

CONF-750958--2

BEHAVIOR OF IRRADIATED LWR FUEL PELLETS
DURING THERMAL TRANSIENTS

By

L. R. Kelman, J. Rest, M.G. Seitz & S. M. Gehl

For Presentation at
1975 Third Water Reactor Safety Research Information Meeting
Washington, D. C.
September 29 - October 2, 1975

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.



DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED
ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

operated under contract W-31-109-Eng-38 for the
U. S. ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

BEHAVIOR OF IRRADIATED LWR FUEL PELLETS
DURING THERMAL TRANSIENTS

L. R. Kelman
J. Rest, M. G. Seitz, and S. M. Gehl

ARGONNE NATIONAL LABORATORY
Argonne, Illinois 60439

Summary

Prediction of the behavior of LWR fuel rods and fission products under off-normal and accident conditions requires a physically realistic description of fuel swelling and fission-product release that currently does not exist. To satisfy this need, a program was initiated at ANL approximately a year ago with the prime objective of developing a comprehensive computer-base model that describes the release of fission products as a function of thermal transients anticipated in hypothetical accident situations. This model will be incorporated into ANC's FRAP accident-analysis code system. The analytical effort is supported by data developed from characterization of irradiated LWR fuel and from out-of-reactor transient heating tests of irradiated LWR fuel under conditions that simulate hypothetical LWR accidents.

Modeling of Fuel-fission-product Behavior

Significant progress has been made in the development of the Steady-state and Transient (SST) fuel-behavior code. A preliminary version of the transient portion of the code has been submitted to ANC for inclusion into FRAP-Transient (FRAP-T). A preliminary LWR steady-state code has been generated by modifying a multiregion version of the LIFE-III mixed-oxide fuel-element code; an analytical description of fuel behavior during an accident depends critically upon the initial fuel-rod conditions. The modifications include the conversion of the physical- and mechanical-property data from (U,Pu)O₂ fuel and 300-series stainless steel cladding to stoichiometric UO₂ fuel and Zircaloy cladding, the conversion of coolant properties and models from liquid sodium to water, the modeling of radial flux depression and the time variation of the axial fission-rate profile, and the inclusion of a physically realistic model (GRASS-Mod 1) that describes the behavior of fission gas in irradiated LWR fuels. Preliminary verification of SST has been completed. A comparison of the SST-calculated centerline-temperature results for four Halden fuel rods with data points obtained from graphs of thermocouple centerline-temperature measurements vs local rod power appears to show good agreement. The SST-LWR fuel-behavior code is being used at present to conduct parametric studies to determine critical conditions for the experimental program. Work has been initiated to develop a model for the behavior of volatile fission products. The model development will rely on basic input data from the experimental program.

Experimental Program

The experimental program consists of (a) characterization of LWR-irradiated fuel, (b) characterization of PBF- and Halden-tested fuel, and (c) direct-electrical heating (DEH) and fission-product collection tests. With a view to contributing useful information as early as possible by applying existing technology, the experimental program is concentrating initially on the behavior and release of fission gases with less emphasis on the volatile fission products. Also, early emphasis in the DEH test program is on the simulation of fuel radial-temperature profiles and heating rates anticipated for hypothetical power-coolant mismatch (PCM) accidents; simulation of other potential accident types will follow.

Fuel for the comprehensive test program is from a 15 by 15-in. rod bundle of H. B. Robinson Reactor No. 2 (a PWR) Zircaloy-4-clad UO₂ fuel rods acquired by NRC-RES for use by all contractors in WRSR's fuel-behavior program. The fuel is reported to have an average burnup of 28,027 MWD/MTM and an average linear power of 7 kW/ft. Two test fuel rods -- a

low-power, low-burnup PWR-irradiated rod and a high-power, high burnup BWR-irradiated rod -- are also being used to develop fuel-specimen preparation methods, to conduct preliminary DEH checkout tests, and to represent different fuel types in the test program.

The fission-gas and volatile fission-product release from LWR fuels during steady-state operation and transient conditions is strongly dependent on the distribution and morphological occurrence of the fission products with respect to available escape paths from the fuel. Extended networks of interconnected porosity and cracks provide the only effective escape paths during the short times involved in temperature excursions following reactor accidents. Thus, the fuel-characterization effort in this program is arrived at by quantitatively describing the amount and distribution of fission products, both gaseous and volatile, the extent of interconnected porosity in irradiated fuel, and the chemical and microstructural changes that occur as a result of simulated transients. Fuel from pre- and post-DEH tests and from ongoing, related WRSR test programs in the Power Burst Facility and in the Halden Reactor will be characterized to provide basic input data and to permit verification of the SST fuel-behavior code.

The chemical characterization of the fuel includes the determination of axial and radial burnup profiles, gamma scanning for fission-product activity, and electron-probe microanalysis. Initial gamma-scan measurements of an irradiated Robinson fuel rod indicated that no major redistribution of the examined isotopes has occurred in the axial direction.

The microstructural characterization uses the techniques of quantitative stereology to estimate the radial profiles of pore fraction, pore specific surface, and grain size. Examination of high-burnup microstructures of the Robinson fuel show that in-reactor sintering has occurred in the central region of the fuel. An outer ring, ~20% of the cross-sectional area, shows higher porosity. The mean grain intercept was ~6 μm in the central region and ~3 μm in the outer ring. Calculations indicate that the fraction of interlinked porosity was over 80% in the outer region but <20% in the densified zone. Thus, only a small fraction of the fission gas arriving at the grain boundaries can escape from the hotter central region. This effect can introduce important deviations from the retained gas concentration calculated with less detailed diffusion models.

Apparatus for direct-electrical heating of irradiated fuel pellets and for collecting and analyzing the released fission gases and volatile fission products has been developed, assembled in a hot cell, and proof tested. The correlation between nuclear-accident-heating and electrical-heating conditions has been established for ANC-calculated fuel heating rates and radial temperature profiles for several hypothetical accidents. The existing DEH apparatus and associated electronics can simulate the fuel-temperature profiles and heating rates of a wide range of PCM accidents with the potential for simulating other types of accidents, e.g., BWR flow blockage, LOCA, and RIA. Heating rates to 1500°C/sec can be achieved, and tests can be interrupted to permit the study of the fuel and the released fission products during progressive stages of the transient. Recorded measurements of power or current input and fuel-surface temperature are used to compute radial temperature profiles; improvements in our capability to measure fuel-surface temperatures at low and high temperatures and to calculate temperature profiles are being developed. Gross fuel behavior can be observed and recorded with high-speed movies during the transient test, and changes in fuel length can be monitored.

The fission product collection system utilizes low-temperature charcoal traps for quantitative collection of released fission gases and tantalum Raschig-ring traps for collection of released condensable solid fission products.

The extensively cracked condition of the irradiated LWR fuel available to this program greatly complicates the recovery of specimens suitable for DEH experiments. Fuel-recovery methods must accomplish the objectives of removing the fuel from its cladding, preserving the cylindrical geometry of the fuel pellets and fixing the relative positions of fuel fragments, and ensuring that the pellets and fragments are in good electrical contact. Several fuel-recovery approaches have been tried, but complete success has not been achieved. The Robinson fuel is not firmly bonded to its cladding so separation is not difficult. Impregnation of cracks in the fuel with a fine UO_2 powder suspended in a solution containing a binder was found to bridge the insulating gaps between pellet fragments and also to cement the fragments together. Work is continuing to improve the impregnation technique and the strength of the bond.