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LIFETIME ANALYSIS OF FUSION-REACTOR COMPONENTS*

by

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ABSTRACT

A one-dimensional computer code has been developed to examine the lifetime of first-wall and impurity-control components. The code incorporates the operating and design parameters, the material characteristics, and the appropriate failure criteria for the individual components. The major emphasis of the modelling effort has been to calculate the temperature-stress-strain-radiation effects history of a component so that the synergistic effects between sputtering erosion, swelling, creep, fatigue, and crack growth can be examined. The general forms of the property equations are the same for all materials in order to provide the greatest flexibility for materials selection in the code. The code is capable of determining the behavior of a plate, composed of either a single or dual material structure, that is either totally constrained or constrained from bending but not from expansion. The code has been utilized to analyze the first walls for FED/INTOR and DEMO.

I. INTRODUCTION

Lifetime analyses for fusion reactor systems have been performed over the years as part of the design studies for the different reactor systems that have been proposed. Most of these analyses have emphasized particular aspects of the component lifetimes. For example, early studies emphasized radiation effects, particularly swelling and ductility loss.¹ The design limits on swelling range from ~ 2-10%, and the limits on uniform elongation range from ~ 0.5 to 2%. The radiation limits were used since they depended only on the operating temperatures and neutron fluence, and they did not require detailed analysis of the material stresses or the operating scenario. Other studies have emphasized the mechanical property limitations, such as fatigue, thermal creep, and creep-fatigue interactions.^{2,3} In work by Mattas and Smith, both radiation effects and mechanical property effects were considered.⁴

A study by Cramer, et al., also examined both radiation effects and mechanical property changes.⁵ In addition, component design, and thermal and structural analysis were an integral part of the study.

More recent studies have examined the synergistic effects that occur during operation.^{6,7} In these studies the interaction of properties such as swelling, radiation creep, crack growth, and fatigue have been analyzed in considerable detail. The material most often studied is austenitic stainless steel because it is the most likely structural material in near term devices and it has the largest data base from which to draw. The most complete work in this area was done by Watson,⁷ who examined 20% cold-worked Type 316 SS. In addition to the properties mentioned above, the effect of surface erosion and the effect of radiation on crack growth behavior were also incorporated into the model.

The present work was undertaken to study the behavior of first wall and limiter/divertor components. The approach used is to calculate the synergistic effects of the various materials properties in a manner similar to previous studies. In this way, the time evolution of the component stresses and strains along with changes in the materials properties could be studied. Two major modifications have been made to this type of materials modeling. First, the model is designed to analyze the behavior of several different materials rather than just a single material. The material property equations have been formulated to be flexible enough so that only the coefficients need to be changed for different materials. Second, the model was designed to examine both single and duplex structures. Many first wall and limiter/divertor designs use a duplex structure where a non-structural material, exposed to the plasma, is attached to a heat sink. The response of such a structure to the fusion environment is expected to be considerably different from a single material. These two

modifications allow for greater flexibility to the user in analyzing various fusion components.

II. LIFETIME MODEL

The major emphasis of the modeling effort was to examine component response for a wide variety of operating conditions and materials. For this purpose, a one-dimensional model of a plate is believed adequate, although the model can, in principle, be adopted to 2 or 3 dimensional calculations. Only normal operating conditions (burn cycle and down periods) are included in the model.

The flow diagram for the lifetime code is shown in Figure 1. The code first calculates the temperature, stress, and strain distribution through a plate based upon the selected operating and design parameters. Property changes are then determined for a specific time increment, and the effects of those changes on the initial distributions are evaluated. This process is repeated until failure occurs or the desired lifetime is reached.

The basic inputs to the code are the operating and design parameters. The design parameters include component material(s) selection, plate thickness, coolant characteristics, primary stress level, initial crack length, and plate constraint. The operating parameters include the burn cycle characteristics (ramp time, burn time, dwell time), down period duration and frequency, surface heat flux, neutron wall loading, and surface erosion rate. Finally, the failure criteria for swelling, deformation, and ductility are also provided.

Either a single or dual material plate can be analyzed by the code. The dual material plate is representative of impurity control components which are likely to have a low-Z material bonded to a structural material. The materials properties which are considered are the thermophysical, mechanical, swelling, and neutronic properties. Temperature is the primary parameter considered for the thermophysical properties. Radiation effects are also considered in the cases of the thermal conductivity and the elastic modulus. The mechanical properties considered are the tensile, crack growth, fatigue, and creep properties. A bi-linear elastic-plastic behavior is assumed for the tensile properties. Temperature, stress, fluence, flux, stress and strain ranges, and stress intensity are the parameters included in determining the mechanical behavior. Radiation swelling is dependent on the temperature and fluence, and the neutronic properties depend on the neutron flux and fluence.

The material properties are represented by empirical equations that incorporate the temperature, fluence, and stress dependencies. Whenever possible the predicted property values are based upon available experimental data. In many cases, the experimental data are sparse, and therefore data from similar materials or best estimates of property values are employed. The lack of data for many materials means that there is considerable uncertainty built into the model, and the predictions from the model should be viewed as showing qualitative trends rather than as an accurate representation of component behavior. Details of the material property equations and the property data base are given in Ref. 8.

A simple thermal-hydraulics subroutine calculates the temperature distribution through the plate at various times during the

Fig. 1. Lifetime code flow diagram

burn cycle. The temperature distribution is then used to calculate the thermal strain distribution, which is used as input to calculate the stress distribution. The stresses can be determined for a plate which is either totally constrained from expansion, allowed to expand but not bend, or is unconstrained. These three conditions span the possible range of component constraint in the reactor.

Once the temperature, stress, and strain distributions are determined, the long term response of the material to the reactor environment can be evaluated. The code determines the swelling change, creep change, fatigue damage, and crack growth for a time increment Δt . The swelling and creep which occurs during the period Δt results in a modified strain distribution that is used to calculate the changes in the stresses. The code also determines the amount of surface erosion during the time increment. The reduction in plate thickness results in a modified temperature distribution. The property change calculations are then repeated using the modified distributions until either the failure criteria are met or the component reaches the goal lifetime. For calculational purposes, the temperature and stresses are assumed to remain constant during the time increment. Therefore, Δt must be chosen such that the stress change is small compared with the total stresses in order for this approach to reasonably approximate in reactor behavior.

III. RESULTS

The long term response of the FED/INTOR and DEMO first walls have been analyzed. The relevant operating parameters used for the calculations are shown in Table 1.^{9,10} The structural material for both designs is 20% cold worked Type 316 stainless steel. The FED/INTOR first wall is uncoated, and the DEMO first wall is coated with 2 mm of Be (70% TD). Useful heat is extracted only in DEMO as reflected by the difference in the coolant temperatures. Another major difference in the two designs is the high number of burn cycles in FED/INTOR compared with DEMO which operates in the steady state mode.

A. FED/INTOR First Wall

The temperature distributions through the first wall at several times during the reactor lifetime are shown in Figure 2. The surface of the plate exposed to the plasma for this and all other relevant figures in this section

Table 1. DESIGN AND OPERATING CONDITIONS FOR FED/INTOR AND DEMO FIRST WALLS

Parameter	FED/INTOR	DEMO
Material	20% CW Type 316 SS	Be (70% TD) on 20% CW Type 316 SS
Thickness	12 mm	2 mm (Be) 4 mm (SS)
Coolant Temperature	90°C	280°C
Surface Heat Flux	.11 MW/m ²	.25 MW/m ²
Neutron Wall Load	1.3 MW/m ²	2.1 MW/m ²
Availability	50%	50%
Lifetime Goal	10 y	10 y
Lifetime Cycles	7×10^5	250
Lifetime Fluence	52 dpa	84 dpa
Burn Time (s)	200	6×10^5
Dwell Time (s)	40	3.6×10^3
Down Periods/Year	6	2
Stress Condition	Free to expand but not bend	Free to expand but not bend
Primary Stress Level (MPa)	0	50
Erosion Rate (m/s)	3×10^{-11} m/s	1×10^{-11} m/s

Fig. 2. INTOR first wall temperature distribution

is the zero point of the abscissa. The temperature at the plasma side surface is 500 K, and it drops to 373 K adjacent to the coolant. The temperature gradient deviates from linearity because of the temperature dependence of the thermophysical properties and the significant contribution from nuclear heating. As the plate thickness is reduced by surface erosion, the surface temperature decreases such that after 15 y of operation the peak temperature is only 435 K.

The thermal gradients through the plate are largely responsible for the generation of the operating stresses. The stress distributions for a plate that is allowed to expand but not bend are shown in Figure 3 for various times during the reactor life. Early in life ($t = 0$), the stress is compressive on the plasma side and tensile on the coolant side during the burn. The stress levels are near zero during the dwell period. The peak thermal stress of ~ 400 MPa is well within the $3 S_{mt}$ design guideline. Radiation creep results in stress relaxation during the burn period such that the stress levels are near zero after ~ 1.6 y of operation. The stress levels during the dwell cycle move towards the opposite of the initial stress distribution during the burn. As the plate erodes, the peak stress levels are reduced ($t = 15$ y). Void swelling is predicted to reach only .02% after 15 y and hence it has no impact on the first wall stress distribution.

Fig. 3. INTOR first wall stress distribution

A major concern of a high cycle device like FED/INTOR is the possibility of fatigue or crack growth failure. For the present analysis, the initial crack length is assumed to be zero, so that only fatigue damage accumulation was evaluated. The increase with time in the fatigue damage factor (a value of 1 means failure) for the plasma side of the plate is shown in Figure 4. It should be noted that a safety factor of two in strain or a factor of 20 in fatigue lifetime was used in the calculations. The results indicate that the FED/INTOR first wall is not expected to crack during its lifetime. On the other hand, if stress relaxation and erosion were not included in the analysis, a failure would be predicted in ~ 10 y.

Fig. 4. INTOR first wall fatigue damage accumulation

Loss of ductility is another major concern for the first wall structural material. The change in the uniform elongation in the FED/INTOR first wall is shown in Figure 5. The uniform elongation is predicted to reach a level of only 1% after ~ 10 y. The DEMO first wall (see below) will reach the same level after ~ 4 y of operation. The loss of ductility to these levels must be considered as being serious, but there are no established guidelines for acceptable ductility values.

Fig. 5. Loss of ductility in 20% CW Type 316 SS

B. DEMO First Wall

The DEMO first wall employs a duplex structure with 2 mm of Be attached to 4 mm of 20% cold-worked Type 316 stainless steel. Both 70% theoretical density and 100% theoretical density Be have been analyzed. The temperature distributions through the first wall using 100% TD Be at several times during the reactor lifetime are shown in Figure 6. The peak temperature is 630 K. The thermal gradient is low in the Be due to its large thermal conductivity compared with Type 316 stainless steel. Erosion during the reactor life leads to a reduced surface temperature in a similar fashion to the FED/INTOR first wall. The temperature distributions for the first wall with 70% TD Be is similar to the distribution shown in Figure 6, but the peak surface temperature is calculated to be ~ 5 K higher with the low density material.

The stress distribution in the first wall using 70% TD Be is presented in Figure 7. The most obvious difference between the DEMO and FED/INTOR first walls is the large stress discontinuity that occurs at the Be-stainless steel interface. The stress in both the Be and stainless steel are below the ASME guidelines for thermal stresses and they are in fact considerably lower than the stresses

Fig. 6. Temperature distribution in the DEMO first wall

Fig. 7. DEMO first wall stress distribution

in the FED/INTOR first wall. (Note that the stress free temperature, i.e., the temperature where the thermal mismatch between the Be and stainless steel is zero, is assumed to be room temperature.) The stresses relax towards the primary stress after ~ 0.7 y of operation. The average stress level increases with time since surface erosion reduces the thickness of material that can carry the forces from the pressurized water.

The loss of ductility in the 316 stainless steel is shown in Figure 5. The reduction in uniform elongation occurs approximately a factor of two more rapidly than in the FED/INTOR first wall. Further work is required to evaluate the impact of this ductility loss.

IV. SUMMARY AND CONCLUSIONS

The differences in the operating parameters and designs are predicted to result in considerably different behavior of the FED/INTOR and DEMO first walls. The thinner wall and the use of beryllium in DEMO results in reduced stresses in the first wall structure. The low stresses and reduced number of cycles mean that fatigue is not considered to be a major concern in DEMO. On the other hand, the FED/INTOR design is very close to the allowable fatigue limits such that increases in either the thickness, heat flux, or number of cycles cannot be tolerated. Radiation creep is expected to cause stress relaxation during operation. The thermal stresses are predicted to relax out of the FED/INTOR first wall in ~ 1.6 y, and they are predicted to relax out of the DEMO first wall in ~ 0.7 y. During stress relaxation, stresses of equal magnitude to the thermal stresses but opposite in sign occur during the reactor dwell periods. In both cases, radiation swelling is predicted to be insignificant (.02% in FED/INTOR vs. 0.27% in DEMO). This is due to the fact that both designs operate at temperatures which are well below the swelling peak for 20% cold worked, Type 316 stainless steel. The porosity in the 70% theoretical density beryllium is assumed to accommodate the helium generated during irradiation so that the predicted swelling in beryllium is zero. The uniform elongation of the stainless steel is predicted to drop below 1% after ~ 10 y and ~ 4 y of operation for FED/INTOR and DEMO, respectively. The major conclusions of this particular study are that radiation swelling should not be a concern for near term devices, that fatigue and crack growth is the primary concern for a high cycle

device such as FED/INTOR, that loss of ductility could be a concern towards the end of life for both devices, and that the DEMO first wall appears to have a large operating safety margin in terms of permissible heat loads and operating cycles.

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