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IN-REACTOR PERFORMANCE OF LWR-TYPE
TRITIUM TARGET RODS

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IN-REACTOR PERFORMANCE OF LWR-TYPE TRITIUM TARGET RODS

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Pacific Northwest Laboratory (PNL)^(a) has conducted several one-year irradiation tests of light-water reactor (LWR)-type tritium target rods. These tests have been sponsored by DOE's Office of New Production Reactors. The first test, designated water capsule-1 (WC-1), was conducted in the Advanced Test Reactor (ATR) at DOE's Idaho National Engineering Laboratory from November 1989 to December 1990. The test vehicle contained a single four-foot target rod within a pressurized water capsule. The capsule maintained the rod at PWR-type water temperature and pressure conditions.

The target rod concept being tested was the "getter-barrier" design, consisting of high-density sintered annular lithium aluminate pellets (enriched in Li-6) stacked inside a special stainless steel cladding tube. The cladding was pretreated to obtain aluminized layers on the inside and outside surfaces, which acted as barriers to tritium permeation through the steel. Between the pellets and the cladding was a nickel-plated Zircaloy getter tube, the purpose of which was to fix the free tritium and to keep the internal partial pressure of tritium very low.

(a) PNL is operated by Battelle Memorial Institute for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

The WC-1 test was the premier irradiation of this target design in water at prototypic neutronic and thermal conditions. During its 13-month residency in the ATR, the WC-1 target rod accumulated 281 full-power days, of which 222 days (79%) were at goal temperature. The operating history is shown in Figure 1. The axial-average n, α power was 0.6 kW/ft, and the Li-6 burnup was approximately 20% (of initial Li-6 atoms). The four-foot rod produced approximately 0.34 gram (3300 Curies) of tritium.

The permeation leakage from the rod to the capsule water was assessed by thoroughly draining and flushing the capsule at the end of each ATR operating cycle (7 to 40 days) and measuring the tritium concentration and volume of the retained water. The cumulative measured permeation leakage at end-of-test was only 0.00139 Curie (compared to the 3300 Curies produced). This low release rate meets current criteria for tritium retention in a new production reactor (NPR) (see Figure 2).

This high tritium retention was obtained in spite of "conservatisms" in the test environment: notably, lack of oxygen control on the capsule water and through-cycle excursions in the capsule water pH. Capsule water would rise in pH from the normal pH of 5.5 of the feedwater to as high as pH 10.0 at end-of-cycle. The end-of-cycle water flushing procedure also subjected the rod to cycles of external pressure well in excess of those projected during normal operation in an NPR.

Post-test non-destructive examinations of the WC-1 rod involved visual examinations, dimensional checks, gamma scanning, and neutron radiography. The results indicate that the rod maintained the integrity of its pressure

seal and was otherwise unaltered both mechanically and dimensionally by its irradiation and post-test handling. However, a black, adherent corrosion layer was noted on the outside of the rod, and waterside cladding corrosion was undoubtedly enhanced (by the lack of oxygen and pH control on the capsule water) over that to be expected in an NPR.

Destructive examinations will include rod puncture and plenum gas analysis followed by transverse sectioning and tritium/helium analysis of component pieces from several short sections. Metallographic and ceramographic examinations will be conducted on companion sections. The preliminary puncture results are available and tend to confirm the calculated level of tritium production and the success of the getters in fixing the tritium as metal hydrides.

In subsequent target rod irradiations, steps were taken to control the water chemistry to which the rods were subjected, but the WC-1 irradiation demonstrates that even without these controls, the getter-barrier target rod can produce and retain tritium adequately in a PWR environment.

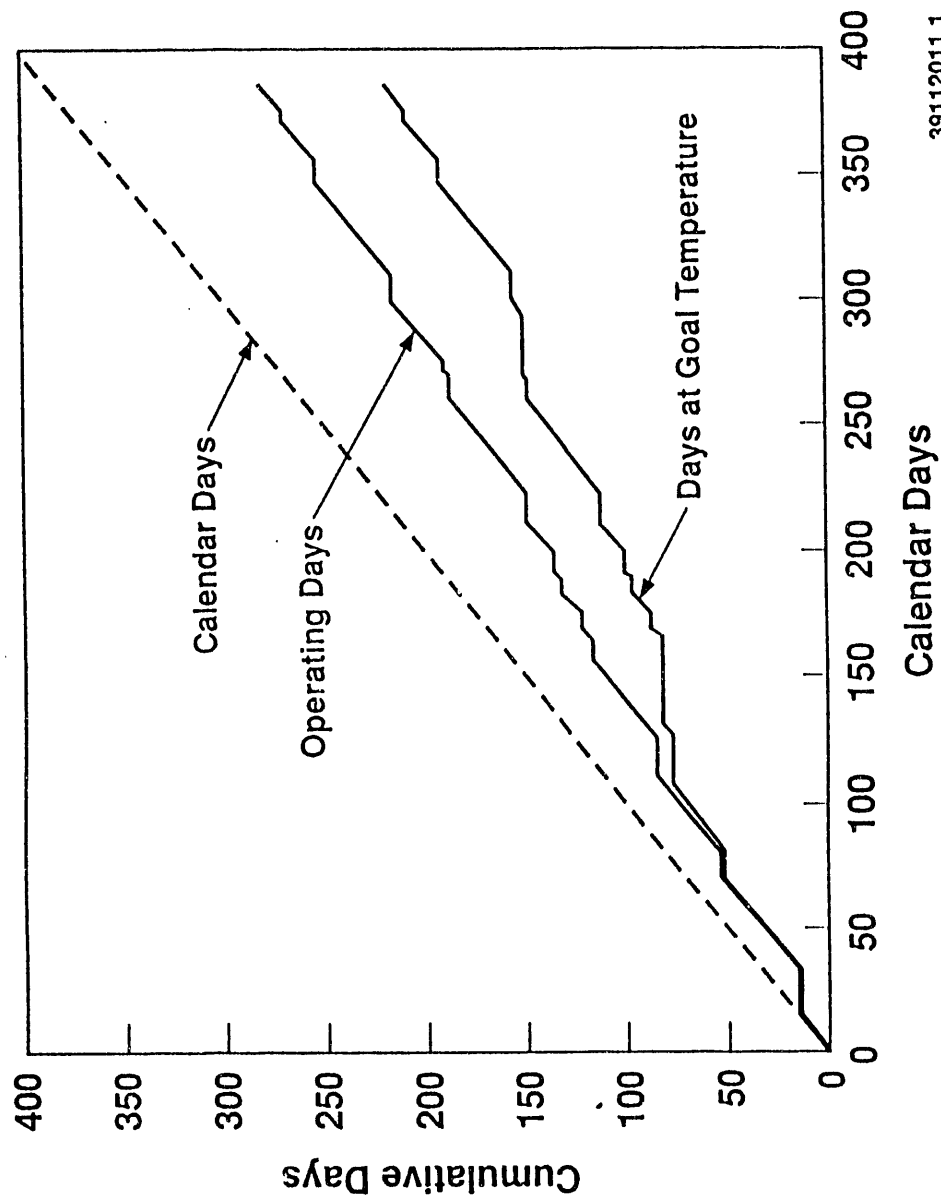
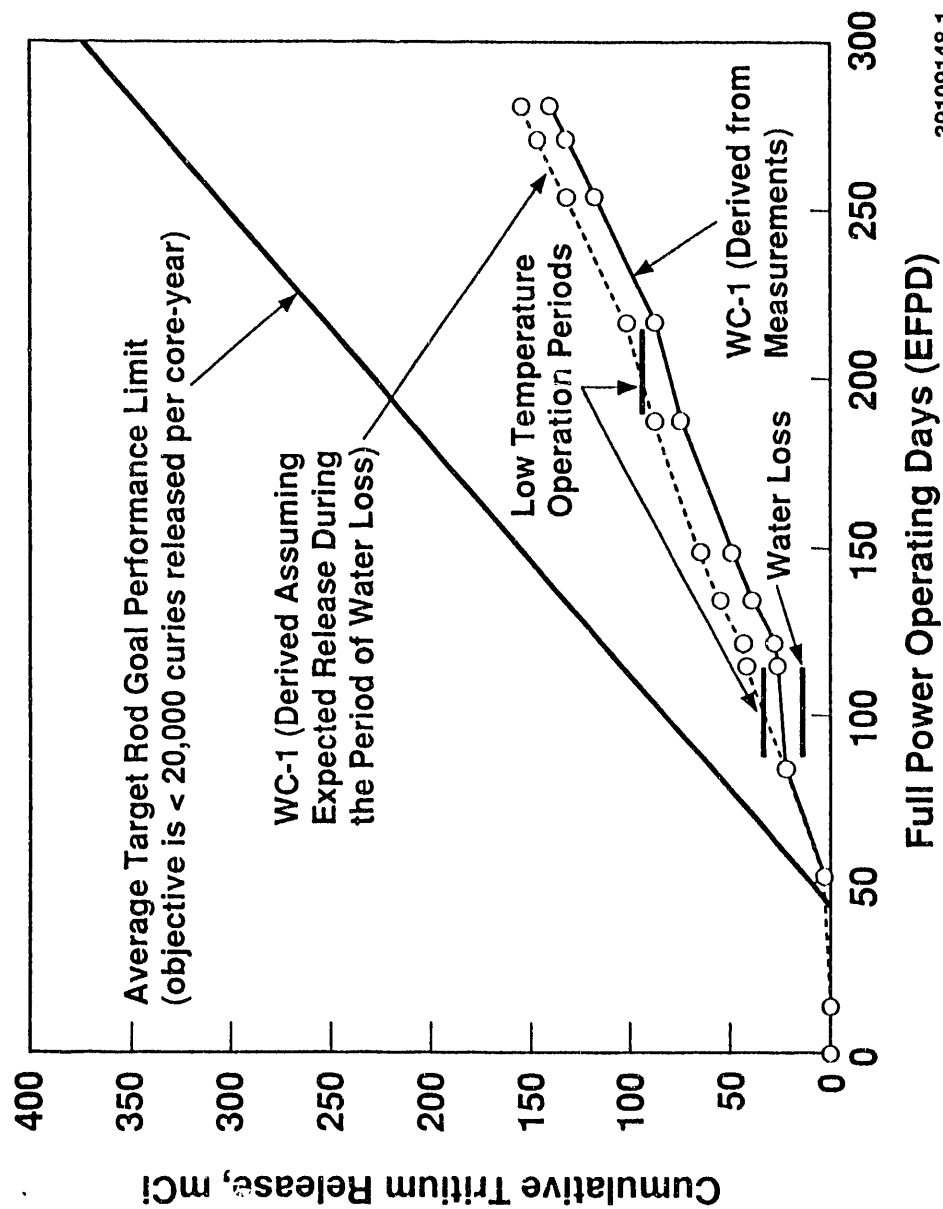


FIGURE 1. WC-1 Operating History

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FIGURE 2. WC-1 Tritium Release Versus Operating Days

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