MASTER

OAK RIDGE NATIONAL LABORATORY

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UNION CARBIDE CORPORATION
NUCLEAR DIVISION

for the

U.S. ATOMIC ENERGY COMMISSION

ORNL - TM - 2363

**DESIGN OF A CAPSULE FOR IRRADIATION
TESTING OF URANIUM NITRIDE FUEL**

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Reactor Division

DESIGN OF A CAPSULE FOR IRRADIATION TESTING OF URANIUM NITRIDE FUEL

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Oak Ridge, Tennessee
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UNION CARBIDE CORPORATION
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ABSTRACT

This report covers the design of an experimental capsule assembly for use in assessing the irradiation performance of uranium nitride (UN) fuel. Design conditions for the first irradiation test include operation of the fuel cladding at 1370°C with a linear heat rating of 8.5 kw/ft. A Zircaloy-2 calorimeter provides continuous monitoring of the heat generation rate in the fuel pins. The first test (capsule O5-11) is being conducted in the O5 poolside position of the Oak Ridge Research Reactor to an approximate burnup of 3 at. % U.

Primarily, the report describes the irradiation assembly; however, some initial operating experience is included. Detailed reporting of the fuel characterization, operational test results and the postirradiation examination will be the subjects of other reports.

Because of difficulties experienced with six of the high temperature thermocouples at the outset of neutron exposure certain revisions were made to the original design. The nature of this problem and the design revisions to alleviate it are discussed.

INTRODUCTION

An experimental program is underway at ORNL to evaluate nitride fuels. As part of this program an experimental assembly was designed for irradiation testing of uranium nitride (UN) fuel specimens. Since the thermal efficiency rises rapidly with increased operating temperatures, there is appreciable incentive attached to use of a fuel-cladding-coolant system that operates successfully at elevated temperatures. Thus, the present high-temperature test work represents a contribution to the development of high-temperature reactor technology. It is hoped that results from these

experiments will provide direction for establishing design criteria for reactors of this type and will provide useful information with which to predict the design limitations.

This report covers the design of a test assembly suitable for irradiation testing at high temperatures of several short, vertically-stacked uranium nitride fuel elements in a poolside position of the Oak Ridge Research Reactor (ORR). Since initial operating experience with the first test (capsule O5-11) indicated the desirability of a design revision, some operational information is given. Capsule O5-11 contained UN fuel clad with W-26% Re and was designed for a clad surface temperature of 1370°C, linear heat rating of 8.5 kw/ft and exposure to about 3 atom % U burnup.

Detailed reporting of the preirradiation characteristics, irradiation conditions and postirradiation examination will be the subject of other reports.

CAPSULE DESIGN CRITERIA

The criteria for design of this particular test of uranium nitride (capsule O5-11) and other tests in the series included the following:

Primary Criteria for all Tests in the Series:

1. Linear heat generation rate - 8.5 kw/ft.
2. Fuel cladding temperature - 1370°C (2500°F).
3. Reasonably reliable measurement of cladding surface temperatures on all test fuel elements.
4. Measurement of fuel center temperature in at least one fuel element.
5. Continuous measurement of the linear heat generation rate for each test element.

Secondary Criteria for at Least the First Test:

1. Number of test fuel elements - 3.
2. Cladding material - W-26% Re.
3. Fuel diameters - 0.3 in. for top and bottom elements and 0.25 in. for middle.
4. Maximum ^{235}U enrichment - 20% but to be varied as required for each element to achieve uniform linear heat generation rates.

IRRADIATION FACILITY

The general arrangement of the poolside irradiation facilities in the ORR is shown in Fig. 1. Capsule O5-11 is being irradiated in the O5 position of the ORR poolside facility, the location of which is shown in Fig. 2. Means are available for moving the capsule radially within the neutron flux gradient of the facility to obtain the desired flux level. The measured thermal-neutron flux and gamma-heating rate for this position are given in Fig. 3 as functions of distance from the reactor face. A schematic flow diagram for capsule O5-11 as it was incorporated into the poolside facility is given in Fig. 4. Additional description of the poolside facility is given in a previous report.¹

GENERAL DESCRIPTION OF EXPERIMENTAL ASSEMBLY

The general configuration of the irradiation capsule is shown schematically in Fig. 5. The capsule is suspended from a 1 1/8-in.-OD stainless steel support tube which serves as a lead tube to conduct the thermocouple wires and gas lines from the capsule. The gas lines and thermocouple leads from the capsule are connected to the O5 facility leadout lines. The locations of these lines and their locations relative to the reactor and other poolside facility equipment, such as the valve box, charcoal traps, and control panels, are shown schematically in Fig. 1.

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¹D. B. Trauger, Some Major Fuel Irradiation Test Facilities of the Oak Ridge National Laboratory, USAEC Report ORNL-3574, Oak Ridge National Laboratory, November 16, 1964.

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Fig. 1. General Arrangement of Poolside Irradiation Facilities in the ORR.

Fig. 2. General Arrangement of ORR Experimental Facilities.

Fig. 3. Measured Thermal Neutron Flux and Gamma Heating Rate for ORR Poolside Position 05.

ORNL-DWG 67-9223

ORNL-DWG 67-8616

Fig. 5. General Configuration of Capsule
05-11.

DESCRIPTION OF THE TEST ELEMENTS AND CAPSULE

Capsule Design

The fueled portion of the capsule assembly consists of three separate elements, shown in Fig. 5, which are designated top, middle, and bottom elements. The top fuel element contains a thermocouple well for fuel central temperature measurements. Table 1 is a summary of the design and planned test conditions for each element. A detailed description of the fuel elements is to be reported in the postirradiation examination report. The fuel elements are immersed in NaK, which provides a good liquid heat-conduction medium in which to insert the thermocouples in order to obtain, as nearly as possible, the surface temperatures of the fuel cladding.

The NaK and its blanket gas of helium is enclosed in a vessel, shown in Fig. 5, made of T-111 alloy (Ta-8% W-2% Hf). The choice of T-111 was made because of its high temperature strength properties since the vessel must withstand a pressure of 450 psig at 1260°C. On the OD of the vessel are nine centering spacers 30 mils thick which provide the desired gas gap at operating temperatures. The upper 2 5/16 in. of the vessel, which is above the fuel elements, has a thicker wall to reduce the gas gap in this region. This was done to lower the operating temperature of the brazed joint that seals the seven tantalum-sheathed thermocouples and the two tantalum tubes for the blanket gas, which penetrate the T-111 alloy bulkhead.

The primary container is made of type 304 stainless steel as indicated in Fig. 5. The seven thermocouples for measuring cladding surface and fuel center temperature, the two tantalum tubes for the blanket gas, and the two type 304 stainless steel inner control-gas lines were brazed into the primary container bulkhead.

Table 1. Summary of Design and Planned Operating Conditions
for the Fuel Elements of Capsule 05-11

Fuel element designation	Upper	Middle	Lower
Identification	ORNL 1	ORNL 2	ORNL 3
Fuel element description			
Fuel pellets			
Material	UN	UN	UN
²³⁵ U enrichment, %	20	20	13.9
Number of pellets	10	12	10
Outside diameter, in.	0.300	0.250	0.300
Inside diameter, in.	0.105	(a)	(a)
Length, in.	0.300	0.250	0.300
Cladding			
Material	W-26% Re	W-26% Re	W-26% Re
Outside diameter, in.	0.365	0.365	0.365
Inside diameter, in.	0.305	0.305	0.305
Wall thickness, in.	0.030	0.030	0.030
Length, in.	4.500	4.500	4.500
Internal fluid			
Material	He	Li, He	He
Pressure as assembled, psia	~15	~15	~15
Planned Operating Conditions			
Linear fission heat rate, kw/ft	8.5	8.5	8.5
Fuel cladding temperature, °C	1370	1370	1370
Neutron flux, neutrons/cm ² .sec			
Effective thermal x 10 ⁻¹³	1.16	1.37	1.57
Fast (E > 1 Mev) x 10 ⁻¹²	1.2	1.4	1.6
External cladding pressure, psig	450	450	450
Assembly drawing number	10435-R-002	10435-R-002	10435-R-002

^a Solid pellets.

Calorimeter

The outer container, which is also shown in Fig. 5, serves the dual function of secondary containment for the NaK and as a calorimeter. For the latter function a heavy wall-section is required, and because of its relatively low neutron absorption cross section, Zircaloy-2 was used for this vessel. The principal of operation of the calorimeter is to measure the temperature drop across a continuous metal wall. The Zircaloy-2 portion of the outer container is joined to the stainless steel lead-tube by a special Zircaloy-to-stainless steel transition joint which was used because Zircaloy-2 cannot be welded directly to stainless steel. Transition joints of this type are made by a commercial coextrusion process. At the appropriate step in assembly of the capsule the calorimeter was calibrated. This was done by placing an electrical heater inside the sleeve and then placing the subassembly including the 24 calorimeter thermocouples into a drum of water heated to $43^{\circ}\text{C} \pm 6^{\circ}$ to approximate the temperature of the ORR pool. The heater had three heated sections to simulate the fueled regions of the capsule. By energizing the heater to produce given heat levels and making the assumption that 9% of heat passing through the Zircaloy-2 sleeve is due to gamma heating, the relationship of linear fission heat-generation rate to temperature difference between the inner and outer thermocouples was obtained. Figure 6 is the resulting plot of linear fission heat-generation rate versus temperature difference across the calorimeter wall.

Four pairs of thermocouple junctions are provided on a common plane at each of the three axial calorimeter positions corresponding to the mid-length of a fuel element (see Fig. 5). The average of the four temperature-difference values obtained from the four thermocouple pairs was used to obtain the average heat generation of each fuel pin at its midlength.

Gas Systems

The He pressure over the NaK is held at 450 psig to prevent the NaK from boiling. The pressure of the helium gas, called the inner control gas, in the region between the T-111 sleeve and the primary container is held at 400 psig.

Fig. 6. Calibration Plot for Calorimeter Sleeve Used on Capsule 05-11.
(ΔT is the average temperature difference of the four pairs of thermocouples
for each element.)

The gas gap between the primary stainless steel container and the secondary Zircaloy-2 container is called the outer gas gap. The outer gas gap, which has a width of 1 mil at operating temperature, is filled with helium at 50 psig.

Instrumentation

The surface temperatures of the fuel element cladding are measured by six thermocouples. The central temperature of the upper fuel element is measured by a seventh thermocouple. The locations of all seven thermocouples are given in Fig. 5. The lower 5 ft of all seven thermocouples consists of W-3% Re vs W-25% Re wires surrounded by BeO insulation in a 0.055-in.-OD tantalum sheath. The thermocouples are connected by means of a Conax-type fitting to approximately 12 ft of stainless steel-sheathed type 300N/300P lead wire and then connected through an epoxy seal to glass-insulated copper lead-out wires.

The 24 Chromel-P vs Alumel thermocouples in the calorimeter are sheathed with 40-mil-diam stainless steel and are approximately 30 ft long. After placement of a junction into the Zircaloy-2 sleeve, the metal in the sleeve surrounding it is peened to hold the junction in position. The remaining length of the sheath is carried in a conduit that extends the full length of the capsule assembly and lead tube.

The pressures of the NaK blanket gas and the inner and outer control gases are monitored with strain-gage-type pressure transducers mounted in the gas lines and are continuously recorded.

FLUX MONITORING METHODS

The thermal neutron flux to which the capsule assembly is exposed is measured by two methods. Total thermal-neutron exposure is measured by recovering, after exposure, portions of the 0.005-in.-thick tantalum strips used to strap the thermocouples to the fuel elements. These are analyzed for Ta¹⁸². Instantaneous thermal neutron flux is measured with an argon-activation monitor attached to the exterior of the capsule on the

side away from the reactor core. This facility consists of a stainless steel tube loop with cadmium-shielded inlet and outlet lines. A known flow of argon is established through the loop, and the neutron flux is determined from a gamma spectrometry measurement of the ^{41}Ar in a sample of the exit gas.

ASSEMBLY INTEGRITY ANALYSIS

The fuel is doubly contained in accordance with ORNL policy. The primary container is a stainless steel cylinder having a wall thickness of 0.0865 in. and an outside diameter of 0.8340 in. The secondary containment shell is a Zircaloy-2 vessel 1.430 in. in outside diameter with a 0.2925-in. wall. Maximum allowable working pressures under the ASME Code were calculated for the primary and secondary containment tubes opposite the hot fuel region and compared with actual operating pressures. The results of these calculations are given in Appendix A.

THERMAL ANALYSIS

Linear heat generation rate and temperature of the cladding were specified conditions for this experiment; therefore thermal design calculations were directed toward sizing the insulation gas gaps. In sizing these gaps and obtaining a thermal analysis, the one-dimensional computer program GENGTC² was used. This program was written for in-pile experiments and was ideally suited for the design of this capsule. Data which were specified for the GENGTC program are given in Table 2. The results of the computer analysis including dimensions of the various components and the calculated operating temperature of each surface at design conditions are also given in Table 2.

---

²H. C. Roland, GENGTC: A One-Dimensional Computer Program for Capsule Temperature Calculations in Cylindrical Geometry, USAEC Report ORNL TM-1942, Oak Ridge National Laboratory, September 1967.

Table 2. Input and Output Data for GENGTC Computer Program Used in Design of Capsule 05-11

Data Card	Description	Node Number ^a								
		0	1	2	3	4	5	6	7	8
<u>Input</u>										
600	(See Note 1)									
601	Radius at room temperature, in.	0.15	0.1525	0.1825	0.2525	0.3025	0.3335	0.42	0.423	0.715
603	Emissivity	0.6	0.6	0	0	0.4	0.64	0.6	0.6	0
604	Designator number ^b	1	0	1	1	1	0	1	0	1
605	Gamma heat, w/g	.5	0	.5	.5	.5	0	.5	0	.5
606	Selector material number	9	3	11	4	8	3	6	3	12
		UN fuel	He	W-26% Re	NaK 44	T-111 alloy	He	Type 304 ss	He	Zr-2
<u>Output</u>										
	Radius at design conditions, in.	0.1522	0.1536	0.1838	0.2548	0.3053	0.3356	0.4226	0.4235	0.7159
	Temperature at design conditions, °F	2680	2560	2545	2401	2380	770	673	598	254
	Temperature at design conditions, °C	1471	1404	1396	1316	1304	410	356	315	124

^aSee Fig. 7 for location of node; data given for material inside of node.

^bFor solids and liquids number is 1, for gases number is 0.

Note 1: Data Card 600

Number of nodes	8
Surface heat transfer coefficient, Btu/hr·ft ² ·°F	2 x 10 ⁶
Linear fission heating rate, kw/ft	8.5 (283 w/cm)
Water saturation temp. at surface of capsule, °F	242 (117°C)
Designator number for surface heat transfer coefficient	1
Center hole radius of fuel, in.	0.0525

Fig. 7. Location of Nodes for Input to GENGTIC Program.

NEUTRON FLUX ANALYSIS

A neutron flux analysis was performed to determine the power density that could be obtained with a maximum enrichment of 20%. An effort was made to vary the enrichments in the three elements so as to obtain the same linear heat generation rate in all three elements.

A thermal-neutron flux profile measurement of the 05 poolside position was made using cobalt monitors. Three measurements were made at 1.5, 4.0, and 8.0 in. from the face of the reactor. From these measurements a reasonable picture of the unperturbed thermal-neutron flux of the 05 position was obtained and is shown in Fig. 8.

Use was made of the THERMOS⁴ code to calculate the flux depression from an unperturbed thermal flux to an effective average flux in the fuel as well as cross sections to be used for the fuel. From these results, enrichments were chosen for each fuel element such that the linear heat-generation rate of each element would be the same. The following tabulation gives the results of these calculations:

Element	²³⁵ U Enrichment (%)	Thermal Flux Depression (ratio of unperturbed to average flux in fuel)
Upper	20	0.191
Middle	20	0.198
Lower	13*	0.219

²³⁵U. *The material actually delivered and used was 13.9% enriched

OPERATION AND RESULTS

For startup, the capsule was in the fully retracted position with 100% helium in the gas gap region. After the reactor obtained full power, the capsule assembly was to be inserted until the desired temperatures were

⁴H. C. Honeck, THERMOS: A Thermalization Transport Theory Code for Reactor Lattice Calculations, USAEC Report ENL-5826, Brookhaven National Laboratory, September 1961.

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Fig. 8. Unperturbed Thermal Neutron Flux of the 05 Position 3.0 in. from the Reactor Face (approximate operating position of capsule 05-11).

obtained as listed in Table 1. However, during the first few hours of operation all six tantalum-sheathed thermocouples attached to fuel cladding surfaces (and immersed in the NaK) failed. The mode of failure was determined to be NaK corrosion of the oxygen contaminated end closure welds. The oxygen contamination resulted from making the closure welds in direct contact with the BeO insulation which resulted in the formation of hypostoichiometric BeO. The oxygen released was picked up by the Ta sheath in the region of the weld. The one tantalum-sheathed thermocouple in the fuel (and not immersed in NaK) did not fail. After some delay to revise the mode of operation, the capsule was inserted to achieve design power conditions. The measured linear heat-generation rate for each element (by the calorimeter) was used for selecting the proper insertion position.

Operational results with capsule 05-11 confirmed the essential features of the design in spite of the loss of the cladding thermocouples. The linear heat-generation rate of 8.5 kw/ft was readily obtained although the desired cladding temperature of 1370°C (2500°F) at 8.5 kw/ft was probably not obtained, being approximately 130°C low. When the bottom element operated at 8.5 kw/ft, the heat generation rates of the top and middle elements varied between 6.5 and 7.5 kw/ft during a reactor cycle. This difference is due in part to the fact that 13.9% instead of the desired 13.0% enriched UN was loaded into the bottom element. The calorimeter appeared to operate satisfactorily in giving axial as well as circumferential distributions of heat generation; however, its accuracy is yet to be determined from burnup analysis of the fuel. The general performance of the capsule was satisfactory. The central temperature and calorimeter measurements, along with pressures of helium gas gaps were steady and easily maintained.

A check on the thermal design is indicated by a comparison of calculated temperatures with measured values during the initial operation of capsule 05-11 as shown in Table 3. Increased heat transfer due to natural convection in the NaK and/or in the large He gap could possibly be the reason for the difference between calculated and measured values of central temperature at the higher temperature levels.

Table 3. Comparison of Calculated-to-Measured Temperatures for Capsule 05-11

Fuel Element	Heat Generation Rate as Measured by Calorimeter (kw/linear ft)	Calculated Temperature of Cladding (°C)	Measured Temperature of Cladding (°C)	Central Temperature (°C)	
				Calculated	Measured
Top	2.4	700	670	772	800
	7.2	1274		1402	1210
Middle	2.8	751	728		
	7.4	1293			
Bottom	3.6	873	800		
	8.5	1396			

CONCLUSIONS AND RECOMMENDATIONS

The following conclusions are drawn from the operation of capsule 05-11:

1. The six cladding surface thermocouples should be sheathed with an alloy of refractory metals other than tantalum to avoid the corrosion problem of oxygen contaminated tantalum in a NaK environment. Nb-1% Zr has been selected as a likely material for this application. In addition a Nb-1% Zr plug should be used to separate the BeO from the weld region. Following the end closure weld, we recommend a two-hour anneal at 1200°C to tie up the oxygen with the zirconium.

2. The ^{235}U enrichment of the lower element should be reduced from 13.9 to 12.9% to flatten the power and temperature gradient.

3. The inner helium gas gap should be increased from 30 to 35 mils at design conditions to increase the fuel cladding temperature $\sim 100^\circ\text{C}$ at an 8.5 kw/ft linear heat-generation rate.

4. The material of the NaK vessel should be changed from T-111 to Nb-1% Zr to avoid possible compatibility problems of tantalum and oxygen. This will necessitate joining the NaK blanket gas system and inner control gas system to eliminate the differential pressure across the weaker Cb-1% Zr wall.

ACKNOWLEDGEMENTS

The results reported in this document are due to efforts of numerous people in the Metals and Ceramics and Reactor Divisions of the Oak Ridge National Laboratory. In particular, the authors gratefully acknowledge the contributions of T. G. Chapman who made the stress analysis calculations and E. A. Franco-Ferreira who specified and made the braze joints in this capsule.

APPENDIX A

Appendix A

STRESS ANALYSIS FOR CAPSULE 05-11

The temperatures for the containment vessels as obtained from the ASTRA¹ heating code (see Fig. A-1) were utilized in the analysis.

Primary Container

1. Material — Type 304 H stainless steel.
2. Maximum service temperature — 1100°F (593°C).
3. Material stress intensity, S_m — 9,450 psi (from ASME Pressure Vessel Code, 1965 Summer Addenda for 1100°F).

4. Stress intensities

Maximum internal pressure — 600 psi

Primary membrane stress, P_m 2,620 psi

Primary stress intensity, P_L 3,550 psi

Maximum thermal stress, Q_t 17,750 psi

Maximum discontinuity stress,   1,750 psi

Q_d

Maximum combined stress,   21,000 psi

$P_L + Q_t$

5. Comparison of stress intensities with allowable stresses

$P_m = 2,620$ psi $S_m = 8,250$ psi $P_m < S_m$

$P_L = 3,550$ psi $1.5 \times S_m = 12,375$ psi $P_L < 1.5 \times S_m$

$P_L + Q_t = 21,000$ psi $3.0 \times S_m = 24,750$ psi $P_L + Q_t < 3 \times S_m$

Therefore, the primary container meets intent of ASME Code, Section III, 1965.

6. External collapse pressure

Maximum external pressure   600 psi

Collapse pressure   600 psi

Therefore, the primary container will not collapse.

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¹R. R. Liguori, ASTRA: The Heating Program, Report 417-50, Astra Inc., Raleigh, North Carolina, January 1, 1961.

Fig. A-1. Breakdown of Capsule 05-11 for the ASTRA Heating Code. Numbers shown are regions specified for the code.

Secondary Container

1. Material - Zircaloy-2.
2. Maximum service temperature - 650°F (343°C).
3. Material stress intensity for Zircaloy-2.

The rules for establishing stress intensity values of nonferrous materials given in paragraph UA-601, ASME Pressure Vessel Code, Section VIII, 1965 ed., were used as a guideline in arriving at a value to be used in this analysis. The rules require the least of the stress values to be the allowable stress intensity at 650°F (343°C).

1/4 ultimate stress	6,800 psi
2/3 yield stress	9,300 psi
0.1% creep in 10,000 hr	9,000 psi
100,000/hr rupture stress	16,000 psi

Therefore, the allowable stress intensity at 650°F is $S_m = 6,800$ psi.

4. Stress intensities (only material not penetrated by thermocouple holes considered).

Maximum internal pressure -	600 psi
Primary membrane stress, P_m	6,620 psi
Primary stress intensity, P_L	7,240 psi
Maximum thermal stress, Q_t	10,110 psi
Maximum discontinuity stress, Q_d	5,160 psi
Maximum combined stress, $P_L + Q_t$	17,350 psi

5. Comparison of stress intensities with allowable stresses

$P_m = 6,620$ psi	$S_m = 6,800$ psi	$P_m < S_m$
$P_L = 7,240$ psi	$1.5 \times S_m = 10,200$ psi	$P_L < 1.5 \times S_m$
$P_L + Q_t = 17,350$ psi	$3 \times S_m = 20,400$ psi	$P_L + Q_t < 3 \times S_m$

Therefore, the secondary container meets the intent of ASME Code, Section III, 1965 ed.

Cyclic Operation of Secondary Container

The number of pressure and temperature cycles during operation is not expected to exceed the following numbers:

Full range pressure cycles	60
Temperature cycles (25% to 100%)	200

Under the conditions specified, the capsule is not subject to analysis for cyclic operation as per requirements of paragraph N451.1 of the ASME Code, Section III, 1965 ed.

Pneumatic Testing

Pneumatic test pressures required by paragraph N-715 of the ASME Code are:

Primary containment	maximum 1700 psi minimum 1640 psi
Secondary containment	maximum 1875 psi minimum 1800 psi
Maximum external pressure on primary container	1400 psi

for a test temperature of 100°F (38°C).

Based on the results reported above, with proper fabrication and inspection assumed, Capsule 05-11 meets the intent of the ASME Pressure Vessel Code.

Internal Distribution

1. S. E. Beall	30. A. J. Miller
2. R. L. Beatty	31. A. M. Perry
3. E. S. Bomar, Jr.	32. R. A. Potter
4. C. M. Burton	33. P. Patriarca
5. T. G. Chapman	34. M. W. Rosenthal
6. J. A. Conlin	35. D. E. Rosson
7. J. H. Coobs	36. A. W. Savolainen
8. W. B. Cottrell	37. J. L. Scott
9. F. L. Culler	38. R. L. Senn
10. J. W. Cunningham	39. M. J. Skinner
11. V. A. DeCarlo	40. G. M. Slaughter
12. W. S. Ernst	41. K. E. Spear, II
13. R. B. Fitts	42. I. Spiewak
14. A. P. Fraas	43. J. G. Stradley
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