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LIGHT WATER REACTOR MIXED-OXIDE FUEL IRRADIATION EXPERIMENT

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LIGHT WATER REACTOR MIXED-OXIDE FUEL IRRADIATION EXPERIMENT* †

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ABSTRACT

The United States Department of Energy Office of Fissile Materials Disposition is sponsoring and Oak Ridge National Laboratory (ORNL) is leading an irradiation experiment to test mixed uranium-plutonium oxide (MOX) fuel made from weapons-grade (WG) plutonium.[1] In this multiyear program, sealed capsules containing MOX fuel pellets fabricated at Los Alamos National Laboratory (LANL) are being irradiated in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL).[2] The planned experiments will investigate the utilization of dry-processed plutonium, the effects of WG plutonium isotopics on MOX performance, and any material interactions of gallium with Zircaloy cladding.

1. INTRODUCTION

Several technical issues must be resolved before Light Water Reactor (LWR) operating licenses can be amended to allow use of MOX fuel made from WG plutonium and depleted uranium. Demonstration of the in-reactor thermal, mechanical, and fission gas release behavior of the prototype fuel is required. One experiment containing WG MOX fuel has been designed and is being irradiated in the ATR at the INEEL.[1,2]

Figure 1 illustrates the test capsule cross section and lists the responsibilities of the participating laboratories. Table 1 provides the complete test matrix, which will be accomplished in a series of steps. The first experiment, which involves fuel Types 2 and 3, was inserted for irradiation in February 1998. The compositions of these test fuels were selected to focus primarily upon investigation of the behavior and acceptability of the gallium present in the MOX.

The MOX pellets were sealed within Zircaloy fuel pins at LANL before shipment to INEEL, where the fuel pins themselves were sealed within stainless steel capsules. Although the fuel pins are hermetically sealed, the ATR-approved containment for these experiments is provided by the stainless steel capsules. The first test involves nine of these capsules, including five Type 2 fabricated from gallium-bearing PuO_2 and four Type 3 fabricated from PuO_2 that has been thermally treated for gallium removal.[3] Both test fuels were manufactured to the generic LWR MOX fuel pellet specification developed specifically for these experiments.[4]

The nine capsules are located within a basket assembly as illustrated in Figure 2, and the entire test assembly was placed in one of the four small I-holes (3.81 cm diameter) of the ATR reflector, as shown in Figure 3. The I-hole basket was designed and built to accommodate the nine fuel capsules within three internal holes, with three fuel capsules being placed in each hole. The I-24 position was selected as the best

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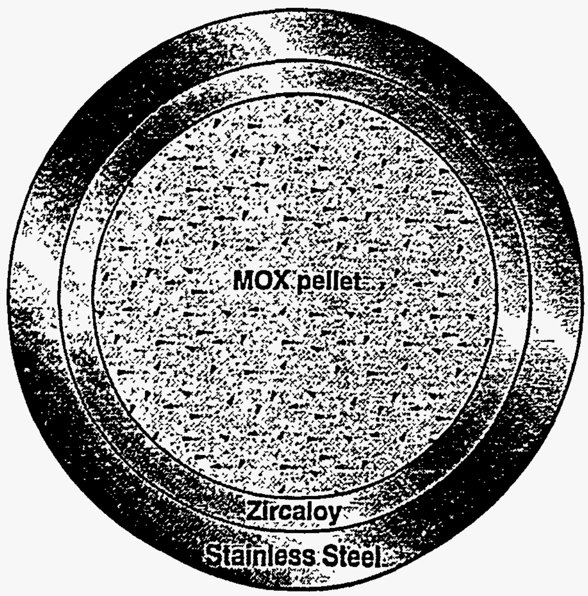
ORNL-DWG 97-2435 EFG

Single Capsule



Oak Ridge:
- Manages overall program
- Manufactures test hardware

Los Alamos:
- Fabricates fuel
- Loads fuel pins
- Welds pins and ships to Idaho

INEEL:
- Loads steel capsules
- Welds capsules
- Operates Advanced Test Reactor
- Ships irradiated capsules to ORNL

Oak Ridge:
- Performs PIE

Figure 1. The planned test irradiations are a multilaboratory effort.

Table 1. The complete test matrix includes three WG MOX fuels plus RG MOX and an LEU control.

Fuel Type	Description	Initial Feed	Pu Purification	Pu to PuO_2 Conversion	Intended Burnup (GWd/MT)
1	5% WG MOX	1% Ga WG Pu	Aqueous	Oxalate	8,20,30
2			None	Hydride	
3			Thermal	Hydride	
4	6% RG MOX	RG Pu	Aqueous	Oxalate	22
Control	8.6% LEU	LEU	n/a	n/a	22,30

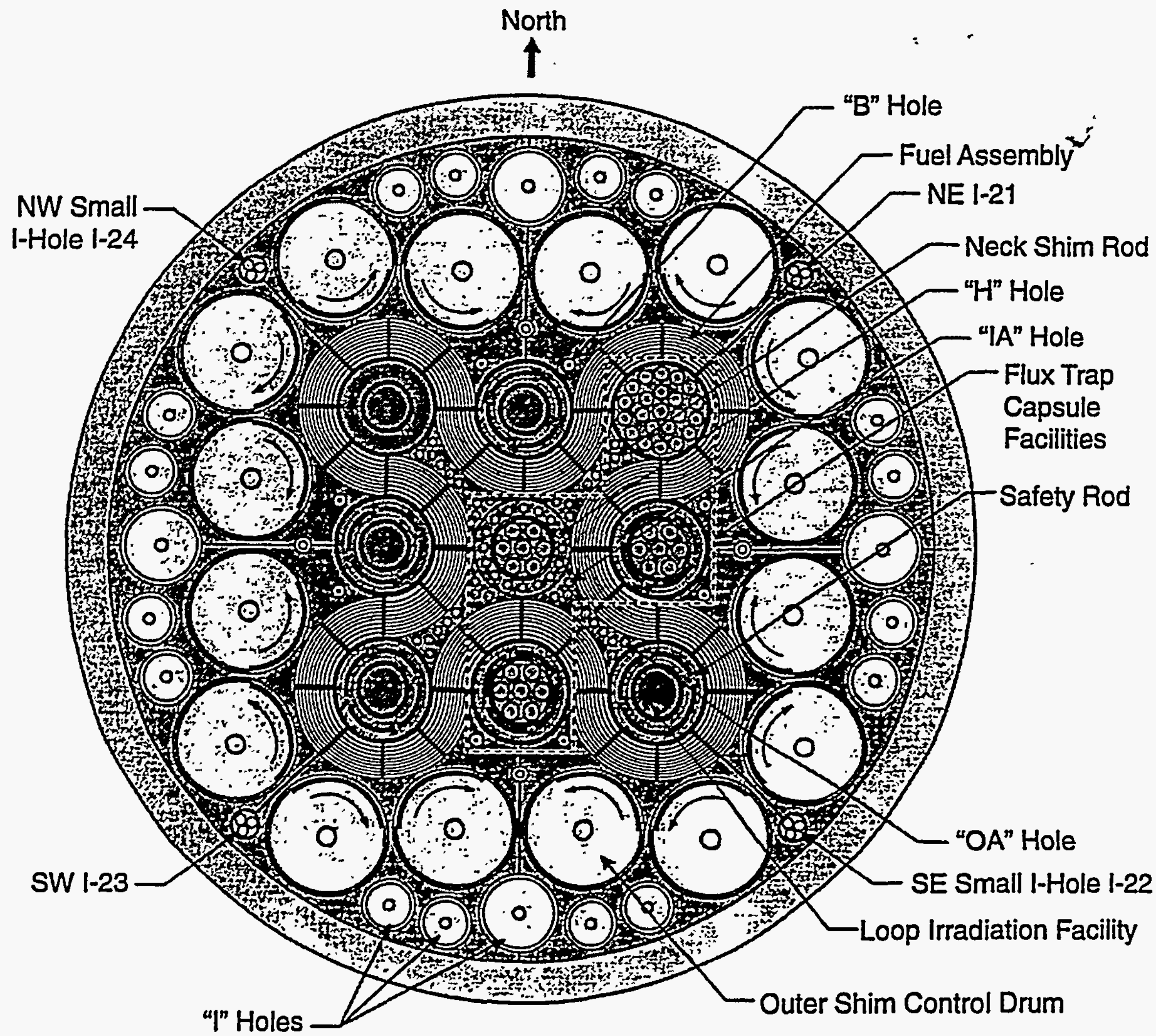


Figure 3. Plan view of the Advanced Test Reactor.

location for the MOX test assembly because it provides the best match to the desired linear heat generation rate (LHGR) of 26 kW/m (8 kW/ft).[5,6] A simple cylindrical Inconel shroud provides the desired neutron flux.

One test fuel capsule of each type will be removed from the reactor in the fall of 1998 after reaching about 8 GWd/MT to provide an opportunity for early indication of the effects of residual gallium at low burnup.[7] One additional fuel capsule of each type will be removed in late 1999 after reaching about 20 GWd/MT to provide intermediate indication of any developing trends. The remaining fuel capsules will be irradiated to approximately 30 GWd/MT, corresponding to two cycles of irradiation in a commercial reactor, and removed in early 2001. All capsules will undergo Postirradiation Examination (PIE) at ORNL.[8]

The fuel is being irradiated at prototypic LWR LHGRs of 19.7–32.8 kW/m, and with near-prototypic LWR pellet temperature profiles and pellet surface temperatures of about 400°C.[9] Fuel pellet dimensions are 0.831 cm in diameter and 1.016 cm in length. The total fuel length in each fuel pin is 15.24 cm (i.e., 15 pellets). The pellets are completely contained in Zircaloy (i.e., within a Zircaloy tube with end caps welded into both ends, thereby forming a fuel pin). An upper spring maintains the stack configuration within a helium atmosphere in each fuel pin.

In order to conform with ATR requirements and to achieve the proper fuel cladding and pellet outer surface temperatures, each fuel pin is contained within a stainless steel capsule. Pressurization of the capsule requires prior failure of the fuel pin boundary, and calculations have demonstrated that the maximum possible pressure arising from fission gas release is 5.29 MPa (767 psi).[10] The capsules have been designed for an internal pressure of 5.86 MPa (850 psi), which meets the applicable American Society of Mechanical Engineers (ASME) code requirements.[11,12]

2. NEUTRONICS ANALYSES

Figure 3 shows a plan view of the ATR, with numerous potential irradiation positions. The experiment requires neutron flux levels and LHGRs that are typical of those in commercial LWRs.[11] The first main neutronics task was to select an irradiation position that best matches the design requirements. A Monte Carlo N-Particle Transport Code (MCNP)[13] model of the ATR core was used to analyze a variety of early experiment designs.[5,6]

From these studies, several observations can be stated:

1. Neutron flux levels vary greatly with core position.
2. Higher plutonium loadings per capsule produce higher LHGRs and temperatures.
3. Because of self-shielding, adding more capsules close together reduces the thermal flux and thus the LHGRs and burnups.
4. Double encapsulation is required to achieve prototypic clad temperatures because of the low temperature of the ATR primary coolant.

These analyses were used to help set MOX testing requirements, select the desired test location, estimate the irradiation time required to achieve a desired burnup level, and ensure that the capsule thermal/hydraulic limits will not be exceeded.

Because the thermal neutron flux expected in the chosen irradiation position, I-24, is too high to satisfy the 19.7–32.8 kW/m design requirement without shielding, MCNP calculations were performed with various shroud materials and thicknesses. These calculations showed that a 2-mm (80-mil) thick Inconel shroud achieves the desired heat rates in the I-24 position. With this shield, the average linear heat rates in each capsule vary from 20.0 to 30.3 kW/m for fresh WG MOX fuel.

To benchmark the neutronics analyses, a means for incorporating removable dosimetry was included in the basket assembly mechanical design. The dosimetry is comprehensive enough to ascertain (with the aid of neutronics analyses) whether or not the fuel capsules have operated within the desired bounds of 19.7–32.8 kW/m. MCNP and the ORIGEN2 isotope depletion code[14] will be used together to track fuel burnup and heat rates as a function of irradiation time.

Detailed radial, axial, and azimuthal power distributions throughout the MOX pins are required to determine temperature distribution. Temperature distributions were needed to make sure the MOX pins meet the ATR safety requirements and to analyze the behavior of gallium migration. The INEEL coauthors developed a new Monte Carlo procedure (minicell method) for accurately determining power distribution in the MOX pins located in the ATR reflector.[15] Conventional LWR methods are not appropriate because of the unique ATR geometry. The minicell method was validated by comparison of radial power profiles generated by minicell and deterministic methods.

MCNP and ORIGEN2 have been linked to allow accurate prediction of fuel depletion in the MOX pins. Applying this capability with the minicell method allows calculation of detailed nuclide concentration distributions within the MOX pins. For example, Figure 4 shows detailed radial power profiles at beginning and end of capsule irradiation as determined by this Monte Carlo depletion methodology. This information will be useful for comparisons with measurements made during PIE. The prediction of plutonium and fission product concentration profiles in irradiated MOX pins may provide insights into MOX fuel performance.

In summary, a simple, uninstrumented, test assembly containing nine MOX fuel capsules with local flux monitor wires was inserted into the ATR. Important neutronics parameters were computed using novel methods developed based on Monte Carlo. These computations have led to an experiment design with WG MOX fuel that meets all design requirements and that will, upon completion of PIE, produce information on the behavior and acceptability of gallium impurities in WG MOX fuel.

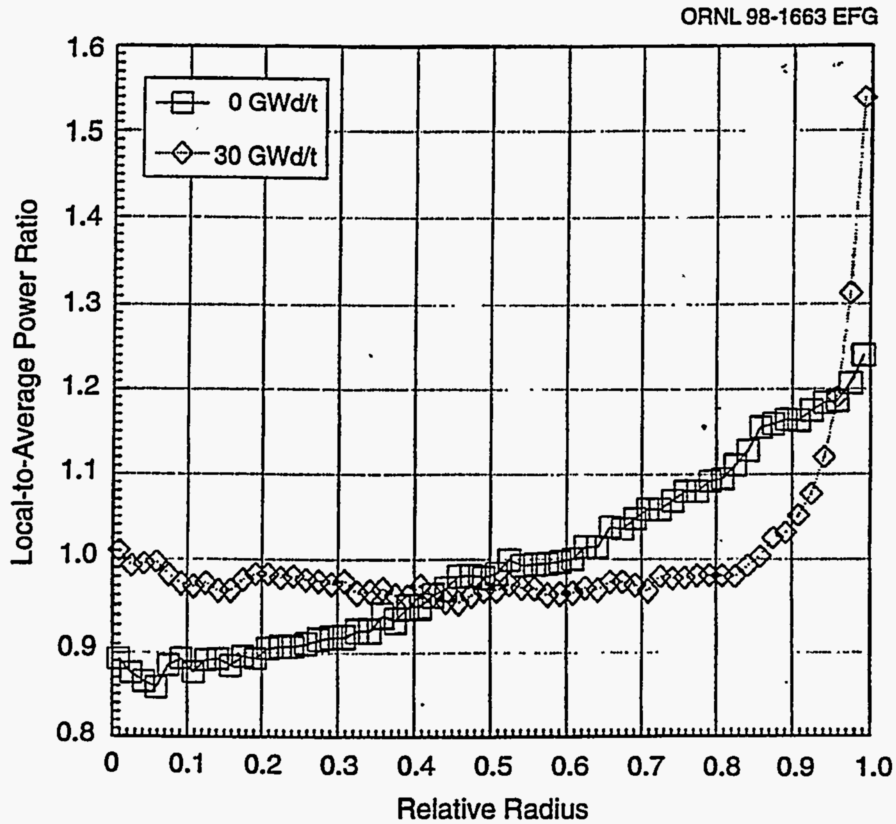


Figure 4. Detailed radial fission power profiles in WG-MOX fuel pins at 0 and 30 GWd/t burnup.

NOMENCLATURE

ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
DOE-MD	United States Department of Energy Office of Fissile Materials Disposition
Ga	gallium
INEEL	Idaho National Engineering and Environmental Laboratory
LANL	Los Alamos National Laboratory
LEU	low-enriched uranium
LHGR	linear heat generation rate
LWR	Light Water Reactor
MCNP	Monte Carlo N-Particle Transport Code
MOX	mixed uranium-plutonium oxide
ORNL	Oak Ridge National Laboratory
PIE	Postirradiation Examination
RG Pu	reactor-grade plutonium
WG Pu	weapons-grade plutonium

REFERENCES

1. B. S. Cowell and S. A. Hodge, "Fissile Material Disposition Program Light Water Reactor Mixed-Oxide Fuel Irradiation Test Project Plan," ORNL/MD/LTR-78, Rev. 1, Oak Ridge National Laboratory (1998).

2. J. M. Ryskamp, S. A. Hodge, and B. S. Cowell, "A Mixed-Oxide Fuel Irradiation Experiment in the Advanced Test Reactor," Proceedings of the Sixth International Conference on Nuclear Engineering, San Diego, CA, May 10–15, 1998, accepted for publication.

3. H. R. Trellue, et al., "Nuclear Fuels Technologies Fiscal Year 1997 Research and Development Test