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Irradiation Testing of Actinide Transmutation Fuels in the Advanced Test Reactor

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Abstract – The first irradiation experiment to evaluate the technical feasibility of proposed actinide transmutation fuels for the U. S. Accelerator Transmutation of Waste program is currently under design. The goal of this irradiation experiment is to obtain initial irradiation performance data on candidate transmutation fuel concepts. The candidate fuels include non-fertile variations of 1) metallic alloys, 2) nitrides, 3) oxides, and 4) metal-matrix dispersion fuels. These fuels will be irradiated in the form of rodlets in the Advanced Test Reactor in Idaho beginning in September 2002. It is expected that postirradiation examinations will be performed on these fuels at the ~7 and 20 at.-% burnup levels. This paper presents the design of the irradiation test vehicle and the fuel rodlets; the test matrix of fuel variations; the target test conditions; and the planned postirradiation examinations.

I. BACKGROUND

The U. S. Accelerator Transmutation of Waste (ATW) program seeks to develop and demonstrate the technologies needed to transmute the long-lived transuranic actinide isotopes contained in spent nuclear fuel into shorter-lived fission products, thereby dramatically decreasing the required design lifetime of a future deep geologic repository. A vitally important component of that technology will be a non-fertile actinide transmutation fuel form containing the plutonium, neptunium, americium (and possibly curium) isotopes to be transmuted. Such non-fertile fuel forms, especially ones enriched in the minor actinide elements (i.e., Np, Am, Cm), have virtually no irradiation performance data available from which to establish a transmutation fuel form design. Thus, initial scoping-level irradiation tests on a variety of candidate fuel forms are needed.

A series of non-fertile fuels tests is planned for irradiation in the Advanced Test Reactor (ATR) in Idaho. Although the ATW program will likely select a fast neutron spectrum system to achieve the most effective transmutation route, initial irradiation testing of candidate fuels in a thermal reactor will be of great benefit, for the following reasons: 1) experiments in a thermal test reactor are much less expensive than fast reactor tests, making them attractive for scoping-level tests on large numbers of small fuel samples; 2) prototypic powers and temperatures can be achieved in thermal reactors, though the power distribution within the fuels will differ as will the isotopes which produce that power; 3) the burnup rate achievable in a thermal reactor is greater, allowing tests to be completed in less time; and 4) at present there is no U. S. fast neutron spectrum test reactor, and the complications associated with international shipments of nuclear materials are costly and formidable. Furthermore, many fuel performance issues are primarily functions of temperature and/or power, and depend upon neutron spectrum only as a lower order effect. Tests on transmutation fuels is expected to produce useful data regarding such fuel performance issues as irradiation growth and swelling, helium production, gas release, fission product and fuel constituent migration, fuel phase equilibria, and fuel-cladding chemical interaction. Of course, cladding mechanical behavior will be very different between the thermal and fast spectrum environments; however, since the cladding material to be employed in the fast spectrum transmutation system will be a stainless steel alloy traditionally used in fast reactors, its irradiation performance is already well established and need not be demonstrated in these initial tests. Finally, if the ATW program elects to pursue a dual strata approach to transmutation in which the plutonium and (possibly) neptunium components of the actinide mix are irradiated first in thermal reactors prior to insertion in the fast spectrum transmuter, then thermal irradiation tests of such fuels are directly applicable; thus, a number of the fuel forms to be tested in these ATR experiments are legitimate candidates as non-fertile fuel forms for use in thermal reactors (e.g., light water reactors).

Currently, four candidate fuel forms are under consideration by the ATW program: metallic alloys, nitrides, oxides and metal-matrix dispersion fuels. The nitride metallic alloy fuel forms are considered to be the front-runners at this early stage of development, and these fuels will be tested first. Initially, four drop-in experiments are planned, designated ATW-1A, -1B, -1C and –1D. ATW-1A and –1C will contain a variety of nitride fuel compositions; both are expected to be identical in design but destined for different discharge burnup levels. Similarly, ATW-1B and –1D will be identical in design, containing a variety of metallic fuel compositions. An overview of these four experiments is shown in TABLE I.

TABLE I

Overview of the ATW-1A, -1B, -1C and –1D Experiments

ATR Experiment Designation	Fuel Form	ATR Insertion	Target Discharge Burnup*
ATW-1A	Nitride	Sept-2002	5-7%
ATW-1B	Metallic	Sept-2002	5-7%
ATW-1C	Nitride	Sept-2002	20%
ATW-1D	Metallic	Sept-2002	20%

*Burnup in percent of initial heavy metal.

The simultaneous insertion of these four irradiation vehicles is expected in September 2002, contingent upon the ATR operating schedule in that timeframe.

II. FUEL RODLET DESCRIPTION

The fundamental component which contains the fuel specimen is termed a "rodlet". Rodlets are simply miniature fuel rods 6.0-in. in length. Externally each of the rodlets in all four experiments will be identical. Internally, however, the fuel column diameters and heights differ between the nitride and metallic fueled experiments. TABLE II shows the materials used in constructing the rodlets along with their key design dimensions; an axial schematic of a nitride fueled rodlet is shown in Fig. 1.

TABLE II

Fuel Rodlet Design Data

Design Parameter	ATW-1A, ATW-1C	ATW-1B, ATW-1D
Cladding Material	HT9	HT9
Cladding O.D.	0.230-in.	0.230-in.
Cladding I.D.	0.194-in.	0.194-in.
Bond Material	Sodium	Sodium
Fuel Type	Nitride	Metallic
Fuel Smear Density	75%	66%
Fuel Porosity	15%	0%
Fuel O.D.	0.168-in.	0.158-in.
Fuel Height	2.000-in.	1.500-in.
Plenum Volume	0.081 $in.^3$	0.096 $in.^3$

III. FUEL TEST MATRIX

The fuel compositions and arrangements of the nitride fueled rodlets in ATW-1A and ATW-1C will be identical, as will the compositions and arrangements of the metallic fueled rodlets in ATW-1B and ATW-1D. These compositions and arrangements are shown in TABLE III. Note that the metallic alloy compositions are expressed in weight percent, and the nitride fuel compositions will be 36% ZrN by weight.

IV. IRRADIATION VEHICLE DESCRIPTION

The six rodlets will be stacked vertically within a single, stainless steel secondary containment vessel in each experiment. This secondary containment vessel serves two purposes: 1) it provides a second, reliable barrier between the water coolant and the fuel, sodium and fission products, and 2) it provides additional free volume for the expansion of helium and fission gases should the cladding of any number of rodlets be

TABLE III

ATW-1A, -1B, -1C and –1D Fuel Test Matrix

Rodlet	Fuel Test Matrix	
	ATW-1A & -1C*	ATW-1B & -1D†
1	$(Pu_{0.2},Am_{0.8})N\text{-}X\cdot ZrN$	Pu-12Am-40Zr
2	$(Pu_{0.8},Am_{0.2})N\text{-}X\cdot ZrN$	Pu-10Am-10Np-40Zr
3	$(Pu_{0.5},Np_{0.5})N\text{-}X\cdot ZrN$	Pu-40Zr
4	$PuN\text{-}X\cdot ZrN$	Pu-12Am-40Zr
5	$(Pu_{0.50},Am_{0.25},Np_{0.25})N\text{-}X\cdot ZrN$	Pu-30Np-40Zr
6	$(Pu_{0.5},Am_{0.5})N\text{-}X\cdot ZrN$	Pu-60Zr

*X denotes weight fraction of inert ZrN diluent.
†Alloy composition expressed in weight percent.

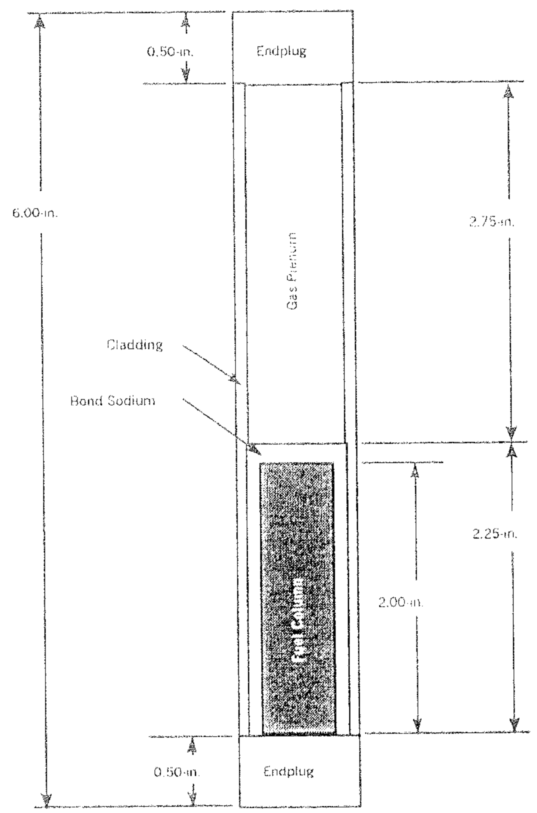

Fig. 1. Nitride fueled rodlet design for ATW-1A and ATW–1C.

breached during irradiation. This containment vessel is shown schematically in Fig. 2, and the relevant design data are given in TABLE IV. The containment vessel design will be identical in all four experiments.

TABLE IV

Design Data for Secondary Containment Vessel

Design Parameter	Value
Containment Material	316SS
Containment O.D.	0.354-in.
Containment I.D.	0.234-in.
Containment Length	58.000-in.
Containment Free Volume	15.56 cm^3
Containment-Rodlet Gap	0.0022-in.

Outboard of this containment vessel is an annular hafnium shroud. The thickness of the hafnium shroud will be designed to produce the target fuel powers in the rodlets, as will be discussed in the next section. Presently it is assumed that the hafnium shroud will be at least 0.070-in. thick.

V. EXPECTED IRRADIATION CONDITIONS

The experiments ATW-1A, -1B, -1C and –1D, as presently conceived, have been designed for irradiation in the four available "outboard" A-holes on the north side of the ATR core (i.e., the two outboard A-holes associated with the NW lobe and the two associated with the NE lobe). The expected thermal conditions at beginning-of-life for these experiments have been calculated and are shown in TABLE V.

The linear heat generation rates (LHGR) shown for each rodlet in TABLE V have been assumed. The design objective for these experiments is to have heat generation rates of 300 W/cm for rodlets 3 and 4 in each experiment, and it is assumed that rodlets 1, 2, 5 and 6 will operate at lower linear powers. The hafnium shroud will be sized to achieve this design objective.

Also shown in TABLE V are conservative plenum pressures estimated for each rodlet at a maximum burnup limit of 25 at.-%; these values were obtained assuming 25% release of all fission gas and 50% release of all helium generated in the nitride fueled rodlets, and 80% and 100% release of fission gas and helium, respectively, from the metallic fueled rodlets. The upper and lower plenums of the containment vessel have been sized to reduce the total pressure on the outermost boundary to below 235 psi in the event that all rodlets fail during irradiation.

The ATW-1A and –1B experiments will be discharged from the ATR upon reaching a target peak

TABLE 5

Estimated Operating Conditions for ATW-1A, -1B, -1C and –1D

Rodlet	LHGR (W/cm)	Maximum Burnup (at.-%)	Plenum Pressure (psi)	Peak Temperatures		
				Coolant (°C)	Containment (°C)	Clad (°C)
---------- ATW-1A and -1C ----------						
1	231	25.0	1500.4	53.7	142.0	425.6
2	250	25.0	505.9	55.6	151.1	449.2
3	300	25.0	161.6	57.8	172.4	506.3
4	300	25.0	161.9	60.1	174.7	508.0
5	250	25.0	599.3	61.9	157.4	454.2
6	231	25.0	1010.3	63.7	151.9	433.4
---------- ATW-1B and -1D ----------						
1	231	25.0	770.7	53.3	141.7	425.4
2	250	25.0	741.3	54.7	150.4	448.7
3	300	25.0	348.2	56.4	171.2	505.4
4	300	25.0	860.3	58.1	172.9	506.7
5	250	25.0	355.5	59.5	155.2	452.4
6	231	25.0	155.3	60.3	149.0	431.1

burnup of between 5 and 7 at.-%. ATW-1C and –1D will be discharged upon reaching a minimum peak burnup of 20 at.-%.

VI. POSTIRRADIATION EXAMINATION

Upon discharge of each experiment from the reactor and a cooling period in the ATR canal, the fueled rodlets will be shipped to a hot cell facility for postirradiation examination (PIE). Of particular interest during PIE will be fission gas bubble morphology and gas release; helium gas generation and release; fuel swelling; fuel irradiation growth; fuel-clad chemical interaction; fuel constituent migration; solid fission product behavior; and fuel phase stability. Anticipated exams include cladding profilometry; neutron radiography; plenum gas pressure measurement and gas compositional analysis; burnup determination; and fuel metallography, ceramography and microscopy.