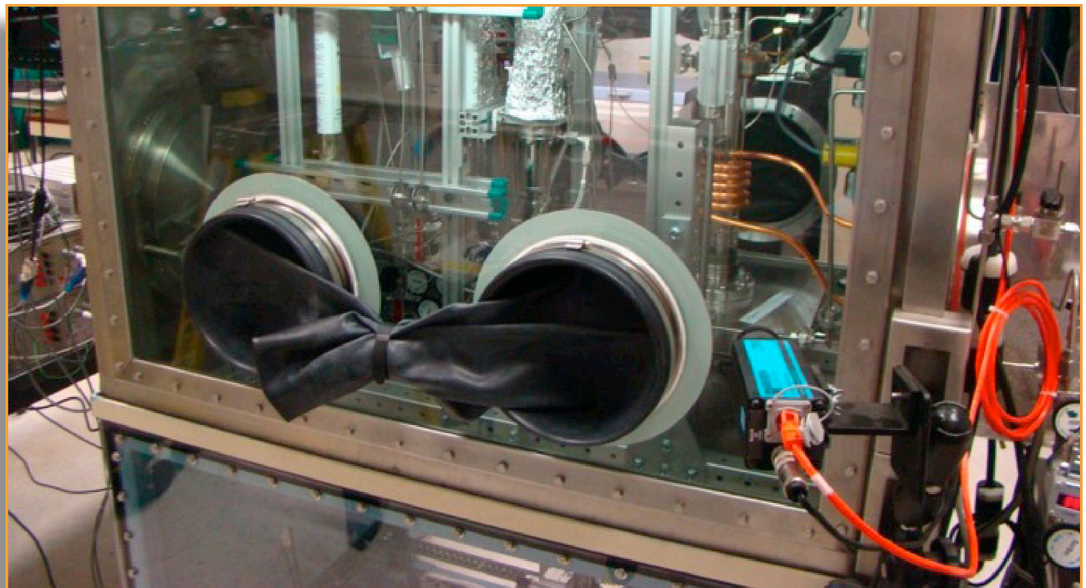

Tritium permeation test glovebox

The Energy of Industry

Fission Product Transport and Source Term

Next Generation Nuclear Plant

The Very High Temperature Reactor (VHTR) Program Next Generation Nuclear Plant (NGNP) Project is working to develop high temperature gas-cooled reactor (HTGR) technology that will meet the commercial needs of a wide range of industrial end users. Idaho National Laboratory manages the research and development of NGNP for the Department of Energy (DOE).

Fission product transport experiments and source term modeling activities are used to predict the fate of fission products from production in the fuel to the reactor boundary. Experi- mental data provide the technical basis for source terms under normal and accident conditions.

The smallest unit of the HTGR design is a uranium ceramic particle that is surrounded by graphite and ceramic coatings to contain the products of nuclear fission. Millions of these tristructural- isotropic (TRISO) particles approximately 1 mm in diameter are surrounded by graphite components to form a reactor core. A major activity associated with coated particle fuel is determination of the potential for release of radioactive material. The initial Advanced Graphite Reactor (AGR-1) experimental test operated at temperatures in excess of 1000°C for over two years with no indication of release of fission product gases.

The data from this experiment series will validate the mechanistic source term and the inputs for the NGNP safety analysis and licensing submittal.

From previous operations and irradiation testing, it is known that some fission products were not completely retained within the engineered barriers of the previous designs of fuel particle coatings. The fission products of concern are cesium, silver, strontium, iodine, europium, and tellurium. Through the NGNP irradiation test protocol, the fission product retentiveness of newly-developed engineered coatings and graphitic materials will be established and qualified.

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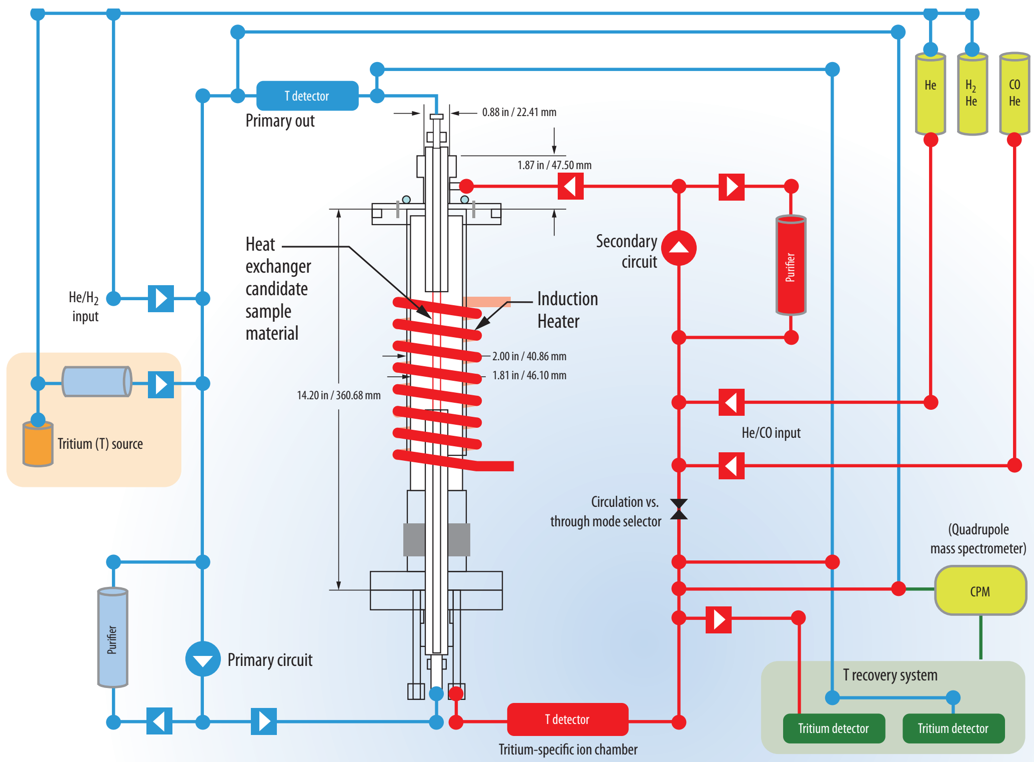
Tritium-specific ion chamber



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Another significant source term issue is the retention and transport of tritium in the structural components of the core and heat exchangers. Tritium in a HTGR comes from ternary fission, neutron activation of graphite lithium impurities, helium-3 in the coolant and neutron capture reactions in boron carbide control material. As much as 60% of the tritium production comes from fission. Tritium is largely contained by the TRISO coatings of the fuel particles and absorbed by the core structural graphite. Despite containment and absorption, tritium equilibrium in the primary coolant is reached during reactor operation. The tritium in the coolant gas is able to permeate through the primary heat exchanger at a given rate, depending on the material properties, tritium concentration and temperature. Permeation testing of high nickel superalloys such as Incoloy 600 was performed in the late 1970's, but only up to temperatures of 700°C. Permeation data for the elevated temperatures seen in a VHTR was needed. A permeation measurement system was built for this purpose. Photos and a schematic of that system are shown above. Permeation characteristics of hydrogen through high-nickel alloys Inconel 617, Incoloy 800H, and Haynes 230 alloys in the temperature range of 650 to 950ºC was completed in 2010. These data matched well with historical test values and were reported in Hydrogen Permeability of Incoloy 800H, Inconel 617, and Haynes 230 Alloys. Testing with tritium in this system in 2011 yielded results that were significantly lower than predicted by theoretical models. Validation of these data is being performed in 2012, and the results indicate that the release rates may be orders of magnitude lower than the model values, but have a higher sensitivity to increasing temperature. The rate of permeation is affected by the concentration (partial pressure) of the permeating element, the bulk material characteristics, and the chemical composition at the surface of the barrier. Corrosion resistant alloys typically form a metal oxide surface that prevents oxidation corrosion. This oxide layer also affects permeation. Maintaining a uniform oxide layer may be key to minimizing release. Testing of different coatings and oxide layer management are expected to contribute to the success of very high temperature reactors.