

## SYNOPSIS AND REPORT HIGHLIGHTS

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This summary section is provided to aid the reader in finding specific aspects of reactor pressure vessel surveillance within this report and to provide a brief review of its major highlights. For clarity, the synopsis is divided to correspond with the six chapters of the report.

1. Basis for Vessel Surveillance.
2. Neutron Embrittlement of Reactor Vessel Steels—A Brief Review.
3. Army Reactor Vessel Surveillance.
4. Commercial Reactor Vessel Surveillance.
5. Neutron Dosimetry and Spectrum.
6. Surveillance Programs Critique and Recommendations.

In addition to this overall review of report highlights, the introduction to each chapter provides an abbreviated review of the contents of that chapter.

## CHAPTER 1—Basis for Vessel Surveillance

### Introduction

This report was undertaken at the request of the Metal Properties Council to review and analyze available reactor pressure vessel radiation damage surveillance data in order to guide new reactor owners and others having an interest in this aspect of nuclear power. In Chapter 1, some important factors are outlined relative to the basis for reactor vessel surveillance.

### Highlights

1. Ferritic steels such as those used for nuclear power reactor pressure vessels undergo a marked transition from ductile-to-brittle fracture be-

havior over a relatively narrow temperature range (100 to 150 F) usually near 0 F.

2. Neutron radiation gradually elevates the transition temperature range, lowers the energy required to fracture, hardens, and increases the strength of a steel.

3. These effects of radiation tend to exaggerate the chances for brittle fracture of the vessel; hence, the need for some technique to measure, progressively, radiation effects and the propensity of the vessel steel to serve its function with assured reliability.

4. The usual basis for assessment of the vessel's condition in service is to expose notched impact bars (Charpy V-notch) of the vessel steel (plate and weld) at the vessel wall during operation and to remove and test groups of these specimens periodically. Radiation induced changes in the specimens are accepted as a measure of change in the vessel and thus provides a basis for reactor operational decisions which help assure vessel reliability in service.

5. The changes considered most significant are the increase in transition temperature and the reduction in fracture energy, especially the energy absorbed at temperatures where ferritic steels exhibit full shear fracture, usually associated with high ductility, but, with severe irradiation, may, in fact, permit low energy tearing. In the latter situation, assuring higher vessel temperatures before significant pressurization may be of no consequence. These two types of changes often are referred to as radiation embrittlement. The term embrittlement is relative and is used here to denote either small or large changes. It is important in either context because of the important function of the vessel and because of the economic costs of overly conservative design or of operational limitations which result from uncertainty.

6. The Charpy V-notch specimen forms the primary basis for vessel surveillance and for measuring radiation effects on notch ductility. This test is made more meaningful by the fact that, when properly correlated with the drop weight test, a nil-ductility transition (NDT) temperature can be determined for a given vessel steel. The use of NDT is important because it gives a *point* for measuring an increase in transition temperature rather than a *range*, and, since it was originally developed on the basis of structural failures, it provides a reference point for fracture analysis using the fracture analysis diagram (FAD) for relating stress, temperature, and flaw size for failure. An NDT or Charpy V-notch based increase in transition temperature is valuable but does not describe the radiation induced reduction in energy absorption at the ductile (shelf) end of the transition range. (Criteria are now being developed which will guide vessel embrittlement projections using the Charpy V-notch data (correlated with fracture toughness data) so as to integrate both reduced shelf energy absorption and transition or NDT temperature increases.)

7. Recent developments relative to reactor pressure vessel safety include industry programs for describing statistically the properties of representative vessel steels and for inspecting before and during service for critical flaws. A related United States Atomic Energy Commission (USAEC) sponsored program seeks to evaluate the potential for failure resistance in heavy section steels.

8. Within the USAEC program, a most significant development has been recorded on the basis of full thickness plane strain fracture mechanics  $K_{Ic}$  tests and dynamic tear (DT) tests. Both tests demonstrate a dramatic upsweep in fracture energy representing the ductile-brittle transition range, thereby demonstrating that rising ductility with increasing temperature overcomes the tendency of the high mechanical constraint in heavy section to suppress significantly the inherent ductility of the steel as had been suggested by certain theoretical studies.

9. Static linear elastic fracture mechanics tests which seek to define a critical stress intensity value at an artificial sharp flaw in a specimen of given dimensions were applied using progressively larger specimens (1 to 12 in.). The result was a curve sweeping rapidly up at about the NDT temperature and lying about 60 to 70 F below the rapid upsweep defined by the dynamic tear test results conducted using 3, 6, and 12-in. specimens. Thus, the upper temperature for valid  $K_{Ic}$  tests (using full thickness tests) lies no higher than the temperature of NDT +40 to 60 F which normally corresponds to the upper-middle region of the transition as defined by the Charpy V-notch curve.

10. Several important observations are noted from the dynamic-tear test results:

(a) With increasing specimen size a progressively sharper transition curve is defined, reflecting increased mechanical constraint in the larger specimens.

(b) DT specimens of 3, 6, and 12 in. thickness all define similar transition curves which rise sharply only after a temperature of approximately 120 F above the NDT temperature. Thus, a thickness effect (elevation of the transition temperature of about 70 F) is noted. This is most important since it demonstrates the need for accepting a criterion based upon the Charpy specimens tests which will assure vessel ductility in spite of this thickness induced elevation of the transition range. Criteria are now being considered by the USAEC to provide such assurance under any projectable vessel condition based upon surveillance data. New criteria will depend upon the ductility inherent in steels at a point above about half the energy at the midtransition point or a related dynamic fracture toughness value for describing full shear (shelf level) fracture resistance.

(c) The DT test results generally portray a transition which is higher than that provided by the Charpy V-notch test results, thereby suggesting that a somewhat more conservative basis is provided by the former. How-

ever, the new criteria mentioned in (b) will give consideration to a more conservative and quantitative interpretation of Charpy V-notch data for defining notch ductility of vessel steels.

(d) Gross plastic loading is required to fracture DT specimens at temperatures representing the ductile shelf of the transition curve. An energy of 760,000 ft·lb (full machine capacity) was absorbed by an A533-B steel specimen having an unfractured cross section of about 12 by 12 in. It is estimated that, at the test temperature (215 F), energy of about 1,000,000 ft·lb would have been required for full fracture.

(e) These results suggest that reasonable assurance against vessel failure can be obtained by following the dynamic tear results and a related conservative application of Charpy V-notch surveillance data. Similarly, projection of critical flaw sizes for failure at the temperatures involved in reactor vessel applications requires flaws measured in feet rather than inches.

(f) With severe irradiation (not expected under normal reactor conditions), however, the possibility of low energy tearing overshadows these transition temperature aspects. In this regard, properties of welds and of plates in the weak orientation (transverse to primary rolling direction) are the most critical. Data for this orientation must be developed to provide knowledge which will permit action which obviates potential consequences of low energy tearing.

## CHAPTER 2—Neutron Embrittlement of Pressure Vessel Steels—A Brief Review

### Introduction

This chapter was written as background for specific reactor vessel surveillance results to help explain, using experimental results, why radiation damage surveillance is needed.

Studies have been and are being conducted to assess the effects of high energy nuclear radiation upon the properties of reactor vessel steels under conditions which simulate nuclear service. Selected highlights of these program results are outlined.

### Highlights

1. Neutron embrittlement of vessel steels varies from very small to very large depending upon a number of factors which can be classified as

environmental or materials related. The most significant environmental factors are the exposure temperature and neutron fluence, and the most significant materials factors are the composition and microstructure of the specific vessel steel of interest.

2. Neutron fluence or exposure level for steel specimens exposed at a given temperature determines how much embrittlement will be sustained. Embrittlement is highly nonlinear with very rapid changes to about 1 to  $2 \times 10^{19}$  neutron (n)/cm<sup>2</sup> > 1 MeV with progressive deceleration of change to about 7 to  $8 \times 10^{19}$  where there are indications of damage saturation. With elevated temperature irradiation ( $\geq 550$  F) this pattern occurs at lower fluences.

3. Studies conducted on one heat of A302-B steel in different types of reactors have shown that the distribution of neutrons by energy (neutron spectrum) may affect significantly the embrittlement results giving more embrittlement in some cases and less in others than would be expected by assuming a spectrum and conducting the usual activation dosimetry. Further, in certain environments, significant errors in judging expected embrittlement have resulted from the conventional use of fluences in terms of neutrons >1 MeV. Thus, it is suggested that the vessel wall spectrum be computed and considered to lower energy levels (for example, 0.5 MeV) for defining damage per unit flux at the reactor vessel. (Advancements in neutron dosimetry and spectral analysis, which are reviewed in Chapter 5, can be expected to continue with the ultimate result that new standards for defining the damaging neutron environment will result).

4. Irradiation at higher temperatures results in significantly reduced neutron induced change as a result of thermal annealing processes which become quite evident and progressively more advantageous as the irradiation temperature is raised above about 450 F.

5. Advantage may be taken of the potential for "annealing-out" neutron damage through the elevation of the temperature above the normal reactor vessel operating or exposure temperature. The degree of thermal correction or annealing achieved depends largely upon the same factors which influence embrittlement plus the conditions of annealing; that is, annealing is most effective if carried out much above the steel exposure temperature for an extended period. Further, annealing effectiveness appears to be related directly to the ease with which the steel is embrittled. For example, in steels showing a high propensity for neutron embrittlement, annealing is highly effective and vice versa. Similarly, steels irradiated at low temperatures respond more favorably to annealing than steels irradiated at elevated temperature even if the same temperature increment (say 150 F) above the exposure temperature is applied for correction in both cases.

6. Specimens of pressure vessel steels have been irradiated in test reactors under applied loads to simulate reactor service. Results of these

relatively short term treatments have indicated no significant effect of applied stress upon the radiation induced change. Longer term tests of this factor are underway with fracture mechanics specimens exposed in vessel surveillance programs under static loads.

7. Very large differences in irradiation sensitivity have been observed between different steels, between heats of the same steel, and between forms or conditions (as weld to plate) of the same type of steel. The magnitude of these differences has been so great in some comparisons as to overshadow the effects due to the normal environmental variations. Variations in embrittlement sensitivity between steels have ranged as high as an order of magnitude as has the variation between weld and plate. The more usual situation, however, involves a factor of 2 to 3 difference in sensitivity from best to worst.

8. The most significant factor in radiation embrittlement sensitivity is composition though microstructure has been shown to affect the level of change as well. The observation of low and high sensitivity to radiation embrittlement in commercial steels has led to studies of special laboratory heats in which copper, phosphorus, and vanadium have been identified as critical to the level of sustained embrittlement.

9. The identification of critical elements in radiation embrittlement has been verified empirically by studies of composition and embrittlement in many steels and weldments and recently through special production plates and welds in which specifications were applied to minimize the undesirable elements. Control of copper (to less than about 0.080 percent) and phosphorus (to less than about 0.010 percent), and general minimization of other residual elements (such as arsenic, lead, antimony, tin, sulfur, etc.) provides assurance of low radiation embrittlement sensitivity of reactor pressure vessel steels.

10. Studies of radiation embrittlement sensitivity, while incomplete, have progressed to the point where it appears that carefully applied composition specifications can assure routine production of radiation insensitive pressure vessels. Further, these studies have provided important guidelines for projecting in advance the level of radiation embrittlement which may be expected in a particular reactor vessel.

11. Radiation effects significant changes in tensile properties, increasing the yield and tensile strength and reducing the elongation (total and uniform), the strain hardening exponent, the TS/YS ratio, and the reduction of area. At the usual reactor operating temperatures ( $>500$  F), however, these changes do not appear to be significantly detrimental at neutron fluences below about  $3$  to  $5 \times 10^{19}$ .

12. The fatigue strength of vessel steels do not appear to be affected significantly by radiation. However, the important parameter of fatigue crack propagation rates in nuclear versus nonnuclear environments is yet to be studied.

These significant experimental aspects of radiation effects on pressure vessel steels provide a strong basis for meaningful comparison and correlation of data from individual reactor vessel surveillance programs.

## CHAPTER 3—Army Reactor Vessel Surveillance

### Introduction

This chapter was written primarily because of the advanced exposure condition in the compact Army reactors and the accelerated embrittlement experience which some day may be directly applicable to the larger commercial nuclear power plants. Of special interest is the in-place annealing of the SM-1A reactor vessel and the postservice testing of the PM-2A reactor vessel. An important consideration is the fact that these two Army reactors were made up of forged rings, thereby eliminating welds in the highest flux regions of the vessel.

### Highlights

1. The compact Army reactors provide early accumulation of very significant, if not severe, nuclear exposure levels; thus, they offer advance knowledge of conditions which may eventually accrue to the larger reactor vessels.
2. The SM-1A reactor vessel surveillance was limited by space to the placement of dosimeters along the wall and specimens above the core. Supplementary experiments using test reactors, however, provided the needed background on the radiation embrittlement sensitivity of this vessel steel and, with the dosimetry, provided the basis for later application of the first in-place annealing (to correct radiation damage) of any reactor vessel. The SM-1A vessel steel, ASTM Type A350-LF1 (modified with added nickel), proved to be very sensitive to radiation at the temperature of reactor vessel operation ( $\sim 430$  F) but also proved very responsive to thermal correction by annealing at a temperature of about 140 F above the operating level.
3. The SM-1A was annealed to correct embrittlement after the vessel reached a neutron fluence of  $1.6 \times 10^{19}$  n/cm<sup>2</sup> (assumed fission spectrum) in 40.4-MW years of operation and caused an increase in transition temperature ( $\Delta T$ ) of about 260 F. This  $\Delta T$  was 80 F short of the Army imposed limit of 340 F. The Army's conservative basis for projecting the annealing point along with superimposed AEC conservatism resulted in

annealing before limits based upon experimental data would have required. Annealing recovery was equal to about 70 percent of radiation induced  $\Delta T$  or  $\Delta NDT$  after annealing for one week at a temperature of 572 F. This low annealing temperature was made possible by the very favorable response to heat treatment at about 140 F above the operating temperature. This permitted annealing using nuclear heat (low operating power with throttled coolant flow) which allowed the needed rise in temperature with a relatively small related rise in operating pressure.

4. It was required that neutron spectrum and fluence calculations be performed and projected on the basis of  $n/cm^2 > 0.5$  MeV. This exercise indicated the use of  $n/cm^2 > 0.5$  MeV to be the more favorable treatment for the SM-1A reactor because the relative abundance of neutrons having energies greater than 0.5 MeV was lower in the SM-1A spectrum than in the Watt fission spectrum.

5. In spite of the success of the SM-1A vessel annealing sequence as verified by experiment and by specimens simulating the irradiation and annealing which were removed from the SM-1A after annealing, the high vessel wall flux requires planning for subsequent repeat annealing cycles. Experiments have been conducted which validate this plan. The absence of surveillance specimens at the vessel wall complicates planning and dictates continued conservatism toward vessel embrittlement however.

6. A similar sister reactor to the SM-1A, the SM-1, has not accumulated the high vessel exposures of the SM-1A because of a smaller core (yielding a lower vessel wall flux) and reduced time at power because of its training function. Nevertheless, concern for the embrittlement to a point requiring annealing has resulted in analytical evaluations of the vessel exposure flux, fluence, and spectrum in an effort to project the vessel condition in future years.

7. The PM-2A reactor vessel was tested nondestructively and destructively (as a single specimen and as a source of materials for multiple laboratory specimens) after the reactor had served its function at Camp Century, Greenland, in the early 1960's.

8. Nondestructive tests of the PM-2A included ultrasonic examination which showed no significant included flaws in the vessel, out-of-roundness tests which showed none which would significantly accentuate stresses in planned destructive tests, and gamma ray spectrometry which indicated (unfortunately erroneously on the high side) the neutron fluences sustained at various positions around the vessel.

9. Destructive testing of the PM-2A vessel was to show how fracture mechanics would predict the flaw, stress, and temperature conditions for failure. Predictions proved to be in error, but the test sequences were complicated by the errors in projected fluence, the gradient in exposure through the vessel wall, the significant portion of ductile Type 304 stainless steel lining in the remaining vessel ligament at failure, the complex cyclic



test procedures which involved loading and unloading at various temperatures, fatigue cycling to sharpen the machined flaws, and, finally, stress corrosion to sharpen the machined flaw. In spite of all these difficulties, the full vessel test suggested a very reliable component in that failure required higher stresses than had been predicted.

10. Laboratory tests of the failed PM-2A vessel showed:

(a) Increases in transition temperature in line with experimental data projections for the steel and its service condition.

(b) Tensile ductility not greatly affected by the service environment ( $\sim 1 \times 10^{19}$  n/cm<sup>2</sup> > 1 MeV at 510 F).

(c) Neutron fluences across the vessel wall thickness presenting a linear relationship on a semilog plot which should aid similar projections for thicker vessels.

(d) Fracture mechanics data providing relatively good agreement with the failure conditions though the possibilities for projection to temperatures having meaning in an operational sense were shown to be quite limited.

11. In general, the small Army reactors lead their larger commercial counterparts in the problems which may be anticipated due to neutron radiation damage of the vessel.

## CHAPTER 4—Commercial Reactor Vessel Surveillance

### Introduction

This is a key chapter in that it meets the primary objectives of this report to describe and analyze results of commercial reactor vessel surveillance programs. Since Chapter 6 provides a critique of each individual reactor program, emphasis in this synopsis is upon description, interim results, and relative status of programs.

### Highlights

1. Only five commercial reactors have operated long enough with surveillance programs to have produced significant results. These include the General Electric constructed reactors: Big Rock Point, Dresden 1, and Humboldt Bay; the Westinghouse constructed reactor-Yankee-Rowe; and the Allis-Chalmers constructed reactor-Elk River. In spite of the limited number involved, these reactors as pioneers in surveillance (and in many other aspects) provide valuable data for viewing the future; hence, a detailed analysis is of special value.

2. Information describing individual surveillance programs of 25 com-

mercial reactors are tabulated along with vessel material, planned life, anticipated maximum vessel fluence, assumed plant factor, and projected maximum vessel NDT.

3. A comparison of the range of anticipated vessel fluences shows a high of  $6.0 \times 10^{19}$  n/cm<sup>2</sup> > 1 MeV for Shippingport and a low of  $1 \times 10^{17}$  for Vermont-Yankee. A similar extreme of projected maximum NDT values ranges from 438 to 40 F for these two reactor vessels.

4. Projected fluence values of some of the earlier reactors have been in error on the low side, however. For example, calculated values for Big Rock Point were about  $3 \times 10^{18}$  > 1 MeV while measured values (discussed in detail in Chapter 6) are  $3 \times 10^{19}$  > MeV. Similar, though smaller, variations have been determined for other plants including Dresden 1 and Yankee-Rowe. Related errors in maximum NDT temperature are predicted.

5. Surveillance program plans reported to date are largely those of General Electric and Westinghouse, which follow the ASTM Recommended Practice, E 185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors), in general, but are expanded over this in several specific instances. Standard General Electric and Westinghouse programs are described. Variations from the standard are prevalent in General Electric's case as those vessels having electroslog welds receive special attention and those which have anticipated fluences  $< 5 \times 10^{17}$  receive minimum treatments unless an electroslog weld is present in the belt region of the vessel. Standardization is developing rapidly, but many early plants contained no weld or weld heat affected zone specimens (a potentially serious omission).

6. Surveillance programs contain predominantly Charpy V-notch and tension specimens (in widely varying total numbers), but some contain fatigue specimens (Elk River) and some fracture mechanics specimens (certain Westinghouse and Babcock and Wilcox produced reactors). While the type of specimen used is most important, the inclusion of specimens truly representing the vessel materials is even more important.

7. Three surveillance capsule positions normally are used: just inside the vessel wall, an accelerated (usually inside thermal shield) exposure position, and a thermal control (out of significant flux) position. The latter two are not required but provide guidance to aid the projection of vessel wall surveillance results to later years with confidence. Capsules usually contain, in addition to vessel steel specimens, reference (correlation monitor) specimens of a well known steel heat, low melting point or eutectic alloys to provide guidance as to temperatures attained plus neutron dosimeters to aid in determining the flux and fluence attained in the capsule exposure period. Dosimeters are also valuable for projecting NDT temperatures in future years provided the assumption of no significant change in core loading or power density is valid.

8. The Big Rock Point reactor surveillance program was well planned; however, the determination of the actual vessel flux was an order of magnitude greater than calculated and left the program inadequate for the full life of the plant. It also lacked specimens of a well-known reference steel. During 1969 the last surveillance specimens were removed and evaluated. Fortunately, the availability of specimens in accelerated locations, and the presence of plate, weld, and weld heat affected zone (HAZ) specimens which could be removed at several year intervals provided the basis for reasonably good lifetime projections. This program shows the importance of having extra surveillance specimens available plus some means for adding them. The unexpectedly high flux made it necessary that specimens be removed much ahead of the planned removal, thereby depleting in only 6 years what was intended to cover approximately 30.

9. The Yankee-Rowe vessel surveillance program was rather limited containing ten Charpy V-notch and tension specimen capsules, eight of which were in accelerated and two in vessel wall irradiation locations. The program was further limited by the loss of four accelerated capsules and one vessel wall capsule due to failure of the capsule anchor assembly. All four accelerated capsules were removed after only one core life (Core II) and the single wall capsule was removed after three cores (Cores II, III, and IV). The specimens were prepared only from vessel plate (no weldment specimens), and the surveillance plate was separately heat treated so its properties did not meet exactly those of the vessel. In spite of these several limitations, careful dosimetry analysis and full evaluation of the well-known reference steel provided the key to meaningful projection of later life conditions in the Yankee vessel. Many lessons are inherent in the results of the Yankee-Rowe surveillance.

10. The Dresden 1 program was very large and though a similar underestimation (factor of four) of flux and fluence occurred, the availability of many specimens and a means for adding more if needed proved very farsighted. Thus, this second oldest U.S. nuclear power plant has an on-going surveillance program which, with careful analysis (for specimens added after the initial startup), can and will provide a continuing and acceptable vessel radiation damage surveillance program. One weakness was the lack of a well-known reference steel. Fluences projected from the early measured dosimetry results made it important to continue surveillance even if on a delayed insertion basis. The reactor owner has been especially diligent in seeking full answers to the radiation damage question.

11. The Elk River reactor vessel surveillance program is large (40 capsules) but suffers the rather severe handicap of not containing specimens of the actual vessel material. Instead, it includes three heats of A302-B steel (neither truly representative), two of A105, and one of A212-B. In addition, certain transition welds, A105 to Type 304 stainless, were exposed in Elk River. Another early limitation of this program was the lack of

definitive dosimetry; this has been corrected in recent efforts. Also, difficulty in exact measurement of temperature has hampered analysis of results in the Elk River case.

A unique aspect of the Elk River surveillance was the inclusion of fatigue specimens. This was instigated by the observation of small cracks in the stainless steel vessel overlay. Fatigue results have been obtained.

12. The Humboldt Bay vessel surveillance is in a relatively infant state with accelerated irradiation position specimens having been tested after two years of operation. Many capsules containing specimens from plate, weld, HAZ, and from the Dresden steel provide an ample basis for good surveillance; results to date are too limited for really significant conclusions.

## CHAPTER 5—Neutron Dosimetry and Spectrum

### Introduction

This chapter provides a description of the neutron dosimetry and spectrum analysis techniques necessary to meaningful reactor vessel surveillance. Without good neutron dosimetry and spectrum analysis misleading conclusions may result even with the best materials surveillance. Basic definitions and accepted techniques are described.

### Highlights

1. As early surveillance data are used to project properties changes in the vessel in later life, success depends upon knowing the number or density and the energies of neutrons which pass through a given cross section of the steel vessel; this is neutron dosimetry and spectrum analysis.

2. Neutron dosimetry is accomplished through the placement of wires or foils of selected metals or alloys for which the action of neutrons in causing activation and secondary radiation emissions is well known. Such materials, known as dosimeters, monitors, or detectors, react to neutron bombardment by emitting radiation of a particular type; the intensity coupled with the exposure period provides an indirect measure of neutrons which have passed through the detector.

3. Since single detectors are responsive to neutrons of a particular energy range, proper dosimetry depends upon knowing the response (activation) of a group of detectors so as to assure coverage of the spectrum at the surveillance location. Thus, the best approach involves use of several well-known detectors which are activated by neutrons of widely different

energy levels. No such group of detectors remotely approaching an ideal response has yet been found for long-term applications such as reactor vessel surveillance. A listing of the preferred detectors and their pertinent nuclear characteristics is provided, however. In addition, it is necessary that a computation of the spectrum at the vessel wall be used as a basis for selecting the proper activation constants.

4. Computation of the spectrum at the location of surveillance specimens is essential as it provides the basis for use of the proper dosimeter activation cross-section value, permits a reasonable evaluation of the contribution to damage by various components of the spectrum (thermal versus intermediate versus high energy, etc.), and provides a good test of the reported fluence in relation to observed embrittlement.

5. Recent developments in computer technology provide sophisticated techniques for computing the neutron energy spectrum at any reactor location and even alternate approaches for check evaluations. Developments in neutron dosimetry have provided a series of detectors which cover the general range of energies of interest for steel irradiation damage studies and thereby permit a dosimetry validated spectrum computation. Use of a damage function (cross section for damage production) for a given spectrum further aids the analysis. Damage function development depends upon a computed spectrum augmented by dosimetry and, where possible, by changes in properties in a particular steel. Computer analysis of such data represents the latest and best approach to radiation damage analysis.

6. A more complex spectrum analysis was applied and is reviewed for the SM-1A reactor surveillance. Due to lack of space in the SM-1A reactor, it was necessary to expose representative materials in a test reactor, the low intensity test reactor, to simulate exposure of the SM-1A vessel and then to calculate and adjust for spectrum differences. This exercise provided valuable experience and added confidence in the use of computed spectra in conjunction with measured changes in properties and good neutron dosimetry at the reactor positions of interest.

## CHAPTER 6—Surveillance Programs Critique and Recommendations

### Introduction

This chapter outlines conclusions as well as critiques of results from various reactors which have thus far produced surveillance data. Upon

this background, important features or factors to be considered in producing a comprehensive vessel surveillance program are outlined. Since the Army reactor programs have been reviewed in the synopsis of Chapter 3, no further highlight discussions are provided. A brief critique is provided for each of the five commercial reactors, however.

### Highlights

1. Some overriding general comments about what a good surveillance program should contain include the following:

(a) It should provide for the irradiation of steel (plate, weld, and HAZ) from the actual pressure vessel.

(b) It should utilize an irradiation location and condition which duplicates as closely as possible the physical and nuclear conditions of the vessel.

2. Other important factors include:

(a) The assurance that specimens are completed in accordance with pertinent standards.

(b) That these specimens are tested on carefully calibrated test apparatus.

(c) That the nuclear constants used in neutron dosimetry and spectral analysis are the best and latest available.

(d) That a complete record of the materials history is recorded and retained (along with archive segments of the actual vessel steel) for future analysis.

3. Critique of Big Rock Point Reactor Vessel Surveillance

(a) Materials were of the actual vessel and weldments though certain heat treatments were simulation of the vessel.

(b) The number of vessel wall specimens, thought to be adequate for the projected lifetime fluence of  $3 \times 10^{18}$  n/cm<sup>2</sup> > 1 MeV, proved inadequate for good lifetime coverage in view of the order of magnitude higher measured fluence ( $3 \times 10^{19}$  projected for vessel lifetime).

(c) The mode of specimen notching in some cases was nonstandard (parallel rather than perpendicular to the surface) and missed the fusion zone in some HAZ specimens.

(d) Nevertheless, the program was complete enough to permit projection of a lifetime fluence and NDT ( $\sim 200$  F) in the weld metal (the worst case), which for the rather high ( $\sim 575$  F) operating temperature should not present more than an operational control question in later years of reactor life unless core changes produce higher fluences or for some reason the operating temperature is reduced significantly.

4. Critique of Yankee-Rowe Reactor Vessel Surveillance

(a) The strongest feature of this effort was the inclusion of a well known A302-B reference steel; the weakest factor was the mode of capsule attachment which resulted in the loss of half of the ten capsules and a questionable positioning of the remainder.

(b) This program lacked weld specimens and the plate specimens were from a separately heat treated plate.

(c) The lack of neutron dosimeters and temperature monitors, especially the latter, also weakened the program.

(d) In spite of the noted weaknesses, however, the availability of the well characterized reference steel saved the effort, making possible reasonably good vessel embrittlement projections. These projections (at most an NDT of 305 F), assuming exposure conditions to continue unchanged, or if changed, changed to minimize radiation damage, suggest that nothing more than a conservative operating practice need be applied. Doubt about the weld sensitivity and the operating temperature, nevertheless, requires conservative interpretation and operational practice.

(e) It should be noted in favor of this effort that all facets were well documented and good records were maintained, though the availability of surplus "archive" specimens or material was very limited.

### 3. Critique of Dresden 1 Reactor Vessel Surveillance

(a) This pioneer program was initially lacking in weldment specimens and in temperature monitors, but this was corrected and the Dresden became the most carefully monitored of the early power reactors.

(b) One aspect of specimen preparation, however, creates some doubt for the interpretation of data. Charpy specimens were taken from top and bottom layers of the vessel steel. These areas of quenched and tempered steels usually are superior in strength and ductility to the major section of the vessel. Further, specimens are notched contrary to standard practice, that is, parallel rather than perpendicular to the plate surface direction. Nevertheless, any significant departure from expected results would be determined by the vast program. (Dresden has been used somewhat as a test reactor for evaluating a number of vessel steels in addition to that of the Dresden vessel).

(c) Calculated projections of neutron flux levels were found to be low by a factor of four when measurements were made. Even the higher measured fluxes are expected to produce an NDT of only about 265 F at most. Thus, while requiring care in devising suitable operating practice for avoiding approaching NDT (or  $NDT + 60$  F) while the vessel is significantly stressed, nothing more than a careful reactor operating plan appears necessary provided there are no changes to increase fluence or decrease temperature.

### 4. Critique of Elk River Reactor Vessel Surveillance

(a) The Elk River program violates the two major requirements of vessel surveillance: (1) the use of representative materials and (2) the placement of specimens at the vessel wall.

(b) The large scope of the program helps somewhat to cover these major weaknesses, however, in that a number of steels and welds along

with good neutron measurement (of the wall in later phases of capsule insertion) make up the ongoing surveillance effort.

(c) Fatigue specimens exposed in the Elk River reactor will test this property on a prepost test basis for materials simulating plate and welds in the vessel.

(d) Projections of NDT at or near planned end of life (20 years) of 420 F ( $\Delta T$  of 320 over initial 100 F) suggest the need for interim operational controls and continuing evaluation of the vessel condition and the possibility for correcting embrittlement by annealing if the plant is to operate anywhere near the planned lifetime.

#### 5. Critique of Humboldt Bay Reactor Vessel Surveillance

(a) The materials and surveillance locations meet the primary criteria in that representative plate, weld, and HAZ are located at positions to test the effects in the actual environment.

(b) The specimens, however, are taken from the plate surface, thereby possibly giving misleadingly high initial properties.

(c) Early accelerated exposure data suggest no materials sensitive to radiation, though relatively low Charpy V-notch ductile "shelf" results for the HAZ specimens suggest a need for continuing examination of this factor.

(d) No dosimetry data are yet available from the vessel wall; however, comparison with the Big Rock Point reactor layout indicates no particular reason for concern for the vessel in its 40-year life. This tentative conclusion must be tempered by the results of future surveillance tests.