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ASSESSMENT OF FERRITIC STEELS FOR STEADY-STATE FUSION REACTORS*

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Abstract

Ferritic steels have been widely used in industrial applications for their excellent physical and mechanical properties. Alloying and heat treatment has led to superior combinations of strength and ductility over a wide range of temperature. Recently, it has been experimentally shown that this class of alloys resist neutron (and ion)-induced void swelling. This, in turn, has aroused great interest for the use of ferritic alloys in the nuclear industry. The merits and advantages of utilizing ferritic, or, more broadly, chromium-based steels in steady-state fusion reactors are disucssed. General mechanical property change correlations are developed along with specific design equations. For steady-state fusion reactors, where mechanical fatigue does not play a major role, three properties limit the component lifetime: irradiation embrittlement at low temperatures; void swelling at intermediate temperatures; and high-temperature creep-rupture. The design equations are applied to analyze the SAYTR D-D Cycle Tandem Mirror reactor. It is shown that the shift in the Ductile-to-Brittle Transition Temperature (ΔDDBT) due to neutron bombardment dictates a lower temperature limit of about 350°C. An upper temperature limit of about 520°C is caused by the slow creep-rupture process.

I. INTRODUCTION

The development of new alloys that are specifically tailored to meet the challenging fusion reactor environment is of prominent importance in achieving practical fusion energy. One such candidate alloy system now being considered for fusion applications is a class of ferritic steels. The rapidly expanding irradiation data base has been encouraging in some areas.

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However, recent data [1] and analysis [2] have pointed out the possibility of potential problems associated with the use of ferritic steels at low temperatures. It is important, therefore, to develop an analysis strategy to better judge the feasibility of using ferritic steels in fusion applications.

Iron-chromium alloys have been extensively used in non-nuclear applications because they have excellent strength and toughness at low temperature. Morevoer, creep resistance in combination with the ability to withstand corrosion attack make them economically competitive with austentitic steels at high temperature. Iron-chromium alloys have excellent resistance to swelling when bombarded with heavy ions or neutrons [3-8] leading to interest in their application as cladding and duct material for breeder reactor [3-8] and as first-wall and blanket structural material for fusion reactors [9,10].

In order to give carbon steels excellent creep strength, molybdenum (0.5%) is added as an alloying element. However, an addition of 1% Cr is to improve the ductility and to eliminate graphitization [10]. Efforts toward optimization of properties has resulted in the use of 1% Mo for creep-strength and 2 1/4% Cr for oxidation resistance up to 550-600^{o}C [11]. For more aggressive chemical environments high chromium alloys (such as 5% Cr - 1/2 Mo, 7 Cr-1/2 Mo, 9 Cr-1 Mo, 12 Cr-1 Mo (HT-9) have been developed. For lower temperatures and non-corrosive environments, alloys are used with low chromium and molybdenum concentrations.

The combination of chromium concentration and heat treatment can lead to either a martensite or bainite structure. Sometimes, this class of materials are referred to as "martensitic", rather than "ferritic" steels. Undoubtedly, compositional differences and heat treatment procedures are expected to affect the radiation response of such alloys. In our work, we de-emphasized differences in the radiation response due to either composition or heat treatment. Design equations are developed using the data on a variety of Fe-Cr steels. The purpose is to provide an overall view of the merits and limitations of these alloys.

II. Fe-Cr ALLOYS IN THE "SATYR" FUSION REACTOR

One way to analyze the performance of Fe-Cr alloys in a fusion environment is to study the characteristics of the structure in a particular reactor design. The UCLA design, SATYR [12], is a convenient means to achieving this end. SATYR is a tandem mirror device based on a deuterium-deuterium fusion

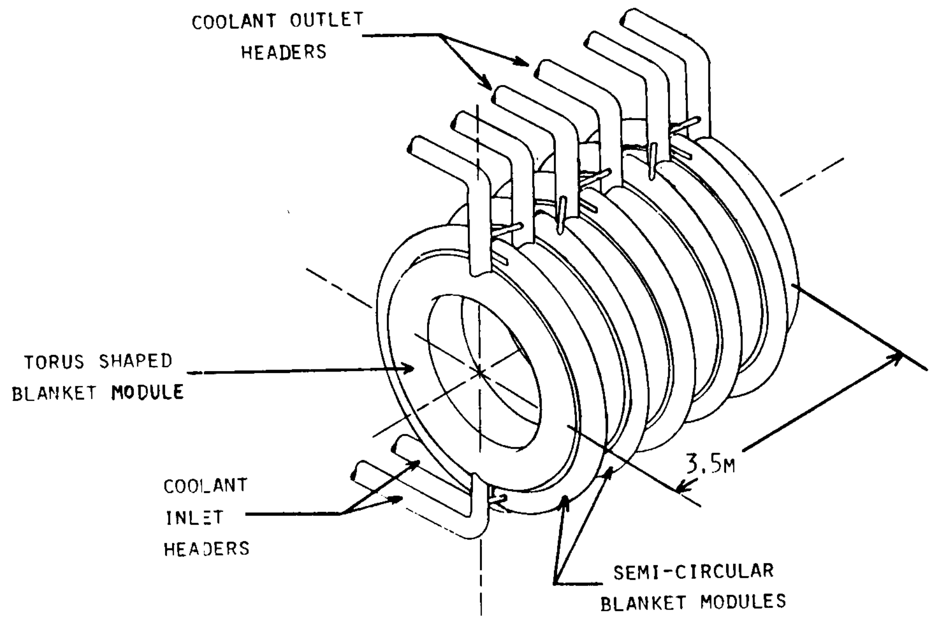

Fig. 1. An isometric view of the ferritic steel blanket structure of the D-D SATYR fusion reactor.

fuel cycle. The reader is referred to reference [12] for details.

Fig. (1) shows a pressure vessel blanket design for SATYR, where a 2.2 cm thick ferritic steel toroidal shell is used. The radiation damage parameters are shown in table (I). The boiling water concept employs a Sintered-Aluminum Product (SAP) alloy for low long-term radioactivity requirements. A specific ferritic alloy AISI 4130 is used in the helium-coolant concept. Its composition (0.28% C, 0.4% Mn, 0.2% Si, 0.8% Cr, 0.15% Mo, Fe) offers the following advantage.

(1) Elimination of phosphorus and sulfer reduces the initial DBTT before irradiation, and increases the upper shelf energy.

(2) The absence of nickel and the possibility of replacing the 0.15% Mo by V for high temperature strength result in a substantial reduction in the long-term radioactivity of the structural material.

(3) The low chromium content is an advantage from a resources viewpoint when compared to HT-9 (~12% Cr). Approximately 99% of the world's chromium reserves are located in Zimbabwe and the Republic of South Africa [10].

In general, Fe-Cr alloys (ferritic steels) offer many advantages that are desirable from a design standpoint. The motivations for using such alloys are:

(1) Low void swelling under both ion and neutron bombardment.

(2) Thermal stress resistance due to the relatively high thermal conductivity and small thermal coefficient of expansion.

(3) The BCC crystal structure of ferritics is more resistant to high temperature helium embrittlement, when compared to FCC crystal structures.

(4) Ferritic steels are less costly than austenitics.

(5) Manufacturing and fabrication technologies are well developed.

(6) Slight compositional modifications, such as the elimination of Ni and Mo, can lead to a significant reduction in the long-term radioactivity.

III. DEVELOPMENT OF DESIGN EQUATIONS

We present here empirical design equations for the change in the mechanical properties of materials as functions of the irradiation environment. A combination of the existing irradiation data base and theoretical understanding is used to develop such equations. As pointed out in the introduction, fatigue life analysis is not included and property changes that are only relevent to steady-state devices are considered. Design equations are developed for void swelling, the shift in the DBTT, and thermal creep-rupture at high temperature.

III.1 Swelling

Recent fast reactor irradiation experiments have shown low swelling. The work of Little and Stow [14] divides the swelling behavior of ferritic alloys into three materials categories: (a) pure iron and mild steels; (b) binary iron-chromium alloys; (c) commercial ferritic and martensitic steels. The temperature dependence of the swelling of the zone-refined iron irradiated to a displacement dose of 30 dpa over the temperature range of 380-460^{o}C defines a swelling peak ($\Delta V/V \approx 1\%$) centered at ~420^{o}C Another higher temperature peak of a smaller magnitude is found to be centered around 510^{o}C [14].

All the binary iron-chromium alloys exhibit well defined swelling peaks following irradiation to 30 dpa at temperatures in the range of 380-460^{o}C [14]. The peak swelling temperatures are found to be essentially independent of chromium concentration and coincident with the equivalent peak present in pure iron. The magnitude of the swelling is found to be a function of the chromium concentration, with a minimum around a chromium content of ~5%. The most noticeable feature apprant in many of the microstructures of commercial ferritic and martensitic steels is the marked coarsening and redistribution of the $M_{23}C_6$ precipitates. The preliminary results of Little and Stow [14], and the results of Smidt et al. [17] confirm the conclusion of low overall swelling of such alloys.

The following design equation for void swelling has been developed by us from the neutron and ion irradiation data [13-17].

$$\frac{\Delta V}{V}(\%) = \exp\left\{-\left(\frac{T-T_p}{W}\right)^2\right\}\left\{0.036\delta-0.074\right\}\left\{\phi(Cr)\right\} \quad (1)$$

where,

$$\phi(Cr) = \begin{cases} 0.067Cr^2-0.457Cr+1.0 & (Cr<5\%) \\ 0.037Cr+0.237 & (Cr>5\%) \end{cases} \quad (2)$$

Cr = Chromium concentration, %.

In equation (1) W (=59^{o}C) is the full width of the assumed Gaussian distribution, δ is the displacement dose (dpa), T is the irradiation temperature (^{o}C), and T_p is the peak swelling temperature given by:

$$T_p(^{o}C) = 420 + 7.27\ln\left\{\frac{P(dpa/s)}{10^{-6}}\right\} \quad (3)$$

Equation (1) is then used to determine the first wall structure lifetime, given a swelling limit. A more detailed approach to the problem should consider analysis of the stresses induced by swelling gradients, and the interplay with irradiation creep [16,17], his approach is beyond the scope of the present study. Fig. (2) shows the lifetime of the SATYR structure as function of temperature for swelling limits of 2%, 5% and 10%. We will further assume that a $\Delta V/V$ of 5% is reasonable. The resulting minimum lifetime is 50-60 years.

III.2 Radiation Embrittlement

Brittleness is a normal property of most solids, including metals and alloys at low temperature. Only face-centered-cubic

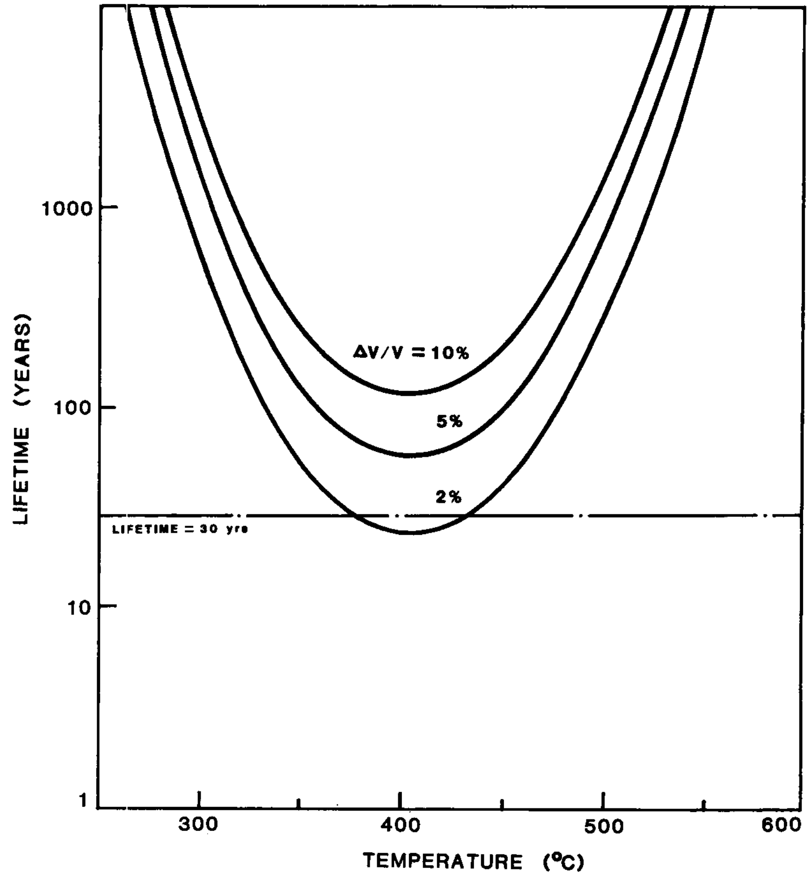

Fig. 2. Blanket structure lifetime as a function of the irradiation temperature for various levels of swelling strain limits.

(fcc) metals are commonly ductile at low temperature. One of the most serious concerns is the embrittling effect of neutron irradiation because it produces brittle fracture even at relatively high temperatures. A measure of brittle fracture is the transition temperature between brittle behavior at low temperature and ductile behavior at high temperature. The ductile-to-brittle-transition-temperature (DBTT) can be defined as the intersection of the fracture (σ_1) and yield (σ_y) stresses, when each is plotted in Fig. (3). At low temperature, the material undergoes brittle fracture before reaching the yield point. The effect of irradiation is to harden the alloy by introducing vacancy and interstitial loop obstacles to dislocation motion. The effect is a significant increase in the yield stress ($\Delta\sigma_{y\phi}$), and only a small elevation of the fracture strength

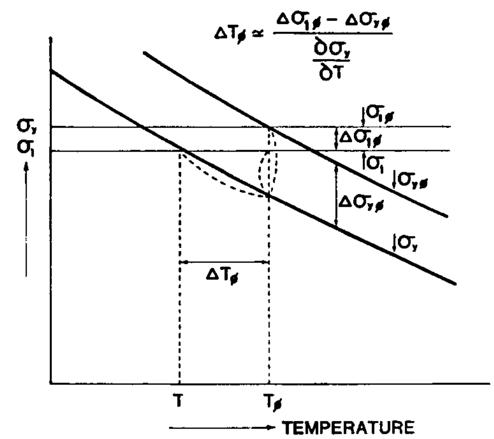

Fig. 3. A schematic illustration of the concept of the shift in the DTBB. σ_y = yield stress, σ_1 = fraction stress, ϕ for fluence, and ΔT is the DBTT shift.

($\Delta\sigma_{1\phi}$). The shift in DBTT can then be geometrically defined as (see Fig. (2)):

$$\Delta DBTT = \Delta T_\phi \simeq \frac{\Delta\sigma_{1\phi} - \Delta\sigma_{y\phi}}{\frac{\partial\sigma_y}{\partial T}} \tag{4}$$

A rather consistent trend-band for increases in DBTT with integrated neutron fluence has been observed for several irradiated steels. However, the rate of embrittlement decreases with each added increment of neutron dosage. Data from the LWR surveillance programs shows a declining trend in the inrease in DBTT with integrated fluence [18,19]. The data indicate that saturation effects are important and that saturation in DBTT occurs at a lower fluence for a higher temperature. It must be remembered, however, that the effects of hydrogen and helium gas produced under typical fusion reactor conditions are not included in such studies. Recent neutron results on the embrittlement behavior of the high chromium alloy, HT-9, show a tendency toward temper embrittlement at 538°C [1]. A DBTT shift of 108°C is measured at an irradiation temperature of 419°C.

Composition, heat treatment and type of steel are known to influence the neutron irradiation embrittlement response of steels [20]. For the purpose of the present analysis, we will ignore such effects, and assume the validity of one design equation regardless of compositional variations. We will further

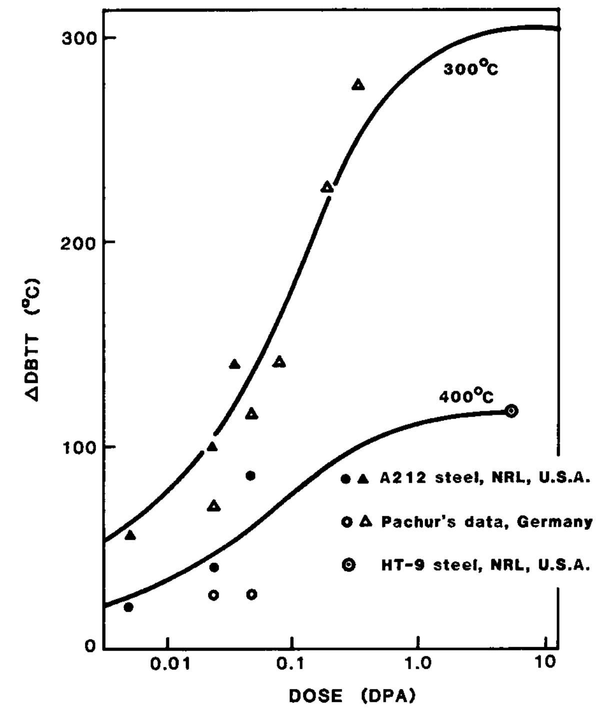

Fig. 4. The shift in the DBTT as a function of irradiation dose.

consider that the DBTT is about 0^{o}C before irradiation. Again, composition and heat treatment affect the initial DBTT (within ~ $\pm$ 50^{o}C). A design equation for the shift in DBTT, that is based on hardening theories [21], can be then written in the form:

$$\Delta DBTT = \frac{1.87 \times 10^4}{T-288} \left\{ 1 - \exp\left[- \left(\frac{4T-350}{T} \right) \delta^{1/2} \right] \right\} \qquad (5)$$

This equation is valid above approximately 250^{o}C. The data, together with the fitting equation, are shown in Fig. (4). The shift in the DBTT as a function of the irradiation temperature for the SATYR concept is shown in Fig. (4). It is evident that saturation of embrittlement is achieved after about one year of irradiation. On the same figure, the straight line, DBTT=T, is shown. In our analysis, the fialure of the structure results when the DBTT is nearly equal to the operating temperature.

It is also possible to relate the DBTT to crack growth and propagation rates [22]. Fig. (5) shows that embrittlement is an extremely sensitive function of the operating temperature.

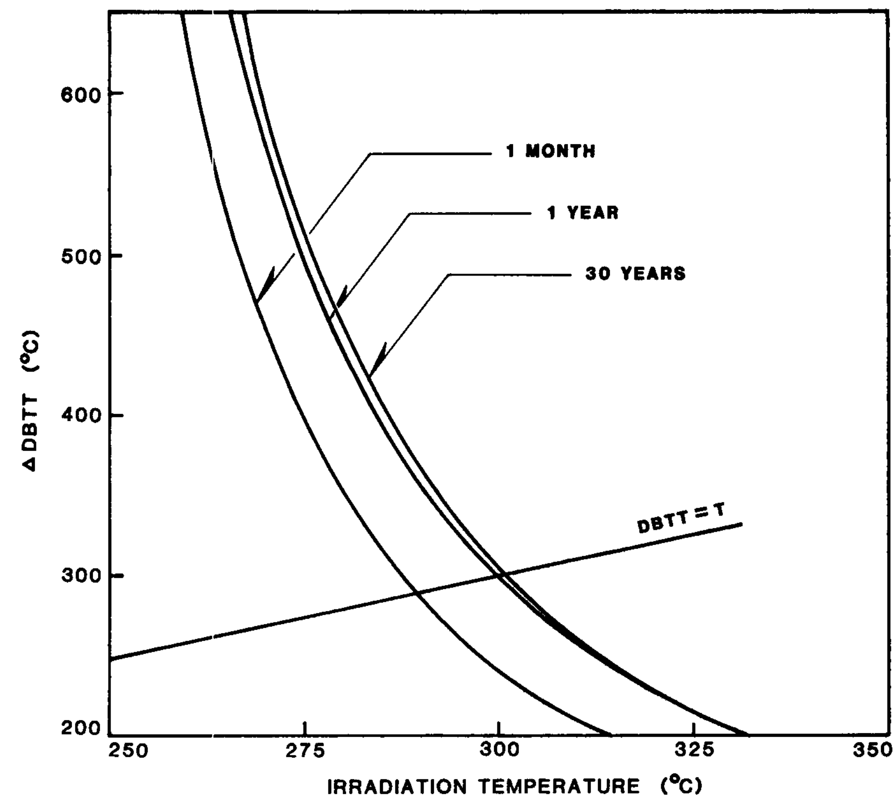

Fig. 5. ΔDBTT as a function of irradiation temperature. Failure is defined where DBTT = T.

Above approximately 325°C, ΔDBTT will never reach the temperature of the structure.

III.3. Creep-Rupture

The slow deformation of structural components under a small applied stress usually leads to failure at high temperatures. Diffusional and/or slow plastic deformation processes result in this "creep-rupture" behavior. It has been experimentally observed that irradiation accelerates the time-to-rupture (t_R) in austenitic steels. There are indications that the acceleration of failure due to the accumulation of helium on grain boundaries may not be operating in ferritic alloys [23]. The high temperature strength of ferritic steels is primarily due to the precipitate stability. Rosenwasser [9] concluded that a martensitic Cr-Mo steel (HT-9) would have favorable properties up to 520°C. However, Klueh [10] showed

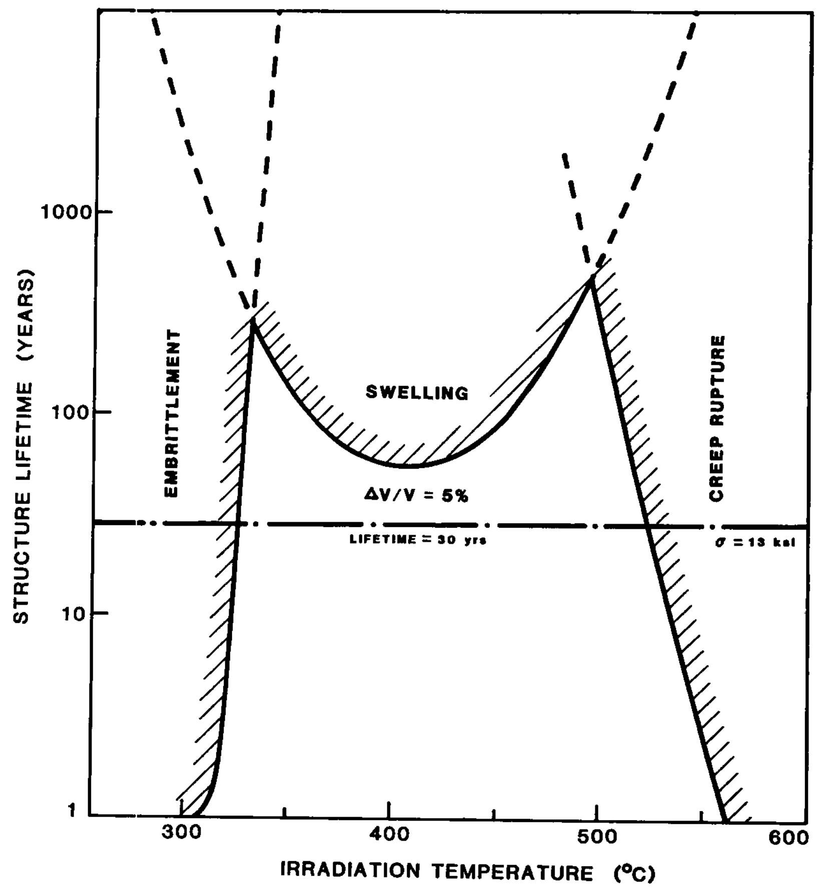

Fig. 6. Overall lifetime analysis of the SATYR blanket. The lifetime is given as a function of irradiation temperature.

that very little difference exists between various Cr-Mo alloys out to ~10,000 hours rupture lifetime. This conclusion was reached by comparing creep-rupture data at 538°C for 2 1/4 - 1 Mo steel [24-26], Sandvik HT-9 [27], and modified 9 Cr-1 Mo [10] steel in the appropriate heat-treated condition.

The time-to-rupture (t_R) is a function of both the applied stress, σ, and the temperature. The following is our proposed design equation, that is primarily based on the data available for AISI 4130 steel [28]:

$$t_R = \left\{ \frac{\sigma}{\exp(37.8-0.0243T-\frac{Q}{MRT})} \right\} \left[\frac{919.1}{T^{1.1}} - 0.756 \right]^{-1} \qquad (6)$$

where m = 4.5, Q = 100 KCal/Mol, T (^oK), σ (ksi), and t_R (hours). Equation (6) also reproduces Klueh's data for 2 1/4 Cr - 1 Mo steel [10].

IV. CONCLUSIONS AND RECOMMENDATIONS

To realize the potential of ferritic alloys in fusion reactors, an integrated structure lifetime analysis must be performed. We will use the specific radiation damage parameters of the SATYR reactor, and the proposed design equations for a preliminary assessment of these alloys. Fig. (6) shows the SATYR blanket lifetime as a function of the irradiation temperature. The following considerations are utilized:

(1) Swelling Limit $\frac{\Delta V}{V} = 5\%$

(2) Maximum stress, σ = 13 ksi

(3) Low temperature failure; DBTT = T

It is clear that a maximum temperature window of about 200^{o}C is permissible for the use of ferritic alloys as structural materials. Moreover, the lowest temperature for the structure must be above ~350^{o}C due to the severe radiation embrittlement below this temperature. The existence of an operational temperature window is not unique to ferritic alloys. However, the low temperature limit is of great concern during a reactor shutdown.

Our analysis shows the need to develop fusion reactor blanket designs that can cope with the irradiation embrittlement problem at low temperature. Flow reversal, intermittent annealing, and auxiliary heating systems during shut-downs are examples of methods to get around the DBTT problem. It should be emphasized that alloy development of ferritics must address the solution to the DBTT problem. Other important considerations for the development of a ferritic alloy for fusion are: (1) lowering the chromium content for conserving resources; (2) elemental tailoring by removing nickel and molybdenum for radwaste disposal.

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