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SPECTRAL TAILORING FOR FUSION RADIATION-DAMAGE STUDIES: WHERE DO WE STAND?*

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ABSTRACT

The Oak Ridge Research Reactor is currently being used to irradiate an experiment (MFE-4A) to simulate in stainless steel-316 the He to DPA ratio of 14 appm He/DPA expected at the first wall of a fusion reactor. The experimental design criterion is to remain within $\pm 20\%$ of this ratio after an initial irradiation period of ~ 1.2 years. At present (March 31, 1981) the experiment has been in the reactor for 19 cycles corresponding to 131,543 Mwh, 4.05×10^{25} n/m² in thermal fluence, 1.21×10^{26} n/m² in total fluence, 3.1 DPAs and 21.0 appm He. The current He to DPA ratio is 6.8. This paper summarizes some of the analysis calculations for this experiment and presents the current status of this work. The analysis calculations utilize the codes VENTURE, ANISN, GAS, UNFOLD, etc. and the obtained results include gas production, DPA, and heating rates, and neutron spectral modifications resulting from different reactor fuel loadings and core piece compositions.

1. INTRODUCTION

The radiation damage of the first wall of a D-T fired fusion reactor is one of the quantities limiting the feasibility of these machines. The damage parameters of interest in this paper are the displacement per atom (DPA) and helium production (appm) levels. For a stainless steel-316 first wall and a 14-MeV neutron wall loading of 1 MW/m², the anticipated annual levels for these parameters are 10 and 140, respectively, for a typical fusion device¹ yielding a constant He to DPA ratio of 14 appm He/DPA. Since there currently exist no experimental data that simultaneously represents these damage levels, the purpose of this work is to present a method that is being used to yield, in realistic time, the necessary damage information.

Fourteen-MeV neutron generators can yield the correct He-to-DPA ratio, but the time frame to obtain the high DPA and gas production levels precludes total reliance on data obtained in this fashion. Also, the amount of irradiated material tends to be quite limited. Fast reactors can yield in short times high DPA levels for large samples, but only low He production levels via direct fast neutron interactions. Thermal reactors can also yield high DPA levels in large samples, and if the alloy under consideration contains Ni, high gas levels are obtainable¹ via the thermal neutron reactions

Therefore, if some control over the thermal and fast neutron flux (which controls the He production and DPAs, respectively) can be accomplished, thermal reactors may be used to irradiate Ni-bearing alloys to obtain within a reasonable time frame the desired DPA and He production levels.²

The Oak Ridge Research Reactor (ORR) is currently being used to irradiate an experiment (MFE-4A) containing SS-316 (and PCA). The neutron flux around the experimental volume is being modified so that the DPA and He

production levels will closely resemble those anticipated in fusion reactors. The experimental design criterion is to remain within $\pm 20\%$ of the anticipated He to DPA ratio of 14 appm He/DPA for a fusion reactor after an initial irradiation period of ~ 1.2 years. The time lag to approach the proper ratio is required due to the two-step He production reaction.

Presented in the following sections are some of the calculational codes and calculated results that have been used to design and analyze the ongoing MFE-4A experiment. The results obtained so far include fast and thermal flux modifications and heating rate changes in the irradiated samples, and anticipated He production and DPA levels extrapolated in time.

2. CALCULATIONAL CODES

The ORR reactor has been modeled in x-y-z geometry using the VENTURE³ code system which solves the three-dimensional diffusion equation applied to neutron transport. The cross sections used in VENTURE are represented in a seven-energy group format (15-MeV to thermal) and were processed from ENDF/B-IV data. Calculated results obtained with VENTURE have been compared to ORR experimental results and reasonable (5-15% agreement has been obtained.¹ This code system is used to monitor each ORR core cycle, which varies from 5 to 14 days, and to determine the effect of the fuel loading and core piece composition on the neutron flux at the experimental location. Using the control rod position, and fueling and experimental makeup of the reactor supplied by the reactor operators, the code correctly predicts k_{eff} to be $1.0 \pm 1\%$.

UNFOLD is a program which takes the seven-group neutron flux data generated by VENTURE and converts it to 100-group neutron flux data. The neutron flux data base used by UNFOLD to produce the unfolded flux was obtained from a one-dimensional spherical ANISN⁶ calculation of the ORR reactor.

ANISN is a one-dimensional discrete ordinates code which solves the Boltzmann transport equation using the finite difference approximation. The neutron and gamma ray cross sections used in the ANISN code are taken from the DLC-37 library⁷ which was generated from ENDF/B-IV data. Some additional data sets (for example, Hf cross sections and kerma factors) have been added to this library as well as some of the thermal cross sections in this library have been modified to account for different temperatures. The ANISN code using cylindrical geometry and in conjunction with the unfolded neutron flux from VENTURE has been used to analyze in more detail the neutron and gamma flux levels and energy deposition rates in the areas immediately around the experiment.

SIGMA and GAS are analysis programs which convert neutron flux data to DPA and gas production rates.

3. RESULTS

Without neutron spectrum modification, the He production to DPA level ratio for SS-316 irradiated in a thermal fission reactor would quickly exceed 14, the value expected on the first wall of a fusion device. To determine the effect on the neutron spectrum of various materials placed around the experiment, several VENTURE and ANISN calculations have been carried out. Not only is it desirable to have control over the thermal flux, which will determine the gas production, but it is also advantageous to control the fast flux which will determine the DPA rates.

A typical ORR core which is representative of the ones used in the VENTURE calculations is shown in Fig. 1a and a typical one-dimensional cylindrical geometry mockup for ANISN of the 8 fuel elements which surround the experiment is given in Fig. 1b. The neutron source distributions which are distributed around the fuel rod have been obtained from VENTURE calculations.

The results of several calculations are shown in Figs. 2 and 3. Figure 2 gives the variation of the thermal flux and DPA rates when the core piece is composed of H_2O and Al and when the core piece is composed of H_2O , Al and Hf. During the initial phase of irradiation, the thermal flux needs to be increased so as to burn in Ni^{59} and the displacement rate needs to be reduced. This can be accomplished by using a core piece composed partly of Al and H_2O (see Fig. 2a). The current core piece which is now being used is composed of 37% Al and 63% H_2O . Since thermal neutron capture materials are to be used during the latter phases of the experiment, it is desired that the thermal flux be reduced while the displacement rate is either increased or at least held to a very moderate decrease. As can be seen in Fig. 2b the thermal neutron flux can be reduced substantially with only minor reductions in the displacement rate. To compensate for this reduction in the displacement rate additional fuel can be added around the experiment. The resulting effect of additional fuel can be seen in Fig. 2b.

Since strict control is maintained on the temperature of the samples, it is very important to have an indication of the energy deposition rates within the samples. This type of data are presented in Fig. 3. As the percent reduction in thermal flux increases due to the use of solid Al and Hf core pieces, the energy deposition rates in the samples are reduced due to a loss of fissions and resulting gamma rays in the fuel surrounding the experiment. The calculations indicate a reduction of ~13% in the gamma heating when the solid Al core piece replaces the 63% H_2O , 37% Al core piece. This calculated value agrees well with the experimental value of 14% and yields confidence in the data presented in Fig. 3.

Since VENTURE only gives neutron flux data, the gamma ray source strength for the one-dimensional ANISN calculations must be defined from other means. There are several methods which can be used. Method 1 calls for a reflected boundary conditions at the neutron source location shown in Fig. 1b to be placed upon the gamma rays. This simulates the gamma rays

which come in from surrounding areas. Method 2 differs from Method 1 in that a gamma ray source spectrum was obtained from the base case calculation and was employed in succeeding calculations thus preventing a reduction in the number of gamma rays incident on the outer edge of the fuel as the core piece composition was changed. In Method 3, the outer radius of the fuel required for criticality was determined in the base case calculation and a fuel concentration search was performed to maintain criticality in the succeeding calculations. This procedure allowed the loss of fissions in the fuel surrounding the experiment due to the thermal flux reduction to be compensated by increased fissions elsewhere. In any case, the results presents in Figs. 2b and 3 are remarkably insensitive to the methods employed.

Other potential neutron capture materials (W,Ta) have been analyzed. However, due to the ease of fabrication and lower cost, Hf is the currently acceptable material for the core pieces.

At present (March 31, 1981), the MFE-4A experiment has been in the ORR for 19 cycles corresponding to 131,543 Mwh, 4.05×10^{25} n/m² in thermal fluence, 1.21×10^{26} n/m² in total fluence, 3.1 DPA in SS-316, and 21.0 appm He in SS-316 (not including 2.0 appm from ¹⁰B). Using these data and previous calculations, real time projections have been made to estimate the dates the core pieces should be changed and the first samples removed. These data are reflected in Fig. 4. The insertion of the solid Al core piece which will decrease the thermal flux (slow the He production) and increase the fast flux (speed up the DPA rate) will not occur until the end of November, 1981. The first Hf core piece will be inserted at the end of March, 1982, and the first sample will be removed at the 10 DPA level sometime around June, 1982. The experimental He to DPA ratio will be within the design limits of $\pm 20\%$ sometime around September 1981. The He and DPA levels at that date will be 67 appm He and 6 DPA.

Ni wires that were inserted at the center of the experimental capsule have already been removed and will be sent for gas analysis so that a check can be obtained on the calculated He production levels. The currently calculated He production levels are based on HFIR data.

Following the removal of the first samples and re-encapsulation, the experiment will continue to higher DPA and gas production levels with the ultimate goal being >50 DPA.

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Figure Captions

Fig. 1.a A typical ORR core configuration used in the neutronic calculations. The MFE-4A experiment is located in L-3. b) Typical example of the one-dimensional model of MFE-4A experiment and surrounding ORR fuel elements.

Fig. 2. Variation of the DPA and thermal neutron flux in the E-3 position in ORR as a function of core piece composition and fuel loading around E-3.

Fig. 3. Variation in the energy deposition rate due to various thicknesses of Hafnium core pieces.

Fig. 4. Current and projected He to DPA levels expected in the MFE-4A experiment.

Fig. 2. Variation in the DPA and Thermal Neutron Flux in the E-3 Position in CER as a Function of Core Piece Composition and fuel loading around E-3.

SOLID Al Core piece
x (additional fuel)

Base Case (63% H₂O, 37% Al)

Relative Power

Fig. 3 Variation in the energy deposition rate due to various thicknesses of Hafnium core piece.

Fig 4 Current and projected He to DPA levels experiment in the JFE-4A experiment.