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A REVIEW OF LIFETIME ANALYSES FOR TOKAMAKS

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A REVIEW OF LIFETIME ANALYSES FOR TOKAMAKS*

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System studies have vividly shown that economic fusion power can only be achieved from the use of long lived components. Lifetime goals of 90 MW-yr/m^2 for a tokamak based power plant should be established. The stresses generated in a first wall module are a complex function of its geometry, the chosen structural material and the tokamak burn cycle characteristics. A formalism based on the foundation ASME Code Case 1592 has been established. Methods of incorporating some of the changes expected from irradiation are discussed. The cyclic stress pattern imposed by tokamak operation is expected to cause fatigue related properties to govern the life of the structure. Stress assisted bubble growth is suggested as the possible critical mechanism in establishing the stress-to-rupture life of a fusion first wall component.

1. INTRODUCTION

To be a viable part of the nation's energy supply system, fusion based reactors will have to prove themselves both as reliable and an economical alternative energy sources. A very important factor in whether or not fusion reactors ever become important producers of electricity is the expected lifetime of key reactor components. This paper is an attempt to review the status and results of the lifetime analysis work performed, to date, for tokamaks.

System study [1,2] results are useful for putting into perspective the importance of achieving long lived designs. Figure 1(a) and 1(b) compare the operating costs as a function of neutron wall loading assuming the existence of a 10 MW-yr/m^2 design in 1(a) and a 30 MW-yr/m^2 design in 1(b). The achievement of relatively high 2-3 MW/m^2 wall loadings has been increasingly widely recognized [3] as a key means of reducing the size and, in turn, the capital cost of a tokamak-based reactor.

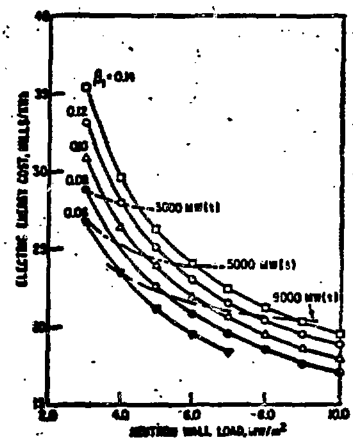

Fig. 1(a)

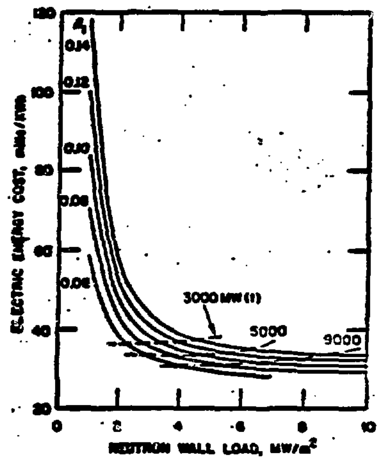

Fig. 1(b)

The main reason for the economic incentive shown in Fig. 1 for longer lived designs is the long time required for rebuilding of a tokamak system. To date, no one has been able to identify a means of in-situ repair so that the tedious and laborious disassembly of large, welded, radioactive components must be envisioned. A length of 80 days to rebuild a tokamak system was used in analysis that led to Fig. 1. Viewed in another way, the lost revenue from one days operation of a 1000 MWe plant where electricity is sold at five cents a KW hr is 1.2 million dollars.

The effect of planned maintenance, alone, is sufficient to justify (indeed dictate) the development of a long lived design. However, there is additional incentive for such a development from a consideration of the relationship between expected component lifetime and the occurrence of unscheduled shutdowns. Failure is by its very nature a statistical phenomenon. It is impossible to say precisely when a component will fail, as one is restricted to a

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statement of the sort that there is a 95% probability that failure will not occur before a certain point in time. Putting this in simplest terms, if one assumes a functional form such as e^{-t/t_L} where t_L is the expected component lifetime and t is the actual life, then a certain probability of failure can be ascribed to a particular component. The actual number of failures to be expected in the system will be the product of the number of components times this probability. Many fusion reactor system designs involve hundreds or even thousands of first wall modules. For a short lived design the number of projected component failures would be much higher than for a long lived component over the whole lifetime of the reactor plant, t_r i.e. the number of expected failures is $\alpha\ t_r/t_p \times e^{-t_p/t_\ell}$ where x is the number of modules in the design and t_p is the length of period between scheduled maintenance operations. Clearly higher values of t_ℓ results in a reduction in the number of expected failures during the life of the plant.

Thus system studies on the economics of fusion power indicate the incentive to achieve an expected component lifetime equal to that of the reactor plant or 30 years. To be useful, this life must be achieved for wall loadings of from 2-3 MW/m^2, thereby creating an overall goal of 90 $MW\text{-}yr/m^2$.

2. LIFETIME ESTIMATES

Lifetime estimates are dependent on criteria developed in three categories. The first of these includes those things that actually lead to failure of the component, either through leakage of the coolant or by a complete failure of an element. Creep-fatigue and flaw propagation would fall into this most obvious class.

The second class involves limits on dimensional change that would cause some planned operation to become impossible. Swelling and creep are the main considerations in this category. Setting limits for allowable dimensional change is very design dependent and can only be done after all planned maintenance operations are well established. A more stringent restriction on the amount of allowable swelling may well result from an analysis of the effect of the stresses induced by the swelling gradients on class one properties.

The third class comprises limits on deformation that experience has shown are important to prevent the structure from failing in an unspecified mode. Thermal ratcheting limits is an important example in this category.

To begin to analyze the performance of a reactor module in terms of these three categories of properties, one must first characterize the environment in terms of the expected stress and thermal history.

3. THERMAL-HYDRAULIC CONSIDERATIONS

Any detailed estimate of the lifetime of a tokamak first wall module must be based on a complete thermal-hydraulic analysis, such as is included in references [4] and [5]. For the purposes of this review, certain fundamental limitations on performance result directly from the selection of a structural material and a wall thickness. The maximum temperature of the first wall module can be no lower than the coolant temperature plus the quantity $W_s t/k$ where W_s is the surface heat flux, t is the wall thickness, and k is the thermal conductivity of the wall. The actual wall temperature will, of course, be higher since the heat transfer between the wall and the coolant stream will never be perfect. If one assumes as a basis of comparison a coolant stream temperature of 225°C, the relationship between temperature and wall thickness for three candidate materials are as presented in Table 1. While each of these three candidates, Type 316 stainless steel, a stabilized 9 Cr ferritic stainless, and a titanium alloy (Ti-6Al-4V) have an upper temperature limit of about 500°C in a fusion reactor environment, it can be seen that widely different wall loading limits are strictly from the radial temperature gradient. Titanium alloys suffer from the fact that they are restricted to service wall loadings below $\sim 4\ MW/m^2$.

The temperature gradients developed within the wall result in secondary stresses of both thermal and radiation swelling origin. The magnitude of these stresses will be considered in the next section.

4. SOURCES OF STRESS

The results of an analysis for one tokamak wall design is presented in Fig. 2 to illustrate the changing stress pattern that must be expected with time. The primary stresses existent in a tokamak system are a result of the coolant pressure and magnetic forces if a magnetic structural material such as ferritic stainless steel

Table 1.

First Wall Alloy	T Across 1 cm wall, at 1 MW/m^2, °C	Allowable* Wall Loading on a 3 mm Wall for Maximum Temperature	
		500 °C	400°C
316	108	8.5	5.4
Ti-6Al-4V	192	4.8	3.0
Ferritic, 9-Cr	86	10.6	6.7

* Assumes coolant inlet temperature of 275°C.

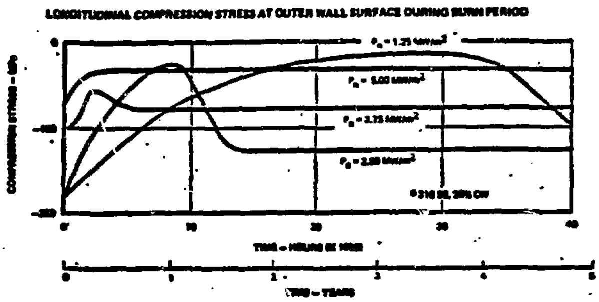

Fig. 2.

is used. The coolant pressures range from atmospheric for a falling bed concept to as high as 13.7 MPa for some water cooled designs. Representation pressures are shown in Table 2 for each of 5 coolant choices as developed by recent blanket studies [6]. The magnetic force that results from the use of a ferromagnetic structural material can also result in significant stresses. Recent calculations [7] for a 9-Cr stabilized ferritic stainless steel structure indicates stresses of ∿ 1 atm will be imposed if this material is used.

A possible source of cyclic primary stresses are pressure fluctuations due to turbulence in the coolant that could contribute to fatigue failure in first wall structures. In addition, plasma disruptions may cause electrical currents to flow in first wall structures resulting in internal forces. These must be combined with other primary loads in determining structural thicknesses. These forces and their resulting dynamic effects on stress have been shown to have a significant effect on structural design

Table 2.

Coolant	Typical Pressure MPa (psi)
Helium	5.2 (750)
Lithium	0.35 (50)
Molten Salt	1.4 (200)
Water	13.7 (2000)
Falling Bed	0.14 (20)

and on resulting component life [8]. Because these disruptions may occur frequently (assumed once each 10 cycles in Ref. [8]), they may have to be considered a normal loading condition.

In addition to these primary stresses as the onset of operation, the incident electromagnetic and particle fluxes that comprise ∿ 25% of the total fusion energy cause large compressive thermal stresses to be generated in the first wall surface facing the plasma and a corresponding tensile stress to be generated on the coolant side of the wall. The initial level of these secondary stresses depends on the material selection, the wall loading, and the wall thickness. Representative values for several materials are shown in Table 3.

Table 3.

$$\sigma_{TH} = \alpha E\,\Delta T = \alpha E\,\frac{W}{K}\,T$$

Radial thermal stresses generated in a 3 mm thick first wall

Structural Material	Yield Strength at 500°C MPa (psi)	Thermal Stress at Wall Loading 1	3	5
316 stainless steel	195 (28,000)	72 (10,400)	216	360
9 Cr stabilized ferritic	310 (45,000)	42 (6,100)	126	210
Titanium	480 (69,600)	34 (4,930)	102	170

Figure 3, as taken from the work of Muhkerjee, et al. [9], illustrates the importance of including the anelastic component in the analysis as it results in a recoverable reduction in the imposed stress of about 15% in the example case of an austenitic stainless steel structure.

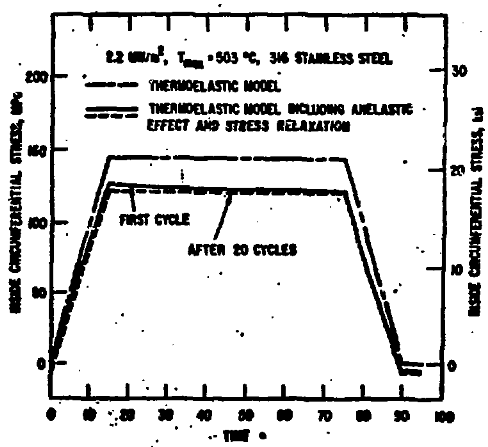

Fig. 3.

This stress is then further relaxed by thermal and radiation enhanced processes that result in permanent deformation of the structure so that eventually the thermally induced secondary stresses are low during the reactor burn and approach the thermoelastic stress less the anelastic recovery during the dwell portion of the burn cycle. Since it is this stress that has been found to govern the fatigue and flow propagation estimates, the importance of the inclusion of the anelastic recovery is evident since crack growth rates are a function of the stress level raised to an exponent ranging from 2 to 4. Thus the reduction of the estimated stress by 15 percent could result in a factor of 2 reduction in the estimated crack growth rates.

The thermoelastic stress is immediately reduced by anelastic effects and as a function of time by thermal and radiation enhanced stress relaxation.

It should be noted that radiation enhanced stress relaxation is particularly effective in reducing the level of stress present during the burn due to both its amplification of the process and its linear dependence on stress. This latter characteristic allows the relaxation process to proceed to much lower stresses than would be reached by the much more highly stress dependent thermal stress relaxation process (Fig. 4). Radiation enhanced stress relaxation eventually causes the plasma side of the wall to be under rather high tensile stress during the dwell period which may have important implications for the fatigue properties as surface damage due to blistering and sputtering may aid both crack initiation and propagation under tensile loading. A very important aspect of the stress relaxation is that it tends to minimize the importance of creep rupture as a life limiting property since the high thermal stresses are only imposed for the relatively short periods between burns and at times when wall temperatures are lower and the neutron flux is absent. This has the effect of translating the tokamak first wall problem into one where fatigue and flaw propagation considerations are of increased importance. In addition to the thermally induced secondary stresses, another important source of secondary stress is due to differential swelling. As shown in Fig. 2, for a swelling material such as Type 316 stainless steel, gradients in void volume within the material result in rather high stresses at long times within the material. Also included for purposes of contrast are the expected stresses for a ferritic stainless steel which is assumed not to swell during neutron irradiation.

The gradients in swelling can result in rather complex stress distributions in a structure constructed of a material such as Type 316 stainless steel. An elastic analysis is presented as Fig. 5 to dramatize this point.

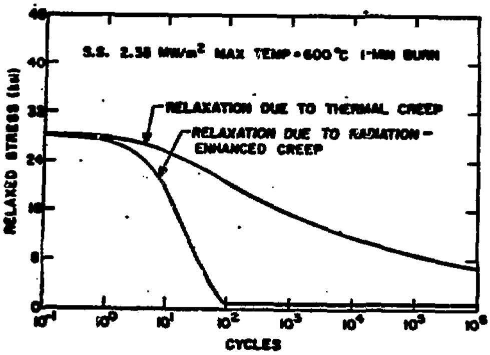

Fig. 4.

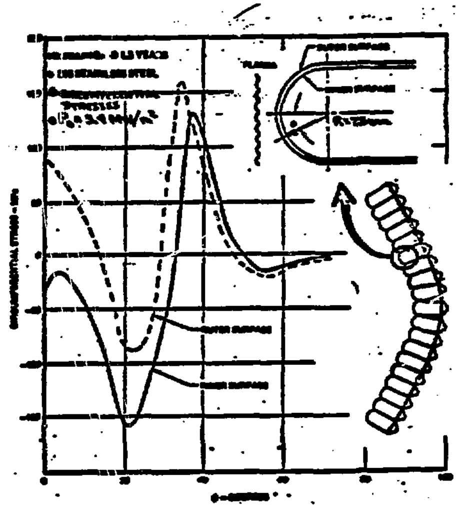

Fig. 5.

Design dependent stress distributions will also result from coolant pressures and temperature distributions. Stress histories in the first wall structures, including effects of irradiation swelling and creep, have been calculated on an inelastic basis in references [9] and [10]. Results of the analyses from reference [9] are shown in Fig. 2. These stress histories indicate the relaxation of compressive stresses with time due to creep and subsequent increase in compressive stresses due to swelling. This relaxation of compressive stresses cause residual tensile stresses to result during the "non burn" portion of the cycle. These stress histories were calculated for a uniaxial element in the first wall. They are shown for various levels of neutron wall loading and include a temperature effect (temperature increases with increasing wall loading) as well as a time dependent irradiation effect. The results shown in Figs. 2 and 5 provide an indication of the complexity of first wall stresses due to their time dependent nature, cyclic nature, and the redistribution of stresses which will occur.

In depth analysis of a near term doublet shaped plasma chamber has been reported [8, 11, 12 and 13]. Because of the rather short operating life (≈ 70 days) of this machine, unirradiated material properties were used. However, a detailed analysis of thermal stresses and plasma disruption stresses was included in predicting wall life. A resulting cyclic stress history for an Inconel 625 wall is shown in Fig. 6. Results included effects of plasticity to determine thermal stress distributions and optimum wall section geometries as well as a dynamic analysis to determine response of the wall to plasma disruptions. Cyclic stresses due to both effects have a significant effect on first wall life.

5. DESIGN APPROACHES TO REDUCE THE LEVEL OF INDUCED STRESSES

The designer has at his disposal several methods of reducing the level of thermal stresses imposed on the first wall of a tokamak.

One of the most important considerations deals with the use of restraint as a means of

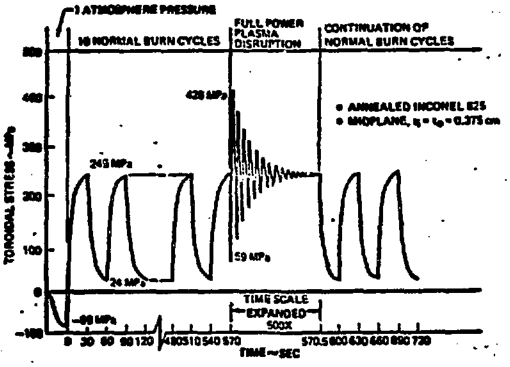

Fig. 6.

limiting bending stresses. For example, if the hemispherically capped first wall module were not rigidly clamped, then the thermal stresses would be as shown in Fig. 7 rather than as in Fig. 3. The use of restraint in system design must be approached carefully when dealing with structural materials that are dimensionally unstable under irradiation. For example, Wolfer and Watson [14] found it was necessary to avoid all restraint when designing a graphite first wall structure due to the stresses that would otherwise be generated in this dimensionally unstable material.

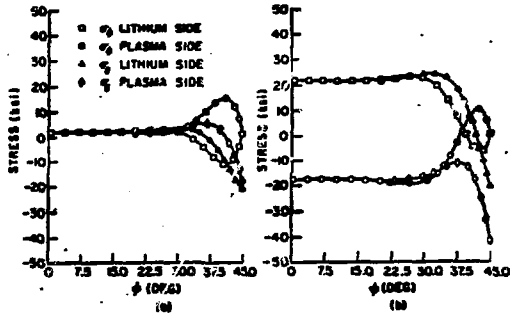

Fig. 7.

A number of ways to minimize the temperature gradient that is established by the surface heating due to the electromagnetic and particle flux from the plasma have been devised. These generally take two forms, either reducing the fraction of energy that is deposited on the wall through the use of a divertor to intercept the particle transport, or to spread out the time that energy is radiated to the wall so that some heating continues during the dwell period by means of a liner or limiters.

The effects of a magnetic divertor on the temperature and temperature gradients generated in the first wall of a tokamak have been analyzed by B. Misra and V. Maroni [4]. Their results reflect the fact that over half of the heat flux incident on the first wall surface is due to particle transport so that the use of a divertor effectively allows a factor of two higher wall loadings to be imposed on the system for a given allowable thermal stress level.

The application of a liner as a protective first wall device has been analyzed by Kearney [5] et al. A liner is usually included in tokamak designs to protect the first wall from plasma disruptions that result in a sudden deposition of the plasma's energy onto the first wall. However, this device, which is usually conceived to be fabricated from a coated graphite or silicon carbide, has the additional benefit of flattening the rate of energy deposition onto the first wall during normal operation. Figure 8 compares the expected first wall surface temperature for the case when a liner completely covers the inside of the tokamak, when refractory limiters comprise 10% of the surface area with that for a bare wall as calculated by Krazinski and Smith [15]. The use of a liner was calculated [5] to increase the crack growth limited lifetime from 2.6 to 89 years.

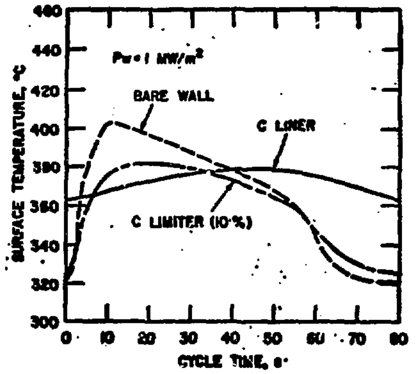

Fig. 8.

It should be noted that after stress relaxation has worked to limit the presence of the thermally induced stresses to the downtime between burns, ingenious methods to generate energy in the blanket such as developed for the Numak [3] reactor might serve to aggrevate the problem. The Numak approach uses the solidification of a Pb-Li eutectic during the dwell period to even the temperature swing seen by the turbine. This could serve to also raise the temperature of the coolant side of the first wall, thereby causing an increase in the stress levels already present since stress relaxation during the burn has caused the coolant side to be under compression during the dwell part of the cycle.

6. LIFETIME CRITERIA

In developing a method for predicting the lifetime of a fusion reactor first wall module, the logical foundation to build upon is the ASME Code Case 1592 for Class 1 Components in Elevated Temperature Service [16]. C. K. Youngdahl and D. L. Smith [17] have developed this approach for a fusion reactor first wall in such a way as to relate the maximum allowable surface heat flux as a function of the average wall temperature for a number of candidate structural alloys.

These authors suggest a two step process. In the first step the structure is analyzed for those processes that result in actual failure of the component such as fatigue, stress-to-rupture and excessive primary loading. Based on the use of Code Case properties for the unirradiated materials where they existed, and estimates based on available experimental data when they didn't, the authors were able to project good performance in this first step for all materials from annealed Type 316 stainless steel (the worst performer) to vanadium alloys (the best) and still comply with the Code Case requirements on tensile, stress-to-rupture and fatigue properties. These authors' results for Type 316 stainless steel are shown in Fig. 9 along with an additional plot of the maximum wall loading possible at a given wall temperature based on the simplified analysis presented in the thermal hydraulics section of this paper. It can be seen that these temperature limits reduce the attractiveness of low temperature systems.

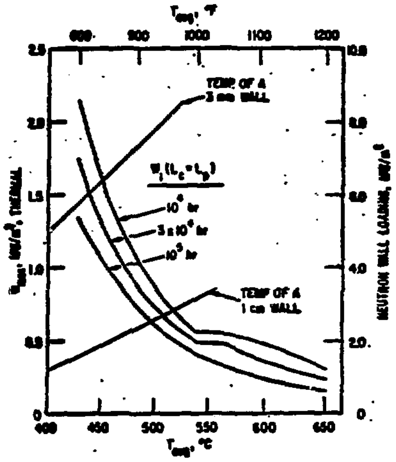

Figure 9 was developed by first establishing the allowable module wall thickness from S_{mc}, the primary membrane stress limit through the relation $\delta = PR/S_{m,t}$. $S_{m,t}$ is the lesser of the time independent tensile limits and the time dependent stress-to-rupture related properties. P is the coolant pressure and R is module radius in the above relationship. This value of wall thickness was then used in a fatigue based equation

$$\epsilon_t\ (L) = \frac{\alpha \delta W}{2k(1-\nu)}$$

where $\epsilon_t(L)$ is the code given allowable strain to achieve the desired lifetime, W is the thermal wall loading (equals $\sim$ 25% of the neutron wall loading), α is the thermal expansion coefficient, k is the thermal conductivity, and ν is Poisson's ratio.

In the second step of the analysis, what were previously termed category three criteria are examined. These deal with deformation limits on thermal ratcheting and thermal creep whose main intent is to minimize the possibility of initiation of cracks within the material. A useful way of analyzing for the effects of ratcheting in a thermally cycled system has been developed by Bree [18]. When Youngdahl and Smith extend their analysis to cover ratcheting, the allowable design point drops from to MW-yr/m^2 where the fatigue curve plotted in Fig. 10 is taken from Fig. 9. Again the results of including a temperature limitation reduces the attractiveness of using a low structural temperature design.

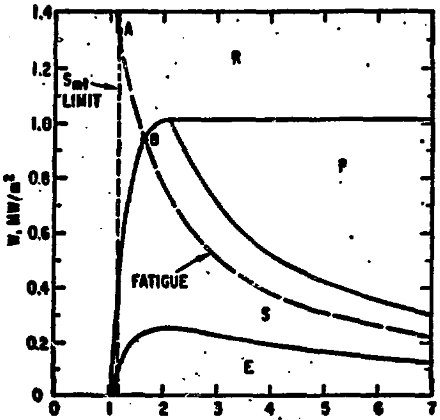

Fig. 10.

This promising approach to design stops short of including the propagation of existent flaws in the analysis. In one study [19] flaw propagation was found to be the life limiting property. This analysis is dependent on both the assumptions made for initial flaw sizes and the applicability of the available data which is for unirradiated material tested in air.

Within these constraints the crack growth properties necessary to achieve a given lifetime can be calculated. For Type 316 stainless steel the assumption of the presence of an 0.124 cm flaw depth, a flaw aspect ratio of 0.5, in a wall of 0.25 cm thickness resulted in the information presented as Fig. 11 [19]. For these conditions, a significant improvement in the crack propagation properties are necessary just to achieve a 10 MW-yr/m^2 life at a wall loading of 2 MW/m^2. Based on this calculation, only low stress, low temperature combinations would allow the achievement of the minimal 10 MW-yr/m^2.

In a study of an Inconel 718 first wall module, Kearney, et al. [5] found that wall loadings were limited to less than ∿ 2.2 MW/m^2 from their imposition of a 600°C maximum temperature. For a neutron wall loading of 2 MW/m^2, it was found essential to detect all flaws larger than 0.75 mm in depth in

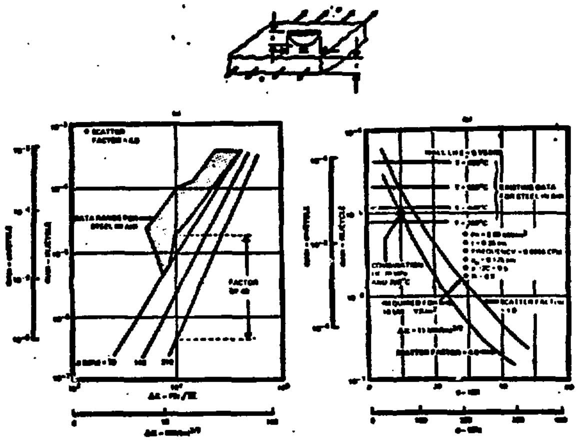

Fig. 11.

order to achieve acceptable lifetimes (> 10 MW-yr/m^2) for their 5 mm thick structure.

The question of what is the appropriate initial flaw size to choose is a difficult one to answer. Fusion devices will, in all probability, be field erected and consist of joining massive components. While it is true that techniques are currently in practice to detect scratches of 0.025 mm in thin walled tubing, such schemes may not be applicable to fusion first wall modules. The benefits to be gained in going to better nondestructive testing systems are shown in Fig. 12 for a wall of 0.25 cm. If a scratch depth of only 0.025 cm (10% of the wall thickness) is assumed, a substantial improvement is still required in crack propagation properties if economically interesting wall lifetimes are to be achieved.

From this discussion, it is evident that flaw propagation must be included in any lifetime analysis of a fusion device. This should be done after the structure has been found acceptable in an analysis such as outlined by Youngdahl and Smith.

With this analysis framework in mind, the next step is to extrapolate the available data to conditions presented by the fusion environment.

7. EXTRAPOLATION OF DATA

As described expertly elsewhere [20,21], the high energy neutron bombardment characteristic of a fusion device will lead to major microstructural changes. These effects are

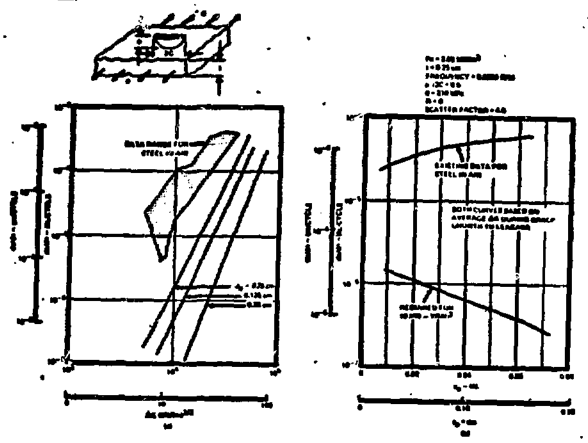

Fig. 12.

best documented for the austenitic stainless steels where swelling and ductility loss are perhaps the most important property changes with fluence. An important [22] paper has recently been presented that deals with the difficult problem of how to incorporate the ductility loss into a design approach. In the referenced approach, each of the ASME 1592 Code Case requirements were addressed from the viewpoint of what additional conservatisms should be included when designing with a low (where the definition of low changes with the requirement) ductility material. A major assumption made in this work is that swelling and irradiation creep do not inherently damage the material.

For example, in setting the criterion for the primary membrane stress intensity for unirradiated materials, the level must be kept below the yield stress, S_y, while for irradiated material whose ductility has dropped below 5% (or as can be shown to be equivalent, whose yield strength to ultimate strength ratio has exceeded 0.6), the allowable level becomes 0.6 of the ultimate strength.

In analyzing the effect of ductility loss on the use of the Bree diagram to protect against ratcheting, the authors conclude that it must be amended when the uniform elongation is reduced to approximately 1% for an austenitic stainless steel. At this point the work of Leckie and Porter [23] is used to modify the boundaries in the Bree diagram. These authors have shown that ratcheting strains will be small when creep is significant if loads are kept less than n/n+1 of the elastic-plastic shakedown loads where n is the exponent in the creep relation $\dot{\varepsilon}_c = A\sigma^n$. If the thermal creep expression is used, then n's of 4-6 are appropriate and relatively small movements of the Bree diagram boundaries result. If the expression for radiation enhanced creep is used, then an n value near 1 should be used and much larger movements of the Bree boundaries are required. The use of any thermal creep related deformation laws for the prediction of performance of a component during neutron radiation may at first seem questionable since, for in the temperature range of interest, the time dependent deformation mechanisms for the matrix of the grains of the structural alloy are not those that operate during thermal creep. In other words, creep during irradiation is not a superposition of a radiation enhanced process and a thermal one. However, the importance of including a limit on ratcheting has more to do with avoiding conditions where cracks are initiated within the material than to the absolute amount of deformation that occurs. Therefore, what one would really like to know is the amount of grain boundary sliding that occurs for some given amount of total strain. If one assumes that the grain boundary sliding process is little affected by radiation, then the calculation of a nonexistent amount of thermal creep strain may serve as an adequate estimate of the level of the grain boundary sliding (some small fraction of the actual calculated creep strain) that can be incurred without crack initiation.

For this reason, the suggestion by Nelson, et al., to use a total strain limit that is based on a summation of the ratio of the plastic strain, $\delta\varepsilon^p$, to the plastic strain limit, ε_L^p, and the ratio of the thermal creep strain $\delta\varepsilon^c$, to the creep strain limit ε_L^c where the limits, ε_L and ε_L^c are in turn allowed to vary with fluence and temperature appears valid. That is

$$\sum \frac{\delta\varepsilon^p}{\varepsilon_L^p} + \sum \frac{\delta\varepsilon^c}{\varepsilon_L^c} < \beta \ \text{(where } \beta < 1\text{)} .$$

This approach replaces the fixed 1% limit on strain imposed by the ASME Code Case in a manner that is more in keeping with the intent of minimizing the chances of crack formation in a material whose ductility changes as a function of time. In a similar way, the authors recommend a linear summation of the fraction of rupture life and fatigue life where each of these is again a function of fluence as well as temperature and stress.

The authors recommend a linear summation of the rupture life fraction and the fatigue life fraction where these lifetimes are allowed to vary as a function of fluence, temperature and stress. While the linear summation aspect [24] of the approach is open to debate, it does serve to emphasize two properties of special importance; the fatigue and stress-to-rupture lifetimes expected under tokamak reactor conditions.

Some experimental information has been obtained on the effect of neutron irradiation in the void forming temperature range on fatigue lifetimes. Michel [25] has suggested an adaptation of the Universal Slopes Equation to effectively use the increases in yield strength and reduction in area parameters to adjust for the effects of radiation on the fatigue properties. This approach results in a prediction of an improvement in properties under high cycle conditions with a reduction at low cycles to failure. The Universal Slopes Approach has the inherent problem of choosing the proper ductility parameter to use. For example, if the reduction in area of an irradiated sample is measured at low magnifications, very low values are found while a higher magnificiation examination of the same sample will reveal relatively high local reductions due to its tendency to deform a dislocation channeling. It is not evident which of the values of reduction in area is appropriate to use.

Furthermore, this difficulty relates to the larger issue that it may be unreasonable to expect an empirically derived relationship to be valid when the basic deformation mechanisms change. As discussed lucidly by Grossbeck, [20] the clustering of point defects into voids and dislocation loops leads to a microstructure in which dislocation channeling occurs. Thus the nature of the plastic zone advancing in front of a fatigue crack would be expected to change dramatically as shown in Fig. 13. Additionally, the shearing of voids might be expected to cause branching cracks to form in front of the main fatigue crack. Other mechanisms proposed for fatigue crack propagation would also be expected to change under irradiation. For example, as reviewed by Weertman, [26] Yokubari has suggested a mechanism dependent on the formation of voids ahead of the fatigue crack tip. These would be expected to form at much different rates in the

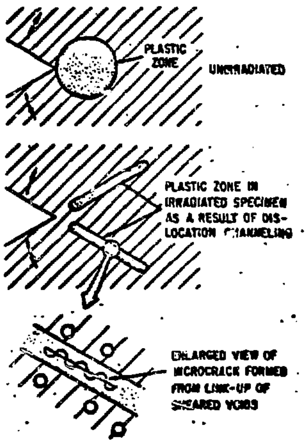

Fig. 13.

presence of a highly mobile self-interstitial population whose migration will be a sensitive function of the stress state present around the crack tip.

Besides affecting the propagation rate of a fatigue crack, irradiation would also be expected to reduce the fracture toughness properties. Several studies have evaluated the growth of existing flaws to a coolant leak condition. Failure by coolant leakage has generally been considered to be more critical than fracture. This is due primarily to the relatively thin gauges of the first wall concepts evaluated so that the critical crack size for fracture is larger than the material thickness. These analyses have generally been based on linear elastic fracture mechanics. This requires appropriate fracture toughness data. Because first wall gauges are relatively thin, stable flaw growth with significant plastic deformation at the crack tip might be expected to occur prior to the onset of crack instability. However, the embrittlement which occurs, due to the irradiation environments, is expected to make brittle fracture, around small cracks more probable. Therefore, the linear elastic fracture mechanics analysis and associated plane strain fracture toughness properties are considered relevant to the first wall structures.

Fracture toughness is expected to be severely degraded by irradiation. Hawthorne and Watson [27] have studied fracture toughness trends in AISI 300 series stainless steels. They observed large decreases in postirradiation fracture toughness in 308 stainless steel welds, based on fatigue precracked Charpy-V specimens and J-Integral assessment procedures. For EBR-II radiation exposures at temperatures above 370°C, fracture toughness values were reduced to as much as 20% of unirradiated values. Reductions of this magnitude will have an extremely detrimental effect on fusion reactor first wall design and life.

Since at present it is impossible to predict on theoretical grounds how these different effects will combine, it is imperative that in-pile fatigue data be obtained at once on candidate structural materials. Until that time, the designer has little option but to use a Universal Slopes type approach to the prediction of fatigue life.

The stress-to-rupture lifetime of a metal exposed to a fusion neutron spectrum may well be determined by the stress assisted growth of helium filled gas bubbles at the grain boundaries. The high rate of helium generation caused by the energetic fusion spectra will make this mechanism much more important than in fission applications. Experimental verification of the potential importance of this mechanism has been demonstrated by Grossbeck, et al. [20] who was able to show good correlation between a calculated lifetime based on a Hull-Rimmer [28] type calculation of bubble growth ratio and the experimentally measured rupture time.

An analysis of this mechanism shows that it is linear in stress and reduced by a decrease in grain size as well as an increase in the fraction of helium that can be trapped in the matrices of grains. Much more attention needs to be given to this gas bubble growth mechanism which heretofore has been mainly an academic interest.

8. CONCLUSIONS

The major points emerging from this review include:

1. Long lived first wall structures are imperative for the achievement of economic fusion power.
2. The stress state in a tokamak first wall component is a complicated function of the module geometry, the choice of structural material and the irradiation conditions.
3. Stresses can be sharply reduced by the use of selected materials, such as the ferritic steels, and the incorporation of design features such as divertors and liners.
4. A framework of lifetime analysis has been developed based on the ASME Code Case 1592 due primarily to the work of Youngdahl and Smith [17] and to Nelson and others [22].
5. When flaw propagation is included, assuming the presence of undetected flaws, much shorter lives than predicted by the ASME based approach result.
6. Fatigue and stress-to-rupture properties appear the most critical to the achievement of long lives by tokamak first wall modules.
7. Radiation effects on crack propagation and fracture toughness need to be much better understood through the gathering of in-reactor experimental information.
8. Stress-assisted bubble growth may determine the stress-to-rupture life in a fusion environment. In reactor studies are needed under high helium generation conditions to verify the importance of this mechanism.

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