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MDO-723-0015
January 9, 2006
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B-MT(EDT)S-028

The Manager
Schenectady Naval Reactors Office
United States Department of Energy
Schenectady, New York

The Manager
Pittsburgh Naval Reactors Office
United States Department of Energy
West Mifflin, Pennsylvania

Subject: Multiple Irradiation Capsule Experiment (MICE)-3B Irradiation Test of Space Fuel Specimens in the Advanced Test Reactor (ATR) - Closeout Documentation for Naval Reactors (NR) Information (U)

Enclosures: (1) Test Details for MICE-3B Space Material Capsules (Beale, Regan, Noe, Kundrat, Petranka)
(2) Post Irradiation Examination Plan for MICE-3B Space Materials (Regan)
(3) Envisioned Future Fuel Testing in the MICE Facility (Regan, Chen, Beale)
(4) Compliance Assessment for MICE-3B UO₂ Capsule (Henning)
(5) MICE-3B Capsule and Fuel-Can Design Report (Petranka)

Dear Sirs:

This letter and enclosures submit to Naval Reactors (NR), for information, documentation of the effort to prepare an irradiation test of space reactor fuel materials using the Multiple Irradiation Capsule Experiment (MICE) test facility in the Advanced Test Reactor (ATR). The purpose of this test was to provide the first UO₂ fuel performance data at space reactor operating conditions. The schedule of this test included beginning of irradiation in March 2006 with post irradiation data becoming available beginning mid 2008. This test is now canceled upon the termination of the Naval Reactors program effort to deliver a space reactor for Project Prometheus.

I. Background of Test Program

The Prometheus-1 reactor fuel was required to perform at high temperatures (1500 – 1700 K) and low fission rates ($1 \times 10^{12} - 2 \times 10^{12}$ fissions/cm³/sec) (Reference (a)). Because there are few data on the performance of nuclear fuels under these specific conditions, these requirements necessitated a testing program to evaluate fuel performance under the proposed Prometheus conditions. At the start of the test program, both UO₂ and UN fuels were being considered; however, before the proposed MICE-3B test was finalized, the Naval Reactors Prime Contractor Team (NRPCT) recommended the use of uranium dioxide (UO₂) (Reference (b)). While significant materials performance uncertainties/risks remain with UO₂, the decision to pursue this fuel system was made because it was judged to be more amenable to engineering solutions in the available timeframe. There was less confidence that UN materials performance issues could be resolved in time to guarantee acceptable performance. Research and planning efforts were underway to retire the remaining uncertainties with UO₂ with MICE-3B

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being the first of several tests necessary for the characterization of Prometheus-1 fuel. While not selected as a candidate fuel for this mission, a limited effort to characterize the UN fuel was also planned to retire some of the key risks associated with this fuel. The information learned from this latter effort may enable the use of UN fuel for a future mission.

The MICE facility in the northeast-loop of the ATR (Figure 1) was to be used as the first test platform to resolve some of the performance uncertainties with UO_2 for space reactor applications. There are two primary locations in the MICE facility where this fuel could be tested and each location can accommodate a test train of five capsules. Each of the five capsules can provide a high-temperature irradiation environment with independent temperature monitoring and active temperature control by varying the helium-neon sweep gas ratio. In addition, there is on-line fission gas detection and analysis capability. (In addition, secondary locations of the MICE facility can accommodate two test trains for core structural materials without instrumentation). Reference (c) gives a detailed description of the MICE test platform and its capabilities. The test train (M-2-103; MICE-3B) was scheduled to start in March 2006; the top two capsules (tiers 4 and 5) were to be used by the NRPCT for Prometheus-1 fuel irradiation tests. Figure 2 shows a sketch of the test train.

To accommodate Space Materials objectives, NRPCT planned to insert one capsule of UO_2 fuel pellets and one capsule of UN fuel pellets into the MICE-3B train. Because properties of UO_2 fuel are more sensitive to irradiation conditions and the MICE Tier 5 position experiences a lower flux than the Tier 4 position, UO_2 fuel pellets would be irradiated in the Tier 5 position to achieve the minimum fission rate feasible within the MICE platform. These rates are generally higher than those cited for Prometheus, but may fall within the upper range of possible values. It should be noted that although the Tier 5 position yields a more prototypical fission rate, the low fission rate and limited duration of the test would yield a final burnup below that intended for Prometheus. Conversely, UN is perceived to be less sensitive to burnup rate. Therefore, the UN fuel pellets would have been tested in MICE-3B Tier 4 position and would have achieved higher fission rates and a burnup close to that expected at end-of-life in a space reactor. Table 1 summarizes the MICE-3B test matrix of Space reactor fuel specimens.

Enclosure (1) of this document, "Test Details for the MICE-3B Space Material Capsules," describes the evaluations and preparations for testing both UO_2 pellets in Tier 5 and UN fuel pellets in Tier 4. UO_2 fuel pellets were manufactured using a process similar to reference (d) involving crushing and pressing gel microspheres. Details on the fabrication can be found in reference (e). The UN fuel pellets were manufactured at Los Alamos National Laboratory (LANL) using a carbothermic reduction process (reference (f)). Design work for fuel cans and irradiation capsules was completed, capsule parts were fabricated, and all components were ready for assembly. However, due to scheduling and changes in program direction, the irradiation test of UN fuel pellets was indefinitely deferred. For completeness, the proposed UN test was also included in Enclosure (1). Enclosure (2) and Enclosure (3) are "MICE-3B Post-Irradiation Examination Plan" and "Future Irradiation Tests of Space Fuel Materials in ATR," respectively. Enclosure (4) was prepared to assess compliance issues only for the capsule of UO_2 pellets, because testing of UN fuel pellets was dropped from the March 2006 insertion. Enclosure (5) is the MICE-3B capsule and fuel-can design report and includes the work performed on UN even though it was not going into test.

A design review was conducted for testing UO_2 pellets in Tier 5 and UN fuel pellets in Tier 4 of the MICE-3B test train to identify any remaining issues. Findings and resolutions of this review are documented in Reference (f). The design review team recognized that these tests were

very difficult and had a significant element of risk relative to the success in obtaining quality data. However, given the lack of data to support the space fuel effort, and possibility that future tests may contain similar or greater degree of difficulty and risk, the design review team concluded that the test program has value and should proceed.

II. Objectives of the MICE-3B Test Plan for Space Fuel Materials

Few data exist for UO_2 or UN within the notional design space for the Prometheus-1 reactor (low fission rate, high temperature, long duration). As such, basic testing is required to validate predictions (and in some cases determine) performance aspects of these fuels. Therefore, the MICE-3B test of UO_2 pellets was designed to provide data on gas release, unrestrained swelling, and restrained swelling at the upper range of fission rates expected for a space reactor. These data would be compared with model predictions and used to determine adequacy of a space reactor design basis relative to fission gas release and swelling of UO_2 fuel and to assess potential pellet-clad interactions.

A primary goal of an irradiation test for UN fuel was to assess performance issues currently associated with this fuel type such as gas release, swelling and transient performance. Information learned from this effort may have enabled use of UN fuel for future applications.

III. Test Description

A. Fuel Specimens and Fuel Cans

The MICE-3B test planned to have one capsule (Tier 5) that would contain a total of 18 UO_2 fuel pellets that were separated into three fuel mini-cans (Figures 3, 4a and 4b). Because gas release and swelling are influenced by grain size, one set of pellets were small grained ($\sim 10 \mu\text{m}$) while the other sets were large grained ($\sim 40 \mu\text{m}$). This would provide a relative comparison of the magnitude of the grain size effect for the relevant conditions. Both pellet types had identical fuel loadings of $0.5 \text{ g } ^{235}\text{U} / \text{cm}^3$. The pellet dimensions were 6.3 mm in diameter by 3 mm in length and were sintered to 96-98% theoretical density. The fabrication report for MICE-3B UO_2 fuel pellets is in Reference (e).

The MICE-3B test also originally planned a capsule (Tier 4) that would contain a total six UN fuel pellets, each pellet in an individual fuel can (Figures 5 and 6). The UN fuel pellets were made at LANL. These pellets had a uranium loading of $1.8 \text{ g } ^{235}\text{U} / \text{cm}^3$. Pellet dimensions were 6.5 mm in diameter by 3 mm in length and were sintered to a minimum 95% theoretical density. The fabrication report of MICE-3B LANL UN fuel pellets can be found in Reference (f).

Nickel alloy Haynes 230 was selected for the MICE-3B fuel can material. Mechanical properties of unirradiated Haynes 230 are readily available and the material was also a candidate for other components of the Prometheus-1 reactor. In addition, specimens of Haynes 230 were included for the JOYO irradiation test (Reference (h)). An objective of the planned MICE-3B irradiation test program was to optimize the accuracy of fuel sample temperature measurement during testing. The fuel sample temperatures during irradiation would be calculated based on the temperatures measured by thermocouples placed outside of the fuel cans. Weld distortion of fuel cans would increase the uncertainty of the fuel sample temperatures. To achieve the goal of accurate fuel sample temperature measurement, welding of Haynes 230 fuel cans was pursued to minimize weld distortion and meet ASME code weld qualification standards.

In the fuel-can design for UN pellets, the lid was attached to the base as an overlap joint (forming a wall of 2.54 mm (0.100 inch) thick) with a circumferential weld that required a minimum depth of penetration of 1.27 mm (0.050 inch) with minimum distortion. Welding was to be done in a high purity helium cover gas to provide a sealed, inert environment for testing as well as a condition for nondestructive helium leak-check evaluation of the weld. LANL successfully developed and optimized a laser beam weld process for this fuel-can design to encapsulate UN pellets. Their process met weld depth penetration and radial distortion requirements (38 μm / 0.0015 inch max) and passed helium leak-check testing. Future work would require mechanical tensile and bend testing of the weld material and further nondestructive testing to meet ASME weld code qualifications. This work was discontinued when UN was eliminated from the test and ultimately terminated when the Naval Reactors effort to deliver a space reactor ended. See figures 5 and 6 for sketch of the UN test assembly.

A similar but longer Haynes 230 fuel can was designed for encapsulation of UO_2 specimens. The fuel can for UO_2 pellets had the same diameter as that used for UN pellet encapsulation but used a V-groove joint for the weld. In order to meet the MICE-3B insertion schedule in March 2006, a manual gas tungsten arc weld process was developed at KAPL to seal weld these fuel cans. The distortion requirements for these UO_2 fuel cans were not as stringent as those for the UN fuel-can design. A diametral distortion of maximum 0.254 mm (0.010 inch) was measured on fuel cans with weld parameters optimized to meet the 1.27 mm (0.050 inch) weld depth-of-penetration requirement. Details on the welding process development efforts and testing for both fuel can types are discussed in Reference (i). See figures 3 and 4 for sketch of the UO_2 test assembly

B. Irradiation Condition

Three fuel cans of UO_2 pellets in the Tier 5 capsule were to be irradiated at nominally the same power rating of 150 W/cm³ (5×10^{12} fissions/cm³/sec) and a best-estimate peak central temperature (PCT) of 1600 K. The estimated burnup after 500 days of irradiation in the ATR was 2.4×10^{20} fissions/cm³, or ~1.1% fissions per initial metal atoms (FIMA). The common PCT and power level of fuel pellets were selected to maximize comparisons between the three fuel cans and minimize uncertainty of operating test conditions. All three fuel cans would use helium as fill gas to help reach desired temperatures and to preclude oxidation. The bottom-most fuel can (termed 'position-1') would contain six pellets of UO_2 (three large grain, three small grain). These pellets were to be arranged in two stacks, separated by grain sizes, one on the top of the other. Each stack would be loaded within a molybdenum constraining cladding. The cladding material would be TZM, a Molybdenum-Titanium-Zirconium alloy. The middle position fuel can (position-2) would contain six pellets with large grain UO_2 . The fuel can in the top-most position (position-3) would contain six pellets with small grain UO_2 . The fuel pellets in both position-2 and position-3 did not use constraining cladding. Test details and uncertainties for irradiating UO_2 fuel pellets in MICE-3B are discussed in Enclosure (1).

Test details and uncertainties for irradiating UN fuel pellets in MICE-3B are also discussed in Enclosure (1).

Fuel cans and capsule internal arrangements were designed to achieve MICE-3B temperature and fission rate requirements for UO_2 specimens in the Tier 5 capsule, and UN specimen requirements in the Tier 4 capsule. Thermal analyses of these capsule and fuel can designs are discussed in Enclosure (5).

C. Pre-Irradiation and Post-Irradiation Examination

Pre-irradiation examinations would be performed on all UO_2 fuel pellets. The minimum required work includes dimensional and weight measurements, visual and SEM examinations. Chemical analysis and ceramographic examination would be performed on unirradiated sister specimens from the same production lot.

The post-irradiation examinations (PIE) for MICE-3B UO_2 fuel pellets has been outlined, but not finalized at the close of the project. Post-irradiation measurements of MICE-3B fuel pellets were to be performed to determine:

- Fission gas release and fuel swelling
- Microstructural stability
- Mechanical stability during transient events
- Fission product distributions and chemistry
- Diffusion coefficients for fission gases
- Thermal conductivity after irradiation.

The current working plan for the post-irradiation examination (PIE) and disposal of the MICE-3B specimens and capsule materials was to ship the capsules directly from the ATR to the intended PIE facility. The facilities that would have potentially been used for PIE of the Space fuel specimens would have most likely involved Idaho National Laboratory, Pacific Northwest National Laboratory, and/or Oak Ridge National Laboratory. This is detailed in Enclosure (2).

A compliance assessment for NR facilities was performed to assure that samples could be brought to those facilities as a last resort option when irradiation were completed. This is detailed in Enclosure (4).

IV. Cost Estimate

Table 2 summarizes predicted major expenses for the irradiation test of MICE-3B Space fuel specimens. For one capsule of irradiated UO_2 pellets a rough order of magnitude PIE cost estimate of \$2,000,000 was obtained. The total cost for the Space Program MICE-3B test would be approximately \$3,000,000 (2005 dollars).

V. Materials Compliance Assessment and Evaluation

Enclosure (4) provides the details of a Compliance Assessment for the MICE-3B UO_2 Capsule. No significant concerns were identified.

VI. Future Tests in the MICE Facility

The MICE-3B irradiation test of space fuel materials was envisioned to be the first of several tests to be performed in a MICE irradiation test platform. Each test would have built on the experiences from previous testing and compensated for their shortcomings. While MICE-3B would have provided valuable information for the development of space nuclear fuel, it was not without room for improvement. Compromises included:

- Low fast-to-thermal (neutron) flux ratio
- High total flux (which affects power density)
- Small test volume (low number of samples)
- No online fission gas monitoring capabilities of individual specimens
- Relatively large temperature uncertainty
- No variety in processing technique of the samples tested
- No variety in pellet size of the samples tested.

Specific goals of future MICE tests for space reactor fuel would have depended on space reactor design developments such as clad material selection and the decision on whether or not to develop a moderated core. However, there are some general characteristics such as decreased temperature uncertainty that would be incorporated in future tests. This would have been approached in several ways including the use of melt wires, direct instrumentation of the fuel specimens, or operating the fuel at can temperatures (and vice versa) to minimize the differential between thermocouple and specimen temperatures. These were all considered but dismissed for MICE-3B due to the extended amount of time needed for development. Enclosure (3) provides a discussion of specific test objectives which would be considered in future tests including:

- Venting and monitoring of fission gas from bare fuel
- Comparing fuel samples made from different processes
- Studying clad-liner-fuel interactions and compatibility
- Studying effects of pellet size and geometry
- In-reactor transient testing.

VII. Review

Enclosures (1), (2), (3) and (5) of this document were reviewed by Michael Kania and David Miller, Enclosure (4) was reviewed by Martin Johnson. Technical review and concurrence has been received from the Managers of Materials Technology - Advanced Materials Technology and Materials Technology - Engineering Development and Testing at Bettis Atomic Power Laboratory; and the Manager of Materials Development Organization - Space Materials at Knolls Atomic Power Laboratory.

Very truly yours,

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References:

- (a) B-SE(RE)-0003, "Space Reactor Preconceptual Design Report," January 2006
- (b) B-SE(RE)-0002, "Fuel Type Recommendation for the Project Prometheus Space Nuclear Reactors", January 2006
- (c) WAPD-MT(EDT)F-478, "MICE Design Update," November 1998
- (d) Suryanarayana, S, et al., "Fabrication of UO_2 pellets by gel pelletization technique without addition of carbon as pore former," Journal of Nuclear Materials, Volume 230, 1996, pp140-147
- (e) MDO-CME-2120, "Fabrication of UO_2 Fuel Pellets for Space Program," January 2006
- (f) MDO-723-0045, "LANL UN Fuel Pellet Fabrication Process, for Information, " January 2006
- (g) MDO-720-1138, "Design Review of MICE 3-B Space Fuel Irradiation Test (Design Review #2008)," November 2005
- (h) MDO-723-0044 / B-MT(SRME)-52, "JOYO-1 Irradiation Test Campaign, For Information (U)," January 2006
- (i) MDO-723-0017, "Haynes 230 Mini-can Welding to Support Planned Irradiation Testing of Candidate Space Fuel Materials," January 2006

Table 1 Conceptual MICE-3B Space Reactor Fuel Test Matrix

Capsule Position	Capsule Contents	Instrument -ation	Nominal Fuel Dimensions	Fuel Types	Can Material	Nominal ²³⁵ U loading	Peal Central Temperature	Target Power Density
MICE-3B (M-2-103) Tier 5	18 UO ₂ pellets	TC, Sweep Gas	6.3 mm dia x 3 mm L	UO ₂ 95% min dense	Haynes 230	0.5 g/cc	1600K	150 W/cm ³
MICE-3B (M-2-103) Tier 4	6 UN pellets	TC, Sweep Gas	6.5 mm dia x 3 mm L	UN 95% min dense	Haynes 230	1.8 g/cc	1475K	500-750 W/cm ³

Table 2 Predicted Major Expense Items for Space Program MICE-3B Test

Expense Items	FY2005	FY2006	FY2007	FY2008	FY2009
Fabrication of capsule components ¹	\$90,000				
Test train assembly ²		\$70,000			
Irradiation Fee for one capsule ³		\$190,000	\$390,000	\$140,000	
Disassemble Train, prep one capsule, load cask and ship ⁴				\$59,000	
Cask rental (per cask) ⁵				\$102,000	
PIE estimate for one capsule of UO ₂ ⁶				\$1,500,000	\$500,000
Yearly total (not including fuel)	\$90,000	\$260,000	\$390,000	\$1,801,000	\$500,000
Project total (not including fuel)		\$3,041,000 (2005 dollars)			

Notes:

- Includes expenses for both the UO₂ and UN capsule components*
- Assuming 1 MY = 2080 hours = \$150,000*
- Assuming the test operates for the full 500 full power days and one capsule is tested. Irradiation Charges based on information and calculations from Reference - INL Letter: CCN 202381 dated August 31, 2005*
- Assumes a GE2000 cask is needed. If a GE 100 is used, cost would be approximately \$46,000.*
- Assumes a GE2000 is needed and must be brought in from its storage location in California.*
- Very rough order of magnitude estimate made based on other projects with similar level of work. Estimate includes disposal but not procurement of any equipment. A schedule had not been developed but was assumed to spread over a year and a half with the majority being done in the first year. A split in the charges was estimated. This estimate is only meant to be a guide and can change significantly.*

Figure 1 ATR Cross Sectional Diagram

Figure 2 Five tier MICE test train

Figure 3
Nominal Layout of Space Materials MICE-3B Tier 5 Capsule (with UO₂ pellets)

Figure 4a
Fuel Can ('mini-can') Design for Unconstrained UO₂ Pellet Testing

Figure 4b
Fuel Can ('mini-can') Design for Constrained UO₂ Pellet Testing

Figure 5
Nominal Layout of Space Materials MICE-3B Tier 4 Capsule (with UN pellets)

Figure 6
Fuel Can ('mini-can') Design for UN Pellet Testing

The Manager, SNR
The Manager, PNR

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Enclosure 1
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Test Details of MICE-3B Space Material Capsules (U)

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Reviewed by:
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1.0 Introduction

The Multiple Irradiation Capsule Experiment (MICE) facility in the Advanced Test Reactor (ATR) northeast-loop provides the capability for high temperature fuel development testing. There are two primary locations in the MICE facility, each of which can accommodate a test train of five capsules. These provide high temperature irradiation environments with temperature monitoring and helium-neon flow-gas for active temperature control, plus on-line fission gas detection and analysis. In a MICE test train, the Tier 5 capsule is located near the top of the reactor and the Tier 1 capsule is near the bottom. Both of these positions receive the lowest flux relative to the Tier 3 at mid-height. The center of Tier 3 capsule is positioned at mid-plane of the ATR core and receives the highest flux.

MICE-3B (NE-M-2-103) is scheduled to be inserted in March 2006 into the M-2 location. The planned irradiation period for MICE-3B is 500 effective full power days (EFPDs) (approximately 22 calendar months) and is projected to end in early 2008.

The Naval Reactors Prime Contractor Team (NRPCT) recommend the use of uranium dioxide (UO_2) fuel in the Project Prometheus Space Nuclear Reactor cores in July 2005, see Reference [1.a]. This recommendation was based on the conclusion that UO_2 fuel is more amenable to engineering solutions to mitigate risks in the available timeframe although significant materials performance uncertainties still remain with UO_2 . A specific plan would be pursued to resolve these uncertainties. Limited effort to characterize UN fuel was also recommended to resolve some of the key risks currently associated with this fuel type. The information learned from the recommended testing may enable the use of UN fuel for a future mission.

The NRPCT originally planned for the MICE-3B test train include one capsule of UN fuel pellets and one capsule with UO_2 fuel pellets. For both capsules, fuel pellets would be hermetically sealed in Haynes 230 fuel cans. These cans have multiple roles, including retention of the nitrogen overpressure generated by UN, retention of fission products to prevent unacceptably high radiological releases for the ATR facility, and prevention of oxidation or contamination of the specimens by impurities in the MICE swept gas system. Nickel-based alloy Haynes 230 was selected for this application because its high temperature strength and good weldability. Welding parameters and unirradiated mechanical properties were readily available. While irradiated properties could not be identified, this was a material of interest for the Prometheus-1 reactor and specimens of Haynes 230 were included for the JOYO irradiation test, see reference [1.b]. In addition, since nickel-based alloys generally show lower nitrogen absorption than iron-based alloys, Haynes 230 is better than stainless steels to be the fuel-can material for UN pellets. Table 1.1 summarizes the overall MICE-3B test matrix in the intended proposal.

Nuclear fuel in a space mission reactor would operate at high temperatures and low fission rates. Since the properties of UO_2 fuel are more sensitive to irradiation conditions, UO_2 fuel pellets would be tested at the low flux Tier 5 location in the MICE-3B test train which represents a near-prototypic fission rate. The UN fuel pellets in the higher flux Tier 4 capsule location of the test train would have high (5 to 10 times prototypical) fission rates but would achieve a burnup close to that expected in a space reactor. Fuel cans in capsules were individually designed, such that the uncertainty in temperature for an individual fuel pellet is estimated to be within 100 K.

UO₂ fuel pellets would be tested at the upper range of fission rates expected for a space reactor (but at a low burnup). The primary objectives for the UO₂ fuel are: 1) to provide performance data of different UO₂ grain sizes at near-prototypical fission rates, and 2) to assess the restrained force of UO₂ fuel pellets on a cladding. The UN fuel pellets would be tested to a burnup expected for a space reactor (but at a high fission rate). The primary goal of the MICE-3B test is to yield near-term information in support of reactor design efforts.

Post-irradiation measurements will be performed on MICE-3B fuel pellets to determine:

- Fission gas release and fuel swelling
- Microstructural stability
- Mechanical stability during transient events
- Fission product distributions and chemistry
- Diffusion coefficients for fission gases
- Thermal conductivity after irradiation.

Table 1.1
Test Specimens and Operating Conditions

Capsule Position	Capsule Contents	Instrumentation	Nominal fuel Dimensions	Fuel Types	Can Material	Nominal ²³⁵ U loading	Peak Central Temperature	Target Power Density
MICE-3B (M-2-103) Tier 5	18 UO ₂ pellets	TC, Sweep Gas	6.32 mm dia x 3 mm L	UO ₂ 95% min dense	Haynes 230	0.5 g/cc	1600K	150 W/cm ³
MICE-3B (M-2-103) Tier 4	6 UN pellets	TC, Sweep Gas	6.47 mm dia x 3 mm L	UN 95% min dense	Haynes 230	1.8 g/cc	1475K	500-750 W/cm ³

2.0 Uranium Dioxide UO₂ Fuel Testing

Uranium dioxide (UO₂) is a potential space reactor fuel for many of the same reasons that it is used in virtually all commercial reactors and as a component in the fuel of large fast-spectrum test reactors (e.g., FFTF) [2.a, 2.b]. UO₂ performance is better known than any other uranium-bearing compound, including UN or uranium carbides (UC/UC₂). However, the database to describe its behavior in the long time-at-temperature / low-fission-rate regime, applicable for a space mission, has not been fully investigated. UO₂ has well-behaved chemical, thermal, and mechanical properties, good solid fission product retention (particularly of rare-earth elements), reasonable uranium density, and an extensive unirradiated and irradiated property database. Because this database is at non-prototypic conditions relative to space needs, specific property correlations have been selected for preliminary space-mission design use; however, these correlations need to be confirmed through testing.

In addition to the extensive, albeit non-prototypical performance database, there is a significant infrastructure for handling and fabricating UO₂ materials. The main challenges with UO₂ relative to a long-term space mission are managing the fission gas release, fuel swelling, microstructural stability, and the mechanical integrity of UO₂ fuel pellets during temperature changes. The low thermal conductivity of UO₂ is well known and must be factored into the design calculations. Qualifying a UO₂ system from the point of view of understanding of its properties and

manufacturing issues is not expected to be limiting, although it must be integrated into a complete system within the known database and its limitations.

2.1 World Experience

Swelling

There is relatively little data available on the swelling of UO_2 in a long duration, low fission rate, unrestrained operating condition; however, a relationship can be extrapolated using the correlation from the FRAPCON-2 [2.c] fuel performance code. The correlation in FRAPCON-2 was developed by comparing and distilling the results from about 100 references. However, the correlation is weak in the long-time-at temperature and low-fission-rate regime of interest for space reactors and may not be applicable in this temperature range. Therefore, additional testing at space-reactor relevant conditions is warranted.

Fission Gas Release

There is wide variability in the reported fission gas release from UO_2 . As in the case of swelling, UO_2 fission gas release has received extensive study (see References [2.a, 2.b and 2.d]), and there are still uncertainties due to the complex inter-relationship of physical, fabrication related, and environmental variables. In particular, because the notional fission rates for the space reactor application are far below those typically used in commercial fuels, applicable gas release data within the fission rates of interest have not yet been located. Again, testing is warranted.

Thermal Conductivity

Thermal conductivity of UO_2 has been measured over a wide range of variables including temperature and burnup. A recent review of several thousand measurements over a range of burnup was published by C. Ronchi *et al.* in Reference [2.e]. In 1964, a correlation for the thermal conductivity of irradiated UO_2 was developed by J. Belle in Reference [2.f]. The Belle correlation was compared to the Ronchi data presentations and found to be in good agreement and has been chosen for space reactor calculations. Because of this, measurement of UO_2 thermal conductivity is not a high priority objective, but would be tested if budget and schedule allow.

2.2 Test Objectives

The MICE-3B test is designed to provide data on gas release, unrestrained swelling, and restrained swelling of UO_2 pellets at the upper range of fission rates expected for a space reactor. Two fuel cans each containing six specimens were intended to compare grain size effects on gas release and unrestrained swelling by testing both large-grain (40 μm) and small-grain (10 μm) fuel. One additional fuel can containing six specimens was intended to determine the restrained swelling of UO_2 and the force that it exerted on the restraining clad for two different grain size fuel pellets. The goals of these tests were to provide performance data that would have determined the adequacy of the space reactor design basis for fission gas release and swelling of UO_2 fuel, and to assess the potential pellet-clad interaction in the reactor design space. Primary post-irradiation measurements of these UO_2 fuel pellets were planned as follows:

Gas release and swelling for unconstrained UO_2 The MICE-3B test design correlation uses a single-stage gas release mechanism to predict gas release, which is a function of time at temperature and grain size. The use of UO_2 pellets with two different grain sizes in this test would permit a quasi-quantitative assessment with respect to the effect of grain size on fission gas release and fuel swelling. The main concern is the uncertainty in test temperature of the UO_2 pellets. The MICE-3B test design is such that temperature uncertainty is as low as reasonably achievable and specific impact is given in Section 2.4.

Restrained swelling and force assessment for UO_2 at low fission rate No data are available for constrained swelling of UO_2 at prototypical space reactor fission rates and temperatures. These data are important to assess the amount of strain/stress induced in a clad material in the event of pellet-clad mechanical interaction (PCMI). While the current reactor design basis does not allow for PCMI, it can occur from pellet rearrangement or cracking and an assessment method needs to be developed. In addition, if stresses were low, designs could permit PCMI for improved heat transfer similar to what is done in all commercial pressurized water reactor (PWR) designs.

Microstructural stability of UO_2 fuel at near-prototypical fission rates Restructuring of UO_2 fuel pellets is a well-known phenomenon and is associated with the migration of fission gas bubbles along a temperature gradient at elevated temperatures. Fuel pellets in commercial PWRs and the design basis for space reactors are to operate at a peak central temperature (PCT) below 1500°C to prevent restructuring, because restructuring results in substantial releases of trapped fission gases. In the commercial application of UO_2 pellets, the comparatively short life-time and relatively cold operating conditions (despite a relatively similar PCT, the circumference of commercial pellets are generally in direct contact with the Zircaloy cladding throughout most of their operational lives, which is at PWR coolant temperatures) aid in the retention of fission gases within the fuel pellets. In addition to restructuring, grain growth can lead to a different gas release behavior. The space reactor design will have a considerably higher average temperature and a much smaller gradient compared to PWR fuel which may lead to different microstructure changes. Severe differences are not expected but confirmation is needed.

Mechanical stability of UO_2 pellets during temperature changes and transient events Fuel pellet cracking is common in commercial reactors. However, the combination of gravity and clad restraint prevent the gross relocation of fuel within the rods. For notional space reactor designs involving free standing or annular pellets, the lack of these restraints could allow for lateral or axial movement within the pin, affecting reactor operation. In addition, the movement of small pieces of fuel between pellets and cladding is known to have a detrimental effect on the PCMI in commercial reactors, where this interaction produces locally high clad stresses. Assessments of pellet integrity would be made after irradiation via neutron radiography and visual examination. If still intact, the UO_2 pellets would be subjected to heating tests in an annealing furnace to determine the conditions under which cracking and relocation occur for near-prototypical fuel.

Fission product distributions and chemistries at near-prototypical fission rates Fission product redistribution is largely a thermal gradient driven phenomenon but is also dependent on the diffusion coefficients of the migrating species in the fuel. The thermal gradients in the space reactor fuel were anticipated to be lower than that observed in PWR fuel and the fission rates were also lower which may reduce the mobility of some species. Comparison of the radial

fission product distribution with commercially published data could be made to assess fission product mobility under near prototypical conditions.

Pending the availability of specimens after meeting the primary test objectives, several interesting tests are proposed for better understanding of the fuel. While thermal conductivity data exists for unirradiated UO_2 , it does not exist for irradiated UO_2 under conditions typical of space reactor designs. Microstructural and porosity differences could influence the measured value. Measurement of diffusion coefficients for fission gas can be determined by post irradiation thermal annealing. Annealing studies on the fuel can give insight into long range gas release and provide diffusion coefficients for released fission products.

2.3 Test description

Fuel Fabrication

UO_2 fuel pellets were manufactured using a process similar to reference (2.g) involving crushing and pressing gel microspheres. The pellet dimensions are 6.3 mm in diameter by 3 mm in length sintered to 96-98% theoretical density. Since gas release and swelling are influenced by grain size, two sets of one set of pellets will be made: one set of small grained ($\sim 10 \mu\text{m}$) and one set of large grained ($\sim 40 \mu\text{m}$). This would provide a relative comparison of the magnitude of the grain size effect for the relevant conditions. Both pellet types had identical fuel loadings of $0.5 \text{ g } ^{235}\text{U} / \text{cm}^3$. The fabrication report of these fuel pellets is in Reference [2.h].

Fuel Cans

The MICE-3B fuel pellets would be hermetically sealed in Haynes 230 fuel cans ('mini-cans') to prevent high radiological releases of fission products in the ATR facility and to prevent oxidation or contamination of the specimens by impurities in the MICE swept gas system.

To optimize the accuracy of fuel sample temperature measurement during testing, development of welding procedure was pursued to minimize weld distortion. The fuel can lid attached to the base as an overlap joint (to form a 2.54 mm thick wall), with a circumferential weld that requires a 0.050" minimum depth of penetration with less than 0.254 mm diametral distortion. The fuel can welding was to be performed in a high purity helium cover gas to provide an inert environment inside the can for testing as well as a condition for nondestructive helium-leak check evaluation of the weld. Los Alamos National Laboratory (LANL) developed a laser beam weld process for sealing the fuel cans containing the UN pellets. In order to meet the MICE-3B insertion schedule in March 2006, KAPL developed a manual gas tungsten arc weld (GTAW) process to seal fuel cans containing UO_2 pellets. The GTAW process was suitable for this application because the weld distortion tolerance for the UO_2 fuel-can design were not as stringent as that for the UN fuel-can design, see Reference [2.i]. The fuel can for UO_2 pellets had the same diameter and weld joint V-groove design as that used for UN pellet encapsulation. One significant difference in the UO_2 fuel-can design was the inclusion of a vent hole that would prevent gas-pressure inside the fuel can to build up during the welding process and eliminate the blowout concern.

Test Configuration

The Tier 5 capsule uses hafnium shroudings to adjust the axial variation of neutron flux reaching UO_2 pellets, so that irradiated UO_2 pellets in three fuel cans would nominally all be at a power rating of 150 W/cm^3 (5×10^{12} fissions/ cm^3/sec) and a best-estimate PCT of 1600 K. The estimated burnup after 500 days of irradiation is 2.4×10^{20} fissions/ cm^3 , or $\sim 1.1\%$ fissions per initial metal atom (FIMA). The common PCT and power level of fuel pellets were selected to maximize comparisons between the three fuel cans and minimize the uncertainty of operating conditions of the test. All three fuel cans use helium as fill gas, to help reach desired temperatures and to preclude oxidation. The bottom most fuel can (termed 'Position-1') contains six pellets of UO_2 (three large grain, three small grain). These pellets are arranged in two stacks, separated by grain sizes, one on the top of the other. Each stack is loaded within a molybdenum constraining cladding (Figure 2.1b). The cladding material is TZM, a Molybdenum-Titanium-Zirconium alloy. The middle position fuel can (Position-2) contains six pellets with large grain UO_2 . The fuel can in the top most position (Position-3) contains six pellets with small grain UO_2 . The fuel pellets in both Position-2 and Position-3 do not use constraining cladding (Figure 2.1a).

The above arrangement of fuel can positions was actually decided before completion of the capsule design. Without planning to use hafnium shroudings, the highest flux position (Position-1) was chosen to maximize the potential swelling for the constrained test to give the best chance of measurable change. The lowest flux Position-3 would test pellets with small grain material to minimize the gas release from these specimens. The fuel can would accommodate the pressure from increased potential gas release better in the third position due to lower fuel can operating temperatures.

Unconstrained UO_2 Fuel Experiment (Position-2 and Position-3 in Tier 5 capsule) The two fuel cans containing unconstrained UO_2 pellets, one with a $10 \mu\text{m}$ grain material and the other with $40 \mu\text{m}$ grain material were to be irradiated under the same nominal conditions. The internal design of the fuel can ($\sim 21.8 \text{ mm}$ (0.86 inch) diameter x 50.5 mm (1.99 inch) long) is shown in Figure 2.1a. The tungsten / 3%-rhenium spring exerts $\sim 0.7 \text{ lb-force}$ to the specimens resulting in an axial stress of $\sim 13 \text{ psi}$. The specimens are free to grow 0.304 mm (0.012 inches) diametrically to accommodate swelling and the predicted stresses suggest that axial growth should be accommodated. The radial clearance gives $\sim 2\text{X}$ margin to worst case estimated swelling based on fuel temperatures exceeding 1650 K PCT. The nominal fuel temperature profile with this design and configuration is provided in Figure 2.2. The desired fuel pellet PCT of 1600 K (2420°F) was to be achieved by adjusting the helium-neon ratio of the capsule sweep gas until capsule thermocouple data matched the thermal analysis results. By the current design basis, integrated gas release for the large grain specimens was predicted by the current design basis to be $\sim 40\%$ and gas release in the small grain specimen was predicted to be $\sim 70\%$. This level of separation would reveal difference in gas release, if any, between these two pellet groups for a temperature uncertainty of 100 K. Temperature uncertainty of fuel pellets will be discussed further in Section 2.4 below.

Constrained UO_2 Fuel Experiment (Position-1 in Tier 5 capsule) The fuel can at Position-1 was designed to allow pellet clad interaction between the UO_2 and a clad with known creep properties at the interface temperatures. The goal is to assess the force exerted by UO_2 onto the clad at low fission rates and compare test results to predictions by models using known creep rate data of UO_2 and the cladding. Molybdenum TZM was chosen as the tubular clad

material because available test data exists in the vicinity of the predicted interface temperatures; however, the predicted stresses were lower than the range of available test data by a factor of three. The fuel can design (Figure 2.1b) for this experiment was similar to that for unconstrained fuel experiments (Figure 2.1a) except that a 0.5 mm (0.02 inch) thick cladding surrounded the fuel. The fuel pellet-clad gap was designed to be zero at irradiation temperature to allow pellet-clad interaction to occur from the beginning of the test. This reduces uncertainty when comparing UO_2 swelling to the predicted creep. The temperature profile for the constrained fuel test is given in Figure 2.2.

Figure 2.1a
Fuel can ('mini-can') design for unconstrained UO_2 pellet testing

Figure 2.1b
Fuel can ('mini-can') design for constrained UO₂ pellet testing

Figure 2.2
MICE-3B irradiation temperature (°F) of UO₂ pellet stack

2.4 Test Methods and Uncertainties

Uncertainty in Test Temperature

General uncertainty in operating temperatures related to radial misalignment of the fuel pellets is similar in all three fuel cans and is shown in Figure 2.3 for the unconstrained case. The PCT of the specimen decreases by approximately 15°F (8.3K) if it is shifted 0.1905 mm (0.0075-inches) radial so that it contacts the thermal resistor; however, there would be little influence on the average pellet temperature.

Figure 2.3
Effect of Radial Misalignment on Temperature (°F) Profile

Fission gas release from the fuel would reduce the radial conductivity due to the low thermal conductivity of the Xe and Kr isotopes. However, radial conductivity is only a part of the thermal losses because of the TZM thermal resistor design (Figure 2.1b). Therefore, with a 40% gas release, the UO_2 fuel PCT would increase by only ~50 K. However, gas release and gas composition could vary from what was predicted and would affect heat transfer, therefore, contributing to the PCT uncertainty. To mitigate gas composition uncertainty, chemical analysis was to be performed on retrieved gas obtained during the gas puncture process on each fuel can. The predicted temperatures would be recalculated based on the gas composition changing from pure helium to the final gas mixture. As seen in Figure 2.4, using a one-step diffusional model, expected release for fuel at 1500 K is about 25% lower than that for fuel at 1600 K under nominal MICE-3B conditions. For swelling, the uncertainty is even greater; fuel at 1700 K is predicted to experience approximately 80% greater swelling than fuel at 1600 K (Figure 2.5).

Figure 2.4

Figure 2.5

A further complication is the fuel temperature variation within the pellet stacks with results from a computation of uncertainties associated with gamma heating, fission heating, thermal conductivities and emissivity. As seen in Figure 2.2, there is a calculated maximum 266 K (512°F) temperature gradient within the fuel stack with a 50 K difference in PCT between the hottest and coldest pellets. Because every pellet within a fuel can would be venting gas to a common plenum space, the fission gas release measurement would be based upon fuel operating throughout the range of these temperatures rather than at a unique temperature. This uncertainty has been estimated and methods to account for the range of expected gas release

due to pellet temperature gradients are within the space reactor design basis. See Enclosure 5 section 3E for further discussion on uncertainty in test design.

Uncertainty in Test Results

Unconstrained UO_2 Fuel Experiment (Position-2 and Position-3 in Tier 5) The primary data to be obtained on irradiated UO_2 specimens was gas volume and composition retrieved from the punctured fuel-can along with individual pellet density, dimensional data and irradiated microstructure. Together these data would be used to determine grain size effect on fission gas release and swelling of UO_2 under the operating condition of interest. Figure 2.6 shows the range of gas release data predicted by the current space reactor design basis for this test. The UO_2 fuel pellets were designed to be similar to reduce the uncertainties when comparing the gas release and swelling data from PIE. All fuel cans are designed for a PCT of 1600 K with a resulting average fuel temperature of ~ 1587 K. Gas release is predicted to be $\sim 40\%$ for the large grained fuel and $\sim 70\%$ for the small grained fuel. Due to the temperature uncertainties, variations up to ± 35 K may occur between the cans which would complicate the comparison between large- and fine-grained fuels. The post-irradiation assessment of the gas constituents is necessary to properly assess the temperatures of each test. Potential specimen misalignment and uncertainty in gamma heating, fission heating, thermal conductivities and emissivities may affect fuel temperatures in the same manner for pellets in both fuel cans. However, the effect has not been fully assessed. Due to the effect of off-centering on temperature profiles, post-test neutron radiography would be necessary to determine end of test positioning within the can.

The swelling data (obtained from density and dimensional measurements) would also be obtained on the unconstrained specimens. A comparison of the swelling data would be made for each fuel pellet from each fuel can. Assessments of position variability of each fuel pellet in the can would be used to gauge measurement sensitivity. The largest calculated axial temperature drop in a 6-pellet stack is 160 K. Irradiated fuel pellets in the same stack may show a measurable swelling difference (Figure 2.5). Observed differences in swelling among fuel specimens of a given grain size will be compared with the correlation based on temperatures.

Figure 2.6
Fission-Gas Fractional Release Prediction

Constrained UO₂ Fuel Experiment (Position-1 in Tier 5) The primary data from these specimens were cladding diameter changes for each grain size of the restrained fuel. The force exerted from the fuel on the cladding could be determined by comparing potential differences with the unrestrained conditions. The estimated pressure exerted by the UO₂ from this test would aid in clad thickness evaluations for potential fuel designs. Test sensitivity was set at ~ 150 psi by the 0.508 mm (0.020 inch) thick molybdenum TZM clad. This was based on the pressures predicted by a creep evaluation of UO₂ unconstrained swelling for the best estimate average fuel temperature. Clad stresses can be determined using a hoop stress calculation:

$$\sigma_{Mo} = r * P_{UO_2} / t$$

where; σ_{Mo} = Stress in the Molybdenum TZM used for the creep evaluation
 P_{UO_2} = Pressure exerted by the UO₂ onto the clad
 r = Tube radius
 t = Tube wall thickness.

The measured deformation in TZM was to be used to determine cladding stresses by comparing the deformation to predicted creep for the actual test time and temperature. Temperature uncertainty affects the magnitude of predicted stresses, and comparison to unrestrained data would be needed. The unrestrained volumetric swelling of UO₂ ($\Delta V/V$) was predicted to be between 2% and 4 % over the temperature range of 1450 K to 1550 K. Assuming ~ 1/3 of the deformation is radial, a 63.5 μ m (0.0025 inches or 1%) diametral displacement would be predicted. Since the TZM would swell ~ 0.2% at the test conditions due to neutron irradiation, there is a minimum deformation that was expected to be measured that yields no data about the UO₂ pressure.

Tubular TZM claddings of MICE-3B UO₂ pellets are 6.35 mm (0.25 inches) in inner-diameter. It would be expected for them swell ~2% or ~ 12.7 μ m (0.0005 inches) in diameter due to irradiation, and this defines the minimum measurable deformation expected for this test. The equivalent stress corresponding to this deformation is ~100 psi (Figure 2.7), and this stress value depends on the UO₂ temperature and the TZM clad temperature. The calculations assumed a TZM clad temperature of 1436 K, based on thermal modeling. Clad temperatures were assumed to scale proportionally for assessing temperature uncertainty effects (a 5% uncertainty in PCT corresponds to 5% in clad temperature as opposed to applying a 100 K uncertainty). If the diametral deformation in the TZM is 63.5 μ m (0.0025 inches), which matches the unconstrained UO₂ swelling, the pressure in the UO₂ must be assumed to be 200 psi or greater for the wall thickness chosen for the test, depending on the test temperature (Figure 2.5). Measured deformation between 12.7 μ m and 63.5 μ m (0.0005 inches and 0.0025 inches) would give a measure of the pressure between 100 psi and 250 psi. This evaluation would require best estimate temperature predictions for the gas temperature and requires min-can gas analysis after irradiation.

Figure 2.7

Predicted pressure from swelling UO_2 as a function of TZM Deformation (inches)

2.5 Characterization of Unirradiated and Irradiated Specimens

Key pre-irradiated fuel pellet attributes are size, shape, porosity, grain size, impurity content, uranium content and enrichment. Table 2.1 summarizes attributes and techniques used for characterization. Prior to capsule loading, all specimens selected for irradiation would be measured for dimensions and weights, photographed, and SEM examined. Thermal conductivity can be determined on unirradiated "sister" specimens later, together with irradiated specimens during the post-irradiation examination.

Table 2.1 UO_2 Pellet Pre-Irradiation Characterization	
Attribute	Techniques/Remarks
Density	Determined by dimensional and weight measurements
Dimensions	Determined by length and diameter measurements at several points
Uranium assay	Determined by wet chemistry
Enrichment	Determined by mass spectrometry
Grain size and structure	Determined from optical ceramography, SEM and image analysis
Distribution of porosity	Determined from optical ceramography, SEM and image analysis
Impurities	Determined from wet chemistry, neutron activation analysis and WDS analysis
Thermal conductivity [†]	Determined using laser flash at temperatures from 300K to 1600K in 100K increments

[†] To be preformed on sister samples during PIE pending resource availability.

After completion of the planned irradiation in early 2008, MICE-3B capsules were to be removed from the train and shipped for PIE. If the fuel did not reach the planned exposure, or could not be reinserted into subsequent tests and was recoverable, PIE was still planned to be performed at the lower burnup. As of the termination of the NR effort to deliver a space reactor for the Prometheus program, a formal plan for PIE of MICE fuel had not been fully developed; however, a suite of standard characterization exams was envisioned to be performed as described in Table 2.2.

Table 2.2	
UO₂ Pellet Post-Irradiation Characterization	
Attribute	Techniques/Remarks
After test configuration	Neutron radiography of capsule
Density	Determined by dimensional and weight measurements (or pycnometry)
Dimensions (Swelling)	Determined by length and diameter measurement at several points
Grain size and structure	Determined from optical ceramography, SEM and image analysis
Thermal conductivity [†]	Determined using laser flash at temperatures from 300K to 1600K in 100K increments
Fission gas release	Gas puncture plus chemical analysis
Fission product distribution	Electron microscopy plus EDS/WDS analysis
Distribution of porosity	Determined from optical ceramography, SEM, computed tomography and image analysis
Transient Behavior	Determined by rapid heating of pellet in a furnace
Annealing behavior	Determined by long term heating of pellet in a furnace capable of monitoring/measuring fission product release.

[†] Not a primary PIE goal. This test was to be performed pending resource availability. Upper test temperature may change.

To determine an optimal post irradiation examination (PIE) location, NRPCT engineers were to visit and evaluate each major American irradiated materials testing facility:

- Idaho National Laboratory's (INL) Material and Fuel Complex
- Oak Ridge National Laboratory (ORNL)
- Pacific Northwest National Laboratory (PNNL)
- Los Alamos National Laboratory (LANL)
- Knolls Atomic Power Laboratory's (KAPL) Radioactive Materials Lab (RML).

There were several criteria which must be met for the facility to be selected. These criteria include: ability to accept low enriched fuel, ability to perform the majority of desired testing (Table 2.2), ability to accept the cask used for shipment (i.e. the GE 2000), willingness to meet the tight schedule, and ability to dispose of test components and fuel. Based on the evaluation presented in Enclosure 2, INL is the favored facility with ORNL or PNNL in second; however, more research would need to be done before a conclusion can be made.

3.0 Uranium Nitride (UN) Fuel Testing

Uranium nitride has a number of attractive characteristics compared to UO_2 including:

- High uranium atom density
 - 40% higher than UO_2
 - 4% higher than UC/UC₂
- Lower fission gas release than UO_2
- High unirradiated thermal conductivity
 - 10 times higher than UO_2
 - 5-6 times higher than UC/UC₂
 - no information on irradiated thermal conductivities has been located
- High melting / decomposition temperature
 - Nitrogen overpressure is required above ~1200 K to avoid decomposition

In addition, some investigators have reported a lower release rate for certain solid fission products compared to UO_2 (see Reference [3.a]). In general, this lower release is most likely related to the inherently lower diffusion coefficients for fission products or the lower thermal gradient in UN. Because of these characteristics, the United States has considered UN for use in various space applications, and Russian, French, and Japanese programs are studying nitride-based fuels for earth-based power generation reactors. Despite these benefits, the difference in chemistry between oxide and nitride fuel systems creates other fission product compounds of concern within nitride fuels that are not present in oxide systems. For example, formation of URu_3 is predicted based upon thermodynamic models of the UN system and has been observed in irradiation tests. This byproduct, which has been shown to be a detriment to several candidate liner and cladding materials, is not expected to form in a UO_2 -based system. The challenge associated with UN is the level of uncertainty in irradiation performance due to the relatively small test database. This lack of data produces an increased uncertainty in properties, irradiation behavior, materials compatibility, sensitivity to manufacturing variables, etc. For example, significant differences in gas release have been noted between fuel batches without an identified cause. The production use of UN would require a significant test program to reduce the level of uncertainty associated with its use and to qualify the material and fuel element configuration. A suitable UN test program could start by performing PIE on any remaining specimens from the SP-100 irradiation test program archived at Hanford [3.b], or on the Japanese or Russian archived UN fuel, if available. Subsequent testing would follow with the fabrication and irradiation testing of new specimens. The first of such irradiations was to be in the ATR – MICE-3B train. Subsequent irradiation testing may have included other reactors' platforms.

3.1 World Experience

Swelling

Swelling correlations for UN have been previously reported by Ross and El-Genk at LANL in Reference [3.c]. This same correlation is also used by Russian programs as reported by Rogozkin, Stepenova, and Proshkin in Reference [3.d]. Although Russian programs have significant data on UN, and have developed a UN gas release correlation from these data, they did not use their own swelling correlation. The UN swelling correlation used by the Russian programs is presented in the design basis tables in Reference [3.e]. This swelling correlation was developed from Pratt and Whitney and Oak Ridge data in support of the SNAP-50 program. This correlation does not include more recent SP-100 data. Calculated fuel centerline temperatures in the data used for the correlation ranged between 1300 and 1500 K and power densities varied between 270 and 1410 W/cm³. The centerline temperature and power densities in the originally envisioned Prometheus-1 space mission design were ~ 1400–1600 K and 25 to 60 W/cm³, respectively.

Figure 3.1 and Figure 3.2 show the Ross swelling correlation plotted with SP-100 data. To plot the correlation and the SP-100 data simultaneously, a correction factor between volume average fuel temperature and the specimen clad outer diameter temperature needed to be calculated. A simple thermal calculation based on power density, fuel, liner and clad conductivities and gas gaps was performed and that resulted in a delta of ~270 K between average temperature and clad surface temperature. The Ross correlation was plotted using this correction factor, along with lines at ± 100 K from the correlation. The SP-100 data is scattered below the -100 K deviation of the correlation. This may indicate that the SP-100 material swells less than other UN material; however, there is also a concern about the accuracy of the actual test temperature. For instance, element bowing [3.f] may have caused the pellets to contact the irradiation capsule and therefore, operate at a lower temperature. Such a shift could bring the SP-100 data in-line with the prediction. To resolve the discrepancies, LANL was to reassess the test pin temperatures. If the temperatures are correct or with little change, the LANL SP-100 data indicates both a lower swelling and a lower gas release than other UN data sets. This seemingly inconsistent result needs to be understood. A thorough review of all of UN data was underway, however, this compilation of data did not bound all ranges of interest and, therefore, additional irradiation testing would be needed to provide new data.

Figure 3.1: UN swelling shown as a function of irradiation temperature. The lines show a predictive correlation and the data points are from the SP-100 program. A thermal calculation for a test pin with gas gaps was used to normalize the data to the prediction; however, there is great uncertainty in the test temperature, so the prediction may be most accurate.

Figure 3.2: UN swelling shown as a function of burn-up. The lines show a predictive correlation and the data points are from the SP-100 program. A thermal calculation for a test pin with gas gaps was used to normalize the data to the prediction; however, there is great uncertainty in the test temperature, so the prediction may be most accurate.

Fission Gas Release

Initial evaluations indicate that fission gas retention in the SP-100 UN is about 50% better than previous UN materials. If the SP-100 program UN fuel actually had superior fission gas retention, the mechanism or reason for this result is unknown. Factors cited for this superior performance include density, grain size, and stoichiometry. However, fuel manufactured during the SNAP-50 program appears to have had similar properties but less fission gas retention [3.g].

It must be emphasized that there are uncertainties associated with the SP-100 test conditions. In any case, it is important to understand the basis for the release properties, otherwise it is difficult to assess how changes in feedstocks or the manufacturing steps required to produce pellets with different geometries than those used in the SP-100 program will affect the gas release rate.

There is no correlation for fission gas release from the SP-100 UN. While several fission gas release equations have been developed by Rogozkin, Stepennova, and Proshkin (Russia) in Reference [3.a], and Weinstein, Kirchgessner, and Tambling (NASA) In Reference [3.h], the Storms equation (Los Alamos) in Reference [3.i] was chosen because the report contained a clearly defined set of data (consisting of 81 data points) with which the predictions were compared. The Russian correlation presented no data while the NASA correlation was based on a relatively small data set (two capsules), and was correlated to release rate (as opposed to total gas release) [3.h]. Figure 3.3 compares the Storms correlations with the SP-100 data. In this figure, the SP-100 specimen test temperature (calculated clad outer diameter temperature) was calculated from the volume average fuel temperature using the same factor given in the swelling section. It should be noted, that while the Storms correlation was judged to be the best available, it still contains significant error because the specimen temperatures were not well known and gas release is approximately proportional to T^3 .

Figure 3.3: Actual UN fission gas release data plotted against the prediction. The line shows the predictive correlation and the data points are from the SP-100 program. A thermal calculation for a test pin with gas gaps was used to normalize the data to the prediction; however, there is great uncertainty in the test temperature, so the prediction may be most accurate.

The Storms equation predicts higher gas release than was reported for the SP-100 UN but much lower levels than the Russian correlation. For example, at 6% burnup and 1400 K the Storms correlation predicts a gas release of about 6% while the Russian correlation predicts about 18%. Because the Russian paper presented no data, it can only be assumed that the Russian UN releases more gas than the American UN. The cause of this disparity warrants further study because the fundamental explanation for why certain UN materials release more gas than others is not known. UN fuels tested in terrestrial reactors commonly contain porosity for the intentional promotion of fission gas release, and this may be the source of higher gas releases observed in the Russian program. In addition, the combination of low swelling and low fission gas release reported for the SP-100 material, although not necessarily physically

unreasonable, is unexpected and inconsistent with every previous study of fission gas release from UN [3.i, 3.j, 3.k, 3.l].

It is important to note that the fission gas release properties associated with non-SP-100 UN are lower than those of UO_2 (by a factor of 8-10), so UN would still be of interest even if the reported properties of SP-100 UN cannot be recaptured. Ultimately a design margin must be established for the UN fuel based upon additional irradiation tests.

An equation for the thermal conductivity of unirradiated UN was developed by El-Genk, Ross, and Matthews [3.m] from six different data sources. A total of 107 data points for fuel temperatures ranging between 10 K and 1923 K were used. All of the data fall within $\pm 10\%$ of the correlation. However, no irradiated thermal conductivity data for UN has been reported. In the absence of data, it is assumed that the conductivity is degraded by the volume fraction of low conductivity gaseous fission products retained within UN with no solid fission product contribution. Conservative irradiated conductivity values can then be approximated using the Downar model [3.l] for calculating thermal conductivity of cermet by applying the fission product bubble volume to the "ceramic" terms and unirradiated UN properties to the "matrix" terms. Actual data would be obtained on the irradiated UN fuel specimens in MICE-3B and potentially measurements on archived SP-100, Japanese, and Russian specimens, as well as future irradiation testing.

While the commercial process variation (crushing and pressing gel microspheres) was pursued by LANL, it is not the only fuel process available. BWXT was expected to produce UN by an alternate method based on an internal gelation process. Pellets from this process could have been tested in subsequent irradiation test programs.

3.2 Test Objectives

There is little data that exist for either UN or UO_2 within the notional design space for a space mission (low fission rate, high temperature, long duration). As such, basic testing is required to validate, and in some cases determine, the performance aspects of these fuels. The primary goal of the MICE-3B UN test was to assess some of the key risks currently associated with this fuel type. The information learned from this effort may have enabled the use of UN fuel for a future mission.

Post-irradiation measurements of MICE-3B UN fuel pellets would be the same as those planned for the UO_2 testing as follows:

- Fission gas release and fuel swelling
- Microstructural stability
- Mechanical stability during transient events
- Fission product distributions and chemistry
- Diffusion coefficients for fission gases
- Thermal conductivity after irradiation.

Fission gas release and fuel swelling are a strong function of temperature, with some estimates suggesting that these properties are proportional to T^3 for UN fuels [3.c]. Therefore, design-quality information on these variables can only be achieved in tests with low temperature uncertainties. The analyses supporting the design of the test capsules showed that both of

these conditions could not be simultaneously met in MICE-3B, i.e. high temperature accuracies could only be achieved in unrestrained (uncanned) fuel. Due to the requirement that the specimens must be canned, this has limited the proposed MICE-3B test to one with relatively high uncertainties in temperature (+120K, -65K). See Enclosure 5, section D for details.

Because one option for the Prometheus-1 reactor was to use unconstrained fuel pellets, the unconstrained swelling needs to be known with a high degree of accuracy. However, since swelling is a strong function of temperature and previous experiments did not have accurate temperature measurements, unconstrained swelling performance is a major concern. Like the fission gas release rate, the LANL SP-100 fuel showed less unconstrained swelling than other UN fuels and was below published predictions. However, like the fission gas release, the swelling data is highly suspect due to uncertainties in the test operating conditions. Because of the discrepancy in data and the lack of data at the space mission conditions, additional irradiation testing is needed. The unconstrained swelling data would be gathered from the same samples as the gas release rate via dimensional measurements and volume measurements (by pycnometry).

MICE-3B test results would provide information on microstructural changes that occur in UN during irradiation. For UO_2 fuels, microstructural changes (grain growth or subdivision, columnar grain formation, fission gas bubble interlinkage or migration) exert a strong influence on almost all of the key properties of interest. For UN, the available data seem to indicate that relatively few microstructural changes occur. In general, UN specimens do not exhibit the same sort of grain growth, gas bubble formation, or pore migration that is associated with irradiation testing of UO_2 . To some degree, this data is currently available on near-prototypical fuels from the SP-100 tests. However, given the uncertainties on the irradiation conditions of these specimens, and given the need to perform a more detailed characterization, more work on irradiated UN specimens is necessary.

PIE of MICE-3B specimens would provide data for UN fuel irradiated under transient temperature conditions. Currently, there is no data available for the transient performance of this fuel. UO_2 pellets are known to fracture under even the relatively mild temperature ramps associated with commercial nuclear reactors due to retained expanding gas. For UN, an even higher degree of gas retention has been cited for this material than for UO_2 , suggesting that transient performance may also be a concern. However, the amount of fission gas bubbles in UN and the size distribution of these bubbles differ greatly from UO_2 . Concurrent with the MICE-3B test, efforts would be pursued to perform transient testing on available UN fuels irradiated in other programs. However, these fuels generally are not similar in fabrication or irradiation test conditions to that which would be required for space mission. As a result, there is a strong need to perform transient testing on near-prototypical fuels irradiated in MICE-3B. As currently envisioned, transient performance would be tested out-of-reactor by subjecting irradiated fuel pellets to rapid temperature changes in an annealing furnace.

Irradiated MICE-3B specimens would provide additional data on fission product distributions and chemistries within and exterior to UN. In general, UN appears to release fewer fission products than UO_2 , presumably due to an intrinsically lower diffusion rate for fission products in UN. There are differences in the expected chemistries between irradiated UN and UO_2 and certain compounds that are effectively tied up in an oxide system may not be immobilized as effectively in a nitride-based system. UN reacts with fission products, especially Ru, Rh, Mo and Tc. The formation of a uranium-fission product liquid has been noted in irradiation tests of UN [3.n, 3.o,

3.p]. In general, loss of nitrogen precedes the formation of liquid phases [3.k]. Redistribution of a uranium- and ruthenium-rich compound was observed during post-irradiation examination of some SP-100 specimens. The formation of uranium ruthenides is uncommon in UO_2 fuels due to the relatively high stability of the UO_2 molecule. Based upon thermochemical considerations, it appears that ruthenium will react with UN to form uranium ruthenides, most notably URu_3 . Out-of-pile testing on this compound, and to some extent the SP-100 data, suggest that URu_3 can be corrosive and is a potent grain boundary embrittler. The planned duration (>15 years) of the Space mission suggests that an accurate assessment of fission product migration and chemistry is critical.

Irradiated MICE-3B specimens would be used to assess diffusion coefficients. As noted previously, these specimens would be hermetically sealed into capsules, so fission gas monitoring cannot be used to measure release. In addition, given the temperature uncertainties inherent in the proposed test, gas puncture measurements are only anticipated to be useful in a somewhat qualitative sense. For the purposes of this objective, the capability to perform long term annealing tests after irradiation along with the monitoring of fission product releases was to be assessed. This data could be useful for determining diffusion coefficients.

Irradiated MICE-3B specimens would be used to determine the irradiated thermal conductivity of UN. The unirradiated thermal conductivity of UN has been measured and is about 10 times that of UO_2 ; however, there are no irradiated thermal conductivity measurements. Results of a theoretical analysis suggest that thermal conductivity of irradiated UN fuel is still 6 to 9 times that of irradiated UO_2 at temperatures greater than 1300 K [3.q]. The same samples of fuel which have been proposed for fission gas release and swelling measurements would be used for thermal conductivity testing.

3.3 Test Description

Fuel Specimens Fabrication

The UN fuel pellets have been made by LANL per Reference [3.r]. The fabrication report can be found in Reference [3.s]. The pellet dimensions are 6.47 mm in diameter by 3 mm in length. The grain size was specified at 50 μm , but only grain sizes of $\sim 25 \mu\text{m}$ could be achieved. Grain size is generally accepted as an influence to fuel properties such as gas release in UO_2 fuel and would likely influence properties in UN as well.

However, because there was only room to test six specimens in the MICE-3B UN test capsule, it was desired to limit the variables. Moreover, LANL did not have the capability to grow larger grains. To study the effect of UN grain sizes, additional time and money would be required to bring the high temperature annealing furnaces online. Due to limitations in test space and resources as well as test schedule, the effect of UN grain size could not be studied in this planned MICE-3B test. (Future tests need to explore this issue.) For the MICE-3B test, in order to control over the heat transfer between the fuel and the structure (i.e., the thermal resistance would not change as the pellet swelled and released fission gas), heat was to be conducted out of the fuel pellet axially instead of radially. A pellet height of 3 mm was chosen to mimic the radial temperature gradient which would be developed in a 6.47 mm-diameter pellet under normal operation.

UN fuel pellets (13.3 w/o enriched) were sintered to ~96% theoretical density and had a fuel loading of $1.8 \text{ g U}^{235} / \text{cm}^3$ in each pellet. Although there was a difference in the enrichment between the MICE fuel and the envisioned space reactor fuel, which is 93 w/o enriched ($11.7 \text{ g U}^{235} / \text{cm}^3$), the ultimate burnup as measured in fissions/cc would be similar. With respect to test goals, there would be relatively minor differences in fission product yields due to the increased number of fissions from U^{238} or the transuranics beginning with Pu^{239} in the MICE-3B. Despite the lower enrichment, the vast majority of the fissions would be from U^{235} . At the expected burnup of 2% to 4% fissions per initial metal atom (FIMA), little transmutation would occur. For the amount that does occur, the fission contribution from Pu would remain small due to the equilibrium level of Pu at the high flux level of ATR. Due to the thermal energy spectrum of the ATR, the MICE-3B fuel was expected to experience a power distribution (and therefore a fission product distribution) that would be peaked towards the outer diameter of the element. For commercial fuel pellets, this is known as the rim effect and does affect the fission product gas retention [3.t] and fuel swelling behavior (potentially higher rates than normally expected for an average %FIMA rate). The fission rate profile in the fast reactor spectrum of the envisioned space reactor is expected to be much flatter across the pellet. For the purpose of this test, the differences are not expected to significantly affect the interpretation of the test results. UN fuel specimens of 93 w/o enrichment would be included in future space-mission irradiation tests.

Fuel Cans

An objective of the planned MICE-3B irradiation test program was to optimize the accuracy of fuel sample temperatures during testing. The temperature of a fuel sample was to be calculated based on the temperature measured by a thermocouple outside the fuel can and the as-designed fuel sample position inside the fuel can. The weld distortion of a fuel can affects the uncertainty of the fuel sample temperature. Haynes 230 was chosen over conventional alloys due to the high-temperature strength required for this application. A welding process development of Haynes 230 fuel cans was pursued to minimize weld distortion. The welding process needed to meet ASME code weld qualification standards. Welding was to be performed in a high purity helium cover gas to provide an inert environment for testing as well as a condition for nondestructive helium-leak check evaluation of the weld. LANL successfully optimized a laser beam weld process for the MICE-3B UN fuel-can design which met weld depth penetration and maximum distortion requirements of $38 \text{ } \mu\text{m}$ (0.0015 inch), and passed helium-leak check testing. However, due to scheduling and changes in program direction, the irradiation test of UN fuel pellets was indefinitely deferred and the laser weld was not fully qualified. Future work would require mechanical tensile and bend testing of the weld and further nondestructive testing to meet ASME weld code qualifications. Details on the laser beam welding development efforts and testing are discussed in Reference [3.u].

Test Conditions

Six pellets of UN fuel were planned for the Tier 4 capsule. UN fuel is sensitive to oxygen and moisture and requires a constant nitrogen overpressure. Because a previous evaluation of fission gas releases from bare UN pellets showed the potential to exceed ATR limits, the fuel would be individually canned or sealed in the capsule. At 1600 K, the uranium nitride fuel must be kept under a nitrogen overpressure of approximately 8×10^{-8} torr to 8 torr to prevent decomposition to uranium metal or U_2N_3 respectively. To accomplish this, the pellet was to be hermetically sealed in a Haynes 230 can. To mitigate the interactions between the fuel and the can (or capsule) and to establish the correct heat flux reaching the can, a tungsten pedestal

would be placed between the fuel and the can (Figure 3.4). A failure in the can would be detected by a spike in the gross activity monitor which monitors the helium/neon sweep gas. Loss of the nitrogen overpressure would result in a decomposition of the fuel. Because the nitrogen partial pressure over UN is fairly low at the test temperature of interest and because experiments at LANL have shown no decomposition while a UN sample cools from sintering temperature to room temperature under vacuum, the rate of decomposition is expected to be slow, and subsequent reactor shutdown can be accomplished before significant decomposition can occur. However, even if fuel decomposition should occur following a loss of can integrity, the tungsten pedestal would temporarily contain the decomposed fuel. While compatibility data is limited for interactions between tungsten and liquid uranium, the phase diagram indicates the solubility of tungsten in uranium near the melting point of uranium is about one atom percent in both liquid uranium and solid uranium. The decomposed fuel is not expected to compromise the integrity of the tungsten pedestal.

Figure 3.4
Schematic of the UN fuel can.

The UN fuel would be tested at approximately the same temperatures but at different burnups (power levels) as outlined in Table 3.1. The proposed conditions do not completely encompass the design envelope for a space mission but represent a compromise between the information that is required and that which can be attained within volume and design constraints of the MICE test platform.

Table 3.1

Desired operating conditions for each UN pellet, time is 500 Effective Full Power Days

Tier / Pellet	Power W/cm ³	Peak Central Temp. K	Burnup (f/cc x 10 ²⁰)	Burnup rate (f/cc*s x 10 ¹³)
4 / 1	500	1475	6.7	1.5
4 / 2	500	1475	6.7	1.5
4 / 3	500	1475	6.7	1.5
4 / 4	750	1475	10.1	2.3
4 / 5	750	1475	10.1	2.3
4 / 6	750	1475	10.1	2.3

In the intended test, the fuel test temperature would be set initially by the length of the conduction path (i.e. the height of the tungsten pedestal). The fuel specimens would experience up to a 5% temperature (+/- 50 K) swing due to a changing flux profile during reactor operation. This can be compensated for during test by changing the flowing sweep gas helium to neon ratio used.

Heat would be conducted through the pellet axially. At the expected power level, the fuel temperature can be controlled at a PCT of 1500 K while the can material remains below 1170 K. At 1170 K, the analysis indicates that Haynes 230 would have adequate strength and be thermally stable with minimal interactions with the surrounding environment.

UN fuel requires a nitrogen overpressure to prevent decomposition to uranium metal, and UN will react with oxygen or water to form urania, even if these contaminants are in a sub-ppm level. The sweep gas monitoring system currently in place at the ATR does not use a nitrogen sweep gas, nor is the impurity content of the sweep gas below the reaction threshold. Therefore, the sweep gas is not allowed to directly contact UN fuel to collect fission gas for the release measurement. However, the sweep gas system would be able to determine if the capsule has leaked (detection of water vapor) or if the mini-can has failed (detection of fission gas). Three thermocouples would be used to measure the temperature just outside of three separate fuel mini-cans.

3.4 Characterization of Unirradiated and Irradiated Specimens

The key pre-irradiated fuel pellet attributes are size, shape, porosity, grain size, uranium content and enrichment. The impurity content of the pellet is also important. Table 3.2 summarizes attributes and techniques used for characterization. Prior to capsule loading, all specimens selected for irradiation were to be measured for dimensions and weights, photographed, and SEM examined. The unirradiated thermal conductivity was to be determined on unirradiated "sister" specimens during the post-irradiation examination of the irradiated specimens.

Table 3.2 UN Pellet Pre-Irradiation Characterization	
Attribute	Techniques/Remarks
Density	Determined by dimensional and weight measurements
Dimensions	Determined by length and diameter measurements at several points
Uranium assay	Determined by wet chemistry
Enrichment	Determined by mass spectrometry
Grain size and structure	Determined from optical ceramography, SEM and image analysis
Distribution of porosity	Determined from optical ceramography, SEM and image analysis
Impurities	Determined from wet chemistry, gamma spectroscopy and WDS analysis
Thermal conductivity [†]	Determined using laser flash at temperatures from 300K to 1600K in 100K increments

[†] To be preformed on sister samples during PIE.

After completion of the planned irradiation in early 2008, MICE-3B capsules would be removed from the train and shipped for PIE. If the test should need to be stopped pre-maturely and the capsule is recoverable, PIE would be performed at the lower burnup. A suite of standard characterization exams would then be performed as described in Table 3.3.

Table 3.3 UN Pellet Post-Irradiation Characterization	
Attribute	Techniques/Remarks
After test configuration	Neutron radiography of capsule
Density	Determined by dimensional and weight measurements
Dimensions	Determined by length and diameter measurement at several points
Grain size and structure	Determined from optical ceramography, SEM and image analysis
Thermal conductivity [†]	Determined using laser flash at temperatures from 300K to 1600K in 100K increments
Fission gas release	Gas puncture plus chemical analysis
Fission product distribution	Electron Microscopy plus EDS/WDS analysis
Distribution of porosity	Determined from optical ceramography, SEM, computed tomography and image analysis
Transient Behavior	Determined by rapid heating of pellet in a furnace
Annealing behavior	Determined by long term heating of pellet in a furnace capable of monitoring/measuring fission product release.

[†] Upper test temperature may change depending on what capabilities are available at the time.

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Enclosure 2
to

MDO-723-0015/ B-MT(EDT)S-028

Post Irradiation Examination Plan for MICE-3B Space Materials

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List of Acronyms

AL	Analytical Laboratory
ATR	Advanced Test Reactor
CCCTF	Core Conduction Cool-down Test Facility
CMR	Chemistry and Materials Research
CNC	Computer Numeric Control
DSC	Differential Scanning Calorimeter
DTA	Differential Thermal Analyzer
EBR-II	Experimental Breeder Reactor – II
EDS	Energy Dispersive Spectrometry
EML	Electron Microscopy Laboratory
EMM	Electro-Mechanical Manipulator
FCF	Fuel Conditioning Facility
FIMA	Fissions per Initial Metal Atom
GT-MHR	Gas Turbine Modular High Temperature Reactor
HFEF	Hot Fuels Examination Facility
HLRF	High-Level Radiochemistry Facility
HTGR	High Temperature Gas Reactor
ICP-MS	Inductively Coupled Plasma – Mass Spectrometer
IFEL	Irradiated Fuels Examination Laboratory
IMGA	Irradiated Microsphere Gamma Analyzer
INL	Idaho National Laboratory
KAPL	Knolls Atomic Power Laboratory
LANL	Los Alamos National Laboratory
LDRD	Laboratory Directed Research and Development
LFA	Laser Flash Apparatus
LWR	Light Water Reactor
M&C	Metals and Ceramics
MFC	Materials and Fuels Complex
MICE	Multiple Irradiation Capsule Experiment
NIST	National Institute of Standard and Testing
NPR	New Production Reactor
ORNL	Oak Ridge National Laboratory
PIE	Post Irradiation Examination
PNNL	Pacific North National Laboratory
RML	Radioactive Materials Laboratory
RPL	Radiochemical Processing Laboratory
SAL	Shielded Analytical Laboratory
SEM	Scanning Electron Microscope
TGA	Thermo-Gravimetric Analysis
WDS	Wavelength Dispersive Spectrometry
XPS	X-ray Photoelectron Spectrometer
XRD	X-Ray Diffraction

The following is a summary and assessment of capabilities of various facilities considered for post irradiation examination (PIE) of MICE-3B fuel specimens. Since a PIE plan was not fully developed before the restructuring of the Prometheus Program, direction for further investigation is indicated where applicable.

I. Knolls Atomic Power Laboratory (KAPL)- Radioactive Materials Laboratory (RML)

The KAPL-RML could not accept low-enriched fuel after irradiation, as they did not have procedures developed to handle the transuranic elements that would be in the spent fuel nor did they have transuranic detection equipment. KAPL-RML has many of the evaluation capabilities needed for fuel evaluation such as ceramography, dimensional and weight measurement, and can puncture; however, they would need to purchase / install some examination equipment such as a laser-flash unit for thermal diffusivity measurements of irradiated fuel (currently they only have this capability only for low-activity structural materials). Therefore, KAPL was not considered for MICE-3B space reactor fuel evaluation.

II. Los Alamos National Laboratory (LANL)

The hot cell facilities are located in Wing 9 of the Chemistry and Materials Research (CMR) building in TA 3 consisting of sixteen 6' x 6' x 11' hot cells connected by a central corridor in a Category III facility. The hot cells were very clean; however, several of them currently do not contain equipment. LANL maintains this cleanliness by performing all operations that could result in significant hot cell contamination in large metal liners that are fabricated to fit within the hot cells. These liners approximately fill the interior volume of a hot cell and contain ports through which the manipulators pass. At the time of the visit the hot cells were largely inactive. Equipment that is of potential use to the space reactor program included metallographic mount preparation equipment, a microhardness tester, and a tensile tester. The large amount of open space within the hot cell and the relative inactivity suggest that this facility would be available for the examination of space reactor program specimens, although a considerable amount of infrastructure development may be required (liners, equipment, etc). The CMR building is currently scheduled for demolition, although the hot cell wing may be retained.

The Sigma building at LANL houses an extensive capability for processing and characterizing materials. Test frames capable of loads in excess of one million pounds can be used for testing or fabrication of large-scale components. They possess excellent capabilities for deformation processing, electron-beam welding of large components, high resolution orientation-imaging microscopy, and focused ion beam sample preparation [1].

While LANL has some limited PIE capability for fuel, they are not the most capable facility. Many pieces of equipment, which are already in place at other national laboratories, would need to be procured and installed. Therefore, NRPCT is not considering Los Alamos for MICE-3B space reactor fuel evaluation.

III. Oak Ridge National Laboratory (ORNL)

Portions herein have been excerpted from reference [2].

The ORNL-Irradiated Fuels Examination Laboratory (IFEL) contains a large horseshoe-shaped array of hot cells which are divided into three work areas, each with a different level of cleanliness. The hot cells are constructed of 3-ft thick high density concrete walls with oil-filled lead glass viewing windows. The inside surfaces of the cell bank are lined with stainless steel to provide containment of particulate matter and to facilitate decontamination. Special penetrations are provided for the sealed entry of services such as instrument lines, lights, and electrical power. A pair of manipulators are located at each of 15 window stations for remote cell operations, and periscopes allow for magnified views of in-cell objects. Heavy objects within each cell bank can be moved by electromechanical manipulators (EMM) or a 3-ton crane. Fuel materials enter and leave the cells through three shielded transfer stations provided at the rear face of the North cell. Two small diameter (6.5 & 14.5 in) horizontal transfer stations are used for small objects (less than 8 ft in length). Items up to 4 ft by 4 ft by 6 ft in size can be transferred through the shielded air-lock door system. The facility has a second level which can be accessed through the hot cells and can be used for storage and repair of equipment. A layout of the facility can be seen in Figure 1.

Figure 1 Layout of Irradiated Fuels Examination Laboratory

Over a period of three decades this facility has handled a wide variety of fuels including aluminum clad research reactor fuel, stainless steel and Zircaloy clad light water reactor (LWR) fuel, coated-particle gas-cooled reactor fuel and numerous one-of-a-kind fuel test experiments. In addition, the facility has also done iridium isotope processing and irradiated capsule disassembly. A variety of shears, machine tools, and cutoff saws are available within the cell for the disassembly of irradiation capsules and the preparation of fuel specimens. The facility has experience in the handling and cutting of a wide variety of capsule and clad materials such as Inconel, stainless steel, Zircaloy, aluminum matrix, and graphite-based materials. A gamma scanner and gamma spectrometer are available for the non-destructive examination of moderate length fuel rods and individual specimens.

Metrology equipment such as mass scales and dimensional tools are available and routinely used. A fully equipped metallographic line is maintained in the West Hot Cell. The IFEL also has other facilities outside the main bank of cells. Three free-standing, shielded cubicles are located on the first and second floors of the building which contain specialized equipment associated with the post-irradiation examination and accident condition testing of HTGR type fuels.

The Scanning Electron Microscope (SEM) cubicle is located on the second floor above the West Hot Cell and is connected to the West Cell by an in-wall shielded elevator. The cubicle contains a precision balance and a vacuum-coating system for both gold and carbon coating of SEM specimens. A special transfer port is mounted on the side of the cubicle which allows remote handling of specimens to be loaded into the SEM system located adjacent to the cubicle. The SEM currently available is a JEOL JXA 840A system with both wavelength dispersive spectrometry (WDS) and energy dispersive spectrometry (EDS) capability. Radioactive specimens up to 1 R/hr can be examined in this system. While this is located outside the main cell, small amounts of fuel can be examined here.

The Irradiated Microsphere Gamma Analyzer (IMGA) cubicle is located on the second floor above the East Hot Cell and is connected to the East Cell by an in-wall shielded elevator. The cubicle contains several pieces of equipment designed for handling individual HTGR microspheres (coated particles). The cubicle has a shielded stereomicroscope with a 3 axis stereo-stage and micro-manipulator for handling individual particles. It also houses the IMGA system which is a fully automated device for examining individual HTGR microspheres using a high-resolution gamma-ray spectrometer. The IMGA system measures the absolute activities of the particles it examines, and, by means of special user programmed instructions, is able to segregate particles based on the measured activities.

The Core Conduction Cooldown Test Facility (CCCTF) cubicle is located on the first floor of the IFEL. Samples to be tested are transferred to and from the cubicle using small shielded casks. The cubicle contains a fully programmable furnace facility with special sampling features. The furnace is capable of temperatures up to 2000°C when using an inert atmosphere such as helium and is equipped with fission product monitoring equipment for both solid and gaseous elements.

The facility contains modular hot cells which allow for rapid reconfiguration. Equipment such as in-cell SEM, laser flash diffusimeter and gas puncture equipment can be inserted into this cell and then removed when work is complete. This would allow for thermal conductivity measurement of irradiated MICE fuel and measurement of the mini-can gas.

The IFEL has no neutron radiography capability; however, X-Ray Computed Tomography can be done on both irradiated and non-irradiated materials. Fuel materials can be examined if the specimen size is small (a MICE pellet would be too large). Materials properties such as tensile strength in vacuum and Charpy impact strength can be tested in-cell. Thermal property analysis of low radioactivity materials are usually performed out-of-cell in laboratories. However, some equipment can be moved in cell for testing fuel.

Oak Ridge has an established waste stream and can easily handle GE 100 and NRBK casks. The GE 2000 cask can also be handled, but is more difficult.

The ORNL personnel are available with experience in a wide variety of fuel PIE programs and analysis techniques along with the detailed reporting and quality control requirements for nuclear programs. The Metals and Ceramics (M&C) division contains a wealth of experience in fuel fabrication, metal and ceramic material behavior, irradiated material behavior, and material testing. Ongoing programs at ORNL maintain experience in hot cell techniques and analysis. In addition, academic and industrial consultants are available to meet special program needs and to conduct reviews. Recent work includes extensive support for the Gas Turbine Modular High Temperature Reactor (GT-MHR) program, the New Production Reactor (NPR), a cooperative gas cooled reactor agreement with Japan, (U,Pu)O₂ fuel development for the Fissile Materials Disposition Program and the handling of legacy fuel under the National Spent Fuel program.

When the Prometheus Program was restructured, the ORNL facility had not yet been fully evaluated by NRPCT personnel for testing of MICE-3B space reactor fuel. However, based on information obtained thus far, ORNL appears to be a viable candidate for fuel evaluation. The major disadvantage is the lack of neutron radiography which would prohibit the determination of end of test capsule configuration. This measurement would allow the determination of final gap dimensions which may be lost once the can is opened.

IV. Pacific North National Laboratory (PNNL)

Portions herein are excerpted from reference [3].

Examination of irradiated fuel is performed in the High-Level Radiochemistry Facility (HLRF) and the Shielded Analytical Laboratory (SAL) of the Radiochemical Processing Laboratory (RPL) which house a variety of equipment in several areas as described below.

The RPL (Building 325) is a rectangular-shaped, metal frame structure erected on a concrete base and has a total of 144,092 square feet of space. The first floor of the main building contains primarily research laboratories, in addition to a front wing, an east wing, and a west wing. The east and west wings house the HLRF and SAL hot cells, respectively. The basement of the main building is mostly mechanical space but also houses several laboratories, including a small hot cell laboratory and a nondestructive analysis (NDA) laboratory.

High-Level Radiochemistry Facility (325A-annex): The HLRF contains three interconnecting hot cells (A-Cell, B-Cell, and C-Cell) and supporting facilities originally designed to conduct bench- and pilot-scale aqueous processing. The 325A-annex is a 25.9-m by 14.7-m (85- by 48-ft) steel-framed structure, with partial height reinforced concrete walls. The remaining height of the walls is finished with metal siding. The HLRF hot cells are located on the first-floor level and supported by heavy reinforced-concrete piers, columns, and pilasters. The "front face" and sides of these cells are constructed of 1.2-m- (4-ft) thick, high-density-concrete walls. The "rear face" of the cells is constructed as a (3-ft) thick, high-density concrete wall. The inside cell dimensions are 4.6-m (15-ft) high by 2.1-m (7-ft) deep; A-Cell is 4.6 m (15-ft) wide, and B- and C-Cells are each 1.8 m (6 ft) wide.

In-cell operations are performed using medium duty electro-mechanical manipulators in the "front face" operating gallery. In-cell viewing is through leaded-glass, oil-filled windows. Closed circuit television cameras, monitors, and video recorders are also used for performing detailed inspections or for documenting experimental results. A- and C-Cells contain overhead, bridge-

mounted, 2200-kg (1000-lb) capacity hoists. Each hot-cell is equipped with electrical, vacuum, air, and distilled and process water services. The front face walls also contain shielded service penetrations for installation of additional process, electrical, and instrumentation services as required for a particular project. Each hot cell contains several process vessels, located below the cell work decks, which range in capacity from 4 to 320 liters depending on the cell.

Each hot cell is equipped with a floor drain routed to separate critically safe sump tanks (slab tanks) located in a separate shielded room directly below the hot cells. These drain tanks are located within a stainless steel containment, which provides a secondary containment for any potential leakage from the tanks or piping. Recovery piping to the tanks provides capability to remove tank contents for process or disposal requirements.

Access to each cell, for renovation or installation and removal of large items, is provided through individual 45.7-cm (18-in) thick by 2.1-m (7-ft) high by 1.5-m (5-ft) wide Meehanite iron main shield doors in the rear face wall. Material and equipment are normally installed or removed at the rear face, through a 33-cm (13-in) diameter shielded double-door transfer port, which is built into the main shield door of each hot cell. A- and C-Cells also contain shielded double-door transfer mechanisms built into the rear face wall. The transfer mechanisms are primarily used for rapid retrieval and transfer of samples. The transfer mechanism openings measure approximately 25-cm (10-in) wide by 30-cm (12-in) high.

The rear face operating gallery, or hot cell canyon, is used for handling casks, contains the manipulator decontamination and repair shop, and is equipped with 3 fume hoods and a stainless steel glove box with approximate dimensions of 96-in long by 41-in square. An overhead pendant-controlled, bridge-mounted crane services the length of the rear hot cell canyon. The bridge crane is equipped with a 30-ton capacity main and 5-ton capacity auxiliary hoist. The crane is used to load and unload casks from transportation vehicles and to position casks at the "rear face" of the cells for loading or unloading into the cells through the transfer ports. Cask payloads measuring up to approximately 33-cm (13-in) diameter by 254-cm (100-in) long and weighing up to 2200-kg (1000-lb) can be transferred or removed from the cells. A truck lock is connected to the hot cell canyon. The truck lock serves as a ventilation buffer between the hot cell canyon and the outside. Casks are on- and off-loaded by backing the trailer through the truck lock and into the cask-handling area so that the cask can be accessed with the bridge crane.

Shielded Analytical Laboratory (325B): The SAL contains six interconnecting "hot" cells, each about 1.7 m (5.5 ft) square by 2.9 m (9.5 ft) high, enclosed in shield walls of either 66-cm (26-in) magnetite (Fe_3O_4) concrete or 30.5-cm (12-in) Meehanite iron. The hot cells are served by front and rear face operating galleries, similar to those described for the High-Level Radiochemistry Facility. The six hot cells are located on the first floor and are enclosed in a 16.5-m by 10.7-m (54- by 35-ft) steel-framed, reinforced-concrete structure. The front face of each hot cell is equipped with two medium duty manipulators and a high-density, lead-glass viewing window which has the same shielding effect as the cell wall.

The SAL is designed for performing a wide variety of analytical chemistry operations using highly radioactive samples. Specialized analytical instrumentation located within or next to these hot cells permits several types of analyses to be completed without transferring the materials from the SAL hot cells to other analytical laboratories. For those analyses that must be completed in other laboratories within the RPL, turntables built into the rear face of the cells provide safe, rapid sample transfer out of the cells.

Axial Gamma Scanning

Axial gamma scanning is performed in the HLRF A-Cell. While the capability currently does not exist, plans are in place to install a new system. The gamma scanning system will be based on the design of a system that was built and used successfully at PNNL in the recent past. The system will consist of a fuel rod holder and drive mechanism that will be placed inside A-Cell. A collimator with variable slits will be located in a penetration of A-Cell to define the field of view of an external gamma detector. The system will be automated such that a computer-controlled analyzer will move the rod and collect a gamma spectrum in programmable step sizes and counting times. This will result in profiles of all the gamma emitters as a function of position along the length of each fuel rod that is scanned. Such data can be used to determine fuel burnup by comparison with burnup calculations.

The hardware will consist of the fuel rod holder, drive mechanism, gamma collimator, and the gamma detector system. The gamma detector will be a high-purity germanium detector with a liquid nitrogen cryostat. Gamma spectral analysis will be performed by a computer running commercial software, such as the Canberra APEX and GENIE software packages, that are capable of running the entire gamma scanning system, collecting data, analyzing the data, and preparing reports of the measured isotopes. Digital spectrum analyzer systems will be used to ensure state-of-the-art counting rate capabilities to minimize system dead time and allow the scanning to be performed in the shortest time possible.

Metrology

Rod length and axial laser profilometry measurements are performed in the HLRF A-Cell. The rod holder, drive mechanism and positioning control system described above for the axial gamma scanning measurements are used for the laser profilometry measurements as well. From the profilometry results, data concerning diametrical rod swelling as well as rod bowing can be extracted. The laser profilometry data acquisition system (to be developed) will be based on similar systems developed and deployed recently by PNNL. Testing and calibration of the length and profilometry systems are performed using dummy pins and NIST-traceable standards.

Rod Puncture and Fission Gas Collection

Rod puncture and fission gas collection are performed in the HLRF A-Cell. The gas collection system which is being developed will be similar to arrangements used in the past at PNNL for in-cell gas collection and analysis. The A-Cell is presently equipped to perform gas sampling, so no hot cell modifications will be required. The measurements will provide the volume of gas in the pin as well as the gas composition. The composition data can be used as a cross-check on fuel burnup calculations and gamma scan results. Gas analysis will be conducted using an existing Finnigan MAT-271 mass spectrometer. Not only will the spectrometer provide elemental composition of the gas, but the isotopic composition of each element can be determined as well. The only equipment purchases required for gas collection and analysis will be pressure transducers and vacuum pumps.

To puncture the rods, a mechanical system has been used in the past and is still operational. However, if laser flash thermal diffusivity measurements are to be made on the fuel, a laser head could be used to provide a laser rod puncture capability. The advantages of laser rod puncture include faster processing and easier sealing. Both of these advantages result in less likelihood of

gas sample contamination with atmospheric constituents. If the laser puncture method is used, the laser head would remain outside the hot cell and the beam would be directed into the cell by appropriate optics.

Microscopy and Spectroscopy

Following the A-Cell examinations described above, selected fuel pins will be transferred to the HLRF B-Cell. B-Cell is equipped with a saw for sectioning the pins at locations of interest and the equipment required for potting and performing initial polishing of metallographic mounts and preparation of electron microprobe specimens. Metallographic and electron microprobe samples can then be transferred from the HLRF B-Cell to the SAL hot cells 4 and 5 for additional examination. Optical and scanning electron microscopy equipment is in place and can be used to examine the state of the fuel. Microscopy and spectroscopy of the spent fuel solids provide information on the nature of accumulated damage and the disposition of fission products and other species. To identify the location of light elements (e.g. nitrogen and oxygen), light element x-ray detectors or wavelength dispersive detectors need to be employed. The techniques used to examine spent fuel must be performed with careful sample preparation to avoid damaging the microstructure. Additional information on the state of the fuel may be obtained from x-ray photoelectron spectroscopy and x-ray or electron diffraction analyses.

Optical microscopy can be used to obtain cross-sectional views of fuels. Optical microscopy is typically performed in the hot cell with a specially adapted camera. Optical images provide information on the overall structure of a fuel segment, including regions of porosity. Light microscopy remains one of the most useful tools for examining any material including spent fuel.

Electron microscopy is a useful technique for examining the microstructure of fuels. However, the technique is subject to sampling problems and sample preparation artifacts. Images are dependent on the sample preparation route, accelerating voltage, and resolution of the instrument; therefore, care is needed when using observed microstructures to interpret spent fuel behavior. All electron beam techniques that provide compositional information (e.g. x-ray wavelength dispersive analysis) are subject to problems that can make quantitative analysis difficult. Generally, the analysis of light elements in a heavy element matrix is problematic because of absorption of either incoming beam or the characteristic emission.

Electron microprobe spectroscopy is a well-established technique for obtaining compositional information on solids. The wavelength detection system has better sensitivity than x-ray energy dispersive systems. However, sample preparation of spent fuels could lead to large compositional changes, especially for the gaseous species. Any analysis of Xe or Kr over the pellet must consider the possibility of gas release during polishing. The excitation depth for a typical microprobe is 1 μm ; however, in a uranium-solid, this depth is reduced significantly. Detection efficiency of L - lines (~ 4 keV) will improve in the less dense regions of the fuel.

The x-ray photoelectron spectrometer (XPS) can be used to probe the oxidation states of actinides using the $N_{45/2}$ and $N_{47/2}$ lines. The technique is surface sensitive. Synchrotron radiation offers the possibility of chemical analysis of a region in a fuel with an x-ray energy dispersive detector, serial analysis with x-ray absorption spectroscopy of individual elements, and high energy diffraction of phases in the fuel. Relatively large samples may be examined as long as sample handling issues can be resolved.

X-ray diffraction (XRD) has been used to measure lattice parameters for determining oxidation of fuels. X-ray diffraction is a bulk analysis method; the signal intensity depends on the atomic number of the material, the size of the crystallites, and the degree of crystallinity. XRD is less sensitive compared to neutron or electron diffraction for detecting oxidation of fuels. During the initial stages of burn-up, XRD detects a slight peak broadening that may correspond to defect formation in the lattice.

Thermal Diffusivity

Thermal diffusivity measurements can be made in the SAL hot cells 4 and 5, after sample preparation in the HLRF B-Cell. Preparation of the thermal diffusivity samples is similar to that for optical or scanning electron microprobe samples. Measurement of thermal diffusivity (in order to calculate thermal conductivity) of irradiated fuel will be a new capability in the RPL hot cells, although PNNL has significant experience in thermal diffusivity testing of low-activity irradiated materials such as SiC. The most widely used system that is designed to measure the thermal diffusivity is the Laser Flash Apparatus (LFA). Compared to the direct measurement method, the main advantages of this non-contact, non-destructive method for radioactive samples include simple sample geometry, easy sample preparation, small sample size, applicability over a wide range of temperatures (room temperature to 1600°C with a SiC heating element), and excellent accuracy and reproducibility. While the capability does not currently exist, the instrument that will be purchased is the Netzsch LFA 427 system. It consists of four main components: 1) measuring unit with furnace sample carrier and infrared detector, 2) controller for the measuring unit, 3) laser system connected by fiber optics, and 4) data acquisition system. The system will allow cut radioactive samples to be mounted on a carrier system inside a furnace within the hot cell. The sample is heated to the desired temperature and then a burst of laser energy is directed to the face of the sample, resulting in rapid, homogeneous heating. The relative increase in temperature on the opposite face is measured as a function of time by an infrared detector. The thermal diffusivity is computed using the data acquisition software using a variety of computational methods for comparison. Thus, the multiple diffusivity values can be compared to evaluate the system for errors associated with laser pulse or heat loss effects. Samples can be selected from specific locations of the fuel to provide any indication of structural effect on the thermo-physical properties.

Cladding Mechanical Properties

The de-fueled 'cladding' pieces such as the Haynes 230 mini-cans can be transferred to the SAL hot cells 4 and 5 for mechanical property sample preparation and testing, if desired. A computer numeric control (CNC) milling machine can be used to fabricate specimens for the tensile and creep rupture tests, and for other machining requirements as necessary. This would minimize the amount of hot cell work, and will result in consistent specimen dimensions. Existing or new load frames would be installed in the hot cell, and furnaces tailored to the specific testing temperatures and environments can be purchased and installed on the load frames. PNNL has significant experience in testing miniature-sized mechanical property samples in-cell for DOE fast reactor development and fusion materials programs.

The mechanical properties of irradiated cladding are to be evaluated at ambient and elevated temperatures in vacuum or inert environments. The testing will be done using closed-loop mechanical test systems, which are much more adaptable to use in hot cells than servo-hydraulic test systems. Each test system will employ a computer-controlled stepper motor and digital data

acquisition system to apply the desired stress or displacement rate. Load cells and extensometers can be used to measure load and displacement. Tensile tests and creep-rupture tests can be used to measure the mechanical properties. The results of these tests will include yield strength, ultimate tensile strength, elongation to fracture, and stress-rupture properties.

When the Prometheus program was restructured, the facility had not yet been fully evaluated by NRPCT personnel for testing of MICE-3B space reactor fuel. However, based on information obtained thus far, PNNL does not appear to be the best candidate for fuel evaluation. They can install and operate a multitude of equipment but do not have much of the desired equipment pre-installed. In addition, like ORNL, they lack neutron radiography capability.

V. Materials and Fuels Complex at Idaho National Laboratory (INL-MFC)

Idaho National Laboratory's Materials and Fuels Complex has three hot cells in operation: the Hot Fuels Examination Facility (HFEF), the Fuel Conditioning Facility (FCF), and the Analytical Laboratory (AL).

HFEF is a large (about 64,000 ft²) Hazard Category 2 Nuclear Facility that provides shielding and containment for highly radioactive fuels, materials and waste form remote examination, processing, and handling. Two large hot cells are the heart of this facility. The main cell has 15 workstations, is stainless steel lined, hermetic, and is filled with high-purity argon gas with water and oxygen levels kept to below 60 ppm although 10 ppm levels are possible. The decontamination cell has six workstations, is air-filled and supports activities in the main cell as well as direct program activities. The main cell is capable of handling most full-length LWR fuel assemblies and rods. HFEF was completed in 1974 and has only required minor upgrading. Figure 2 shows the layout of the facility.

HFEF Process Areas and Equipment Locations

Figure 2 Layout of HFEF [4]

FCF is a 53,000 ft² facility containing two large hot cells, one donut-shaped with an argon atmosphere and the other rectangular-shaped with an air atmosphere. The argon cell is an annular, 16-sided polygon that is completely lined with a seal-welded steel liner, and in its center

is an observation room. This cell has 15 workstations around the outer periphery and four inside the central observation room. The air cell has six work stations. A large mock-up area near the cells is used for testing and checkout of equipment prior to installation. FCF was activated in 1963 and has undergone several refurbishment projects to upgrade equipment and improve the cells over the years.

The AL facility has about 20,000 ft² and contains six negative pressure, air-filled hot cells that are equipped for remote chemical analysis of radioactive samples. The facility also contains a wing of glovebox-equipped and instrument-equipped laboratories for analyses of samples that do not require extensive shielding. The cells are connected to the larger hot cell facilities at MFC—FCF and HFEF, via a pneumatic transfer system (rabbit tube). The AL is a Hazard Category 3 facility as well as a Category III Nonreactor Nuclear Facility.

NRPCT personnel visited the Materials and Fuels Complex at Idaho National Lab (INL-MFC) and reviewed the capabilities for post irradiation examination. INL was eager to entertain NRPCT work and had the capability to do so. The hot cell facility was well-maintained and all equipment appeared to be in good working order; however, much of the equipment was installed for the International Fusion Reactor Program in the early 1990's and it hasn't been used much since then. Staffing was low and there were few materials specialists on hand. Only 10 hot cell operators were employed at the time of the visit but INL was confident that they could staff up if needed. INL-MFC is set up for handling highly radioactive fuel elements, which is a good fit for evaluation of fuel specimens and elements being irradiated in the Multiple Irradiation Capsule Experiment (MICE) in the Advanced Test Reactor and other ATR platforms. They could accept the GE-2000 cask, dispose of the waste, perform all testing with current equipment except thermal conductivity and work with low enriched fuel. PIE capabilities are addressed below.

Ceramography / Metallography and Microscopy

An enclosed section of the hot cell (called the containment box) is dedicated for sectioning and prepping of metallographic samples for SEM or Metallography. This box is kept at a negative pressure relative to the cell as to ensure no contaminants leave the box. It is connected with the metallographic box and the rest of the hot cell via a pneumatic tube. Optical characterization of the samples are conducted using a stereomicroscope (4, 10 and 25x) or a Leitz metallographic (20x to 800x) microscope located in the met-box. Image analysis performed on photomicrographs is used to quantify microstructural features such as grain size and structure, porosity, etc.

Electron microscopy can be conducted using an SEM in the hot cell which can image up to 30,000x and can accommodate specimens up to 3.8 cm x 3.8 cm (1.5 x 1.5 inches); however, there is no energy dispersive spectroscopy (EDS) or wavelength dispersive spectroscopy (WDS) capability. The Electron Microscopy Laboratory (EML) also has an SEM which does have the EDS and WDS capabilities. To use this facility, the specimens must be reduced in size such that the total Curie content is below a critical value. For fuel, the size limitation is generally too small to be practical so EDS and WDS characterization are generally only performed on metals. There is currently no in-cell microprobe capability; however, INL is actively pursuing this equipment and it will be installed either in the hot cell or in the electron microscopy lab. It is currently in the funding recommendation stage of procurement and, if procurement proceeds, would be available sometime after 2007.

Vickers hardness testing is performed in the metallographic box. It has magnification up to 400X and can use a hardness tester weight of 5 grams to 400 grams.

Metrology

Dimensional measurements of fuel pellets could be made in a variety of ways, but normally they would be made using an optical comparator on neutron radiographs. Contact profilometry is used for measurement of diameters. It can accept element diameters ranging from 4.4 mm (0.174 in) to 21 mm (0.84 in) having a maximum diametral swelling of 0.5 mm (0.02 in). The accuracy of this is 0.0076 mm (0.3 mil) and is achieved by calibrating the equipment to a standard before measurement. This method is used in the hot cell as optics used in laser profilometry tend to brown and require frequent replacement. INL works to NQA-1 standards.

Fuel pins and capsules are weighed using an in-cell balance. Accuracy on objects less than 1,000 grams is ± 0.02 grams. More accurate weights of fuel pellets could be obtained by transferring pellets to the Analytical Laboratory.

Specific Gravity is measured using a high precision pycnometer. They can receive samples up to 17.7 cm (7 inches) long and 7.6 cm (3 inches) in diameter.

Wet Chemistry Capabilities and Burnup Measurements

In the analytical lab, they can analyze for carbon, nitrogen and oxygen at the ppm level. ICP-MS is used for burnup calculations and isotope concentrations can measure at the ppt level. Typically they measure burnups in the 5-20% FIMA range but can measure lower values.

Thermal Properties

Thermal diffusivity and specific heat measurements exist for unirradiated materials only up to 1200°C. No irradiated thermal diffusivity capability is installed; however, installing a laser flash diffusivity apparatus in cell is possible. INL is currently planning research to develop an alternate method of in-cell thermal diffusivity measurement which will allegedly be more accurate than laser flash. As of September 2005, they have submitted a LDRD proposal for this work. Thermal analysis (DTA, DSC, TGA) equipment exists in the analytical lab and can only be done on low activity items.

Annealing Furnaces

As of April 2005, INL did not have an annealing furnace in cell; however, they were in the process of procuring a high temperature (2000°C) furnace for another program. The furnace uses a flowing sweep gas and is capable of fission product monitoring and inventory. Installation was expected to be complete in early 2006.

Can Puncture

INL has this capability installed. The fuel can is placed in a chamber with a precisely known volume and pressure. The can is then punctured with a laser to release the gas and the pressure of the chamber is measured, which yields the fission gas release. The gas can be sampled for elemental analysis. The system can accept elements from 4.4 mm (0.174 inches) to 21.1 mm

(0.832 inches) diameter and from 2.5 cm (1 inch) to 386 cm (152 inches) long. The same laser can re-seal the can if needed. A sample of the fission gas can be taken during the puncture and then transferred to the analytical lab for compositional analysis.

Radiography and Spectrometry

Both gross gamma scanning and isotopic gamma scanning exist in cell and can be used for relative fuel burnup and power profiles of fuel, structural activation profile of core components, position and dimension of internal structures within fuel assemblies, relative distribution of various isotopes of interest, and identification of breached elements or capsules. The equipment can handle a maximum vertical travel of 3.8 meters (152 inches).

Neutron radiography can be done using a system which was designed to radiograph highly radioactive specimens. It uses a 250 kW TRIGA reactor, located in the basement of the hot cell and can radiograph specimens up to 16.5 cm (6.5 inch) in diameter, 3.8 meters (152 inches) long and up to 600 lbs. Two different films are used, one which is sensitive to fast neutrons and one that is sensitive to epithermal neutrons. Once used, these films become active and they emit gamma rays which are in turn used to expose x-ray film thus allowing the image to be transferred. This allows clear pictures to be taken with resolutions up to 0.025 mm (0.001 inch).

X-radiography and X-ray diffraction capabilities exist out of cell and can only be used for low activity items.

Handling, Disposal and In-Cell Machining Capabilities

INL can handle all commercial casks as well as NR casks up to 17 ft long, 56 inches in diameter and 32 tons. Water in the casks is not a problem. Items can be moved using a 40-ton crane in the high bay, 25-ton crane in the truck lock, and a 4-ton freight elevator. In cell, there is a 5-ton crane and two electro-mechanical manipulators rated for 750 lbs each.

There is the metallographic sectioning saw for precision cutting in the containment box. In the main hot cell there is a chop saw for cutting capsule open and a milling machine with 3 μ m (0.0001 inch) precision.

They have the capability to disassemble irradiation test capsules and the possibility of reconstituting specimens back into a new test capsule. They have limited experience doing this in the past with EBR-II, with some development they may be able to do this with the ATR.

Materials which have been irradiated in Idaho (i.e. the ATR) can be disposed of by INL. They have a storage facility and waste path already established for fuel.

Overall, the Materials and Fuel Complex is an impressive facility which could meet the MICE-3B space reactor fuel testing requirements. At the time of this evaluation, thermal diffusivity is the only test equipment that would need to be installed; however, INL was trying to get funding for the development of this capability and may have it installed when needed. If it does not get funding, NRPCT would need to investigate if it is more practical to send the fuel to a different location such as PNNL or ORNL (assuming their plans to install thermal diffusivity go through) than to fund the installation of a laser-flash unit for thermal diffusivity measurements in HFEF. In addition, since

thermal conductivity of uranium oxide is fairly well established, this test may have been deemed unnecessary.

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Enclosure 3
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MDO-723-0015/ B-MT(EDT)S-028

Envisioned Future Fuel Testing in the MICE Facility

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Compromises in MICE-3B

The MICE-3B irradiation test of space fuel materials was envisioned to be the first of several tests to be performed in a MICE test train. While MICE-3B was a test which would have provided valuable information for the development of space nuclear fuel, it was not without compromise.

Compromises include:

- Low fast-to-thermal (neutron) flux ratio
- High total flux (which affects power density)
- Small test volume (low number of samples)
- No online fission gas monitoring capabilities
- Relatively large temperature uncertainty
- No variety in processing technique of the samples tested
- No variety in pellet size of the samples tested.

Future tests would build on previous experiences and attempt to compensate for these compromises.

The Prometheus-1 reactor is envisioned to be a fast reactor whereas the ATR is a thermal reactor. This will inherently create a lower fast-to-thermal flux ratio in the ATR than would be experienced in Prometheus-1. This ratio will affect power density and fission product distributions in the tested fuel specimens as well as transmutation of materials. The fast-to-thermal flux ratio can be enhanced to some extent through the use of hafnium filters; however, the amount enhancement is limited since too much hafnium will affect reactor performance. A more favorable fast-to-thermal flux ratio will be incorporated into fuel element testing as described in reference [1].

The total neutron flux in the MICE test facility is relatively high. This results in a high power density for fuel with an enrichment typical of a space reactor. As a result, enrichment of the fuel in MICE-3B was significantly reduced to meet power density requirements. This also has an effect on the fission product composition and distribution since ^{238}U (more specifically, ^{239}Pu bred from fertile ^{238}U) will have to produce different concentrations of fission products than ^{235}U . Like the fast-to-thermal flux ratio, the total fluence received by specimens can be reduced by adding hafnium filters but the limitations on the amount of hafnium which can be used are the same as previously stated.

The capsules in the top and bottom tiers of a MICE test train have the lowest flux and, therefore, are the most favorable for space reactor fuel testing, thereby limiting the potential test volume to two fifths of total test capacity. In addition, there are other test programs interested in these locations. This limited the space program to testing UO_2 fuel in only one of the MICE test train capsules. The most favorable design allowed the testing of only 18 pellets in three fuel cans. This limited measurements such as gas release to three data points, unconstrained swelling performance to twelve data points and constrained swelling performance to six data points. Measurement of these properties typically have a large scatter and therefore, these few data points would be of high uncertainty.

Online fission gas monitoring capabilities, temperature uncertainty, variety in processing technique, and variety in pellet size were compromises in MICE-3B, as a result of time constraints. Future testing in the MICE platform would aim to reduce these compromises. This is detailed below.

Goals of Future Testing in the MICE Facility

Specific goals of future MICE tests for space reactor fuel would have depended on space reactor design developments such as the selection of a clad material and the decision on whether or not to develop a moderated core. However, there are some general characteristics such as decreased temperature uncertainty which would be incorporated in future tests. This could be done in a number of ways such as using melt wires, direct instrumentation of the fuel instead of the can or operating the fuel at can temperatures (and vice versa). These were all considered but dismissed for MICE-3B due to the extended amount of time needed for development.

Some more specific test objectives which would be considered in future tests include:

- Venting and monitoring of fission gas from bare fuel
- Comparing fuel samples made from different processes
- Studying clad-liner-fuel interactions and compatibility
- Studying effects of pellet size and geometry
- In-reactor transient testing.

I. Venting and Monitoring of Fission Gas from Bare Fuel

As mentioned above, fission gas containment was a compromise made in MICE-3B. This containment has two effects on the ability to measure the desired properties. First, low thermal-conductivity fission gas mixes with the helium fill gas in a fuel-can and increases thermal resistance of gaps between fuel pellets and the support structure as well as gaps between the support structure and the fuel can. This would increase the fuel temperature even though the fuel-can temperature remains the same, thus creating uncertainty in temperature measurement. Secondly, since fuel pellets are sealed in a fuel can, only the cumulative gas release can be measured after the test.

If fission gas is allowed to be vented through the sweep gas of a capsule, the online instantaneous activity measurements would be used in calculations of the fission gas release rate over time. Implementation of this technology requires control of the radioactive species through cold traps, filtering, or recirculating the sweep gas. The size and number of fuel specimens affects the degree of control necessary for fission-product release, since the fuel volume required to make the release a non-issue may be too small to produce practical results. Further consideration of this subject is necessary. Recirculating contaminated sweep gas or filtering fission products out of the gas stream are possible but would require installation of new equipment into the reactor area; additional radiological and safety calculations must be made. Details on this can be found in reference [2].

II. Comparing Fuel Samples from Different Processes

The NRPCT was investigating the advantages of various fuel manufacturing methods including pellet fuel derived from a variety of powder processing methods as well as internal gelation methods. Initial UO_2 pellets considered for testing in MICE-3B were derived from an internal gelation process [3]. There are several other vendors in the country which produce fuel using a variety of other methods. BWXT and Nuclear Fuel Systems (NFS) both produce highly enriched uranium dioxide fuel. Global Nuclear Fuel, Westinghouse, Hematite Nuclear Fuel Manufacturing,

Siemens Power Corporation and Framatome Cogema Fuels and Technologies Groups all produce low enriched (commercial) uranium dioxide fuel. Work at BWXT and NFS was started and that progress is reported in references [4] and [5], respectively. While the fuel specifications would be the same regardless of the producers and processes, performance can vary due to differences in the final product such as grain size, porosity, and impurity levels. These variations could manifest themselves in the mechanical performance of the fuel, specifically in the transient performance or fission product retention.

One envisioned goal of future MICE tests would be performance evaluation of fuel made by various vendors and processes to determine if one has an advantage over the other.

III. Studying Clad-Liner-Fuel Interactions and Compatibility

In future tests of space fuel materials, cladding and liner materials would be down selected probably to a prime material and a backup material. While these would be used in element testing in I-holes in ATR or Removal Beryllium (RB) positions in HFIR [1], the MICE platform would allow more flexibility of the test geometry and allow for a wider range of configurations. The primary locations for space fuel element testing in MICE test trains are Tiers 5 and 1 which can provide near prototypic fission rates and can be used to test a variety of cladding options. Tier 4 typically has a neutron flux too high for testing space UO_2 fuel, but there is a possibility to test natural uranium in this position. If silicon carbide is chosen as a candidate cladding, the MICE platform could be utilized for accelerated corrosion testing using natural uranium. Palladium, which is known to have a corrosive effect on SiC, would have an increase in palladium production rate of seven fold over enriched fuel. Therefore, end of life palladium concentrations can be produced without end-of-life burnup. While palladium-silicon carbide interactions have been studied in open literature, the studies have not captured the volume to surface ratios which were to be used in a space-mission reactor.

IV. Studying Effects of Pellet Size and Geometry

Since MICE test trains use capsules of ~1 inch diameter (versus capsules of ~0.5 inch diameter envisioned with fuel element test facilities in ATR and HFIR), pellet size and pellet geometry can be easily varied. By increasing the pellet diameter, temperature gradients and peak central temperatures can be changed. Because temperature gradients create stress, this would test the fuel's strength and resistance to cracking. Alternate pellet geometries such as annular pellets can also be tested. One test capsule could contain pellets of different configurations whereas a capsule in the fuel element test train could not.

V. In-Reactor Transient Testing

Fuel performance during thermal transients is important. It was estimated that Prometheus-1 reactor fuel could rise in temperature 60-120 K at a rate of 0.5 - 3 K/sec [6] due to 1) loss of flow from one Brayton engine unit, 2) reactivity addition, or 3) over speed from one Brayton engine unit. Out-of-reactor behavior would be representative but different from in-reactor behavior and therefore, an in-situ transient test would be preferred. The difference comes from the direction of the gradient. During furnace heating, the heat flows from the outside of the fuel inward, whereas

in nuclear heating, the heat flows from the inside of the fuel outward. However, furnace heating would be useful for modeling. In theory, it is possible to test both nuclear and thermal transients in Tiers 1, 2, 4, and 5 of a MICE test train by changing sweep gas composition. While the individual capsule flow rates and compositions can vary, the total gas flow rate of the test train is 60 standard cubic centimeters per minute (sccm).

A purely thermal transient test could be accomplished by initially operating the sweep gas using pure helium, then quickly switching over to a neon rich mix which would decrease the conductivity of the gas, and therefore, increase the fuel temperature since the fuel power would remain the same. After the transient, the gas mix can be returned to pure helium. The average time for gas change in a capsule is slightly over a minute, and the thermal transient rate of 0.5 - 3 K/sec is expected to be achievable.

Therefore, it seems feasible to generate a purely thermal transient, which would be representative of a decrease in coolant thermal load. However, it may be more difficult to simulate a transient that would result from an addition of reactivity. One possible method to cause a rapid power increase in fuel pellets in MICE capsules without disturbing the reactor operation is to use helium-3 (nuclear poison) for sweep gas during normal operation. Then, to spike fuel power, the sweep gas can be changed from helium-3 to natural helium. In theory, this action should rapidly increase the fuel power without affecting the entire reactor. The magnitude of the effect and the cost for this test have not been calculated and may show that this approach is not practical.

References

- [1] MDO-723-0016 / B-MT(EDT)S-027, "In Pile Test Plans for Prometheus Fuel Elements (U)", January 2006
- [2] MDO-720-1138, "Design Review of MICE 3 B Space Fuel Irradiation Test (Design Review #2008) (U)," November 2005
- [3] MDO-CME-2120, "Fabrication of UO₂ Fuel Pellets for Space Program," January 2006
- [4] MDO-729-05-0083, "Uranium Nitride Fuel Pellet Development at BWXT for Space Applications," January 2006
- [5] MDO-729-05-0084, "Uranium Oxide Fuel Pellet Development at Nuclear Fuel Services for Space Applications," January 2006
- [6] SPP-SRE-0008, "Response to Request for Transient Specifications for Furnace Design," June 2005

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Enclosure 4
to

MDO-723-0015/ B-MT(EDT)S-028

Compliance Assessment for MICE-3B UO₂ Capsule

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**Reviewed by:
M. Johnson**

This enclosure addresses the compliance assessment of the MICE-3B Tier 5 capsule for Space Materials specimens.

Compliance

A complete listing of the weights of materials for all specimens, specimen assemblies, instrumentation and test train hardware was not available when this assessment was performed. The capsules had not been designed and some materials had not been selected. There is a preliminary materials list below that was used in the test prospectus compliance assessment. Final materials information will be input to the Materials Declaration Listing to support reassessment of compliance issues prior to irradiation. Post-irradiation examination plans were for shipment from the Reactor Test Complex to a non-NR Program facility. The non-NR Program facility was also to be responsible for disposal of the test materials and hardware. The MICE-3B Space Materials were to be handled at NR Program facilities as a backup.

Materials List

Specimens – UO_2 (5.34% ^{235}U).

Capsules, Instrumentation, other Hardware - Graphite, Haynes 230, Inconel 600, LCAC Molybdenum, TZM Molybdenum, Tungsten-3% Rhenium, ZrO_2 - Magnesia stabilized. (Table 1)

Osmium-191

Several years ago a charcoal sample from ECF Stack #2 was counted by NRF Chemistry and determined to contain about 300 pCi of ^{191}Os . Since the ^{191}Os was observed on the charcoal filter, the detected isotope had to be in gaseous form. The most likely chemical form of the detected isotope was probably OsO_4 that melts at 40 °C and boils at 130 °C.

This isotope can be produced by activation of another material that would lead to the production of ^{191}Os . The primary materials that could lead to the production of ^{191}Os are rhenium and tungsten. Preliminary calculations using ORIGEN have shown that millicuries of ^{191}Os could be produced per gram of rhenium by the chain below.

Tungsten could also be a source of ^{191}Os , if it were present. If tungsten is irradiated in the MICE 3B train significant quantities of ^{191}Os would be produced per gram of tungsten. Preliminary calculations using ORIGEN have shown that millicuries of ^{191}Os could be produced per gram of tungsten using the chain below.

Other possible, but not expected sources of ^{191}Os are: $^{190}\text{Os}(n, \gamma)^{191}\text{Os}$; $^{191}\text{Ir}(n, p)^{191}\text{Os}$; $^{194}\text{Pt}(n, \alpha)^{191}\text{Os}$.

In the presence of oxygen from trace moisture or air in the flowing gas stream, some of the metallic osmium could be converted to OsO_4 which could be transported through the flowing gas stream to the piping system outside the reactor or to the Advanced Test Reactor stack. Although this may be unlikely it should be recognized as a possibility, since it was generated in samples from the pressurized water loops.

Radiological Controls

Although many of the MICE-3B Space materials of construction are considered to be standard, the irradiated material may present unique personnel exposure concerns based on the relatively high cobalt content (0.1646%) in Haynes 230, and the expected generation of significant transuranic radioactivity from the long term irradiation of low-enrichment fuel. Thus, a preliminary radiological control system evaluation has been performed to support insertion of MICE-3B Space materials into the ATR. The results of the evaluation for decay times of 30 and 730 days, documented in RCO-05-101, are that the MFP control system is sufficiently conservative for capsule cutting operations and handling of capsule internals at the Expanded Core Facility.

This evaluation will be updated as required prior to insertion to include as-built materials information and again prior to final removal from the reactor to reflect actual irradiation parameters. Evaluation data will be made available to NR Program activities that may be selected to conduct either capsule disassembly or specimen examination, in sufficient time to establish the proper radiological controls for these tasks.

ESH

The Space materials in MICE-3B and the handling and examination of the specimens are very similar to those in previous MICE test proposals. The new material is Haynes 230. Haynes 230 is a corrosion resistant, high temperature material consisting of elements common to the NR program.

NESHAPS

There may be some NESHAPS concerns due to the greater than normal activity of osmium, rhenium and tungsten isotopes. The Haynes 230 material used to can the fuel contains 13.86% tungsten. See the discussion of Osmium-191 above. No other NESHAPS concerns are expected because the new materials consist of common elements that will activate to radionuclides that are common in the hot cells. No concerns have been identified with previously used materials. The total number of curies is likely to be accommodated. However, NESHAPS assessments will be performed for each NR program facility based on as-built materials data and actual irradiation histories prior to introduction to that facility.

Transportation

Many of the capsules and other non-specimen hardware are not defined at this time. Post-irradiation examination plans are not expected to involve shipments from NR program facilities. Therefore, potential concerns for shipment of the irradiated materials must be addressed later.

There are some concerns for shipment of the unirradiated MICE-3B Space materials components because they contain both fissile and moderating materials. Shipment may be made under the General License Limits of Type A packages if the following conditions are met:

The MICE 3B Space Materials Test Program will irradiate 0.86 grams of ^{235}U (18.24 gm of UO_2) in the Tier 5 capsule. The MICE-3B train, which includes four other test capsules, may contain greater than 15 grams of ^{235}U . If the as-built train exceeds 15 grams ^{235}U or if the mass of the shipping container (excluding the mass of any beryllium, graphite, and hydrogenous material enriched in deuterium) is less than 200 grams per gram of ^{235}U , the shipping container must be limited to 60 grams ^{235}U (38 grams ^{235}U if the fissile material is mixed with moderating substances having an average hydrogen density greater than water) and the amount of beryllium, graphite, and hydrogenous material enriched in deuterium must be limited to 500 grams. For the shipment of the MICE train assembly from NRF to TRA, a transportation plan with government approval would be needed to make the shipment.

If the above limits are exceeded, then a Certificate of Compliance Package or DOT Specification Package authorized for fissile material must be used.

Waste Management

Low enrichment fuel and Haynes 230 are new materials to the test program. Waste from the low enrichment fuel will be managed with the same methods as higher enrichment fuel since the wastes are unlikely to be segregated. Haynes 230 is a corrosion-resistant metallic material containing chromium as an alloying material. It is not expected to leach heavy metals in excess of limits for hazardous wastes.

The capsule and instrumentation materials will probably be similar to previously evaluated MICE materials. The only materials that exceed a criterion for hazardous material contain chromium as an alloying material. A number of the capsule and instrumentation materials will become GTCC waste when irradiation is completed.

Criticality

The MICE-3B Tier 5 Space Materials capsule is not expected to include beryllia or other neutron moderators. The irradiated fuel material will contain a significant mass of plutonium relative to the mass of uranium-235. This fact may need to be considered for nuclear materials accountability.

Long Term Fuel Storage

The MICE-3B Space Materials Test Program will irradiate 0.86 grams of ^{235}U and 0.10 in³ total volume of fuel specimens. Plans have not been formulated for post-irradiation examination or long term storage. The materials are not expected to be examined or returned to NRF following examination. The amount of additional irradiated waste ultimately requiring disposal represents less than 0.01% of the current inventory of similar waste from ATR test programs (not including expended core waste). If the materials were returned to NRF, this additional amount of material would have an insignificant impact on the total waste requiring disposal, and it would be very unlikely that an additional disposal canister would be needed. The handling and storage of the irradiated test articles

from this program in the NRF water pits will be done in accordance with approved procedures. The only additional NRF effort specific to MICE-3B Space specimen disposal would be unloading a small portion of a MIN container with scrap waste and transferring to the water pits and loading it into a disposal canister for shipment when needed. 100 NRF man-hours is an order of magnitude estimate for this MICE-3B Space materials specific work. The shipping and disposal in the geologic repository of these and numerous other miscellaneous scrap fuel specimens will also be performed with approved procedures. Disposal and handling of the specimens in the other capsules can be performed using the same procedures and containers as other miscellaneous fuel specimens.

Table 1: Preliminary Estimate for the ATR Materials Declaration List

Material Name	Composition	Approximate Weight
Haynes 230†	Ni-21.88Cr-13.86W-1.25Mo-1.38Fe-0.1646Co-0.5Mn-0.39Si-0.4Al-0.11C-0.012La-0.0035B	240 gm
Molybdenum TZM	Mo - 0.02W - 0.1Zr - 0.5Ti	3 gm
Molybdenum LCAC	Mo	55 gm
Graphite	C	190 gm
Tungsten / Rhenium	97 W – 3 Re	15 gm
Zirconia	ZrO ₂ -3MgO	20 gm
Inco 600	76Ni-15.5Cr-0.2Si-0.3Mn-0.2Al-0.08C-8Fe-0.2Ti	900 gm
²³⁵ U Isotopic		0.86 gm
UO ₂		18.24 gm

† Composition from the heat being used.

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Enclosure 5
to

MDO-723-0015/ B-MT(EDT)S-028

MICE-3B Capsule and Fuel-Can Design Report

MG Petranka

Reviewed by :

**MJ Kania
DV Miller
MC Chen
CM Regan**

Acronym Listing

ATR	Advanced Test Reactor
BEA	Batelle Energy Alliance
BPVC	ASME Boiler and Pressure Vessel Code
EFPD	Effective Full Power Days
GTAW	Gas Tungsten Arc Weld
LANL	Los Alamos National Laboratory
MEWP	Maximum Expected Working Pressure
MICE	Multiple Irradiation Capsule Experiment
NRPCT	Naval Reactors Prime Contract Team
PCT	Peak Central Temperature
TZM	0.5 w/o Titanium – 0.1 w/o Zirconium - Molybdenum alloy
UN	Uranium Nitride
w/o	Weight Percent

I. Background

The MICE test platform is used to irradiate high temperature fuel and structural specimens in the Advanced Test Reactor (ATR). The MICE test platform consists of four test locations, two Prime (M-2 and M-4) and two Secondary (M-1 and M-3), in the northeast lobe of the ATR (Figure 1). Each Prime test train consists of five welded capsules with instrumentation lines (three thermocouples and two gas tubes) vacuum brazed to each capsule head. Each Secondary test train consists of five smaller diameter welded capsules without instrumentation. Capsules in a test train are numbered sequentially from "Tier 1" at the bottom to "Tier 5" at the top.

The test trains are discussed in detail in References [a] thru [c]. The test train which was to be used to test space fuel, designated NE-M-2-103 (MICE-3B), is scheduled for insertion into the ATR in March 2006. Prior to cancellation of Naval Reactors contract to deliver a reactor for Project Prometheus, the NRPCT planned to irradiate two candidate Prometheus-1 fuel materials in the NE-M-2-103 train. Specifically, the Tier 4 capsule would house six uranium nitride (UN) specimens fabricated by the Los Alamos National Laboratory (LANL) and the Tier 5 capsule would house eighteen uranium oxide (UO₂) specimens. This enclosure describes in detail these two capsule designs.

II. MICE-3B Tier 4 Canned UN Capsule

A. General Description of Capsule

This section provides an overview of the MICE-3B Tier 4 canned UN capsule. Each unique component inside the capsule is described in more detail in Section II.D. Sketch SK-MGP2197 is the top level engineering sketch for the capsule. A cross section of the capsule assembly is illustrated in Figure 2. Capsule internals are housed inside an Inconel 600 capsule body. Electron beam welded to the capsule body is a head assembly with five Inconel 600 strain reliefs, two 304L stainless steel gas lines, and three standard Type K thermocouples vacuum brazed to an Inconel 600 head. A helium-neon gas mixture flowing at 60 sccm through the gas lines is used to adjust the specimen temperature in-situ by regulating the ratio of helium to neon in the mixture. Likewise, the electron beam weld design is identical to the MICE-3A capsule

weld qualified to Section IX of the ASME Boiler and Pressure Vessel Code (BPVC), Reference [d]. The Nitronic 60 capsule spacer centers the capsule body in the MICE strongback.

Inside the capsule were six laser welded Haynes 230 can assemblies. With each assembly was located a single cylindrical 0.255-inch (6.47 mm) diameter by 0.118-inch (2.99 mm) thick uranium nitride fuel specimen fabricated by LANL. The can assemblies were numbered sequentially from "Level 1" at the bottom of the capsule to "Level 6" at the top. The Level 3 can assembly is illustrated in Figure 3. In each can assembly, the UN specimen was sandwiched between a tungsten pedestal and a tungsten cap. A tungsten - 3% rhenium compression spring was seated on radial fins integral to the tungsten cap. The spring accommodates component thermal expansion while maintaining good thermal contact between the specimen and tungsten pedestal and cap. Assembly of the UN specimen, tungsten pedestal, tungsten cap and tungsten-rhenium spring into the Haynes 230 can was to be performed by LANL inside of a glovebox filled with ultra high purity helium. Autogenous laser welding of the Haynes 230 lid to the Haynes 230 can was to be performed by LANL inside the glovebox. After inspection of each can per Sketch SK-MGP2198, each Haynes 230 can assembly was to be shipped to Bettis Atomic Power Laboratory for assembly into the MICE-3B Tier 4 capsule. At Bettis, each can assembly was to be installed into a POCO graphite holder centered in the capsule body with two 316 stainless steel centering disks. A slot in the Levels 2, 4 and 6 graphite holders (Figure 4) each house a standard Type K thermocouple vacuum brazed to the capsule head. Inconel 600 spacers at the top and bottom of the capsule minimize axial heat losses from the top and bottom can assemblies. An Alloy 286 wave spring identical to that used in the molybdenum structural capsule (SK-MGP2185, Group 1) was to be positioned at the top of the capsule to accommodate internal component thermal expansion during operation. Two external capsule shrouds fabricated from Zircaloy/hafnium sheet slide over the capsule body. The percentage of hafnium in each sheet was selected so that the fission power is approximately 500 W/cm^3 for the bottom three specimens (100% hafnium sheet) and 750 W/cm^3 (10% hafnium sheet) for the top three specimens.

B. Nuclear and Thermal Analyses

Nuclear analyses of the MICE-3B Tier 4 canned UN and Tier 5 canned UO_2 capsules consisted of the calculation of neutron fission rates using a NRPCT RCP01 model of the ATR with the northeast lobe operating at 18 MW. The non-fission (gamma) heat rates were derived from previous results that have been correlated for use in the MICE tests. Therefore, the nuclear analysis method was the same as that used for previous MICE insertions. See References [a] thru [c] for additional information on the nuclear analysis methodology.

Thermal analyses of the capsules were performed using ANSYS finite element models with fission and gamma heat generation rates formulated from the nuclear analyses. Thermal plots for the MICE-3B Tier 4 canned UN capsule are reproduced in Figures 5 thru 7 assuming an 80 vol% helium - 20 vol% neon gas mixture. Figure 5 is a thermal plot of the entire capsule assembly. Figure 6 shows the temperature distribution of the Level 4 can assembly. The remaining five internal can assemblies exhibit similar thermal profiles. Figure 7 is a representative temperature plot for the uranium nitride specimens.

C. UN Specimen Operating Conditions

Specimen fission power density and peak central temperature (PCT) targets for each UN specimen were established to simulate the performance response in the Prometheus-1 reactor. Predicted fission power density and PCT values for each specimen were obtained from the RCP01 nuclear and ANSYS thermal models assuming an 80 vol% helium - 20 vol% neon gas

mixture. A comparison of the target and predicted values is highlighted in Table 1. The planned test duration was 500 effective full power days (EFPD), or 12000 hours. Note that gamma heating of the UN specimens contributes an additional 80-100 W/cm³ to the overall specimen heat generation rate. For example, the Tier 4 Level 1 uranium nitride specimen has a predicted fission power density and gamma heat rate of 545 W/cm³ and 100 W/cm³, respectively, for a total heat generation rate of 645 W/cm³.

Table 1: Comparison of Target and Predicted Fission Power Density and PCT Values (Tier 4)

Level	Target Specimen Fission Power Density (W/cm ³)	Predicted Specimen Fission Power Density (W/cm ³)	Target Specimen PCT (°F / K)	Predicted Specimen PCT (°F / K)
1	500	545	2196 / 1475	2206 / 1480.8
2	500	532	2196 / 1475	2204 / 1479.7
3	500	517	2196 / 1475	2208 / 1481.9
4	750	836	2196 / 1475	2186 / 1469.7
5	750	806	2196 / 1475	2189 / 1471.3
6	750	773	2196 / 1475	2186 / 1469.7

D. Specific Details about Unique MICE-3B Tier 4 Canned UN Capsule Components

This section provides detailed information about the purpose and manufacturing difficulties, if applicable, specific to each unique UN capsule component.

Haynes 230 Can (SK-MGP2189, Items 2 and 3) and Haynes 230 Lid (SK-MGP2189, Item 1)

The Haynes 230 can and lid served three purposes. First, uranium nitride will dissociate unless a small (less than eight torr) overpressure of nitrogen is in equilibrium with the fuel. The Haynes 230 can is a hermetic vessel after welding the lid shut, so dissociation of an insignificant mass of the UN specimen inside the can provides the nitrogen overpressure, preventing further dissociation during operation. Note that the MICE sweep gas system uses only helium and neon, not nitrogen, necessitating the requirement that the specimens be canned. In other words, uncanned UN specimens would completely dissociate into elemental uranium as the nitrogen released from a specimen via dissociation is swept out of the capsule. Secondly, if the specimens were not canned, steady state fission gas emission from the specimens would violate emission thresholds imposed by the operating contractor for the ATR, Battelle Energy Alliance (BEA). The fission gases products are collected inside each hermetic Haynes 230 can during operation, so no BEA emission thresholds are exceeded. Thirdly, 60 sccm of helium-neon gas flows continuously through the MICE capsules via the two gas lines brazed into the capsule head. This gas stream contains approximately 1-10 ppm of impurities such as water vapor and oxygen. The Haynes 230 can prevents chemical interaction between the UN specimens and impurities within the gas stream impurities.

Haynes 230 (UNS N06230) is a solid-solution strengthened nickel-base alloy. It was selected as the can and lid material primarily because of its high creep strength and weldability. During operation in the ATR, the can and lid are internally pressurized, and therefore primary stresses are generated, from two sources. First, helium inside the can would be at atmospheric pressure (approximately 12.5 psia during filling at LANL) when the can is at room temperature. However, during operation, the average gas temperature inside the can increases to 1500-2000°F (1088 K – 1366 K), resulting in a corresponding increase in pressure to approximately 60 psia. Second, fission gas products released from the uranium nitride specimen during the 12000 test

hours account for an additional 10 psia maximum. Therefore, the maximum expected working pressure (MEWP) inside the can at temperature is 70 psia. The design pressure of the Haynes 230 can assembly is 85 psia, which provides a 15 psi margin to the MEWP.

The design temperature of the Haynes 230 can and lid is 1650°F (1172 K), and was selected because Section VIII, Division 1 of the BPVC limits operation of pressure vessels fabricated from Haynes 230 to this temperature. Specifically, Code Case 2063-3 specifies a maximum allowable stress of 2.0 ksi at 1650°F. Above this temperature, the creep strength of Haynes 230 is sufficiently low that fabrication of pressure vessels from the material is impractical. The capsules were designed to limit the Haynes 230 pressure boundary temperature to approximately 1550°F (1116 K), providing a 100°F (55.5 K) margin to the BPVC temperature limit. This margin is adequate to offset uncertainties in the temperature measurement that may result in the pressure boundary operating at a higher temperature than predicted. Figure 8 shows the resulting von Mises stresses in the Haynes 230 pressure boundary when subjected to the design pressure of 85 psi. Figure 8 demonstrates that even the peak stresses are less than the maximum allowable stress of 2 ksi at 1650°F (1172 K) dictated by Code Case 2063-3.

Note that technically the pressure boundary enveloping the UN specimen does not have to adhere to the rules of the ASME BPVC, either Section VIII or Section III. BEA only required that the Inconel 600 capsule body and head assembly meet the requirements of the BPVC Section III Class 3, since this pressure boundary prevents water from leaking into the capsule and out of the reactor. Nonetheless, it is good engineering practice to use the maximum allowable temperatures and stresses dictated by the Code, if available, and the accompanying rules of construction, due to their built-in conservatism and the Code's proven history.

Although Haynes 230 was deemed the optimal material for this application given its superior creep strength among non-refractory metal alloys, weldability, published allowable stresses at elevated temperature and product availability, Haynes 230 does have its drawbacks. First, like most ferrous and nickel-based alloys, it has low thermal conductivity, resulting in high thermal stresses during operation in the ATR. Figure 9 illustrates the thermal stresses in the Level 4 can assembly during normal operations. A plastic analysis of the can shows that these stresses, which exceed 20 ksi, cause creep relaxation. Creep damage coupled with fatigue damage due to reactor start-ups and shutdowns can cause cracking of the pressure boundary via a mechanism called creep-fatigue failure. However, it is difficult to predict the onset of this failure mode given the limited irradiated creep-fatigue data for Haynes 230 in the open literature and the high level of uncertainty associated with the calculation methodology. The low number of reactor start-ups and shutdowns, less than twenty, in this application would reduce the likelihood of such a failure, but it was still a risk that must be considered.

The second drawback of Haynes 230 is the uncertainty in the temperature measurement that results from its design temperature limit of 1650°F (1172 K). Three thermocouples in the canned UN capsule are positioned in slots in the Levels 2, 4 and 6 graphite holders (Figure 4). Each thermocouple during operation measures approximately 1250°F (949 K), approximately 50°F (28 K) lower than the outside surface temperature of the Haynes 230 can (Figure 6). However, the average UN specimen temperature [approximately 2180°F (1466 K)] was the critical parameter needed since fuel swelling and fission gas release are significantly affected by specimen temperature. It was not feasible, given the test schedule, to develop a technique to penetrate a thermocouple through the Haynes 230 pressure boundary to directly measure the specimen temperature. Instead, the specimen temperature must be estimated from correlations generated by the ANSYS thermal model [e.g., ANSYS predicts that the specimens operate at approximately 2200°F (1477 K) PCT when the thermocouples read 1250°F (949 K)]. Uncertainty in the specimen temperature correlation increases with the temperature difference

between the fuel specimen and thermocouple. Note that this drawback is inherent to any iron or nickel-based can material, since all would have metal design temperatures, and therefore thermocouple temperatures, significantly lower than the fuel specimen temperatures. Ideally, the can would be fabricated from a high thermal conductivity refractory metal that could operate satisfactorily at a temperature just below the specimen temperature. Then, with the thermocouple installed adjacent to the refractory metal can, the fuel specimen temperature uncertainty would be significantly reduced [i.e., because the thermocouple would be operating at a temperature 200°F (111 K) less than the average specimen temperature instead of the current 930°F (499 K)]. However, to meet the original MICE-3B insertion scheduled for January 2006, it was judged that insufficient time was available to develop and qualify a refractory metal can to lid weld. Therefore, to maximize the likelihood of the can remaining hermetic for 12000 hours in-pile, Haynes 230 was selected.

Finally, the third drawback of Haynes 230 is the compatibility of Haynes 230 with fission products such as cesium, rubidium and tellurium. It is possible, although not likely, that fission products could corrode the Haynes 230 can, as discussed in Attachment A. Out-of-pile testing of the chemical compatibility of Haynes 230 with cesium, rubidium and tellurium is necessary to quantify the corrosion rate.

Tungsten Pedestal (SK-MGP2191, Items 1-3) and Tungsten Cap (SK-MGP2191, Items 4-5)

The tungsten pedestal acts as a thermal resistor meaning that the cross section of each pedestal is tailored so that the specimen operates at an approximately 2200°F (1477 K) PCT even though the Haynes 230 can temperature is limited to 1550°F (1116 K). Examining the pedestal on Sketch SK-MGP2191 in more detail, the UN specimen sits on an undercut polished surface (flatness of 0.0005 inches (12 µm)) designed to minimize the thermal contact resistance between the specimen and tungsten pedestal. For the same reason, the underside of the tungsten cap was polished to a 0.0005 inch (12 µm) flatness tolerance. Fission and gamma heat generated by the UN specimen would conduct axially from the specimen's two flat faces into the tungsten cap and base. Some of the heat transferred to the tungsten cap is radiated to the colder Haynes 230 can and lid, but a significant percentage conducts through the tungsten cap's skirt and back into the pedestal. This combined heat load then conducts axially through the 0.030-inch (0.76 mm) thick resistor section of the tungsten pedestal, and spreads out into the pedestal's flange and flows into the bottom of the Haynes 230 can. The pedestal's resistor section was purposely made thin so that a significant axial temperature drop, approximately 400°F (222 K), occurs over the length of the pedestal (Figure 6). This feature is what maintains the UN specimen at 2200°F (1477 K) PCT even though the bottom of the pedestal is seated on the 1550°F (1116 K) can. In addition, the length of the pedestal's resistor section varies depending on which can assembly the pedestal is used in, as listed below:

Table 2: Length of the Resistor Section in the Levels 1 thru 6 Tungsten Pedestals (Tier 4)

Level of Can	Tungsten Pedestal Resistor Length (inches / mm)
1	0.450 / 11.43
2	0.539 / 13.69
3	0.539 / 13.69
4	0.335 / 8.51
5	0.335 / 8.51
6	0.450 / 11.43

An examination of Fourier's law shows why the tungsten pedestal was designed with a thin, variable length resistor section. Fourier's law for this application states that:

$$q = -kA \frac{dT}{dx} \approx -k\pi Dt \frac{\Delta T}{L}$$

where

q = heat conducted axially through resistor section (Btu/hr, W)

k = thermal conductivity of tungsten (Btu/hr/in/R, W/mK)

D = outside diameter of pedestal resistor section (in, m)

t = thickness of pedestal resistor section (in, m)

ΔT = temperature drop over length of pedestal resistor section (R, K)

L = length of pedestal resistor section (in, m)

The resistor section is thin (0.030 inches, 0.76mm) so that the temperature drop over the resistor length is relatively large (400°F, 222K). In addition, the length of each pedestal is varied to maintain a relatively constant temperature drop over the pedestal length even though the heat generation of each specimen is not constant (i.e., L is varied to maintain a constant $\Delta T/q$ ratio for all six specimens even though q varies by level). For example, the Level 4 and 5 can assemblies have the shortest pedestal lengths to compensate for their higher fuel heat generation rates (Table 1). The tungsten pedestal has a polished flange on the bottom to minimize the thermal contact resistance between the pedestal and Haynes 230 can, since contact resistance decreases with increased contact area.

The tungsten cap has two radial fins on which the high-temperature spring sits. Radial fins were utilized as a spring seating surface to minimize heat conduction losses from the specimen, through the radial fins and directly into the spring or the wall of the Haynes 230 can.

Tungsten was selected as the pedestal and cap material for three main reasons. Most importantly, tungsten will not react chemically with the UN specimen at the proposed operating temperatures. Second, tungsten has sufficient creep strength at temperature that it would not buckle the pedestal resistor section nor the cap's radial fins due to the spring load. Finally, the thermal conductivity of tungsten makes it an appropriate thermal resistor material. Re-examining Fourier's law, if a high thermal conductivity material such as copper was utilized as a pedestal material, the thickness of the resistor section would be unacceptably thin or the length unacceptably long to achieve a 400°F (222 K) temperature drop. Likewise, a low conductivity pedestal material such as zirconia would require an unacceptable decrease in length.

Tungsten is brittle at room temperature, so it is very difficult to machine. A machine shop specializing in refractory metal fabrication was used. Contact information for this vendor is listed below:

*Sylhan LLC
210 Rodeo Dr.
Edgewood, NY 11717
Tel: 631-243-6600*

Tungsten – 3% Rhenium High Temperature Springs (SK-MGP2187, Item 1)

The high temperature spring served two functions. First, it ensures that good thermal contact would be maintained between the UN specimen and the tungsten pedestal and cap while permitting thermal expansion of the components. Second, the spring would apply a preload to the specimen so that it would be constrained during shipping, thereby preventing in transit damage. The spring operates at 1500-1600°F (1088K-1144 K) (Figure 6) and was designed to provide a 0.7 pound load at a working height of 0.42 inches (10.6 mm) (0.10 inches (2.5 mm) of deflection). In addition, the spring was designed for a 0.012-inch (0.3 mm) nominal diametral clearance with the inside bore of the Haynes 230 can. Maintaining a tight clearance between these two components reduces the temperature of the spring, thereby increasing its allowable stress. Figure 10 shows a typical load-deflection curve for this spring design. For a given free height, empirical load values are within 20% of loads predicted by the fundamental spring equation. The springs were designed to have relatively low stress intensity (30 ksi) at the working height. Conventional spring materials such as music wire, beryllium copper, stainless steel and Alloy X750 have insufficient yield and creep strength at 1600°F (1144 K). Springs fabricated from these materials would thermally relax during operation, so that the spring load would be negligible. Thus, the spring was fabricated from the refractory metal alloy tungsten – 3% rhenium, which has an estimated 12000 hour creep rupture strength of 64 ksi at 1650°F (1171 K).

Conventional spring manufacturers were unable to fabricate the spring because tungsten-rhenium must be wound at an elevated temperature. Instead, a company that specializes in winding high temperature heater elements was contracted to fabricate the springs. Contact information for this vendor and the vendor supplying the tungsten – 3% rhenium wire is listed below:

*Spring Fabrication Vendor:
Union City Filament Corp.
1039A Hoyt Avenue
P.O. Box 777
Ridgefield, NJ 07657
Tel: 201-945-3366*

*Wire Supplier:
H. Cross Company
363 Park Avenue
Weehawken, NJ 07086
Tel: 201-863-1134*

Graphite Holders (Sketch SK-MGP2192, Items 1-6)

The graphite holders positioned around each Haynes 230 can assembly serve four purposes. First, the temperature of each Haynes 230 can is limited to 1550°F (1116 K), 100°F (55 K) lower than the design temperature, by varying the outside diameter of each graphite holder, as listed in Table 3. The gas gap is the radial clearance between the outside diameter of the graphite holder and inside diameter of the capsule body.

Table 3: Comparison of Levels 1 thru 6 Gas Gaps and Corresponding Can Temperatures (Tier 4)

Level of Can	Graphite Holder Diameter (in / mm)	Helium/Neon Gas Gap (in)	Maximum Haynes 230 Can Temperature (°F / K)
1	1.150 / 29.210	0.070 / 1.778	1535 / 1108.0
2	1.224 / 31.089	0.033 / 0.838	1536 / 1108.6
3	1.150 / 29.210	0.070 / 1.778	1559 / 1121.3
4	1.246 / 31.648	0.022 / 0.559	1500 / 1088.6
5	1.200 / 30.480	0.045 / 1.145	1507 / 1092.4
6	1.194 / 30.327	0.048 / 1.219	1482 / 1078.6

As the radial gap in a given level is increased, Fourier's law dictates that the graphite holder and Haynes 230 can temperature is increased. Therefore, each gas gap could be altered until the maximum temperature of all six Haynes 230 cans was approximately 1550°F (1116 K). This was important for two reasons. If a Haynes 230 pressure boundary operated at a significantly higher temperature than 1550°F (1116 K), its creep strength would degrade and the can could rupture during operation. Similarly, if a Haynes 230 can operate at a lower temperature, the uncertainty in fuel specimen temperature increases since the uncertainty is dependent on the difference between the specimen and the can temperatures. By optimizing the outside diameter of each graphite holder and the tungsten pedestal length, it was possible to meet the simultaneous requirement of a 2200°F (1477 K) PCT specimen and 1550°F (1116 K) maximum temperature Haynes 230 can on all six levels. Note, however, that this required significant iteration because modifying the dimensions in one level affects the temperatures on adjacent levels.

The second purpose of the graphite holders was to radiate heat to the inside diameter of the capsule body. Graphite has a high emissivity (approximately 0.9) so a considerable percentage of the heat transferred across the gas gap would be via radiation. The heat transferred across the gap via conduction would be correspondingly reduced, so the average gas gap increased. This was significant because component tolerances and thermal expansions can cause appreciable changes in predicted temperatures when they are a significant percentage of the nominal gas gap. For example, the Haynes 230 can temperature would be appreciably affected by the 0.002-inch (50.8 μ m) radial thermal expansion of the graphite holder if the nominal gas gap was only 0.010 inches (0.254 mm).

Thirdly, the graphite holders contact the flat base and lid of each Haynes 230 pressure boundary. Contact with the high thermal conductivity graphite tends to reduce the radial temperature distribution in the Haynes 230, thereby reducing thermal stresses, although they were still relatively high. Finally, as mentioned above, slots in the Levels 2, 4 and 6 graphite holders house the three thermocouples used to estimate the UN specimen temperatures.

Stainless Steel Centering Disks (SK-MGP2194, Items 1 and 2)

A top and bottom centering disk centers each graphite holder in the capsule body, thereby establishing a uniform gas gap around each holder. The disks are designed with fins to minimize heat loss from the graphite holder, through the centering disk, and into the capsule body.

Inconel 600 Capsule Body (SK-MGP2196, Item 1) and Head Assembly (SK-MGP2217, Group 1)

After electron beam welding the head assembly to the capsule body, these two components form the main pressure boundary to the ATR coolant. BEA requires that these components meet the requirements of Section III, Class 3 of the BPVC. Design and service loadings for the capsule pressure boundary are equivalent to that for the MICE-3A capsules, which are discussed in reference (c). BPVC calculations for the pressure boundary are identical to that for the capsule described in reference (a). The three Type K thermocouples brazed to the head have crushed alumina insulation with a 316L stainless steel sheath. These thermocouples, which would operate at approximately 1250°F (949 K), are the same type of thermocouples used in standard loop tests in the ATR.

Capsule Shrouds (6266E46, Item 2 and Item 4)

The hafnium percentage of the two capsule shrouds (i.e., 100% hafnium and 10% hafnium in the bottom and top shrouds, respectively) was selected to meet the target of three specimens each operating at 500 W/cm³ and 750 W/cm³. The bottom three specimens were selected to operate at the lower fission heating rate of 500 W/cm³ to offset the higher gamma heating rate that results from the closer position of these specimens to the core centerline. Similarly, the top three specimens were shrouded to provide the higher 750 W/cm³ fission heating rate since their gamma heating rate was lower. This technique was used in an effort to keep the total heat generated (fission plus gamma) in the top three and bottom three can assemblies identical so that the range of the six gas gap values was not extreme.

D. Specimen Temperature Uncertainty in MICE-3B Tier 4 Canned UN Capsule

The temperature of the uranium nitride specimens could not be directly measured. Instead, an ANSYS finite element model would be used to correlate the specimen temperature from the temperature recorded by three thermocouples embedded in the graphite holders. As a result, there would be significant uncertainty in the specimen temperature. The nontrivial sources of uncertainty are as follows:

- Thermal contact resistance between the specimen and tungsten pedestal and cap. The thermal contact resistance is modeled in ANSYS as a uniform 0.001-inch (25.4 μm) helium gas gap between the specimen and tungsten components. Modeled as such, ANSYS predicted that each specimen would generate sufficient heat that an approximately 125°F (69 K) temperature drop would occur across the gap. High uncertainties are inherent to contact resistances since they depend on the microscopic contact area. The uncertainty would increase considerably if the specimen's flat surfaces became bowed due to swelling, so that there would be only point contact between the specimen and tungsten. The contact resistance uncertainty is the single largest contributor to the overall measurement uncertainty. Assuming the specimen remained a right circular cylinder during operation, an upper bound on this uncertainty would be 100%, or approximately 125°F (69 K).
- Thermal contact resistance between the tungsten pedestal flange and the Haynes 230 can. Modeled as a 0.001-inch (25.4 μm) helium gap, ANSYS predicts a 25°F (14 K) temperature drop across this interface. An upper bound on this uncertainty is 100%, or approximately 25°F (14 K).
- Thermal conductivity of irradiated tungsten and Haynes 230. The uncertainty in the unirradiated thermal conductivity value of pure tungsten and Haynes 230 should be fairly small. It is unclear, however, what the effects of irradiation are on these values, although it is expected to be less than a 15% change.

- Emissivity of tungsten – The uncertainty in the emissivity of the tungsten was estimated at 15%.
- Fission and gamma heating rates of materials inside capsule – The uncertainty in fission and gamma heating rates is estimated at 15%. Unlike the other contributors to the overall temperature uncertainty, an empirical fission or gamma heating rate that does not match that used in the ANSYS model would be detected by the thermocouples. For example, if the three Type K thermocouples record a 1200°F (922 K) temperature and the ANSYS model predicts they should read 1250°F (949 K), it can be concluded that the actual Tier 4 gamma and fission heating rates are lower than predicted.

It is difficult to assign a realistic overall specimen temperature uncertainty to the Tier 4 canned UN design given the limited knowledge about each contributor. When the uncertainties quantified above are incorporated into the ANSYS model, the specimen temperature uncertainty is approximately +250°F/-150°F (+139 K / -83 K). The uncertainty is lower on the negative side because the assumed contact resistance values cannot be appreciably smaller unless components diffusion bond, which is improbable. This overall temperature uncertainty would be significantly reduced if the can pressure boundary had a higher design temperature limit, closer to that of the specimen.

E. Impracticality of Radial Heat Transfer from UN Specimen

All Project Prometheus core concepts exhibit radial heat transfer through the fuel specimens into a flowing coolant. To be prototypic, heat transfer from the MICE UN specimens should also be predominantly radial. However, for reasons discussed below, this was not practical. Instead, axial conduction from the specimen's two flat surfaces would be the dominant heat transfer mechanism, resulting in planar isotherms (surfaces of constant temperature) instead of the more prototypic concentric isotherms.

Heat removal from the UN specimens in the MICE capsule is accomplished by conduction and radiation only. In addition, the total specimen heat generation rate (fission plus gamma) would be much higher than prototypic (610-925 W/cm³ versus 30-50 W/cm³) to accelerate depletion. As a result, radial heat transfer through a gas gap results in very high temperature drops through the gap [approximately 90-140°F (50-78 K) per 0.001 inch (25.4 μm) of helium]. This phenomenon makes radial heat transfer from the UN specimen impractical.

For example, consider a conceptual design that consists of a UN specimen operating at 750 W/cm³ centered inside a 1550°F (1116 K) Haynes 230 can backfilled with pure helium. Heat generated by the specimen would be transferred radially through the specimen, across a 0.0075-inch (0.19 mm) radial helium gap to the inside diameter of the can. If perfectly centered in the can, the specimen surface temperature would be approximately 2300°F (1533 K), a 100°F (55 K) temperature drop for each radial mil (25 μm) of gas gap. The 0.015-inch (0.381 mm) diametral clearance is required to permit in-pile swelling of the specimen. However, consider the following uncertainties. First, if the specimen is not centered in the can, and is instead offset in the 0.0075-inch (0.19 mm) radial gap, the average specimen temperature increases by more than 200°F (111 K). Second, the specimen will swell in-pile, which reduces the radial clearance between the specimen and can. If the specimen swells 0.010-inches (0.254 mm) diametral, for example, the nominal radial gap will decrease from 0.0075-inches (0.19 mm) to 0.0025-inches (63.5 μm), resulting in an approximately 500°F (278 K) decrease in specimen temperature. Finally, fission products released from the specimen into the gas gap including xenon, krypton, and possibly cesium and rubidium vapors that all have thermal conductivities 15-45 times lower

than helium. Over the 12000 hour life of the test, this can cause a 30% reduction in the effective thermal conductivity of the gas gap, which increases the specimen temperature by approximately 300°F (162 K). With the current design utilizing axial heat transfer, the specimen temperature would essentially be independent of these three factors (i.e., specimen shift, specimen swelling and fission gas release). Note that the level of uncertainty associated with a radial heat transfer design is reduced proportionately with reductions in specimen heat generation. The Tier 5 UO₂ capsule design, which operates at much lower heat generation rates, takes advantage of this.

F. Summary of MICE-3B Tier 4 Canned UN Capsule Design

Table 1 demonstrates that the test designers were able to meet the target test conditions, albeit with significant uncertainty in the UN specimen temperature. Axial heat transfer from the specimens was necessitated in lieu of more prototypic radial heat transfer because the specimen heat generation rate is approximately 20 times greater than prototypic, to accelerate depletion. The UN specimens were required to be canned, to prevent specimen dissociation, release of fission products in excess of BEA emission thresholds and chemical interaction with impurities in the helium-neon gas stream. Haynes 230 was selected as the can material because of its relatively high creep strength among iron and nickel-based alloys, its published allowable stresses at elevated temperature, its weldability and product availability.

III. MICE-3B Tier 5 Canned UO₂ Capsule

A. General Description of Capsule

This section provides an overview of the MICE-3B Tier 5 canned UO₂ capsule. Each unique component inside the capsule is described in more detail in Section III.D. Sketch SK-MGP2216 is the top level engineering sketch for the capsule. A cross section of the capsule assembly is illustrated in Figure 11. The Inconel 600 capsule body, Nitronic 60 capsule spacer, capsule to head electron beam weld and helium-neon gas system are identical to those used for the Tier 4 canned UN capsule assembly. The Inconel 600 head assembly is also identical to that used in Tier 4, except for the axial termination point of the three Type K standard thermocouples. Inside the capsule are three gas tungsten arc welded Haynes 230 can assemblies, each accommodating six cylindrical 0.249-inch (6.32 mm) diameter by 0.118-inch (3.0mm) thick uranium oxide fuel specimens. The can assemblies are numbered sequentially from "Level 1" at the bottom of the capsule to "Level 3" at the top.

In the Level 2 and 3 can assemblies, which are identical except for the thickness of the molybdenum thermal resistor, the UO₂ specimens are unconstrained, so that the specimens can swell freely in-reactor, Figure 12. The Level 2 can assembly houses six large grain UO₂ specimens and the Level 3 assembly houses six small grain specimens. The six specimens were to be installed into a molybdenum thermal resistor, with magnesia-stabilized zirconia insulators surrounding the fuel stack to prevent axial thermal losses. A tungsten - 3% rhenium spring, identical to that used in the Tier 4 canned UN capsule, accommodates component thermal expansion. The spring would be seated on a flange integral to the molybdenum top hat, which helps center the upper zirconia insulators in the Haynes 230 can. Assembly of the UO₂ specimens, zirconia insulators, molybdenum thermal resistor, molybdenum top hat, and tungsten-rhenium spring into the Haynes 230 can would be performed inside of a glovebox backfilled with ultra high purity helium. The Haynes 230 lid was to be manually gas tungsten arc welded with ERNiCrWMo-1 filler metal to the Haynes 230 can inside a helium filled glovebox and the vent hole subsequently welded shut, Figure 13. Reference [e] describes the welding process. Each Haynes 230 can has an integral rib that houses a Type K thermocouple.

The six UO_2 specimens in the Level 1 can assembly are radially constrained by two molybdenum - 0.5 w/o titanium - 0.1 w/o zirconium (TZM) restraint tubes, so that the specimens cannot swell freely, Figure 14. Instead, the specimens swell into the TZM tubes, which act as fuel cladding, so that specimen-clad interactions and fuel viscosity can be studied. Two constraint tubes were to be used so that small and large grain specimens could be investigated independently (i.e., three small grain specimens were to be installed in the upper restraint tube and three large grain specimens in the lower restraint tube). To accommodate the Level 1 molybdenum thermal resistor has a larger diameter than the Level 2 and 3 resistors. Otherwise, components comprising the Level 1 can assembly are essentially identical to those in Levels 2 and 3.

After assembly, welding and inspection of the three Haynes 230 can assemblies, they were to be assembled into the MICE capsule. Each can assembly would be installed into a POCOTM graphite holder and subsequently a molybdenum or Inconel 600 thermal shield that would be centered in the capsule body with six small centering tabs integral to the shield. Inconel and zirconia spacers at the top and bottom of the capsule minimize axial heat losses from the top and bottom can assemblies. An Alloy 286 wave spring identical to that used in the Tier 4 canned UN capsule would be positioned at the top of the capsule to accommodate internal component thermal expansion during operation. Three external capsule shrouds fabricated from Zircaloy/hafnium sheet would then slide over the capsule body. The percentage of hafnium in each sheet was selected so that the average specimen fission power density in each set of six specimens would be approximately 150 W/cm^3 .

B. Nuclear and Thermal Analyses

Methods for performing nuclear and thermal analyses of the MICE-3B Tier 5 canned UO_2 capsule are discussed in Section II.B. Thermal plots for the Tier 5 capsule are reproduced in Figures 15 thru 18 assuming a 30 vol% helium - 70 vol% neon gas mixture. Figure 15 is a thermal plot of the entire capsule assembly. Figures 16 and 17 show the temperature distributions of the Level 2 and Level 1 can assemblies, respectively. Figure 18 displays the thermal profile of all eighteen specimens.

C. UO_2 Specimen Operating Conditions

A target fission power density of 150 W/cm^3 and PCT of 2420°F (1600 K) was established for all eighteen UO_2 specimens. Predicted fission power density and PCT values for each specimen were obtained from the RCP01 nuclear and ANSYS thermal models assuming a 30 vol% helium - 70 vol% neon gas mixture. A comparison of the target and predicted values is highlighted in Table 4. The planned test duration was 500 EFPD, or 12000 hours. Note that gamma heating of the UO_2 specimens contributes an additional $20\text{-}40 \text{ W/cm}^3$ to the overall specimen heat generation rate.

Table 4: Comparison of Target and Predicted Fission Power Density and PCT Values (Tier 5)

Level	Target Specimen Power Density (W/cm ³)	Predicted Specimen Power Density (W/cm ³)	Target Specimen PCT (°F / K)	Predicted Specimen PCT (°F / K)
1 – bottom 3	150	153-158	2420 / 1600	2385-2405 / 1580-1591
1 - top 3	150	147-152	2420 / 1600	2400-2422 / 1589-1601
2	150	154-169	2420 / 1600	2340-2427 / 1555-1604
3	150	137-161	2420 / 1600	2300-2404 / 1533-1591

D. Specific Details about Unique MICE-3B Tier 5 Canned UO₂ Capsule Components

This section provides detailed information about each unique UO₂ capsule component. Note that Section II.D will be referenced extensively in the paragraphs below, since many of the UO₂ capsule components are similar or identical to those used in the Tier 4 canned UN capsule.

Haynes 230 Can (SK-MGP2205, Item 2) and Haynes 230 Lid (SK-MGP2205, Item 1)

The Haynes 230 can and lid serve three purposes. First, the hermetic Haynes 230 pressure boundary retains specimen fission products, so BEA emission thresholds are not exceeded. Second, the Haynes 230 can prevents chemical interaction between the UO₂ specimens and the impurities in the flowing helium-neon gas. Finally, an integral rib on the outside diameter of each Haynes 230 can houses a standard Type K thermocouple, Figure 13, so that the temperature of each can would be measured directly. This method is preferable to that used in the Tier 4 UN capsule, where the Haynes 230 can temperature is extrapolated from a thermocouple embedded in the surrounding graphite holder. Ribs were not used on the Tier 4 Haynes 230 cans due to the increased complexity it would require in the fixture that secures the can during the automatic laser welding process. The cans in the Tier 5 UO₂ capsule are manually gas tungsten arc welded, so the rib could be accommodated in the fixturing. Note that, in the Tier 4 capsule, the Haynes 230 pressure boundary also prevents dissociation of the UN specimens, Section II.D. The UO₂ specimens, unlike the UN specimens, do not require a small overpressure of a constituent gas to prevent dissociation, so the Tier 5 Haynes 230 can does not serve this function.

The Tier 5 Haynes 230 pressure boundary has the same outside diameter, shell thickness and head thickness as the Tier 4 can and lid. The only differences in the two pressure boundary designs are the can lengths (Tier 5 cans are longer to accommodate multiple fuel specimens) and the bevel weld preps and vent hole in the Tier 5 design to accommodate a gas tungsten arc weld (GTAW). A vent hole was used to prevent helium pressure buildup inside the Tier 5 can during welding of the lid. The pressure buildup, which results from heat input during welding, can cause the molten weld metal in the seam to be ejected. Filling the small diameter (0.06-inch / 1.52 mm) vent hole with filler metal after completion of the lid weld does not result in significant internal pressure buildup due to the limited heat input of this operation. Note that LANL determined a vent hole was not required in the Tier 4 can design, presumably because the laser weld requires less heat input than the GTAW.

The design pressure and temperature of the Tiers 4 and 5 Haynes 230 pressure boundaries are identical, specifically 85 psia at 1650°F (1172 K). The MEWP inside the Tier 5 can at temperature is 70 psia (60 psia due to the increase in temperature of the helium fill gas and 10

psia due to specimen fission gas release). The Tier 5 canned UO_2 capsule was designed to limit the Haynes 230 pressure boundary temperature to approximately 1550°F (1116 K), 100°F (55.6 K) below the design temperature of 1650°F (1172 K). This margin is adequate to offset uncertainties in the temperature measurement that may result in the pressure boundary operating at a higher temperature than predicted. The advantages and disadvantages of a Haynes 230 pressure boundary material are discussed in detail in Section II.D. Note that the Tier 5 cans have similar primary and secondary stresses due to pressure and thermal loads, respectively, as the Tier 4 cans.

Molybdenum Thermal Resistor (SK-MGP2204, Items 3-5)

Each molybdenum-TZM thermal resistor was custom designed so that the UO_2 fuel stack operates at approximately 2420°F (1600 K) PCT even though the Haynes 230 can temperature is limited to 1550°F (1116 K). Its function is similar to that of the tungsten pedestal used in the Tier 4 canned UN capsule, except for one significant difference – the molybdenum thermal resistor was designed so that heat transfer from the specimens to the resistor is primarily radial (as compared to axial in the Tier 4 design, Section II.E). This is evident by comparing the isotherms in Figures 18 and 7. In Figure 18, the isotherms in the UO_2 fuel stacks are approximately vertical, except for the extreme edges, where some axial heat transfer to the zirconia insulators takes place. A vertical isotherm signifies radial heat transfer through the fuel into the molybdenum thermal resistor. In Figure 7, the isotherms in the UN fuel are approximately radial (horizontal), signifying axial heat transfer through the fuel into the tungsten pedestal and cap.

Examining the heat transfer path from the UO_2 specimens in more detail, heat generated by the specimens conducts radially through the TZM restraint tube (Level 1 only), across a 0.015-inch diametral gas gap and into the molybdenum thermal resistor. The 0.015-inch (0.381 mm) diametral gas gap was necessary to permit unconstrained swelling of the specimens in Levels 2 and 3, and unconstrained deformation of the TZM restraint tube in Level 1. The section of the molybdenum thermal resistor that surrounds the specimens is comparatively thick (approximately 0.10 inches / 2.54 mm) to promote axial heat transfer within the resistor, so that the six specimens all operate at approximately the same temperature. There are three heat transfer mechanisms between the thick section of the thermal resistor and the Haynes 230 can. Approximately 30% and 15% of the total heat generated by the specimen would be conducted and radiated, respectively, across the helium gas gap into the can. The remaining 55% of the heat would conduct axially through the thin section of the molybdenum thermal resistor, spread out into the resistor's flange and flows into the bottom of the Haynes 230 can. In the Tier 4 design, the length of each tungsten pedestal was tailored to achieve an approximately 400°F (222 K) drop over the length of the flange. Similarly, in the Tier 5 design, both the length and thickness of each molybdenum thermal resistor was tailored to achieve a $200\text{-}250^\circ\text{F}$ (111-139 K) drop over the resistor's reduced section, as listed below.

Table 5: Length and Thickness of the Moly Thermal Resistor's Reduced Section (Tier 5)

Level of Can	Length of Thermal Resistor (inches / mm)	Thickness of Thermal Resistor (inches / mm)
1	0.275 / 6.985	0.038 / 0.965
2	0.300 / 7.620	0.062 / 1.545
3	0.300 / 7.620	0.030 / 0.762

By tailoring the dimensions of each molybdenum thermal resistor, the fuel specimen stacks in each can assembly operate at approximately 2420°F (1600 K) PCT despite the variation in the total heat generated by the fuel specimens and other components in each can. Molybdenum was selected as the resistor material because of its chemical compatibility with uranium oxide, its thermal conductivity provided the appropriate temperature gradient over the length of the resistor and its relative ease of machining compared to tungsten.

Radial heat transfer from the fuel specimens was practical in the Tier 5 canned UO₂ capsule because the fission heat generation rate was limited to approximately 150 W/cm³, versus 500-750 W/cm³ in the Tier 4 design. Therefore, as noted in Section II.E, the uncertainties associated with specimen shift, specimen swelling and fission gas release are reduced as follows. Assume a Tier 5 UO₂ specimen operating at approximately 175 W/cm³ (150 W/cm³ fission heat rate plus 25 W/cm³ gamma heat rate) is perfectly centered within the 1915°F (1319 K) molybdenum thermal resistor. The resulting specimen PCT and average outer surface temperature are 2433°F (1607 K) and 2120°F (1433 K), respectively. The 0.0075-inch (0.1905 mm) radial helium gas gap therefore supports a 205°F (114 K) temperature drop, or approximately 27°F (15 K) per mil (25 µm). If the specimen were to become offset in the gas gap so that it contacts the thermal resistor, the specimen PCT and average outer surface temperature would decrease to approximately 2418°F (1599 K) and 2084°F (1413 K), respectively, Figure 19. The uncertainty of 15°F (8.3 K) in specimen PCT due to specimen shift is manageable. Likewise, if the UO₂ specimen swells completely into the thermal resistor, its PCT would decrease by approximately 200°F (111 K) (versus approximately 800°F (444 K) for a 750 W/cm³ fission heat generation rate). Note that the uncertainty due to specimen swelling would be significantly less than 200°F (111 K). Since the specimen dimensions are remeasured post-irradiation, to determine swelling and swelling rate (by dividing by the test time), the Tier 5 ANSYS thermal models would be updated with these dimensions, resulting in more accurate specimen temperature predictions. Finally, the release of fission gas products into the Haynes 230 can will have a smaller impact on specimen temperature because of the reduced fission heating rate. Specifically, it is predicted that the UO₂ specimen temperature would increase by approximately 100°F (55.6 K) after 12000 hours of operation due to the decrease in gas mixture thermal conductivity. This increase would be countered somewhat by the decrease in specimen temperature due to swelling.

In summary, radial heat transfer through the fuel specimens was ideal because it is representative of all Project Prometheus core concepts. The Tier 5 UO₂ specimens have sufficiently low target heat generation rates that specimen temperature uncertainties due to specimen offset, specimen swelling and fission gas release in a radial heat transfer design are manageable.

Zirconia Insulators (SK-MGP2203, Items 1-7)

The ceramic insulators above and below each UO₂ fuel stack were fabricated from fully dense zirconia stabilized with three weight percent magnesia. The proprietary material, designated Zycron Composition 1030, was supplied by:

Zircoa, Inc.
31501 Solon Rd.
Solon, OH 44139-3526
Tel: 440-248-0500
Fax: 440-248-8864

The purpose of the insulators was to minimize the axial heat conduction from the ends of the specimen stack, so that heat transfer through each specimen would be predominantly radial. Zycron Composition 1030 was selected for several reasons. First, it has extremely low thermal conductivity at elevated temperatures (2.0 W/mK versus approximately 5.0 W/mK for alumina and 110 W/mK for molybdenum). Second, components fabricated from 1030 can be ground to tight tolerances (± 0.001 inch (25.4 μm)). Finally, this particular type of zirconia has been successfully irradiated in prior MICE insertions without exhibiting significant irradiation damage. Components fabricated from zirconia partially stabilized with yttria, on the other hand, have experienced severe cracking as a result of irradiation in the ATR.

Tungsten – 3% Rhenium High Temperature Springs (SK-MGP2187, Item 1)

The high temperature spring is identical to that in Tier 4 and is discussed in detail in Section II.D. The spring accommodates thermal expansion of the internal can components, and prevents in-transit damage of the specimens.

Molybdenum-TZM Restraint Tubes (SK-MGP2204, Items 1-2) and Divider Plate (SK-MGP2204, Item 7)

The two restraint tubes in the Level 1 can assembly were designed to have less than a 0.0005-inch (12.7 μm) diametral clearance with the UO_2 specimens at temperature. Three small grain specimens were to be installed in the upper restraint tube and three large grain specimens in the lower tube. During irradiation, the specimens swell into the TZM tubes, which act as fuel cladding, so that specimen-clad interactions can be studied. TZM-molybdenum was chosen as the tubular clad material because available test data (creep and irradiated creep) exists in the vicinity of the predicted interface temperatures, albeit the predicted stresses were lower than the range of available test data by a factor of 3. The molybdenum divider plate isolates the three small grain specimens from the three large grain specimens. The divider plate was plasma sprayed with a 0.001-inch (25 μm) layer of zirconia so that it would not diffusion bond to the molybdenum restraint tubes.

Molybdenum-TZM Top hat (SK-MGP2204, Item 6)

The molybdenum top hat serves three functions. First, it helps center the upper zirconia insulators in the Haynes 230 can. Second, it has an integral flange that provides a bearing surface for the high temperature spring. The tungsten cap in the Tier 4 design serves a similar function, except the tungsten cap has two radial fins instead of a flange to minimize heat conduction losses from the specimen, through the tungsten cap and into the can. In the Tier 5 design, the zirconia insulators thermally isolate the molybdenum top hat from the specimens, so a flange can be utilized in lieu of two fins, which are more difficult to machine. Finally, a 0.10-inch (2.54 mm) diameter boss integral to the top hat limits axial translation of the fuel stack to 0.07 inches (1.778 mm), and thereby prevents the specimen from exiting the thermal resistor, should the high temperature spring inadvertently break.

Graphite Holders (SK-MGP2207, Items 1-3)

Each Haynes 230 can assembly was to be installed into a unique graphite holder. The graphite holders serve three purposes. First, the high thermal conductivity, low gamma heating graphite conducts heat from the can assembly to the thermal shields with minimal temperature drop across the graphite. As a result, most of the temperature difference between the Haynes 230 can and Inconel 600 capsule body is due to the temperature drop across the helium/neon gas gap on the outside of the thermal shield. The gas gap widths are therefore maximized, which

reduces the effect that component tolerances and thermal expansion have on specimen temperature. Secondly, the graphite holders contact the flat base and lid of each Haynes 230 pressure boundary. Contact with the high thermal conductivity graphite tends to reduce the radial temperature distribution in the Haynes 230, thereby reducing thermal stresses, although they are still relatively high (approximately 20 ksi maximum). Finally, the graphite holders have a slot feature to accommodate the thermocouple rib in the Haynes 230 can.

Molybdenum and Inconel 600 Thermal Shields (SK-MGP2208, Items 1 and 2; SK-MGP2206, Item 1)

Positioned around each graphite holder is a 0.03-inch (0.762 mm) thick thermal shield with three fins at the top and bottom that center the shield in the capsule body, thereby establishing a uniform gas gap around each shield. The fins are designed to minimize heat conduction through the fins directly into the capsule body. The dimensions and material of each thermal shield are selected to limit the maximum temperature of each Haynes 230 can to approximately 1550°F (1116 K), as listed below.

Table 6: Comparison of Levels 1 thru 3 Thermal Shields (Tier 5)

Level of Can	Thermal Shield Material	Thermal Shield Outside Diameter (in / mm)	Helium/Neon Gas Gap (in / mm)	Maximum Haynes 230 Can Temperature (°F / K)
1	Inconel 600	1.198 / 30.429	0.046 / 1.168	1549 / 1116
2	Molybdenum	1.226 / 31.140	0.032 / 0.813	1543 / 1112
3	Molybdenum	1.150 / 29.210	0.070 / 1.778	1492 / 1084

Non-fuel materials in Levels 2 and 3 have lower gamma heat rates than equivalent nonfuel materials in Level 1, because they are further away from the ATR core centerline. Thus, the UO₂ fuel stacks in these levels tend to run colder than the target PCT of 2420°F (1600 K). To compensate, molybdenum was selected as the thermal shield material because of its relatively high gamma heat rate and low emissivity. Likewise, because it is located closer to the ATR core centerline, the specimen stack in Level 1 tends to run hotter, so Inconel 600, with its lower gamma heating rate and higher emissivity (approximately 0.7 versus 0.23 for molybdenum) was selected for the Level 1 thermal shield. As discussed in Section II.D, maximizing the temperature of the Haynes 230 can minimizes the uncertainty in the specimen temperature measurement. By optimizing the helium/neon gas gap thickness, thermal shield material, and the length and thickness of each molybdenum thermal resistor, it was possible to meet the simultaneous requirements of a 2420°F (1600 K) PCT specimen and a 1550°F (1116 K) maximum temperature for the Haynes 230 can on all three levels. Note that this required significant iteration because modifying the dimensions in one level affects the temperatures on adjacent levels. The estimated temperature of the Level 3 Haynes 230 is approximately 50°F (27.8 K) lower than desired [1492°F (1084 K) versus 1550°F (1116 K)] due to the low gamma heat rates in Tier 5 Level 3. However, it was not possible to increase the can temperature by increasing the helium/neon gap thickness, since the rib in the Haynes 230 can would have interfered with the inside diameter of the molybdenum thermal shield.

Inconel 600 Capsule Body (SK-MGP2196, Item 1) and Head Assembly (SK-MGP2217, Group 2)

After electron beam welding the head assembly to the capsule body, these two components form the main pressure boundary to the ATR coolant. BEA requires that these components

meet the requirements of Section III, Class 3 of the BPVC. Design and service loadings for the capsule pressure boundary are equivalent to that for the MICE-3A capsules, which are discussed in Reference [c]. BPVC calculations for the pressure boundary are identical to that for the capsule described in Reference [a]. The three Type K thermocouples brazed to the head have crushed alumina insulation with a 316L stainless steel sheath. These thermocouples, which operate at approximately 1400°F (1033 K), are the same thermocouple types used in standard loop tests in the ATR.

Capsule Shrouds (6266E46, Item 2 and Item 4)

The material composition of the three capsule shrouds (i.e., Zircaloy - 70% hafnium, Zircaloy - 25% hafnium and aluminum for the Levels 1, 2 and 3 shrouds, respectively) was selected to meet the test sponsor target of 150 W/cm³ fission heat generation rate. Note that the shrouds farther removed from the ATR core centerline require less hafnium content, a neutron absorber, because of the cosine flux distribution of the core.

E. Specimen Temperature Uncertainty in MICE-3B Tier 5 Canned UO₂ Capsule

The temperature of the UO₂ specimens would not be directly measured. Instead, an ANSYS finite element model was to be used to correlate the specimen temperature with the temperatures recorded by three thermocouples embedded in the ribs integral to the Haynes 230 cans. As a result, there would be significant uncertainty in the specimen temperature. The nontrivial sources of uncertainty are as follows:

- Thermal contact resistance between the molybdenum thermal resistor flange and the Haynes 230 can. Modeled as a 0.001-inch (25 μm) helium gap, ANSYS predicts a 30°F (16.7 K) temperature drop across this interface. An upper bound on this uncertainty is 100%, or approximately 30°F (16.7 K).
- Thermal conductivity of irradiated molybdenum and Haynes 230. The uncertainty in the unirradiated thermal conductivity value of pure molybdenum and Haynes 230 should be fairly small. It is unclear, however, what the effects of irradiation are on these values, although it is expected to be less than a 15% change.
- Emissivity of molybdenum – The uncertainty in the emissivity of the molybdenum was estimated at 15%.
- Temperature uncertainty due to specimen offset inside molybdenum thermal resistor – As discussed in Section III.C, the PCT of the specimen decreases by approximately 15°F (8.3 K) if it is shifted 0.0075-inches (0.1905 mm) radially so that it contacts the thermal resistor.
- Temperature uncertainty due to irradiation-induced swelling of the specimen – Swelling of the UO₂ specimen would cause its temperature to decrease, as the gas gap between the specimen and molybdenum thermal resistor is reduced. However, the temperature uncertainty at the termination of the 12000 hour test is not significant, since the diameter of the specimen could be measured post-irradiation. There would be nontrivial temperature uncertainty (perhaps 50°F / 28 K) at intermediate test durations, since available data on UO₂ swelling versus fluence has significant scatter.
- Temperature uncertainty due to specimen fission gas release – Fission gases have significantly lower thermal conductivities than helium, so specimen temperatures would increase as the UO₂ fuel fissions and fission gasses are released. By the end of the test, it was estimated that fission gas release will result in a 100°F (56 K) increase in the specimen temperature. The uncertainty due to this temperature increase could be significantly reduced post-irradiation by puncturing each can and measuring the gas pressure and the elemental composition of the gas mixture.

- Fission and gamma heating rates of internal capsule materials— The uncertainty in fission and gamma heating rates is estimated at 15%. Unlike the other contributors to the overall temperature uncertainty, an empirical fission or gamma heating rate that does not match that used in the ANSYS model would be detected by the thermocouple readings, since the empirical and theoretical temperature drops across the gas gaps predicted by ANSYS would not be consistent.

It is difficult to assign a realistic overall specimen temperature uncertainty to the Tier 5 canned UO_2 design given the limited knowledge about each contributor. The Prometheus project was restructured prior to predicting the overall UO_2 specimen temperature uncertainty using the ANSYS thermal model. However, assuming that the specimen diameter would be measured and the gas mixture within the Haynes 230 can analyzed post-irradiation, it was estimated that the overall uncertainty in Tier 5 would be slightly less than that in the Tier 4 design, perhaps $+200^\circ\text{F} / -150^\circ\text{F}$ ($+111 \text{ K} / -83 \text{ K}$). The uncertainty is lower on the negative side because the release of fission gas would not result in a decrease in specimen temperature. In the Tier 4 design, uncertainty in the thermal contact resistance between the UN specimen and the tungsten pedestal and cap would be the largest contributor to the overall temperature uncertainty. In the Tier 5 design, uncertainties due to specimen offset, specimen swelling and fission gas release dominate those associated with uncertainties in the contact resistances.

F. Summary of MICE-3B Tier 5 Canned UO_2 Capsule Design

The comparisons shown in Table 4 demonstrates that the test designers were able to meet the target test conditions, albeit with significant uncertainty in the UO_2 specimen temperature. Radial heat transfer through the specimens, which is prototypic of Project Prometheus core concepts, was feasible in MICE-3B Tier 5 because of the reduced specimen heat generation rate. The UO_2 specimens were required to be canned to prevent release of fission products in excess of BEA emission thresholds and specimen contamination from impurities in the sweep gas.

References:

- (a) B-MT(EDT)I-1189, "Multiple Irradiation Capsule Experiment Design Report, MICE Prime, Appendix B," April 2002
- (b) B-MT(EDT)I-1316, "Multiple Irradiation Capsule Experiment Design Report, MICE-2, Appendix C," March 2003
- (c) B-MT(EDT)F-766, "Transmittal of Appendix E to The Multiple Irradiation Capsule Experiment Design Report MICE-3A (U)," December 2004
- (d) B-MT(EDT)R-1732, "Certification of Review For Irradiation Test Hardware: MICE-3a Head To Capsule Weld Qualification (U)," November 2004
- (e) MDO-723-0017, "Haynes 230 Mini-Can Welding to Support Planned Irradiation Testing of Candidate Space Fuel Materials (U)," To be issued

Figure 1: ATR Core Cross Sectional Diagram

ATR CORE CROSS SECTIONAL DIAGRAM

Figure 2: MICE-3B Tier 4 Canned UN Capsule Assembly

Figure 3: Level 3 Haynes 230 Can Assembly, MICE-3B Tier 4 UN Capsule

Figure 4: Location of Thermocouple in Levels 2, 4 and 6 Graphite Holders, UN Capsule

Figure 5: MICE-3B Tier 4 Canned UN Capsule Temperature Distribution (°F)

Figure 7: Level 4 UN Specimen Temperature Distribution in the (r, z) plane, Tier 4 UN Capsule (°F)

Figure 8: von Mises Stresses in Haynes 230 Can Subjected to 85 psia Internal Pressure (psia)

Figure 9: von Mises Thermal Stresses in Level 4 Haynes 230 Can, Tier 4 UN Capsule
(psia)

Figure 10: Empirical Load-deflection Curve of High Temperature Spring SK-MGP2187,
Item 1

Figure 11: MICE-3B Tier 5 Canned UO_2 Capsule Assembly

Figure 12: Level 3 Haynes 230 Can Assembly, MICE-3B Tier 5 UO₂ Capsule

Figure 13: Location of Thermocouple in Haynes 230 Can, Tier 5 UO₂ Capsule

Figure 14: Level 1 Haynes 230 Can Assembly, MICE-3B Tier 5 UO₂ Capsule

Figure 15: MICE-3B Tier 5 Canned UO₂ Capsule Temperature Distribution (°F)

Figure 16: Level 2 Can Assembly Temperature Distribution, UO₂ Capsule (°F)

Figure 17: Level 1 Can Assembly Temperature Distribution, UO₂ Capsule (°F)

Figure 18: Temperature Distribution of Specimens, MICE-3B Tier 5 Canned UO₂ Capsule (°F)

Figure 19: Comparison of Centered and Offset UO₂ Specimen Temperatures (°F)

Listed below are the sketches and drawings for the NE-M-2-103 (MICE-3B) Tier 4 Canned UN and Tier 5 Canned UO₂ capsule assemblies. The far right column of the table lists the status of each sketch and drawing.

Sketch or Drawing Number	Title	Rev.	Status Code
SK-MGP2081 (Sh. 1-3, 8, 11-12 only)	MICE Prime Head Details	F	2
SK-MGP2179	UN Specimen Details	-	2
SK-MGP2187	High Temperature Spring, Canned UN Capsule	A	1
SK-MGP2189	Haynes 230 Can Details	A	1
SK-MGP2191	Canned UN Tungsten Components	B	1
SK-MGP2192	Graphite Holders, Canned UN Capsule	-	2
SK-MGP2194	Spacers, Canned UN MICE Capsule	A	1
SK-MGP2196	Capsule Body, MICE Tier 4 UN Test	-	1
SK-MGP2197	MICE-3B Tier 4 Canned UN Capsule	-	2
SK-MGP2198	MICE-3B Tier 4 Haynes 230 Can Assemblies	-	2
SK-MGP2202	Haynes 230 Can Gauge	-	1
SK-MGP2203	Zirconia Pieces, UO ₂ Capsule	A	1
SK-MGP2204	Molybdenum Pieces Inside Haynes Can, MICE UO ₂ Capsule	A	1
SK-MGP2205	Haynes 230 Pieces, MICE UO ₂ Capsule	A	1
SK-MGP2206	Alloy 600 Components, MICE UO ₂ Capsule	-	1
SK-MGP2207	Graphite Holders, MICE UO ₂ Capsule	A	1
SK-MGP2208	Molybdenum Shields, MICE UO ₂ Capsule	-	1
SK-MGP2215	Can Assemblies, MICE UO ₂ Capsule	-	2
SK-MGP2216	MICE-3B Tier 5 Canned UO ₂ Capsule	-	2
SK-MGP2217	Welded Head Assy Details, MICE-3B Tiers 4 & 5	-	2
6266E46	MICE Capsule Rolled Shroud Details	A	1
6270E38	Multiple Irradiation Capsule Experiment (MICE) Strongback Capsule Spacer Details	A	1

Definition of Status Codes

- 1 – Sketch or drawing has been issued
- 2 – Sketch or drawing has not been issued; it is complete but has not been reviewed by a peer reviewer or supervisor

Attachment A
to

Enclosure 5
to

MDO-723-0015/ B-MT(EDT)S-028

Fission Product Compatibility with Haynes 230

CM Regan

Reviewed by :
MJ Kania
DV Miller
MC Chen
D Noe

There are no studies done in open literature where the compatibility of Haynes 230 with cesium, rubidium or tellurium were evaluated; however, the literature does investigate the interactions with stainless steel along with a few other materials [A-N].

Cesium and rubidium are both alkaline metals and should have similar corrosion characteristics; therefore, the response of one should be typical of the response of the other. The commonalities among the articles suggest that pure cesium is relatively unreactive with stainless steel; however, cesium containing a moderate amount of oxygen can cause grain boundary attack via the formation of cesium chromate.

One study exposed alloys Nimonic PE16, FV448 and stainless steel 316¹ in cesium containing 33 mol% oxygen (Cs₂O) for 168 hours at 948K. All three alloys severely corroded with the degree decreasing in the order Type-316>FV448>PE16. The corrosion was inter-granular causing severe longitudinal disruption between the grains. The chemistry of corrosion was dominated by formation of cesium chromate and ferrite [A]. Based on chemistry, Haynes 230 should perform as well as PE16, i.e. better than stainless steel, therefore information derived from studies on stainless steel can be considered a limiting case.

Little oxygen is produced from elemental transmutation during fission. The majority of oxygen liberated is from the Uranium – Oxygen bonds breaking during a fission – two oxygen atoms (one molecule) per fission. Therefore, little oxygen would be generated in UN fuel and any cesium generated would contain low oxygen.

In the proposed UN experiment, all cesium and rubidium would be vapor. The hottest mini-can area is ~1550°F (1115 K) while the coolest area is ~1300°F (975 K). The vapor pressure of cesium and rubidium at 1100 K is 360,000 Pa (52 psi) and is 168,000 Pa (24 psi) at 1000 K [N]. Estimates on elemental yields from ²³⁵U can be seen in Table 1.

Table 1: Approximate Yield per fission of ²³⁵U [O]

Chemical Group	Yield / Fission
Zr + Nb	0.298
Y + rare earths	0.534
Ba + Sr	0.149
Mo	0.240
Ru + Tc + Rh + Pd	0.263
Cs + Rb	0.226
I + Te	0.012
Xe + Kr	0.251

Each can has an interior gas volume of 2.5 cm³ and therefore, assuming an ideal gas, there would be 5x10⁻⁵ moles of Cs+Rb needed in the gas at 1000 K to generate the saturated partial pressure. The experiment was to operate at a maximum of 750 W/cm³ for 500 days and would have had an end of life fission density of 1x10²¹ fission/cc. Therefore, a fuel pellet 0.1 cm³ in volume would produce 3.7x10⁻⁵ moles of Cs + Rb by the end of test. Assuming all cesium and

¹ Nimonic PE16 = 43.5Ni, 34Fe, 1.2Al, 16.5Cr, 3.3Mo, 1.2Ti;
FV 448= Fe, 10.5 Cr, 0.13 C, 1.0 Mn, 0.5 Si, 0.75 Mo, 0.15 V, 0.45 Nb, 0.05 N
Type-316 = Fe, 18 Cr, 14 Ni, 0.08 C, 2.0 Mn, 1 Si, 3 Mo;
Haynes 230 = Ni, 22Cr, 14W, 1.25Mo, 1.4Fe, 0.16Co, 0.5Mn, 0.4 Si, 0.4Al, 0.11C, 0.012La, 0.0035B

rubidium diffuse out of the fuel, this is still less than the saturated partial pressure and therefore the gas would not condense on the can walls while operating. Cesium and rubidium would condense while the test cooled, but would evaporate again when the test started. Attack would be limited to vapor phase. As Haynes 230 would not be exposed to liquid cesium and there would be little oxygen in the can, there should be little or no attack on the mini-can by cesium.

For the UO_2 case, an ORIGEN model was run to determine the actual concentrations (and therefore, the total molar content) of the fission products. This is shown in Table 2.

Table 2: Calculated Elemental Fission Product Content per mini-can of UO_2 Fuel, 0.5 gms $^{235}\text{U}/\text{cm}^3$ (~5.7% enrichment) after irradiation at 150 Watts/ cm^3 for 500 EFPD in an ATR MICE Tier 5 Test *

Element	moles/can	Element	moles/can	Element	moles/can
Zr	4.81E-05	Sm	5.47E-06	Tb	1.57E-08
Xe	4.46E-05	Rb	5.15E-06	Ge	6.84E-09
Mo	4.22E-05	Te	4.88E-06	Dy	6.48E-09
Cs	3.60E-05	Pm	2.64E-06	As	2.23E-09
Nd	3.05E-05	I	2.57E-06	Ho	1.13E-10
Ru	3.03E-05	Ag	1.22E-06	Li	8.38E-11
Ce	2.82E-05	Nb	1.14E-06	Er	2.29E-11
Sr	1.45E-05	Se	9.16E-07	Be	2.16E-11
Ba	1.29E-05	Eu	8.13E-07	C	2.86E-12
La	1.15E-05	Cd	6.70E-07	Ga	2.13E-12
Tc	1.15E-05	Sn	5.70E-07	Zn	9.72E-13
Pd	1.14E-05	Br	3.43E-07	Tm	5.71E-14
Pr	9.89E-06	Gd	2.74E-07	Yb	2.83E-14
Y	7.34E-06	Sb	2.00E-07	Ni	1.54E-16
Rh	6.91E-06	H	6.70E-08	Cu	9.90E-17
Kr	5.73E-06	In	2.85E-08		

*Calculated by G.D. Carpenter using ORIGEN

In the proposed UO_2 experiment, the fuel volume is 0.6 cm^3 and would yield 4.12×10^{-5} moles of $\text{Cs} + \text{Rb}$ at the end of the test, an order of magnitude less than is needed to reach the saturated vapor pressure at 1000 K for an interior gas volume of 5.8 cm^3 . Therefore, very little of the gas would condense on the can walls while operating. Since oxygen influences the corrosion it must also be considered. By the end of test, $\sim 4 \times 10^{-4}$ moles of oxygen atoms would be liberated. The amount of oxygen that would be tied up in oxides of the fission products is dependent on fission product activity. Haynes 230 is a chromium bearing alloy and since the free energy of formation of Cs_5CrO_4 is -140 kcal/mol [H], it is reasonable to assume that fission product oxides with lower free energy would bind up any liberated oxygen. Therefore, oxides of the yttrium and rare earth category ($\Delta G \sim -280$ kcal/mol) and zirconium / niobium category ($\Delta G \sim -170$ to -220 kcal/mol) would be stable (assuming an ideal solution). This reduces the free oxygen to 2×10^{-4} moles, still more than enough to saturate the cesium. However, the reported free energy for grain boundary attack is -84 kcal/mol at 1070 K [C] and -100 kcal/mol at 970 K [H]. Barium, strontium and molybdenum are in the range of these energies and therefore, being in the fuel, may prevent some of the oxygen from leaving the fuel. If these elements tie up oxygen, there would be 7×10^{-5} moles free, still enough to saturate the cesium. Another complication in this estimate is the activation energy for inter-granular attack in Haynes 230 is

unknown. It is reasonable to believe the energy is lower than that of stainless steel based on the performance of PE 16. However, these values are for the end of test and are small values compared to the surface area of the can; in addition, not all the elements will diffuse out. Based on the limited amount of data, an attack on the mini-can from cesium and rubidium is not likely, but possible.

References [I, J] investigated the role of tellurium on corrosion in Nimonic PE16, FV448 and stainless steel 316 at Cs / Te ratios of 4:1, 2:1 and 1:1 both with and without oxygen. In pure cesium at 723 K to 1273 K with oxygen levels below 10 ppm no attack was observed after 1000 hours. Intergranular and transgranular corrosion started at oxygen levels above 500 ppm with increasing severity with increasing oxygen levels. The dilute solutions of Te in Cs (Cs/Te ratios of 2:1 and 4:1) with no oxygen caused little corrosion on all the alloys; however, once oxygen is added, the corrosion increases and continues to increase with increasing oxygen content with PE16 performing better than FV448 followed by Type 316 stainless steel. In both cases, the oxygen potential to initiate corrosion was higher (i.e. required less oxygen) than with pure cesium. In the case of the Te / Cs ratio of 1:1, there was severe corrosion of all alloys even without oxygen present and the corrosion rate was relatively insensitive to oxygen level when oxygen was added. Like pure cesium, tellurium - cesium mixtures with oxygen present were found to associate with chrome (and to a lesser extent nickel) on the grain boundaries and form precipitates [I, J].

In the MICE test, the ratio of cesium to tellurium would be high (~7:1) and therefore, should behave like a dilute solution. Based on these studies, the Hayes 230 in the UN capsule should exhibit little corrosion from the fission products. Like the case for pure cesium, the outcome for the UO₂ capsule is less conclusive. While the tellurium vapor would be dilute in the cesium vapor and there would be a lower vapor activity of reactant relative to the UN capsule, the presence of oxygen may cause increase reactivity toward the can. However, given the relative activation energies and the large surface to volume ratios of constituents involved, it is most likely that the integrity of the Hayes 230 fuel-can would not be affected.

Based on the available experimental data gathered from open literature and calculations performed on thermodynamic data, Haynes 230 should provide adequate corrosion resistance against the fission products generated by both the UO₂ and UN. However, to confirm this, simple static corrosion tests using cesium, tellurium, rubidium and oxygen in similar quantities and ratios expected in the MICE test should be performed. Depth of penetration and degradation of mechanical properties should be performed on the test coupon.

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Attachment B
to

Enclosure 5
to

MDO-723-0015/ B-MT(EDT)S-028

**Sketches and drawings for the NE-M-2-103 (MICE-3B)
Tier 4 Canned UN and Tier 5 Canned UO₂ capsule assemblies**

MG Petranka

A - COMPOSITION: UNS N06600, TYPE 11 OR 111, CONDITION BG, GRADE 2 B - MATERIAL SUPPLIED BY BUYER C - COMPOSITION: UNS N06600, ANNEALED				NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS					
								GP1	GP2	MATERIAL	STD	NUMBER	COMP	FORM	FIN
				A/B	1	SK-MGP2081-H01	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				A/B	2	SK-MGP2081-H02	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				A/B	3	SK-MGP2081-H03	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				A/B	4	SK-MGP2081-H04	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				C	5	SK-MGP2081-H05	SINGLE INSTRUMENT STRAIN RELIEF	1		NI-CR-FE	ASTM	B166		BAR	A
				A/B	6	SK-MGP2081-H06	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				A/B	7	SK-MGP2081-H07	HEAD, MICE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR	
				A/B	8	SK-MGP2081-H08	HEAD, MICE UN SPACE CAPSULE	1		NI-CR-FE	MIL	MIL-DTL-23229		BAR

- NOTES:
1. ELECTRICAL DISCHARGE MACHINING OF SLOTS AND HOLES NOT PERMITTED.
 2. DELETED
 3. THE VALUE OF DIAMETERS TXL (HIGH TEMP T/C LONG), TXS (HIGH TEMP T/C SHORT), STC (STANDARD T/C), TBI (INLET TUBE) AND TBO (OUTLET TUBE) SHALL BE SPECIFIED IN THE ORDERING INFORMATION (APPLIES TO ITEMS 1-4 AND 6-8).
 4. DIAMETERS TXL, TXS, STC, TBI, TBO AND R SHALL BE MEASURED AND RECORDED TO FOUR DECIMAL PLACES USING CLASS X PIN GAUGES.
 5. THE VALUE OF ØR (ITEM 5) SHALL BE SPECIFIED IN THE ORDERING INFORMATION.
 6. DELETED
 7. FOR 0.10 MINIMUM FROM [F], DIM "B" SHALL BE AS FOLLOWS: Ø0.288 REF, MACHINE FOR 0.0015 TO 0.0020 DIAMETRAL CLEARANCE WITH THE Ø0.290 HOLE OF ITEMS 1-4 AND 6-8, OR THE Ø0.290 HOLE OF PART SK-DRK1027-H01. THIS AS-BUILT DIMENSION SHALL BE DENOTED "B1". FOR REMAINING LENGTH OF STRAIN RELIEF DIM "B" SHALL BE WITHIN ±0.003/-0.000 OF "B1". A SMOOTH TRANSITION BLEND SHALL BE APPLIED BETWEEN THE DIAMETERS STARTING ABOVE THE 0.10 MINIMUM LENGTH WHERE "B1" TERMINATES.

2 MARKING		MIL-STD-792 TYPE 1	0.10 HIGH CHARACTERS
1 MARKING		MIL-STD-792 TYPE 1	0.12 HIGH CHARACTERS
NO. OPERATION		SPEC. NO.	REMARK
MANUFACTURING SPECIFICATIONS			
ENGINEERING SKETCH			
TITLE			
MICE PRIME HEAD DETAILS			
BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
CAGE CODE 94259			
SK-MGP2081 Rev F			
ENGINEER		DATE	
M. PETRANKA		10/18/01	
SUPERVISOR		DATE	
J. HOLITORIS		10/22/01	
PEER REVIEW		DATE	
D. R. KIRBY		10/17/01	
PROJECT (SECTION)		MICE (EDT)FEE	
WORK CATEGORY C		UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.	
ENGINEER		SUPERVISOR	
SUPERVISOR		PEER REVIEW	
REVISION BLOCK

ENGINEER	DATE	NO. OPERATION	SPEC. NO. REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS	
PEER REVIEW	DATE	ENGINEERING SKETCH	
TITLE			
MICE PRIME HEAD DETAILS			
P. O. NO.			
MICE-HEAD-WELD-PRIME			

WORK CATEGORY C	UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.
-----------------	---

BECHTEL BETTIS, INC.
PITTSBURGH, PA. U.S.A.

CAGE CODE 94259
SK-MGP2081 Rev F

VIEW C-C

0.010 $\pm$ 0.003 X
0.010 $\pm$ 0.003 CHAM

DETAIL F
ENLARGED

DETAIL G
ENLARGED
APPLIES 5 PLACES

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE		MICE PRIME HEAD DETAILS	
MT (ED) FEE		P.O. NO.		SK-MGP2081 Rev F	

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED, ALL
TOLERANCES AND MANUFACTURING
PRACTICES SPECIFIED ON 296E600
APPLY TO THIS DRAWING.

MICE PRIME HEAD, WELDED PRIME

WICE HEAD, WELDED PRIME

SHEET 3/12

BECHTEL BETTIS, INC.
PITTSBURGH, PA., U.S.A.

CAGE CODE 94259

SK-MGP2081 Rev F

SECTION Z-Z

MARK SERIAL NUMBER SPECIFIED
IN THE ORDERING INFORMATION IN
LOCATION SHOWN PER MFG SPEC 2

0.10 REF. SEE NOTE 7

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
PROJECT (SECTION)		TITLE		
MT(ED)FEE		MICE PRIME HEAD DETAILS		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296EG00 APPLY TO THIS DRAWING.		P.O. NO.		
WORK CATEGORY C		CAGE CODE 94259		
MINIFILLER-HEAD-WELDED-PRIME		SK-MGP2081 Rev F		
MICE-HEAD-WELDED-PRIME-REV		SHEET 8/12		

BECHTEL BETTIS, INC.
PITTSBURGH, PA., U.S.A.

MARK SERIAL NUMBER
SPECIFIED IN THE ORDERING
INFORMATION IN LOCATION
SHOWN PER MFG SPEC NO. 1

MARK SERIAL NUMBER
SPECIFIED IN THE ORDERING
INFORMATION IN LOCATION
SHOWN PER MFG SPEC NO. 1

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
MT(EDT)FEE		MICE PRIME HEAD DETAILS			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.		P.O. NO.			
WORK CATEGORY C		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SK-MGP2081 Rev F

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL
TOLERANCES AND MANUFACTURING
PRACTICES SPECIFIED ON 296E600
APPLY TO THIS DRAWING.

MINIFILLER_HEAD_WELDED_PRIME

MICE_HEAD_WELD_PRIME_REV

SHEET 1212

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATION			
				GPI	GP2	MATL	STD	NUMBER	COMP FORM FIN
A-H	1	SK-MGP2187-H01	HIGH TEMPERATURE SPRING	1		W-35RE			WIRE

NOTES A THRU H ON SHEET 2
REVISION BLOCK ON SHEET 2.

ALL DIMENSION IN INCHES.

WORK CATEGORY C	NO.	OPERATION	SPEC. NO.	REMARKS
	MANUFACTURING SPECIFICATIONS			
ENGINEER <i>M. G. Petras</i>	ENGINEERING SKETCH			
M. G. PETRAS 5/11/05	TITLE:			
SUPERVISOR	HIGH TEMPERATURE SPRING, CANNED UN CAPSULE			
G. C. WESSEVELT 5/11/05	PO NO.			
PEER REVIEW	SK-MGP2187 Rev A			
D. F. BLASCO 5/11/05	BECHTEL BETTIS INC. PITTSBURGH, PA., U.S.A.			

MSPACE_HOLDER_SPRING4

MSPACE_HIGH_TEMP_SPRING

SHEET 1/2

TABLE NOTES:

- A - NOMINAL WIRE DIAMETER: 0.040
- B - TOTAL COILS: 7.05
- C - DIRECTION OF COILING - OPTIONAL
- D - BOTH ENDS CLOSED AND GROUND
- E - CALCULATED LOAD AT ROOM TEMPERATURE: 0.67 LBF AT 0.424 WORKING HEIGHT (0.10 DEFLECTION)
- F - CALCULATED LENGTH OF WIRE: 13.4 REF
- G - STARTING FORMING, THERMAL SET AND STRESS RELIEVE SPRING IN HYDROGEN AT 1250°C
- H - REMOVE OXIDE FILM, SCALE, ETC. FROM FINAL SURFACES

A	MATERIAL "W-33RE" WAS "W-25IRE" NOTES A, B, E AND F UPDATED	ENGINEER <u>M. P. Jones</u> 6/3/65 SUPERVISOR <u>W. D. Doyle</u> 6/3/65 PEER REVIEW <u>P. F. Bettis</u> 6/3/65
REVISION BLOCK		

NO.	OPERATION	SPEC. NO.	REMARKS
MANUFACTURING SPECIFICATIONS			
ENGINEERING SKETCH			
TITLE: HIGH TEMPERATURE SPRING, CANNED UN CAPSULE			
PO NO.			
ENGINEER	M. P. Jones	6/3/65	BECHTEL BETTIS INC. PITTSBURGH, PA., U.S.A.
SUPERVISOR	X		
PEER REVIEW	X		
SK-MGP2187			Rev A

TABLE 1

ITEM NO.	GLM "L"
2	0.583±0.002
3	0.672±0.002

672 113015

REVISION BLOCK ON SH 2

[illegible]

SEE TABLE 1

NOTES:-
1. PERMANENT WARNINGS (I.E. SERIAL NUMBERS) ON ITEMS 1-5 NOT PERMITTED.

[illegible]

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296100 APPLY TO THIS DRAWING.

JVM NO.	DIM * L ¹
1	0.450 ± 0.003
2	0.539 ± 0.003
3	0.335 ± 0.003

TABLE 1

ITEM 1 SHOWN

3540-HISGML-15465N

SPACE-TUNING PIECES

50161 213

A - POCO GRAPHITE INC., UNOCAL CO., DECATUR TX B - MATERIAL IMPURITY IS 5 PPM MAXIMUM		ITEM NO.		PART OR ID. NO.		NOMENCLATURE OR DESCRIPTION		NO. REQ.		MATERIAL SPECIFICATIONS			
NOTE								GPI	GP2	MATERIAL	STD	NUMBER	COMP FORM
A/B	1	SK-MGP2192-H01				LEVEL 1 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR
A/B	2	SK-MGP2192-H02				LEVEL 2 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR
A/B	3	SK-MGP2192-H03				LEVEL 3 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR
A/B	4	SK-MGP2192-H04				LEVEL 4 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR
A/B	5	SK-MGP2192-H05				LEVEL 5 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR
A/B	6	SK-MGP2192-H06				LEVEL 6 GRAPHITE HOLDER		1		GRAPHITE	POCO	AXF-501	BAR

PRELIMINARY - NOT FOR CONSTRUCTION

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PIER REVIEW	DATE	ENGINEERING SKETCH		
PROJECT (SECTION)		TITLE		
WITNESS		GRAPHITE HOLDERS, CANNED UN MICE CAPSULE		
P.O. NO.		SK-MGP2192 Rev -

SECTION B-B

ITEM 2 SHOWN

TABLE 1

ITEM NO.	ϕ
2	1.22 ± 0.001
3	1.150 ± 0.001

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION) MT(EDT)FEE		TITLE GRAPHITE HOLDERS, CANNED UN "MICE" CAPSULE			
WORK CATEGORY C		P.O. NO.			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296600 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SK-MGP2192 Rev			

NSPACE 14UN GRAPHITE SLEEVE B

NSPACE GRAPHITE HOLDERS

SHEET 2/4

ITEM 4 SHOWN

TABLE 2

ITEM NO.	Ø 42	Ø 436
4	Ø 1.246 ± 0.001	1.0400 ± 0.0005
5	Ø 1.200 ± 0.001	1.0740 ± 0.0005

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
WORK CATEGORY C		TITLE		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.		GRAPHITE HOLDERS, CANNED UN MICE CAPSULE		
PROJECT (SECTION) MT (EDT) FEE		P.O. NO.		
SPACE 1AUB, GRAPHITE, SLEW 1B		SPACE GRAPHITE HOLDERS		
BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		CAGE CODE 94259		
SK-MGP2192		Rev -

SECTION F-F

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
MT (ED) FEE		GRAPHITE HOLDERS, CANNED UN-MICE CAPSULE			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2960680 APPLY TO THIS DRAWING.		P.O. NO.			
WORK CATEGORY C		SK-MGP2192 Rev -			
HSPACE - TUBS, GRAPHITE, SLEEVES		CAGE CODE 94259			
HSPACE - TUBS, GRAPHITE, SLEEVES		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
HSPACE - TUBS, GRAPHITE, SLEEVES		SHEET 214			

A - COMPOSITION: UNS S31600, ANNEALED B - COMPOSITION: UNS N06600, ANNEALED		NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS						
						GPI	GP2	MATERIAL	STD	NUMBER	COMP	FORM	FIN	
A		1	SK-MGP2194-H01	LEVELS 5-6 CENTERING SPACER		1			CRES	ASTM	A 216		BAR	A
A		2	SK-MGP2194-H02	LEVELS 2-4 CENTERING SPACER		1			CRES	ASTM	A 216		BAR	A
B		3	SK-MGP2194-H03	LOWER SPACER, TIER 4 UN CAPSULE		1			NI-CR-FE	ASTM	B 166		BAR	A
B		4	SK-MGP2194-H04	UPPER SPACER, TIER 4 UN CAPSULE		1			NI-CR-FE	ASTM	B 166		BAR	A

TABLE 1

ITEM NO.	DIMENSION "C"	DIMENSION "D"
1	1.273	1.042 ± 0.0005
2	1.274	1.0750 ± 0.0005

IT 4:
6X 0.085 ± 0.003 WAS 6X 0.082 ± 0.001

ENGINEER: M. G. PETRAMA 7/26/05
SUPERVISOR: G. C. WESTERVELT 7/26/05
PEER REVIEW: D. F. BLASKO 7/26/05

REVISION BLOCK

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
M. G. PETRAMA	7/26/05				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
G. C. WESTERVELT	7/26/05				
PEER REVIEW	DATE	ENGINEERING SKETCH			
D. F. BLASKO	7/26/05				
PROJECT (SECTION)		TITLE		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.	
MT(ED)FEE		SPACERS, CANNED UN MICE CAPSULE		CAGE CODE 94259	
		P.O. NO.		SK-MGP2194 Rev A	

MSPACE_LOWER_ASDO_SPACER

MSPACE_SPACERS

SHEET 123

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
W. J. P. P.	9/14/5				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
		ENGINEERING SKETCH			
		TITLE			
		SPACERS, CANNED UN WICE CAPSULE			
		CAGE CODE 94259			
		SK-MGP2194 Rev A			
		P.O. NO.			
		NSPAC. SPACERS			
		NSPAC. LOWER ASSEMBLY SPACER			
WORK CATEGORY C					
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2901600 APPLY TO THIS DRAWING.

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
M. J. B. B.	5/1/55				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
MTID 7111		SPACERS, CANNUL UN MICE CAPSULE			
WORK CATEGORY C		P. O. NO.			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SK-MGP2194 Rev A			

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.	MATERIAL SPECIFICATIONS							
					GPI	GP2	MATERIAL	STD	NUMBER	COMP	FORM	F/N
A/B	1	SK-MGP2196-H01	CAPSULE BODY MICE TIER 4 UN TEST	1			NI CR-FE	MIL	MIL-DTL-23229		BAR	

A - COMPOSITION: UNS N06600, TYPE II OR III, CONDITION B6, GRADE 2
B - MATERIAL SUPPLIED BY BUYER

SECTION C-C

SECTION B-B

DETAIL B
ENLARGED

MARK SERIAL NUMBER
SPECIFIED IN THE ORDERING
INFORMATION IN APPROXIMATE
LOCATION SHOWN, ON OUTSIDE
SURFACE, PER MTG SPEC NO. 2

ENGINEER	DATE
M. A. P. 7/10/65	7/10/65
SUPERVISOR	DATE
P. J. P. 7/10/65	7/10/65
PEER REVIEW	DATE
P. J. P. 7/10/65	7/10/65

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL
TOLERANCES AND MANUFACTURING
PRACTICES SPECIFIED ON 2361G00
APPLY TO THIS DRAWING.

2	MARKING	MIL-STD-792 TYPE VII	0.50±0.05 HIGH CHARACTERS
1	DESIGN	ASME BPVC SECTION III DIVISION 1 SUBSECTION ND CLASS 3	DESIGNED FOR 390 PSIG EXTERNAL PRESSURE AT 600°F
NO. OPERATION SPEC. NO.		REMARK	
MANUFACTURING SPECIFICATIONS		ENGINEERING SKETCH	
TITLE		MICE TIER 4 UN CAPSULE BODY DETAILS	
P.O. NO.		SK-MGP2196 Rev	
CAGE CODE		94259	
BECHTEL BETTIS, INC.		PITTSBURGH, PA., U.S.A.

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
<i>M.A. P. P.</i>	<i>1/1/68</i>				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
		ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
MTCED11EE		MICE TIER 4 UN CAPSULE BODY DETAILS			
WORK CATEGORY C		P.O. NO.			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296F600 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SK-MGP2196 Rev			

WSPACE, CAPSULE C, WELDED

MICE, WSPACE, CAPSULE C, WELDED

SHEET 2/2

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS					MATERIAL VOLUME (IN ³)
				GPI	GP2	MATERIAL	STD NUMBER	COMP	FORM	FIN	
	1	SK-MGP2196-H01	CAPSULE BODY, MICE TIER 4 UN TEST	1							5.754
	2	SK-MGP2194-H03	LOWER SPACER, TIER 4 UN CAPSULE	1							0.128
	3	SK-MGP2198-G01	LEVELS 1 AND 6 HAYNES 230 CAN ASSEMBLY	2							
	4	SK-MGP2192-H01	LEVEL 1 GRAPHITE HOLDER	1							0.420
	5	SK-MGP2194-H02	LEVELS 2-4 CENTERING SPACER	3							0.028
	6	SK-MGP2198-G02	LEVELS 2 AND 3 HAYNES 230 CAN ASSEMBLY	2							
	7	SK-MGP2192-H02	LEVEL 2 GRAPHITE HOLDER	1							0.616
	8	SK-MGP2192-H03	LEVEL 3 GRAPHITE HOLDER	1							0.493
	9	SK-MGP2198-G03	LEVELS 4 AND 5 HAYNES 230 CAN ASSEMBLY	2							
	10	SK-MGP2192-H04	LEVEL 4 GRAPHITE HOLDER	1							0.597
	11	SK-MGP2192-H05	LEVEL 5 GRAPHITE HOLDER	1							0.531
	12	SK-MGP2194-H01	LEVELS 5-6 CENTERING SPACER	2							0.028
	13	SK-MGP2192-H06	LEVEL 6 GRAPHITE HOLDER	1							0.512
	14	SK-MGP2194-H04	UPPER SPACER, TIER 4 UN CAPSULE	1							0.192
A/B	15	SK-MGP2197-H15	WAVE SPRING	1		FF-NI-CR	AMS 5810	A286		APPROX. 0.005	
	16	SK-MGP2217-G01	WELDED HEAD ASSY. TIER 4 CANNED UN CAPSULE	1						A600 0.784	
	17	6270E38-H02	SPHERICAL CAPSULE SPACER	1						0.120	
C	18	6266E46-H04	SHROUD	1						1.197	
D	19	6266E46-H02	SHROUD, WITH TABS	1						0.849	

A - COMPOSITION: UNS S66286
B - NOMINAL OD : 0.709, NOMINAL ID : 0.609, NOMINAL FREE HEIGHT = 0.415
NOMINAL LOAD AT WORKING HEIGHT : 3.5LB AT 0.300
C - ROLLED SHROUD, Ø 1.730 ID, 3.819 LG
MATERIAL IS 100% MAGNETUM
D - ROLLED SHROUD Ø 1.703 ID, 2.691 LG
MATERIAL IS 2R - 10 VOL% HF

PRELIMINARY - NOT FOR CONSTRUCTION

ENGINEER		DATE	NO.		OPERATION	SPEC. NO.	REMARK
SUPERVISOR		DATE	MANUFACTURING SPECIFICATIONS				
PEER REVIEW		DATE	ENGINEERING SKETCH				
WORK CATEGORY C		TITLE		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 286C600 APPLY TO THIS DRAWING.		MICE-3B TIER 4 CANNED UN CAPSULE		CAGE CODE 94259			
PROJECT (SECTION) WT(CED)T(FEE		P. O. NO.		SK-MGP2197 Rev			
HSPACE_WELDED_ASSY		HSPACE_WELDED_ASSY		SHEET 1/3

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
PROJECT (SECTION) MT(ED)T(EE)		TITLE		
		MICE-3B TIER 4 CANNED UN CAPSULE		
WORK CATEGORY C		P.O. NO.		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		
		CAGE CODE 94259		
		SK-MGP2197 Rev -

TABLE 1

LEVEL	DIM "L"
1	1.635
2	2.702
3	3.769
4	4.632
5	5.610
6	6.703

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
WORK CATEGORY C UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON Z396600 APPLY TO THIS DRAWING.		TITLE		
		MICE-3B TIER 4 CANNED UN CAPSULE		
		P.O. NO.		
PROJECT (SECTION) MT(ED)FEE		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		
NSPACE WELDED ASSY		CAGE CODE 94259		
NSPACE WELDED ASSY		SK-MGP2197 Rev -		
SHEET 3/3				

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.			MATERIAL SPECIFICATIONS			MATERIAL VOLUME (H3)
				GP1	GP2	GP3	MATERIAL	STD NUMBER	COMP FORM	FIM
	1	SK-MGP219-H01	UN SPECIMEN	1	1	1				0.006
	2	SK-MGP219-H04	TUNGSTEN CAP, LEVELS 1, 2, 3 AND 6	1	1	1				0.019
	3	SK-MGP219-H05	TUNGSTEN CAP, LEVELS 4 AND 5	1	1	1				0.018
	4	SK-MGP219-H01	TUNGSTEN BASE, LEVELS 1 AND 6	1	1	1				0.035
	5	SK-MGP219-H02	TUNGSTEN BASE, LEVELS 2 AND 3	1	1	1				0.037
	6	SK-MGP219-H03	TUNGSTEN BASE, LEVELS 4 AND 5	1	1	1				0.032
	7	SK-MGP2187-H01	HIGH TEMPERATURE SPRING	1	1	1				0.015
	8	SK-MGP2189-H02	HAYNES 230 CAN, LEVELS 1, 4, 5 AND 6	1	1	1				0.193
	9	SK-MGP2189-H03	HAYNES 230 CAN, LEVELS 2 AND 3	1	1	1				0.214
	10	SK-MGP2189-H01	HAYNES 230 LID, LEVELS 1 THRU 6	1	1	1				0.083

NOTES:

1. SOLVENT CLEAN EACH COMPONENT USING AN APPROVED PROCEDURE PRIOR TO ASSEMBLY.
2. ASSEMBLY SHALL BE FILLED WITH ULTRA HIGH PURITY HELIUM (99.999%) PRIOR TO WELDING.
3. VISUALLY CENTER ITS 2-1 IN ITS 8-9, AS APPLICABLE.
4. VISUALLY CENTER IT 1 FOR ITS 4-6, AS APPLICABLE.
5. ITS 2-3 ARE NOT PERMITTED TO CONTACT ITS 8-9, AS APPLICABLE.
6. ØD. 585 GROUND SURFACE OF 11 3 SHALL FACE DOWN, ORIENT 11 3 SO THAT THE GROUND (FLAT) SURFACE OF THE SPRING CONTACTS THE TWO LEGS OF ITS 2-3, AS APPLICABLE.
7. ALIGNMENT MARKS ON ITS 8-9 AND 10 SHALL BE COLLINEAR.
8. OUTER Cylindrical SURFACES OF IT 10 AND ITS 8-9, AS APPLICABLE, SHALL BE MAINTAINED AS FLUSH AS POSSIBLE PRIOR TO INITIATING WELD SEQUENCE.
9. USING A CLEAN FILE OF APPROVED MATERIAL, HAND-WORK THE CROWN OF THE WELD UNTIL IT IS FLUSH WITH THE ADJACENT BASE METAL.

PRELIMINARY - NOT FOR CONSTRUCTION

4	HELIUM LEAK TEST		
3	RADIOGRAPH		
2	LIQUID PENETRANT INSPECTION		
1	CASER WELD	ASME BPVC SECTION III CLASS 3	
NO.	OPERATION	SPEC. NO.	REMARK
MANUFACTURING SPECIFICATIONS			
ENGINEERING SKETCH			
TITLE			
NICE-38 TIER 4 HAYNES 230 CAN ASSEMBLIES			
P.O. NO.			
BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
CAGE CODE 94259			
SK-MGP2198 Rev.			

ENGINEER	DATE
SUPERVISOR	DATE
PEER REVIEW	DATE
PROJECT (SECTION)	MT (EDT) FEE

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 236C600 APPLY TO THIS DRAWING.

45PAC1.MATHE0230.CAN.335T

SHEET 1/2

FIGURE 1

INSPECTION REQUIREMENTS

- PRIOR TO AND SUBSEQUENT TO WELDING, PERFORM THE FOLLOWING INSPECTIONS ON EACH CAN ASSEMBLY. ALL RECORDED DIMENSIONS SHALL BE IN INCHES. POST-WELD DIMENSIONS SHALL BE RECORDED AFTER FILING THE WELD FLUSH PER NOTE 9.
1. PLACE THE CAN ASSEMBLY ON DATUM A AND MEASURE THE HEIGHT "L" TO THREE DECIMAL PLACES. THE HEIGHT "L" IS DEFINED AS THE DISTANCE FROM DATUM A TO THE HIGHEST POINT ON THE CAN LID.
 2. MEASURE THE DIAMETER OF THE CAN ASSEMBLY AT THE FOUR LOCATIONS SHOWN IN FIGURE 1 TO FOUR DECIMAL PLACES. DIAMETER SHALL BE MEASURED ADJACENT TO AND 90° FROM SCRIBE LINE AT:
 - a. TOP OF LID
 - b. AT THE WELD SEAM
 - c. APPROXIMATELY 0.25 INCHES BELOW THE WELD SEAM
 - d. BOTTOM OF CAN
 3. MEASURE THE FLATNESS OF DATUM A TO FOUR DECIMAL PLACES.
 4. MEASURE THE FLATNESS OF DATUM B TO FOUR DECIMAL PLACES.
 5. MEASURE THE PARALLELISM OF DATUM B RELATIVE TO DATUM A TO FOUR DECIMAL PLACES.
 6. DETERMINE THE SMALLEST DIAMETER BY MGP2202-1101 GAUGE THAT THE FINISHED ASSEMBLY CAN COMPLETELY SLIDE INTO WITHOUT THE USE OF FORCE. THIS INSPECTION SHALL BE PERFORMED SUBSEQUENT TO WELDING ONLY.

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR		MANUFACTURING SPECIFICATIONS			
PEER REVIEW		ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
NTED/TICE		MICE-38 TIER 4 HAYNES 230 CAN ASSEMBLIES			
WORK CATEGORY C		P.O. NO.			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 294E400 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SK-MGP2198 Rev -			

ITEM		PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS			
NOTE	NO.			GPI	GP2	WATER	STD	NUMBER	COMP FORM FIN
A/B	1	SK-MGP2203-H01	LOWER ZIRCONIA, LEVELS 2 & 3	1		ZIRCONIA			
A/B	2	SK-MGP2203-H02	UPPER INNER ZIRCONIA, LEVELS 2 & 3	1		ZIRCONIA			
A/B	3	SK-MGP2203-H03	UPPER OUTER ZIRCONIA, LEVELS 2 & 3	1		ZIRCONIA			
A/B	4	SK-MGP2203-H04	LOWER ZIRCONIA, LEVEL 1	1		ZIRCONIA			
A/B	5	SK-MGP2203-H05	UPPER INNER ZIRCONIA, LEVEL 1	1		ZIRCONIA			
A/B	6	SK-MGP2203-H06	UPPER OUTER ZIRCONIA, LEVEL 1	1		ZIRCONIA			
A/B	7	SK-MGP2203-H07	THERMAL BARRIER, LEVEL 3 LID	1		ZIRCONIA			

A - FULLY DENSE STABILIZED ZIRCONIA
B - MATERIAL REQUIREMENTS SPECIFIED IN ORDERING INFORMATION

ENGINEER M.D. Petros 9/12/95
SUPERVISOR W.D. May 9-12-95
PEER REVIEW D.F. Blasko 9/12/95

WORK CATEGORY C
UNLESS OTHERWISE SPECIFIED ALL
TOLERANCES AND MANUFACTURING
PRACTICES SPECIFIED ON 2968600
APPLY TO THIS DRAWING.

ENGINEER DATE
W.G. PETRANKA 8/26/05
SUPERVISOR DATE
D.F. BLASKO FOR GUN 8/26/05
PEER REVIEW DATE
D.F. BLASKO 8/26/05
PROJECT (SECTION)
WILLIETHIEL
P.O. NO.
ZIRCONIA PIECES,
UO2 CAPSULE
CAGE CODE 94259
SK-MGP2203 Rev A

REVISION BLOCK

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
M. J. P. J.	9/10/65				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PLCR REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE		BECHTEL BETTIS, INC.	
MTEDTJCE		ZIRCONIA PIECES, UO2 CAPSULE		PITTSBURGH, PA., U.S.A.	
		P.O. NO.		CAGE CODE	94259
		SK-MGP2203		Rev A	

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296600 APPLY TO THIS DRAWING.

A - ALTERNATE MATERIAL: MOLY, ASTM B 387, 365.
BAR, STRESS RELIEVED

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.	MATERIAL SPECIFICATIONS							
					GPI	GP2	MATERIAL	STD	NUMBER	COMP	FORM	FIN
	1	SK-MGP2204-H01	LOWER RESTRAINT TUBE		1		MOLY	ASTM	B 387	363	BAR	SR
	2	SK-MGP2204-H02	UPPER RESTRAINT TUBE		1		MOLY	ASTM	B 387	363	BAR	SR
A	3	SK-MGP2204-H03	LEVEL 3 THERMAL RESISTOR		1		MOLY	ASTM	B 387	361	BAR	SR
A	4	SK-MGP2204-H04	LEVEL 2 THERMAL RESISTOR		1		MOLY	ASTM	B 387	361	BAR	SR
A	5	SK-MGP2204-H05	LEVEL 1 THERMAL RESISTOR		1		MOLY	ASTM	B 387	361	BAR	SR
A	6	SK-MGP2204-H06	MOLY TOPHAT		1		MOLY	ASTM	B 387	361	BAR	SR
A	7	SK-MGP2204-H07	SPECIMEN SPACER		1		MOLY	ASTM	B 387	361	BAR	SR

0.0200^{+0.0015}
-0.0010

- NOTES:
1. DELETED
 2. DELETED
 3. STRESS RELIEVE ITS 1-7 AT 1840° ±25°F FOR 1 HOUR ±25 MIN AT 1X10-5 TORR MAXIMUM. FURNACE COOL. FURNACE AND TCS SHALL BE CALIBRATED. COMPONENTS SHALL BE CLEANED TO REMOVE FOREIGN MATERIALS, INCLUDING GREASE, OILS AND ACIDS. FROM ALL SURFACES PRIOR TO PERFORMING THE STRESS RELIEF OPERATION. IT IS RECOMMENDED THAT COMPONENTS WITH HARD TO REACH SURFACES BE CLEANED USING AN ULTRASONIC TECHNIQUE TO ASSURE CLEAN THRESH.
 4. A COMPONENT SHALL BE MACHINED AS CLOSE AS POSSIBLE TO FINAL SHAPE PRIOR TO STRESS RELIEVING PER NOTE 3. IT IS FEASIBLE THAT SOME COMPONENTS, ESPECIALLY THOSE WITH THIN WALLS, WILL DISTORT DURING THE STRESS RELIEF OPERATION. THEREFORE, THE SELLER IS PERMITTED TO LEAVE EXCESS MATERIAL ON A COMPONENT DURING THE STRESS RELIEF OPERATION AND THEN MACHINE TO FINAL DIMENSIONS.
 5. CLEANING FOLLOWED BY FLASH PICKLING OF ITS 1-7 TO REMOVE 0.0002 INCH OR LESS PER SURFACE AND ACHIEVE A MATTIE FINISH SHALL BE PERFORMED AFTER ALL OTHER FABRICATION OPERATIONS ARE COMPLETE. SEE ORDERING INFORMATION FOR ADDITIONAL REQUIREMENTS.
 6. AFTER FLASH PICKLING, VISUAL INSPECT ACCESSIBLE SURFACES AT 5X MINIMUM. NO INDICATIONS PERMITTED.
 7. ALL DIMENSIONS APPLY AFTER PICKLING.
 8. SEE ORDERING INFORMATION FOR RESTRICTIONS ON ELECTRICAL DISCHARGE MACHINING (EDM)
 9. NO PERMANENT MARKING PERMITTED ON ITS 1-2.

0.0200^{+0.0015}
-0.0010

REVISION BLOCK ON SHEET 2

M.D. P. 9/10/05

ENGINEER	DATE
M.G. PETRANKA	8/30/05
SUPERVISOR	DATE
M. S. SAYLOR	8/30/05
PEER REVIEW	DATE
H. R. WETZGER	8/30/05
WORK CATEGORY C	
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2966800 APPLY TO THIS DRAWING.	

NO.	OPERATION	SPEC. NO.	REMARK
MANUFACTURING SPECIFICATIONS			
ENGINEERING SKETCH			
TITLE			
MOLYBDENUM PIECES INSIDE HAYNES CAN, MICE UO2 CAPSULE			
P.O. NO.			
SK-MGP2204 Rev A			

MSPACE.UO2.LEVEL3.THERM.REGIST

MSPACE.UO2.MOLY.COMPONENTS

SHEET 1/4

TABLE 1

ITEM NO.	DIMENSION
3	0.030 ± 0.001
4	0.052 ± 0.002

ITEM 3 SHOWN

A	SH 1: BELTED NOTES 1 AND 2	ENGINEER	24.3.2000	24/05
	SH 2: ADDED "0.030 ± 0.001 TO TABLE 1"	SUPERVISOR	24.3.2000	24/05
	SH 2: ADDED "0.062 ± 0.002 TO TABLE 1"	PEER REVIEWER	24.3.2000	24/05
	SH 2: REMOVED "AND NOTE 1" FROM TABLE 1			
	SH 3: "0.030 ± 0.001" WAS "12, SEE NOTE 2"			

REVISION BLOCK

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEWER	DATE	ENGINEERING SKETCH		
PROJECT (SECTION)		TITLE		
MILITARY		HOLYDEN NUM PIECES INSIDE HAYNES CAN. NICE UO2 CAPSULE		
P.O. NO.		SK-MGP2204 Rev A		

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2986600 APPLY TO THIS DRAWING.

MSPACE, UO2 LEVELS, THERM RESIST

MSPACE, UO2 LEVEL COMPONENTS

ENGINEER M. J. P. [Signature]	DATE 1/10/54	NO.	OPERATION MANUFACTURING SPECIFICATIONS	SPEC. NO.	REMARK
SUPERVISOR	DATE	ENGINEERING SKETCH			
PEER REVIEW	DATE	TITLE NON YBDEHUM PIECES INSIDE HATHES CAN, MICE UO2 CAPSULE			
PROJECT (SECTION) NY (ED) FEE		P. O. NO.		CAGE CODE 94259	
WORK CATEGORY C		UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 214-680 APPLY TO THIS DRAWING.		SK-MGP2204 Rev A	
REPLACE AND REWORK THE ENTIRE SET					
REPLACE AND REWORK THE ENTIRE SET

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATIONS								
				NO. REQ.	GP1	GP2	WATERIAL	SID	NUMBER	COMP	FORM	FIL
A	1	SK-MGP2206-H01	LEVEL 1 THERMAL SHIELD		1		NI-CR-FE	ASTM	B 166		BAR	A
A	2	SK-MGP2206-H02	LOWER SPACER, MICE UO2 CAPSULE		1		NI-CR-FE	ASTM	B 166		BAR	A
A	3	SK-MGP2206-H03	UPPER SPACER, MICE UO2 CAPSULE		1		NI-CR-FE	ASTM	B 166		BAR	A

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATIONS								
				NO. REQ.								
				GP1	GP2	WATERIAL	SID	NUMBER	COMP	FORM	FIL	
A	1	SK-MGP2206-H01	LEVEL 1 THERMAL SHIELD	1		NI-CR-FE	ASTM	B 166		BAR	A	
A	2	SK-MGP2206-H02	LOWER SPACER, MICE UO2 CAPSULE	1		NI-CR-FE	ASTM	B 166		BAR	A	
A	3	SK-MGP2206-H03	UPPER SPACER, MICE UO2 CAPSULE	1		NI-CR-FE	ASTM	B 166		BAR	A	

NO. REQ.

NOMENCLATURE OR DESCRIPTION

PART OR
ID. NO.

NOTE	ON ITEM
------	---------

A - COMPOSITION: UNS N06600, ANNEALED

NOTES:
1. 3 SKIRT LENGTH 40.200 NOMINAL) SHALL BE REDUCED
PER BUYER INSTRUCTIONS

ENGINEER		DATE					
M. D. P. 200		9/13/05			NO.	OPERATION	SPEC. NO.
SUPERVISOR		DATE			MANUFACTURING SPECIFICATIONS		
W. A. 200		9/13/05			ENGINEERING SKETCH		
PLR REVIEW		DATE			TITLE		
L. P. 200		9/13/05			ALLOY 600 COMPONENTS.		
PROJECT		SECTION			MICE U02 CAPSULE		
MT		EDT	JEE		P.O. NO.		

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296-600 APPLY TO THIS DRAWING.

BECHTEL BETTIS, INC.
PITTSBURGH, PA., U.S.A.

CAGE CODE 94259

SK-MGP2206 Rev -

UNSPAC(102,LOWER,460,SPACR

COMPACT_102_A600_COMPONENTS

67-1385

SECTION C-C

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
M. J. Betts	9/15/65				
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
		ENGINEERING SKETCH			
		TITLE			
		ALLOY 600 COMPONENTS, MICL UO2 CAPSULE			
		P. O. NO.			
		BECHTEL BETTS, INC. PITTSBURGH, PA., U.S.A.			
		CAGE CODE 94259			
		SW-MGP2206 Rev -			

WORK CATEGORY C
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2961 600 APPLY TO THIS DRAWING.

SECTION D-D

ENGINEER M. J. P. Smith	DATE 9/10/65	NO. OPERATION MANUFACTURING SPECIFICATIONS	SPEC. NO. ENGINEERING SKETCH	REMARK
SUPERVISOR	DATE	TITLE ALLOY 600 COMPONENTS, WICE UO2 CAPSULE		
PEER REVIEWER	DATE	P. O. NO.		
PROJECT (SECTION) WICE UO2 CAPSULE		CAGE CODE 94259		
WORK CATEGORY C		SK-MGP2206 Rev -		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 244500 APPLY TO THIS DRAWING.		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		

A - POCO GRAPHITE INC., URBAL CO., DECATUR TX														
B - MATERIAL IMPURITY IS 5 PPM MAX														
NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS								
				GPI	GP2	MATERIAL	STD	NUMBER	COMP	FORM	FIN			
A/B	1	SK-MGP2207-H01	LEVEL 3 GRAPHITE HOLDER	1		GRAPHITE	POCO	AXF-SOI				BAR		
A/B	2	SK-MGP2207-H02	LEVEL 2 GRAPHITE HOLDER	1		GRAPHITE	POCO	AXF-SOI				BAR		
A/B	3	SK-MGP2207-H03	LEVEL 1 GRAPHITE HOLDER	1		GRAPHITE	POCO	AXF-SOI				BAR

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 294E840 APPLY TO THIS DRAWING.

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
M. J. P. 10/10/65	9/10/65	MANUFACTURING SPECIFICATIONS		
SUPERVISOR	DATE	ENGINEERING SKETCH	TITLE	
W. J. P. 10/10/65	9/10/65		GRAPHITE HOLDERS, WICE UO2 CAPSULE	
PROJECT	SECTION	P.O. NO.	CAGE CODE	Rev
MT(ED)TICE		SK-MGP2207	94259	

DECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.

WSPACEL-002-LEVEL 3, GRAPHITE

SHEET 1/3

ENGINEER M.A. Patake	DATE 9/10/65	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
PROJECT (SECTION) MT(FDT)FEE		TITLE		
WORK CATEGORY C		GRAPHITE HOLDERS, NICK UO2 CAPSULE		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296600 APPLY TO THIS DRAWING.		P.O. NO.		
		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		
		CAGE CODE 94259		
		SK-WGP2207 Rev -

SECTION C-C

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
M. J. P. P.	9/10/65		MANUFACTURING SPECIFICATIONS		
SUPERVISOR	DATE	ENGINEERING SKETCH			
		TITLE			
		GRAPHITE HOLDERS, MICE UO2 CAPSULE			
		P. O. NO.			
		SK-MGP2207 Rev -			

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E500 APPLY TO THIS DRAWING.

MSPACE .002 LEVEL 3, GRAPHITE

MSPACE UO2 GRAPHITE HOLDERS

SHEET 3/3

ALTHEA
BAM. 55

NUMBER	COM
B 387	36
B 387	36

FURNACE COOL
 OILING GREASE, OIL
 COMPONENTS THAT COMPONENTS
 STRESS RELIEVING PER NOT
 STRESS RELIEF OPERA
 STRESS RELIEF OPERATION
 SURFACE AND ACHIEVE
 ORDERING INFORMATION F
 INDICATIONS PERMITTED.
 (DM).

ENGINEER	DATE
M.A. Ponce	9/12/05
SUPERVISOR	DATE
W.A. Taylor	9/12/05
PEER REVIEW	DATE
R.P. Smith	9/12/05

UNLESS OTHERWISE SPECIFIED, ALL TOLERANCES ARE AS SHOWN. UNLESS SPECIFIED, SURFACES ON PART ARE TO APPLY TO BOTH SIDES.
--

WORK CATEGORY C

NO. OPERATION	SPEC. NO.	MARK
MANUFACTURING SPECIFICATIONS		
ENGINEER NG SKETCH	BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.	
TITLE	CAGE CODE 94259	
POLYBENZENE SHIELDS, MICE UO2 CAPSULE	SK-NGP2208 Rev -	
P. O. NO.

ITEM 1 SHOWN

ITEM NO.	Ø A	Ø B	Ø C	Ø
1	1.282 ± 0.001	1.100 ± 0.001	1.090 ± 0.001	2.069
2	1.282 ± 0.001	1.226 ± 0.001	1.166 ± 0.001	2.114

ENGINEER DATE 9/12/65		NO. OPERATION SPEC. NO. REMARK	
SUPERVISOR DATE		MANUFACTURING SPECIFICATIONS	
PEER REVIEW DATE		ENGINEERING SKETCH	
PROJECT (SECTION) MT (C/D) DEC		TITLE MOLYBDENUM SHIELDS, WICE UO2 CAPSULE	
WORK CATEGORY C		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.	
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2966600 APPLY TO THIS DRAWING.		CAGE CODE 94259	
P.O. NO.		SK-MGP2208 Rev -	

WSPACE UO2 LEVEL 3 SHIELD

WSPACE UO2 WOLY SHIELDS

SHEET 2/2

SID	NUMBER	COPY NO
1	1	1
2	2	2
3	3	3
4	4	4
5	5	5
6	6	6
7	7	7
8	8	8
9	9	9
10	10	10
11	11	11
12	12	12
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98	98	98
99	99	99
100	100	100

E	POCO	AXF-301	B
E	POCO	AXF-501	B
E	POCO	AXF-501	B

SECTION A-A

[illegible]

WORK CATEGORY C

UNLESS OTHERWISE SPECIFIED FOR ALL
TO: OFFICE OF THE SECRETARY
PRICES \$3.00 PER COPY \$29.50
APPLY TO THE SECRETARY

MSPACE_002_004 TO HOUSERS SHEET 1/3

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.			MATERIAL SPECIFICATIONS		
				GPI	GP2	GP3	MATERIAL	STD NUMBER	COMP FORM FIN
	1	SK-MGP2205-H02	HAYNES 230 CAN, UO2 CAPSULE	1	1	1			
	2	SK-MGP2205-H01	HAYNES 230 LID, UO2 CAPSULE	1	1	1			
	3	SK-MGP2187-H01	HIGH TEMPERATURE SPRING	1	1	1			
	4	SK-MGP2204-H06	WOLY TOPHAT	1	1	1			
	5	SK-MGP2203-H03	UPPER OUTER ZIRCONIA, LEVELS 2 & 3	1	1	1			
	6	SK-MGP2203-H02	UPPER INNER ZIRCONIA, LEVELS 2 & 3	1	1	1			
	7	SK-MGP2204-H03	LEVEL 3 THERMAL RESISTOR	1					
	8	SK-MGP2204-H04	LEVEL 2 THERMAL RESISTOR	1					
	9	SK-MGP2215-H09	UO2 SPECIMEN	6	6	6			
	10	SK-MGP2203-H01	LOWER ZIRCONIA, LEVELS 2 & 3	1	1	1			
	11	SK-MGP2203-H06	UPPER OUTER ZIRCONIA, LEVEL 1			1			
	12	SK-MGP2203-H05	UPPER INNER ZIRCONIA, LEVEL 1			1			
	13	SK-MGP2204-H02	UPPER RESTRAINT TUBE			1			
	14	SK-MGP2204-H07	SPECIMEN SPACER			1			
	15	SK-MGP2204-H01	LOWER RESTRAINT TUBE			1			

A - SPECIMENS ARE $\phi 0.240 \times 0.249 \times 0.117-0.121$ THICK

PRELIMINARY - NOT FOR CONSTRUCTION

- NOTES:
1. SOLVENT CLEAN EACH COMPONENT USING AN APPROVED PROCEDURE PRIOR TO ASSEMBLY.
 2. $\phi 0.585$ GROUNDED SURFACE OF IT 3 SHALL FACE DOWN.
 3. ALIGNMENT MARKS ON ITS 1 AND 2 SHALL BE COLLINEAR.
 4. OUTER CYLINDRICAL SURFACES OF ITS 1 AND 2 SHALL BE MAINTAINED AS FLUSH AS POSSIBLE PRIOR TO INITIATING WELD SEQUENCE.
 5. WELD PROCEDURE SPECIFICATION AND INSPECTION REQUIREMENTS TO BE DETERMINED.
 6. ASSEMBLY SHALL BE FILLED WITH ULTRA HIGH PURITY HELIUM (99.999%) PRIOR TO PLUG WELDING.
 7. USING A CLEAN FILE OF APPROVED MATERIAL, HAND-WORK THE CROWN OF THE WELDS UNTIL THEY ARE FLUSH WITH THE ADJACENT BASE METAL.

ENGINEER	DATE	NO. OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS		
PEER REVIEW	DATE	ENGINEERING SKETCH		
WORK CATEGORY C		TITLE		
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 256C600 APPLY TO THIS DRAWING.		CAN ASSEMBLIES, NICE UO2 CAPSULE		
PROJECT (SECTION)		P.O. NO.		
ATTENTION		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.		
		CAGE CODE 94259		
		SK-MGP2215 Rev -		

NSPAC 102 LEWELL WITH PREP

NSPAC 102 CAN ASSY

SHEET 173

GROUPS 1 & 2

ENGINEER		DATE		NO.		OPERATION		SPEC. NO.		REMARK	
SUPERVISOR		DATE		MANUFACTURING		SPECIFICATIONS					
FIELD REVIEW		DATE		ENGINEERING		SKETCH					
PROJECT (SECTION)		MIT (C) JEE		TITLE							
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2961660 APPLY TO THIS DRAWING.				CAR ASSEMBLIES, HICE UO2 CAPSULE						BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.	
WORK CATEGORY C				P. O. NO.						CAGE CODE 94259	
				SK-MGP2215						Rev

GROUP 3

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE			
MITLED/EE		CAN ASSEMBLIES, MICE U02 CAPSULE			
P. O. NO.		SK-MGP2215 Rev -			
WSPACE, U02 LEVELS, WITH, PREP		WSPACE, U02, CAN, ASST			

WORK CATEGORY C	
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.	

BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.	
CAGE CODE 94259	
SK-MGP2215 Rev -	

ITEM NO.	NOTE	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS			
				GPI	GP2	MATERIAL	STD NUMBER	COMP FORM	FIN
1		SK-MGP2196-H01	CAPSULE BODY, MICE TIER 4 UN TEST	1					
2		SK-MGP2206-H02	LOWER MOLY SPACER, MICE UO2 CAPSULE	1					
3		SK-MGP2206-H01	LEVEL 1 THERMAL SHIELD	1					
4		SK-MGP2207-H03	LEVEL 1 GRAPHITE HOLDER	1					
5		SK-MGP2215-G03	LEVEL 1 CANNED FUEL ASSEMBLY	1					
6		SK-MGP2208-H02	LEVEL 2 THERMAL SHIELD	1					
7		SK-MGP2207-H02	LEVEL 2 GRAPHITE HOLDER	1					
8		SK-MGP2215-G02	LEVEL 2 CANNED FUEL ASSEMBLY	1					
9		SK-MGP2208-H01	LEVEL 3 THERMAL SHIELD	1					
10		SK-MGP2207-H01	LEVEL 3 GRAPHITE HOLDER	1					
11		SK-MGP2215-G01	LEVEL 3 CANNED FUEL ASSEMBLY	1					
12		SK-MGP2203-H07	THERMAL BARRIER, LEVEL 3 LID	1					
13		SK-MGP2206-H03	UPPER SPACER, MICE UO2 CAPSULE	1					
A/B		SK-MGP2216-H14	WAVE SPRING	1		FE-NI-CR	AMS 5810	A286	
15		SK-MGP2217-G02	WELDED HEAD ASSY, MICE CANNED UO2 CAPSULE	1					
16		6266E46-H04	SHROUD	1					
17		6266E46-H04	SHROUD	1					
18		6266E46-H02	SHROUD, WITH TABS	1					
19		6270C38-H02	SPHERICAL CAPSULE SPACER	1					

A - COMPOSITION: UNS S86286
 B - NOMINAL OD = 0.709, NOMINAL ID = 0.609, NOMINAL FREE HEIGHT = 0.415 REF
 NOMINAL LOAD AT WORKING HEIGHT = 3.5LBS AT 0.300
 C - ROLLED SHROUD, Ø 1.730, 2.442 LG
 MATERIAL IS ZR-10% HAFNIUM
 D - ROLLED SHROUD, Ø 1.730, 2.067 LG
 MATERIAL IS ZR-25% HAFNIUM
 E - ROLLED SHROUD, Ø 1.730, 2.001 LG
 MATERIAL IS ALUMINUM

PRELIMINARY - NOT FOR CONSTRUCTION

ENGINEER	DATE	NO.	OPERATION	SPEC. NO.	REMARK
SUPERVISOR	DATE	MANUFACTURING SPECIFICATIONS			
PEER REVIEW	DATE	ENGINEERING SKETCH			
WORK CATEGORY C		TITLE			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 298E600 APPLY TO THIS DRAWING.		MICE-38 TIER 5 CANNED UO2 CAPSULE			
PROJECT (SECTION) WT (C/D) FEE		P. O. NO.			
		CAGE CODE 94259			
		SK-MGP2216 Rev -			

BECHTEL BETTIS, INC.
PITTSBURGH, PA., U.S.A.

MSPACE-UD2-ASSY

MSPACE-UD2-ASSY

SHEET 173

ENGINEER		DATE		NO. OPERATION		SPEC. NO. REMARK	
SUPERVISOR		DATE		MANUFACTURING SPECIFICATIONS			
PLER REVIEW		DATE		ENGINEERING SKETCH			
PROJECT (SECTION)		TITLE		BECHTEL BETTIS, INC.		PITTSBURGH, PA., U.S.A.	
MT/ERT/EE		MICE-3B TIER 5		CAGE CODE 94259			
		CANNED D02 CAPSULT		SK-MGP2216		Rev -	
WORK CATEGORY C		P.O. NO.					
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2981600 APPLY TO THIS DRAWING.							

HSPACE. D02. ASST

HSPACE. D02. ASST

SHEET 223

ENGINEER		DATE		NO. OPERATION		SPEC. NO. REMARK	
SUPERVISOR		DATE		MANUFACTURING SPECIFICATIONS			
PEER REVIEW		DATE		ENGINEERING SKETCH			
PROJECT (SECTION)		DATE		TITLE			
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 2966600 APPLY TO THIS DRAWING.		MICE-38 TIER 5		CAGE CODE 94259			
WORK CATEGORY C		CANNED UO2 CAPSULE		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.			
P.O. NO.		SK-MGP2216		Rev -			

MSPACE.D02.J55T

MSPACE.D02.J55T

SHEET 313

NOTE	ITEM NO.	PART OR ID. NO.	NOMENCLATURE OR DESCRIPTION	NO. REQ.		MATERIAL SPECIFICATIONS						
						GP1	GP2	MATERIAL	STD	NUMBER	COMP	FORM
	1	SK-MGP2081-H08	HEAD, WICE UN SPACE CAPSULE	1	1							
	2	SK-MGP2081-H05	SINGLE INSTRUMENT STRAIN RELIEF	5	5							
A	3	SK-MGP2217-H03	TUBE	2	2			CRES	ASME	SA-213	IP304L	SMLS
B	4	SK-MGP2217-H04	STANDARD ATR TYPE K THERMOCOUPLE	1	1							
C	5	SK-MGP2217-H05	BRAZE FILLER METAL	AR	AR				AWS	A5.8		

A - COMPOSITION: UNS S30403 (304L)
 Ø0.063±0.001 OUTSIDE DIAMETER, 0.015 MINIMUM WALL THICKNESS, TUBING LENGTH : 360±2
 B - SHEATH COMPOSITION: UNS S31603 (316L)
 Ø0.063±0.001 OUTSIDE DIAMETER, 0.015 MINIMUM WALL THICKNESS, TUBING LENGTH : 360±2
 THERMOCOUPLE INSULATION: CRUSHED ALUMINA
 JUNCTION IS UNGROUNDED
 TOTAL LENGTH OF THERMOCOUPLE: 360±2
 C - AWS CLASSIFICATION BN1-5

PRELIMINARY - NOT FOR CONSTRUCTION

6	HELIUM LEAK TEST	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS D	
5	HELIUM LEAK TEST	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS B	
4	BRAZE INSPECTION	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS D	VISUAL INSPECT BRAZEMENT BOTH SIDES
3	BRAZE INSPECTION	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS B	VISUAL INSPECT BRAZEMENT BOTH SIDES
2	BRAZE	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS D	VACUUM BRAZE
1	BRAZE	PER GUIDANCE OF NAVSEA 0989-027-8000, CLASS B	VACUUM BRAZE
NO. OPERATION SPEC. NO.		REMARK	
MANUFACTURING SPECIFICATIONS			
ENGINEERING SKETCH			
TITLE			BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.
WELDED HEAD ASSY DETAILS, WICE-3B TIERS 4 & 5			CAGE CODE 94259
P.O. NO.			SK-MGP2217 Rev -

ENGINEER	DATE
SUPERVISOR	DATE
PEER REVIEW	DATE
PROJECT (SECTION)	
ATTACHED	
WORK CATEGORY C	
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296100 APPLY TO THIS DRAWING.	

SPACE-HEAD WELDED ASSY

WICE-PRIME 3B-HEAD-ASB-SPACE

SHEET 173

NOTES:
1. AFTER BRAZING EACH ASSEMBLY, THE DISTANCE FROM DATUM "C" TO THE TIP OF THE INTERCOUPLE/TUBE SHALL BE PER THE FOLLOWING TABLE:

PART NUMBER	DESCRIPTION	LOCATION ØTIP (INLET TUBE)	LOCATION ØTBO (OUTLET TUBE)	LOCATION ØTTL (SPECIAL T/C, LONG)	LOCATION ØTTS (SPECIAL T/C, SHORT)	LOCATION ØSIC (STANDARD T/C)
SK-MGP2217-G01	WELDED HEAD ASSY, NICE-38 TIER 1 US SPACE CAPSULE	6.803±0.050	0.100±0.050	5.792±0.050	3.658±0.050	1.483±0.050
SK-MGP2217-G02	WELDED HEAD ASSY, NICE-38 TIER 3 UO2 SPACE CAPSULE	6.803±0.050	0.100±0.050	6.670±0.050	4.591±0.050	2.312±0.050

2. ROTATIONAL ORIENTATION OF ITEM 2 SHALL BE DETERMINED BY THE SELLER SO THAT THE TOP OF EACH BRAZE CAN BE INSPECTED.
3. DIMENSIONS APPLY AFTER BRAZING.
4. SEE ORDERING INFORMATION FOR ADDITIONAL REQUIREMENTS.

ENGINEER		DATE	NO.		OPERATION	SPEC. NO.	REMARK
SUPERVISOR		DATE	ENGINEERING SKETCH		MANUFACTURING SPECIFICATIONS		
PEER REVIEW		DATE	TITLE		BECHTEL BETTIS, INC. PITTSBURGH, PA. U.S.A.		
PROJECT (SECTION)		WELDED HEAD ASSY DETAILS, NICE-38 TIERS 4 & 5		CAGE CODE		94259	
ATTENTION		P.O. NO.		SK-MGP2217		Rev -	
WORK CATEGORY C		UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 298C600 APPLY TO THIS DRAWING.		VICE-PRESIDENT, HEAD, ASST.		SHEET 222

GROUPS 1 & 2 WELDED HEAD ASSY. MICE-3B CANNED UN AND UOZ CAPSULES

SECTION A-A

REFERENCED IN NOTE 1

WORK CATEGORY C		ENGINEER		DATE	NO.	OPERATION	SPEC. NO.	REMARK
UNLESS OTHERWISE SPECIFIED ALL TOLERANCES AND MANUFACTURING PRACTICES SPECIFIED ON 296E600 APPLY TO THIS DRAWING.		SUPERVISOR		DATE	MANUFACTURING SPECIFICATIONS			
PROJECT (SECTION)		PEER REVIEW		DATE	ENGINEERING SKETCH			
ATTACHED FILE		TITLE		WELDED HEAD ASSY DETAILS, MICE-3B TUBES 4 & 5				
MSPACE-HEAD-WELDED ASSY		P.O. NO.		SK-MGP2217 Rev				
MICE-3B HEAD ASM SPACE		SHEET 3/3		BECHTEL BETTIS, INC. PITTSBURGH, PA., U.S.A.				
CAGE CODE 94259		CAGE CODE 94259		CAGE CODE 94259				

FABRICATION NOTES:
 STEP 1 - MACHINE ENGRAVE AS SPECIFIED.
 STEP 2 - REBRASS, CLEAN AND VACUUM ANNEAL FORMED TUBE AT 1400°F ±25°F.
 STEP 3 - REBRASS, CLEAN AND VACUUM ANNEAL FORMED TUBE AT 1400°F ±25°F.
 STEP 4 - CLEAN PER MIL SPEC NO. 1, PICKLE SHROUD PER MIL SPEC NO. 2.
 STEP 5 - MACHINE ENGRAVE AS SPECIFIED.
 STEP 6 - AFTER PICKLING AND ANNEALING, VIBRATION, OR SHROUD BOTTOM SURFACE OF ALL
 CLEAN TUBE.

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
5	REVISIONS SHOWN BLOCK UPDATES				
6	REVISIONS SHOWN BLOCK UPDATES				
7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

VIEW G-4

VIEW G-5

A. MATERIAL, UNDER DIMENSIONS "B" AND "C" SPECIFIED IN ORDERING INFORMATION, MATERIAL TO BE EITHER (1) ALUMINUM SHEET ALLOYED WITH ZIRCONIUM, RANGING FROM 1/16" TO 1/8" AND SUPPLIED BY SUBSTITUTIONS, OR (2) ALUMINUM TUBE OF BAR.

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
5	REVISIONS SHOWN BLOCK UPDATES				
6	REVISIONS SHOWN BLOCK UPDATES				
7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

NOTE: ALL DIMENSIONS APPLY PRIOR TO PICKLING. DIMENSIONS MAY BE MEAN ENGRAVED DIMENSIONS. DIMENSIONS MAY BE MEAN ENGRAVED DIMENSIONS. DIMENSIONS MAY BE MEAN ENGRAVED DIMENSIONS.

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
5	REVISIONS SHOWN BLOCK UPDATES				
6	REVISIONS SHOWN BLOCK UPDATES				
7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
5	REVISIONS SHOWN BLOCK UPDATES				
6	REVISIONS SHOWN BLOCK UPDATES				
7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
5	REVISIONS SHOWN BLOCK UPDATES				
6	REVISIONS SHOWN BLOCK UPDATES				
7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

ITEM	DESCRIPTION	REVISIONS	DATE	BY	APP'D
1	REVISIONS SHOWN BLOCK UPDATES				
2	REVISIONS SHOWN BLOCK UPDATES				
3	REVISIONS SHOWN BLOCK UPDATES				
4	REVISIONS SHOWN BLOCK UPDATES				
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7	REVISIONS SHOWN BLOCK UPDATES				
8	REVISIONS SHOWN BLOCK UPDATES				
9	REVISIONS SHOWN BLOCK UPDATES				
10	REVISIONS SHOWN BLOCK UPDATES				

CAD DRAWINGS CHANGES SHALL BE INCORPORATED BY DESIGN ACTIVITY
APPLICABLE DOCUMENTS: MIL-STD-100G, ASME Y14.5M-1994.
DAP. SEE PURCHASE ORDER FOR ANY APPLICABLE REQUIREMENTS.

[illegible]

CONCURRENCE/DESIGN CHECK FORM FOR DOCUMENT NO. MDO-723-0015 / B-MT(EDT)S-028 Date: Jan. 9, 2006

DOCUMENT TITLE: Multiple Irradiation Capsule Experiment (MICE)-3B Irradiation Test of Space Fuel Specimens in the Advanced Test Reactor (ATR) - Closeout Documentation for Naval Reactors (NR) Information (U)

REFERENCES: B-SE(RE)-0002; B-SE(RE)-0003;
WAPD-MT(EDT)F-478;
MDO-CME-2120; MDO-723-0045;
MDO-720-1138; MDO-723-0017;
MDO-723-0044 / B-MT(SRME)-52

ENCLOSURES: (1) Test Details for MICE-3B Space Material Capsules
(2) Post Irradiation Examination Plan for MICE-3B Space Materials
(3) Envisioned Future Fuel Testing in the MICE Facility
(4) Compliance Assessment for MICE-3B UO₂ Capsule
(5) MICE-3B Capsule and Fuel-Can Design Report

1. ADSARS: PERMANENT RECORD: Yes X No Repository MFLIB Corporate Author: KAPL NR PROGRAM 08S

Key Words:	<u>MICE</u>	<u>Uranium Nitride</u>	<u>Fuel</u>	<u>Prometheus</u>	<u>Space</u>	<u>Uranium Dioxide</u>
Need to Know Categories	<u>SGN</u>	<u>REC</u>	<u>REP</u>	<u> </u>	<u> </u>	<u> </u>
Available Sites:	<u>PNR</u>	<u> </u>	<u> </u>	<u> </u>	<u> </u>	<u> </u>
Design File Location(s)	<u> </u>	<u> </u>	<u> </u>	<u> </u>	<u> </u>	<u> </u>

2. DESIGN CHECK

Type of Check	Signature(s)	Comments: (Including Reference to Check Document If Appropriate)
A. No check considered necessary		
B. Check vs. previous results/issues		
C. Checked calculations made		
D. Checked computer input and/or output		
E. Computer Programs approved/qualified		
F. Performed independent audit		
G. Spot checked significant points	<u>X</u> <u>CMR</u>	<u>For D.V. Miller 1/6/06 per e-mail</u>
H. Reviewed methods used	<u>X</u> <u>CMR</u>	<u>For D.V. Miller 1/6/06 per e-mail</u>
I. Reviewed results for reasonableness		
J. Comparison with test data		
K. Reviewed vs. drawings		
L. Verified procedures		
M. Technical content reviewed	<u>X</u> <u>CMR</u>	<u>For D.V. Miller 1/6/06 per e-mail</u>
N. Management verification of adequate review by others		
O. Performed Lessons Learned Search		
P. Used Measurement Uncertainty Methods		
Q. Other Checks (Describe)		

3. CONCURRENCE REQUIREMENTS:

Indicate signatures required by X:

ARP MANAGER
NUCLEAR ENGINEERING
REACTOR TH/MECH DESIGN
REACTOR EQUIPMENT
POWER PLANT MECHANICAL
POWER PLANT ELECTRICAL
FINANCE
NEW SHIP PROGRAMS
PROGRAM COORDINATION

NCSG
ADVANCED CONCEPTS
NOISE & ELEC. TECH.
SHIELDING
REACTOR SAFETY
TO
RSO
FSO
MDO

FLUID DYNAM
STRUC. ENGRG
DRAFTING
QA
OTHER
BETTIS
BPMI
ADMIN REVIEW

Cognizant Manager

(Must Be Subsection or Higher for External Letters)

4. AUTHORIZED CLASSIFIER:

Reviewed By:

CLASSIFICATION: UNC

5. RELATED SUBJECTS:

UTRS Implication (Y/N)	<u>N</u>	Design Basis Info. (Y/N)	<u>N</u>	Commitment Complete (Y/N)	<u>N</u>
Safety Council Review (Y/N)	<u>N</u>	Design Review (Y/N)	<u>N</u>	UTRS Doc. #	<u>NA</u>

6. Distribution:

NR

J.D. Yoxtheimer	08S
D.I. Curtis	08S
T.N. Rodeheaver	08I
C.H Davis	08S
J.P. Mosquera	08C
M.D. Natale	08I
S.T. Bell	08I
C.H. Oosterman	08C

PNR

S. C. Berasi	PNR
J.A. Andes	PNR
J. F. Koury	PNR

BETTIS

D. C. Noe	ZAP 05P
M. G. Petranka	ZAP 01B
W. S. Saylor	ZAP 01B
* J. L. Hollenbeck	
* B.V. Cockeram	
* D.V. Miller	
* A.J. Mueller	
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* D. Mangone	
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