8.
EFFECTS OF IRRADIATION ON MECHANICAL PROPERTIES

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1 INTRODUCTION

This chapter deals with the effects of neutron irradiation on the mechanical properties of reactor pressure vessel steels. Neutron exposure is a growing concern for operating reactors because of the change in the strength and fracture toughness properties of the vessel wall materials. This change in properties is known as radiation embrittlement, and the long-term effects of neutron exposure could challenge the structural integrity of reactor pressure vessels. Thus, it is important to monitor and control the amount of radiation damage since the success of current industry efforts to assure full life time operation of nuclear plants depends importantly upon the long-term integrity of reactor pressure vessels.

In order to control the effects of irradiation, it is essential to have a good understanding of the mechanisms causing embrittlement in the plates, welds, and forgings that are used in fabricating reactor pressure vessels. Early vessels were constructed from moderate strength carbon steels. Newer vessels were made from manganese-molybdenum steels with the addition of nickel to increase strength. Unfortunately, all steels have the same basic response to neutron irradiation, that is, radiation hardens the steel causing an increase in strength which, in turn, reduces the fracture toughness. This trend is common for ferritic steels that tend to undergo a classic ductile-to-brittle transition over a range of temperatures. Thus, it is appropriate to establish general trends in mechanical properties in describing the radiation embrittlement effects of vessel steels. Once the basic features of these effects are understood, differences in radiation sensitivity as the result of variations in chemistry and microstructure, flux and fluence, and irradiation temperature are examined.

Our knowledge of the effects of neutron irradiation damage comes from studies of unirradiated and irradiated materials in power and test reactors. Mechanical test specimens in these reactors have provided a significant amount of data for determining the relevant damage parameters as well as mechanical properties trends. Most of the data currently available are from Charpy V-notch and tension test specimens which are now used to measure the change in properties due to irradiation exposure. These data have been utilized to develop trend curve prediction methods for the embrittlement process which include the effects of variables such as copper, nickel, phosphorus, and neutron fluence in a given temperature range.

While not all reactor vessel steel types are covered by this chapter the majority of those in light water reactors are covered by emphasis on the types of steels and vessels used in the USA and the former USSR. Variations of these steels are used in Europe and Japan, however the extent of these variations are such that the conclusions of this chapter are still valid since the nature of the steels used throughout the world for reactor vessels are similar to those discussed here.

2 NEUTRON IRRADIATION DAMAGE

2.1 How does irradiation exposure cause damage?

Research to understand the mechanisms of neutron irradiation embrittlement in vessels fabricated in the United States has resulted in improved radiation damage trend curves as given in USNRC Regulatory Guide 1.99, Revision 2 [1]. Similarly, trend curves have been developed in the former USSR [2,3] for assessing the radiation damage in nuclear vessels. The basic mechanisms of embrittlement can be described as follows:

1. Embrittlement is primarily due to microstructural changes which result in irradiation-induced increases in yield strength which can be quantitatively related to changes in the Charpy V-notch curve, as well as hardness.
2. Yield strength increase-related changes in fracture properties are due to irradiation-induced fine scale microstructures (about 1 nm) which act as obstacles to dislocation motion.
3. The most likely resultant microstructures are precipitates, vacancy clusters (micro-voids), and interstitial clusters (dislocation loops). Copper-rich precipitates, possibly alloyed with elements such as nickel (or combined with vacancies), are believed to be a dominant hardening (strengthening) feature in sensitive steels containing significant concentrations of these impurity elements. Other possible types of irradiation-induced or

enhanced precipitates are phosphides and small carbides.

4. Other important effects of irradiation are enhanced diffusion rates and defect clustering.

5. These microstructural changes are kinetic phenomena and are functions of neutron exposure and temperature as well as composition and initial steel microstructure. In general, the changes in the microstructure can be understood from basic principles of alloy thermodynamics and precipitation kinetics, coupled with rate theories of radiation damage.

A noticeably greater difference in radiation sensitivity has been observed in weld materials as compared to plates and forgings for approximately the same chemistry and neutron exposure. While trace amounts of copper are the primary contributor to irradiation embrittlement in USA steels, it has also been discovered that nickel in the presence of copper tends to increase irradiation sensitivity, particularly in the case of welds. This observation has led to physically-based models of the embrittlement process which are slightly different for welds and base materials. Test data from the former USSR and European plants exhibit other effects due to the presence of higher concentrations of elements such as phosphorous and vanadium. These effects are discussed separately.

2.2 What properties are affected?

Embrittlement due to radiation hardening can be quantitatively related to changes in the yield strength following neutron exposure.

This hardening effect can be measured in tension tests and can also be inferred (but not directly quantified) using hardness (or micro-hardness) test techniques. The embrittling effect of irradiation on reactor vessel steels is a complex process, and numerous theoretical studies, testing programs, and microstructural evaluations have been performed to better understand the structure-property correlation. The changes in microstructure described above have been confirmed using atom probe/field ion microscopy (APFIM), small angle neutron scattering (SANS), and high resolution transmission electron microscopy (TEM). These studies have revealed radiation-induced precipitation or clustering of copper, nickel, and phosphorous on a very fine scale. As a result, the more recent models of radiation embrittlement include a dispersion strengthening component based on the age-hardening model developed by Russell and Brown [4] which has been modified to account for the increased diffusion rates of vacancies due to neutron bombardment. Some researchers now believe that age-hardening is the dominant process of radiation embrittlement in vessel steels [5, 6].

The age-hardening effect can be measured by changes in the mechanical properties. In fact, good correlations have been observed between tension tests and Vickers microhardness test data on reactor vessel welds. The changes in ductility and fracture toughness are important measures of the material's ability to resist fast fracture, but these properties are difficult to measure directly. While hardness is not the critical material property, it is a convenient means of monitoring embrittlement, and a method to correlate with other key changes in mechanical properties. Our ability to model the embrittlement process has led to an improved understanding of how steels are damaged by neutron irradiation, but as yet we cannot predict changes in fracture toughness solely from a mechanistic standpoint. These property changes must be measured using more expensive larger size specimens such as used in Charpy V-notch tests or even larger ones such as those used in compact fracture toughness tests.

2.3 How is radiation damage measured?

The fracture toughness of carbon steels is very temperature-dependent as exhibited by a marked change in fracture behavior from brittle (cleavage) at low temperature to ductile(tearing) at higher temperatures. Characteristic changes in fracture toughness from low to high values with increasing temperature are exhibited. This transition behavior is most apparent with a test specimen containing a sharp notch or crack. The most common specimens used to monitor the effects of radiation embrittlement are the Charpy V-notch (ASTM E370-88a) and the drop weight tests (ASTM E208-87a). The Charpy test measures the energy required to produce fracture of a notched three-point bend specimen. Charpy data as a function of temperature can be obtained for unirradiated and irradiated pressure vessel steel materials

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embrittlement effects are noted as an upward shift of the Charpy curve in temperature and a reduction of the maximum upper shelf energy at the higher temperatures. In the U.S., the extent of radiation damage is determined by the shift in the 30 ft-lb (41 Joule) Charpy energy level frequently referred to as the transition temperature shift. This shift is important as it relates to the change in fracture toughness (K_{IC}) of the steel in the irradiated condition which exhibits the same kind of transition temperature shift [8] The change in the toughness curve due to irradiation must be taken into account when establishing safe operating limits to prevent brittle fracture of the vessel. For obvious reasons of reactor vessel integrity, the fracture toughness of the material is the most important mechanical property. Unfortunately, it is difficult to determine vessel toughness directly since "valid" fracture toughness measurements require large specimens at the temperatures of vessel operation.

It is valuable, even essential, to correlate changes in fracture toughness with other, more easily measured, mechanical properties. For example, Charpy specimens are routinely used in surveillance programs to monitor the effects of embrittlement on mechanical properties with the assumption that the change in fracture toughness transition temperature is equivalent to the change in Charpy temperature shift. Based on the work of Cottrell [9], it was recognized that the transition temperature shift as the result of radiation damage should be directly proportional to the change in yield strength. In fact, a robust correlation between yield strength increase ($\Delta\sigma_y$) and transition temperature shift (ΔT) has been observed [9] which follows the simple form:

$$\Delta T = C\sigma_y$$

For U.S. reactor vessel welds, the correlation factor C was determined to be 0.65°C/MPa based upon the existing U.S. surveillance database. A similar analysis for plate materials yielded C = 0.5°C/MPa. This correlation provides evidence that changes in hardness can also be related to transition temperature shift. Caution must be exercised whenever using such correlations since they may be only applicable to specific environmental parameters (e.g., temperature and fluence rate) and specific material types (e.g., welds versus base metal or even specific material types within welds or base metals). This correlation process has been utilized in the former USSR to measure recovery in toughness properties following reactor vessel thermal annealing by performing *in situ* hardness measurements before and after the anneal.

Thus, it is apparent that small-scale testing techniques can be used in conjunction with larger specimens to monitor the effects of embrittlement on reactor vessel steels so that the contributing factors to irradiation damage can be accurately assessed.

2.4 What are the key factors affecting reactor vessel embrittlement?

The results from the many test programs and mechanistic modelling studies identify three key factor which affect the radiation damage of reactor pressure vessel steels. These factors include the neutron environment (fluence, fluence rate, fluence energy spectrum) and the irradiation temperature. The other key factor is the material itself which can be different in chemical elements, microstructure, and processing history. The time integrated neutron flux exposure results in continuing loss in toughness and other changes in mechanical properties for most pressure vessel steels until, in some cases, a saturation level is reached. The degree of radiation damage depends largely on the particular materials involved, especially their chemistry and microstructure, plus the specific neutron environment. The effects of these factors will be described in the next part of this chapter with emphasis on how mechanical properties change based upon the neutron environment and materials. These differences will be highlighted based upon results obtained primarily in the USA and the USSR. The design of reactor vessels along with the size and placement of the core are crucial to radiation effects.

3 USA VESSEL DESIGN AND FABRICATION

Essential to the understanding of the background within which vessel steel properties are changed by neutron exposure is a description of the design and fabrication of the vessel.

Consequently a summary of these features for both USA and former USSR reactors are provided.

3.1 USA Vessel Design

Most reactor vessels now in service in the USA were fabricated in the 1960s and 1970s. A general description of these vessels in outlined in Reference [10].

Table 1 presents typical sizes and weights of LWR pressure vessels for plants in excess of 1000 MW of electric power capacity. The boiling water reactor (BWR) vessel is much larger, and somewhat heavier, than the pressurized water reactor (PWR) vessel. The direct-cycle BWR vessel contains not only the reactor core, but also the steam separators and dryers, and the feedwater spargers. In addition, most domestic (USA) BWR vessels accommodate internal jet pumps, which are part of the coolant recirculation system. The BWR typically operates at 1000 psi (6.8 MPa) and <550°F (<290°C).

The PWR system uses external steam generation and separation, so the PWR vessel is much shorter than the BWR vessel. The two types of vessels are shown schematically in Figure 6 [11]. The weld locations in a typical PWR [10]. The PWR operates at a higher pressure, typically 2250 psi (15.2 MPa) and in a temperature range of 520-600°F (270-315°C). The PWR wall thickness is typically about 10 in. (250mm), while the BWR wall thickness is about 6 in. (150mm).

3.2 Reactor Vessel Fabrication in the USA

Similar techniques are used to fabricate BWR and PWR pressure vessels, and also PWR steam generators and pressurizers. The PWR vessel can be constructed entirely from forged rings, flanges, and nozzles. Alternatively, rolled and welded plate can be used for the shell courses, and forgings for flanges and nozzles. All BWR vessels have used the latter construction method, with shell courses of rolled and welded plate.

The reactor vessel manufacturer typically receives steel plate in the as-rolled and stress-relieved condition. The steel making and hot rolling of the plate is not discussed here. The first step in the fabrication sequence is to hot-form the plate into 120° steel segments. Hot forming is performed at a temperature of approximately 1650°F (900°C). The formed segment is austenitized at 1600°F (870°C) followed by a water quench. The quenching treatment is followed by a tempering treatment of 1250°F (675°C) for two hours. Three 120° segments are then welded together onto a shell course using a welding procedure to be described later. The shell courses are clad with an austenitic stainless steel weldment using either the multiple wire or strip cladding submerged arc process. Three shell courses make up the cylindrical portion of a typical PWR vessel

The upper (or nozzle) shell course is generally 20% to 50% thicker than the intermediate and lower shell courses. The additional thickness is required to reduce the stresses associated with nozzle penetrations. Six to eight nozzle forgings are welded into large diameter holes (60 in., or approximately 1.5 m) in the nozzle shell course. The vessel flange is welded to the top of the nozzle shell course, and the intermediate shell course is welded to the bottom of the nozzle shell course, forming one of two vessel assemblies.

The second vessel sub-assembly consists of the lower shell course and the bottom head. The final steps are to weld the upper and lower vessel subassemblies together and to stress-relieve the assembly.

A forged vessel is constructed in a similar fashion, except that the shell courses are one-piece ring forgings. This construction avoids the longitudinal beltline region welds.

It generally requires between three and five forged rings to construct a reactor vessel. The forging procedure for such a ring is described next. The ingot is cast and the upper and lower ends are discarded. The remainder of the ingot is repeatedly upset in a large hydraulic press until a disk approximately four to five feet thick is prepared. This disk is pierced, and the pierced disk is placed on a mandrel and continually worked in a large hydraulic press until the shell forging is formed. The finished forging is austenitized at 1650°F (900°C) for two hours. The ring is then quenched and tempered at 1275°F (690°C). As with the plate-formed vessel courses, the forged ring must be clad with austenitic stainless steel to prevent general corrosion.

3.3 Material Specification

Reactor vessel shell plate specifications have evolved since the beginning of the commercial nuclear power industry in the mid-1950s. Table 2 [10] lists the principal steel plates used in construction of nuclear pressure vessel components. The original steels, the A212 as well as the A302B, were in widespread use in the construction of fossil power plant components at that time. All of the steel plates are of the low-alloy ferritic variety. The "A" designation indicates the ASTM material specification. The "S" prefix indicates material acceptable by the ASME Boiler and Pressure Vessel Code for construction of power plant components. The A212 steel was used only in very early plants which are now decommissioned. The current equivalent of this carbon-manganese-silicon steel is SA 515 Grade 60. A much more widely used material, A302B, is a carbon-manganese-molybdenum steel that was used in the quenched and tempered condition. In the mid-1960s, with the size of the nuclear components increasing, greater hardenability was required. The addition of nickel in quantities between 0.4 and 0.7 wt% provided the necessary increased hardenability to achieve the desired mechanical properties. This steel was known initially as SA 302B Modified. Later, it became the present grade SA 533 Grade B Class 1, which is the most widely used material for construction of reactor pressure vessels and pressurizers in the USA.

In 1973, limits were imposed in the percentage of copper and phosphorus permissible for use in the so-called beltline region of reactor pressure vessels where the neutron flux is high. The reduction of copper and phosphorus minimized the embrittlement sensitivity of the steel.

The last entry in Table 2 is SA 533 Grade A which is used in the construction of the steam generator shell. This material is also used in the quenched and tempered condition, and is the current equivalent of the original A302 Grade B. Typical heat treatments of these materials are as follows: The material is austenitized at a nominal 1600°F - 1650°F (870°C - 900°C) followed by quenching, generally in an agitated water bath. The material is then tempered at 1220°F - 1250°F (660°C - 675°C) for two hours. After welding, the entire vessel is stress-relieved at 1150°F (620°C). Typical ranges on these heat treatment temperatures are ±25°F (±14°C).

The pressure vessel forging materials (Table 3) have also evolved since the mid-1950s. The earliest grade, A105 Class 2, was used in the normalized and tempered condition for flanges and nozzles. This simple carbon-manganese steel has been used to a very limited extent in vessels. A forging material of greater usage in the 1950s and 1960s was the SA 182 F1 Modified, used in the quenched and tempered condition. This manganese-molybdenum-nickel steel was used principally for flanges and nozzles. Another forging material in use at this time is a carbon-manganese-molybdenum steel, SA 336 F1, used in both the normalized and tempered, and the quenched and tempered conditions.

One of the production problems encountered with the SA508-2 forging material is the occurrence of small underclad cracks with certain cladding procedures. In the early 1970s, typical European practice was to apply cladding in a strip fashion using the submerged arc process. Small cracks on the order of several millimeters in depth occurred with great frequency in some of the heat-affected zones of strip-clad welds. It was eventually discovered that the presence of chromium in this forged material was the root cause of the cracking. Such underclad cracking has never been observed in the SA 533 Grade B Class 1 plate material or the submerged arc weld metal.

To eliminate underclad cracking, SA 508 Class 3 material is used in place of SA 508 Class 2. The Class 3 material is essentially the same as the earlier SA 182 F1 (Modified) specification. The industry appears to have come full circle with regard to forging materials. Hydrogen blistering is no longer a problem in SA 508 Class 3 material.

3.4 Welding Procedures in USA Reactors

Full thickness welding is required to assemble the shell courses, the nozzle forgings, the flange forgings, the top and the bottom heads, and any internal or external support pads. Table 4 lists typical welding techniques used in the construction. By far, the most frequently used technique is the automatic submerged arc welding procedure. The protective environment is provided by a granular flux, part of which is vaporized during welding. The weld wire has the required alloy content. The only heat treatment required is the normal stress relief, which reduces residual stresses and tempers the martensite found in the heat-affected zone. Automatic submerged arc welding is used wherever possible because it provides excellent mechanical properties and has a very high deposition rate relative to other welding techniques. A tandem wire technique may be used to increase the deposition rate.

Narrow gap submerged arc welding is a variant of the technique used primarily in the fabrication of circumferential or girth seam welds. The benefit of the narrow gap technique is reduced weld volume and fabrication time.

A frequently used manual welding technique is the shielded metal arc procedure, which is used for complex configurations, for repairs of base material, or possibly for areas of weld buildup. Although the deposition rate is low, the technique is extremely flexible and the weld metal has excellent mechanical properties.

The electroslag technique was used for full thickness welds in some of the earlier BWR pressure vessel and PWR steam generator shells. This automatic procedure provides extremely high deposition rates. Due to its large coarse cast microstructure, electroslag welds must be austenitized, quenched, and tempered similarly to the treatment for base metal. Unfortunately, electroslag welding requires such close dimensional tolerances on weld preparations that the technique is often not cost-effective.

All interior surfaces of the PWR vessel are clad with austenitic stainless steel to inhibit general corrosion and the buildup of radioactive crud. BWR vessels are clad below the steam-water interface. Three principal types of cladding processes are used. The shielded metal arc process is used wherever possible due to its high deposition rate. The process uses either multiple wires of strip electrodes of Type 308 or 309 stainless steel. In areas where an automatic process is not possible, shielded metal arc or gas tungsten arc welding is utilized.

3.5 Heat Treatment in USA Reactors

The heat treatment of the base material has been previously discussed. The postweld treatment is described here. Following any welding procedure, such as the through-thickness weldment and cladding procedure, or even a small repair, an interstage stress relief is required before dropping preheat. Generally, this requires exposing the welded component to 1130°F (610°C) for approximately one hour. When the vessel is completely assembled, the entire unit is postweld stress-relieved. Typically, the reactor vessel would be heated to 1130 ± 25°F (610 ± 14°C) for approximately one hour per inch of thickness. The total stress relief time of a typical reactor vessel, including the final as well as the intermediate stage stress relief, is typically 20 to 25 hours. Due to the uncertainty of this time, the qualification test for the reactor vessel materials may require that all materials receive a simulated postweld heat treatment of 40 to 50 hours. This stress relief time for reactor vessels can increase if the unit is subjected to a number of repair cycles.

4 GENERAL INFLUENCE OF NEUTRON EXPOSURE ON USA VESSEL STEELS

4.1 USA Background on Radiation Damage Assessment

Ferritic materials are susceptible to radiation-induced embrittlement. The beltline region of a PWR receives fluence ranging from 9×10^{18} to 4×10^{19} n/cm^2 (E> 1MeV) during its design life. This level of exposure to fast neutrons can cause significant reduction of fracture toughness. The embrittlement sensitivity of a particular heat or weldment depends on the presence of trace elements, particularly copper, in the alloy. The safe operating life of some reactor vessels, and the justification for full life attainment or extension, depends on a prediction of the adequacy of fracture toughness for the specific heats of material in the beltline region of a reactor vessel at a specified future date.

A Reactor Vessel Surveillance Program (RVSP) is conducted for each vessel to monitor radiation damage using Charpy V-notch, tensile, and in some cases, fracture toughness specimens. Statistical models have been developed to relate such data, and these statistical models are thus related to static, dynamic, and crack arrest fracture toughness. Interpolation schemes are used to predict radiation damage at locations where flux, temperature and even microstructure differ from the RVSP test specimens.

In the period from the mid-1950s to 1972, there were no limits imposed on the presence of copper. Although copper is not an alloying element, small amounts of copper (up to 0.4 wt%) were present as a "tramp" element. In the plate and forging materials, the origin of the copper is traceable to the automotive scrap used in the production of ingots. It is impossible to remove all electrical, hence copper, wiring from the automotive scrap. The base material for the submerged weld wire has always been very low in copper content; however, copper was intentionally plated on the wire to minimize oxidation of the wire during storage. Also, copper plating provided the additional benefit of improved electrical conductivity in the welding head. This copper plating was standard practice for ferritic weld wire. In the period of 1950 to the mid-1960s, large research programs were introduced to evaluate the impact of fast neutrons on mechanical properties of pressure vessel material. During this period of time, it was observed that there was a large variability of radiation embrittlement sensitivity. In the late 1960s, the element copper was determined to be the prime contributor to this variability.

Experiments designed and conducted in the late 1960s and early 1970s confirmed the fact that copper as a trace element was a primary contributor to radiation embrittlement sensitivity. It was also believed that phosphorus, an element well-known for its deleterious effects in other embrittlement mechanisms, was a contributor to radiation embrittlement, and this was also confirmed. Responding quickly to these observations, limits were imposed by the ASME Boiler and Pressure Vessel Code on the level of acceptable copper and phosphorus in those materials exposed to significant fast neutron bombardment. Consequently, all reactor vessels made since 1972 have significantly reduced embrittlement sensitivity.

In the mid- to late 1970s, a greater radiation embrittlement database was developed. Careful scrutiny of data indicated that accounting for copper did not completely account for the variability of observed sensitivity. In the late 1970s, it was suspected that nickel, a close relative to copper on the periodic table, was a secondary contributor to radiation embrittlement. It should be remembered that nickel was intentionally added as an alloy element in the mid-1960s to improve the hardenability of the base materials and also to improve the initial mechanical properties of the weldments. In the 1980s, nickel was, in fact, confirmed as a secondary contributor to the embrittlement of steels, particularly in combination with copper. The effects of both copper and nickel are recognized in terms of producing small microvoids or copper-nickel precipitates in the presence of neutron flux. These act as the primary source of embrittlement. Improved radiation damage trend curves have been developed using these physically based models and data from reactor surveillance programs. These improved radiation damage trend curves provided the basis for Regulatory Guide 1.99, Rev. 2 for calculating vessel RT_{NDT} (nil-ductility transition temperature) and the implementation of the Regulatory Guide has had an impact on the plant operating pressure and temperature limits which vary from plant to plant.

4.2 Neutron Effects in Light Water Reactors

The nature of neutron irradiation effects for the temperature range of operation in light water (LWR) reactors and for the types of ferritic steels used in "western" Boiling and Pressurized Water Reactors (BWR and PWRs) is essentially the same. (It should be noted that the PWR is the most widely used system of all LWRs.) However, the significant factor of change (usually defined as embrittlement) varies between BWR and PWR types with the PWR vessel showing more embrittlement. Unfortunately, many early commercial reactors in the "west" must be treated as "individuals". Nevertheless the design of the two types of reactors (e.g., the water gap) explains the greater fluence of energetic (>1MeV) neutrons impinging in the PWR vessel wall. This generally higher level of neutron exposure dictates that the level of embrittlement is greater in the PWR. Further, the nature of the composition and microstructure leads to greater sensitivity to damage. (Composition differences in WWER reactor vessels are shown in Tables 6 and 7.) Consequently, following the noted development of a statistical body of data from surveillance, measured changes in fracture toughness provide the bases for the widely used U.S. Nuclear Regulatory Guide 1.99, Rev. 2 trends which often are used to project both the vessel embrittlement condition and the response to a pressurized thermal shock (PTS) event.

The most significant effects of neutron fluence on mechanical properties of reactor vessel steels have led to a general view of these effects which are affected by the level and energy of neutron fluence accumulation over time, the specific composition and microstructure of the steel exposed and the temperature during exposure. Some data indicate, as well, that the rate of exposure to the number of energetic neutrons bombarding the steel each second may affect the ultimate changes in critical mechanical properties for some materials in certain ranges. The most significant of these properties include: strength (yield strength increases along with ultimate tensile strength), notch toughness (the crucial measurement) decreasing in terms of Charpy V-notch (C_v) curve position and shape (with a shift to higher temperatures and a lower ductile shelf level), and static (K_{Ic})and dynamic (K_{Id}) fracture toughness levels which unfortunately must be deduced from C_v data since valid K_{Ic} and K_{Id} numbers currently can only be determined using specimens too large for the surveillance programs. Accordingly, the strength and toughness measures which are applied to assess fracture potential or to judge structural integrity must be deduced from small surveillance specimens. Such knowledge combined with that of steel composition, exposure temperature, flaws present (and stress levels at these flaws) are essential to assessment of vessel integrity under normal and accident conditions.

4.3 Tensile Properties of Irradiated Steels

Neutron radiation effects were first identified as radiation hardening and strengthening. Casual interpretation of this change would suggest a good (or positive) effect on the key properties of pressure vessels steels. However, both engineering and fundamental analysis of these increases in tensile properties carried to the conditions of pressure vessel service can be interpreted as potentially detrimental.

First, while both yield and ultimate strength are raised by neutron exposure, yield strength is increased more rapidly to the point that it may in some cases approach the ultimate strength. This increase then limits the degree of uniform elongation possible thereby creating a "less forgiving" medium under conditions which could be serious to the integrity of a potentially flawed vessel since elastic deformation is reduced. Thus, there is less tolerance for correction in a deformed and flawed vessel if an "overpressure" transient should occur. This projection may be extended to the potential ductile rupture or brittle fracture of a residual (unruptured) section coupled with a rapid separation in the vessel. This scenario must be avoided at all costs in the primary system nuclear pressure vessel.

Fundamentally linkage of the radiation strengthening and potential fracture can be described as follows: microscopically it is possible to describe the role of radiation produced defects as barriers to dislocation motion which results in hardening and strengthening of the irradiated material. The increase in the yield strength resulting from the presence of defect aggregates has been shown to be temperature independent in irradiated iron. This temperature independent increase in yield stress can in turn be used to explain the shift in transition temperature as shown schematically in Figure [11]. If the fracture stress is assumed to be roughly temperature independent and the intersection of the fracture stress curve and the flow stress curve is taken as the brittle-to-ductile transition temperature for the unflawed condition, an increase in flow stress produced by irradiation can be shown to shift the point at which the fracture stress and flow stress curves intersect and increase the brittle-to-ductile transition temperature.

Tension tests provide measurements of yield stress, ultimate tensile strength, percent elongation, and percent reduction in area. The change in yield strength is usually considered to be the most sensitive property to neutron irradiation. Trends for these properties have been observed from various national and international programs [11].

Yield strength increases after neutron irradiation of advanced pressure vessel materials are shown to provide results which follow a pattern that is most similar to the shift in Charpy transition temperature when plotted against neutron fluence. Such curves are usually described by a relationship

$$\Delta\sigma_{0.2} = A\ (\phi t)^n$$

where *A* and *n* are constants and ϕt is the neutron fluence in terms of E> 10^{19} n/cm^2. The increase in yield strength for a number of advanced steels is given by

$$(\Delta\sigma_{0.2})\quad 20^{\circ}C = 44.24\ (\phi t)^{0.356}$$

As will be seen later, the value for the exponent, n, for the Charpy transition temperature shift may be as high as 0.5; there is, however, a larger scatter in the results which may arise from tension test variables as well as from scatter from heat-to-heat variations in the steel. A comparison of the change in yield strength with irradiation between a reference steel (HSST-03) and several advanced steels shows that irradiation of the older HSST-03 steel induces a larger increase in yield strength at lower neutron fluences, and an increased sensitivity with increasing neutron fluence.

4.4 Changes in Transition Temperature (Fracture Potential)

The ferritic low alloy steels most widely used for reactor pressure vessels commonly exhibit the phenomenon of ductile-to-brittle transition over a relatively narrow temperature range (about 100°C or less). This range involves a transition from ductile (fracture only under conditions of plastic overloading or tearing) to brittle (fast fracture under elastic loading conditions) as the temperature is reduced. Over this range the micro- and macroscopic nature of the fracture gradually changes from ductile dimpled rupture, with attendant plastic deformation of the adjacent matrix, to cleavage along crystallographic boundaries. There is a broad range of mixed fracture modes between these points. Two distinct interrelated patterns of embrittlement are definable: a) the very distinct increase in the ductile-to-brittle transition temperature (ΔT) and b) the less predictable but important reduction in fracture resistance known as the ΔUSE (reduced upper shelf energy). A crucial development of the former is the relative equivalence in the measured NDT and the Charpy V-notch transition temperature after irradiation. This fact made possible the RT_{NDT} (reference transition - nil ductility temperature) which is initially defined at the 50 ft-lb (68J) level in combination with the measured NDT and adjusted by the 30 ft-lb (41J) shift after irradiation and forms the principal criterion for defining the vessel's embrittlement condition.

The design and operation of nuclear reactor pressure vessels includes the consideration of radiation-induced embrittlement of ferritic, low alloy steels in the fast neutron fluence range of 10^{18} to 10^{20} n/cm^2 for energies (E) greater than 1 MeV. The effects of neutron irradiation are manifested principally by an increase in yield and ultimate tensile strengths and decrease in fracture toughness. The change in toughness generally is monitored by using Charpy V-notch (C_v or CVN) test results as a function of temperature for specimens that have been irradiated in surveillance capsules. The results from irradiated CVN testing show both an increase in the ductile-to-brittle transition temperature measured at the 30 ft-lb (41J) energy level and a drop in the upper shelf energy (USE). Both the shift in T_{30} (41J) and the drop in USE are key regulatory issues for reactor pressure vessels operation. Ideally, the shift in measured linear elastic fracture toughness (K_{Ic}) as well as the drop in measured upper shelf J-resistance fracture toughness should be used, but surveillance programs are generally limited to only a few small fracture toughness test specimens usually compact fracture specimens with thicknesses less than 1-in. (25 mm). Recent developments have enhanced the potential for use of such "quantitative" specimens in irradiation studies.

4.5 Upper Shelf Toughness

Since the ideal J-R tests are not acceptable for direct use in vessel surveillance, USE values or USE limits are imposed to avoid conditions for ductile tearing of a vessel ligament. Under USA Federal Code 10 CFR 50, Appendix G, a minimum upper shelf Charpy impact energy requirement of 68 J (50 ft-lb) (for specimens with transverse TL orientation) is specified. Further, this code requires notification to the US NRC three years in advance of the date when it is estimated that the 50-ft-lb criterion will be transgressed. If data fall below 50ft-lb level, then the completion of the following three steps is required:

1. A 100% volumetric inspection of the reactor beltline region materials whose USE is less than 50 ft-lb following the requirements of American Society of Mechanical Engineers Section XI [12]
2. Supplemental tests to provide additional evidence of fracture toughness changes in the beltline region materials (i.e. test reactor irradiations of archive material tested to validate the adjusted K_{IR} curve).
3. A fracture mechanics analysis that conservatively demonstrates that adequate margin still exists for continued safe operation; this requirement sometimes can be used without the inspection and toughness results (1 and 2 above) if

conservative estimates of both are used and an adequate margin is easily justified.

The first step may require a revision of the inspection schedule. The second step is difficult because most surveillance capsules do not include fracture mechanics specimens; however, correlations between USE and fracture toughness have been developed. Some surveillance capsules do include small fracture mechanics specimens, but adequate testing techniques for standardization of these specimens are still under development. Therefore, characterization of the upper shelf toughness is a major problem. Guidelines and requirements for a plant-specific analysis to assure safe continued operation when a plant has violated the 50 ft-lb level have only recently been developed through ASME Code Section XI activities. US Federal Code 10CFR50 indicates that thermal annealing of affected beltline material is an acceptable approach for restoring toughness properties. In the USA the number of older plants facing USE limits is significant so that a quantification of these effects or plant operational implications of these effects becomes urgent for a number of older plants.

4.6 Other Factors Affecting Mechanical Properties of USA Vessel Steels

The <u>microstructure</u> of reactor vessel alloy steels generally determines its basic material properties. Several types of microstructural imperfections can be created by irradiation, resulting in changes in tensile and fracture properties. Examples of these microstructural changes are: clustering of vacancies (empty lattice sites) clustering of interstitials (extra atoms forced between lattice sites), forming microvoids, and forming dislocation loops. These imperfections can be created when high energy neutrons collide with individual atoms, knocking some of them out of their lattice positions. Because of the relatively slow speed of the displaced atoms and their uniform size, the displaced atoms interact with other lattice atoms in a billiard ball manner, creating what is called a cascade phenomenon. Once the cascade has dissipated, local zones of depleted atoms draw interstitial atoms by diffusion through the metal. The interstitials and microvoids act as barriers to dislocation movements.

Another example of microstructural changes that can occur during radiation exposure is irradiation-enhanced precipitate formation and alteration. Detailed characterization of these actions is difficult because of the small precipitate sizes (1 to 2 nm). However, research has indicated that precipitation of small, complex copper-enriched clusters and their interactions with dislocations may be a dominant embrittling mechanism in most radiation-sensitive vessel steels. Increased vacancy population created by neutron irradiation can enhance the copper precipitation process at PWR operating temperatures. The thermodynamic situation created by irradiation drives the system away from normal equilibrium and as such is strongly dependent upon the displacement damage rate and the irradiation temperature. Precipitation influences with other elements such as nickel and phosphorus are also important.

Other factors than microstructure and composition affect the nature and extent of properties changes in a nuclear environment. Key among these factors are exposure temperature, radiation fluence, and possibly flux. A general pattern of the temperature effect is illustrated in Figure 2 [11]. Of course, effects resulting from copper content may dominate the physical environment.

While the large reductions of ΔT suggest thermal annealing as a solution, the implications in terms of the general vessel's fracture toughness, both ΔT and ΔUSE requires a more quantitative general approach using fracture mechanics parameters (K_{Ic}, J_{Ic}) and T). However, serious limitations exist at this time since, as noted, earlier; larger specimens than can be accommodated in vessel surveillance programs are required. The solution then has been to test large K and J specimens in non-power reactors and thereby establish, by correlation, limit curves which can be applied using smaller surveillance specimens. The combination of a carefully established K_{Id} - K_{Ia} curve (collectively called the ASME Code K_{IR} limit curve) coupled with a projected RT_{NDT} (index to reference temperature, NDT) can accommodate ΔT or ΔRT_{NDT} values based on CVN specimens from surveillance programs. A similiar stat initiaiton K_{Ic} curve has been established in the ASME Code, Section XI, Appendix A, for assessing indications discovered during inservice inspection. This approach has been accommodated in regulatory limit curves. In the USA limits from adjusted surveillance data curves derived for US Regulatory Guides for irradiated RT_{NDT} and for PTS (Pressurized Thermal Shock) limit curves. The weakest link in this approach is the underlying basic dependence on values derived from bounding curves from dynamic fracture toughness, K_{Id}, and crack arrest toughness, K_{Ia}, for unirradiated steels or from large specimens irradiated in non-power reactors (i.e., at accelerated exposure rates). The types of curves for USA regulatory limits thus derived are shown in Figure 10. An elaborate correlation for chemical content, copper and nickel, is an underlying guide to these (and other) limit curves. Also shown are earlier correlations for comparison. Note that the current correlation predicts more damage at lower fluences and less at higher fluences.

It should be noted that after projecting the RT_{NDT} for irradiation, a radiation-adjusted value from the K_{IR} curve is used for an identified or postulated defect to evaluate potential failure during emergency and faulted conditions. The K_{IR} and K_{Ic} curves are used in lieu of measured fracture toughness data for a plant specific material.

Neutron energy spectrum differences clearly should and do have an effect on measured properties changes of a given steel at a given temperature. This effect is because the usual technique of measuring dosage involves a neutron exposure in terms of the number of neutrons having energies above a single level, usually 1 MeV.

This technique has worked well because of the similar environments of light water reactors in which the most damaging neutrons are of similar magnitudes and the energy spectrum shape is quite similar. However, it is important to note that the 1MeV neutrons are probably more damaging because of their numbers and because their collision cross-sections (probability for displacing iron atoms, and thereby the cause for damaging secondary collisions) are dominant in the usual LWR neutron spectrum. Thus, an explanation exists for the reproducibility of data >1MeV from reactor-to-reactor. Of course, neutron spectra do vary and lower energy neutrons do cause damage. Hence, there is a recent move to use displacements per atom (dpa) to measure damage. However, the difficulty in retroactive dpa analysis, or even in spectrum analysis, degrades the chances for gaining any significant benefit from "backfitting" using dpa techniques.

Studies over many years have sought possible <u>flux (fluence rate) effects</u>. Many results between test and power reactor exposures of the same (reference) steel suggest that variations in flux between one and one thousand (3 orders of magnitude) caused no significant effect. However, caution and logic have led to the use of only surveillance, near one-to-one exposure comparisons, for projecting trends. This approach eliminates the rate variable though it has not been demonstrated conclusively to be a significant variable for all steels in radiation-induced changes to key mechanical properties.

In order to minimize effects of neutron radiation to mechanical properties of a given reactor steel, it is essential to isolate key factors; these factors are: the steel's composition (especially content of copper and to a lesser extent nickel and phosphorus), the neutron fluence (E >1MeV) or dpa, the dominant temperature of exposure and the index for these factors for commonly used classes of steel establishing mechanical performance curves. These factors affecting fracture potential when applied to the key operating factors (such as applied stress on known flaws and operating temperature gradients) establish probabilities for serious failures under abnormal operating conditions and the bases for critical conditions which must be controlled.

In conclusion, a number of factors impinge on vessel life assessment which revolve around irradiated mechanical properties of the vessel steel and its condition during operation. These factors together permit establishment of options for estimating and extending vessel life considering neutron radiation effects. Some of the technical issues essential to quantification of vessel degradation by neutron damage and emerging options for life assessment and possible life extension in irradiated vessels are outlined in Figure 4.

5 GENERAL INFLUENCE OF NEUTRON EXPOSURE IN THE FORMER USSR VESSEL STEELS

5.1 Former USSR Vessel Design Considerations

(Design considerations for the WWER pressure vessels are presented in Chapter 4 of this book.)

5.2 Effect of Irradiation Temperature

Temperature has a marked influence upon radiation embrittlement, the level of embrittlement being reduced progressively with higher temperatures. The reason for this phenomenon is that with higher temperatures, the ability of displaced atoms to return to a vacancy site or other relatively innocuous location is enhanced, thereby relieving part of the damage.

It is reasonable to suppose that the process of self-annealing takes place at elevated temperatures resulting in a decrease in the shift of the brittle-to-ductile transition temperature.

Figures 5 and 6 [13] give the data on the effect of irradiation temperature on radiation embrittlement factor of steels, used for fabrication of reactor pressure vessels and weld metals in the former USSR.

The data for former USSR steels are given as dependence of radiation embrittlement factors against temperature. A_F is the factor which determines the rate of shift of brittle-to-ductile transition temperature due to radiation according to the following formula:

$$T = A_F \left(\frac{F}{F_c}\right)^{1/3} ; \quad F_o = 10^{18}\ n/cm^2$$

(Note that F is fluence for E > 0.5 MeV.)

5.3 Fluence Rate (Flux) and Spectrum Effects in Former USSR

The scheduled time of power reactor operation exceeds, as a rule, the time of specimen irradiation in a research reactor with the fast neutron fluence corresponding to the end-of-life on the inner wall of the reactor vessel by up to three orders of magnitude. Thus, the issue of radiation stability of reactor vessel materials also involves the allowance for a possible effect of the neutron flux (fluence rate) on irradiation embrittlement.

In all countries where vessel-type LWRs are operated, the monitoring of the reactor vessel materials condition is currently accomplished by a surveillance program. However, as is typical in most types of reactor vessels, the flux in the surveillance capsules exceeds by several times the flux in the vessel wall. The question of a flux effect on the mechanical properties of a steel, such as variations in the transition temperature shift, is also possible when the surveillance capsule program utilizes high lead factors (e.g., greater than a factor of 10 times). Most U.S. surveillance programs do not exceed a lead factor of about 3 to 5, and therefore little flux effect is expected in USA LWRs.

It must be pointed out that the experiments on detection of a flux effect are very complicated and time consuming. For a correct establishment of a flux effect, constant specimen temperature should be maintained in both flux conditions during the whole irradiation period and the neutron energy spectrum must be nearly equal.

Though efforts in this direction have been made both in the former USSR and in other countries, nobody has succeeded in detecting a consistent flux effect. The results depend upon the chemical composition of the steel, ranges of flux, and integrated fluences within which the experiment is carried out. Information in the literature does not provide data on the accuracy of maintaining capsule temperature during irradiation. A small change in the irradiation temperature within the range from 200°C to 300°C has a very strong effect on steel as was noted earlier.

Figure 7 shows the calculational integral power spectra of the fast neutron fluxes for the channels with surveillance capsules in the WWER-440 reactors fully loaded (curve 1) and loaded with 36 fuel shielding assemblies (curve 2). It is clear that the shapes of the neutron flux spectra in Figure 3 are similar, thus indicating that the use of the shielding fuel assemblies does not adversely affect the spectrum shape of neutrons hitting the surveillance specimens.

The maximum values of the neutron flux were 4.2×10^{12} and 4.75×10^{11} n/cm²/s, respectively. It follows from this that installation of the shielding fuel assemblies results in reduction of the neutron flux by about nine times for the surveillance irradiation channels. Using the calculated shape of the spectrum shown in Figure 15 and the cross section displacements for iron, the numerical values for the rates of formation of displacements per atom (dpa) have been calculated [14].

The maximum rates of formation of dpa exist in WWER reactor surveillance channels in which those of a normal core are

3.7×10^{-9} and 4.1×10^{-10} for the shielded fuel elements. The difference in fluence rate by using dpa is also approximately nine times.

The values of ΔT depended importantly on the total content of phosphorus and copper in the steel for the specimens irradiated in the WWER fully charged (i.e., flux = 4×10^{12} n/cm^2/s). The analysis of the data shows that the experimental points obtained from the specimens irradiated in a WWER with a load of 313 fuel assemblies (neutron flux equal to 4×10^{11} n/cm^2/s), in most cases, lies above the upper envelope of weld data. This discrepancy is particularly significant for the steels with elevated contents of phosphorus and copper.

It should be pointed out that the values of A_F (ΔT) for a low flux were obtained at neutron fluences between 1×10^{19} n/cm^2 and 5×10^{19} n/cm^2, while for high flux, the data were obtained at fluences greater than 5×10^{19} n/cm^2. Therefore, a more correct assessment of the flux effect can be achieved by the actual changes in measured shift.

The program of surveillance investigations of the reactor vessel materials at the Armenian NPP-2 (ANPP-2) prescribed use of weld metal surveillance specimens cut from the same weld as in the vessel. Thus specimens of the identical weld were irradiated under identical conditions at ANPP-2 with a flux = 4×10^{12} n/cm^2/s and at ROVNO (RNPP-1) with a flux = 4×10^{11} n/cm^2/s. The spectral characteristics of neutrons affecting the specimens in both reactors are essentially identical. The only difference was in the irradiation periods and the neutron fluxes.

Figure 9 presents the experimental values of transition temperature shift (ΔT) as a function of the fast neutron fluence for surveillance specimens at RNPP-1 (□) and ANPP-2 (O) and in the channel of the research reactor (MR) at the I.V. Kurchatov Institute of Atomic Energy. In the latter experiment the tests were from control specimens of the weld metal from the RNPP-1 irradiations. The irradiation of the specimens was carried out in contact with the coolant at a temperature of 270°C - 276°C. The neutron flux in the specimen locations was 7×10^{12} n/cm^2/s. It follows from the analysis of Figure 9 that at a flux of 4×10^{12} n/cm^2/s a saturation effect is not observed up to a fluence of 4.9×10^{20} n/cm^2 (>0.5 MeV). This level is nearly twice as high as the lifetime fluence for the base metal of the WWER-440 reactor. At a flux of about 4.10^{11} n/cm^2/s the embrittlement process occurs more intensely than at a flux of 4×10^{12} n/cm^2/s within the fluence range of 1 to 5×10^{19} n/cm^2.

The experimental values of ΔT are identical for fluences of 3.4×10^{19} and 5×10^{19} n/cm^2. But whether it is a tendency toward saturation of irradiation effects at a flux of about 4×10^{11} n/cm^2/s will be shown by testing the available assortment of specimens from RNPP-1 with a higher fluence level. This question is especially important in view of the fact that on the inner surface of the WWER-440 pressure vessel, the flux is equal to 3×10^{11} n/cm^2/s, that is, it is very close to the flux values obtained with the same types of specimens in RNPP-1.

6 CONCLUSIONS

This chapter covers the effects of neutron irradiation on mechanical properties of reactor pressure vessel steels as viewed from the USA and former USSR perspectives for specific primary systems used in these countries. It is intended that the summary provided serves as a proxy for all Light Water Reactor Systems (LWRs) and provide background on the critical elements controlling such neutron damage effects.

The major properties affected by high energy neutron exposure are hardness (a physical property) and the related mechanical properties strength and toughness. These and variants caused by the environmental and materials factors can have serious effects upon vessel integrity and hence the life of a LWR plant. The factors of neutron environment (flux, fluence, and energy spectrum) and temperature also have major effects. However both types of nuclear systems (USA and the former USSR) operate in similar ranges of neutron and temperature conditions so that no major discrepancies in conclusions are reached between the two. Further the evolution of vessel steels, construction methods in both countries, and aspects related to metallurgy and microstructure for plate, weld, and forging materials are crucial to the level of embrittlement attained.

A most critical factor in the quantitative assessment of embrittlement is steel composition, especially the levels of copper, nickel, and phosphorus. The former USSR investigators have considered other elements as well: vanadium, silicon, aluminum, manganese, nitrogen, boron, and oxygen. Copper, nickel, phosphorus plus vanadium are agreed to be most prominent in causing embrittlement, though the most detrimental threshold levels of content are yet to be defined. Further, the role of microstructure is a qualitative factor in the change of mechanical properties with neutron exposure, but more research is required to provide specific and quantitative guidance to these effects.

The mechanisms of radiation hardening and embrittlement are defined physically as submicrostructural features: precipitates, vacancy clusters, and interstitial clusters. The shape, size, and distribution of these features are sensitive to thermal and nuclear dynamics of service environments. Suggestions of fluence rate effects have been made, but surveillance conditions in the USA show limited rate effects. In the former USSR, while the rate differences may reach an order of magnitude, investigators believe temperature variations can be so overwhelming as to cover any flux effect. More research on this subject is required before quantitative assessment of a rate effect can be assigned.

The extent of our knowledge of the most critical factors to steel embrittlement response, however, is such that damage or embrittlement can be assessed and assigned in most vessels based upon knowledge of the steel (composition and metallurgical microstructure), neutron environment (primarily fluence) and temperature. From this knowledge, trend curves relating these factors to fracture toughness provides a limiting condition from which operators may judge integrity provided flaw sizes and stresses at critical vessel locations can be assigned with confidence.

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Chapter 8 Tables

Table 1
Sizes and Weights of Typical Reactor Vessels

Type (Vendor)	*Size MW(e)*	*Overall Height (m)*	*Inside Diameter (m)*	*Wall Thickness (mm)*	*Dry Weight (kg)*
BWR (G.E.)	1065	22.2	6.4	150	680,000
PWR (B&W)	1213	13.8	4.9	248	540,000
PWR (W)	1106	13.3	4.4	222	390,000
PWR (C.E.)	1270	15.3	4.6	254	510,000

G.E. - General Electric
B&W - Babcock and Wilcox
W - Westinghouse
C.E. - Combustion Engineering

Table 2
Pressure Vessel Plate Materials

Grade	*Heat Treatment*	*Vessels*	*Usage*	*Type*
A212B	N&T, Q&T	All	Mid 50s-60s	C-Si
SA302B	Q&T	All	Mid 50s-60s	Mn-Mo
SA302B (modified)	Q&T	All	Mid-Late 60s	Mn-Mo-Ni
SA533B-1	Q&T	RPV, pressurizer	70s to present	Mn-Mo-Ni
SA533B-1 (low Cu, P)	Q&T	RPV beltline	1973 to present	Mn-Mo-Ni
SA533A	Q&T	Steam Generator	70s to present	Mn-Mo

N - Normalized
Q - Quenched
T - Tempered

Table 3
Pressure Vessel Forging Materials

Grade	*Heat Treatment*	*Applications*	*Usage*	*Type*
A105 II	N&T	Flanges, nozzles	50s (limited)	C-Mn
SA182F1 (modified)	Q&T	Flanges, nozzles	50s-60s	Mn-Mo-Ni
SA336	N&T, Q&T	Flanges, nozzles	50s-60s	C-Mn-Ni
Code Case 1236	Q&T	Flanges, nozzles	1957-60s	Low Ni-Cr-Mo
SA508-2	Q&T	Flanges, nozzles, rings	Current	Same as code case 1236
SA508-2a	Q&T	Tube Sheets, flanges	Current	
SA508-3	Q&T	Flanges, nozzles, rings	Current	same as SA182F1 modified)

Table 4
Welding Techniques for Nuclear Components

Technique	*Type*	*Heat Treatment*	*Application*	*Comments*
Sub Arc	Auto	Stress relief	Wherever possible	Good properties high deposition
Sub Arc - narrow gap	Auto	Stress relief	Girth seams	Reduced weld volume
Shielded metal arc	Manual	Stress relief	Complex or irregular	Very flexible
Electroslag	Auto	Q&T	Longitudinal seams in some BWRs	Very high deposition rate

Table 5
Technical Characteristics of Reactor Vessels

Parameter	Reactor Type	
	WWER-440	WWER-1000
1. Coolant temperature at the reactor inlet under nominal conditions (°C)	267 ± 2	289 ± 2
2. Coolant pressure in the reactor vessel under nominal conditions (MPa)	12.3	15.7
3. Fast neutron fluence for the reactor vessel base metal during design service life of 40 years, n/cm^2 ($E \geq 0.5$ Mev)	2.6×10^{20}	5.7×10^{19}
4. Fast neutron fluence for the weld metal during a service life of 30 years 40 years	 1.86×10^{20} 2.48×10^{20}	 – 5.7×10^{19}

Specifications for heat composition of base metals used in the beltline materials of former USSR reactor vessels

Steel	Elements content, %														
	C	Si	Mn	Cr	Mo	V	Ni	Cu	Co	As	S	P	Sb	Sn	P+Sb+Sn
								not more than							
15X2M0A (WWER-440 old design)	0.13 0.18	0.17 0.37	0.30 0.60	2.50 3.00	0.60 0.80	0.25 0.35	not more than 0.40	0.30	0.025	0.040	0.020	0.020	–	–	–
15X2M0A-A[1] (WWER-440)	0.13 0.18	0.17 0.37	0.30 0.60	2.50 3.00	0.60 0.80	0.25 0.35	not more than 0.40	0.10	0.025	0.010	0.015	0.012	0.05	0.05	0.015
15X2HM0A-A (WWER-1000)	0.13 0.18	0.17 0.37	0.30 0.60	1.8 2.3	0.5 0.7	0.10 0.12	1.0 1.5	0.10	0.03	0.010	0.012	0.010	0.005	0.005	0.015

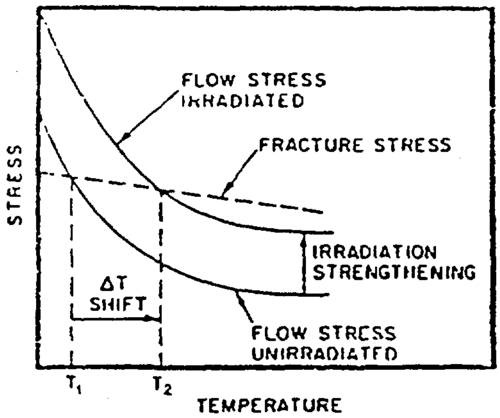
[1]Steel 15X2M0A-A is currently being used (from about 1980) for the manufacture of reactor shells positioned opposite the core. 15X2M0A-A and 15X2HM0A-A steel are smelted using pure original iron ore mixture (i.e., not scrap metal).

Specifications for composition of submerged arc weld metal in the beltline for former USSR reactor vessels

Base metal joining to base metal	Content of elements, %										
	C	Si	Mn	Cr	Ni	Mo	V	Ti	Cu	S	P
									not more than		
15X2M0A with 15X2M0A (WWER-440 design)	0.04–0.12	0.20 0.60	0.6–1.3	1.2–1.8	not more than 0.30	0.35–0.70	0.10–0.35	not more than 0.05	0.25	0.035	0.042
Any joint with 15X2HM0A-A (WWER 440)	0.04–0.12	0.20–0.60	0.6–1.3	1.2–1.8	not more than 0.30	0.35–0.70	0.10–0.35	not more than 0.05	0.10	0.015	0.012
15X2HM0A-A with 15X2HM0A-A (WWER-1000)	0.05–0.12	0.15–0.45	0.50–1.00	1.40–2.10	1.20^x 1.90	0.45–0.75	–	–	0.08	0.015	0.012
15X2HM0A-A with 15X2HM0A-A (WWER-1000)	0.04 0.10	0.15 0.45	0.45 1.10	1.20 2.00	1.00^x 1.3	0.40 0.70	–	0.01 0.06	0.08	0.015	0.012

[x] By new requirements upper content of nickel is not more than 1.5%.

Fig. 1 Schematic diagram illustrating how irradiation strengthening produces a shift in transition temperature [1].



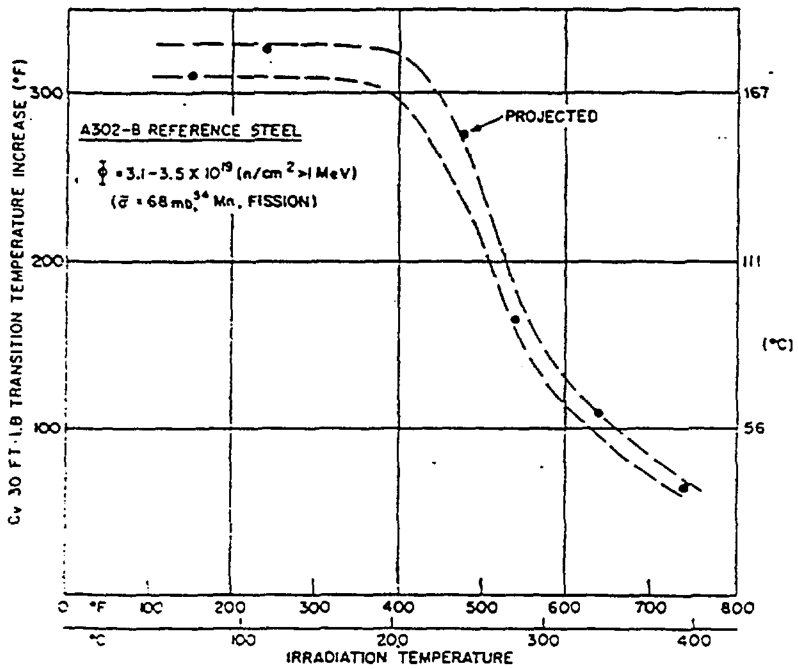


Fig. 2 Effect of irradiation temperature on transition temperature increase for an A302-B reference steel [10].

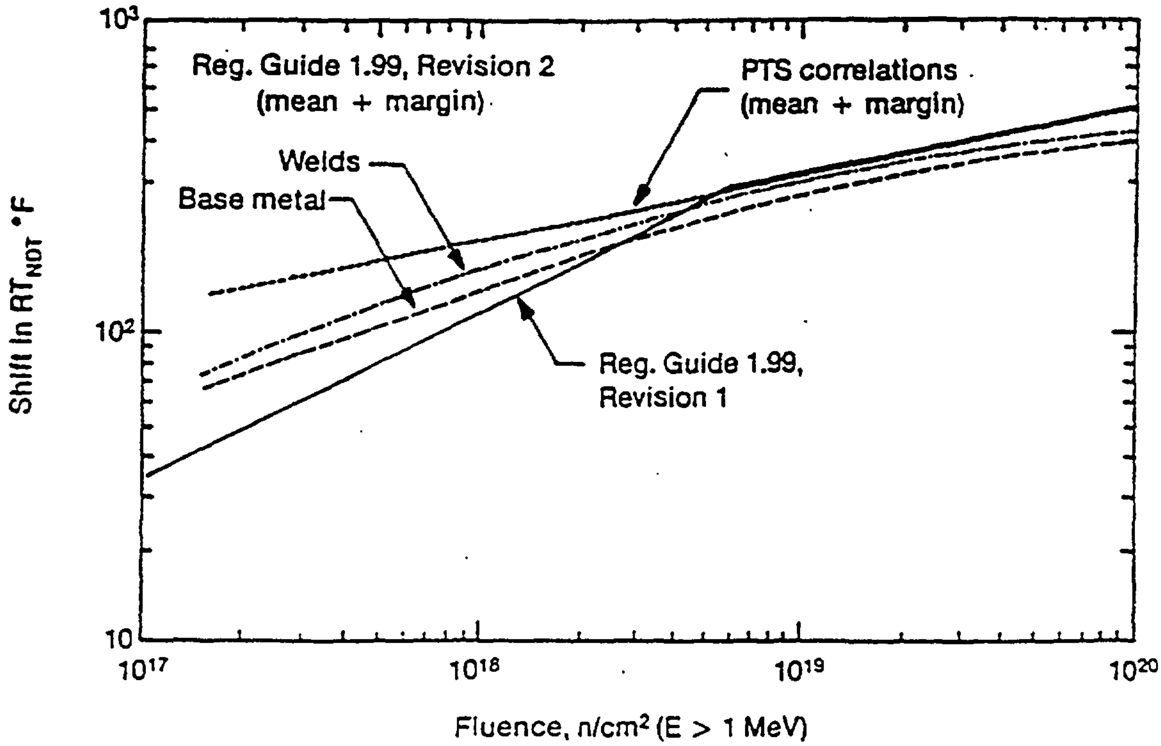


Fig. 3 A comparision of shift predictions for welds and base metal (0.35 wt% Cu and 0.60 wt% Ni) calculated using Revisions 1 and 2 of Regulatory Guide 1.99 and other PTS correlations.

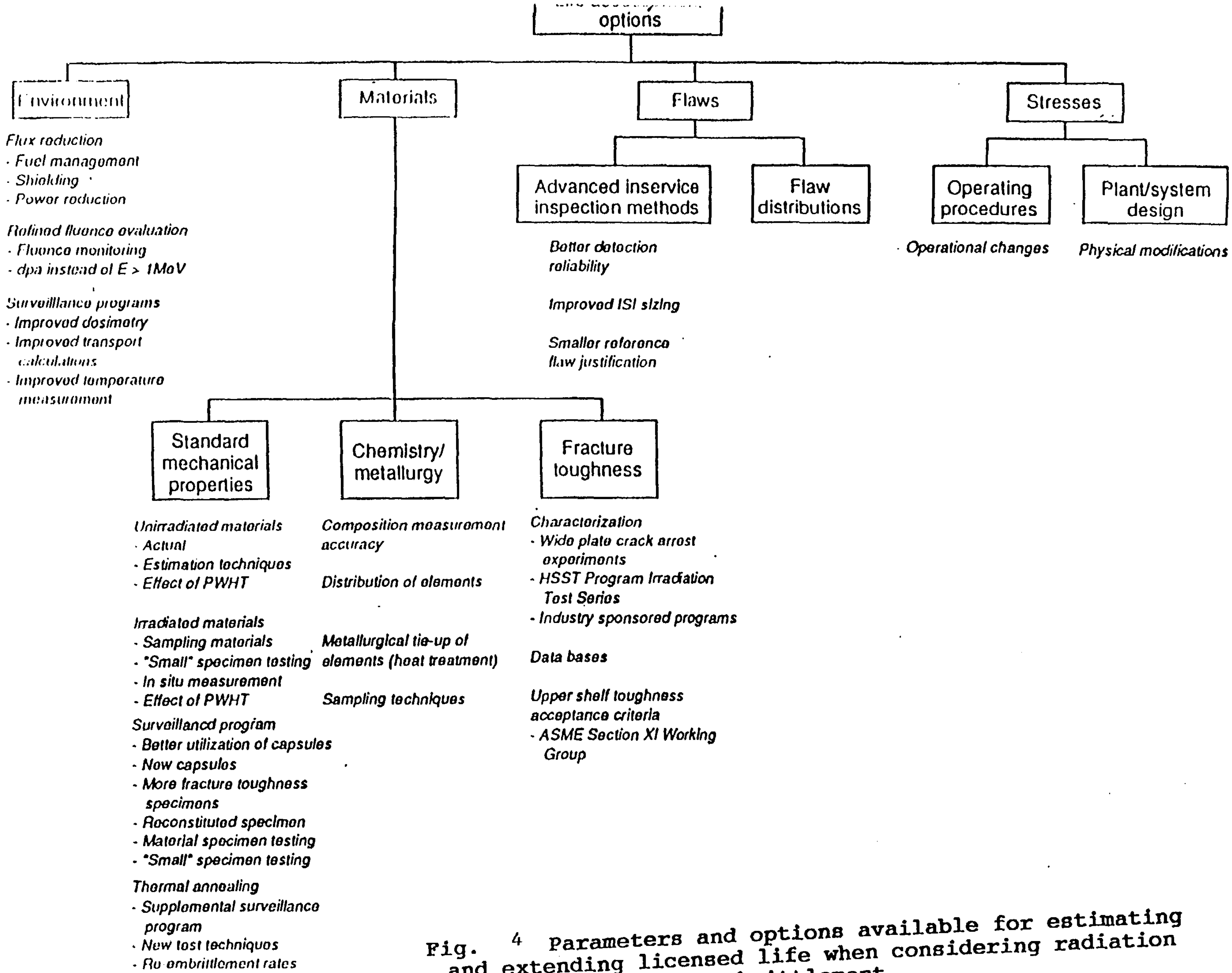
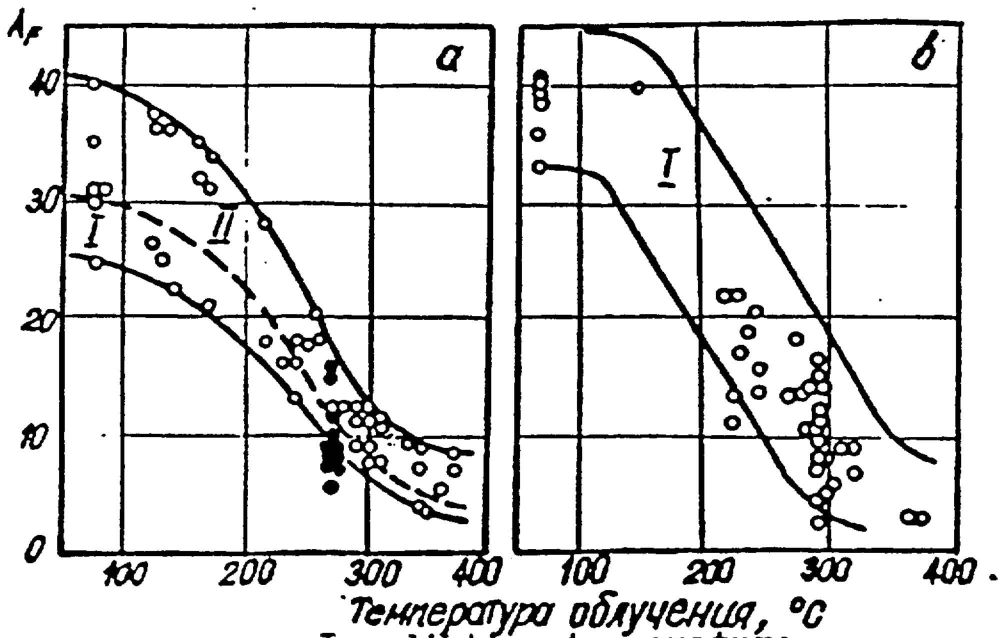


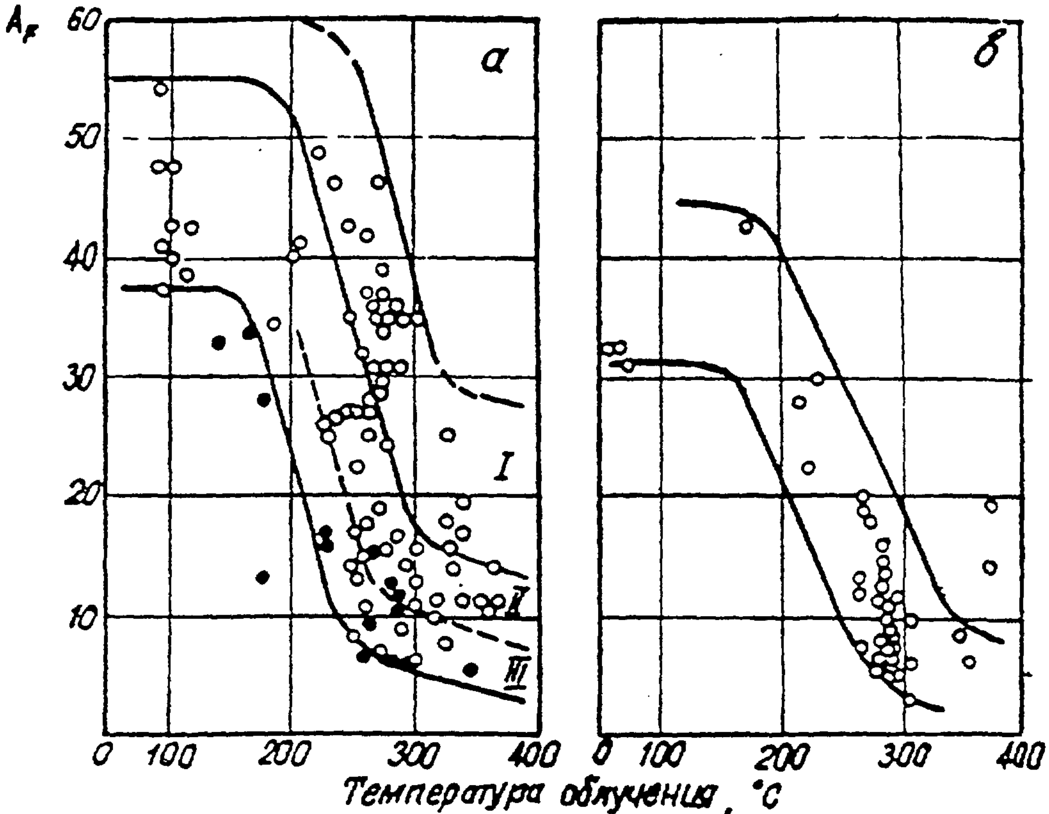
Fig. 4 Parameters and options available for estimating and extending licensed life when considering radiation embrittlement.

Fig. 5 Effect of irradiation temperature on radiation embrittlement factor of steels 15X2MOA-A (a) and 15X2HMOA-A (b)

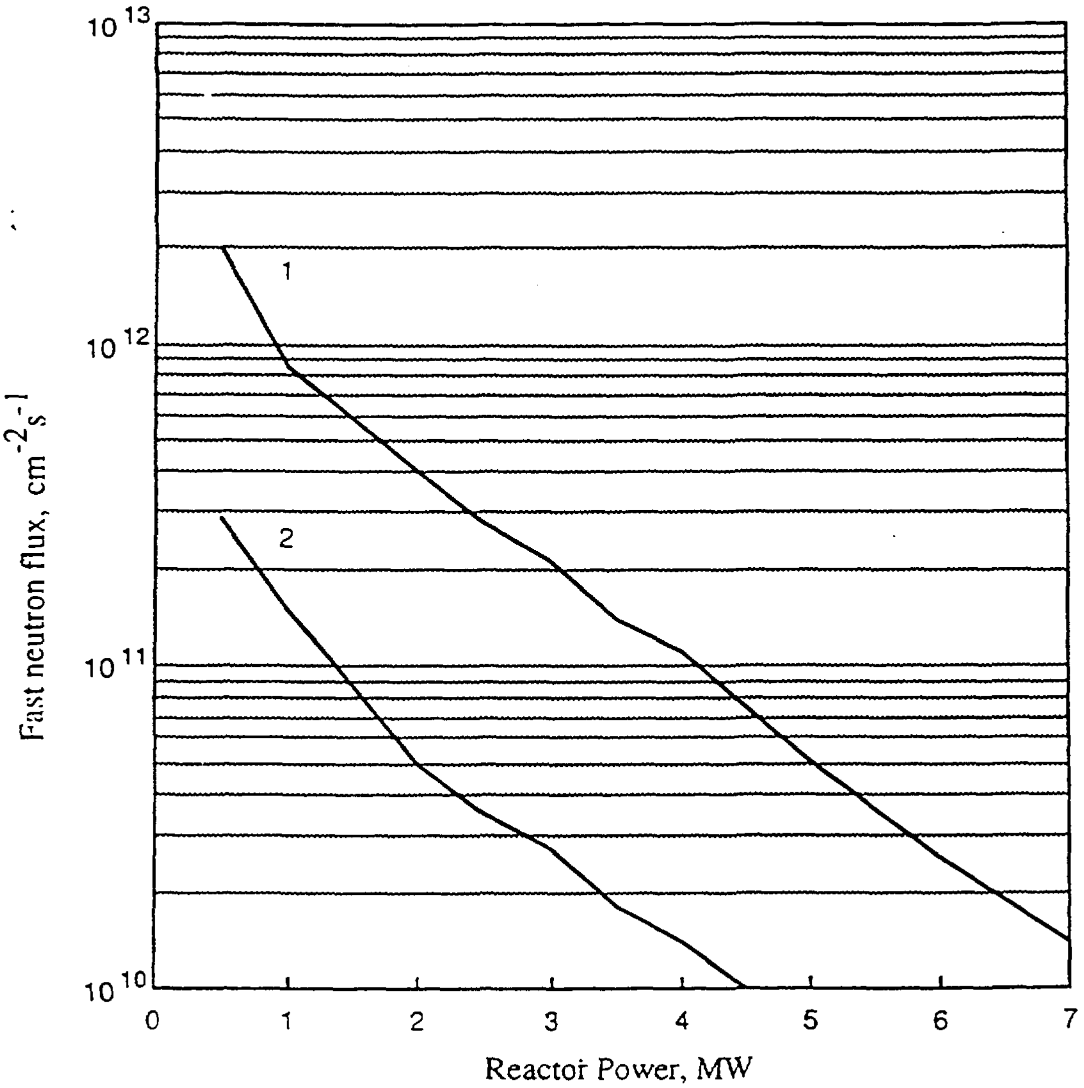


Irradiation temperature

Fig. 6 Effect of irradiation temperature on radiation embrittlement factor of weld metal of steels 15X2MOA (a), and 15X2HMOAA (Cu ≤ 0.08%; P ≤ 0.012%) (b)

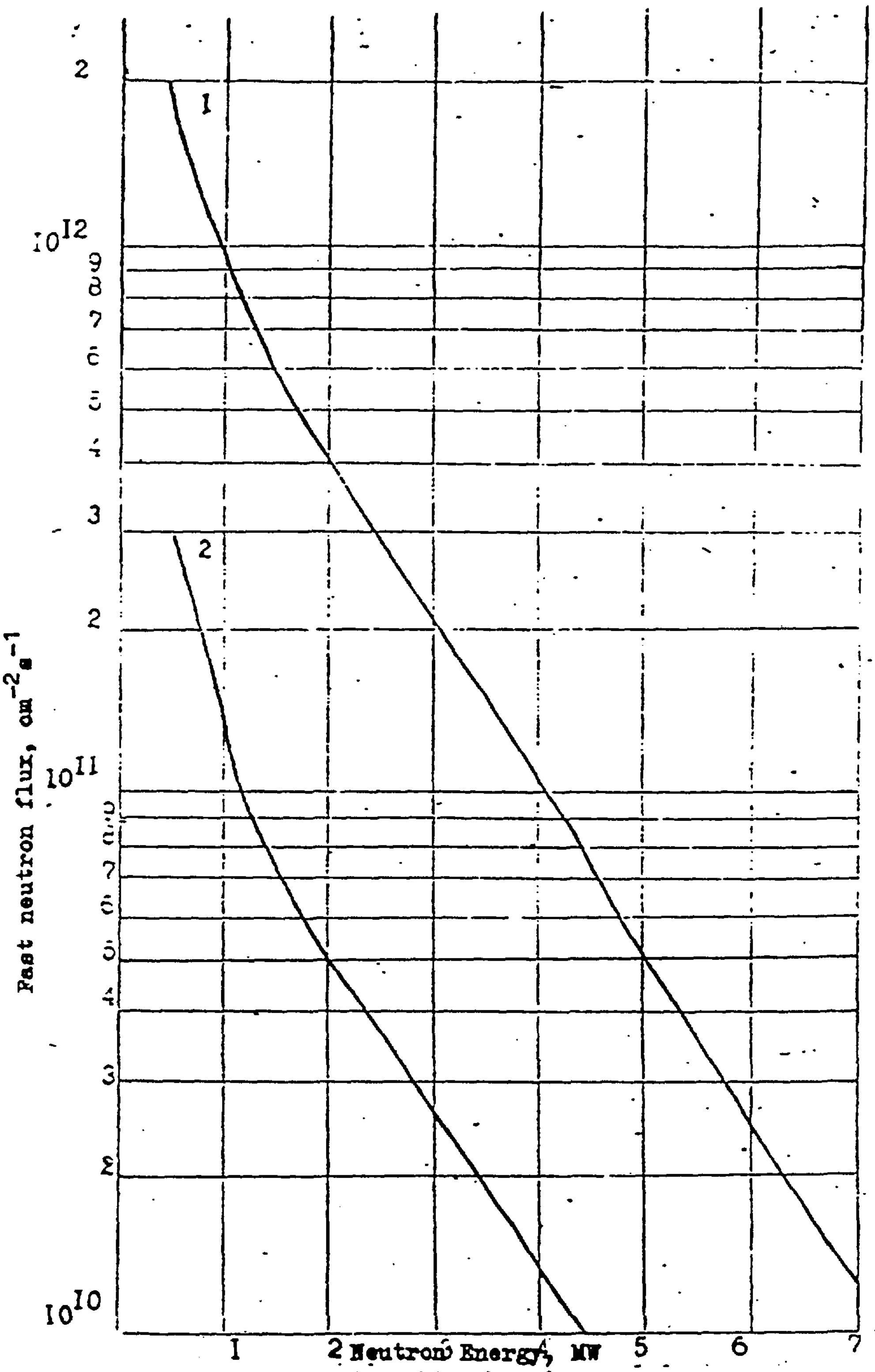


Irradiation temperature, °C



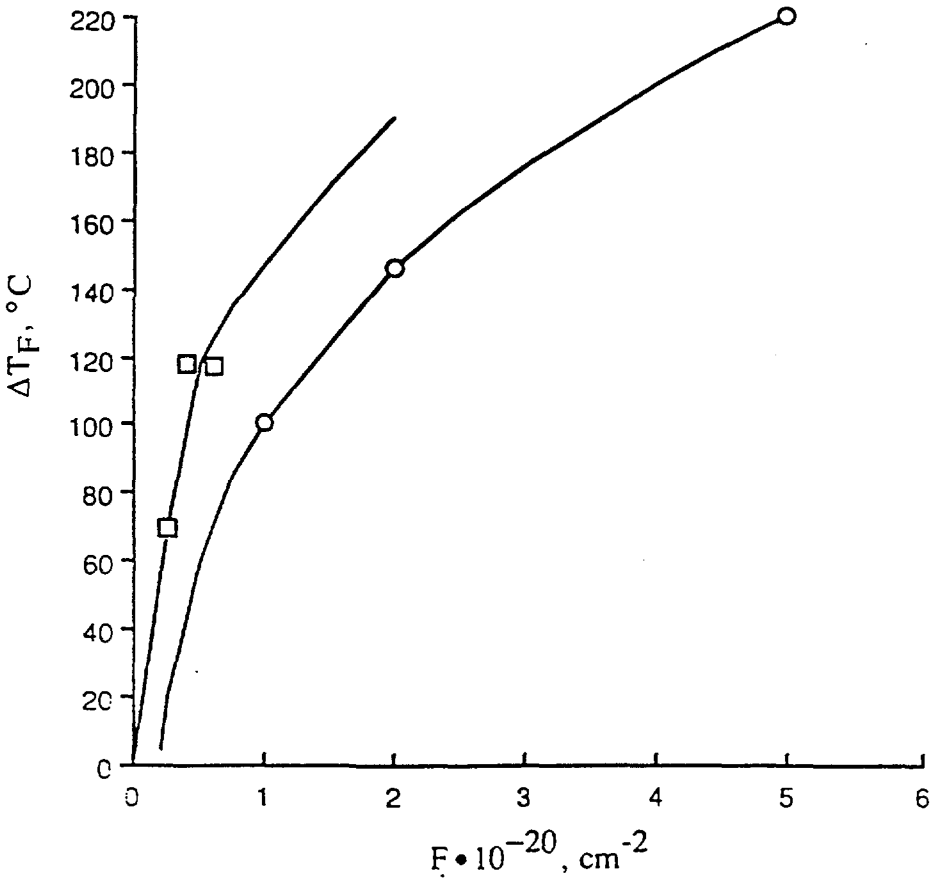
1. Normal loading of core
2. Loading of core with shielding assemblies

Fig. 7 – Integrated energy spectrum of fast neutrons E > 0.5 MeV at irradiation locations of WWER-440 surveillance specimens



1. Normal loading of core
2. Loading of core with shielding assemblies

Fig. 8 – Integrated energy spectrum of fast neutrons at irradiation locations of VVER-440 surveillance specimens



P = 0.028%
Cu = 0.18%
□ – $\varphi \approx 4.10^{11}$ $cm^{-2}s_{-1}$, RNPP-1
○ – $\varphi \approx 4.10^{12}$ $cm^{-2}s^{-1}$, ANPP-2
△ – $\varphi \approx 7.10^{12} cm^{-2}s^{-1}$, MTR
– calculated curve in accordance with norm

$$\Delta T = 800\ (\%P + 0.07\%Cu)\ (F/F_0)^{1/3}$$
$$= 800 \times 0.04\ (F/F_0)^{1/3}$$
$$= 32\ (F/F_0)^{1/3}$$
$$E > 0.5\ MeV$$

Fig. 9 – Radiation embrittlement of weld metal at irradiation temperature of 270° C showing fluence rate effects