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Radiation Damage Surveillance of Power Reactor Pressure Vessels

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Metallurgy Division

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The deleterious effect of high energy neutrons upon the mechanical properties of reactor pressure vessel steels has prompted the employment of material surveillance programs in many nuclear power plants. These programs provide for the exposure of test specimens representative of the reactor pressure vessel at in-reactor locations, wherein they will experience the same thermal and radiation damage history as the vessel itself. Evaluation of these specimens, which reveals the progressive changes in the mechanical properties of the vessel, provides a basis upon which operational procedures and maximum lifetime exposure may be formulated for the plant.

Pressure vessel surveillance specimens from the Yankee, Army SM-1 and SM-1A, and Big Rock Point reactors have been tested. Analysis of the Yankee program was hampered by the fact that only accelerated irradiation rate specimens were available for testing, so the pressure vessel condition was difficult to determine. Unexpectedly, the weld metal specimens from the Big Rock Point program showed a significantly higher mechanical property change, but because of operating temperature variations and overall adherence of material behavior to the nominal, this divergence of performance of the weld should be taken only as a warning for care in evaluating future surveillance specimens. In the Army reactors, the SM-1 and the SM-1A, surveillance programs suffer from not having specimens at the vessel walls. However, when neutron flux measurements and calculations were combined with test reactor material behavior as well as non-vessel wall surveillance material behavior, no significant deviations were observed; thus projections of results to later periods in the reactor lifetime could be made.

A review and an analysis of several instances of shortcomings in surveillance programs are presented along with a set of recommendations for consideration in planning new surveillance programs. In utilizing these recommendations, pressure vessel surveillance programs can be made to provide valuable information for use in determining plant operations; at the same time results from these programs may add to the general knowledge of radiation effects in pressure vessel steels or other materials subject to radiation. Recognition of the value of surveillance programs and their conscientious application should further the public acceptance of nuclear reactors as safe alternative power systems.

INTRODUCTION

Radiation damage studies initiated during the early days of nuclear power reactor development revealed the deleterious effect of high energy neutrons upon the notch ductility of reactor vessel steels (1-5), as manifested by a rapid rise in the ductile-brittle transition temperature with increasing neutron exposure. In addition, tests of tensile properties revealed a significant loss of uniform elongation and reduction of area for irradiated steels with increasing neutron exposure.

Carbon and low alloy steels are particularly susceptible to transition temperature increases and to a large extent also to the loss in tensile ductility properties. Therefore, since the majority of pressurized and boiling water power reactors planned and presently in service have vessels of these types of steel, it is important that the extent of radiation damage attendant upon them be measured and assessed in order to know more accurately the condition of the vessel and therefore to be able to optimize operating conditions. It is the purpose of a reactor-surveillance program to provide the required data from which these assessments may be made.

A minimum reactor surveillance program then should provide for the irradiation of test specimens which will yield transition temperature

NRL Problem M01-14; Projects RR 007-01-46-5409, SR 007-01-01, Task 0858, AT(49-5)-2110, and USA-MIPR-ERG-5-65. This is an interim report on one phase of the problem; work in this and other phases is continuing. Manuscript submitted August 23, 1965.

TABLE 1
Summary of Reactor Pressure Vessel Data and Surveillance Program Features

Reactor	Location	Pressure Vessel Material	Reactor Type	Vessel Thickness (in.)	Thermal Power (Mw)	Surveillance Material	Surveillance Locations
Yankee	Rowe, Mass.	A302-B	PWR	7-7/8*	600	Original and ASTM reference	Vessel wall and accelerated
Big Rock Point	Charlevoix, Mich.	A302-B	BWR	6	240	Original and reactor designer's standard reference	Vessel wall, accelerated, and above-core thermal control
SM-1A	Ft. Greely, Alaska	A350-LF1 (Modified)	PWR	2-3/8	20.2	Fabrication test plate A201 and A212-B reference	Flux at wall, specimens above core
SM-1	Ft. Belvoir, Virginia	A212-B	PWR	2-5/8	10	ASTM reference	Above core
MH-1A	USS STURGIS	AISI-316 Stainless Steel	PWR	3-1/8	45	nozzle cutout	Vessel wall, accelerated, and above core. Flux at wall.

*Material furnished by the steel mill was 8-1/8 in. thick. Subsequent machining operations reduced the thickness to the final as-built dimension of 7-7/8 in.

increase* and tensile data characteristic of the particular reactor vessel and pertinent to the operational history of that reactor. If specimens may not be exposed directly at the vessel wall, then a minimum effort should involve the measurement of the neutron flux at the vessel wall by the best available dosimetry techniques.

This report outlines certain characteristics of several power reactors (Table 1), describes the features of their surveillance programs, and presents the test results of specimens irradiated therein. Some engineering aspects of these results are considered, along with several problem areas in reactor vessel surveillance which have become apparent. Based upon the experience gained through these programs, as well as concurrent experimental radiation damage studies, recommendations are made for optimizing radiation damage surveillance program performance so as to provide the most meaningful results.

*A differentiation has been made between nil-ductility transition (NDT) temperature increases and transition temperature increases. An NDT temperature increase infers that drop-weight specimens of a material have been tested, and the initial nil-ductility transition temperature determined. This temperature then may be correlated to Charpy V-notch results for the same material by selecting the energy level ("fix" point) on the Charpy curve which most closely represents the NDT and using this level for the determination of the neutron-induced NDT increase. Transition temperature increase refers to the method of simply selecting an *arbitrary* energy level (generally based upon a representative NDT correlation point for that class of steels), and measuring the amount of temperature increase translated along that energy level to the curve for the irradiated condition.

POWER REACTOR SURVEILLANCE PROGRAMS

Yankee Reactor, Rowe, Massachusetts

The Yankee reactor is a large, 600-megawatt, thermal, pressurized water plant having a 7-7/8-in.-thick pressure vessel of ASTM type A302-B steel with a 1/4-in. stainless steel cladding. The surveillance program provided for two capsules located between the thermal shield and the vessel wall (designated wall) plus eight capsules in positions designated as accelerated. The latter positions, wherein the neutron flux and dose rate were considerably higher than those at the pressure vessel wall, were interior to the thermal shield and adjacent to the fuel core (Fig. 1). Each of the capsules contained Charpy V-notch impact specimens and tension specimens made from a section of the Yankee pressure vessel steel.

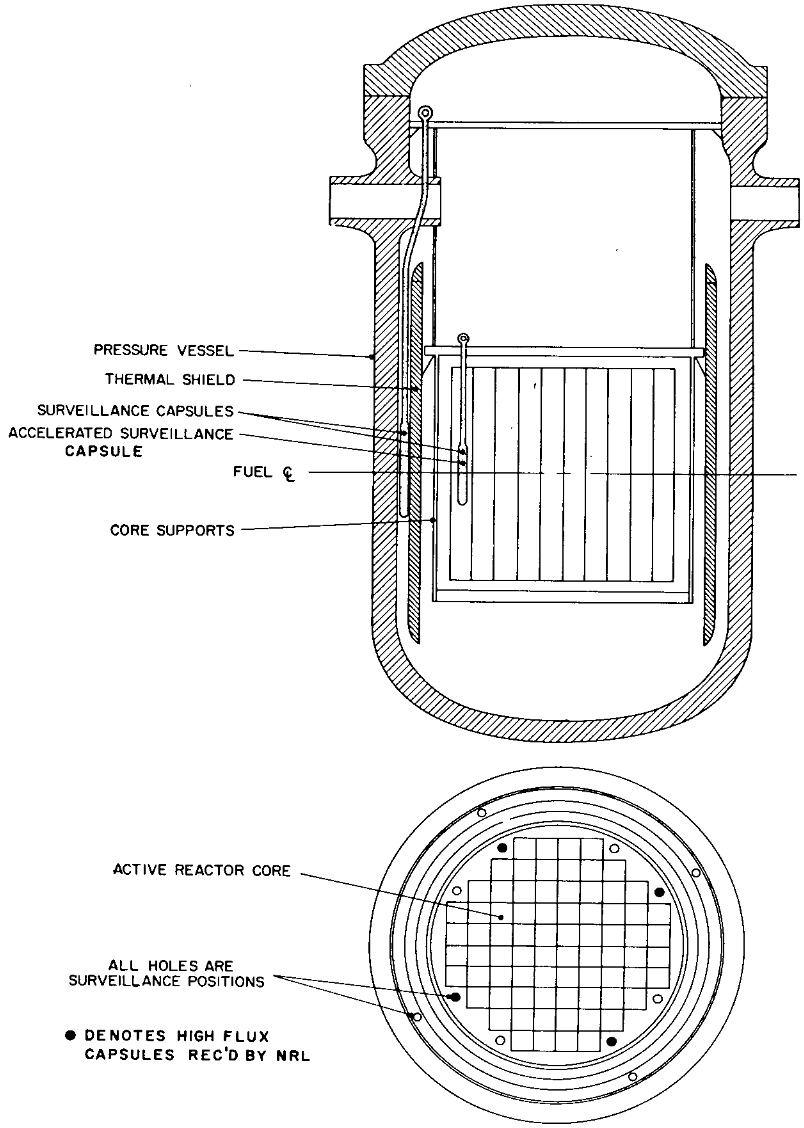


Fig. 1 — Schematic view of the Yankee Atomic Power Reactor, showing surveillance positions and the locations of capsules received by NRL (from Ref. 8)

Charpy-V specimens of a widely distributed and well documented reference heat of A302-B steel (6) were also included in each capsule for NDT determinations by the Charpy-V energy "fix" method (7). Details of this surveillance program have been reported (8). During the irradiation of these capsules (the second core life), four of the accelerated capsules as well as both of the wall capsules broke off, leaving only four accelerated capsules for the entire surveillance program. The limited number of specimens in each capsule required that all four be withdrawn at that time in order to perform a meaningful analysis. The problem of capsule loss and others to be mentioned will be discussed in the section titled Problem Areas.

Testing of the specimens was hampered by the fact that although the capsules were exposed in positions of nominally equivalent neutron flux, the four capsules actually were subjected to three different levels of neutron flux; consequently they received three different levels of total neutron exposure. It was assumed that this occurred as a result of displacement from the nominal position because of variations in water flow at those locations.

Testing schedules were devised to yield Charpy V 30 ft-lb temperature increase data for the three levels of exposure to the Yankee steel and the reference steel as well as annealing data for the Yankee steel. The limited number of specimens in each capsule did not permit full evaluation of as-irradiated properties plus annealing results for both materials. The transition temperature increase exhibited by the Yankee specimens from one capsule is presented in Fig. 2, along with the estimated increases (based upon reference steel behavior) for specimens from the other three capsules. The results of postirradiation annealing of Yankee specimens at three different conditions are shown in Fig. 3. It should be noted in particular that regardless of the uncertain nature of the increase for Yankee specimens from capsules 1, 2, and 6, annealing at 850°F for 168 hours resulted in essentially 100 percent recovery of initial properties. Transition temperature increases

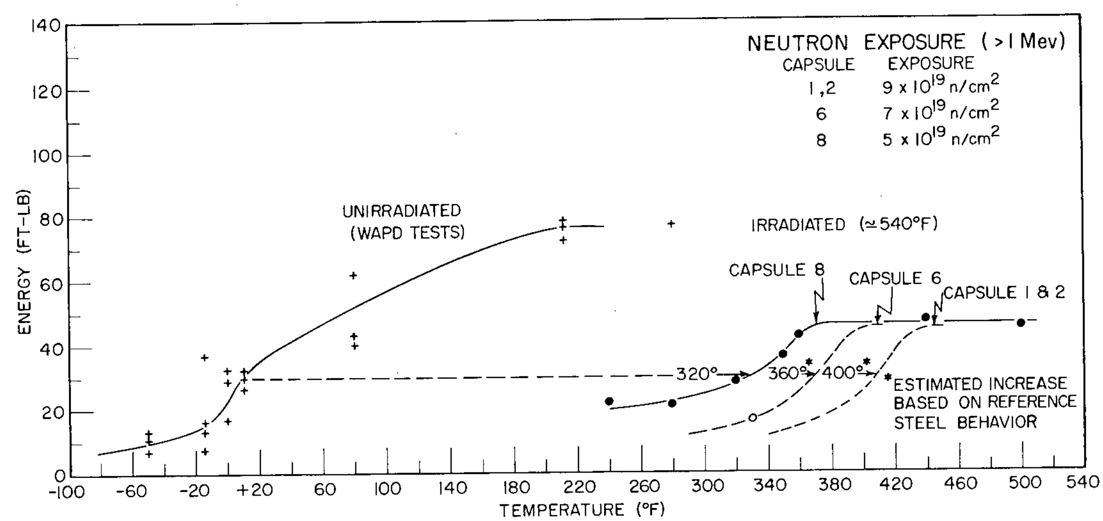


Fig. 2 — Irradiated notch-ductility characteristics of the Yankee pressure vessel steel, A302-B plate

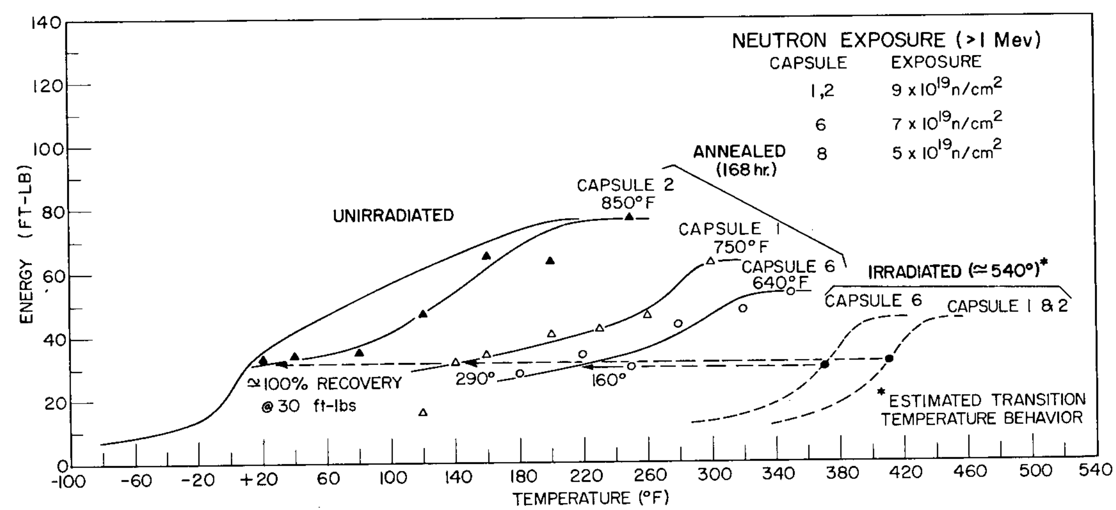


Fig. 3 — Effects of postirradiation heat treatment on the notch ductility characteristics of Yankee pressure vessel steel, A302-B plate

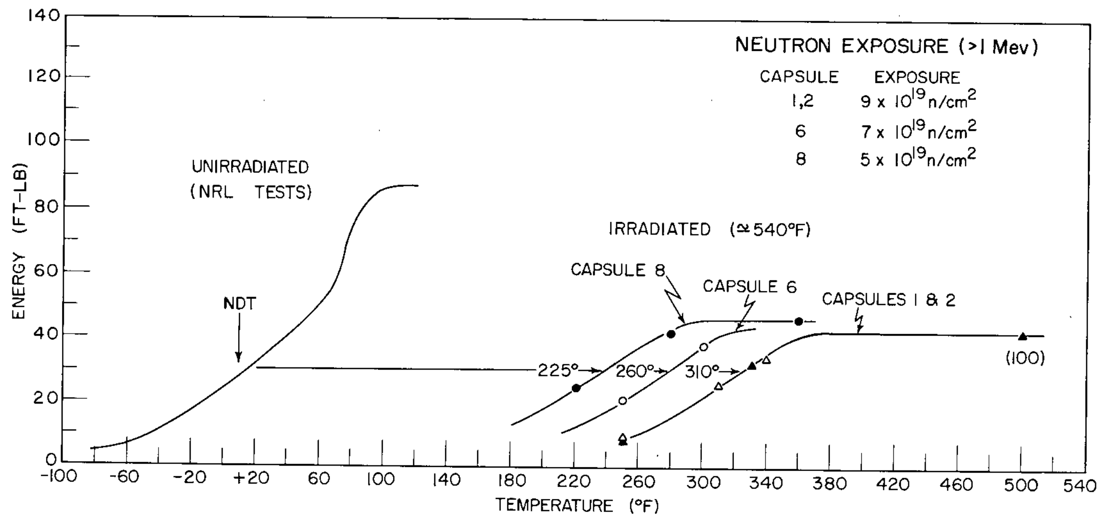


Fig. 4 – Irradiated notch-ductility characteristics of the reference steel (6-in. A302-B plate) exposed in Yankee surveillance capsules. The (100) indicates 100 percent shear fracture appearance.

for the reference steel specimens (Fig. 4) were significantly different from the Yankee steel results at the same exposure levels. These differences are shown in Fig. 5.

The significance of the data points shown in Fig. 5 is that the Yankee pressure vessel steel is apparently more sensitive to radiation, showing a larger transition temperature increase for a given neutron exposure than that observed for the reference heat of A302-B steel. On the other hand, the energy absorption at full shear fracture is almost the same for both steels after irradiation, in spite of the fact that the reference steel had a somewhat greater unirradiated notch toughness value (9).

In assessing the significance of the Yankee surveillance data, it must be remembered that the large increase in the transition temperature in the Yankee steel occurred from exposure in accelerated positions. Additionally, the reactor temperature gradually became lower toward the end of the core life, and transition temperature increases tend to be greater for a given neutron exposure sustained at lower temperatures (10). Thus, a direct comparison of Yankee steel results with true 550°F results (as shown in Fig. 5) is not completely valid, but is the best comparison which can be made for the Yankee operating conditions. The differences in transition temperature increase between the Yankee and reference steels is, however, real, since both types of specimens

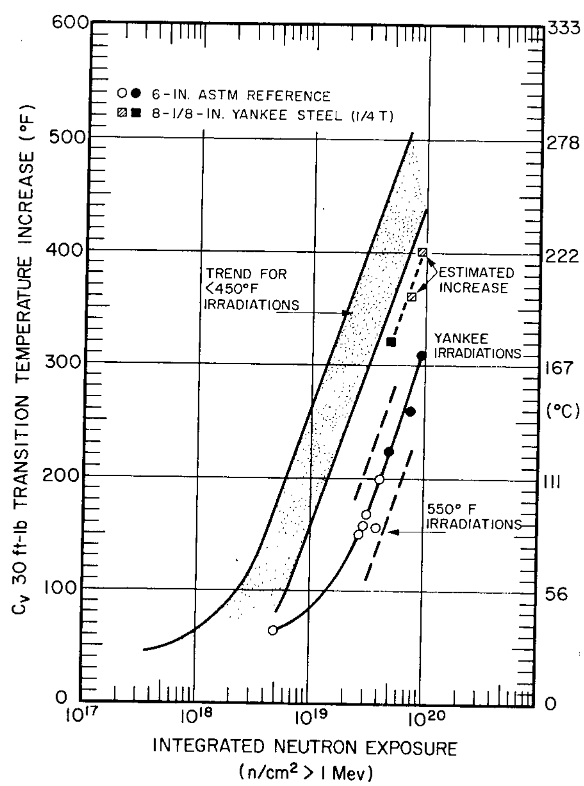


Fig. 5 – Trend band for 550°F irradiations of A302-B steel showing difference in neutron embrittlement between Yankee and reference steels. Solid points represent Yankee surveillance irradiations. Dotted lines indicate width of 550°F band determined by NRL irradiations of reference steel (open circles) and other A302-B heats in the LITR.

had the same exposure history during this irradiation period.

Big Rock Point Reactor, Charlevoix, Michigan

The Big Rock Point reactor is a 240 megawatt thermal, boiling water plant having a 6-in. thick vessel of ASTM Type A302-B steel. The comprehensive surveillance program for this reactor has been described previously (11). The program utilizes Charpy V and tensile specimens of base plate, weld metal, weld heat-affected zone (HAZ), and a standard reference material of the reactor designer, as well as neutron flux and temperature monitors. Irradiation locations are at the pressure vessel wall, in accelerated positions within the thermal shield, and above the core for out-of-flux thermal-control data (Fig. 6). Individual materials are loaded into separate capsules (Fig. 7), then the capsules, grouped as sets are secured in baskets, and the baskets are placed at the various locations so that all the types of materials within a basket are exposed to the same nominal neutron flux. Thus, the removal of one basket from a position yields specimens of all types for a complete material analysis for that position at the time of withdrawal.

NRL received the initial sets of capsules from one accelerated (designated shield) and one wall (designated wall) position basket irradiated during the first one and one-half years of reactor operations. Base plate, weld metal, and HAZ Charpy V results are shown in Figs. 8, 9, and 10, respectively. Data from the base-plate and HAZ specimens irradiated at the wall position fall within the scatterband of the unirradiated data; although the weld metal exhibited a lower initial transition temperature, specimens from both locations show measurable transition temperature increases. The reason for the greater radiation sensitivity of the weld metal is not understood at this time, although differences in sensitivity have also been observed with other materials (9,12). A further complicating factor was noted in the testing of unirradiated weld and HAZ specimens. Metallographic inspection of six HAZ specimens revealed that the notch root of five of the specimens was located just in the weld metal (at the weld metal-HAZ interface) rather than at the desired fusion line (Fig. 11). Assuming this pattern of slight misalignment of the notch root to be consistent with all HAZ specimens, the slight gradation of unirradiated properties from true weld area to true HAZ area (Fig. 12) is probably not significant, as indicated by the similarity of initial 30 ft-lb transition temperatures. However, metallurgical differences between the true weld area and the fusion line-weld metal interface area may be the reason for the difference in irradiation response noted for the two types of specimens (Figs. 9, 10).

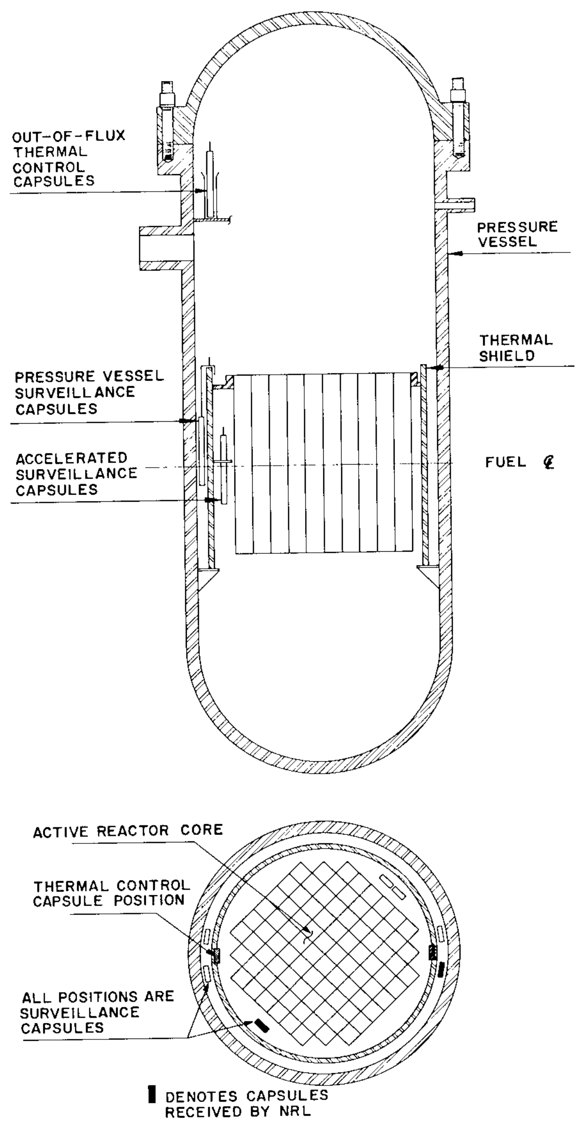


Fig. 6 — Schematic view of the Big Rock Point nuclear reactor, showing the relative locations of the surveillance capsule positions

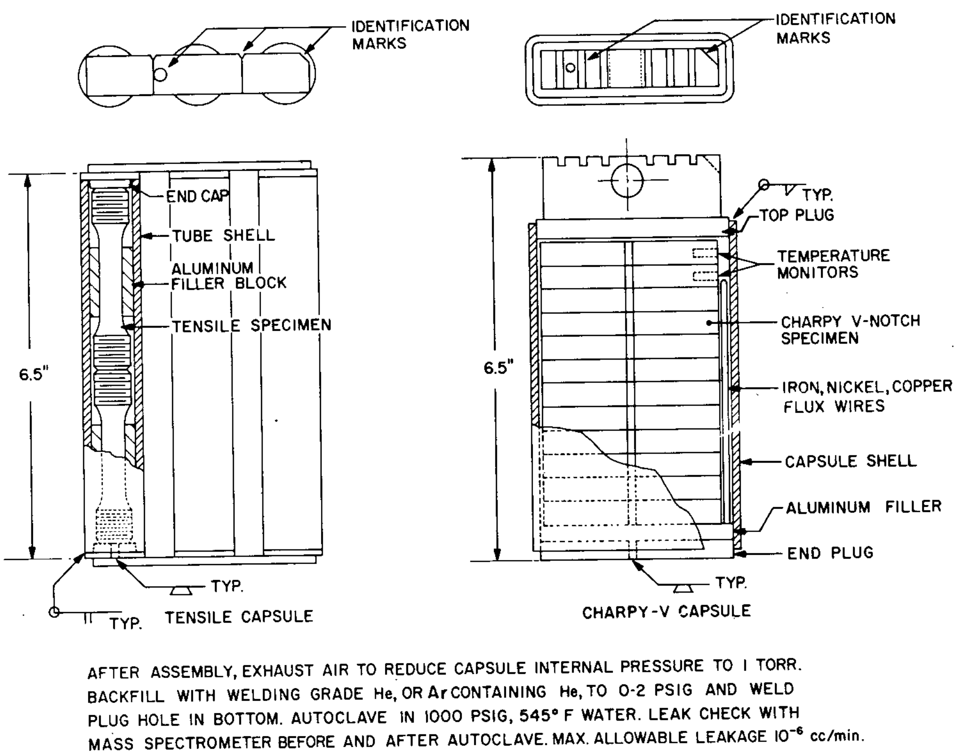


Fig. 7 — Big Rock Point reactor vessel steel surveillance assemblies, showing construction details, position of specimens, and temperature and flux monitors (from Ref. 11)

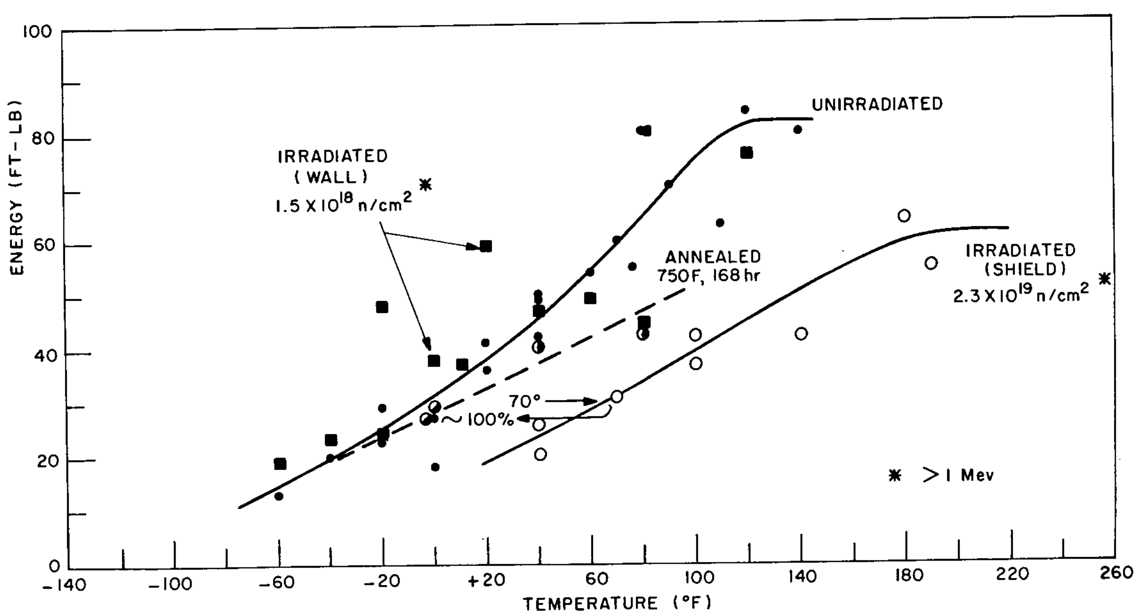


Fig. 8 — Notch-ductility characteristics of the Big Rock Point pressure vessel steel (6-in. A302-B base-plate specimens) before and after irradiation. Data points for the vessel wall position (solid squares) fall within the scatterband of unirradiated data, while data points from the accelerated exposure, shield position (open circles) show a 70-degree Charpy-V 30 ft-lb transition temperature increase. Postirradiation annealing data (half-closed circles), although very limited, indicate almost full transition temperature recovery.

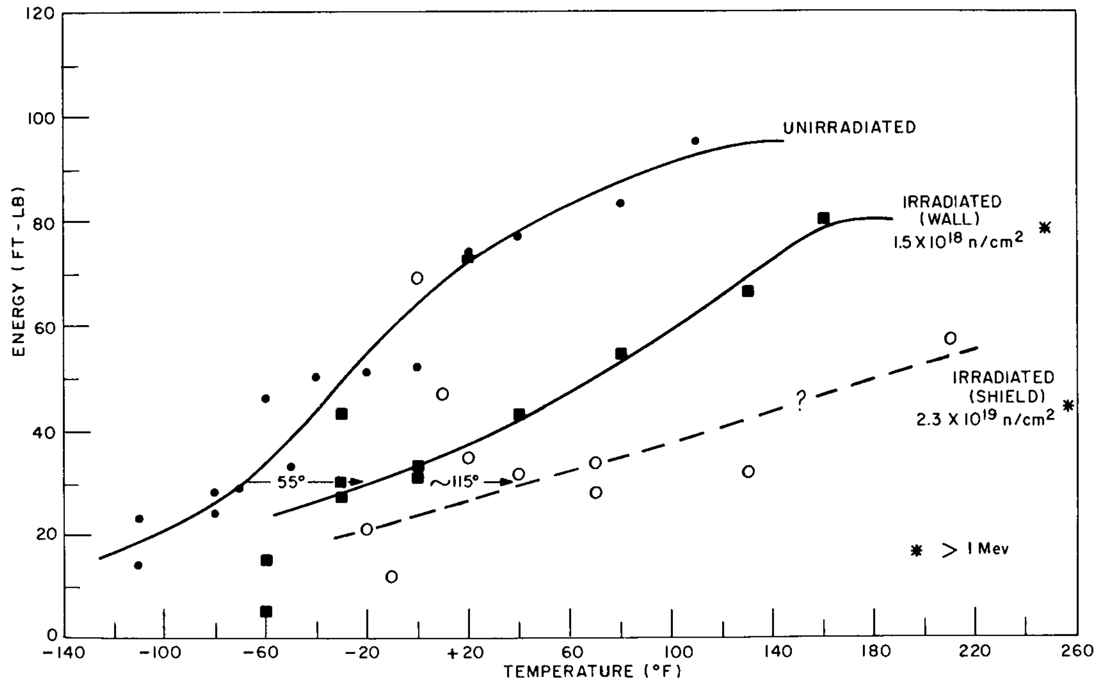


Fig. 9 – Notch ductility characteristics of Big Rock Point weld metal specimens before and after irradiation. Note positive Charpy-V 30 ft-lb transition temperature increase of both vessel wall and shield position specimens.

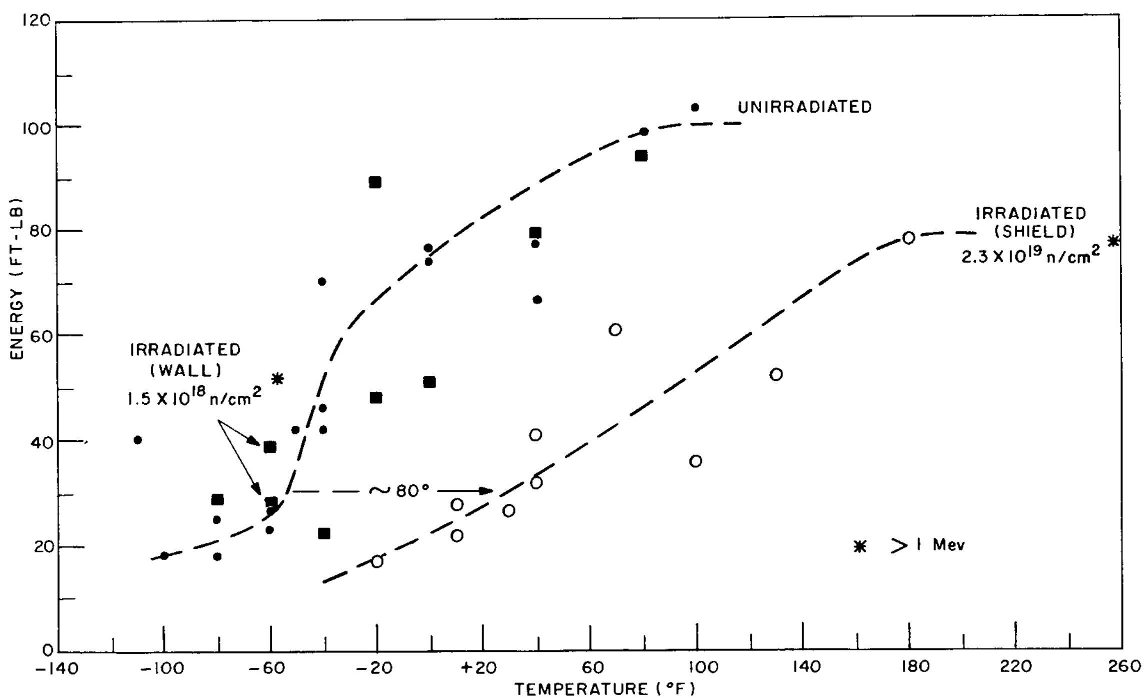
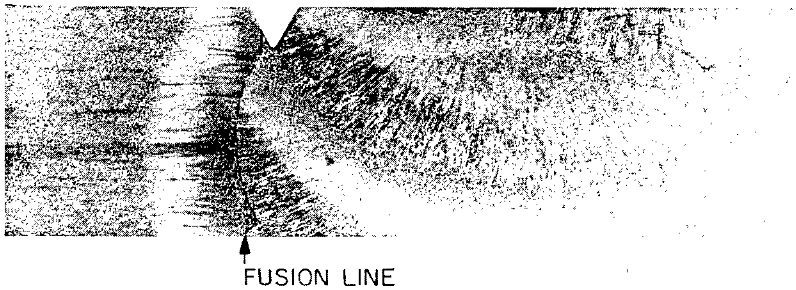


Fig. 10 – Notch ductility characteristics of Big Rock Point heat-affected-zone (HAZ) specimens before and after irradiation



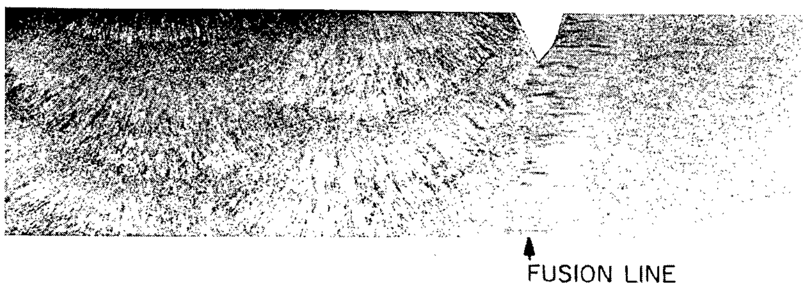


Fig. 11 – Photomacrographs of two Big Rock Point weld heat-affected-zone specimens. Dotted line indicates approximate location of fusion line. The notch root of the lower specimen is within Charpy V-notch specifications, although the overall geometry of the notch is not.

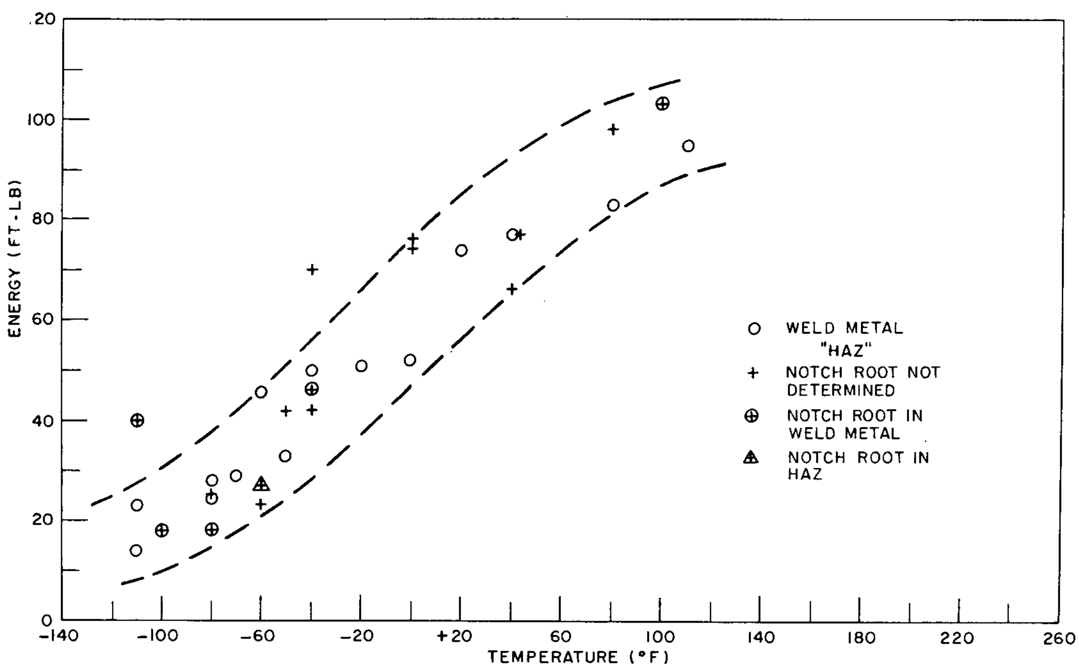


Fig. 12 – Notch ductility chracteristics of unirradiated Big Rock Point weld and weld heat-affected-zone specimens. Although a similarity of behavior is suggested, the notch root of several HAZ specimens was located in weld metal rather than at the fusion line.

Postirradiation heat treatment of a limited number of base plate surveillance specimens at 750°F for 168 hours resulted in essentially 100 percent recovery of the initial Charpy-V 30 ft-lb transition temperature (Fig. 8); however, the trend of the data points suggests that full recovery of energy absorption at the upper portion of the curve was not achieved.

The Charpy-V transition temperature behavior characteristics of the Big Rock Point reactor steel irradiated at a nominal temperature of 550°F to an exposure of 2.3×10^{19} n/cm² appear to be superior to the average behavior trend for A302-B steel at that temperature (Fig. 13). However, while the nominal exposure temperature for these specimens was 550°F, several periods of operations occurred at temperatures up to 600°F. The highest temperature operational period occurred about one-third the way through the irradiation and was

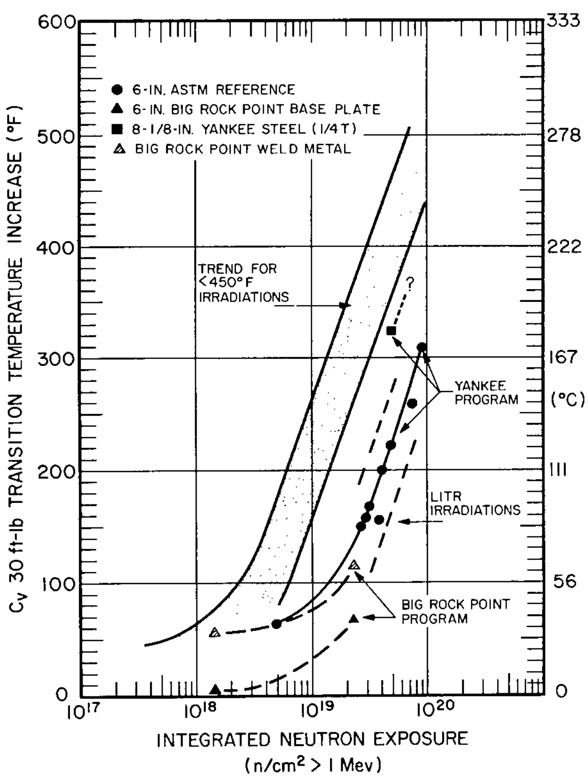


Fig. 13 — Trend band for 550°F irradiations of A302-B steel, showing relative transition temperature increases for Big Rock Point reactor specimens of weld metal and base plate. Dotted lines indicate width of band as determined from NRL irradiations of several heats of A302-B steel at 550°F. Closed circles are ASTM reference heat irradiations performed in Yankee and LITR reactors.

of such duration as to have a probable beneficial annealing effect. Thus, the Charpy-V transition temperature increase indicated for the Big Rock Point reactor vessel base plate at this time is quite low and indicates that little concern for pressure vessel embrittlement should accrue from projecting these results to a later period of the reactor's anticipated life. On the other hand, the weld metal did show a significant transition temperature increase even for the vessel wall exposure. While the initial transition temperature for the weld metal was considerably lower than that of the base metal, the rate of transition temperature increase for the weld metal is significantly higher. Thus, since this portion of the vessel appears to be weakest in terms of radiation damage, it must be used as the limiting consideration in any review of the vessel condition if future surveillance tests confirm the higher damage rate of the weld metal.

Army SM-1A Reactor, Fort Greely, Alaska

The SM-1A reactor, a stationary, medium power (20.2 megawatt thermal), pressurized water plant, designated as Model 1, field emplacement, is one of several small pressurized water plants constructed by the Army. The SM-1A has a 2-3/8-in. thick pressure vessel which is made from A350-LF1 (modified) steel.

The Army Nuclear Power Program philosophy was to construct very compact reactors capable of rapid construction and, in some cases, provide actual portability. This concept dictated that the vessel be made as small as possible to reduce weight and handling problems. The nearness of the vessel to the fuel core thus presents the potential problem of relatively more rapid increase in NDT temperature caused by the fast neutrons in the core neutron flux. Surveillance of this type of reactor then is most critical, but also quite difficult.

Direct placement of surveillance specimens next to the vessel wall of the SM-1A was impossible due to the limitation mentioned above. However, NRL was able to place two tube assemblies containing neutron flux monitors between the wall and the thermal shield. Furthermore, calculations (13) showed that the peak flux position at the wall could be approximated at a position above and to the side of the core where specimen capsules were then placed (Fig. 14).

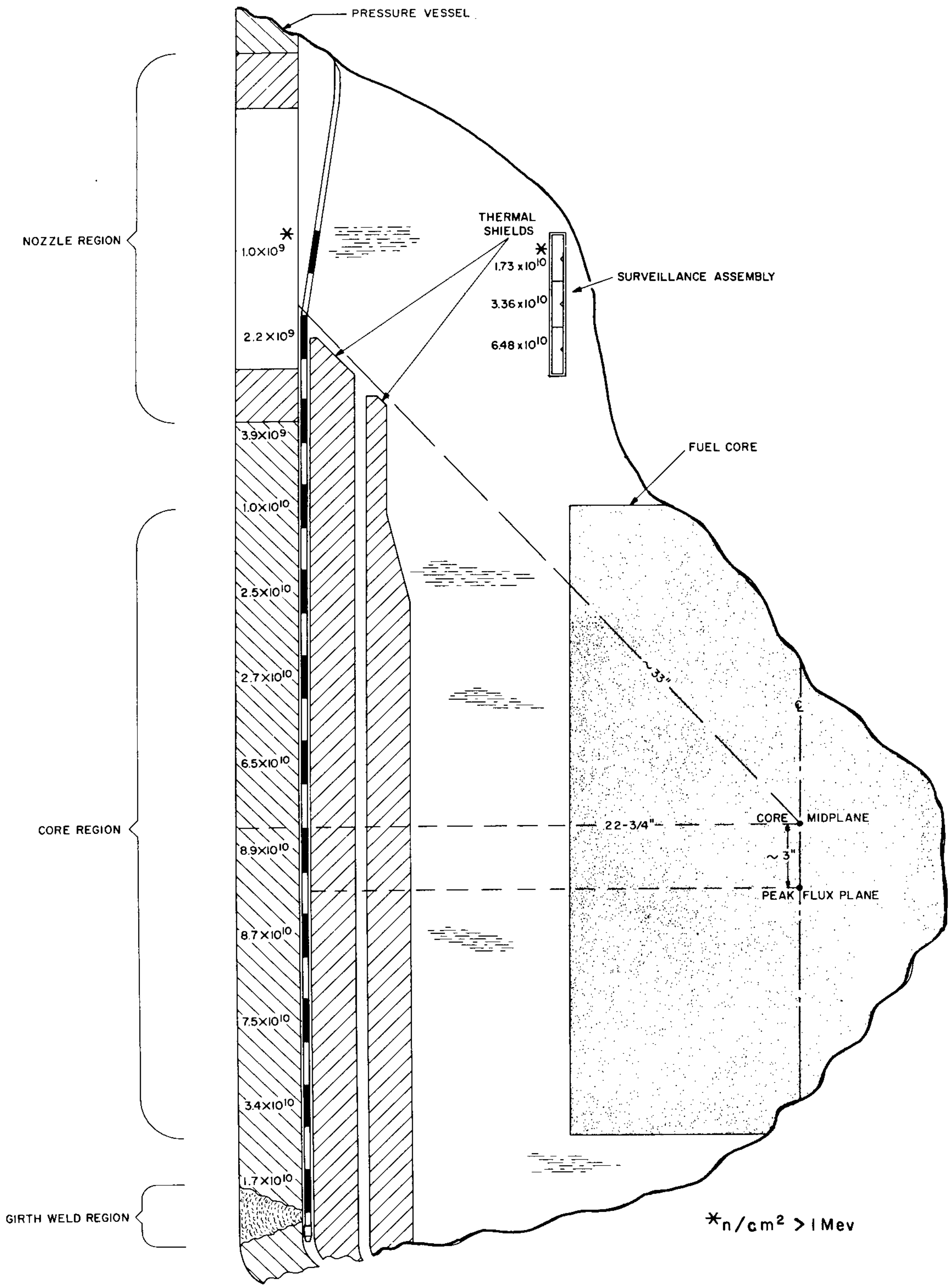


Fig. 14 – Schematic view of the Army SM-1A reactor, Fort Greely, Alaska, showing relative location of surveillance assembly and neutron flux levels measured at various locations. Vessel peak flux (8.9×10^{10} n/cm^2) was obtained from average 7.1-Mw exposure; surveillance-capsule fluxes (6.48, 3.36, and 1.73×10^{10} n/cm^2) were obtained from average 10.96-Mw exposure.

The neutron fluxes in Fig. 14 are measured values from different power level exposures. The neutron flux survey along the vessel wall (14) was made by radiochemical separation and analysis of the induced Mn^{54} activity from the stainless steel thermal flux monitor tube, since the high energy neutron monitor wires from a second monitor tube could not be extracted.

Capsules from the above-core positions were recently removed, and the results of specimen tests from the bottom section of those capsules are presented in Fig. 15. The irradiation response of the bottom group of specimens compares favorably with the trend previously established for this same material with accelerated irradiations (14) (Fig. 16). On the other hand, specimens from the top capsule section received such a low neutron exposure that the resultant data points could not be distinguished from the unirradiated results. Analyses of the neutron dosimeters from the bottom capsule position indicated a flux level that is about half the peak flux positions at the vessel wall (when both fluxes are normalized to the same power level).

The surveillance test results combined with the capsule neutron dosimetry and the neutron flux survey at the vessel wall provide sufficient data to permit assessment of the condition of the SM-1A pressure vessel for future operations. One assessment has already been made (14), based upon full-power, full-time (20 Mw) operations.

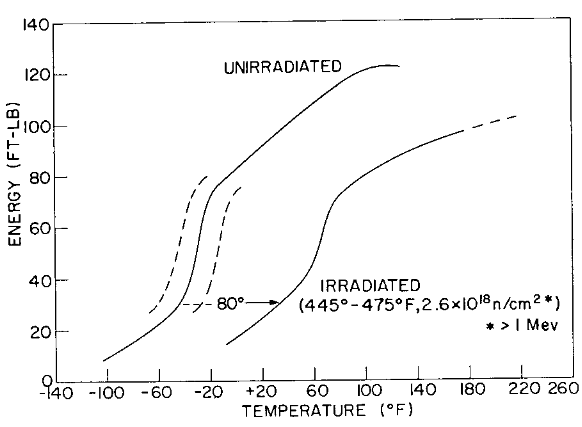


Fig. 15 – Notch ductility characteristics of fabrication test plate (3-5/8-in. A350-LF1 mod. plate) from SM-1A pressure vessel ring forging. Dotted lines indicate width of unirradiated data scatterband. Irradiation temperature was determined from low melting alloy monitors (445°F alloy melted, 475°F alloy did not melt).

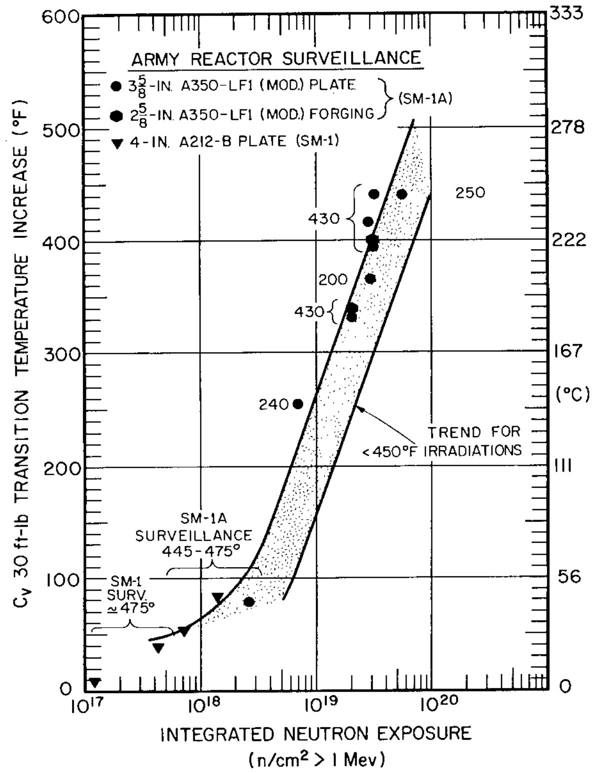


Fig. 16 – Charpy-V 30 ft-lb transition temperature increase *versus* integrated neutron exposure for SM-1A and SM-1 pressure vessel steels referenced to <450°F trend band irradiations. Data points near low exposure end of trend band are from respective surveillance programs. Points indicating material behavior at higher exposures are from accelerated irradiations in test reactors. Data for A212-B steel from SM-1A surveillance program are shown for reference.

The exposure period for the specimens reported herein was for 22 months at an average power level of 10.96 Mw during operational periods and represents about 71 percent of a full-time, full-power (20 Mw) year. If this 22-month surveillance period can be considered as normal operation, then the previously projected NDT temperature increases (based on 20 Mw operation) can be modified, with the result that the reactor vessel should receive the stated degrees of embrittlement over a much longer period of time.

Army SM-1 Reactor, Fort Belvoir, Virginia

The SM-1 reactor is a stationary, medium power (10 megawatt thermal), pressurized water plant used by the Army as a training plant and is also the prototype for the SM-1A plant described above. The 2-5/8-in. thick vessel is made of A212-B steel. Reactor vessel surveillance was truly in its infancy when the SM-1 was in late stages of construction, so that radiation damage surveillance for this plant is quite limited. Again, the compact nature of the vessel and core precluded inclusion of specimens adjacent to the vessel wall. Flux monitor tubes were not installed. Prior to the NRL surveillance effort, eight tubular capsules were placed in positions approximately below and nearer the SM-1 core than the positions indicated in Fig. 14 (SM-1A reactor). Each of these capsules contained three izod impact bars each bar having six notches. Analysis of the early results on these surveillance specimens (15) was hampered by a lack of sufficient specimens at a particular location for good curve delineation and of conclusive neutron dosimetry. In addition, after operating several years, a location above and to the side of the core (also similar to that in the SM-1A, Fig. 14) was made available for specimen placement by NRL.

After a 14-month exposure, the two NRL capsules were removed from the above-core location. These capsules contained specimens of ASTM A212-B reference material, which is the same type as the vessel material. Charpy-V test results from these specimens are shown in Fig. 17. These data, which are also plotted on Fig. 16, have been found to reinforce trends in transition temperature increase *versus* neutron exposure suggested by previous NRL irradiations on the same material but in accelerated exposures in test reactors (16).

By using more recent neutron flux information (13), it has been possible to reevaluate (17) the izod impact data. NRL Charpy V-notch data (Fig. 16) can be shown to compare favorably to the izod impact energy data points from three samplings (15,18), if neutron exposures are calculated from data in Refs. 13 and 19.

In spite of these comparisons, it is most difficult to draw any concrete conclusions concerning the embrittlement condition of the SM-1 since, for the reasons given above, no measurements of the neutron flux at the vessel wall have been made. However, mock-ups of both the SM-1 and SM-1A have been made, and neutron exposures per megawatt-year for both plants have been determined (19). Since the mock-up value for the SM-1A correlates well with the measured value, it is reasonable to assume that the mock-up value for the SM-1 should be a fair approximation as well. The limiting exposure to the SM-1 vessel has been placed by the Army at 9×10^{18} n/cm^2, to allow for the initial NDT, the "NDT plus 60" safety criterion and the radiation-induced increase. The mock-up value has indicated to the Army that the vessel life expectancy is

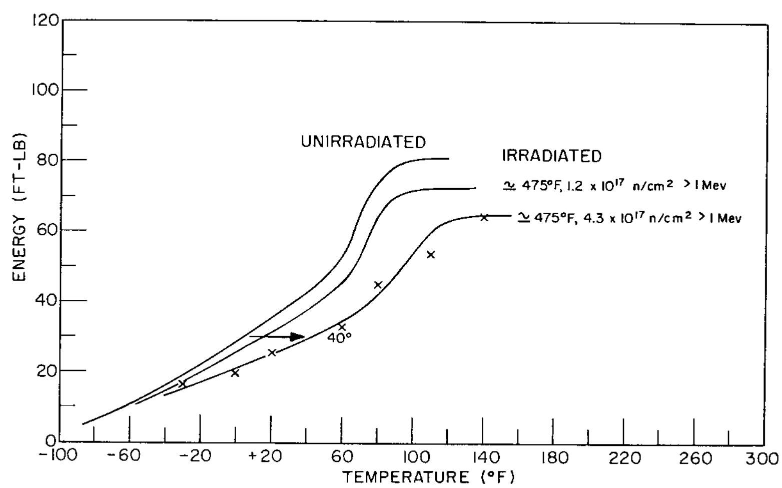


Fig. 17 – Notch ductility characteristics of 4-in. A212-B plate steel representative of the SM-1 pressure vessel. The curve for the 4.3×10^{17} n/cm^2 exposure was from the bottom surveillance capsule section; the lower exposure curve was derived from specimens of the top capsule section.

about 76 Mw-years. Since by mid-1965, the SM-1 has operated for about 25 Mw-years, in about eight calendar years, a continuance of the same type of intermittent, low power, training type operations should permit the reactor to be operated for its anticipated 20-calendar-year lifetime without serious concern for the ductility of the vessel.

Army MH-1A Reactor, Now Under Construction

The MH-1A reactor is a mobile (barge-mounted), high power (45 megawatt thermal, 10 megawatt electric) pressurized water plant having a 3-1/8-in. thick. vessel of AISI type 316 stainless steel. Placed into the converted liberty ship, CHARLES H. CUGLE (recently renamed STURGIS), the plant will be capable of being towed to any deep-water port in the world to serve as an auxiliary supply of power.

NRL was requested to prepare a surveillance program for the MH-1A at a time when the vessel shell courses and thermal shields were still being fabricated. It was thus possible to incorporate certain features into these components to assure a reasonably comprehensive surveillance program.

The vessel fabrication changes required by the program were the provision of two flux monitor tubes to be welded along the inside diameter of the vessel wall from a point below the peak flux plane to the vessel head flange area. Also, pockets were machined into the outer thermal shields to provide access for surveillance capsules. Positions interior to the inner thermal shields and adjacent to the fuel elements were provided for accelerated irradiation capsules, and positions above the core were provided for long-term, thermal control-type surveillance capsules. The relative locations of these positions are shown in Fig. 18.

The MH-1A vessel nozzle cutouts were carefully conserved and used to prepare Charpy V and tensile specimens for the surveillance program. In accordance with the presently recommended ASTM Procedure for Reactor Vessel Surveillance, specimens will be loaded and secured in stainless steel frames, and left open to the reactor coolant water. For this surveillance irradiation, Charpy bars have been left unnotched, and tensile specimens have been prepared with the gage section 0.010 in. oversize in diameter. Notching and machining to size will be performed after irradiation. Neutron flux and temperature monitors sealed in stainless steel tubes have also been placed in every assembly frame.

PROBLEM AREAS

Any component which is installed in a nuclear reactor system ideally is very carefully produced, undergoes a stringent examination, is reviewed for safety, and hopefully is tested under conditions as close to service conditions as possible. However, in many cases, the only real service-condition test comes when the component is placed into operation, since nothing short of operating reactor conditions are true and meaningful. Thus, even the most carefully devised plans and designs often go awry when confronted with the operating nuclear plant environment.

Several examples of the effect of the reactor environment upon surveillance components have been mentioned in the preceding sections. No fault is to be laid at the door of any particular person or organization. Rather, the mistakes of the past should be carefully reviewed as lessons for the future. The following is a brief analysis and discussion of some of these occurrences.

Six of ten surveillance capsules from the Yankee program broke away from their mounts and were subsequently recovered at the end of the fourth core life. These assemblies were designed and inserted only a short time prior to startup for the second core operation of the reactor. The design called for the capsules to be suspended in an unrestricted manner from support disks located near the head flange (Fig. 1). It was subsequently concluded that the velocity and turbulence of the coolant water flow was such that the capsules acted like pendulums and eventually broke away from their mounts, apparently as a result of fatigue of the capsule supports. Substantial evidence of a pendulum effect, or back-and-forth motion, was observed on one capsule which did not break away (Fig. 19). The photograph shows a portion of the stainless steel sheath material worn away, and the resultant hole. Inspection showed a worn area on a specimen taken from that location. Further indications of wear were observed and noted (9). Displacement of the capsules was also thought to be

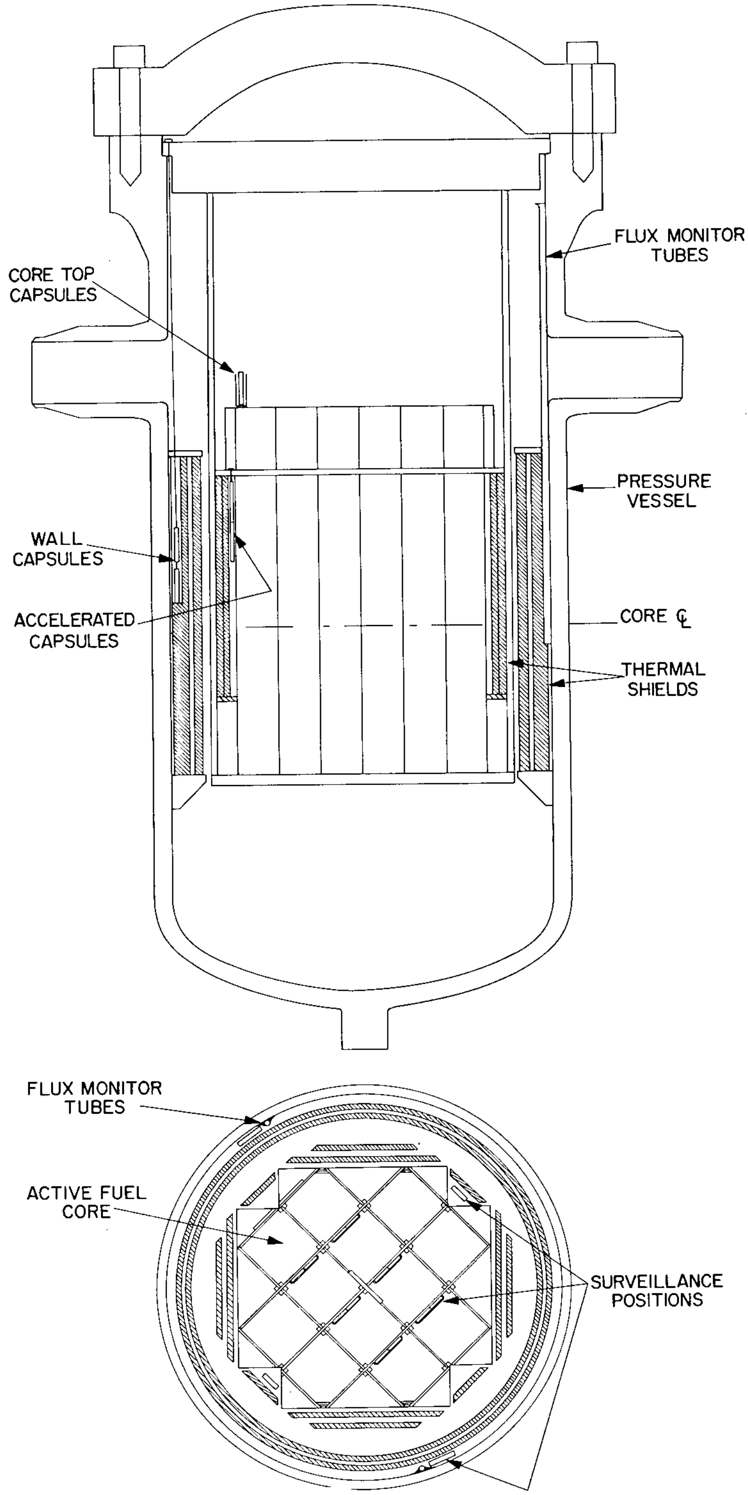


Fig. 18 — Schematic view of the MH-1A floating nuclear power plant pressure vessel showing the relative locations of the surveillance capsule positions (from Martin Company Drawing 393A4153051)

Fig. 19 – Yankee surveillance capsule sheath, showing hole caused by fluttering against a fixed reactor member (in-cell photo)

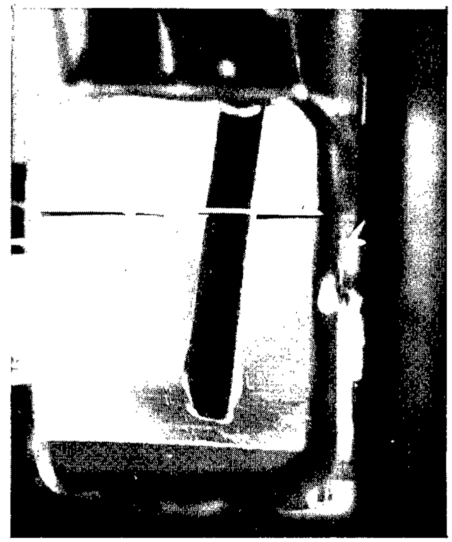


Fig. 20 – Big Rock Point surveillance capsule, showing outer sheath bulged by formation of Al_2O_3 generated by moisture acting upon aluminum spacer within capsule (in-cell photo)

partially responsible for the large gradations in neutron flux in the capsules. This situation probably would not have occurred if the capsules had been more positively secured in place.

The Big Rock Point surveillance capsules were carefully produced according to stringent requirements (Fig. 7). Nevertheless, one capsule received by NRL had a large bulge on one end (Fig. 20). Upon opening this capsule, it was found that the aluminum spacer block had turned to powder, presumably Al_2O_3. It was surmised that water had leaked in and reacted with the aluminum, with subsequent gas generation and oxide formation. Closer supervision and increased quality control are the only means even suggested for the future prevention of this occurrence. It is thought that the area of swelling was limited, since the primary reason for swelling was the formation of the oxide, which was effectively contained from movement by the capsule which was under pressure from the reactor coolant. Fortunately, temperature monitors located in the specimens at the other end of the capsule were found to be in agreement with the temperature indications from other capsules, so the specimens were not considered lost due to overheating from gamma heating effects enhanced by poor heat transfer as a result of the capsule bulging.

Placement of the neutron flux monitor tube adjacent to the pressure vessel wall of the SM-1A was accomplished just prior to the startup of that plant. As previously indicated, access to the position was nearly nonexistent. The method used for monitor installation was to thread together short, 7-in. sections of stainless steel tubing, and to mechanically stake the pieces together. This design proved to be unsatisfactory, since turbulence, or vibrations probably, worked the pieces of the one tube apart, resulting in their falling down inside the vessel near some of the control rod mechanisms. In spite of this problem, the data obtained from the continuous piece of monitor tubing removed earlier for the flux survey was quite valuable. However, the trouble caused by the separation of the remaining tube holder must serve as a caution to avoid the use of surveillance holders which are not fully secured in a reactor system. The technique presently being employed in the MH-1A, that of full penetration welds of the flux monitor tubes to the vessel wall, should be an adequate solution to this problem.

Material problems as well as design problems have also become apparent in several cases. The improper location of the notch root with respect to the fusion line of several Big Rock Point HAZ specimens has been pointed out. Figure 11 shows the location of the notch roots and fusion line of two of these specimens. Considering the difficulty of preparing HAZ specimens, it is very commendable that it has been attempted by the reactor designer. A possible solution for the HAZ specimen preparation problem might be to machine the specimens oversize in length, etch to determine the exact fusion line, notch at the fusion line, and machine to final dimensions using the notch as the point for measurements. Admittedly, this process is tedious, but it would yield specimens which are representative of the property which they are expected to monitor. In spite of the difficulties, it is important that weld and HAZ specimens be included since, as indicated in this report, the weld metal or the HAZ may be the limiting materials for assessing radiation damage in reactor components.

The value of utilizing specimens from the vessel shell components along with reference or correlation-monitor specimens for vessel surveillance programs was vividly demonstrated in the Yankee program. The quite unexpected higher radiation sensitivity of the Yankee steel would have posed a very perplexing problem in analysis had there not been the reference specimens available to provide a base line for comparison. Considerable doubt would have been cast upon the neutron flux analysis as well as the theoretical calculations, the exposure temperature, and a host of other variables if the reference specimens had not been available. Although standard material of the reactor designer was included in the Big Rock Point program, correlation monitors available for the entire industry, as those utilized in the Yankee program, were not included. This is unfortunate, since their inclusion could have helped confirm, or disprove, the apparently low radiation damage rate in terms of transition temperature increase of the base metal.

RECOMMENDATIONS

As pointed out in the Introduction, a meaningful surveillance program must provide for the exposure of representative materials at the point of maximum interest. Many other facets must also be provided as touched upon throughout this paper. The American Society for Testing and Materials is presently revising its existing standard for surveillance programs (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors – E185-62) (20) with the hope that it will be adopted in the near future as a comprehensive guide for all new programs and for the updating, if possible, of existing ones. The recommendations of the authors concerning surveillance programs, which closely follow those of the ASTM, are listed in the Appendix.

The listing in the Appendix provides a reasonable guideline for surveillance programs. While going beyond pressure vessel surveillance alone, it is not intended to be absolutely comprehensive. Rather, it is a compilation of features which must be considered and dealt with, and from which further ideas can be conceived for a meaningful surveillance program. A complete discussion of these items is beyond the scope of this paper; however, a final suggestion may be useful. For one designing a surveillance program, it would be most helpful to discuss the subject with others who have had some previous surveillance program experience.

CONCLUSIONS

Reactor pressure-vessel surveillance programs are expensive and contribute nothing to the immediate plant efficiency. However, the information which is obtained from them can be of considerable value in determining plant operations and at the same time may add to the general knowledge of radiation effects in pressure vessel steels or other materials subject to radiation. Primarily, the operator can know with reasonable assurance the progressively changing mechanical properties of his pressure vessel. Knowing the behavior characteristics of the material and the neutron flux at the vessel wall, he can maintain a running estimate of the NDT temperature increase and the loss of tensile properties, and, if it becomes necessary, devise alternate operating procedures which give cognizance to radiation-induced changes in materials.

Postirradiation heat treatment or annealing of surveillance specimens has been mentioned in several sections of this report. Depending upon the temperature and duration of these heat treatments, 40 to almost 100 percent recovery of initial properties has been recorded. This amount of recovery suggests, then, that annealing of a pressure vessel could result in a substantial recovery of initial properties.

While reactor pressure vessel surveillance is by no means an established procedure with hard and fast rules, it is rapidly moving out of the stage of exploratory practice. Surveillance results interpreted along with experimental radiation damage programs have provided information from which significant changes in operating procedures and anticipated vessel life expectancies have been made. While some of these changes have been in a conservative direction, others have served to suggest a liberal projection of the life of the vessel. Recognition of the value of these programs and their conscientious application will undoubtedly further the public acceptance of nuclear reactors as safe alternative power systems.

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Appendix
SURVEILLANCE PROGRAM RECOMMENDATIONS

I. Materials (include all at each location)
- A. Base plate (all heats, actual material)
- B. Weld Metal
- C. Heat-affected zone
- D. Reference material (well documented – preferably used in other radiation effects investigations)

II. Specimens (include all at each location)
- A. Charpy V-notch (ASTM specifications, number both ends of each)
- B. Tension (ASTM specifications, number both ends of each)
- C. Others, as desired (fatigue, tube burst, *etc.*)

III. Location
- A. At vessel wall (or at component to be surveyed)
- B. Accelerated positions
- C. Out-of-flux thermal control positions

IV. Neutron Flux
- A. Flux monitors in each capsule
  - (1) Iron
  - (2) Cobalt-free copper (or cobalt content predetermined)
  - (3) Others, as dosimetry technology permits
- B. Flux determination at vessel wall or at component surveyed (to be repeated after each major change in reactor operating characteristics)

V. Temperature Determination
- A. Range of low-alloy monitors, each capsule
- B. Instrumented determinations
- C. Operational records (reactor power and coolant temperatures especially)

VI. Capsules
- A. Corrosion-protective sheath
- B. Baked out just below operating temperature and seal welded at that temperature
- C. Identification number
- D. No sheath for stainless steel or noncorroding type specimens
  - (1) Unotched Charpy specimens
  - (2) Tensile gage section oversize
  - (3) Specimens secured in position
  - (4) Flux and temperature monitors secured in stainless steel

VII. Program Report
- A. Diagram of specimen located in parent material
- B. Diagram of each capsule showing location of all components
- C. Diagram of capsule locations and identification system
- D. Unirradiated material data
- E. Specimen description and dimensions
- F. Special instructions and features

VIII. Testing Report
- A. Tension and special testing at operating temperature
- B. Testing instrument calibration certification
- C. Test results, neutron flux and exposure, assumptions, cross sections, half lives, temperature of exposure
- D. Special features or observations

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13. ABSTRACT

The deleterious effect of high energy neutrons upon the mechanical properties of reactor pressure vessel steels has prompted the employment of material surveillance programs in many nuclear power plants. These programs provide for the exposure of test specimens representative of the reactor pressure vessel at in-reactor locations, wherein they will experience the same thermal and radiation damage history as the vessel itself. Evaluation of these specimens, which reveals the progressive changes in the mechanical properties of the vessel, provides a basis upon which operational procedures and maximum lifetime exposure may be formulated for the plant.

Pressure vessel surveillance specimens from the Yankee, Army SM-1 and SM-1A, and Big Rock Point reactors have been tested. Analysis of the Yankee program was hampered by the fact that only accelerated irradiation rate specimens were available for testing, so the pressure vessel condition was difficult to determine. Unexpectedly, the weld metal specimens from the Big Rock Point program showed a significantly higher mechanical property change, but because of operating temperature variations and overall adherence of material behavior to the nominal, this divergence of performance of the weld should be taken only as a warning for care in evaluating future surveillance specimens. In the Army reactors, the SM-1 and the SM-1A, surveillance programs suffer from not having specimens at the vessel walls. However, when neutron flux measurements and calculations were combined with test reactor material behavior as well as non-vessel wall surveillance material behavior, no significant deviations were observed; thus projections of results to later periods in the reactor lifetime could be made.

A review and an analysis of several instances of shortcomings in surveillance programs are presented along with a set of recommendations for consideration in planning new surveillance programs. In utilizing these recommendations, pressure vessel surveillance programs can be made to provide valuable information for use in determining plant operations; at the same time results from these programs may add to the general knowledge of radiation effects in pressure vessel steels or other materials subject to radiation. Recognition of the value of surveillance programs and their conscientious application should further the public acceptance of nuclear reactors as safe alternative power systems.

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