ORNL/MD/LTR-187
Level 2

DESIGN, FUNCTIONAL, AND OPERATIONAL REQUIREMENTS FOR PHASE IV OF THE AVERAGE-POWER MIXED-OXIDE IRRADIATION TEST

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March 2000

Fissile Materials Disposition Program
Engineering Technology Division

Revision History

Revision Number	Date Issued	Reason for Revision
0	March 2000	Initial Issue

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March 30, 2000

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Design, Functional, and Operational Requirements for Phase IV of the Average-Power Mixed-Oxide Irradiation Test **- ORNL/MD/LTR-187**

This document provides the bases for the design and operation of the Phase IV irradiation of the Fissile Materials Disposition Program (FMDP) light water reactor average-power mixed-oxide irradiation test to be conducted in the Advanced Test Reactor (ATR) at Idaho National Engineering and Environmental Laboratory (INEEL).

The requirements for initial installation into the ATR and irradiation Phases I, II, and III were provided in *Design, Functional, and Operational Requirements for the Advanced Test Reactor Mixed Oxide Fuel Irradiation Experiment* (ORNL/MD/LTR-76). At the completion of Phase III the maximum fuel burnup in any one fuel pin will be 30 GWd/MT as stated in ORNL/MD/LTR-76.

In Phase IV the maximum burnup will be extended to as much as 50 GWd/MT, resulting in the potential for higher fission gas release. At the same time, because of fuel burnup, it will not be possible (nor desirable) to operate at the same linear heat generation rates. In addition, since the inception of Phase I, the safety analysis requirements at the ATR have been upgraded. To address all of these issues it was decided to issue this new requirements document.

This is a Level-2 document as defined in the *Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan,* ORNL/MD/LTR-78.

Sincerely,

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Fissile Materials Disposition Program

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U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-96OR22464

DESIGN, FUNCTIONAL, AND OPERATIONAL REQUIREMENTS FOR THE PHASE IV OF THE AVERAGE-POWER MIXED-OXIDE IRRADIATION TEST

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1.0 Project Requester

The United States Department of Energy Office of Fissile Material Deposition (DOE-MD) is sponsoring irradiation experiments in the Advanced Test Reactor (ATR) at the Idaho National Engineering and Environmental Laboratory (INEEL) for the purpose of testing mixed uranium-plutonium oxide (MOX) fuel derived from weapons components. This document establishes the Requirements for Phase IV of the average-power test experiment and is a Level-2 document as defined in the *Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan* (ORNL/MD/LTR-78). The requirements for Phases I, II, and III were established in *Design, Functional, and Operational Requirements for the Advanced Test Reactor Mixed Oxide Fuel Irradiation Experiment* (ORNL/MD/LTR-76).

2.0 Current Need for This Project

The DOE Fissile Materials Disposition Program (FMDP) has announced that reactor irradiation as MOX fuel is one of the alternatives for disposal of surplus weapons-usable plutonium. MOX fuel has been utilized domestically in test reactors and on an experimental basis in a number of commercial light water reactors (CLWRs). Most of this experience has been with plutonium derived from spent low enriched uranium (LEU) fuel, known as reactor grade (RG) plutonium. One of the goals of this irradiation project is to demonstrate that the unique properties of the surplus weapons-derived or weapons-grade (WG) plutonium do not compromise the applicability of this MOX experience base. Other goals include demonstration of the domestic infrastructure necessary to carry out MOX testing and the addition of data obtained from irradiation of gallium-containing fuel to the data base. This information is required for resolution of generic CLWR fuel design issues. Phase IV of this average-power test (APT) will extend the range of investigation associated with the completed Phases I and II, and the currently underway Phase III.

3.0 Project Function

The average-power mixed-oxide irradiation project utilizes a small (1.5-in. diameter) I-hole (initially I-24) to irradiate fuel pins containing various fuel types as described in the *Fissile Materials Disposition Program Light Water Reactor Mixed Oxide Fuel Irradiation Test Project Plan* (ORNL/MD/LTR-78). Each fuel pin is contained in a stainless steel capsule. Several I-hole baskets have been designed and built, each capable of accommodating nine fuel capsules within three internal holes with three fuel capsules being placed in each basket hole. Each basket assembly provides shielding as well as placement locations for the test capsules. Different baskets may be employed during the test in order to apply the shielding strengths most appropriate to prevailing reactor-operating configurations.

While the irradiation testing will be continuous for about six calendar years, selected fuel capsules are removed at various burnup levels for early postirradiation examination (PIE). When irradiated fuel capsules are removed, they will be replaced either with fresh fuel capsules or solid-stainless steel (SS) capsules.

4.0 Design Requirements

4.1 Customer Requirements

The FMDP desires to irradiate a variety of MOX fuel pellets to burnup levels as high as 50 GWd/MT. During Phases I, II, and III of this project fuel pellets were irradiated to burnup levels as high as 30 GWd/MT. In Phase IV it is desired to extend the burnup to as high as 50 GWd/MT while maintaining the linear heat generation rate (LHGR) in the range of 2 to 8 kW/ft. Initial fuel pellet dimensions were 0.327 inches in diameter and 0.400 inches in length with tolerances in accordance with fuel specification ORNL/MD/LTR-75, *Technical Specification: Fuel Pellets for the Light-Water Reactor Irradiation Demonstration Test*. To minimize end-effects, accommodate multiple specimens over the length of the ATR, and ensure reasonable pellet surface contact with generic CLWR Zircaloy cladding; the initial total fuel length in each fuel pin was 6.0 inches (+ 0.0, -0.2 inches), i.e., 15 pellets at 0.400 inches in length each.

The fuel pellets are completely contained in Zircaloy, i.e., within a Zircaloy tube with end caps welded into both ends, thereby forming a fuel pin. The atmosphere inside the Zircaloy fuel pin after assembly was helium at approximately 1 atmosphere pressure with a minimum purity level of 99.995%. Oxygen level in the helium fill gas did not exceed 10 ppm; hydrogen levels in all forms did not exceed 1 ppm; carbon levels in all forms did not exceed 10 ppm. Glove box atmosphere was monitored to ensure purity levels. At no time did the Zircaloy components come in contact with aluminum.

In order to conform with ATR requirements and to achieve the proper fuel cladding and pellet outer surface temperatures, it was necessary to place each pin inside a stainless steel capsule. The capsules were welded inside a glove box. The atmosphere inside the glove box was helium at approximately 1 atmosphere pressure with a minimum purity level of 99.9 %. Oxygen level in the glove box did not exceed 500 ppm.

Because it is intended to remove fuel pins at several burnup levels, it is important that individual capsules can be removed from the irradiation basket assembly during the course of a normal ATR refueling outage. When capsules are removed, they are to be replaced with solid-stainless steel dummy capsules. The design incorporates a means of tracking the orientation of the irradiation basket assembly, and of the capsules within the basket assembly, during the entire irradiation period of each fuel pin. Each capsule assembly was marked with a unique identification [consistent with American Society of Mechanical Engineers (ASME) code requirements] that is visible under 16 feet of water. The basket assembly was also be marked with a unique identification on the top to enable determination of basket identity and orientation relative to the core centerline.

To achieve the proper thermal neutron flux such that the linear heat generation rate did not exceed 10 kW/ft, an Inconel thermal neutron shield was incorporated into the basket assembly for Phase I of the average-power test performed in small I-hole position I-24. Because of fuel burnup, it was not necessary to have a thermal neutron shield for Phases II and III, and therefore an all-aluminum basket assembly was used during this period of irradiation. For Phase IV, neutronic analyses shall be completed to determine the most appropriate irradiation position (either small I-hole, or B-hole) and basket assembly design that will permit operation to continue with LHGRs in the range of 2 to 8 kW/ft.

To benchmark the neutronic analyses, a means for incorporating removable dosimetry was included in the basket design. The dosimetry shall be comprehensive enough to ascertain (with the aid of neutronics analyses) whether or not the fuel pins have operated within the desired bounds of 2 to 8 kW/ft.

4.2 Code Requirements

The ATR Upgraded Final Safety Analysis Report (UFSAR), section 10.1.7.3.2 (Code Compliance of Experiment Containment) states: "Experiment containment that holds pressure greater than 235 psig, or contains material that can generate pressure pulses greater than 430 psig, must have a design that meets the intent of ASME Section III, Class 1 standards, or the ability, demonstrated by prototype testing or other means, to withstand service conditions without failure." Calculations (see ORNL/MD/LTR-185) will show that the maximum possible pressure that can be generated inside the capsule during Phase IV is about 1109 psia. (Capsule pressure this high would only occur with complete fission gas release from the fuel matrix, in addition to failure of the Zircaloy fuel pin.) Based on this information, the stainless steel capsule containment shall meet the intent of the ASME B&PV Code (1998 Edition with addenda up to and including July 1, 1999) Section III, Class 1, standards, and the design pressure shall be 1200 psi.

In the event that capsule design is not in strict compliance with the code, the design analyses must note the differences and include the appropriate justifications, including the rationale as to why the code standards are not compromised. Any deviations from strict code compliance will require INEEL review and approval and could possibly constitute a deviation from the current UFSAR, requiring DOE approval.

Markings applied to the stainless steel capsule assemblies are consistent with ASME code requirements.

4.3 Mechanical Design Requirements

The containment shall meet the intent of the design requirements set forth in ASME B&PV Code Section III, Article NB-3000. The operational service conditions for the stainless steel containment shall meet the requirement of Section III Article NB-3222 (Level A Service Limits), Article NB-3223 (Level B Service Limits), Article NB-3224 (Level C Service Limits), Article NB-3225 (Level D Service Limits), and Subsection NCA Article 2142.4 (b). The service limits, as applied to this experiment, are described as follows:

Level A Service Limits:

These conditions are the desired operating conditions for the system including reactor power, temperature, pressure, reactor scrams, etc. The identified loads for a total of 120 cycles throughout the lifetime of the MOX APT capsule are:

(1) Capsule external operating pressure of 390 psig with an internal pressure of 0 psig (reactor operating conditions at beginning of life) at the capsule design temperature of 500°F.

(2) External pressure of 0 psig with the capsule internal design pressure of 1200 psig at the capsule design temperature of 500°F.

Level B Service Limits:

These events are upsets or deviations from the normal operating conditions which are anticipated to occur often enough that the design shall be capable of withstanding these conditions without damage requiring repair or resulting in operational impairment. Level B events are typically 110% reactor lobe over-power conditions, 110% reactor over-pressure conditions, reactor temperature variations, etc. The identified loads for a total of 31 cycles per year for the MOX APT capsule are:

(1) Capsule external design pressure of 429 psig with an internal pressure of 0 psig at the capsule design temperature of 500°F.

(2) Reactor lobe over-power at 110% of maximum identified power for the capsule.*

(3) A core differential pressure of 130 psid (per the ATR High Flow Event identified in Section 7.2.3 of Report PR-T-78-003, Rev. 6).

Level C Service Limits:

These events have a low probability of occurring (unlikely) but are included to provide assurance that no gross loss of structural integrity will result if the event occurred. These sets of limits permit large deformations in areas of structural discontinuity, which may necessitate the removal of the component or support from service for inspection or repair of damage to the component or support. Typical Level C events are capsule misloadings, 120% reactor lobe over-power conditions, 120% reactor over-pressure events, etc. The identified loads for a total of 1 cycle throughout the lifetime for the MOX APT capsule are:

(1) Capsule external pressure of 468 psig with an internal pressure of 0 psig at the capsule design temperature of 500°F.

(2) Reactor lobe over-power at 120% of maximum identified power for the capsule.*

Level D Service Limits:

These events have an extremely low probability of occurring (extremely unlikely). These loadings and stress limits permit gross general deformations with some consequent loss of dimensional stability and damage requiring repair or component replacement. Typical Level D events are total loss of reactor coolant flow, seismic events, etc. The identified loads for a frequency of 1 cycle in a lifetime for the MOX APT capsule are:

(1) No applicable loads have been identified for the MOX APT capsule.

* As agreed to by INEEL personnel, these requirements may be satisfied for this experiment by using a heat transfer coefficient of 20% below the calculated best value.

4.4 Design Conditions

The stainless steel capsule design shall be verified for an internal pressure loading of 1200 psi at 500°F with no external loading. The intended range of LHGRs for Phase IV of the APT is 2-8 kW/ft. However, the gas and material temperatures used in these analyses shall be those predicted for an LHGR of 9.0 kW/ft, and a surface heat transfer coefficient at least 20% below the calculated best value. The stainless steel capsule shall also be verified for an externally loaded condition of 429 psig at 500°F with no internal loading. The design life of the stainless steel capsule shall be eight years.

During reactor operation in the pressurized mode at a lobe power of 60 MW, the following thermal-hydraulic criteria shall be met for Phase IV:

(i) The departure from nucleate boiling (DNB) ratio shall always be greater than two.

(ii) The rise in bulk primary coolant temperature along the experiment hot track shall be less than half the value that would cause flow instability.

(iii) All criteria shall be met with either two or three primary coolant pumps in operation.

The experiment shall be verified for adequate decay heat removal during no-flow conditions such as reactor outages and storage in the canal.

The Experiment Basket was designed to withstand a core pressure drop of 130 psid (per the ATR High Flow Event identified in Section 7.2.3 of Report PR-T-78-003, Rev. 6).

4.5 Material Requirements

The fuel pellets irradiated in each fuel pin assembly meet the material requirements of fuel specification ORNL/MD/LTR-75. The fuel pins were fabricated from Zircaloy-4 tubing from Sandvik Special Metals Corp. Lot No. DKL85. The stainless steel capsule assemblies were fabricated from Type-304L material meeting appropriate ASME requirements. The basket assembly was fabricated from Type 6061-T6 aluminum. Basket shields for Phase IV, should they be required, shall be constructed of aluminum, inconel, or stainless steel meeting appropriate ASTM specifications.

At no time during fabrication were the Zircaloy components permitted to come in contact with aluminum.

All new Phase IV materials to be used in this irradiation experiment shall be reviewed and approved by cognizant INEEL personnel.

All new Phase IV material requirements shall be documented on the engineering drawings, and the fabrication data package shall contain documentation confirming material compositions.

5.0 Operational Requirements

The completed fuel pin assemblies and stainless steel capsule assemblies were leak tested to the criteria that no leak greater than 1×10^{-8} std. cc/s shall be permitted at 1 atm pressure differential. The outside surfaces of the fuel pin assemblies and the stainless steel assemblies were proven to have contamination levels low enough to be accepted at INEEL for routine operations prior to installation in the ATR.

Phases I, II, and III of the APT were carried out in the small I-hole position I-24. While it is expected that Phase IV will continue in a small I-hole (either I-23 or I-24), should the project desire to accelerate the burnup rate, it may become desirable to move to a higher flux position such as a B-hole. This shall be permitted provided the LHGR does not exceed 8 kW/ft. Because no instrumentation is included, the only method of assessing the actual heat generation rate is through calculations. Prior to each fuel cycle, INEEL personnel shall perform calculations that will predict the LHGR for each fuel pin as a function of time during that cycle. When it is anticipated that the planned reactor power level will result in a LHGR greater than 8 kW/ft as an average in any fuel pin, approval to continue the irradiation must be obtained from the cognizant

Oak Ridge National Laboratory (ORNL) in-reactor test project manager. If the predicted power level is unacceptable, the entire experiment shall either be: 1) moved to another irradiation position that will provide the acceptable neutronic conditions, 2) placed in a basket assembly with an appropriate shield, or 3) removed from the reactor until an acceptable position can be identified.

Neutron dosimetry shall be removed from the test assembly as agreed upon by the cognizant ORNL and INEEL personnel.

It is understood that the change out of capsules cannot be performed inside the reactor vessel, and the entire test assembly will have to be moved through the discharge chute to the canal to perform the change out. It is desired that the test assembly remain within 45 degrees of vertical at all times, so that the individual capsules maintain their orientation inside the basket assembly. INEEL shall provide the necessary tools that ensure the basket remains in the near-vertical position during in-vessel and canal handling.

No anticipated failure of the experiment shall adversely affect operation of the ATR.

6.0 Quality Assurance Requirements

The Quality Assurance (QA) Programs implemented for this experiment at each of the three laboratories involved meet the requirements of 10 CFR 830.120.

The design, parts fabrication, and assembly processes performed at ORNL were in accordance with QA requirements of the Engineering Technology Division (ETD) Project QA Assessment for Irradiation Experiments. The fuel pellets were fabricated by LANL in accordance with fuel specification ORNL/MD/LTR-75. Loading of the pellets into the Zircaloy fuel pins, and the two closure welds for these pins will be performed in accordance with applicable LANL QA documents. The loading of fuel pins into the stainless steel capsules, making the two closure welds on the stainless steel capsules, loading of capsules into the basket assembly, loading of the basket assembly into the ATR, routine analyses for surveillance of the test, fabrication and handling of removable dosimetry, and handling of the test assembly in the ATR canal were, and shall continue to be performed in accordance with applicable INEEL documents. All end cap welds were made using a qualified welding process and leak tested in accordance with the criteria specified in Section 5.

INEEL Nuclear Operation Quality Assurance personnel cognizant of the ATR QA Program requirements reviewed the ORNL and LANL QA programs as they apply to this experiment. Insertion of the test assembly into the ATR was dependent upon INEEL's concurrence with the adequacy of these programs.

7.0 Documentation

Documentation to be prepared to support the ATR average-power (Phase IV) MOX irradiation experiment shall include:

a. Test Project Plan
b. Design, Functional, and Operational Requirements Document (this document)
c. Design Analyses
    i. Neutronics
    ii. Thermal/Hydraulics
    iii. Stress
    iv. Experiment Safety
d. Design Review Closure Letter(s)
e. Capsule Assembly Loading/Operation Schedule
f. Experiment Safety Assurance Package