DOCUMENT CONTROL DATA - R & D

(Security classification of title, body of abstract and indexing annotation must be entered when the overall report is classified)

1. ORIGINATING ACTIVITY *(Corporate author)* Naval Research Laboratory Washington, D.C. 20390	2a. REPORT SECURITY CLASSIFICATION Unclassified 2b. GROUP

3. REPORT TITLE

ANALYSIS OF NEUTRON-EMBRITTLEMENT AND FLUX-DENSITY CONSIDERATIONS OF THE ARMY SM-1 REACTOR PRESSURE VESSEL

4. DESCRIPTIVE NOTES *(Type of report and inclusive dates)*

Final report on one phase of the problem.

5. AUTHOR(S) *(First name, middle initial, last name)*

Charles Z. Serpan, Jr.

6. REPORT DATE	7a. TOTAL NO. OF PAGES	7b. NO. OF REFS
June 22, 1970	23	17

8a. CONTRACT OR GRANT NO. NRL Problem M01-14 b. PROJECT NO. USA-ERG-11-69 c. d.	9a. ORIGINATOR'S REPORT NUMBER(S) NRL Report 7101 9b. OTHER REPORT NO(S) *(Any other numbers that may be assigned this report)*

10. DISTRIBUTION STATEMENT

This document has been approved for public release and sale; its distribution is unlimited.

11. SUPPLEMENTARY NOTES	12. SPONSORING MILITARY ACTIVITY Department of the Army (Nuclear Power Field Office, U.S. Army Corps of Engineers), Fort Belvoir, Virginia 22060

13. ABSTRACT

The Army SM-1 reactor has been evaluated with respect to the increase in transition temperature of the A212-B steel pressure vessel. Although steel from the heat forming the vessel is not available for irradiation-response behavior testing, the initial transition temperature of 40°F (4°C) was determined from vessel steel. A relationship between increasing embrittlement for a 4-in.-thick plate of A212-B steel, representing the ASTM reference heat for this composition, and increasing neutron fluence was established for the irradiation temperature conditions of the SM-1 reactor. Combining with this the Army-imposed transition temperature limit for the SM-1 reactor vessel of 295°F (146°C) results in a fluence value of 2.65×10^{19} n/cm^2 >0.5 MeV for a lifetime vessel exposure. The neutron flux level for the vessel was established by extrapolating a core-region flux measurement using the results of a calculated neutron spectrum at the reactor vessel.

The fluence value of 2.65×10^{19} corresponds to a megawatt-year (MW-yr) operating period for the reactor of 49.4 MW-yr. The SM-1 has generated 33.86 MW-yr of power as of January 1970; the 20-calendar-year design life will be reached in 1977. A continuation of the same level of reactor operations for this time period should not result in the 49.4 MW-yr level being exceeded.

Evaluation was also made of the postirradiation annealing recovery that could be expected for A212-B steel. If irradiation of this steel is carried out at low temperatures, <240°F (116°C), then recovery of its initial properties can be appreciable for 600°F (316°C) postirradiation annealing temperatures. However, for irradiation at the SM-1 reactor operating

(over)

14. KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
Reactor Pressure vessel Steel Neutron embrittlement Neutron flux Neutron spectrum						

temperature of 440°F (227°C), postirradiation annealing at about 575°F (302°C) (a practical annealing temperature) would not result in significant annealing correction of embrittlement. An annealing temperature of 650°F (349°C) or more would be necessary to derive any significant benefit.

CONTENTS

ABSTRACT

The Army SM-1 reactor has been evaluated with respect to the increase in transition temperature of the A212-B steel pressure vessel. Although steel from the heat forming the vessel is not available for irradiation-response behavior testing, the initial transition temperature of 40°F (4°C) was determined from vessel steel. A relationship between increasing embrittlement for a 4-in.-thick plate of A212-B steel, representing the ASTM reference heat for this composition, and increasing neutron fluence was established for the irradiation temperature conditions of the SM-1 reactor. Combining with this the Army-imposed transition temperature limit for the SM-1 reactor vessel of 295°F (146°C) results in a fluence value of 2.65×10^{19} n/cm^2 >0.5 MeV for a lifetime vessel exposure. The neutron flux level for the vessel was established by extrapolating a core-region flux measurement using the results of a calculated neutron spectrum at the reactor vessel.

The fluence value of 2.65×10^{19} corresponds to a megawatt-year (MW-yr) operating period for the reactor of 49.4 MW-yr. The SM-1 has generated 33.86 MW-yr of power as of January 1970; the 20-calendar-year design life will be reached in 1977. A continuation of the same level of reactor operations for this time period should not result in the 49.4 MW-yr level being exceeded.

Evaluation was also made of the postirradiation annealing recovery that could be expected for A212-B steel. If irradiation of this steel is carried out at low temperatures, <240°F (116°C), then recovery of its initial properties can be appreciable for 600°F (316°C) postirradiation annealing temperatures. However, for irradiation at the SM-1 reactor operating temperature of 440°F (227°C), postirradiation annealing at about 575°F (302°C) (a practical annealing temperature) would not result in significant annealing correction of embrittlement. An annealing temperature of 650°F (349°C) or more would be necessary to derive any significant benefit.

PROBLEM STATUS

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

AUTHORIZATION

NRL Problem M01-14
USA-ERG-11-69

Manuscript submitted March 20, 1970.

PROLOGUE

The neutron flux and fluence analysis of this report is based upon the results of a relatively short exposure of several different flux detector materials in core Position 72 of the SM-1 reactor. The detectors, secured into an otherwise empty dummy fuel element, were loaded into the core on July 22, 1968 and irradiated under normal reactor operating conditions through November 10, 1968. The element was removed on December 5, 1968.

The history of operations for this training reactor shows a low percentage of operating time in which there is a demand for steam. However, the reactor is critical at about one-half to one percent of full power for virtually all the rest of the time; but, this time is not recorded since no steam demand exists. Flux detectors are, however, activated by these low-level reactor operations. During the residence of the Position 72 flux-detector assembly, the reactor operated a total of 575 hours at power when there was a steam demand; the average output during this period was 29.3 percent of full power. The reactor was critical at one-half to one percent of power almost all the rest of the time during the period from about mid-August until November 11. This results in a difference of ~1490 hours of operation that were not accounted for. It must be recognized that these hours of operation are lost for purposes of recording the reactor history, but they nevertheless apply to the *actual* condition of the reactor.

Based upon the flux data of the Position 72 flux-detector irradiation, an irradiation damage experiment was conducted in Position 72 to a specific fluence level. The reactor history for this period showed a rise to 100 percent power at midnight January 23, 1970 and continuous 100 percent power operation for 157.3 hours. The maximum flux from this irradiation was one-half of that from the original Position 72 flux-detector irradiation. During the exposure period, there was essentially no unrecorded time at one-half to one percent of power.

It can be concluded then that the peak flux of the SM-1 reactor is not 3.06×10^{13} n/cm^2 >0.5 MeV at 10 Mw as stated in the report; the actual flux intensity is about half that value, and the corresponding service limit for megawatt-years and embrittlement could be projected to higher levels. Nevertheless, the 3.06×10^{13} flux value does represent the flux intensity at the SM-1 vessel wall as a function of the steam power operations which are recorded by the megawatt-year operations integrator and serve as the guide to accumulated neutron exposure to the vessel. Thus, this flux value is realistic to use for reference to the actual operation of the reactor and, hence, for application to the embrittlement condition of the SM-1 reactor vessel steel.

ANALYSIS OF NEUTRON-EMBRITTLEMENT AND FLUX-DENSITY CONSIDERATIONS OF THE ARMY SM-1 REACTOR PRESSURE VESSEL

INTRODUCTION

The SM-1 reactor at Fort Belvoir, Virginia, is a pressurized, light-water moderated plant having a 10-MW (thermal) power output. The 2-1/2-in.-thick reactor vessel is fabricated of ASTM A212-B steel and is internally roll-bond clad with 0.25 in. of A-240 Grade S stainless steel. The reactor is depicted schematically in Fig. 1. The plant is used by the U.S. Army for operating crew training; as a result, the overall plant operations continue at a comparatively low level, since startups and shutdowns are very frequent. The plant was put into service in April 1957 and as of January 1970 had generated 33.86 MW-yr of power.

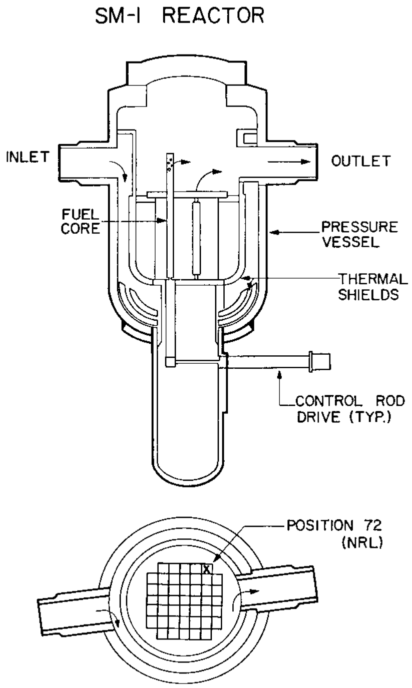

Fig. 1 - Schematic representation of the Army SM-1 reactor pressure vessel

The SM-1 was designed as a prototype compact reactor to verify systems for ease of transportation and installation of similar reactors at remote field sites. The SM-1A, located at Fort Greely, Alaska, was subsequently constructed using the SM-1 as a prototype. The compact nature of the reactor design included a high-power core to be relatively close to the pressure vessel wall. The result of this is a relatively high neutron flux level incident upon the vessel wall.

The steel of the SM-1, ASTM Type A212-B, displays the characteristic brittle-to-ductile transition temperature of ferritic, carbon-silicon steels. This phenomenon is characterized by scant energy absorption during brittle fracture of the steel at some low temperature, but a very rapid rise in energy absorption at increasing temperatures wherein high ductility is exhibited on steel fracture. The initial upsweep of the transition range through which this change occurs is considered most important in assessing steel ductility and is generally called the transition temperature. The effect of fuel core neutrons bombarding the steel in a reactor vessel is to drive this transition temperature upward. The transition temperature rises at a rate characteristic for the steel, the temperature, and the neutron flux, and if unchecked, could rise until it neared the operating temperature of the reactor.

This sequence of events has already occurred for the SM-1A reactor, since it operates at higher power levels for more extended time periods. Accordingly, annealing of the SM-1A reactor vessel was conducted and has been described (1, 2). From the SM-1A analyses it was noted that specimens of a plate of A212-B steel exhibited only about 17% recovery of preirradiation properties for the irradiation and annealing conditions for the SM-1A reactor.

The Army, of course, is interested in maintaining the SM-1 reactor in an acceptable operating condition. Therefore, at their request, a survey is presented of pertinent irradiation and annealing data for A212-B steel thought to be reasonably representative of the SM-1 reactor vessel. From this effort, a conservative but characteristic trend behavior is discerned for irradiation at 440°F (227°C), and patterns of annealing response are developed. An analysis of the neutron spectrum at the SM-1 pressure vessel wall is coupled with neutron flux measurements made in an empty fuel element position at the core boundary to project the flux at the vessel wall and the vessel fluence for future operating periods. These data together permit a projection of the neutron fluence value and the corresponding megawatt-year exposure for the maximum permissible transition temperature of the SM-1 reactor vessel.

IRRADIATION EFFECTS DATA

The steel being evaluated for this study is from a 4-in.-thick plate of ASTM A212-B (3) that was stockpiled by the U.S. Atomic Energy Commission (AEC) as a reference steel. The heat treatment and chemical composition of this plate as well as those of record from the SM-1 reactor vessel manufacturer (4) are included in Table 1. The data reported herein were all developed at the Naval Research Laboratory (NRL) and constitute a coherent data bank referenced to similar irradiation environments, irradiation temperature control characteristics, neutron dosimetry measurement techniques, and neutron spectrum analysis parameters. The study and analysis are confined to these data in the hope that uncertainties and adjustments for other laboratory investigations can be avoided, thereby improving the accuracy of the analysis.

Irradiation Embrittlement of A212-B Steel

The compact nature of the SM-1 reactor and the fact that it was constructed at the time when radiation damage to steels was just being recognized as a potential problem resulted in the exclusion of any direct means of vessel radiation damage surveillance. As a result, essentially all of the embrittlement data for A212-B steel has come from accelerated irradiation-rate experiments in the Low Intensity Test Reactor (LITR) at the Oak Ridge National Laboratory. Additional irradiation data were obtained from an above-core location in the SM-1A reactor at irradiation temperatures very comparable to the SM-1. An example of the irradiation embrittlement increase for A212-B from the SM-1A is shown in Fig. 2 (5).

Table 1
Chemical Composition and Heat Treatment* for ASTM A212-B SM-1 Pressure Vessel Steel and 4-in.-Thick Reference Steel

Type of A212-B Steel	Thickness (in.)	Chemical Composition (wt-%)								
		C	Mn	Si	P	S	N	Cr	Mo	Cu
Heat 21866-12	2-1/2	0.26	0.85	0.20	0.016	0.024	--	—	—	—
Heat 24045-15	2-1/2	0.25	0.78	0.17	0.018	0.029	—	—	—	—
Reference	4	0.26	0.76	0.24	0.011	0.031	0.22	0.20	0.02	0.26

*Heat Treatment:

SM-1 Plates: Clad plate austenitized at 1950°F (1066°C), air-cooled; heated to 1750°F (954°C) for roll forming, temperature not below 1530°F (832°C); postweld stress relief at 1150°F (621°C).

Reference: Austenitized at 1650°F (899°C) for 2 hours; water quenched: tempered at 1175°F (635°C) for 4 hr; furnace cooled to below 600°F (316°C).

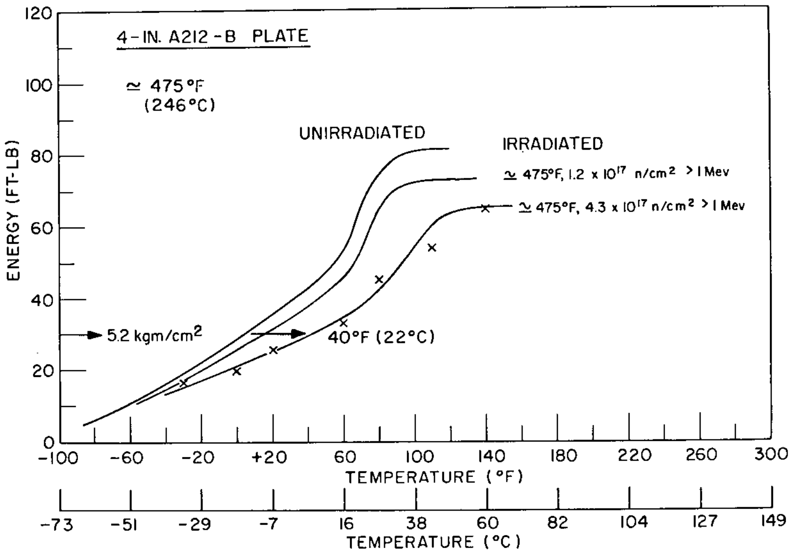

Fig. 2 - Notch-ductility characteristics of 4-in., A212-B plate steel representative of the SM-1 pressure vessel. The curve for the 4.3×10^{17} n/cm^2 exposure was from the bottom section of an SM-1A reactor surveillance capsule; the lower exposure curve was derived from specimens of the top capsule section.

The NRL results of irradiating specimens from the reference heat of A212-B steel are summarized in Table 2 (2,5-10). The transition temperature increase (ΔTT) values shown are based on translation along the 30-ft-lb level from the unirradiated to the irradiated condition. Neutron fluence values are given in terms of an assumed fission-spectrum distribution of neutrons for n/cm^2 >1 MeV, based upon a fission-spectrum averaged cross section of 68 mb for the ^{54}Fe(n,p)^{54}Mn reaction. Included also are the

Table 2
Summary of NRL Irradiations of 4-in. ASTM A212-B Steel Plate

LITR Core Position	Experiment No.	Irradiation Temperature		C_v 30 ft-lb Transition Temperature Increase		Fluence Φ^{fs} n/cm^2 >1 MeV*	$\bar{\sigma}^{cs}$ >0.5 MeV mb†	Fluence Φ^{cs} n/cm^2 >0.5 MeV†
		°F	°C	°F	°C			
55	31	490-350	254-177	230	128	1.4×10^{19}	57.3	2.4×10^{19}
18	8	400	204	180	100	1.0	65.2	1.5
18	8	450	232	200	111	1.0	65.2	1.5
18	12	510	266	210	117	2.2	65.2	3.32
18	11	430	221	305	169	2.6	65.2	3.92
SM-1A	above core	~475	~246	~10	~6	0.012	143.5	0.008
SM-1A	above core	~475	~246	40	22	0.043	143.5	0.029
SM-1A	above core	~475	~246	115	64	0.24	143.5	0.16
18	8	550	288	100	56	1.0	65.2	1.5
55	44	550	288	190	106	3.4	57.3	5.8
18	45	550	288	215	119	3.3	65.2	5.0
55	85	550	288	220	122	4.8	57.3	8.2

*Fission spectrum averaged cross section = 68 mb, $^{54}Fe(n,p)^{54}Mn$.
†Fission spectrum averaged cross section = 82.6 mb, $^{54}Fe(n,p)^{54}Mn$.

^{54}Fe cross sections averaged over the calculated spectra of the individual irradiation locations for n/cm^2 >0.5 MeV. The calculated-spectrum >0.5 MeV fluences were additionally evaluated on the basis of an 82.6-mb fission-spectrum averaged cross section for iron. This has been discussed at length elsewhere (2).

The irradiation data of Table 2 are shown graphically in Fig. 3 for fission spectrum >1 MeV fluences and in Fig. 4 for calculated spectrum >0.5 MeV fluences. Symbols representing irradiation temperatures are the same in both figures. Because it is not known exactly where the irradiation data characteristic of the SM-1 vessel steel would fall within this plot, the trend line for the SM-1 reactor vessel has been conservatively placed along the left side of the data. This appears to be validated, since this line passes

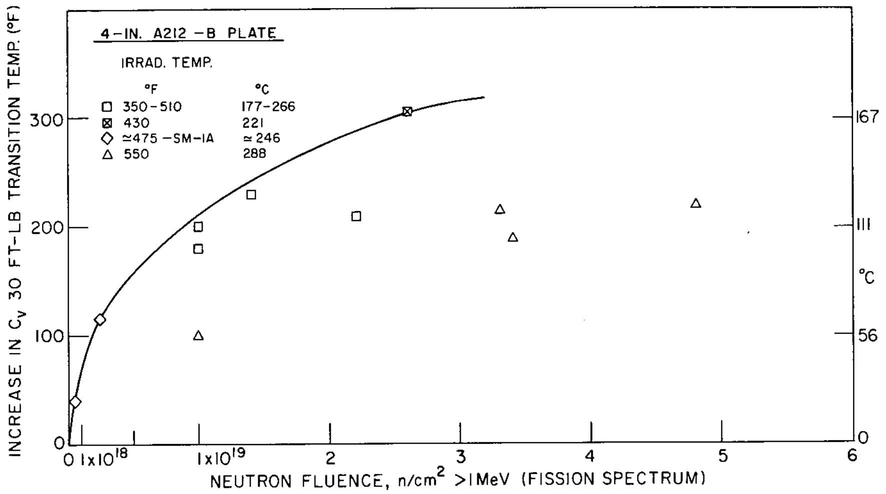

Fig. 3 - Charpy-V transition temperature behavior of A212-B steel plate irradiated at controlled temperatures of 350°F (177°C) and greater, for neutron fluences of n/cm² >1 MeV (fission spectrum)

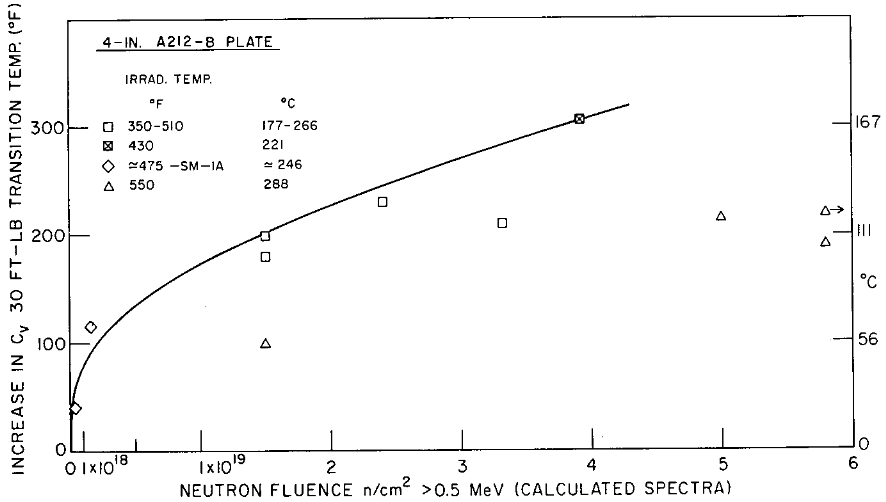

Fig. 4 - Charpy-V transition temperature behavior of A212-B steel plate irradiated at controlled temperatures of 350°F (177°C) and greater, for neutron fluences of n/cm² >0.5 MeV (calculated spectra)

through the single data point at 430°F (221°C) and the two points from irradiations in the SM-1A above-core location (conducted at representative temperatures). These curves can now be used as references for evaluating the vessel steel embrittlement at service-level fluences and for evaluating annealing response.

Irradiation-Postirradiation Annealing of A212-B Steel

Most of the early NRL irradiation-annealing studies were carried out using a test-reactor ambient temperature of ~130°F (54°C) for irradiation rather than the power-reactor operating temperature of 440°F (227°C) because early data indicated that irradiation behavior from the entire range <450°F (232°C) was essentially indistinguishable for a given material. Although this proved to be true for all practical purposes in terms of irradiation response, the SM-1A irradiation-annealing studies clearly revealed that low-temperature irradiation-damage input would be annealed out much more effectively than irradiation-damage input at temperatures near 450°F (232°C). Accordingly, the irradiation-annealing response of reference A212-B steel from the SM-1A irradiation location (2), shown in Fig. 5, is most illuminating. The SM-1A reactor annealing conditions were quite ineffective in relieving the induced embrittlement from the A212-B steel; however, significant embrittlement relief was accomplished for higher temperature post-irradiation heat treatments (700°F, 371°C) for 48 hours. These SM-1A annealing data for A212-B steel form a very useful reference for evaluating the body of elevated temperature irradiation and annealing data compiled for A212-B steel, summarized in Table 3.

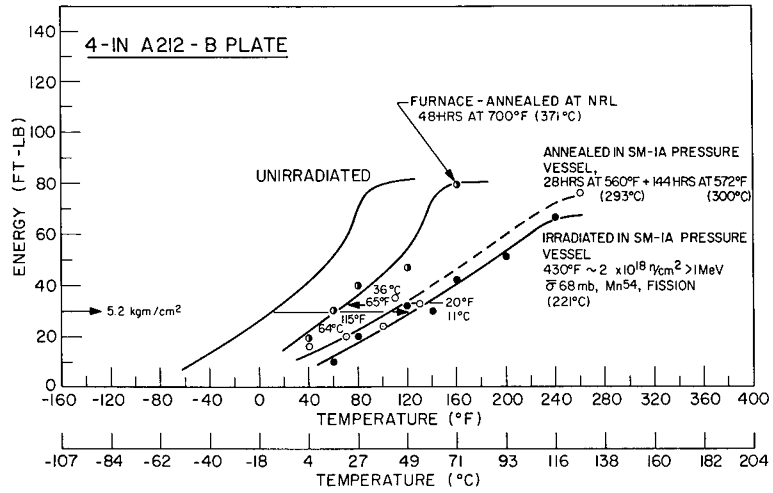

Fig. 5 - Charpy-V transition temperature behavior of A212-B steel specimens from the center layer of an SM-1A surveillance capsule before and after in-reactor (SM-1A) annealing and following a 700°F (371°C), 48-hr furnace anneal

The first four experiment data sets were for irradiations at <240°F (116°C) followed by annealing out of the reactor for the times and temperatures indicated. The total amount of recovery of initial transition temperature is listed and, for convenience, the amount not recovered is also listed, as the residual.

For the next set of data, capsule 21A was irradiated, along with 21B first and then along with 21C. In this way 21B and 21C established the mid- and endpoints for the multiple irradiation annealing cycles. Capsule 21A was irradiated at < 240°F (116°C), then annealed as a whole at 600°F (316°C) for 18 hours, then reirradiated at <240°F (116°C) for the fluence indicated. It was then opened, and specimens were removed and annealed for the different times and temperatures as indicated. The entire experiment, 21A, B, and C, has been summarized schematically in Fig. 6. These data are plotted on the basis of calculated spectrum, n/cm^2 > 0.5 MeV. The basic embrittlement curve of Fig. 4 has been used for reference.

Table 3
Summary of Irradiation and Annealing Experiments by NRL on 4-in. A212-B Steel Plate
C_v 30 ft-lb Transition Temperature Data

LITR Core Position	Experiment No.	1st Cycle Irradiation and Anneal											2nd Cycle Irradiation and Anneal							
		Irradiation Temperature		Neutron Fluence n/cm^2 × 10^{19}		Transition Temperature Increase		Annealing Conditions		Recovery/Residual		Percent Recovery	Neutron Fluence n/cm^2 × 10^{19}		Transition Temperature Increase		Annealing Conditions		Final Residual	
		°F	°C	> 1 MeV*	> 0.5 MeV†	°F	°C	°F hr	°C hr	°F	°C		> 1 MeV*	> 0.5 MeV†	°F	°C	°F-hr	°C-hr	°F	°C
55	15C	240	116	0.75	1.29	180	100	700-168	371-168	150/30	83/17	83.5								
49	17	240	116	0.7	1.48	200	111	600-18	316-18	125/75	69/42	62.5								
						200	111	600-168	316-168	140/60	78/33	70.0								
49	21B	240	116	0.67	1.42	210	117	600-12	316-12	150/60	83/33	71.5								
						210	117	600-18	316-18	150/60	83/33	71.5								
						210	117	600-24	316-24	150/60	83/33	71.5								
						210	117	600-30	316-30	150/60	83/33	71.5								
						210	117	600-72	316-72	150/60	83/33	71.5								
49	21C	240	116	0.65	1.40	205	114	600-3	316-3	150/55	83/31	73.2								
						205	114	600-9	316-9	150/55	83/31	73.2								
						205	114	600-18	316-18	150/55	83/31	73.2								
						205	114	600-96	316-96	150/55	83/31	73.2								
						205	114	750-9	399-9	185/20	103/11	90.2								
						205	114	750-18	399-18	185/20	103/11	90.2								
49	21A	240	116	0.67	1.42	~210	~117	600-18	316-18	—	—	—	1.30	2.84	225	125	600-9	316-9	105	58
															225	125	600-18	316-18	105	58
															225	125	600-96	316-96	95	53
															225	125	750-9	399-9	45	25
															225	125	750-18	399-18	45	25
															225	125	750-96	399-96	45	25
															225	125	900-18	482-18	25	14
SM-1A	above core	430	221	0.24	0.16	115	64	560-28	293-28	20/95	11/53	17.4								
								572-144	300-144											
						115	64	700-48	371-48	65/50	36/28	56.6								
18	45	550	288	—	—	—	—	650-48	343-48	—	—	—	3.3-3.5	5.0-5.3	—	—	650-48	343-48	175‡	97‡
						—	—	750-48	399-48	—	—	—			—	—	750-48	399-48	145‡	81‡
						—	—	800-48	427-48	—	—	—			—	—	—	—	125‡	69‡

*Fission spectrum averaged cross section = 68 mb, $^{54}Fe(n, p)^{54}Mn$.
†Fission spectrum averaged cross section = 82.6 mb, $^{54}Fe(n, p)^{54}Mn$.
‡Final values following third irradiation period.

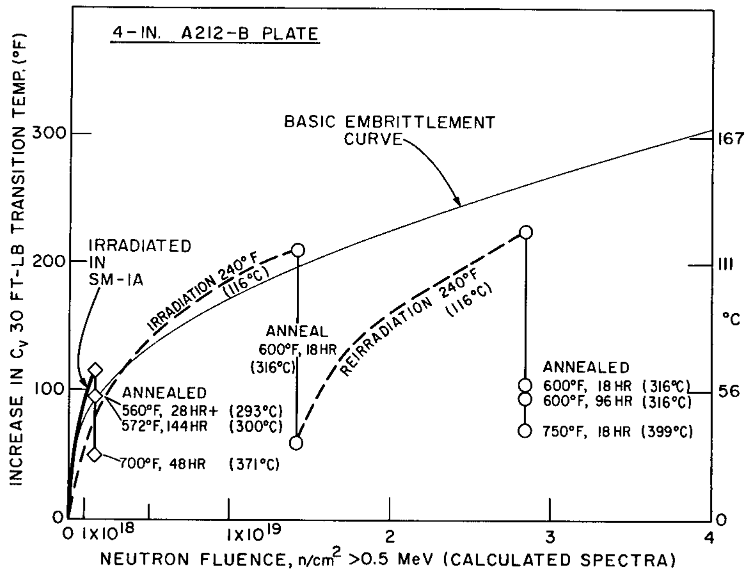

Fig. 6 - Characteristics of A212-B steel following irradiation, postirradiation annealing under various conditions, and reirradiation as referenced to the basic embrittlement curve for SM-1 reactor vessel steel irradiated at 430°F (221°C)

The data from the SM-1A annealing operation are listed next in Table 3 and are also included in Fig. 6. The significant benefit of the higher annealing temperature is apparent.

The final set of data in Table 3 are from irradiation at 550°F (288°C) followed by annealing at intermediate temperatures of 650°F (343°C), 750°F (399°C), and 800°F (427°C). It should be recognized that the percent recovery for specimens irradiated in this experiment reflects the fact that all specimens were subjected to irradiation between the final annealing treatment and testing operations. All other specimens for which data are summarized in Table 3 were annealed *after* irradiation; thus no damage input could be effected prior to final specimen testing. The final recoveries shown for the 550°F (288°C) exposure experiment reinforce the suggestion that recovery of preirradiation properties is retarded by progressively higher irradiation temperatures.

NEUTRON-DOSIMETRY ANALYSIS OF THE SM-1 REACTOR VESSEL

This phase of the SM-1 reactor analysis is concerned with accurate definition of the neutrons responsible for causing the radiation embrittlement of the pressure vessel. The two principal facets are: (a) a realistic description of the neutron energy level distribution at the SM-1 pressure vessel wall, and (b) an accurate value of the flux at the pressure vessel wall for conversion into a fluence per megawatt-year. The two are interdependent but will be discussed separately for clarity.

Neutron Spectrum

In the absence of any other information, it is convenient to assume that the distribution of neutrons by energy levels within the flux of a reactor will conform to the shape described by the Watt form of the fission spectrum. The Watt fission spectrum is shown in Fig. 7 as the smooth curve. Presented also on the same figure as histograms are the calculated neutron energy level distributions for the center of a fuel element located in position 72 at the core boundary and for the interface of the vessel wall and stainless steel cladding. It should be quite clear that the fission-spectrum neutron distribution does not adequately describe the neutron distributions in either of the other two locations.

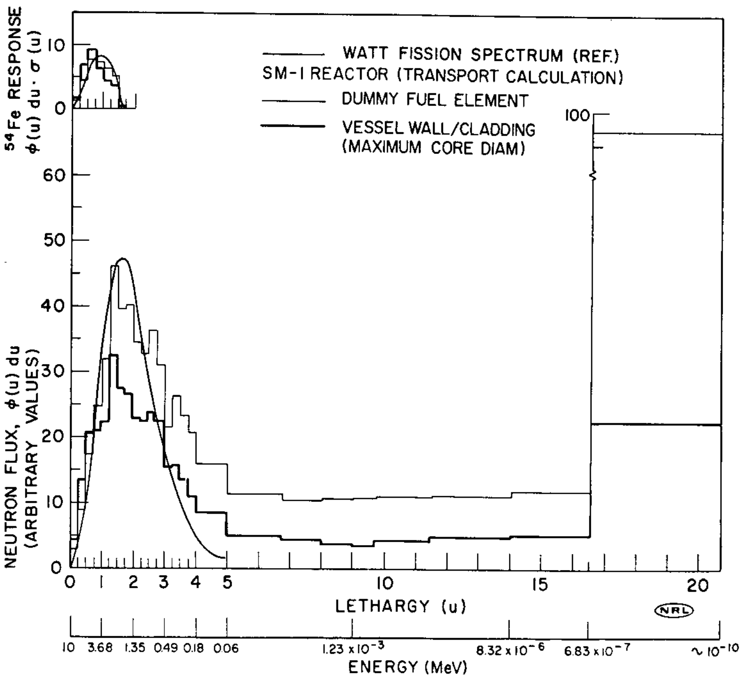

Fig. 7 - Graphical representation of the neutron spectra in the SM-1 reactor dummy fuel element Position 72 (light histogram) and the interface of the vessel wall and stainless steel cladding, assuming a maximum core diameter (heavy histogram). The smooth curve is the Watt fission spectrum.

Spectra are also presented for the LITR core facility C-18 (typical of LITR position as referenced in both Tables 2 and 3) in Fig. 8 and for the SM-1A above-core location in Fig. 9. In both of these figures, the smaller curves that appear in the top portion show the equivalence of the activation of an iron flux-detector wire in the spectrum of interest (the histogram) and the Watt fission spectrum as it was assumed to be present in that irradiation location. The area under these curves is equal because each accounts for the total neutron activation in the iron monitor. (Note that *no* activation occurs from neutrons of energy below ~1.5 MeV). Having this relationship, then, it follows (2) that the curves in the main portion of the figures present, graphically, the flux actually present in a reactor location compared to that which could be predicted by the fission spectrum if the fission spectrum had actually existed at that location. The effect of the spectrum differences is revealed by inspection of the fluence values in Tables 2 and 3. For example, the calculated spectrum fluences for $n/cm^2 > 0.5$ MeV in LITR irradiations are considerably greater than the fission-spectrum > 1 MeV fluences. It is apparent that far more neutrons of energies > 0.5 MeV are present in the calculated spectrum (histogram) for

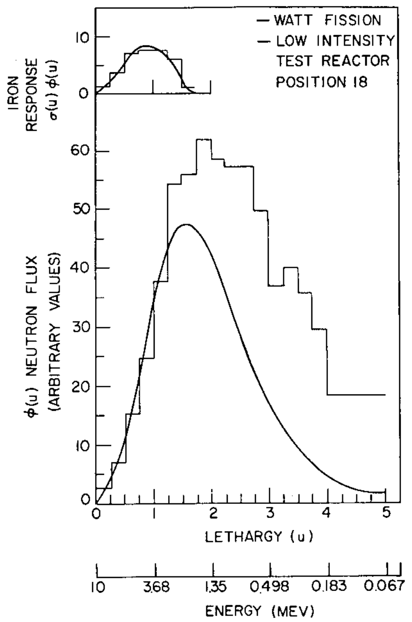

Fig. 8 - The high-energy portion of the neutron spectrum of core lattice Position 18 of the Low Intensity Test Reactor compared to the Watt fission spectrum. The spectra are equated to the same activation of an iron flux detector.

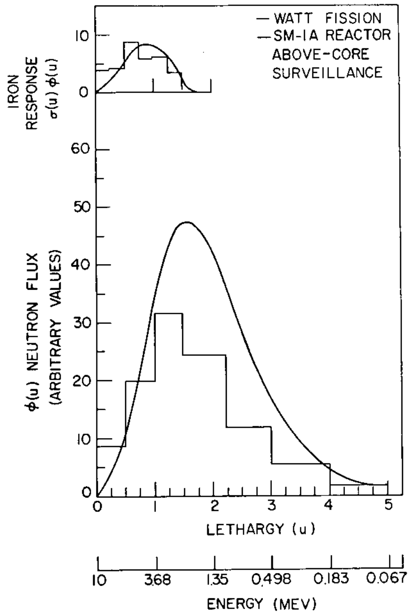

Fig. 9 - The high-energy portion of the neutron spectrum of the above-core location of the Army SM-1A reactor compared to the Watt fission spectrum. The spectra are equated to the same activation of an iron flux detector.

these locations (Fig. 8) than in the fission spectrum. Conversely, note that the vessel wall spectrum of Fig. 7 has fewer neutrons of energies 0.5 MeV than the fission spectrum. Tabulated fluxes per discrete energy grouping of the SM-1 fuel element location and vessel wall -- stainless steel clad interface are shown in Table 4; tabulated spectra for the LITR locations and the SM-1A above-core location are already published (11).

It is particularly important to obtain good accuracy in the calculation of the neutron spectrum for the vessel wall of the SM-1, since no measurement can be made there and only an extrapolation can establish the flux intensity level. Accordingly, the calculation was made by two independent methods and reactor physics codes both of which, however,

Table 4
Calculated Neutron Flux in Arbitrary Units Per Energy Group for Locations in the SM-1 Reactor. Transport Theory Code DTF-IV.

Energy Groups		Relative Flux ($\phi u\, du$)		
Lethargy (u)	Lower Energy Limit (MeV)	Dummy Fuel Element Position 72	Pressure Vessel Wall/SS Cladding	
			Maximum Core Diameter	Average Core Diameter
0.25	7.79	0.9248×10^{11}	1.006×10^{9}	0.7516×10^{9}
0.50	6.07	2.821	3.195	2.386
0.75	4.72	5.505	4.825	3.465
1.00	3.68	7.799	4.864	3.351
1.25	2.87	10.11	5.180	3.458
1.50	2.23	14.55	7.562	4.907
1.75	1.74	12.49	6.432	4.131
2.00	1.35	12.71	6.223	3.966
2.25	1.05	10.86	5.332	3.394
2.50	8.21×10^{-1}	10.33	5.253	3.342
2.75	6.39	11.53	5.641	3.564
3.00	4.98	9.800	5.303	3.345
3.25	3.88	6.791	3.585	2.264
3.50	3.02	8.338	3.681	2.314
3.75	2.35	7.397	3.158	1.981
4.00	1.83	6.526	2.583	1.617
5.00	6.74×10^{-2}	20.34	8.065	5.045
6.75	1.17	25.30	8.703	5.432
8.00	3.36×10^{-3}	16.80	5.411	3.369
9.00	1.23	13.56	3.836	2.351
9.75	5.83×10^{-4}	10.26	2.753	1.708
11.50	1.01	24.53	7.580	4.684
14.00	8.32×10^{-6}	35.71	11.96	7.354
16.50	6.83×10^{-7}	38.34	12.55	7.675
20.72	1.0×10^{-10}	521.7	89.56	62.64
ϕ Total		845.0218×10^{11}	224.2410×10^{9}	148.4947×10^{9}
ϕ >1 MeV		79.8358	45.6700	30.4781
ϕ >0.5 MeV		109.4298	60.8160	40.0607
^{54}Fe $\bar{\sigma}^{cs}$ >1 MeV		129.6	165.9	175.3
^{54}Fe $\bar{\sigma}^{cs}$ >0.5 MeV		94.6	124.6	133.4

used exactly the same input information. The codes were DTF-IV (12) using the transport theory, S_n, method and 2DB (13) using the diffusion theory method. The calculation requirements were to obtain the energy distribution for the center of a water-filled core position at the core boundary of the SM-1, designated in Fig. 1 as Position 72, and then to degrade this flux to the interface of the pressure vessel wall -- stainless steel cladding (PVW/SS Clad) assuming in one case the maximum core diameter and, in a second case, the average core diameter. The results of this calculation from the transport theory DTF-IV code are summarized on p. 12 and are plotted Fig. 7. The spectral shape as well as the intensity values for the core boundary element location (Position 72) and for the PVW/SS Clad location as derived from the diffusion code 2DB calculation were so similar to those from the DTF-IV calculation that their inclusion in this report would simply be redundant. An inspection of the flux intensity values from the maximum and average core diameters clearly reveals a significant difference; the higher flux intensity value must be taken for this report since it represents the limit condition.

Neutron Dosimetry

The neutron flux has not been measured at the vessel wall of the SM-1 because there is no practical way to do it. An accurate value for this location had to be obtained, however, if meaningful projections of the vessel embrittlement were to be made. Accordingly, a series of neutron flux monitor wires was exposed for a two-month period in the core boundary fuel element location described above. As noted above, the calculated spectrum analysis was performed by degrading the core center flux through this location and on to the vessel wall. Thus, the total flux values shown in Table 4 depict the intensity decrease from the element location to the vessel wall. Once a measured value of flux for the element location is obtained, it is a simple matter to extrapolate a value for the vessel wall.

Flux values measured from the element location exposure are shown in Fig. 10 along with the flux values extrapolated to the interface between A212-B steel and the stainless steel cladding, assuming the maximum core diameter condition. Briefly, the extrapolation is performed as follows:

The peak flux value 3.06×10^{13} n/cm$^2 \cdot$ sec >0.5 MeV is reduced by the fraction* of

$$\frac{\text{Calculated flux at vessel wall}}{\text{Calculated flux at fuel element}} = \frac{6.082 \times 10^{10}\ \text{n/cm}^2 > 0.5\ \text{MeV}}{10.94 \times 10^{12}\ \text{n/cm}^2 > 0.5\ \text{MeV}} = 0.00555$$

$$0.00555 \times 3.06 \times 10^{13}\ \text{n/cm}^2 > 0.5\ \text{MeV} = 1.7 \times 10^{11}\ \text{n/cm}^2 > 0.5\ \text{MeV}.$$

The peak vessel flux and the fluence per megawatt-year (one calendar year of operation at one megawatt power) for various neutron -- spectrum criteria are listed below.

	n/cm^2 $\cdot$ sec^{-1}
>1 MeV Fission spectrum $\bar{\sigma}^{fs} = 68$ mb, ^{54}Fe(n,p)	2.15×10^{10}
>1 MeV PIMG-2 (Ref. 4) Calculation Normalized to measurements	3.74×10^{9}

*See Table 4.

>1 MeV Calculated spectrum $\bar{\sigma}^{fs}$ = 82.6 mb, $^{54}Fe(n,p)$	1.27×10^{10}
>0.5 MeV Calculated spectrum $\bar{\sigma}^{fs}$ = 82.6 mb, $^{54}Fe(n,p)$ Measurement extrapolated by DTF-IV and 2DB calculations	1.7×10^{10}

PVW/SS Clad Flux = 1.7×10^{10} n/cm$^2 \cdot$ sec^{-1} >0.5 MeV

1 MW-yr	= 5.36×10^{17} n/cm^2 >0.5 MeV
10	= 5.36×10^{18}
20	= 1.07×10^{19}
30	= 1.61×10^{19}
40	= 2.14×10^{19}
49.4	= 2.65×10^{19}
50	= 2.68×10^{19}
60	= 3.22×10^{19}
70	= 3.75×10^{19}

The very large difference between megawatt-year values as derived by the P1MG-2 diffusion code (4) vs those derived in this report is not understood. It should be noted, however, that different libraries of cross section values were used for the P1MG-2 calculations and those of this report. Furthermore, significant refinements in all the cross section values have been made in the years between the P1MG and the present calculations. Finally, it is noted that the analysis of this report is based on a measurement made from an exposure in the operating reactor under nearly full-power operating conditions. The earlier megawatt-year value was based primarily on the P1MG-2 calculation normalized to measurements made on a mockup of the reactor at very low power levels.

DISCUSSION

The projected embrittlement of the SM-1 pressure vessel can now be referenced to the corresponding increase in megawatt-years of vessel operations. The direct relationship between neutron fluence and megawatt-years is plotted in Fig. 11 using the basic embrittlement curve for n/cm^2 >0.5 MeV from Fig. 4. Note, however, the addition of a second scale in Fig. 11 that gives the actual vessel transition temperature in comparison with the increase in transition temperature. The maximum permissible vessel transition temperature has been set for the SM-1 by the Army at 295°F (146°C), but because of the initial vessel steel transition temperature of +40°F (4°C), the permissible transition temperature *increase* is only 255°F (142°C). This is indicated by the "LIMIT" line drawn across the figure at the transition temperature increase of 255°F (142°C). This limit line crosses the basic embrittlement curve at a fluence of 2.65×10^{19} n/cm^2 >0.5 MeV; this fluence corresponds to an operational period of 49.4 MW-yrs.

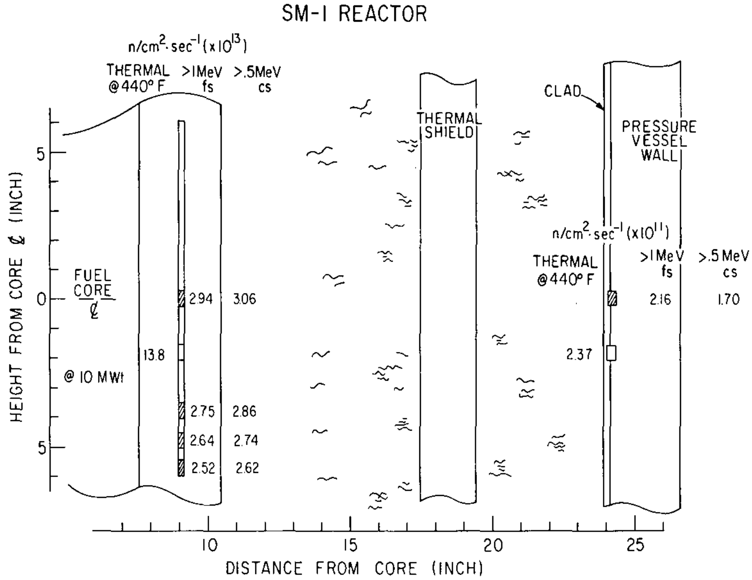

Fig. 10 - Schematic representation of the measured flux values at the SM-1 reactor dummy fuel element Position 72 and the corresponding flux values at the vessel wall and stainless steel cladding interface extrapolated by spectrum calculations of 10-MW operations

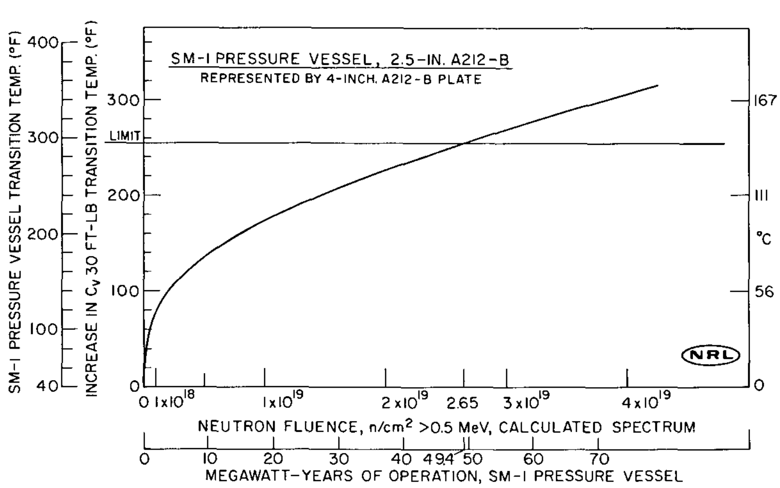

Fig. 11 - Charpy-V transition temperature behavior characteristics of A212-B steel plate to represent the vessel wall status for the SM-1 reactor. Neutron fluence values in terms of n/cm^2 >0.5 MeV are referenced to corresponding megawatt-year exposure values, indicating a limit condition of 295°F (146°C) vessel transition temperature for a fluence of 2.65×10^{19} n/cm^2 >0.5 MeV at 49.4 Mw-yr.

The design life for the SM-1 is 20 calendar years; the reactor went into service in 1957, so that the nominal lifetime will be reached in 1977. The present operational schedule for the reactor calls for the power output of only 1 or 2 MW-yr per calendar year at the maximum. Because the megawatt-year output for the reactor as of January 1, 1970 was only 33.86 MW-yr, it appears that the reactor can easily continue this level of operation for its remaining design years and still be within the vessel transition-temperature limitations imposed.

The review of annealing characteristics for A212-B steel generally confirms the small recovery exhibited in connection with the SM-1A reactor annealing operation. Recovery with 572°F (300°C) annealing after 430°F (221°C) irradiation is far lower than recovery with 600°F (316°C) annealing following lower temperature irradiation. This reaffirms previous observations. If the SM-1 reactor vessel can attain a temperature of only about 575°F (302°C) using nuclear heat as in the SM-1A, any effort to anneal the SM-1 vessel under these conditions would probably not be worthwhile because of the small recovery to be expected. In fact, it would probably be necessary to achieve a temperature of at least 650°F (343°C) if significant recovery were to be achieved. This would probably require installation of auxiliary heaters, and hence, it would be practical to plan on annealing at even higher temperatures to achieve even greater recovery. These annealing considerations should not, however, be a cause for significant activity at this time because the need for annealing of the SM-1 vessel cannot be reasonably foreseen based upon the present megawatt-year power output and the current operational schedule for the reactor for the remainder of its design lifetime.

It is again noted that the SM-1 reactor-vessel embrittlement-increase analysis of this report is based upon irradiation behavior of a 4-in.-thick plate of reference A212-B steel, not upon the 2.5-in.-thick steel plate of the SM-1 vessel. The initial transition temperature of steel used in the SM-1 is known, however, and has been used to establish this critical parameter at +40°F (4°C). This value is higher than the initial transition temperature for the 4-in.-thick plate and coupled with the safety margins placed by the Army on the vessel transition temperature increase, the 255°F (142°C) vessel steel transition temperature increase appears to be conservative.

An additional factor of importance to the SM-1 reactor relates to the maximum degradation of Charpy-V shelf energy and the concomitant increase in yield strength for the A212-B vessel steel as projected for the end of reactor life. Trends in this simultaneous effect have been developed (14). One means for rapid evaluation of these factors for SM-1 reactor steel is the Ratio Analysis Diagram (RAD) (15). Required inputs are the Charpy-V shelf energy level and yield strength corresponding to projected maximum lifetime steel conditions. For irradiation of A212-B steel at representative SM-1 operating temperatures to a fluence in excess of that projected for the SM-1, the Charpy-V shelf level for longitudinally oriented specimens is about 50 ft-lb (16) and the yield strength is about 100,000 psi (17). These two values, when analyzed by the RAD, indicate a shelf level toughness greater than that described by the Diagram K_{Ic}/σ_{YS} infinity ratio line. Assuming this toughness to be indicative of the entire reactor vessel including vessel welds, the metallurgical condition for plane strain fracture would not develop under normal operating conditions of the SM-1 reactor.

SUMMARY AND CONCLUSIONS

The increase in transition temperature for the A212-B steel of the Army SM-1 reactor vessel has been estimated to rise at a rate consistent with that for other carbon-silicon steels used for reactor pressure-vessel construction. Steel from the reactor vessel is not available for irradiation behavior evaluation, but the initial vessel transition temperature was established as +40°F (4°C). A trend curve was constructed giving the relationship between increasing embrittlement and neutron fluence. Based on the maximum permissible transition temperature increase for the SM-1 reactor of 255°F (142°C) as set by the Army, in conjunction with the curve, a neutron fluence of 2.65×10^{19} n/cm^2 >0.5 MeV is indicated to be required to reach this level of embrittlement.

The relationship between neutron flux at the vessel wall and megawatt-years of plant operation was established. The peak flux at the vessel is 1.7×10^{11} n/cm^2 · sec^{-1} >0.5 MeV, for 10-MW operations; this corresponds to a fluence per megawatt-year of 5.36×10^{17} n/cm^2 >0.5 MeV. The parameters of interest for the SM-1 as derived in this report are therefore a plant power output level of 49.4 MW-yr corresponding to 2.65×10^{19} n/cm^2 > 0.5 MeV for a transition temperature increase of 255°F (142°C) and a final vessel transition temperature of 295°F (146°C).

The annealing response of A212-B steel revealed from the data in this report show that small recovery of initial characteristics will occur for annealing near 575°F (302°C) following irradiation at 430°F (221°C). If significant annealing of A212-B steel is required following 440°F (227°C) irradiation (as in the SM-1), the annealing temperature would have to be at least 650°F (343°C). As this would probably require auxiliary heaters, it is reasonable to propose that any necessary annealing of the SM-1 vessel be accomplished at 700°C (371°C) or higher.

ACKNOWLEDGMENTS

The efforts of all persons in the Reactor Materials Branch who have contributed to producing the research data contained in this report are acknowledge with appreciation. The neutron dosimetry values assigned to specific irradiations were determined at the Materials Test Reactor Counting Room, National Reactor Testing Station, Idaho. This study and the supporting research have been sponsored at the Naval Research Laboratory by the Nuclear Power Field Office, U.S. Army Corps of Engineers, Ft. Belvoir, Virginia.

REFERENCES

1. Potapovs, U.,Knighton, G.W., Denton, A.S., "Critique of In-Place Annealing of SM-1A Nuclear Reactor Vessel," Nucl. Eng. Design 8:39-57 (July 1968)

2. Potapovs, U., Hawthorne, J.R., and Serpan, C.Z., Jr., "Notch Ductility Properties of SM-1A Reactor Pressure Vessel Following the In-Place Annealing Operation," NRL Report 6721, May 21, 1968; Nucl. Applications 5 (No. 6):389-409 (Dec. 1968)

3. Alger, J.V., and Porter, L.F., "Evaluation of Reference Pressure Vessel Steels for Neutron Irradiation Studies," U.S. Steel Applied Research Laboratory Technical Report, June 18, 1964

4. McLaughlin, D.W., and others, "Effect of Radiation Damage on SM-1, SM-1A, and PM-2A Reactor Vessels," APAE No. 107, Alco Products, Schenectady, N.Y., Oct. 14, 1961

5. Serpan, C.Z., Jr., Steele, L.E., and Hawthorne, J.R., "Radiation Damage Surveillance of Power Reactor Pressure Vessels," NRL Report 6349, Jan. 31, 1966

6. Steele, L.E., and Hawthorne, J.R., "Irradiation Effects on Structural Materials, QPR, 1 May-31 July 1962," NRL Memorandum Report 1351, Aug. 15, 1962

7. Steele, L.E., and Hawthorne, J.R., "Irradiation Effects on Structural Materials, QPR, 1 Aug-31 Oct 1962," NRL Memorandum Report 1371, Nov. 15, 1962

8. Steele, L.E., Hawthorne, J.R., and Watson, H.E., "Irradiation Effects on Structural Materials, QPR, 1 May-31 July 1963," NRL Memorandum Report 1448, Aug. 15, 1963

9. Steele, L.E., Hawthorne, J.R., and Watson, H.E., "Irradiation Effects on Reactor Structural Materials, QPR, 1 Feb-30 April 1964," NRL Memorandum Report 1534, May 15, 1964

10. Steele, L.E., and others, "Irradiation Effects on Reactor Structural Materials, QPR, 1 Nov. 1964-31 Jan. 1965," NRL Memorandum Report 1593, Feb. 15, 1965

11. Serpan, C.Z., Jr., and Steele, L.E., "Neutron Spectral Considerations Affecting Projected Estimates of Radiation Embrittlement of the Army SM-1A Reactor Pressure Vessel," NRL Report 6474, Sep. 30, 1966

12. Lathrop, K.D., "DTF-IV, A Fortran-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," USAEC Report LA-3373, Los Alamos Scientific Laboratory, 1965

13. Little, W.W., Jr., and Hardie, R.W., "2DB, A Two Dimensional Diffusion-Burnup Code for Fast Reactor Analysis," BNWL-640, Pacific Northwest Laboratory, Richland, Washington, 1968

14. Hawthorne, J.R., "Trends in Charpy-V Shelf Energy Degradation and Yield Strength Increase of Neutron-Embrittled Pressure Vessel Steels," NRL Report 7011, Dec. 22, 1969.

15. Pellini, W.S., "Evolution of Engineering Principles for Fracture-Safe Design of Steel Structures," NRL Report 6957, Sep. 23, 1969

16. Hawthorne, J.R., Steele, L.E., and Pellini, W.S., "Effects of Nuclear Radiation on the Properties of Reactor Structural Materials," NRL Report 5731, Jan. 22, 1962; or "How Neutron Exposure Alters Reactor Steels," Mechanical Engineering 84 (No. 5): 64-65 (May 1962)

17. Klier, E.P., Hawthorne, J.R., and Steele, L.E., "The Tensile Properties of Selected Steels for Use in Nuclear Reactor Pressure Vessels," NRL Report 6649, Dec. 26, 1967

☆ U.S. GOVERNMENT PRINTING OFFICE: 1970-397-177/222