts with an iron atom causing the iron nucleus to radiate a gamma ray. The recoil of the gamma ray forces the iron atom out of its lattice-work, thus, increasing the amount of radiation damage. This (n,γ) recoil effect was postulated by Wechsler (12), and later given substantial credence by Sheely (13).

The proposed thesis that a higher ratio characterizes increased radiation damage is very well borne out by its extension to the case of the Yankee reactor surveillance data. Recall from Fig. 6 that the accelerated capsule data, for the reference steel at least, compared favorably with the trend established in light-water test reactors. Referring to Table 2, the thermal-to-fast flux ratio for the accelerated location was between 7 and 8:1. Although no definitive thermal flux was determined for the vessel-wall location, it is well known that thermal neutron populations rise rapidly in hydrogeneous environments such as the area between the core shroud and the vessel wall of the Yankee reactor.

Since it is very reasonable to propose a thermal-to-fast neutron flux ratio in excess of 9:1 for the vessel-wall location, it is not unreasonable to suggest that increased radiation damage in the form of higher transition temperatures and an increase in NDT temperature is quite realistic for the vessel wall. Thus, a credible explanation can be advanced to explain the apparently higher sensitivity after irradiation for both the reference and the Yankee steels in the vessel-wall location.

Again, from Fig. 6, the exposure temperature should be considered in evaluating the higher increase in NDT temperature for the vessel-wall surveillance data. Note that reactor operations were not constant over any one core life, but rather were composed of two segments within each core period. The first segment, comprising the first two-thirds of operation, yielded rather constant coolant temperatures between 495 and 505°F; the second segment, or the final one-third of each core life, consisted of a period of steadily decreasing temperatures over the range of about 500 to about 450°F. Furthermore, increased radiation damage effects have been observed (14) as a result of lower temperatures at the end of an irradiation period. Thus, the higher transition temperature increases for the Yankee reactor case may be caused by the exposure temperatures near the end of the irradiation period.

Figure 8 shows the inlet-water temperature history (cold leg temperature) for the three core lives being considered in this report. The reactor inlet-water temperature determines the base temperature for the capsules since it will be observed from Fig. 1 that all of the capsules were located outside of the core baffle (shroud) which is the dividing line between the cool inlet water and the warmer outlet water. WAPD has estimated that gamma heating adds about 15°F to the base temperature for the vessel-wall capsule and about 30°F to the base temperature for the accelerated capsules. From Fig. 8, it can therefore be concluded that the exposure temperature for the vessel-wall capsule over Cores II, III, and IV was between 520 and 450°F, while the exposure temperature for the accelerated capsules (Core II) was between 525 and 450°F.

Experimental Simulation of Reactor Operations

To determine the magnitude of the difference in final transition temperature increase to be expected by operating within these two temperature extremes as opposed to steady state operation, a multiple-temperature experiment was performed by NRL using the Oak Ridge Low Intensity Test Reactor (LITR). The preset schedule followed in operating the irradiation assembly for the experimental simulation is shown in Fig. 9. According to this schedule, one assembly section containing twenty A302-B reference steel Charpy-V specimens was maintained at a constant temperature of 500°F, while another duplicate section was subjected to three cycles of constant 500°F operation followed by steadily decreasing temperatures to about 440°F.

The notch ductility behavior of the A302-B reference steel before and after irradiation under constant and under cyclic temperature conditions is depicted in Fig. 10 (15). After irradiation under cyclic temperature conditions, a transition temperature increase of 170°F was observed, but more significantly, the data points for the constant 500°F temperature are seen to fall about the curve described by the data points for the cyclic 500 to 440°F temperature. In other words, no major difference in notch ductility behavior between the constant temperature as opposed to cyclic temperature operations was found. For the unirradiated condition, check tests on the 30-ft-lb transition temperature of the particular section of plate used for this experiment ("x" data points) were about 25°F higher than the average for the entire plate (shaded area). The variations in transition temperature illustrate the need for extensive control tests in such critical experiments.

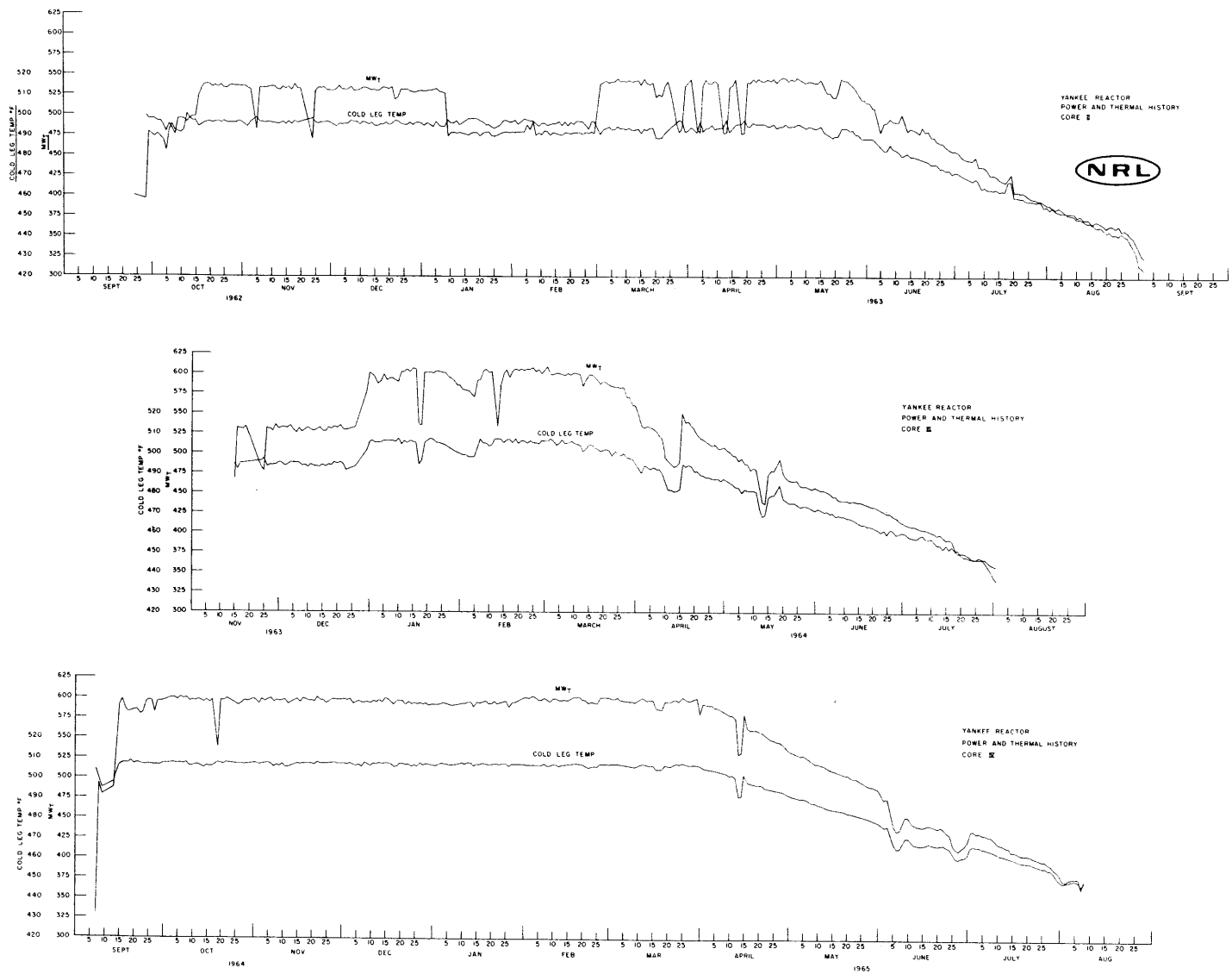


Fig. 8 - Inlet water temperature and thermal power levels for the Yankee Reactor during Cores II, III, and IV

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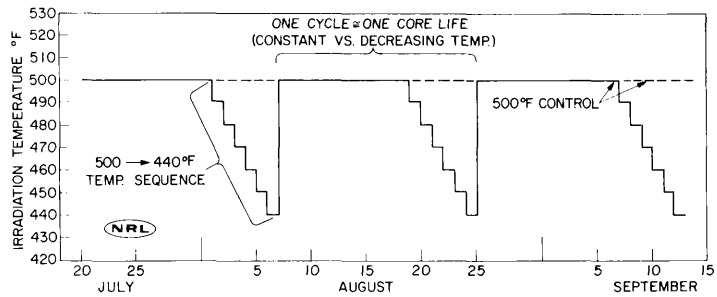

Fig. 9 - Irradiation temperature history of two groups of Charpy V-notch specimens exposed in the LITR to determine the effect of constant 500°F versus cyclic 500 to 440°F service temperatures upon the embrittlement of a reactor pressure-vessel steel

To complete this particular experiment, a number of specimens were annealed at 650°F for 168 hr to compare their relative responses to postirradiation heat treatment following irradiation under constant conditions as opposed to cyclic temperature conditions. The annealing parameters selected were considered within the capabilities of the Yankee reactor using nuclear heat alone. The data in Fig. 10 show that about 53 percent of the initial transition temperature shift was recovered for the specimens representing both the constant and the cyclic operating conditions, and that residual embrittlement amounted to ~80°F in each case.

In order to critically assess the data from this experiment for engineering applicability, several factors which relate to the experimental observations are reviewed. First, it will be noted that the specimens held at a constant 500°F received a measured fluence of 1.08×10^{19} n/cm^2 (>1 Mev), while specimens irradiated under the 500 to 440°F cyclic

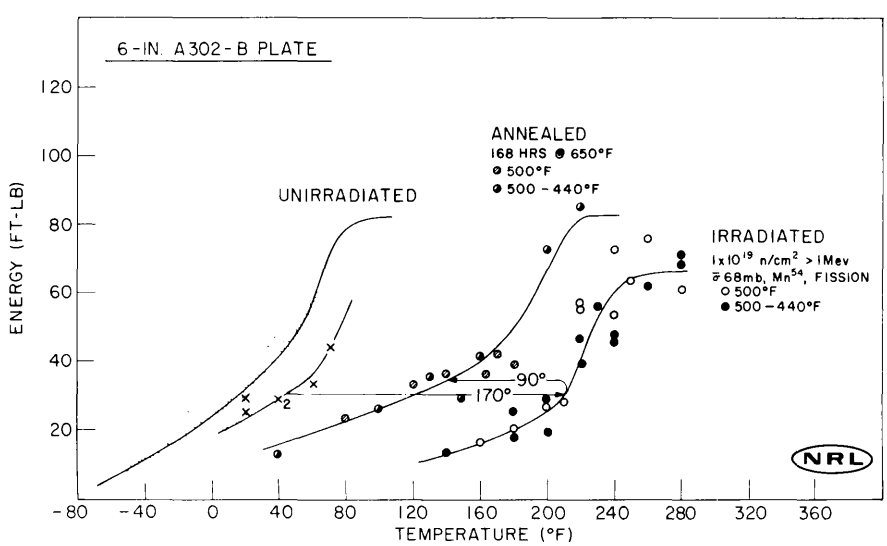

Fig. 10 - Charpy V-notch ductility characteristics of ASTM reference A302-B steel before irradiation and after exposure under constant 500°F and cyclic 500 to 440°F temperature conditions

temperature pattern received a fluence of 0.91×10^{19} n/cm^2. If the specimens in the cyclic temperature group had received the same, higher neutron fluence as did the specimen group held at constant temperature, a somewhat higher transition temperature increase might be conceivable. From the known irradiation response of this steel, however, a significantly higher degree of embrittlement would not be projected for the noted fluence difference.

The time period of irradiation is the second factor to be considered in reviewing the results of this experiment. The experimental irradiation encompassed an 8 wk period and the specimens received a neutron fluence of $\sim 1 \times 10^{19}$, nearly five times that of the Yankee reactor pressure vessel during the referenced three-year-operating period (Cores II, III, and IV). Therefore, the experimental conditions were much more severe, in terms of neutron fluence, than the reactor case in providing for a buildup of accelerated radiation embrittlement during a reduced temperature period.

One further question regarding time at temperature must also be considered. The presence or absence of nonradiation-related property changes such as thermal aging, where the operating reactor case is much more severe than the experimental case, has not been established. In this regard, unpublished results of thermal aging studies by NRL on unirradiated Charpy-V specimens of A302-B steel for 2400 hr at 450, 500, and 550°F have not revealed any increase in transition temperature as a result of these heat treatments. This length of time, while about 50 percent longer than the experimental irradiation period, nevertheless amounted to barely 10 percent of the time at temperature of capsules contained within the Yankee reactor during three full fuel cycles. Long-term metallurgical processes such as thermal aging, therefore, bear reconsideration when seeking possible explanations for the enhanced effects of actual service such as the higher transition temperature increase noted for the Yankee surveillance steel.

The foregoing analysis may be considered as invalidating the earlier thesis of significantly higher pressure vessel embrittlement for those pressurized water reactors (PWR) such as the Yankee reactor which have lower operating temperatures near the end of each fuel cycle. On the other hand, the absence of any deleterious effects directly attributable to the cyclic lower temperatures may be evidence of significant self-annealing during periods of 520 to 500°F operation at the start of each new cycle, or may be directly related to the relatively small amount of neutron fluence received by the Yankee surveillance capsules during lower temperature operations. Finally, in view of the apparently conflicting information discussed concerning the response of A302-B steel to irradiation under cyclic temperature conditions, and how this information might apply to an actual operating reactor, the value of mechanical property data developed from surveillance programs representative of actual operations in a reactor cannot be overstated.

REACTOR VESSEL NEUTRON SPECTRUM AND FLUX

Previous studies (16-18) revealed that all the data from diverse reactor systems can be most satisfactorily interrelated if the neutrons indexed are those determined to have energies >0.5 Mev rather than >1 Mev or >0.183 Mev within the calculated spectra of the irradiation facilities involved. Nevertheless, the irradiation facilities having the >9:1 thermal-to-fast neutron ratios produce results which are at best only poorly interrelated. The conclusions in a later section of this report will be based upon neutron fluences of n/cm^2 (>0.5 Mev), since it is presently the best defined and most accurate index proposed for use.

Neutron Spectrum

Theoretical neutron spectra relating to the Yankee reactor are presented in Fig. 11 for the interface between the low alloy-steel pressure vessel and the stainless-steel

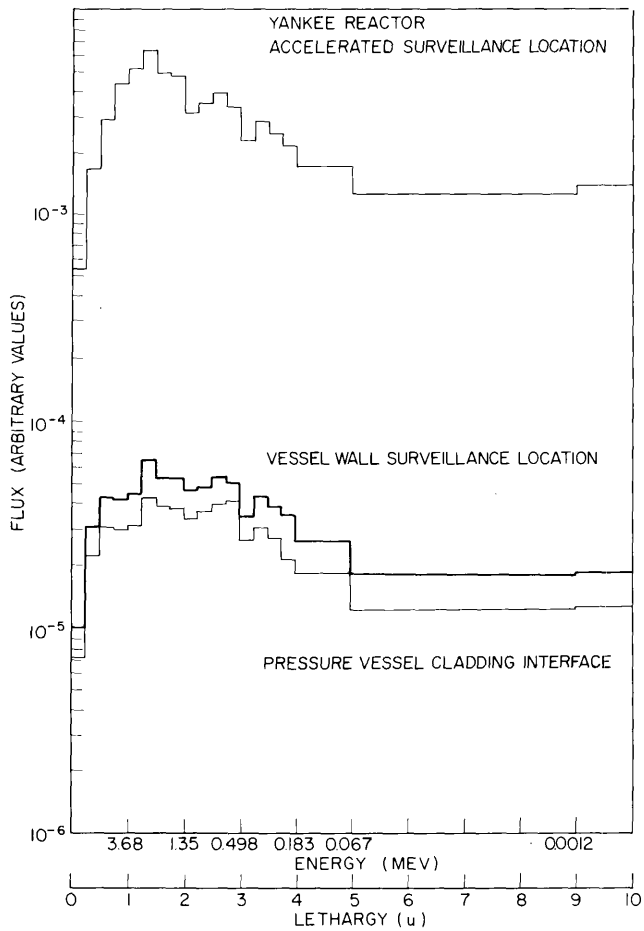

Fig. 11 - Graphical representation and relative intensities of the neutron spectra at the accelerated and vessel-wall surveillance locations in the Yankee Reactor and at the interface between the alloy-steel pressure-vessel wall and stainless-steel cladding

cladding (PVW/SS clad) as well as for the accelerated and vessel-wall surveillance locations. These were calculated, under contract to NRL, by Dahl and Ulseth of the Battelle-Northwest Laboratory using a transport theory, Sn reactor physics code "Program S" (19). The calculated, arbitrary unit fluxes per energy group for those spectra are shown in Table 4. From the output of "Program S," Dahl and Ulseth redetermined the cross-section values for the $Fe^{54}(n,p)Mn^{54}$ neutron flux monitor reaction for correspondence with the actual spectral distribution of neutrons at the three locations (Table 5) as opposed to the nominal distribution in a fission spectrum (Fig. 12). This redetermination makes it possible to transpose the fission spectrum neutron fluxes initially developed from surveillance capsule neutron detectors (as noted on Figs. 2, 3, and 5) into calculated-spectrum neutron fluxes which include neutrons of energies greater than the lower energy limits (E_L) of 1, 0.5, or 0.183 Mev.

Table 4
Calculated Neutron Flux (Arbitrary Units) per Energy Group for Three Locations in the Yankee Reactor

Energy Group Structure			Calculated Relative Neutron Flux* (n/cm^2-sec)		
Group No.	Lethargy (u)	Lower Energy Limit (E_L) (ev)	Accelerated†	Vessel Wall‡	PVW/SS Clad§
1	0.25	7.78×10^{6}	5.409×10^{-4}	1.006×10^{-5}	7.182×10^{-6}
2	0.50	6.07×10^{6}	1.682×10^{-3}	3.121×10^{-5}	2.260×10^{-5}
3	0.75	4.72×10^{6}	2.912×10^{-3}	4.345×10^{-5}	3.132×10^{-5}
4	1.00	3.68×10^{6}	3.385×10^{-3}	4.211×10^{-5}	3.016×10^{-5}
5	1.25	2.87×10^{6}	4.183×10^{-3}	4.503×10^{-5}	3.172×10^{-5}
6	1.50	2.23×10^{6}	6.383×10^{-3}	6.271×10^{-5}	4.357×10^{-5}
7	1.75	1.74×10^{6}	4.923×10^{-3}	5.354×10^{-5}	3.912×10^{-5}
8	2.00	1.35×10^{6}	4.777×10^{-3}	5.364×10^{-5}	3.847×10^{-5}
9	2.25	1.05×10^{6}	3.119×10^{-3}	4.642×10^{-5}	3.446×10^{-5}
10	2.50	8.21×10^{5}	3.491×10^{-3}	4.850×10^{-5}	3.698×10^{-5}
11	2.75	6.39×10^{5}	3.944×10^{-3}	5.473×10^{-5}	4.060×10^{-5}
12	3.00	4.98×10^{5}	3.364×10^{-3}	5.082×10^{-5}	4.194×10^{-5}
13	3.25	3.88×10^{5}	2.298×10^{-3}	3.456×10^{-5}	2.709×10^{-5}
14	3.50	3.02×10^{5}	2.840×10^{-3}	4.329×10^{-5}	3.106×10^{-5}
15	3.75	2.35×10^{5}	2.487×10^{-3}	3.866×10^{-5}	2.784×10^{-5}
16	4.00	1.83×10^{5}	2.182×10^{-3}	3.514×10^{-5}	2.187×10^{-5}
17	5.00	6.74×10^{4}	6.817×10^{-3}	1.052×10^{-4}	7.345×10^{-5}
18	9.00	1230.00×10^{4}	1.968×10^{-2}	2.900×10^{-4}	1.991×10^{-4}
19	13.00	22.60×10^{4}	2.180×10^{-2}	2.927×10^{-4}	2.031×10^{-4}
20	17.00	0.414	2.323×10^{-2}	2.703×10^{-4}	1.828×10^{-4}
21	∞	0.0	8.753×10^{-1}	2.099×10^{-3}	6.802×10^{-4}
		ϕ Total	1.000	3.751×10^{-3}	1.845×10^{-3}
		ϕ ($E_L > 1.0$ Mev)	3.260×10^{-2}	3.979×10^{-4}	2.860×10^{-4}
		ϕ ($E_L > 0.5$ Mev)	4.270×10^{-2}	5.422×10^{-4}	3.981×10^{-4}
		ϕ ($E_L > 0.18$ Mev)	5.247×10^{-2}	6.939×10^{-4}	5.060×10^{-4}
		ϕ ($E_L > 0.067$ Mev)	5.929×10^{-2}	7.991×10^{-4}	5.794×10^{-4}

*The low values for the fluxes are only peculiarities of the computer program output. They are not to be construed as real flux values.
†3.515 in. outside core baffle.
‡0.75 in. outside thermal shield.
§Interface between low alloy steel wall and stainless-steel cladding (clad thickness, 0.110 in.).

Table 5
Average Iron-Activation Cross Section (mb) in the Yankee Reactor
Based on the Two Assumed Fission-Spectrum, $\bar{\sigma}$ Values

Lower Energy Limit E_L	Accelerated		Vessel Wall		PVW/SS Clad	
	$\bar{\sigma}_f$ (68 mb)	$\bar{\sigma}_f$ (82.6 mb)	$\bar{\sigma}_f$ (68 mb)	$\bar{\sigma}_f$ (82.6 mb)	$\bar{\sigma}_f$ (68 mb)	$\bar{\sigma}_f$ (82.6 mb)
1 Mev	125.3	149.3	144.1	171.7	143.5	170.9
0.5 Mev	96.6	114.9	106.2	126.5	103.5	123.3
0.183 Mev	78.6	93.6	82.8	98.6	81.3	96.8

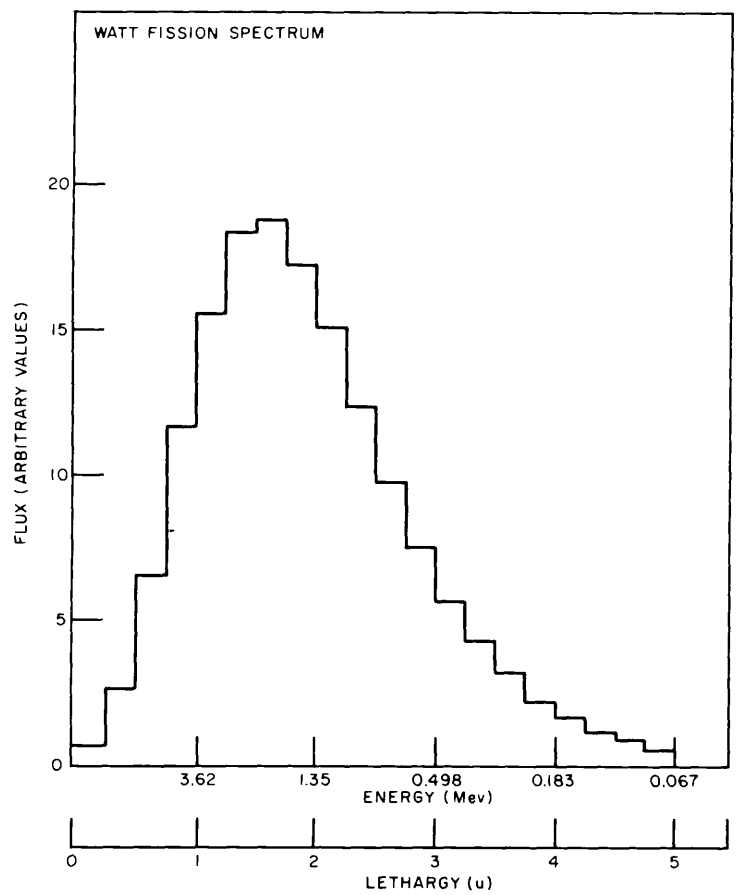

Fig. 12 - Graphical representation of the Watt fission spectrum

Reactor Vessel Neutron Flux

Neutron fluence values for the surveillance embrittlement data given in Figs. 2-6 are in terms of n/cm^2 (>1 Mev), and assume a fission spectrum to be present at all locations. A fission-spectrum-averaged cross section of 68 mb has been used for the

$Fe^{54}(n, p)Mn^{54}$ reaction. Furthermore, the fluence data for the capsules correspond to the average power level during the residence period. Because of the physical displacement of the surveillance capsules from the pressure-vessel wall, the listed fluences do not represent the neutron fluence at the inner edge of the A302-B steel vessel wall but, rather, points nearer to the fuel core.

In Table 6, the fission spectrum flux and fluence data from the two surveillance locations have been individually adjusted by a series of procedures whereby a fluence value is developed for the PVW/SS clad interface of the vessel wall after thirty fuel cycles of operation, the design criterion for the Yankee reactor.

It is necessary to give an explanation of the sequential development of data in Table 6. The calculational sequence begins by giving the flux values in terms of the fission-spectrum-averaged cross section for 68 mb for the $Fe^{54}(n, p)Mn^{54}$ reaction as determined by Shure (20), whose work was largely based on that of Carroll and Smith (21). Although 68 mb is presently in fairly common use, Helm (22) has arrived at a value of 82.6 mb for this same reaction, while 97.2 mb has been deduced by Barrall and McElroy (23). In recognition of the body of evidence for a fission-averaged Fe^{54} cross section higher than 68 mb, the remaining calculations in Table 6 (columns 2-6) have been performed using 82.6 mb.

The vessel-wall capsule exposure period averaged 504 Mw, while the accelerated capsule exposure period averaged 490 Mw as shown in column 1. In both cases, the listed instantaneous flux values in n/cm^2-sec were determined using 68 mb as the $Fe^{54}(n, p)$ Mn^{54} reaction cross section averaged over the entire fission spectrum.

In column 2 the fission spectrum fluxes for neutrons >1 Mev have been extrapolated from 504 and 490 Mw to 600 Mw, the maximum power of the Yankee reactor. Then, the 600-Mw flux at 68-mb fission-averaged cross section was further adjusted to represent 600-Mw flux at the 82.6-mb fission-averaged cross section.

The values for the spectrum-averaged cross sections for flux >0.5 Mev in the calculated spectrum, as taken from Table 5, are in column 3.

In column 4, the fission-spectrum fluxes are adjusted to comply with the calculated spectra. In adjusting neutron fluxes to account for different cross sections, it should be remembered that the products of neutron flux and cross section for one criterion [such as n/cm^2-sec (>1 Mev), fission spectrum] must equal the product of neutron flux and cross section for any other criterion [such as n/cm^2-sec (>0.5 Mev), calculated spectrum] as shown below

$$\phi_f\ (>1\ \text{Mev}) \times \bar{\sigma}_f\ (>1\ \text{Mev}) = \phi_{cs}\ (>0.5\ \text{Mev}) \times \bar{\sigma}_{cs}\ (>0.5\ \text{Mev}). \tag{1}$$

For Eq. (1), ϕ_f (>1 Mev) is the flux of neutrons having energies >1 Mev in a fission spectrum, $\bar{\sigma}_f$ (>1 Mev) is the cross section for neutrons having energies >1 Mev in a fission spectrum, ϕ_{cs} (>0.5 Mev) is the flux of neutrons having energies >0.5 Mev in a calculated spectrum, and $\bar{\sigma}_{cs}$ (>0.5 Mev) is the cross section for neutrons having energies >0.5 Mev in a calculated spectrum.

Completion of column 4 requires calculation of the term ϕ_{cs} (>0.5 Mev), according to Eq. (1). This is possible since values for the other three terms are known. Values for $\bar{\sigma}_{cs}$ (>0.5 Mev) are listed in column 3 and values for ϕ_f (>1 Mev) are listed in column 2. The term $\bar{\sigma}_f$ (>1 Mev) is determined as follows.

First, recall that Helm's value of 82.6 mb is being employed as the cross section for the $Fe^{54}(n, p)$ reaction as averaged over the entire fission spectrum (22). However, the fluxes shown in column 2 of Table 6 are for n/cm^2-sec of energies >1 Mev only, not

Table 6
Development of Maximum Neutron Fluence Values for the Yankee Reactor Pressure Vessel*

Flux Monitor Location	1	2	3	4	5	6
	Fission Spectrum† for $\phi_f > 1$ Mev $\bar{\sigma}_f = 68$ mb (n/cm^2-sec)	$\phi_f > 1$ Mev (n/cm^2-sec)	$\bar{\sigma}_{cs}$ (mb) Spectrum-Averaged Cross Section for $\phi_{cs} > 0.5$ Mev	ϕ_{cs} (>0.5 Mev) Adjustment of Fission-Spectrum flux (n/cm^2-sec)	Extrapolated ϕ_{cs} (n/cm^2-sec) at PVW/SS Clad Interface [n/cm^2-sec (>0.5 Mev)]	Maximum Fluence Φ_{cs} (n/cm^2)
Vessel Wall	2.73×10^{10}	2.67×10^{10}	126.5	2.51×10^{10}	1.84×10^{10}	1.46×10^{19}
Accelerated	3.25×10^{12}	3.28×10^{12}	114.9	3.40×10^{12}	3.16×10^{10}	2.50×10^{19}
	3.06×10^{12}	3.09×10^{12}	114.9	3.20×10^{12}	2.98×10^{10}	2.36×10^{19}
	2.56×10^{12}	2.58×10^{12}	114.9	2.67×10^{12}	2.48×10^{10}	1.96×10^{19}
	1.69×10^{12}	1.70×10^{12}	114.9	1.76×10^{12}	1.64×10^{10}	1.30×10^{19}

*All data except for column 1 are extrapolated to 600 Mw(t) operations and $\bar{\sigma}_f = 82.6$ mb.
†For 504 Mw(t) at the vessel-wall location and 490 Mw(t) for the accelerated location.

of energies over the entire energy spectrum. Since 69.3 percent of the neutrons in a fission spectrum have energies >1 Mev, the fission-spectrum-averaged cross section of 82.6 mb is divided by 69.3 percent to yield the cross section in a fission spectrum in n/cm^2-sec (>1 Mev).

$$\frac{82.6 \text{ mb}}{0.693} = 119.1 \text{ mb} = \bar{\sigma}_f \text{ (>1 Mev)}. \tag{2}$$

Computation of the calculated spectrum fluxes are carried out as shown in a selected example repeating Eq. (1)

$$\phi_f \text{ (>1 Mev)} \times \bar{\sigma}_f \text{ (>1 Mev)} = \phi_{cs} \text{ (>0.5 Mev)} \times \bar{\sigma}_{cs} \text{ (>0.5 Mev)} \tag{1}$$

and substituting the values for the vessel-wall position

$$2.67 \times 10^{10} \text{ n/cm -sec} \times 119.1 \text{ mb} = \phi_{cs} \text{ (>0.5 Mev)} \times 126.5 \text{ mb}$$

yields

$$\phi_{cs} \text{ (>0.5 Mev)} = 2.51 \times 10^{10} \text{ n/cm}^2\text{-sec}$$

In column 5, the fluxes at the accelerated and vessel-wall locations are extrapolated to the PVW/SS clad interface at the vessel wall. This is accomplished using the calculated relative flux values for the energy level from Table 4.

As an example, ratios are obtained for one energy level using the calculated relative flux value at the PVW/SS clad over the calculated relative flux at one of the surveillance locations, such as

$$\frac{\phi_{cs} \text{ (>0.5 Mev) PVW/SS clad surveillance location}}{\phi_{cs} \text{ (>0.5 Mev) accelerated location}} = \frac{3.981 \times 10^{-4*} \text{ n/cm}^2}{5.422 \times 10^{-4} \text{ n/cm}^2} = 0.734.$$

Then, this ratio multiplied by the adjusted fission-spectrum flux at that location for the same energy level gives the extrapolated value at the PVW/SS clad interface, or

$$\phi_{cs} \text{ (>0.5 Mev)} \times 0.734 = 2.51 \times 10^{10} \text{ n/cm}^2\text{-sec} \times 0.734 = 1.84 \times 10^{10} \text{ n/cm}^2\text{-sec}.$$

In column 6, the PVW fluxes for neutrons >0.5 Mev as determined for column 5 are integrated over the time period expected for operation of the Yankee reactor. This time translates into 7.92×10^8 sec (24), and results in a maximum fluence of 1.46×10^{19} n/cm^2 (>0.5 Mev) determined from the vessel-wall capsule data.

Discussion

The final column in Table 6 shows a range of fluence values for various positions for the thirty cycles (30 yr) of expected operations for the Yankee reactor assuming a power level of not less than 600 Mw(t). The four different values for the accelerated location capsules arise from the fair assumption that these four capsules were all displaced to some extent from their intended physical locations within the reactor (6). Since the one vessel-wall capsule apparently was not displaced from its planned position, the monitor data allow the development of a more accurate fluence value for extrapolation to

*The low values for the fluxes are only pecularities of the computer program output. They are not to be construed as real flux values.

the alloy-steel vessel wall. The fact that fluence data for the vessel-wall surveillance location fall within the spread of data extrapolated from the accelerated surveillance locations lends additional confidence to the data (Table 5, column 5).

In an earlier study by WAPD, a maximum fluence of 2.5×10^{19} n/cm^2 (>1 Mev) was calculated (24) for the Yankee reactor vessel using a corrected P1-MG diffusion code with additional correction factors for spatial, axial peaking, and corner effects. This value was also based upon a power level not less than 600 Mw(t). This value and the NRL value for 30-cycle fluences in the Yankee reactor are considered to be in good agreement in view of the complexity of the computer programs and the physical size of the reactor. Furthermore, somewhat different flux values from the vessel-wall capsule were used by both NRL and WAPD for development of the respective 30-cycle fluences (25). These fluxes were developed by independent radiochemical analyses by both NRL and WAPD of the induced Mn^{54} from duplicate samples drilled out of seven of the Charpy specimens from the vessel-wall capsule.

EMBRITTLEMENT AS A FUNCTION OF NEUTRON FLUENCE

Comparison of Surveillance Versus Test Reactor Data

In the previous section, a calculated-spectrum neutron fluence of 1.46×10^{19} n/cm^2 (>0.5 Mev) was developed for 600 Mw(t) operation at the PVW/SS clad interface of the Yankee reactor. Accordingly, it will be of considerable interest to determine the corresponding amount of radiation-induced embrittlement to be expected for these conditions. Since the NDT temperature data corresponding to the fluence level at the PVW/SS clad reactor location cannot be obtained directly, only an estimate may be made based upon as much other embrittlement-fluence data as is available.

Figure 13 is a graphical compilation representing a large body of data from Table 7 which was developed by NRL for A302-B steel (6,7,15,25-29). The sequential development of data in Table 7 is quite similar to that used in Table 6. Column 2 lists irradiation temperatures for individual, well documented neutron exposures of A302-B steel. Columns 4 and 5 give, respectively, the fluence based on the fission-spectrum-averaged

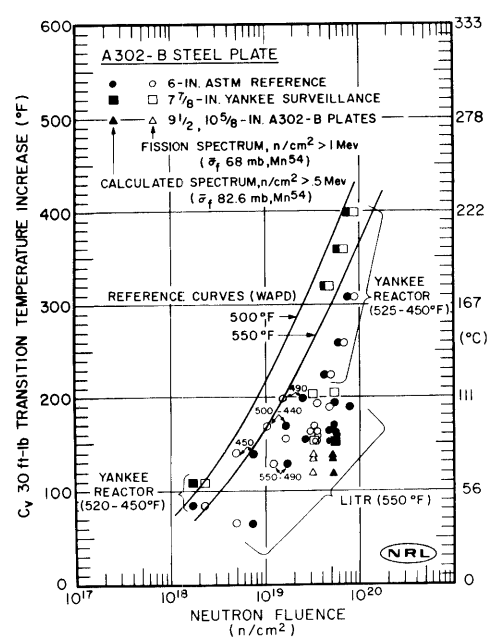

Fig. 13 - Transition temperature and NDT increase (indexed to Charpy-V 30-ft-lb level) versus neutron fluence determined using an assumed fission spectrum for interrelating the >1 Mev data (open points) and determined using calculated spectra for interrelating the >0.5 Mev data (closed points). The data are tabulated and referenced in Table 7.

Table 7
30-ft-lb, ΔNDT, and Neutron Fluence Data for Elevated Temperature Irradiations of A302-B Steel

1	2	3	4	5	6	7	8
Steel	Irradiation Temperature (°F)	ΔNDT (°F) for 30 ft-lb level	Φ_f [n/cm$^2 \times 10^{19}$ (>1 Mev)] for $\bar{\sigma}_f$ = 68 mb	$\bar{\sigma}_{cs}$ [mb (>0.5 Mev)] for $\bar{\sigma}_f$ = 82.6 mb	Φ_{cs} [n/cm$^2 \times 10^{19}$ (>0.5 Mev)]	Irradiation Facility	Reference
ASTM*	450	140	0.5	65.2	0.75	C-18§	7
ASTM	490	200	1.4	57.3	2.5	C-55	7
ASTM	500/440	170	1.0	61.8	1.6	C-57	15
ASTM	520/450	85	0.22	126.5	0.17	VW-Y**	25
YS†	520/450	110	0.22	126.5	0.17	VW-Y	25
ASTM	525/450	225	5.0	114.9	4.2	A-Y††	6
ASTM	525/450	260	7.0	114.9	6.0	A-Y	6
ASTM	525/450	310	9.0	114.9	7.7	A-Y	6
YS	525/450	320	5.0	114.9	4.2	A-Y	6
YS	525/450	360	7.0	114.9	6.0	A-Y	6
YS	525/450	400	9.0	114.9	7.7	A-Y	6
ASTM	550/490	130	1.1	61.8	1.7	C-57	15
SP‡	550	120	3.1	57.3	5.3	C-55	26
SP	550	135	3.1	57.3	5.3	C-55	26
SP	550	140	3.1	57.3	5.3	C-55	26
ASTM	550	155	1.5	58.5	2.4	C-55	27
ASTM	550	155	3.1	57.3	5.3	C-55	26
YS	550	155	3.1	57.3	5.3	C-55	26
ASTM	550	155	3.3	65.2	4.9	C-18	27
ASTM	550	160	3.3	61.1	5.3	C-43	15
ASTM	550	165	3.0	65.2	4.6	C-18	28
ASTM	550	165	3.3	61.1	5.3	C-43	15
ASTM	550	170	3.1	57.3	5.3	C-55	26
ASTM	550	190	4.7	58.5	7.9	C-55	27
ASTM	550	195	3.3	61.1	5.3	C-43	29
YS	550	205	3.1	57.3	5.3	C-55	26

*ASTM reference A302-B steel.
†YS - Yankee Surveillance steel.
‡SP - Special 9-1/2 and 10-5/8 in. thick A302-B plates.
§C-(xx) from Low Intensity Test Reactor.
**VW-Y - Vessel wall, Yankee reactor.
††A-Y - Accelerated location, Yankee reactor.

cross section of 68 mb in n/cm^2 (>1 Mev) and the spectrum-averaged cross section based on Helm's value of 82.6 mb for >0.5 Mev. All of the data from Table 7 have been plotted in Fig. 13, where open points represent fission spectrum->1 Mev fluences and closed points represent calculated spectrum->0.5 Mev fluences. Each point is identified with regard to the reactor irradiation facility and temperature of irradiation; thus, the relationship of the data to the 500 and 550°F WAPD "design curves" (5), which are included for reference only, should be apparent.

From Fig. 13 it is seen that an attempt to predict radiation embrittlement behavior for cyclic exposure temperatures in the range of 520 to 450°F will not be as straightforward as for service conditions at constant temperatures of 450, 500, or 550°F. It is considered fair, however, to assume that the rate of vessel steel embrittlement will not exceed the embrittlement rate exhibited by the vessel-wall exposure (just to the left of the 500°F "design curve"). This embrittlement rate assumption is reinforced by the similarity of the spectral shapes for the accelerated and vessel-wall surveillance locations with that for the actual pressure-vessel wall location (Fig. 11). With this in mind, then, a maximum radiation-induced transition temperature increase of ~265°F can be projected for the Yankee reactor vessel steel at 600-Mw(t) operations based upon the performance of the Yankee surveillance steel for the projected 30 fuel cycle service fluence of 1.46×10^{19} n/cm^2 (>0.5 Mev). Note that this value is the maximum increase possible and assumes all operations will be at 600 Mw(t), a situation which cannot be met. Thus, the increase in transition temperature to be expected after thirty cycles of operation must be less than the 265°F indicated by Fig. 13.

Evaluation of the Significance of Pressure Vessel Embrittlement

All of the evaluations presented thus far have been addressed to determinations of the fluence and associated embrittlement at the PVW/SS clad interface of the pressure vessel. It is pointed out that at no other location in the pressure vessel will either the fluence or the degree of embrittlement be as high. These two values decrease through the thickness of the reactor vessel, thus yielding a satisfactory vessel condition after 30 fuel cycles from the standpoint of transition temperature properties, assuming that core configurations and power levels do not change significantly.

Studies conducted at NRL (29) have determined both the embrittlement and neutron flux changes which may be expected through the thickness of a 6-in. steel plate. For this determination, a test block was located at the face of a pool reactor which contained Charpy V-notch specimens and neutron flux monitors in special capsule assemblies at five locations through the block thickness. The results of two in-depth embrittlement studies revealed that the notch ductility of steel specimens located within the test block corresponded to predictions from other test reactor irradiations of this type steel. Additionally, the neutron spectra calculated by Dahl and Yoshikawa for these same test block locations (16) revealed parallel trends in relative neutron flux values. The measured- and calculated-spectrum neutron flux values for this study depicted quite similar decreases in flux intensity (in the range of 95 percent) from the front to the rear of the 6-in.-thick block.

The excellent agreement between measured and calculated flux decreases for the simulated vessel-wall case suggests that similar agreement should be obtained on examining other vessel-wall situations. In the Yankee reactor, or any other reactor for that matter, it is not possible to measure the flux at finite locations in the vessel wall for obvious reasons. However, if the results of this single study can be applied, it should be possible to calculate the trend in flux decrease through the Yankee vessel and therein make reasonably accurate estimates of the maximum fluence to be expected.

Accordingly, Dahl and Ulseth calculated the neutron flux at successive locations through the thickness of the Yankee vessel (30). These values, shown in Fig. 14 are presented in Table 8 along with corresponding 30-cycle fluences and transition temperature increases as taken from Fig. 13. Note, for example, that 3 in. away from the inner surface of the vessel wall the projected transition temperature increase is 55°F less than at the maximum of the PVW/SS clad interface.

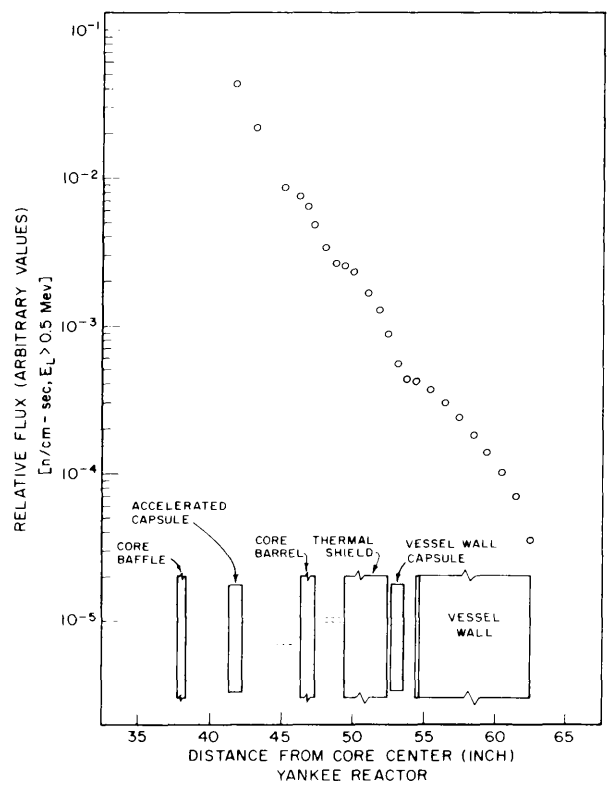

Fig. 14 - Graphical representation of the neutron flux intensity from the fuel core of the Yankee reactor to the exterior of the pressure-vessel wall. The neutron fluxes are derived from a calculated spectrum and only account for neutrons whose energies are greater than the lower energy limit (E_L) > 0.5 Mev. The low values for the fluxes are simply peculiarities of the computer program output and are not to be construed as real flux values.

Although the maximum amount of embrittlement depicted by the Yankee reactor surveillance steel is relatively high, the lower limit of reactor operating temperatures is not closely approached by application of the "NDT plus 60°F" criterion. Furthermore, even if the initial transition temperature of the Yankee reactor were somewhat higher than +10°F due to its "processing" during fabrication (31), the maximum increase in transition temperature including the "NDT plus 60°F" still would probably not reach the operating temperature. Naturally, the steel located through the thickness would be

Table 8
Relative Neutron Fluxes, Fluences, and Corresponding 30-ft-lb ΔNDT Values in the Yankee Reactor Reflector Region and Pressure-Vessel Wall

Location	Distance from Core Center (In.)	Relative Flux* (>0.5 Mev)	Φ_{cs} (>0.5 Mev) (n/cm^2)	ΔNDT† (°F)
Accelerated capsule	41.85	4.27×10^{-2}	–	–
Water	43.35	2.17×10^{-2}	–	–
Water	45.35	8.67×10^{-3}	–	–
Core barrel	46.35	7.42×10^{-3}	–	–
Core barrel	46.85	6.39×10^{-3}	–	–
Core barrel	47.35	4.75×10^{-3}	–	–
Water	48.07	3.36×10^{-3}	–	–
Water	48.78	2.61×10^{-3}	–	–
Thermal shield	49.50	2.51×10^{-3}	–	–
Thermal shield	50.10	2.29×10^{-3}	–	–
Thermal shield	51.20	1.63×10^{-3}	–	–
Thermal shield	51.90	1.27×10^{-3}	–	–
Thermal shield	52.50	8.69×10^{-4}	–	–
Vessel-wall capsule	53.25	5.42×10^{-4}	–	–
Water	53.87	4.24×10^{-4}	–	–
Water/SS clad interface	54.50	4.18×10^{-4}	–	–
0.1 in. in wall	54.61	3.98×10^{-4}	1.46×10^{19}	265
1 in. in wall	55.50	3.67×10^{-4}	1.34×10^{19}	255
2 in. in wall	56.50	2.96×10^{-4}	1.09×10^{19}	240
3 in. in wall	57.50	2.33×10^{-4}	0.85×10^{19}	210
4 in. in wall	58.50	1.80×10^{-4}	0.66×10^{19}	190
5 in. in wall	59.50	1.36×10^{-4}	0.50×10^{19}	170
6 in. in wall	60.50	9.90×10^{-5}	0.36×10^{19}	145
7 in. in wall	61.50	6.81×10^{-5}	0.25×10^{19}	125
8 in. in wall	62.50	3.47×10^{-5}	0.13×10^{19}	90

*The low values of fluxes are only peculiarities of the computer program output. They may be multiplied by some value such as $\sim10^{15}$ to conform to reality.
†From Fig. 13.

embrittled to a lesser degree and would have commensurately better ductility and fracture resistance.

SUMMARY AND CONCLUSIONS

Charpy V-notch specimens from the A302-B Yankee reactor surveillance steel irradiated in two surveillance locations of the Yankee reactor along with Charpy V-notch specimens of a reference A302-B steel have been tested and evaluated by NRL. The impact energy versus test temperature curves show that the Yankee surveillance steel, which was taken from one main reactor vessel plate and heat treated separately, is more sensitive to embrittlement by high energy neutron radiation than is an A302-B reference steel of the same nominal composition. Despite this fact, substantial evidence suggests that the fluence of high energy neutrons to the Yankee pressure-vessel wall after 30 fuel cycles of operation will not raise the transition temperature of the vessel steel to a level which would be of concern in terms of fracture-safe operations.

Embrittlement data for the Yankee surveillance steel reported herein have been referenced (Figs. 2–6) to neutron fluxes and fluences for neutrons >1 Mev assuming a fission spectrum at every irradiation location. Theoretical neutron-spectral calculations for the two surveillance locations in the Yankee reactor as well as for the interface between the alloy-steel pressure vessel and stainless-steel clad revealed significant deviations from a fission spectrum. By adjusting fluxes based on a fission spectrum at the surveillance locations to comply with calculated spectra for those locations, it has been possible to extrapolate these adjusted flux values from surveillance locations to the pressure-vessel wall and to project a maximum neutron fluence for the Yankee reactor [1.46×10^{19} n/cm^2 (>0.5 Mev) from Table 6].

The maximum neutron fluence calculated for the Yankee reactor in terms of n/cm^2 (>0.5 Mev) was derived from a calculated spectrum for the reactor. These data have been graphed along with all of the transition temperature increase data developed by NRL for elevated temperature irradiations of A302-B steel for which calculated spectrum fluences are available. A careful evaluation of the data (Fig. 13) revealed that the maximum transition temperature increase for the Yankee reactor after thirty cycles of operations at 600 Mw(t) will be ~265°F. Further investigations regarding the degradation of embrittlement and neutron fluence through the thickness of the Yankee reactor vessel wall also revealed that the neutron fluence level, based upon a calculated neutron spectrum, drops significantly (76 percent in 6 in.) across the thickness of the alloy-steel vessel wall. Accordingly, the embrittlement level also drops significantly across the thickness of the vessel wall.

Application of the "NDT plus 60°F" criterion to the calculated maximum transition temperature of the Yankee reactor vessel yields a value which should not be in excess of the lower limits of operating temperatures. If a somewhat higher initial NDT temperature due to fabrication is considered as well as a possibly higher initial NDT temperature for the midthickness region of the heavy-section steel plate (11,26), the same lower limit of operating temperature still will not be reached or approached.

In reviewing the Yankee reactor maximum fluence results, one final and highly interesting conclusion may be drawn concerning pressurized, light-water-moderated reactors. From Fig. 11, it will be noted that the spectrum at the Yankee pressure-vessel wall has fewer neutrons of energies >0.5 Mev than does a fission spectrum (Fig. 11). This is also true of a much smaller PWR, the Army SM-1A reactor (19), indicating that PWR spectra may well be quite similar in general shape regardless of the physical size of the plant.

In conclusion, the Yankee reactor represents one of the earliest PWR's to be built; thus it had little benefit during design stages in terms of experience concerning radiation damage to pressure vessels. Nonetheless, two independent evaluations of maximum projected vessel fluence and transition temperature increase show that no undue concern should arise in relation to fracture-safe operation of the plant over its intended lifetime. If in the future, however, the gradual rise in transition temperature of the vessel presents problems (for example, in pressurization during heat-up of the plant), relief may be possible, since it has been demonstrated that a substantial amount of the initial properties can be recovered by annealing of the vessel. The data (Figs. 4 and 10) indicate that better than 50 percent of the initial property values of the Yankee surveillance steel can be recovered using a heat treatment temperature felt to be attainable using nuclear heat alone. Furthermore, all the studies of annealing thus far have not shown any tendency for an increased rate of reembrittlement after annealing.

The value of in-reactor surveillance is well demonstrated by the results presented in this report. Both a surveillance steel closely representative of the main vessel plates and a reference steel were irradiated in the actual neutron flux environment of the reactor concerned. Thus, data denoting mechanical property changes cannot be disputed. A conservative behavior trend for radiation response, based on these data, has been drawn and referenced to the damage-causing fluence of neutrons. Equally important, the availability of neutron flux measurements from the Yankee surveillance program serves to increase the value and validity of all the data in that little, if any, doubt can exist concerning the future condition of the Yankee reactor pressure vessel.

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13. ABSTRACT

Charpy V-notch specimens, representative of one of the several heats of A302-B steel forming the Yankee reactor pressure vessel and irradiated as part of the Yankee surveillance program, have been tested by NRL. Specimens of this particular heat, irradiated in near-core (accelerated) as well as in vessel-wall locations, showed more embrittlement than did specimens of a reference steel heat of the same nominal A302-B composition irradiated simultaneously in the same surveillance capsules.

Those specimens from both the Yankee vessel heat and the reference heat irradiated at the vessel-wall location depicted a higher damage rate than that for the accelerated location. The cause of this difference in embrittlement response could not be attributed to an effect of cyclic, service irradiation temperatures, but could be traced to a qualitative relationship of thermal to fast (>1 Mev) neutron fluxes. This ratio was in excess of about 9:1 at the vessel-wall location versus a ratio less than about 9:1 for the accelerated location.

The computation of a maximum service fluence of 1.46×10^{19} n/cm^2 (>0.5 Mev) was made possible by establishment of the neutron spectrum at the reactor vessel wall using computer calculations. The maximum fluence derived by this technique

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Neutron spectrum						
Surveillance program						
Maximum service fluence						
Pressure vessel steel						

compared favorably with another value given by an independently-developed calculated neutron spectrum. The NRL computed service fluence in concert with the embrittlement data projects a maximum transition temperature increase of 265°F, which represents a level of embrittlement considered acceptable for the Yankee reactor vessel after thirty fuel cycles of operation at 600 thermal megawatts [Mw(t)].