We have performed molecular dynamics computer simulation studies of 25 keV displacement cascades in Cu at low temperature. At this energy we observe the splitting of a cascade into subcascades and show that displacement cascades in metals may lead to the formation of vacancy and interstitial dislocation loop. We discuss a new mechanism of defect production based on the observation of interstitial prismatic dislocation loop punching from cascades at 10 K. We also show that below the subcascade threshold, atomic mixing in the cascade is recoil-energy dependent. We obtain a mixing efficiency that is proportional to the square root of the recoil energy.

A model is presented which describes the major facets of the relationships between irradiation creep. void swelling and applied stress. The increasing degree of anisotropy in distribution of dislocation Burger's vectors with stress levels plays a major role in this model. Although bcc metals are known to creep and swell at lower rates than fcc metals. it is predicted that the creep-swelling coupling coefficient is actually larger.

No summary.

Six binary Fe-Cr alloys with chromium levels ranging from 3 to 18% were tensile tested at room temperature following irradiation at 365, 403, and 574°C to doses of 7.4, 31.1, and 34.3 dpa, respectively. Irradiation at 365 and 403°C produced significant strengthening at all chromium levels, with concurrent reductions in ductility. Strength levels were insensitive to composition, irradiation temperature or neutron exposure. Irradiation at 574°C produced strength increases at only the two highest chromium levels. The results are attributed to dislocation loop and void development at 365 and 403°C, and phase instability at 574°C. NO evidence for channel fracture was found.

The 9Cr-1MoVNb and 12Cr-1MoVW steels were austenitired at 1040 and 1100°C to produce two different prior-austenite grain sizes. Pieces of the steels with these different heat treatments were then given different tempering treatments (1 h at 760 and 2.5 h 780°C). Charpy impact specimens with these different heat treatments were irradiated in FFTF at 365°C up to 5 dpa. For 9Cr-1MoVNb steel in the unirradiated mndition, the smaller the prior-austenite grain size, the lower the ductile-brittle transition temperature (DBTT). Also, the higher the tempering temperature, the lower the DBTT prior to irradiation. Regardless of the starting value, however. the DBTI shift during irradiation was the same for all specimens. This means that heat treatment can be used to insure a lower DBTT before and after irradiation. The 12Cr-1MoVW steel showed little effect of heat treatment on DBTI in the unirradiated condition, and the shift in DBTI was relatively constant. Thus, heat treatment cannot be used to reduce the effect of irradiation on the DBTT.

Miniature tensile and microscopy specimens of the tungsten-stabilized martensitic steel F82H were examined after irradiation in FFTF to fluences as high as 37 dpa at temperatures ranging from 365 to 750°C. Tensile tests performed at room temperature revealed a small amount of hardening at or below 405°C and significant softening at the higher irradiation temperatures. Electron microscopy results were Consistent with the tensile data. The tempered manensile structure was quite stable below 405°C. The manensile lath structure recovered slightly at 550°C accompanied by the coarsening of Carbide particles, while above 670°C, the manensile lath structure was eliminated. Cavity formation was observed only below 405°C. The maximum welling measured was as high as 0.14% at 405°C. Analysis of precipitate extraction replicas showed superior phase stability during irradiation. Precipitates identified in the irradiated specimens were $M_{23}C_6$ and Laves. the Same as were identified following thermal aging. Other precipitates were not observed.

New data on radiation-induced hardening, low-temperature creep and potential susceptibility (sensitization) to aqueous mrrosion have been obtained on various heats of austenitic stainless steel (including type 316) irradiated at 60 to 400°C to 7 to 13 dpa. The data were obtained from spectrally tailored reactor experiments, whose radiationdamage parameters are similar to those in the proposed International Thermonuclear Experimental Reactor **(ITER)** first-wall (FW) and blanket design. Austenitic stainless steels were found to increase significantly in strength at 60 to 330°C, to have higher irradiation-creep rates at 60°C than at 200 to 400°C, and to show radiation-induced changes in electrochemical properties at 200 to 400°C. These data on several radiation-induced property changes **suggest** that **type** 316 **steel** may **be** an adequate material for the FW of ITER. However. there is **a** definitely a need for new data on franure-toughness and on fatique behavior below 400°C, as well as more data on irradiation-creep and effects **of** irradiation on corrosion properties, **to** better define temperature and dose dependencies for more detailed design analyses. Cold-working should remain an optional as-fabricated mndition for the FW of ITER. Many properties of SA and CW 316 become Similar after irradiation at 60 to 400°C. The higher initial yield-strength of CW 316 will allow higher design stress and elastic strain limits.

Specimens of solution-annealed (SA) and 25% cold-worked (CW) PCA austenitic stainless steel were irradiated in ORR-MFE-4 at 330 and **400°C** to 13 dpa. and in ORR-MFE-6J/7J **at** 60, 200, 330 and 4 W C **to** 7.4 dpa. Swelling measured by the ORNL precision densitometer of CW PCA irradiated at 400°C was negligible after 7.4 and 13 dpa. whereas SA PCA showed 0.3 to 0.4% swelling after 13 dpa. At 330°C. both SA and CW PCA showed about 0.2% swelling after 13 dpa. Transmission electron microsmpy (TEM) analysis of the cavity microstructure showed many voids in SA PCA irradiated **at** 400°C, while only tiny helium bubbles were fouhd in the other specimens irradiated at 330 to 400°C. No cavities were detectable in CW PCA irradiated at 60 or 200°C. Fine MC precipitates were found in **SA** and CW PCA specimens at 330 and 400°C, hut not at lower temperatures. Both SA and CW PCA contained mixtures of larger Frank loops and dislocation networks at 330 and 400°C, whereas the dislocation structure of CW PCA was dominated by tiny "black-dot" loops at 60 and 200°C. The CW PCA showed significant radiation-induced recovery of the initial dislocation network structure at **all** temperatures; no large Frank loops were found at 60°C, while the **maximum** mncentration of larger loops was observed **at** 330°C.

A number of collaborative international experiments are being conducted to assess the feasibility of Fe-Cr-Mn austenitic steels for fusion service. A review is presented ***of*** the current **status** of these **various** efforts. The **first** time irradiation of an Fe-Cr-Mn steel in the form of pressurized lubes is highlighted.

Fe-Cr-Mn alloys have **been** proposed **as** a structural material for fusion reactors that **will** exhibit **reduced** long-term radioactivity. However, Fe-Cr-Mn ternary equilibrium has nor previously been well defined at temperatures below 650°C. The current experiment characterized phase evolution and equilibrium in three alloys; Fe-15Cr-15Mn, Fe-10Cr-30Mn and Fe-30Mn in the 20% cold-worked mndition aged from 300 to 700°C for 1,000 to **30,000** hours at temperatures from 300 to 600°C. Results indicate thato phase formation is extremely sluggish, and cold-working and long aging times are required to initiate σ phase formation in alloys not previously predicted to exhibit such precipitation. σ phase appears to form on high diffusivity paths such as recrystallizing grain fronts and grain boundary triple paints. Three isothermal Sections **at** 300, 500 and 600°C were constructed from the experimental data that extend the temperature range of the Fe-Cr-Mn ternary diagram to lower temperatures than previously published.

Microstructural evolutions in V, V-Cr, V-Ti, V-Ti-Cr, and V-Ti-Si alloys have been characterized **by** TEM after irradiation at 420°C in **FFTF/MOTA**. The results have been correlated with swelling behavior of the alloys to provide a better understanding of the superior **resistance** of some of the high-Ti heats to void swelling.

Niobium and Nb-base alloys are under consideration as candidate materials for the ITER divertor structure. Unalloyed Nb and binary alloys of Nb-2.5Ti, Nb-2.5Zr, Nb-2.5Hf, Nb-2.5V, Nb-2.5Ta, Nb-2.4Mo, Nb-2.5W, Nb-2.5Fe, Nb-2.5Ti, and ternary alloys of Nb-2.5Ti-2.5Ta, Nb-2.5Ti-2.5Mo, Nb-2.5Hf-2.5Mo (expressed in terms of atomic percent alloy addition) were exposed to high-puny water with < 30 ppb dissolved oxygen in a refreshed Stainless steel autoclave (with a 1300-psi overpressure) to determine the extent of oxidation as a function of time at 300°C. Corrosion data are presented for pure Nb and for ten binary and three ternary Nb-base alloys exposed for up to 2906 h. All alloys exhibited weight gains except Nb-2.5Ni, Nb-2.5Ta, and Nb-2.5Fe, which lost weight after 0, 1500, and 2200 h, respectively. Of the alloys that gained weight, the V and Mo additions were the most effective in minimizing weight gain. The ternary alloys incorporating Mo were superior in terms of bath resistance to weight gain and embrittlement.

BEATRIX-II is an irradiation experiment designed to study the in situ tritium release behavior from selected ceramic solid breeder materials. The second irradiation cycle of the experiment. Phase II, will include a temperature change capsule with a ring specimen of Li_2O and a temperature gradient capsule with Li_2ZrO_3 spheres. The temperalure change capsule is designed to achieve temperatures in the range from 485 to 650°C while the temperature gradient capsule will include a temperature range from 450 to 1200°C. The effect of specimen temperature, sweep gas composition, irradiation damage, and sweep gas flow rate on the tritium recovery behavior will be determined in a fast neutron flux to burnups of *8%.*

BEATRIX-II is an in situ tritium recovery experiment to determine the tritium release characteristics of Li_2O in a fast neutron flux A large diameter specimen of Li_2O has been irradiated under a steep temperature gradient to 4% burnup in a variety of sweep gas compositions and now rates. Decreasing tne amount of hydrogen in the sweep gas significantly decreases the tritium recovery rate. During irradiation the temperature of the specimen has remained stable and the results suggest that Li_2O is a viable fusion solid breeder material

Tritium release from single crystal lithium aluminate has been investigated by a series of isothermal anneal and constant rate heating experiments. The results of these experiments indicate that after anneals at low temperature, a large fraction of the tritium present before the anneal remains in the sample. We have modeled this behavior based on first order release from three types of sites. At low temperature the release is dominated by one site. while the tritium in the other sites is retained in the solid. Dopants appear to alter the distribution of tritium between the sites. Adding MgO decreases the fraction of tritium released at 777°C, while inneasing the fractions released at *538* and 950°C.

The energetics and kinetics of the evolution of H_2O and H_2 from $LiAlO_2$ have been studied by the temperature programmed desorption technique. After treating the sample with helium containing 990, 495, or 247 vppm H_2 at *923* K, H_2O and H_2 evolution was observed during 473 to 1073 K (200 to 800°C) ramps at rates of 5.6 K/min. The sweep gas was either pure helium or the same He-H, mixture that was used in the preliminary treatment. The H_2O and H_2 desorption peaks were shown to be the sums of first-order subpeaks which had reproducible activation energies and pre-exponential terms. For H_2O desorption. these peaks, of types labeled B. C, and D had activation energies of 22, 28, and *32* kcal/mol. (Earlier work identified type A peaks with an activation energy of 18 kcal/mol.) Enhancement of desorption of H_2O in rate and quantity by H_2 in the sweep gas was confirmed. It appears that the effectiveness of H_2 in this enhancement is not in modifying the activation energy and pre-exponential term for a given kind of site. but rather in changing the populations of sites participating in the desorption processes so that sites with lower activation energies are increasingly involved. Only C and D subpeaks were involved in H_2 desorption and they had activation energies close to and perhaps only 1 kcal/mol higher than the analogous peaks for H_2 evolution.

Polycrystalline spinel ($MgAl_2O_4$) and alumina (Al_2O_3) were irradiated under three, different sets of ion irradiation conditions in order Io examine the effecu of displacement damage and transmutation product formation on their microstructural evolution. Specimens were implanted with H, He and C alone, or irradiated with 4 MeV Ar^+ ions alone, or preimplanted with H, He, C and subsequently irradiated with 4 MeV Ar^+ ions. Irradiated specimens were observed with transmission electron microscopy following irradiation to damage levels of 0.6 to 30 dpa at 260 to 630°C for spinel and 0.6 to 18 dpa at 260 to 810°C for alumina. In both spinel and alumina. no radiation effecu were observable in specimens implanted with H, He and C alone. and the preinjected transmutation products did not have a significant effect on dislocation loop size or density in Ar^+ irradiated specimens. In alumina, cavities were formed in all of the observed Ar^+ irradiated specimens for both the preimplanted H, He and C case and the Ar^+ alone case. In spinel, cavity formation was only observed in preimplanted specimens irradiated with Ar^+ ions at relatively high temperatures and/or doses.

Irradiation of ceramics such as Al_2O_3 and $MgAl_2O_4$ with energetic light ions causes a suppression in the nucleation of dislocation loop. The results of recent electron microscope studies On ion irradiated oxide ceramics are summarized. It is proposed that the anomalous suppression of dislocation loop formation may be associated with the high proponion of energy lost to electronic ionization events compared to displacement damage events during light ion irradiation.

Magnesium aluminate spinel ($MgAl_2O_4$) with an ^{17}O enrichment ($^{17}O/O_{Tot}$) of about 23 at. % was prepared by reacting fine mixtures of aluminum hydroxide (enriched with ^{17}O) and MgO of normal isotopic content. The material was prepared for experiments in which the radiation damage produced in a fusion reactor is simulated by fission reactor exposures. The powder mixtures were obtained by hydrolyzing, with water containing the ^{17}O isotope, a mixture of aluminum isopropoxide and MgO powder. The mixture was convened into pure spinel by a series of heat treatmenu and grindings. Essentially fully dense bodies, which contained about 45% of the ^{17}O isotope initially present in the water, were successfully fabricated provided that all thermal treatmenu were conducted in argon or vacuum atmospheres.

Al_2O_3 and $MgAl_2O_4$ specimens containing ^{17}O isotope concentrations of 0 to 25 at. % ($^{17}O/O_{TOT}$) have been prepared for irradiation in HFIR in order to investigate the effects of helium formation on their microstructure and dielectric and mechanical properties. SiC/SiC flexure ban, ceramic fibers and a wide range of commercial ceramic TEM disks has also been included in the irradiation capsules. The specimens will be irradiated at temperatures of 100, 350, and 600°C in two irradiation capsules that will receive respective maximum neutron fluences of 2.4×10^{26} and 7.2×10^{26} n/m² (E > *0.1* MeV).

A wide range of advanced structural ceramics have been prepared and shipped to FFTF for irradiation to a nominal fluence $\sim 6 \times 10^{26}$ n/m² (E > 0.1 MeV) at temperatures of 420 to 800°C. Most of the ceramics will be irradiated as TEM disks, but SiC/SiC flexural bars, electrical conductivity disks, and bundles of commercial ceramic fibers will also be irradiated in separate irradiation capsules.

The application of a microindentation technique to measure the elastic modulus of carbon beam implanted silicon carbide is presented. Samples of Chemically Vapor Deposited (CVD) Silicon Carbide/Nicalon composites have been implanted to a damage level of 30 dpa at room temperature. The samples were then prepared in cross section and the microstructure was analyzed and modulus was measured along the damage path of the carbon ions. Both CVD silicon carbide and Nicalon silicon carbide was seen to amorphize at room temperature with the threshold for the CVD being approximately 15 dpa. Amorphization of Nicalon silicon carbide fiber was seen to occur over the entire range of the carbon path implying a much lower threshold for amorphization. Elastic moduli were seen to decrease significantly (from 440 GPa to 280 GPa) for the CVD material with the minimum modulus corresponding to the maximum damage region of the carbon beam. The modulus of Nicalon showed the opposite behavior. The modulus for the fiber was increased over most of the damaged region from an undamaged value of 170 GPa to a maximum of 210 GPa.

Recent data by E R. Hodgson[1,2] and by G. P. Pells[3,4] shows enhanced dielectric **loss** and electrical breakdown in ceramics during irradiation with protons and electrons and while subject to an externally applied DC electric field. We have investigated this radiation-induced conductivity effect at frequencies between 100 Hz and 10 MHz by irradiating sapphire with 3 MeV protons at 300 K Calculations of energy deposition and displacement damage in the sapphire were made using TRIM 90 on an IBM-AT. **Results** show an immediate increase in **loss** tangent at onset of irradiation that depends on both frequency and time.

Based on a preliminary **survey** of more than 15 oxides and nitrides, four ceramic materials (CaO, MgO, Y_2O_3, and BN) were identified **as** candidates for insulator coating development. These compounds were fabricated by **a** variety of techniques and *exposed* to flowing lithium **at 400°C to assess** chemical compatibility. Yttrium oxide exhibited excellent mrrosion resistance in flowing liquid Li at 400°C; its mrrosion rate was calculated to be 0.042 **m/yr.** Resistivity measurements were taken before and aher Li exposure to determine the effects of Li on the electrical properties of Y_2O_3. No deterioration in the resistivity muld **be** measured; in both **cases** the resistivity **was** $>10^{12}$ Ω·m. Boron nitride coatings on V-20Ti Substrates were fabricated by the newly developed ion-beam-assisted-deposition process. These coatings are expected to exhibit better compatibility with liquid Li than the previously RF-sputtered coatings. The resistivity of in-situ-formed $(V,Ti)_xN$ reanion-product layers on various V-base alloys, including **V-7Cr-5Ti,** V-2.5Ti-1Si, and V-20Ti, is being determined.