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An Assessment of Fuels for Nuclear Thermal Propulsion

by S. K. Bhattacharyya

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TECHNOLOGY DEVELOPMENT DIVISION TECHNICAL MEMORANDUM
NO. 2

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EXECUTIVE SUMMARY

Studies dating back to the late 1940s performed by a number of different organizations have established the major advantages of Nuclear Thermal Propulsion (NTP) systems particularly for manned missions. NTP systems are inherently superior to chemical propulsion systems because of the greater energy potential per unit mass of nuclear fuel, and the fact that use of hydrogen coolant/propellant helps attain high exhaust velocities relative to that achievable by the higher molecular weight combustion products characteristic of chemical systems. Specific advantages of NTP systems demonstrated by analyses include:

- Reduced transit times for long stay missions for the same initial mass to low earth orbit (IMLEO). This minimizes crew exposure to galactic cosmic rays, solar flares and zero gravity.
- Reduced round trip times for short stay missions for the same IMLEO.
- Reduced IMLEO requirements for the same mission duration.
- Allowance for greater mission flexibility. This includes increased Earth and Mars departure windows and increased propulsion margins available for mission aborts.

In recent preliminary technology evaluations for the In-Space program for NASA, NTP systems ranked low for near term applications, although their importance for longer term robotic science missions was recognized. The bimodal systems, which provide moderate thrust NTP functions, and also produce steady state power (at the tens of kilowatts levels) during non NTP operations, ranked significantly higher for near term applications.

The principal technology element for NTP and bimodal systems is the nuclear fuel. The performance requirements on the fuel are extremely demanding. They have to operate for mission durations up to two hours at temperatures of near 2800 – 3000 K, during which time they need to maintain geometric and physical integrity. The fuel has to be able to withstand multiple restarts, and, in the case of bimodal systems, transient operation with inventories of fission products built up during the low power steady state operations.

There are two classes of fuels that have the potential to meet the demanding requirements. The first class is carbide based fuel (sometimes imbedded in graphite). The second class is ceramic fuel (UO_2 or UN) imbedded in a high temperature metallic matrix (W, Mo or Re). The potential of these fuels has been recognized from the earliest days and they have been used or considered for every NTP program that has been conducted over the years. An early decision was made in favor of the carbide based fuel (from ease of fabrication and low thermal-neutron absorption cross-section considerations) and the bulk of the development work done to date has focused on these fuels. This includes the two largest programs, the Rover/NERVA program in the US and the Soviet program. Several billion dollars were spent on each of these two programs and considerable fuels development work was undertaken. In addition, a number of reactor tests were conducted with complete fuel assemblies. Figures 1 and 2 show photographs of the Rover/NERVA composite fuel and the Russian twisted ribbon fuel. Lack of well defined missions caused both programs to be terminated without actual flight tests of NTP systems; however sufficient work was performed to demonstrate the feasibility of NTP systems with specific impulse Isp values in the mid 800 seconds range. The more recent Space Nuclear Thermal Propulsion (SNTP) Project sponsored by the US Air Force also selected carbide fuel; however a very limited test program was run prior to its termination. Table I shows a summary listing of the fuels considered for the various NTP programs.

The tungsten based cermet fuels were developed in the US as a back-up to the reference carbide fuels. A much more modest program than for the carbide fuels was undertaken. However the data collected were very promising. Long term exposure to hot flowing hydrogen (non nuclear) showed no failure modes for periods exceeding lifetime requirements. Steady state nuclear irradiation up to burnups of up to 1.6% showed excellent performance. Transient tests for short periods (0.2 s) at very high heat rates (up to 16000 K/s) demonstrated the robustness of the fuel. While more development is needed, sufficient data exists to provide indications of potential. Figure 3 shows photographs of cermet fuels fabricated for the ANL cermet fuel program.

Table II shows a summary of the best performances achieved by the various fuels in the NTP programs. The temperature and associated lifetimes represent impressive levels from which advanced fuels can be developed. It should be noted here that the information on diagnostics of the Russian twisted ribbon fuel is limited. While there is no question that impressive successes have been obtained by the Soviet researchers, the lack of diagnostic data will necessitate additional work to validate some of their results.

Table III summarizes our assessment of the key classes of fuels. The highest temperature (and hence specific impulse, Isp) potential is in the binary or ternary carbide fuels. These fuels also need significant additional effort to reach practically usable forms. The largest relevant database exists for the (U,Zr)C,C composite fuel. For near term bimodal and NTP applications, this fuel should be a good choice. Also, despite a lack of significant nuclear test data for appropriate lifetimes, the cermet fuels show significant promise and should be a good choice for near term bimodal and NTP applications as well. Our recommendation is to focus the bulk of the current development resources on the composite and cermet fuels since they can meet all the projected near to intermediate term requirements. Smaller investments should be made on all-carbide fuels (in universities) to capitalize on their potential for the future.

Figure 1. ROVER/NERVA Composite Fuel

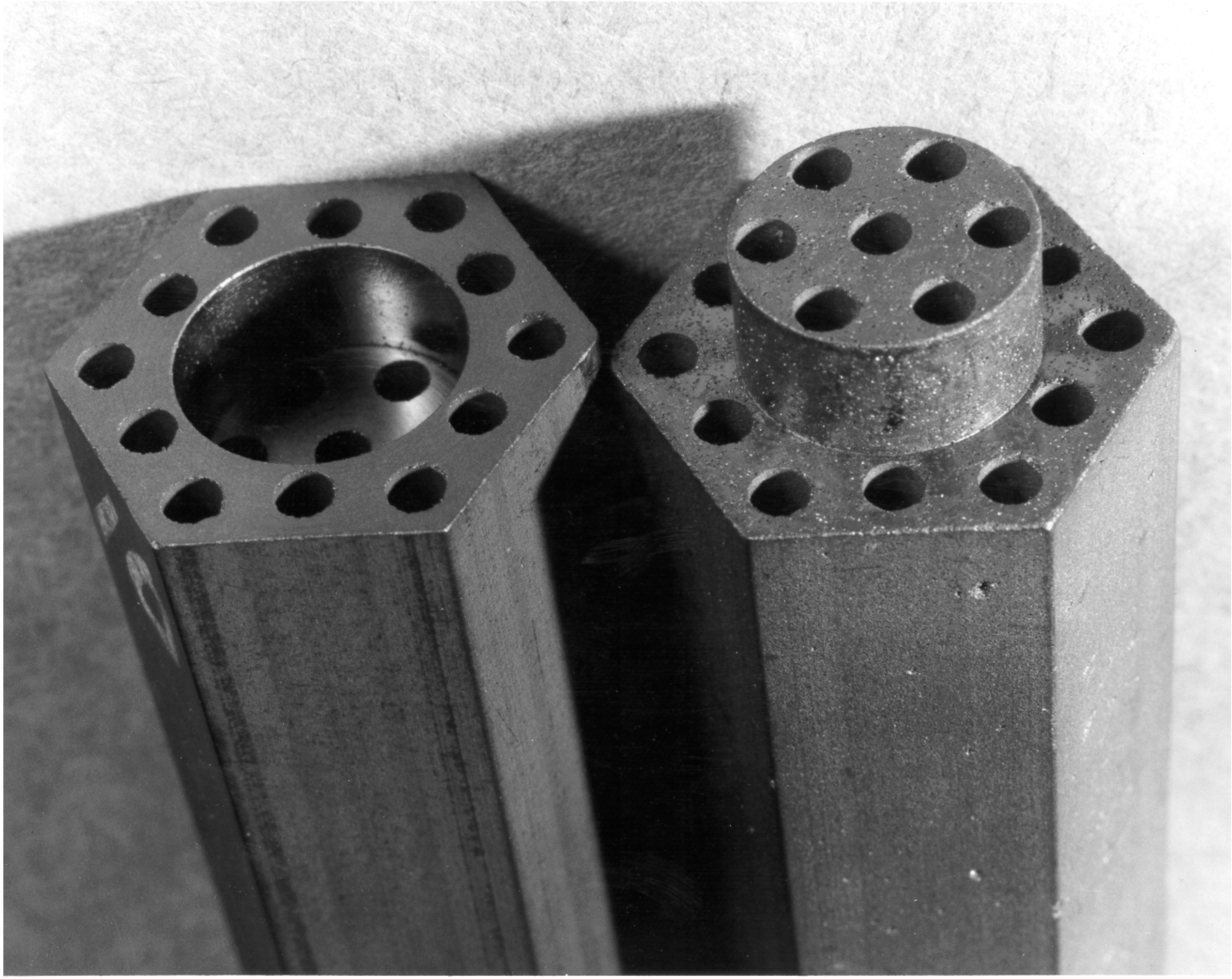


Figure 2. Russian Twisted Ribbon Fuel Located in Fuel Assembly

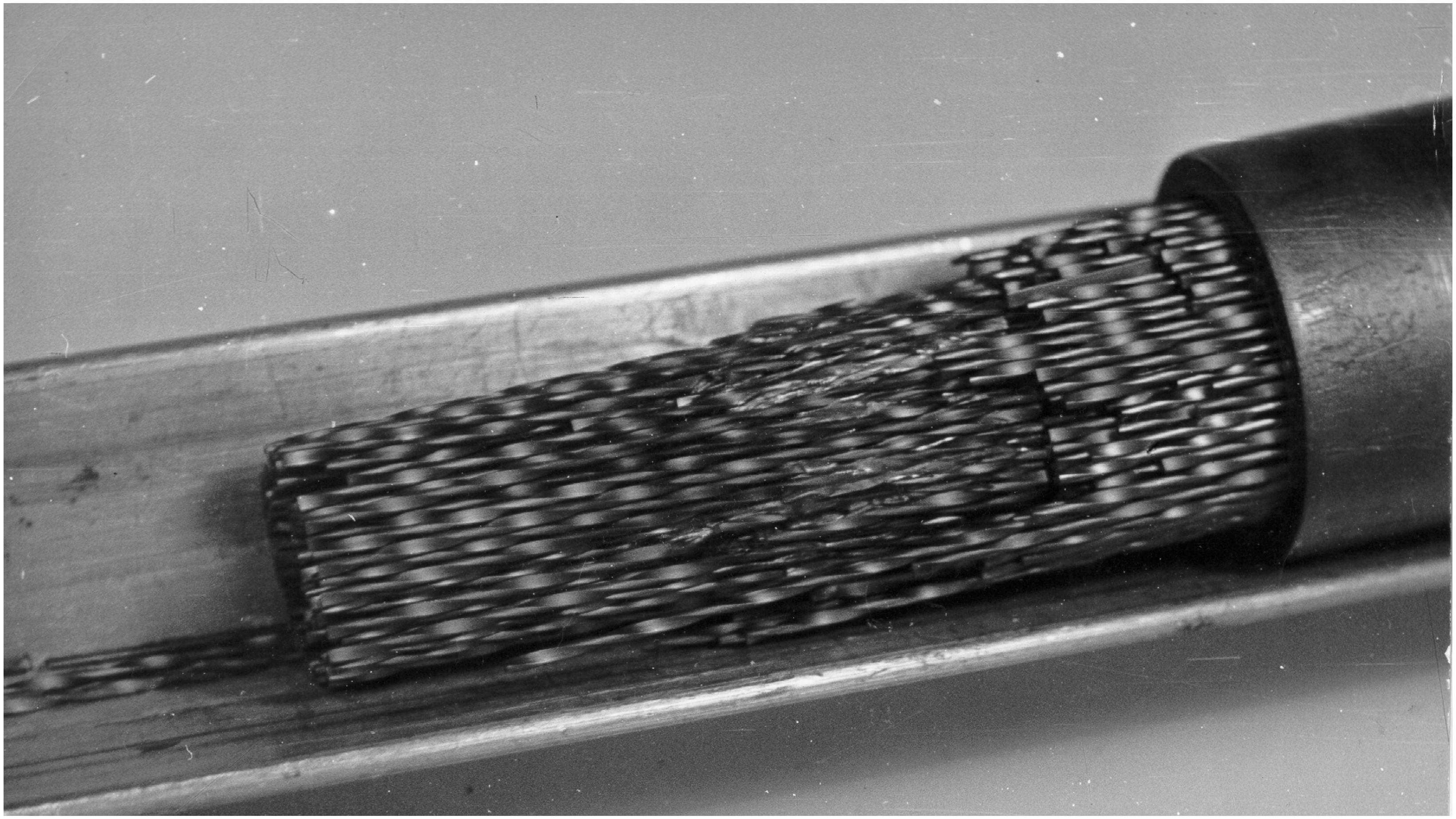


Figure 3. UO_2-W Cermet Fuel Samples Prepared for the ANL Cermet Fuel Program

Table I. NTP Fuel Description

Concept/Program	Fuel Composition	Fuel Geometry
ROVER/NERVA	UC_2 in graphite (U, Zr) C, graphite (composite) (U, Zr) C, solid solution	Solid blocks with coolant channels
PBR	Coated UC_2 Binary, ternary carbides	Particle
CIS	Coated (U, Zr, Nb) C (Other ternary carbides) Carbonitrides	Twisted ribbon
Cermet	UO_2 in W	Solid blocks with coolant channel

Table II. NTP Fuel Performance Parameters (Nuclear Test Data – Best Performance)

System	Fuel Type	Coating/ Clad	Temp. K	Lifetime	# of Cycles	Year
ROVER/ PEWEE 1	Pyrolytic C-UC_2 Particle Dispersion	ZrC	2600	40 min.	1	1968
ROVER/ NF-1	(UC, ZrC)C Composite (U, Zr)C Solid Solution	ZrC	2450	109 min.	6	1972
Cermet	UO_2	W	2700	0.2 s	8	1967
Russian	(U, Zr, Nb)C, C	?	3100	4000 s	?	1980’s

Table III. Summary of Assessment of Fuels

Fuel Type	Features		Recommendation
	Positive	Negative	
(U, Zr)C,C Composite Fuel			
Prismatic	• Most experience • Proven operating experience at 2550K for 109 m.	• Limited steady state irradiation experience • Thermal shock and cracking problems	Pursue for near term applications
Mixed Carbide Fuels	Highest temperature potential		Pursue for longer term applications
Prismatic	• In reactor experience	• Extensive cracking in tests • Limited experience • Fabrication process development needed	
Particle Bed	• High T/W operation temperatures	• Limited experience • Fuel/FP loss anticipated • Mechanical Issues	Subset of above (same fuel type but geometric differences)
Twisted Ribbon	• Potential large database • Excellent high temperature performance reported	• Long effort at getting quantitative data not very successful • Uncertain mass losses	Develop Russian cooperative program
Cermet Fuels	• Robust fuel • Safety features • Steady state operating experience (bimodal and/or surface power)	• Ultimate temperature potential is lower than mixed carbide	Pursue for near term (also potential application in intermediate term MMW steady state system)

1. INTRODUCTION

The earliest references to Nuclear Thermal Propulsion can be found in the papers written by Dr. Robert H. Goddard et. al., before World War II (1). In around 1944, a scant two years after the demonstration of a controlled nuclear chain reaction, scientists at Los Alamos Scientific Laboratory (LASL) and the University of Chicago Metallurgical Laboratory (the forerunner to Argonne National Laboratory) discussed the use of fission "piles" to heat rocket propellant fluids to high temperatures (2). Following WWII there were a number of studies performed to look at nuclear power and propulsion in space. Most of these, in keeping with the world situation at the time, had a decidedly military bent. The 1945 study of nuclear rocket propulsion undertaken by the US Air Force Scientific Advisory Board, under the direction of Theodore Von Karman and H.S. Tsien recommended no action in this area at that time because of perceived technical difficulties. NACA and APL reached similar conclusions. The primary reason for the pessimistic conclusion was the large gap between projected fuels and materials needs and the demonstrated capabilities at the time.

These negative assessments did not completely dampen the enthusiasm of visionaries' thinking and several follow on papers were written that probed feasibility questions further. Interest was sparked in long duration flight nuclear aircraft and two programs were conducted in this area – NEPA (Nuclear Energy for the Propulsion of Aircraft, 1946-51) and ANP (Aircraft Nuclear Propulsion, 1952-68). R.W. Boussard is generally credited with restoring real interest in nuclear rocketry, during his work on the nuclear aircraft program (3,4). His work led to Von Neuman's support to the program. The nuclear thermal propulsion program officially began in 1955 at LASL (now LANL) and LRL (now LLNL). For an excellent summary of the history of the development of nuclear thermal propulsion in the United States, the reader is referred to Bennet et al (5).

One of the first technical questions that arose in the program was the choice of fuels, since it was clear, even then, that this would be the most complex technology choice. There were two paths that the technical people considered: 1) Graphite based fuels (UC_2 in C) and 2) Refractory alloy based fuels (UO_2 in W or W-Re). Both fuel types had the requisite properties of high temperature strength and fabricability with uranium fuels. The primary reason for the initial selection of graphite-based fuel was the high thermal absorption cross-section of tungsten, its weight, and perceived difficulties in fabrication. The major problem with graphite – its reaction with hot hydrogen – was considered to be more easily solvable with coatings.

Overall Requirements for NTP Fuels

Requirements for NTP fuels flow from mission requirements for the entire system. Over the past several decades, a number of potential missions have been postulated and analyzed and the following attributes emerge as very desirable attributes for candidate fuels:

- High Fission Product Retention
- Thermal Stability (low mass loss)
- High Melting Point (>3200 K)

- High fuel Density (Su > 10%)
- Thermal Shock Resistance
- Slow Degradation Mechanisms
- Chemical Compatibility with Coatings
- High Surface Area to Volume Ratio
- Fabricability

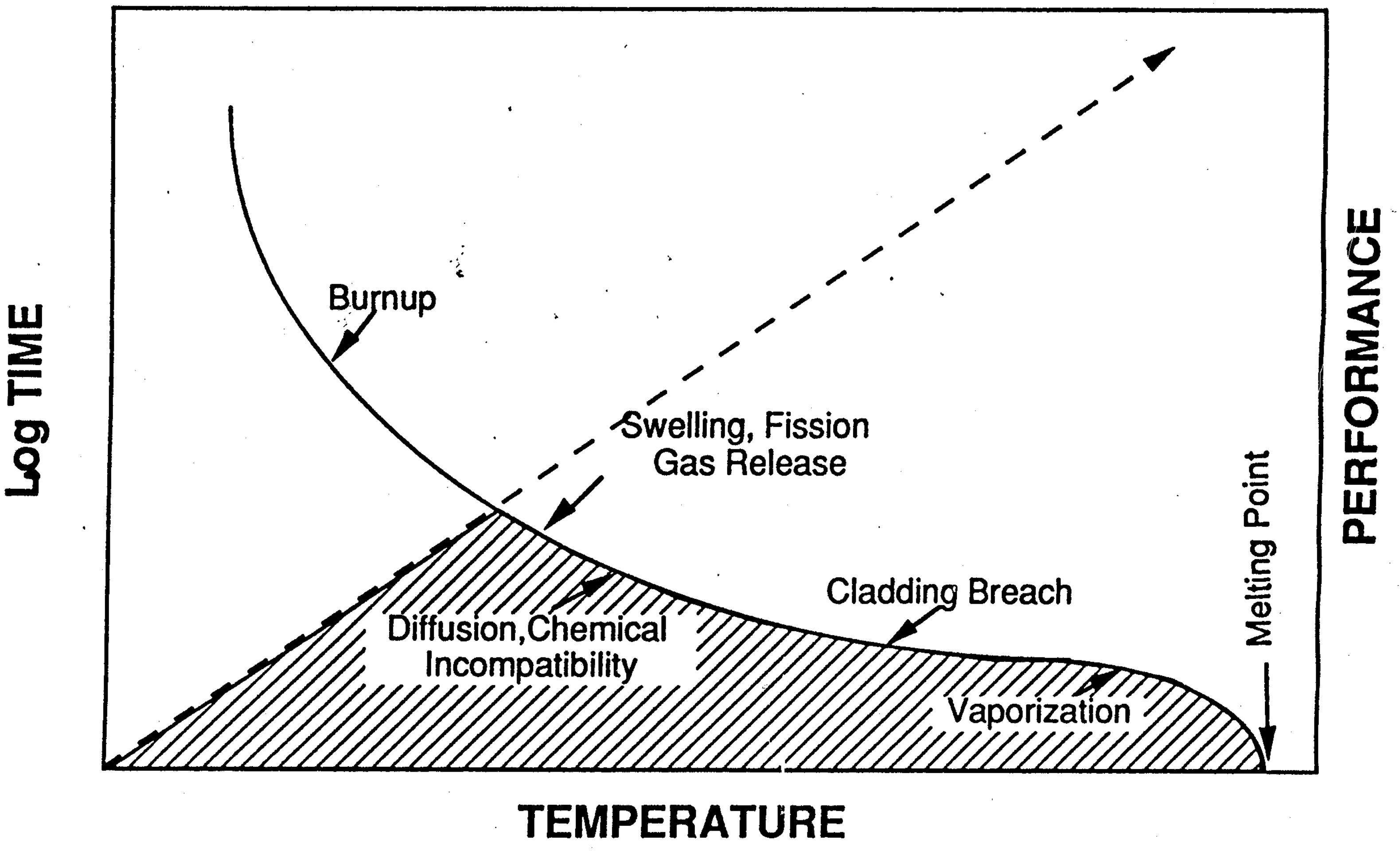
Safety, reliability and performance will need to be optimized for future NTP systems. Safety requirements will drive fuel choices to those with high fission product retention, coolant-clad-coating compatibility, high temperature stability, high melting point and high thermal shock resistance. Performance requirements drive fuel choices to those with high uranium inventory, high operating temperature, high thermal conductivity and high temperature stability. Thus, assured operating lifetimes at the highest possible temperatures emerge as the key requirements for NTP fuels.

Figure 1.1 shows a schematic representation of lifetime and performance vs. temperature behavior of nuclear fuels (6). The basic physical phenomena that affect fuel behavior are listed. Fuel melting is the ultimate limit for most concepts, but the actual operating limit is considerably lower in temperature, governed by vaporization of fuel and coatings leading to significant mass loss. It is noted that higher lifetimes can be achieved at lower temperatures, albeit with sacrifice of performance. At “low” temperatures (at which NEP and surface power systems will operate), orders of magnitude longer lifetimes are anticipated, and the life limiting mechanism is high burnup-induced clad failures.

The <u>bimodal</u> reactor concept (one which operates in both the thermal propulsion mode for several cycles during a mission and also a low power steady state mode for extended periods between those cycles) is being re-examined seriously by mission planners. The low power steady state operation can supply both housekeeping power for the spacecraft, (thus eliminating the need for an auxiliary power source) and primary power for electric propulsion. This bimodal operation introduces additional requirements on the fuel - principally that the high temperature operations need to be conducted with a non-negligible fission product burden created by the burnup of the fuel in the course of the steady state operation. It is understood that the power level (and hence burnup) in these systems will be low; however, the impact on peak NTP operating temperatures must be assessed.

Practical realities could introduce another consideration into the NTP fuel selection process. The development and qualification of fuels represents one of the largest cost and schedule items in a space nuclear program. It makes sense to look for fuels that have a broad range of applicability (perhaps with some sacrifice of performance for specific missions), rather than to develop individual fuel types for niche missions. Fuels that are candidates for bimodal applications and have the potential for both NTP and high power steady state applications deserve special attention from development cost considerations. However, to date, this aspect of optimal development strategies has not been emphasized.

Figure 1.1
Conceptual Trade Off Between Performance and Efficiency



Fuel composition and melting point will change during reactor operation by evaporation, corrosion, diffusion, fission product migration, and chemical interactions

2. MAJOR CLASSES OF FUELS

As shown in Table 2.1, a large number of fuel compositions have been proposed in the past for NTP applications. All of the solid core fuels are based on the three commonly known fuel components, oxides, nitrides and carbides of uranium. For all cases U^{235} has been used as the fissile isotope, although interest in U^{233} has remained because of potential improvement in core compactness. The use of plutonium fuels for fission reactor systems has not been considered in space. A further narrowing of the classification of the fuels is achieved by noting that for NTP applications carbide fuels have generally been associated with graphite matrices and the UO_2 and UN fuels with metal matrices. Thus, the range of NTP fuels for solid cases (that encompass the bulk of the concepts studied) can be categorized as:

1) Carbide (graphite matrix)
2) Oxide/Nitride (metal matrix)

There are several NTP concepts which have been studied in detail over the past decades. The relevant programs are listed in Table 2.2. The first four concepts had the benefit of significant in-reactor and non-nuclear hot hydrogen tests. Figure 2.1 shows schematic sketches of the four concepts for which designs were developed by industrial contractors. In three of these four concepts carbide (graphite) fuels were selected. The geometric forms of the fuels varied in the three cases, but the basic fuels were variants of carbide compositions. The fourth concept used UO_2 fuel in a W matrix. Table 2.3 summarizes the overall features of the fuels. It should be noted that consideration has also been given to nitride fuels in cermets. Because of the need to maintain nitrogen overpressure in the fuels to impede the dissociation of UN, it has been postulated that the use of carbonitride fuels might be the most effective way to utilize the uranium density advantages of these fuels in a cermet form. There is some reported work with these fuels in Russia, although the basic application was with ammonia propellants (i.e. N_2 containing gases).

In the following sections, the technology status of these four classes of fuels will be discussed in detail.

Table 2.1
Basic Fuels Proposed for NTP Applications

- UO_2
- UN
- UC_2 (with TaC, ZrC, NbC coatings)
- (U, Zr)C-C Composite
- UO_2 or UN cermets/W, W-Re or Mo
- (U, Zr)CN
- (U, Zr, Nb)C Solid solution
- (U, Zr)C Solid Solution ; X = Ta, W, Hf
- Liquid core U
- Gas core U, UF_4, UF_6

(U is generally ^{235}U ; however ^{233}U is an option)

Table 2.2
Past Programs on Space Nuclear Thermal Propulsion

I. Significant Programs with In-pile Tests

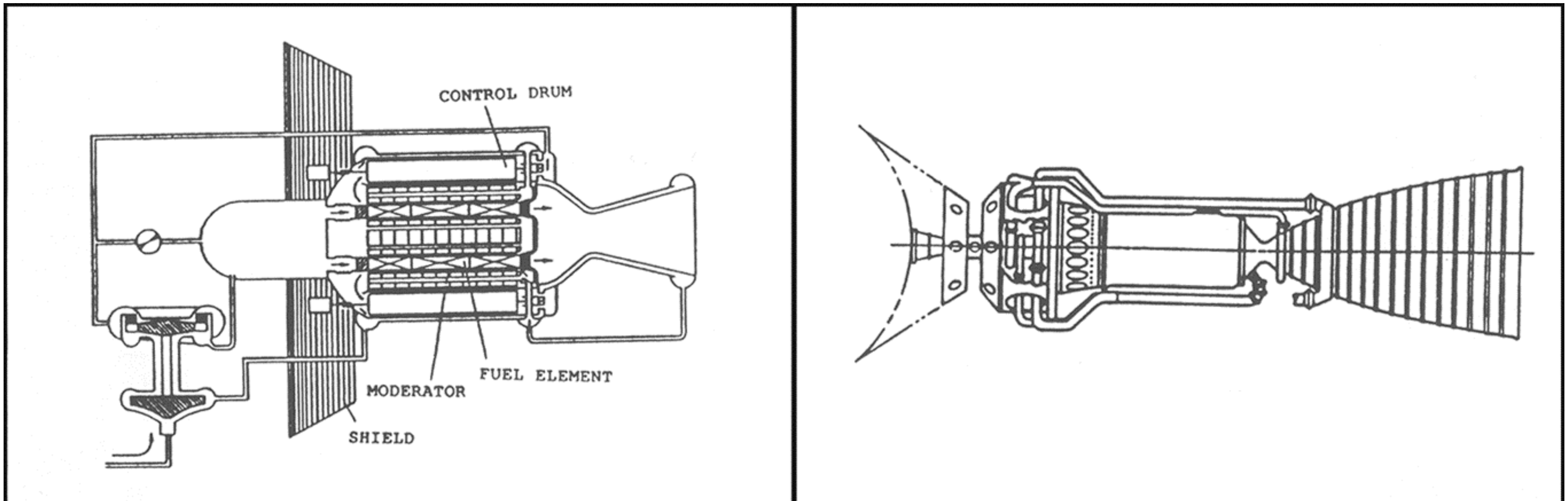
<u>Propulsion</u>

- Rover/NERVA
- Russian Program
- Particle Bed System (SNTP)
- Cermet Fuel System

II. Study Programs

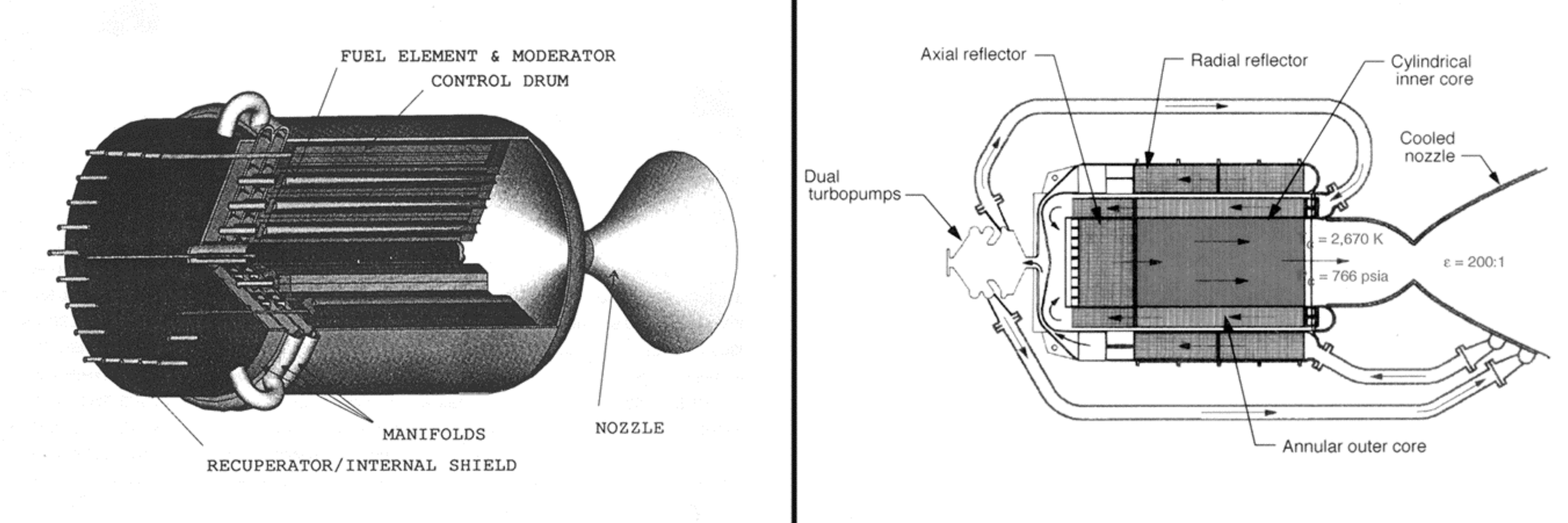
- MMW
- Space Exploration Initiative (SEI)
- Several NASA Studies
- Etc.

Figure 2.1
Nuclear Thermal Propulsion Concepts



CIS: "Twisted Ribbon" (Aerojet/Energopool/B&W)

NERVA - Derived (Rocketdyne/Westinghouse)



Particle Bed (Aerojet/B&W)

CERMET (Pratt & Whitney/B&W)

Table 2.3
NTP Fuel Description

Concept/Program	Fuel Composition	Fuel Geometry
Rover/NERVA	UC_2 in graphite (U, Zr) C, graphite (composite) (U, Zr) C, solid solution	Solid blocks with coolant channels
PBR	Coated UC_2 Binary, ternary carbides	Particle
CIS	Coated (U, Zr, Nb) C (Other ternary carbides) Carbonitrides	Twisted ribbon
Cermet	UO_2 in W	Solid blocks with coolant channel

3. DISCUSSION OF TECHNOLOGY STATUS OF NTP FUELS

3.1 ROVER/NERVA

The Rover program started in 1955 at LANL (then LASL) and LLNL (then LRL). The complexities of a nuclear rocket design was recognized at that early stage and carbide fuels in graphite matrices were selected for reasons of weight advantage, fabricability and perceived advantages in high temperature behavior (7). Initial interest in a ground test site was in the ANP test facilities in Idaho, but on more detailed evaluation, the Nevada Test Site (NTS) proved to be more suitable. In order to curtail costs, the program was localized at LANL in 1956. Principal subcontractors included the Rocketdyne Division of North American Aviation, Aerojet General, ACF-Erco and EG&G, plus several small contractors (8). LANL's approach was two pronged:

- A limited research program with advanced propulsion systems and material
- A specific basic reactor testing effort, called the KIWI program

In 1961, NASA and the AEC awarded the NERVA (Nuclear Engine for Rocket Vehicle Applications) contract to Aerojet and Westinghouse to develop the RIFT (Reactor-in-Flight Test) flight system, based on the LANL KIWI-B design. The RIFT program was cancelled in 1963, and the NERVA program was redirected to be a reactor ground test program. This was a significant blow to the development of NTP systems, since RIFT had been planned to provide a backup to the chemical third stage on the Saturn V as well as the primary means to conduct post Apollo manned missions to the Moon and beyond.

The overall program, at its cancellation in 1972, had spent $1.4B (in 1972 dollars). It had designed several reactor engines and built and tested 20 of them. Of them 17 were test reactors, one safety reactor and two were ground test engines. Figure 3.1.1 (9) shows a chronological view of the tests and Table 3.1.1 presents additional data on the tests. Not unexpectedly, as the program evolved, a large number of problems were encountered in all areas of system design and operation (10). Solutions were developed and tested for most of them.

3.1.1 Summary of Development of Carbide Fuels

The focus of this paper is on NTP fuels. It is interesting to note that the challenging constraints on fuel designs were recognized early in the program (11). The first challenge was to develop a fuel material that could produce the fission energy needed without loss of mechanical integrity. Given that fuel material, it was still necessary to be able to reproducibly fabricate fuel shapes that were rugged enough to handle the violent mechanical and thermal stresses and yet thin enough to reduce thermal gradients and provide large surface areas for heat transfer.

Carbon based fuels have been researched since the earliest days of nuclear power. There are a number of significant advantages of such fuel, the principal one being excellent retention of fission products coupled with the high temperature capability and fabricability. On the other hand, there are also the disadvantages of reactions with hot hydrogen, inherent weakness, brittleness and response to thermal stresses.

Four distinct classes of fuels were used in the Rover program (12) as depicted schematically in Figure 3.1.2. The initial attempt was to produce plate geometries of UC_2 mixed in with graphite. An NbC coating was used to keep the hot hydrogen separated from the graphite. This fuel type performed poorly with rapid ejection of fuel fragments into the exhaust gas stream. This failure mode was attributed to structural cracks in the fuel caused by rapid fission gas build up followed by local decrease in thermal conductivity and strength. The problem was solved by coating the UC_2 particle with a thin pyrolytic graphite coating.

The fuel element designs were modified to provide minimum distance between coolant passages (in order to reduce thermal gradients) while simultaneously preserving structural integrity. The resulting geometry was the now famous long cylindrical tubes of small diameter, with multiple coolant holes running axially throughout the fuel.

Early work was performed with UC_2 and UO_2 spheres imbedded in graphite. The effects of fission product interactions with the graphite matrix very quickly led to the use of pyrolytic graphite coated UO_2 particles imbedded in graphite in much the same manner that the HTGR fuels development was progressing. This solved the problem of the cracking of fuel elements that occurred very early in the irradiation tests. The pyrolytic graphite was able to contain fission products prior to their causing dislocations in the graphite structural materials and initiating cracking. The pyrocarbon coating also provided protection against oxidation of the UO_2 particles. The graphite matrix was isolated from the flowing hot hydrogen with a high temperature carbide coating as shown in Figure 3.1.2. The early choice of coatings was NbC. Such dispersed fuels were the most highly developed during the program and provided the larger bulk of the test data. Performance of the fuel was generally good.

The problem of “mid band” corrosion, attributed to the larger mismatch in the coefficient of thermal expansion between the fuel material and the NbC coating, led to the development of (U,Zr)C, in graphite composite fuel. The coefficient of thermal expansion for the appropriate composite material was increased to 6-6.5 :m/m°K from 3 :m/m°k for the dispersed fuel material. This increased value was a better match to the 7.1 :m/m°k CTE value for NbC, and thereby reduced the corrosion cracking problem. The composite fuel with 30-35% (U,Zr)C proved to be much more robust and chemically stable than the less loaded fuel rods because of the fact that both the carbide and graphite components were continuous and interconnected.

The outer coating material was changed to ZrC from NbC because of the demonstrated superior resistance to fission product diffusion at high temperatures. The mid band corrosion problem was reduced to manageable levels. Figure 3.1.3 shows the progressive reduction carbon loss with improvements in fuel design.

As a final step in the fuels evolution, single phase (U,Zr)C was developed to achieve higher temperatures (by eliminating the limit set by the carbide-carbon eutectic formation). The all-carbide solid solution compounds have significantly higher melting points and potentially a stronger physical structure. Only seven such fuel rods were fabricated and tested in the program, so the database is insufficient to evaluate the performance of these fuels. The tentative conclusion based on available data are mixed: the solid solution fuel tended to crack, was difficult to fabricate and showed corrosion effects in flowing hydrogen (13). However, based on the phase diagrams, single phase carbide fuels have the greatest promise among solid fuels for the attainment of highest temperatures for NTP applications.

Figure 3.1.4 shows a photograph of the Rover/NERVA fuel element. The comprehensive test program conducted for the program has been described in great detail elsewhere (14,15) and a summary of the evolution of the reactor fuel test condition and fuel times over the direction of the program is shown in Table 3.1.2. Listed are many of the key parameters of interest to NTP design (fuel temperature, Isp, time at full power, power and thrust) for the various fuels. Clearly a broad range of parameter space was covered in these experiments (16). A summary of the best test performances achieved in the Rover/NERVA program is presented in Table 3.1.3 (17).

Figure 3.1.5 shows a qualitative comparison of the projected useful lifetimes of the various Rover/NERVA fuels as a function of the hydrogen exit temperature. This is based on available data and calculated extrapolations (particularly for the carbide fuel). This evaluation, performed by Koenig (18), took into account the thermomechanical behavior of the actual fuel elements used in the reactor.

3.1.2 Assessment of Performance Potential for Carbide Fuel

As was shown schematically in Figure 1.1, the ultimate limit on conventional fuels is the melting point of the fuel. The melting points of UC, UC_2, UN and UO_2 range from 2710°K to 3100°K. One way to increase the operating temperature of the fuels is to produce solid solutions compatible compounds with much higher melting points. This is possible for UC which forms stable solid solutions with various carbides (ZrC, TaC, HfC, etc.) all of which have significantly higher melting points. From an examination of the phase diagrams of the various carbide constituents it is clear that stable high melting point solid solutions can be produced. As an example, Figure 3.1.6 shows a psuedobinary solidus liquidus phase diagram for UC-ZrC (19). This diagram is a projection across the ternary phase diagram from UC to the maximum melting point of ZrC_x, and thus represents the maximum solidus and liquidus points. It is noted that the specific melting point value is a function of the composition and is at the highest point at zero UC concentration. This illustrates the fact that low concentrations of uranium might be required to achieve the high melting points. Lowering the uranium concentration affects the criticality of the system (negatively) and this imposes a practical limit to which the melting point of these fuels could be raised.

A second significant point can be observed from isothermal projections of the U, Zr, C phase diagram (Figure 3.1.7). The figure shows the projections at 2600 C and 3000 C (19). It is observed that there is a narrow composition range in which the solid solution is stable, and this range shrinks dramatically as the temperature is increased. The actual melt temperature depends sensitively on the stoichiometry of the carbide compound, and there are several mechanisms (particularly vaporization) for carbon migration at elevated temperatures that cause changes in stoichiometry and concomitant lowered melting temperatures. Thus, there is a potential lifetime limit imposed by the constraint of keeping the binary carbide stoichiometry within a narrow band.

3.1.3 Recent Work

With NASA/GRC funding, work is in progress in LANL to recapture the fabrication technology for composite fuels. All indications are that the use of modern technologies can actually improve the quality of the fuel produced and that the fuel can be produced fairly quickly for the proposed non-nuclear (in hot hydrogen) and nuclear tests. The concern about loss of infrastructure and capability in this area appears not to be as serious as previously believed.

There has been recent promising work performed at the University of Florida on the development of laboratory scale processes for the production of solid solution mixed carbide fuel pieces (20). Samples of (U, Zr, Nb)C have been produced with various mole fractions of uranium. Stoichiometric samples were processed form the constituent carbide powders while hypostoichiometric samples were processed from uranium hydride, graphite and constituent refractory powders. Process optimization studies have been conducted to produce low porosity homogeneous single phase solid solution mixed carbide fuels for testing. In addition, a moderated Square-Lattice Honeycomb (M-SLHC) reactor design has been developed using these carbide fuels and designs to be responsive to the processing complexities of these fuels. Indications from these early efforts are that in the longer term, solid solution ternary carbides can be fabricated and should perform well under higher temperature NTP conditions.

Figure 3.1.1
Chronology of Major Nuclear Rocket Reactor Tests

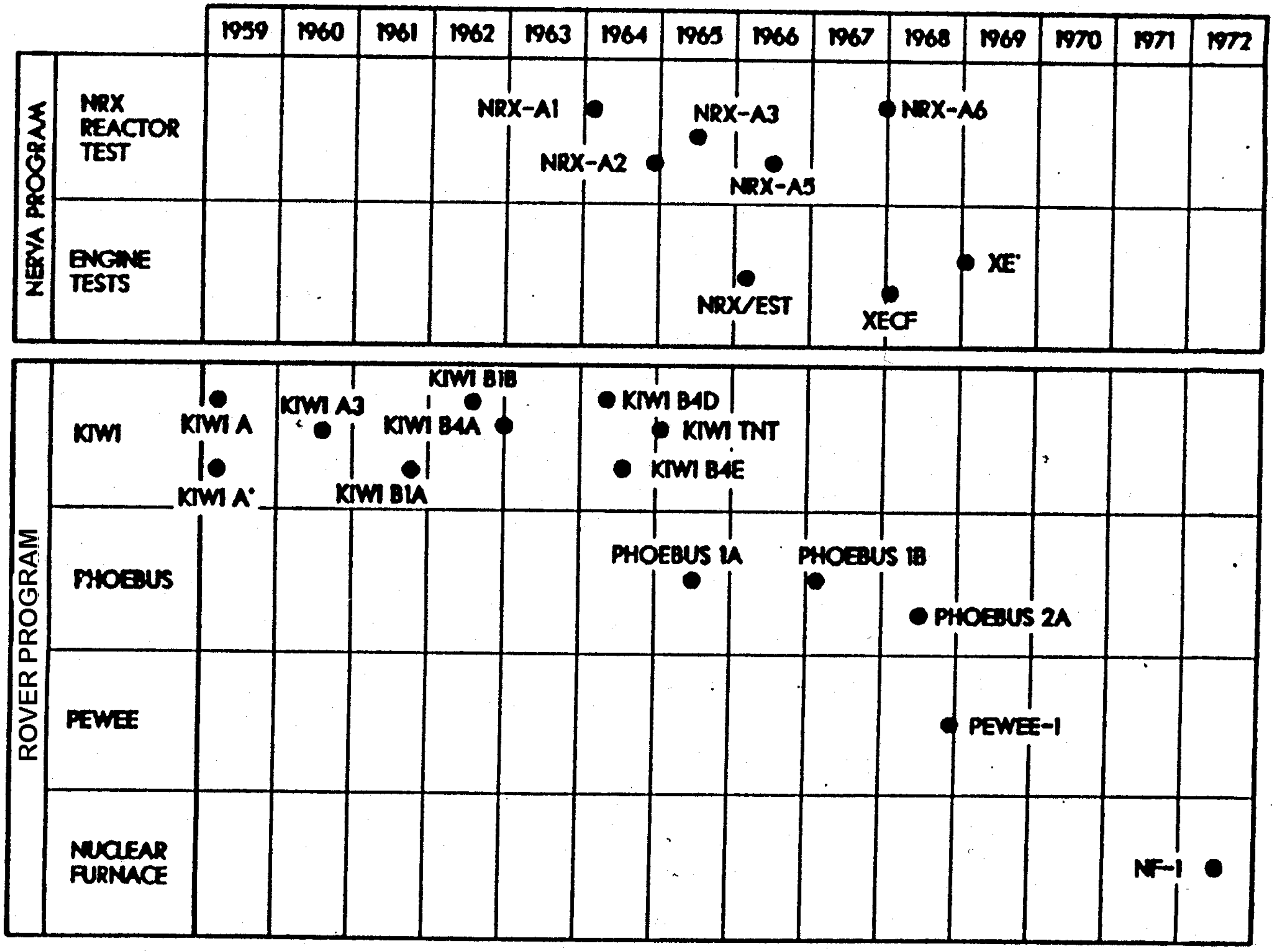
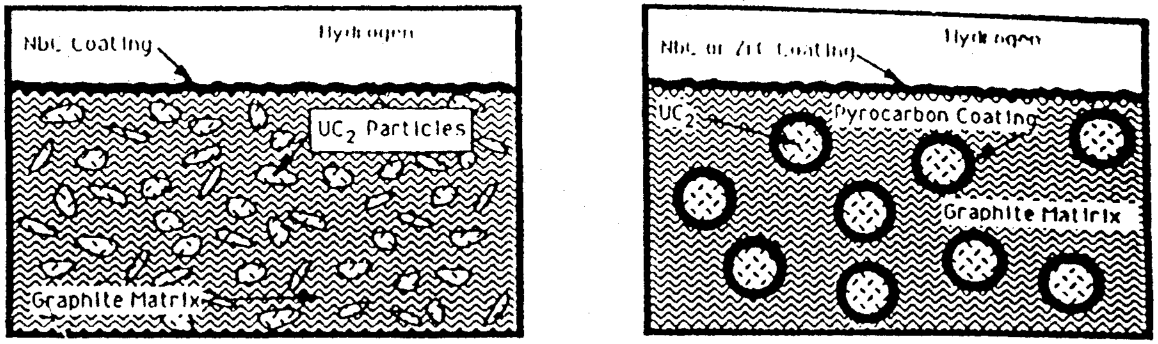
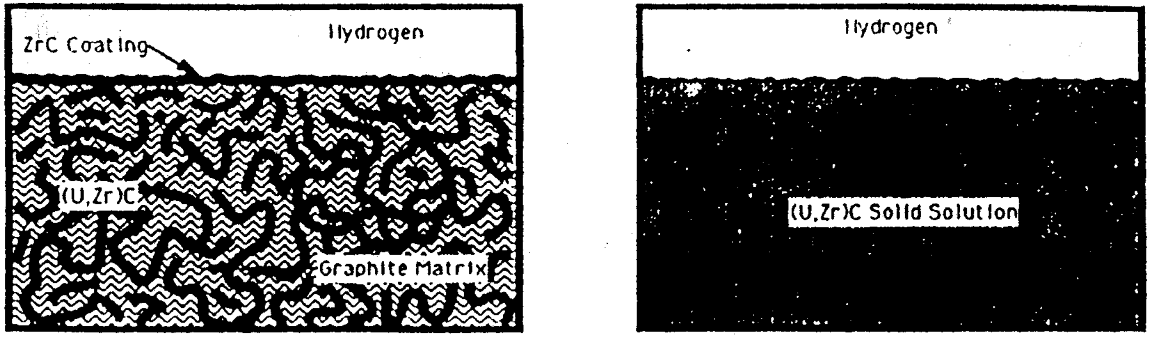


Figure 3.1.2
ROVER/NERVA Fuel Type



UC$_2$ Particles/Graphite Matrix

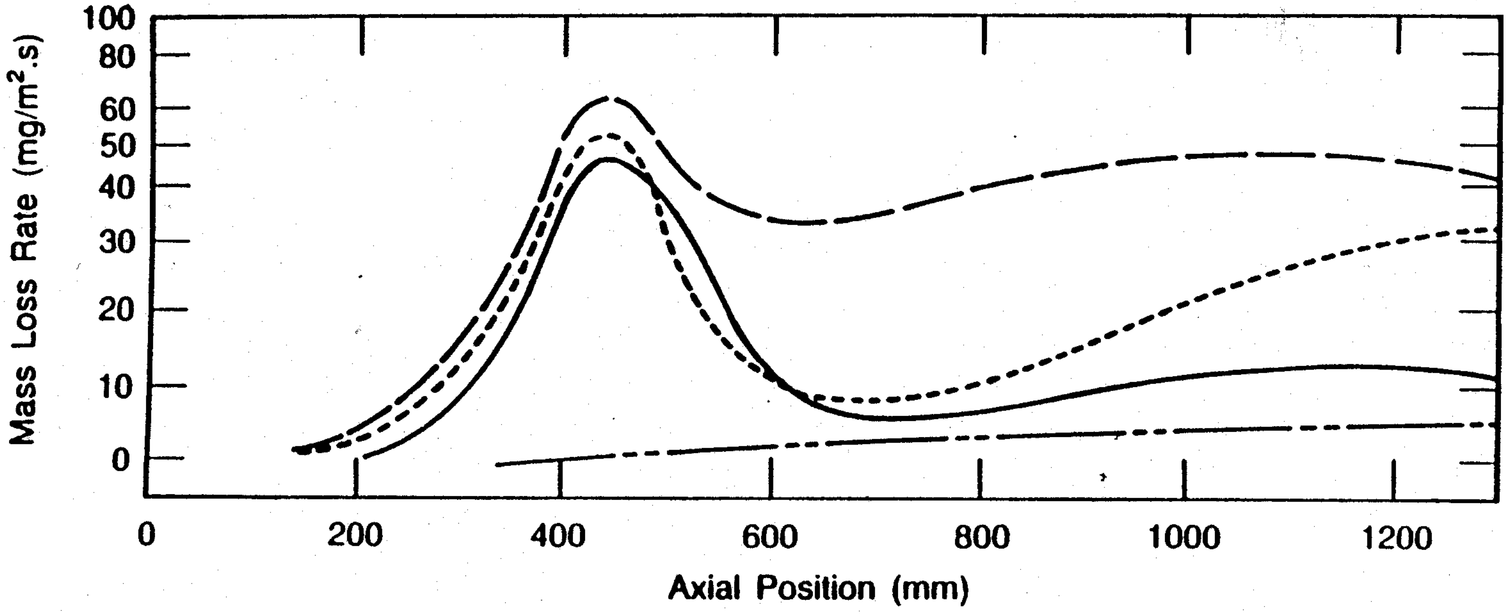
PyC Coated UC$_2$ Spheres/Graphite Matrix



Carbide/Graphite Composite

Carbide Solid Solution

Figure 3.1.3
Carbon Loss as Function of Axial Position in Fuel for Various Fuel/Coating Designs



____ ____ ____	Graphite Fuel Coated With NbC
-----------------------	Graphite Fuel Coated With ZrC
______________	High CTE Composite Fuel Coated With Zrc
_____ _ _ _____	Post NF-1 Composite Fuel 10 Hour, Cyclic Test Results (Electrical Tests) With “Super Coat”

Figure 3.1.4
Photograph of ROVER/NERVA Fuel Element

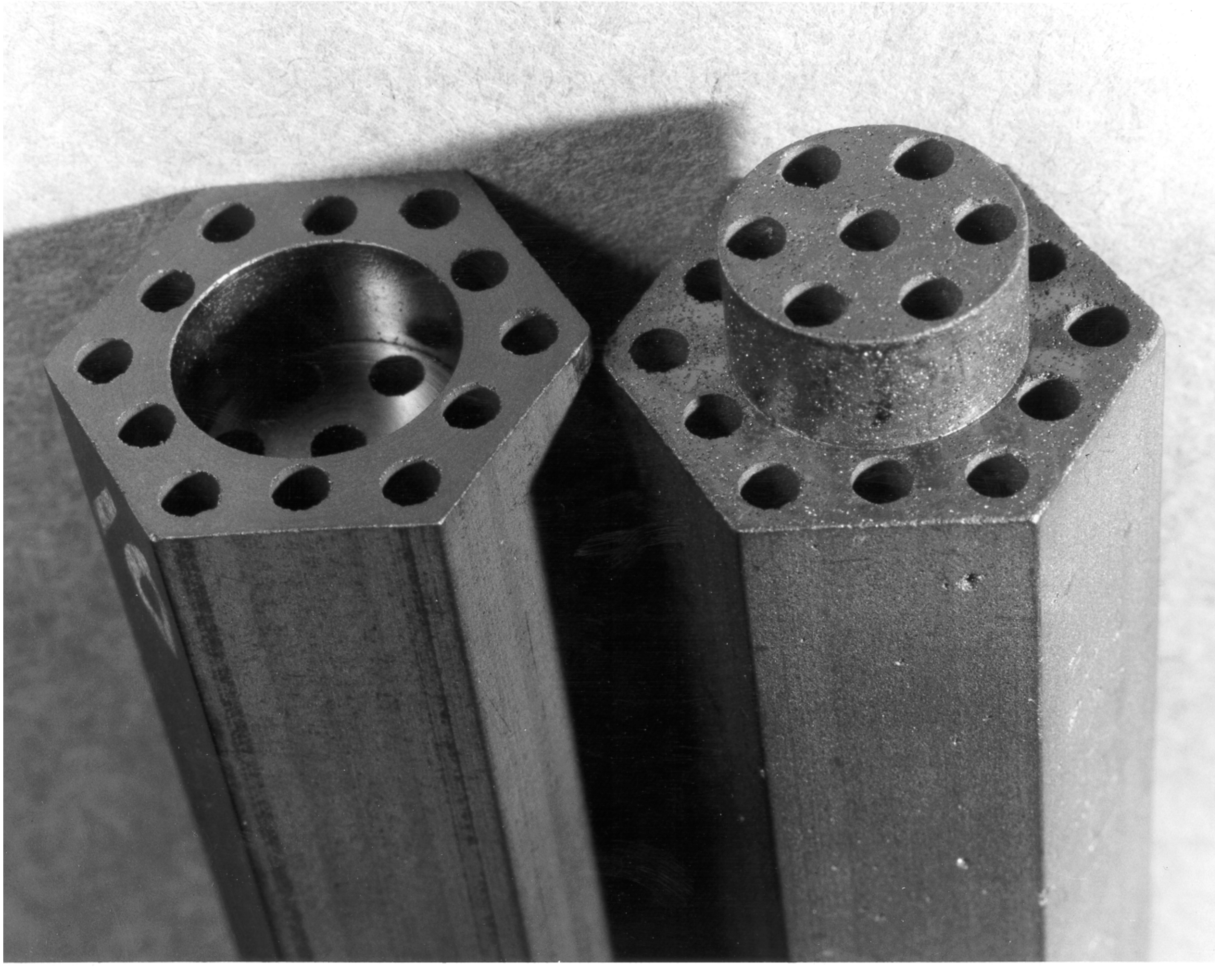


Figure 3.1.5
Qualitative Comparison of Projected Lifetimes at Temperatures for the Rover/NERVA Fuels

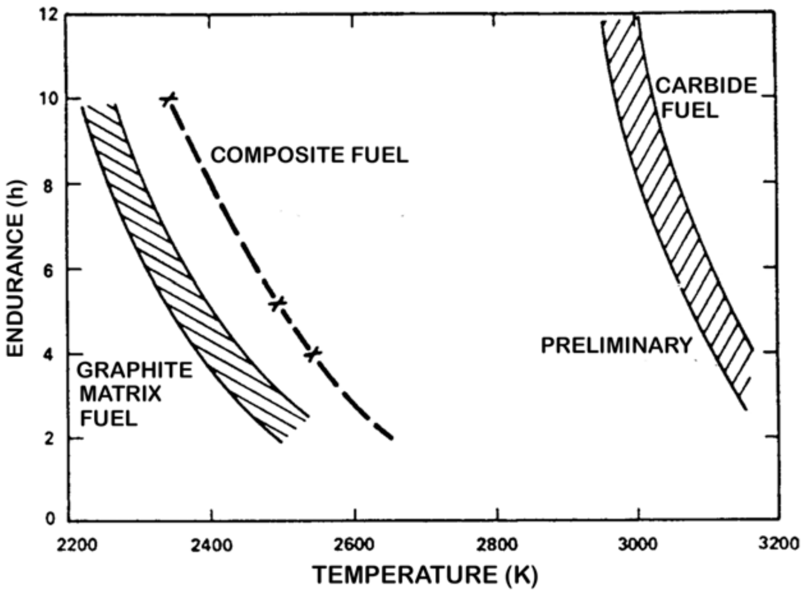


Figure 3.1.6
The UC-$ZrC_{0.31}$ Psuedobinary Phase Diagram Showing Solidus and Liquidus Temperature Data for the (UyZr$_{1-y}$)C$_x$ Solid Solution

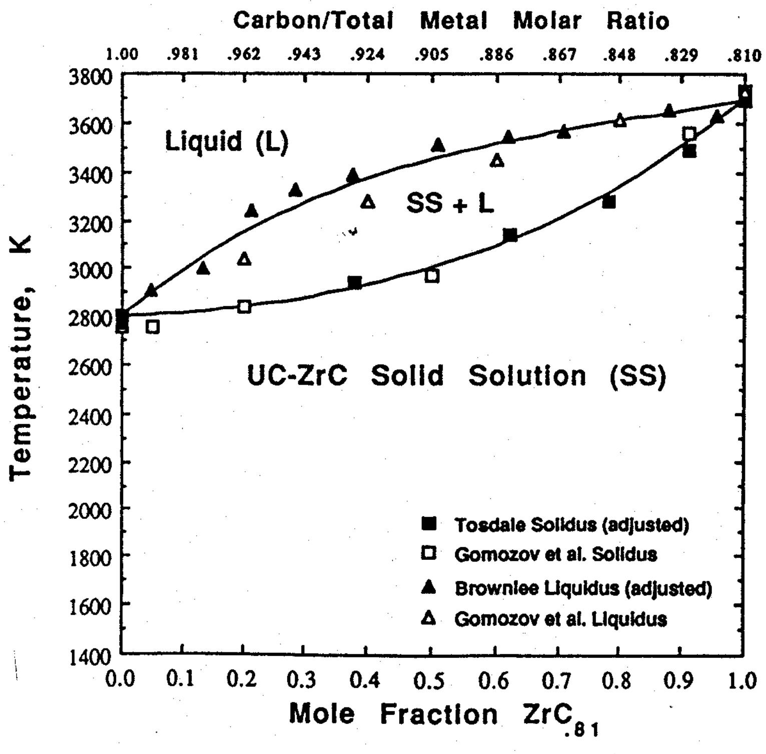


Figure 3.1.7
Isothermal Sections of the U-Zr-C System at 2873 K and 3273 K

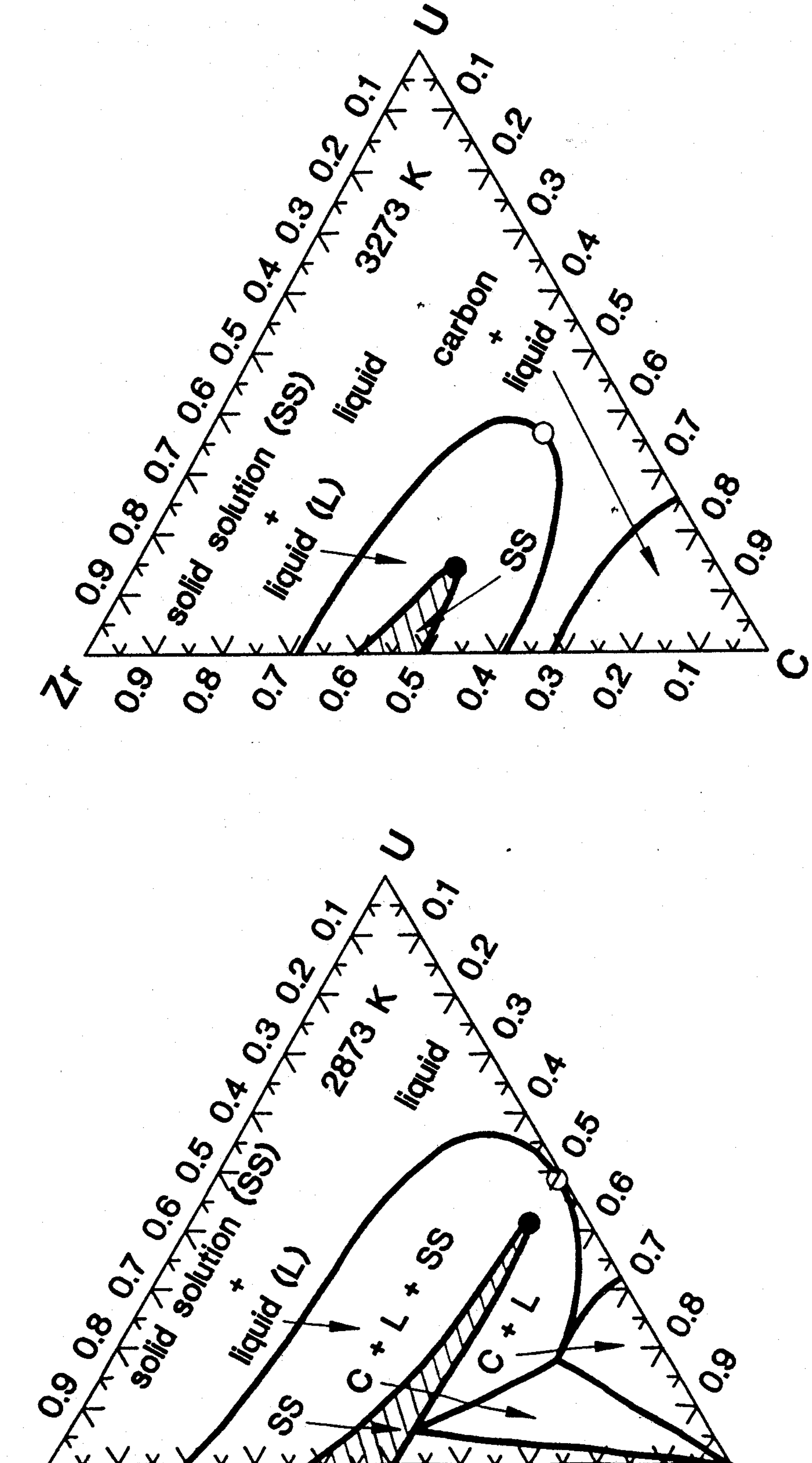
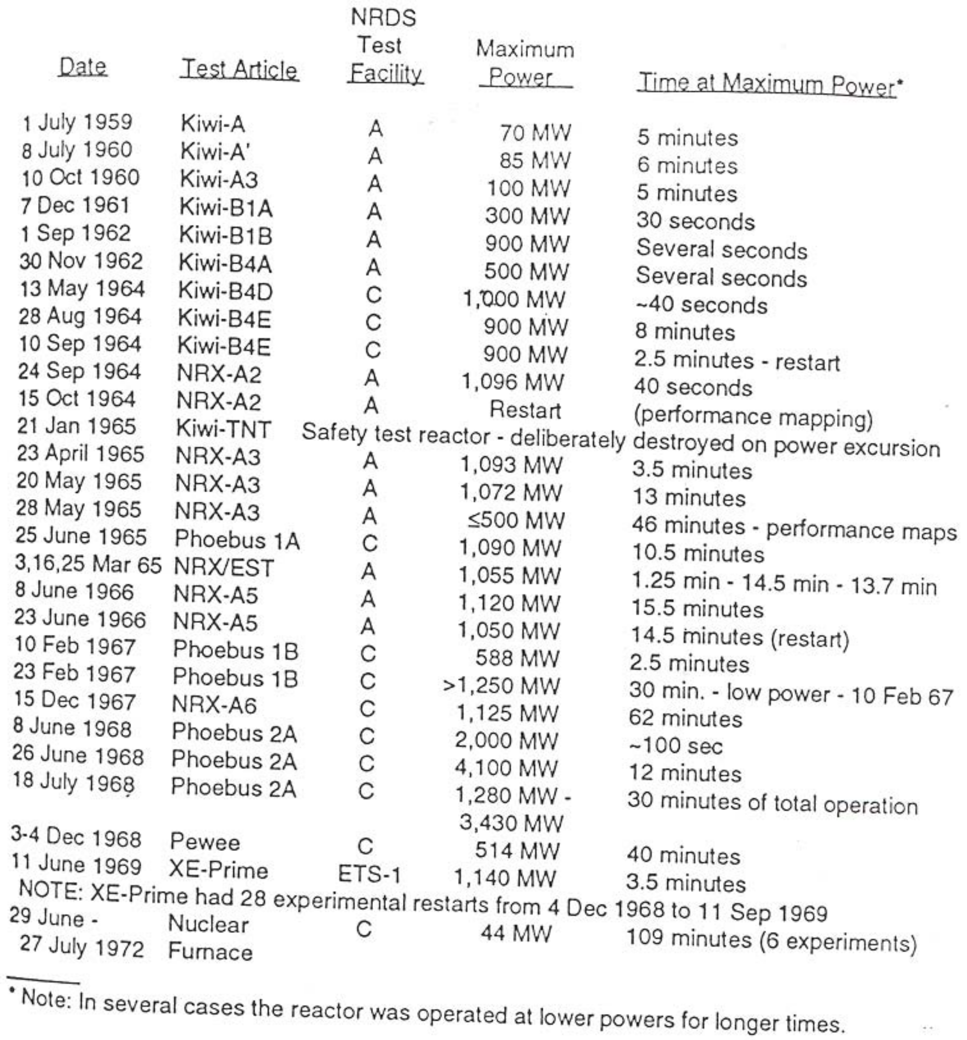


Table 3.1.1
Chronology of Rover and NERVA Reactor/Engine Tests



Date	Test Article	NRDS Test Facility	Maximum Power	Time at Maximum Power*
1 July 1959	Kiwi-A	A	70 MW	5 minutes
8 July 1960	Kiwi-A'	A	85 MW	6 minutes
10 Oct 1960	Kiwi-A3	A	100 MW	5 minutes
7 Dec 1961	Kiwi-B1A	A	300 MW	30 seconds
1 Sep 1962	Kiwi-B1B	A	900 MW	Several seconds
30 Nov 1962	Kiwi-B4A	A	500 MW	Several seconds
13 May 1964	Kiwi-B4D	C	1,000 MW	~40 seconds
28 Aug 1964	Kiwi-B4E	C	900 MW	8 minutes
10 Sep 1964	Kiwi-B4E	C	900 MW	2.5 minutes - restart
24 Sep 1964	NRX-A2	A	1,096 MW	40 seconds
15 Oct 1964	NRX-A2	A	Restart	(performance mapping)
21 Jan 1965	Kiwi-TNT	Safety test reactor - deliberately destroyed on power excursion		
23 April 1965	NRX-A3	A	1,093 MW	3.5 minutes
20 May 1965	NRX-A3	A	1,072 MW	13 minutes
28 May 1965	NRX-A3	A	≤500 MW	46 minutes - performance maps
25 June 1965	Phoebus 1A	C	1,090 MW	10.5 minutes
3,16,25 Mar 65	NRX/EST	A	1,055 MW	1.25 min - 14.5 min - 13.7 min
8 June 1966	NRX-A5	A	1,120 MW	15.5 minutes
23 June 1966	NRX-A5	A	1,050 MW	14.5 minutes (restart)
10 Feb 1967	Phoebus 1B	C	588 MW	2.5 minutes
23 Feb 1967	Phoebus 1B	C	>1,250 MW	30 min. - low power - 10 Feb 67
15 Dec 1967	NRX-A6	C	1,125 MW	62 minutes
8 June 1968	Phoebus 2A	C	2,000 MW	~100 sec
26 June 1968	Phoebus 2A	C	4,100 MW	12 minutes
18 July 1968	Phoebus 2A	C	1,280 MW - 3,430 MW	30 minutes of total operation
3-4 Dec 1968	Pewee	C	514 MW	40 minutes
11 June 1969	XE-Prime	ETS-1	1,140 MW	3.5 minutes
NOTE: XE-Prime had 28 experimental restarts from 4 Dec 1968 to 11 Sep 1969				
29 June - 27 July 1972	Nuclear Furnace	C	44 MW	109 minutes (6 experiments)

---

* Note: In several cases the reactor was operated at lower powers for longer times.

Table 3.1.2
Rover/NERVA Summary Test Results

Reactor ID	Chamber Temp (K)	Fuel Exit Temp (K)	Space Equiv. ISP (sec)	Time at Full Power (min)	Fuel Type	MWt/ Thrust(kN)
KIWI-B4D	1890-2130	2222	780	1	UO2/Graphite	914/204
KIWI-B4E	1890-2100	2389	820	8,2.5	UC2/Graphite	914/204
NRX-A2	2090	>2200	775	3.4	UC2/Graphite	1100/245
NRX-A3	2244	>2400	820	16.3	UC2/Graphite	1100/245
PHOEBUS-1A	2366	2478	835	10.5	UC2/Graphite	1340/298
NRX-A4(NRX-EST)	2264-2290	>2400	820	28.6	UC2/Graphite	1100/245
NRX-A5	2280-2333	>2400	820	29.6	UC2/Graphite	1100/245
PHOEBUS-1B	2222-2290	2445	828	30	UC2/Graphite	1340/298
NRX-A6	2300-2405	2556	847	62.7	UC2/Graphite	1100/245
PEWEE-1	1835	2550 2750	845 890	43	UC2/Graphite UC2/Graphite	500/111
XE-PRIME	2278	>2400	820	7.8	UC2/Graphite	1100/245
NF-1	-----	2450	830	109	Composite/ Carbide	
PHOEBUS-2A TESTED	2256	2306	805	12.5	UC2/Graphite	4100/913
DESIGN	2500	2550	840		UC2/Graphite	5000/1113

Table 3.1.3
Summary Record Test Performance of Rover Program

- 20 Reactors built and tested
  (17 test reactors, 1 safety reactor and 2 ground test engines)

- $1.4B expended in program (then year dollars)

- Best Parameters Achieved (nuclear tests)

- Power	PHOEBUS 2A	4100 MWt
- Peak Fuel Temperature (composite fuel)	PEWEE	2750 K
- Specific Impulse	PEWEE	848 s
- Maximum Restarts	XE	28
- Accumulated Time at Full Power	NF-1	109 minutes
- Continuous Operation	NRX-A6	62 minutes

3.2 Particle Bed Reactor Fuels (SNTP Program)

The Space Nuclear Thermal Propulsion (SNTP) program was aimed at the demonstration and validation of technologies necessary for a reliable, cost effective Particle Bed Reactor (PBR) based nuclear thermal rocket (NTR) engine. It was managed by the DoD (US Air Force with SDIO funding) in an aggressive, goal- oriented fashion during the period 1987-1993. A design and development team was assembled, led by the integrating contractor, Grumman Aerospace Corporation, and comprising of industrial contractors B&W, Aerojet, Hercules, Garrett, General Dynamics and L Systems, with Sandia and Brookhaven National Laboratories playing significant roles. Much of the program was classified secret, although a significant amount of data have subsequently been unclassified. DOE and NASA personnel were provided periodic briefings. An Independent Advisory Group to the project management was assembled under XERAD Corporation. Figure 3.2.1 shows a sketch of the organizational structure (21). The program was terminated in 1993 after expenditures of ~$200 M.

A significant amount of work was performed by the team during the course of the program. Designs of a compact reactor core coupled to all of the other required non-nuclear systems were performed and reviewed. Many of the technology issues with the system were identified and development work initiated to address them. Several non-nuclear and nuclear tests were conducted on fuel elements. Design of a new nuclear test facility, PIPET, was initiated with the aim of allowing fuel assembly cluster tests in a prototypic nuclear environment. Studies were also conducted for the siting and logistics of a ground test with effluent clean up (22).

Since the focus of this assessment is fuels technology we will concentrate on that for the SNTP program. The overall engine requirements translated to specific terms for the fuel called for the attainment of peak temperatures up to 3500 K (exit H_2 gas temperature of 3000 K), for a 1000s operating life, up to 10 startup/shut down cycles and a power density of 40 Mw/ℓ. A maximum uranium loading of 1.5 g/cm^2 was considered necessary to meet criticality requirements for reasonable core size. Table 3.2.1 shows the preliminary target design requirements for SNTP fuel.

The baseline fuel for the project was a modified TRISO fuel, utilizing the NERVA/Rover experience. A UC_2 kernel was first coated with a buffer layer of low-density pyrocarbon, and then with a low temperature isotropic (LTI) sealant of high-density pyrocarbon. A CVD process was then used to apply a ZrC coating to the particle fuel. Figure 3.2.2 shows a sketch of the fuel particle and its incorporation into the fuel element with its moderator block and hot and cold frits and the assembly of the elements into the reactor core. The limiting centerline temperature for this fuel was estimated to be 2500 K. This was clearly inadequate to meet the stated requirements and advanced fuel development was needed. The fuel development strategy did not consider a stepwise evolution from this baseline fuel to related advanced fuels; rather the baseline fuel was used to facilitate development of production processes, coating processes, general fuel element test procedures and early fuel element testing, while a parallel path was assumed to be taken for advanced fuels development.

Figure 3.2.1
DoD/Industry/National Laboratory Team Assembled for the SNTP Program

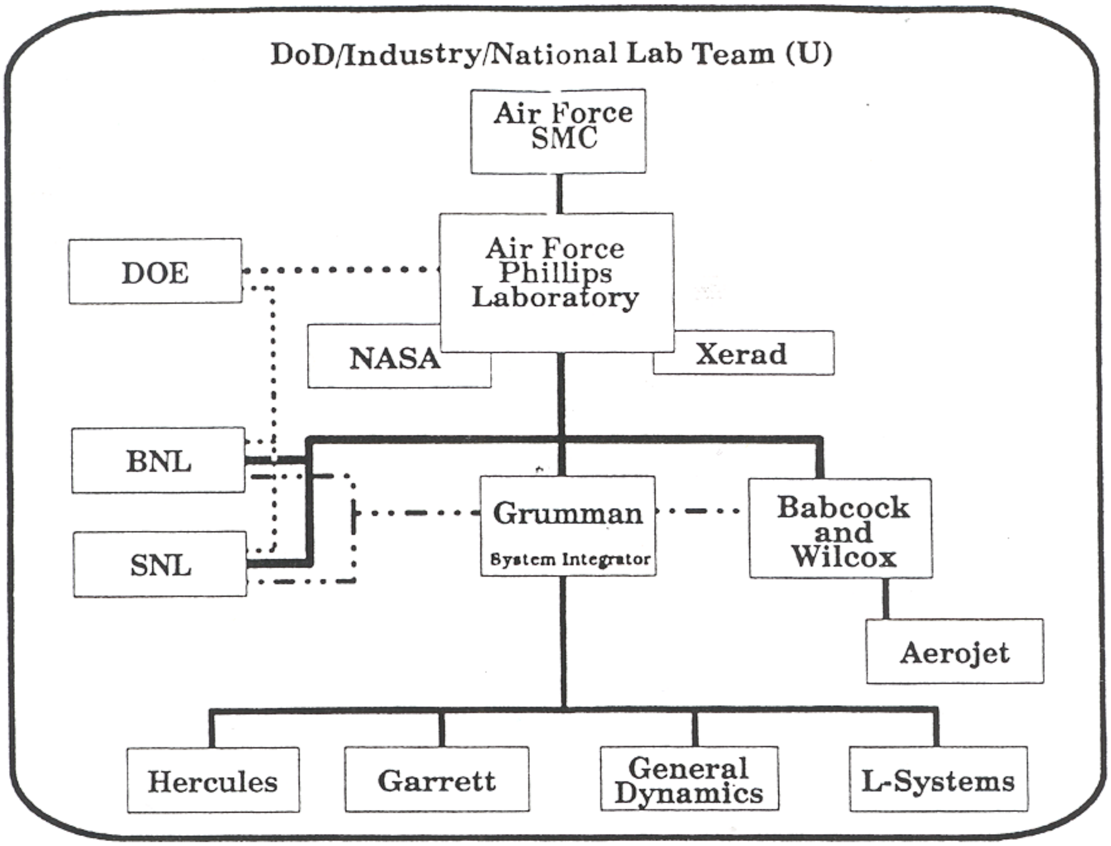


Figure 3.2.2
Evolution of the Particle Bed Reactor – Fuel Particle to the Reactor Core

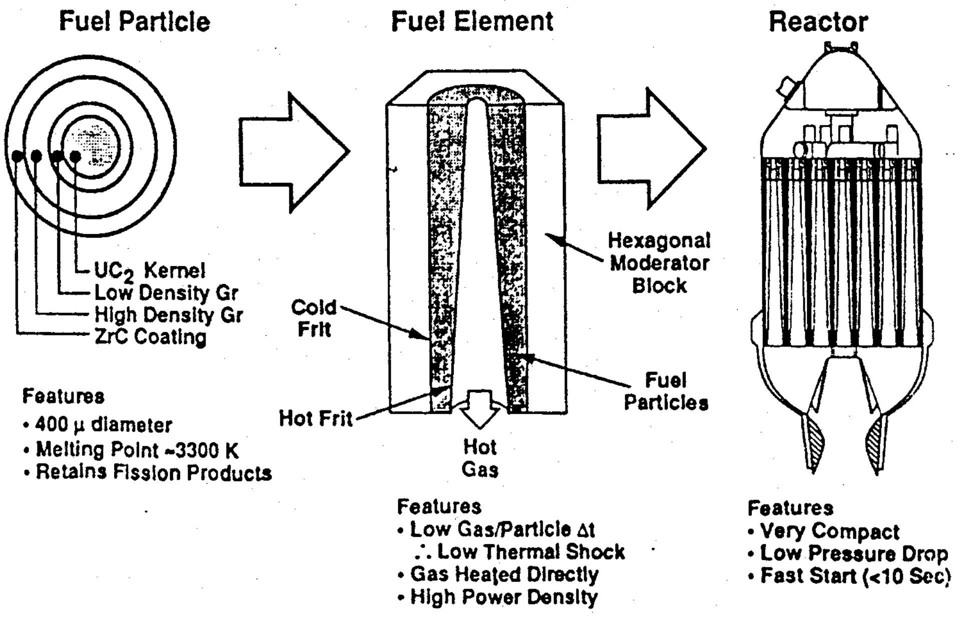


Table 3.2.1
Preliminary Target Design Requirements for the SNTP Particle Nuclear Fuel

Mixed-Mean Exit H_2 Temperature	3000 K
Operating Fuel Temperature	3500 K(1)
Total Operating Time	1000 s
Startup - Shutdown Cycles	10
Power Density	40 MW/I
Material Compatibility	Fuel kernel coating compatible with Hot Frit inner wall coating at operating temperatures
Fuel Kernel Diameter	450 μm
Coating Thickness	25-50 μm
Maximum Particle Diameter	550 μm
Minimum U Content in Particle	1.5 gm/cm^3
Dimensional Stability at Operating Temperatures	Minimum Deformation(2)
Fission Product Release at Operating Temperature	Minimum(2)
U Content Variation	±5%
Density Variation	±5%
Particle Spericity (r_{max}/r_{min})	±5%

1. Preliminary - final temperature will depend on fuel particle and reactor design.
2. Actual values TBD.

Table 3.2.2 shows the summary of the SNTP program approach to advanced particle fuel development. The initial idea of an enhanced baseline fuel particle development (with a double thickness LTI layer) was abandoned early in the program because the anticipated performance (2800 K peak fuel temperature) was not high enough to warrant the effort.

The advanced fuel development task took several parallel paths to develop coated particle fuels that would meet the SNTP Program performance goals. Three fuel particle concepts were considered: the infiltrated kernel (IK) particle fuel, the mixed-carbide particle fuel, and the interstitial dispersoid (ID) particle. The first two advanced fuel development efforts were pursued with equal emphasis, while the third concept was a relative newcomer. Ultimately, one of these fuel forms was going to be chosen for production for the fuel element testing in the Particle-bed reactor Integral Performance Element Tester (PIPET) and in the future demonstration engine.

The IK fuel particle was a porous graphite matrix infiltrated with UC_2, coated appropriately for fission product retention and to prevent particle hydrogen corrosion (23). The basic premise of the concept is that a coated IK fuel particle has the theoretical potential to operate at temperatures up to 3500 K. The temperature limitations of the IK particle is determined by the properties of the graphite matrix, which remain physically stable at very high temperatures. The UC_2 in the graphite pores is molten at temperatures above 2800 K, but is held in the pores of the graphite matrix by capillary action. No nuclear tests were performed with these fuels, but non-nuclear blow down tests with hydrogen yielded encouraging results (24,25).

The mixed-Carbide fuel kernels are a solid solution mixture of uranium-carbide, zirconium-carbide, and/or niobium-carbide. The mixed-carbide fuel systems that were being considered by the SNTP Program are ternary mixtures, $(U_xZr_{1-x})C_z$ or $(U_xNb_{1-x})C_z$, and quaternary mixtures, $(U_xZr_yNb_{1-x-y})C_z$. The mixed-carbide fuel kernel was planned to be coated appropriately to prevent particle hydrogen corrosion and to retain the fission products. Fuels development work on the Rover/NERVA and Soviet programs were used as the basis for this selection.

The ID particle was formed by putting the fissile salts in solution with graphite precursor materials. The dispersed solution was dried into spherical kernels, in a hot oil emulsion, and then heat- treated. The development of this particle was not as far along as the other two advanced fuels (26). Table 3.2.2 presents a summary description of the three fuel concepts.

The project plan was to test and validate each of the advanced fuel forms, in a series of Particle Heating Tests (PHT), Particle Nuclear-heating Tests (PNT), and several other tests that would characterize the fuel particles and assess their performance capabilities. As stated earlier, the baseline fuel would be used to develop the capabilities needed to characterize the fuel and to test the fuel element, etc., essential for fully developing the advanced fuel. Before the fuel would be considered ready for inclusion in a ground demonstration engine test, the advanced fuel would be tested and qualified under conditions that progressively approached the prototypical operating conditions. Included in the planned tests were the Nuclear Element Tests (NET) in the ACRR at Sandia National Laboratory, and the fuel element testing in PIPET which was going to be a new test facility that is planned in support of the program.

Table 3.2.2
SNTP Fuels Summary

Particle Fuel Type	Peak Temperature Potential	Comments
3. **Baseline** UC_2 Kernel PyC Coating(s) ZrC Final Coating	2400 – 2800 K	Adaptation of HTGR and NERVA/Rover technology.
4. **Advanced**		
i Infiltrated Kernel (IK) Porous graphite matrix Infiltration of coated UC_2	3500 K*	UC_2 expected to remain contained in molten state within matrix.
iii Mixed Carbide (U, Zr) C (U, Nb) C (U, Zr, Nb) C	3100 – 3400 K	Potential inferred from final phase of Rover/NERVA program and CIS experience.
iii Interstitial Dispersoid (ID) skeleton UC_2 fuel in graphite	3200 K*	Synthetic percussion method was used to produce spherical graphite skeleton info which UC_2 is introduced using a solution chemistry process. The process is claimed to be optomizable to produce high performance fuel.

Note: The SNTP project management convened a review group (of which Dr. Bhattacharyya was a member) that evaluated the technology status of the candidate fuels in June 1993. Based upon available data the review group recommended pursuing the mixed carbide path.

The coated fuel particles would be tested for the first time, under prototypical operating conditions, in PIPET; however, the risk of fuel failure would be minimized by the progressive testing program planned in NET by pursuing the several parallel advanced fuel development paths to minimize the development risks to the program. Table 3.2.3 shows the planned phased requirements for the SNTP fuel particle development and the principal nuclear tests planned. Most of the planned tests in table 3.2.3 were not conducted during the program. However, a significant number of tests were conducted on the baseline particle fuel. Table 3.2.4 presents a summary of the tests conducted along with principal conclusions reached (27).

Because of funding constraints, the SNTP project decided to pursue one primary path for advanced fuels development from the above choices. A Fuels Evaluation Team was convened to help project management make the selection. Table 3.2.5 shows the membership and affiliations of the team. The team reviewed the available data (which were sparse) and recommended the mixed carbide fuel as the best choice from performance and probability of success considerations (28). Despite the protests of the proponents of the IK and ID fuels, the SNTP project management concurred with this recommendation and set up contracts with Institutes in the Former Soviet Union to help with the mixed carbide fuels. However, the program was cancelled before any significant follow on work could be performed.

In view of the very high potential performance of particle fuels, the concept and fuel geometry should be examined in the future when the need for very high performance is clearly demonstrated.

Table 3.2.3
Phased Requirements for the SNTP Fuel Particle

Performance Parameter	Net 3 1Q FY ‘95	Net 5 1Q FY ‘97	PIPET Core A 3Q FY ‘98	Demo Engine 2Q FY ‘01	Flight Engine FY ’03
Mixed Mean H_2 Temperature	2500 K	2800 K	2800 – 3000 K	2800 – 3000 K	3000 K
Operating Fuel Temperature	2800 K	3000 K$^{(1)}$	3200 – 3400 K$^{(1)}$	3200 – 3500 K$^{(2)}$	3500 K$^{(2)}$
Total Operating Time	200 – 300 s	300 s	1000 s	1000 s	1000 s
Startup – Shutdown Cycles	20 – 30	30	10	10	10
Power Density	5 MW/l	5 MW/l	40 MW/l	40 MW/l	40 MW/l

1. Net’s 6 & 7 will require the same fuel particle temperature as for PIPET (3200 – 6400 K)
2. Preliminary. Final temperature will depend on fuel particle and reactor design.

Table 3.2.4
Summary of Fuel Tests Conducted During the SNTP Program

Test Name	Nature	Primary Goal(s)	Major Conclusions
Particle Element Tests (PET)	• Non-nuclear • In He and He/4%H_2 gas • Fuel particles tested	• Test adequacy of fuel particle fabrication and QA/QC • Property measurements	• Fabrication procedures and process control are very important • Modern computer controlled processes are necessary in fabrication
Particle Nuclear Tests (PNT)	• Nuclear (in ACRR reactor) • Fuel particles in a bed (10^4 particles) • He/4%H_2 gas	• Performance limits under nuclear heating • Heated all the way to fuel melting • Fission produced release and special measurements	• Fuel kernel melting causes spread of failures • Gas purity is extremely important (oxidation occurs very rapidly)
PIPE	• Fuel element tests (includes hot and cold frits and end fittings) • Flowing H_2 gas • In ACRR reactor	• Performance limits evaluated of element with Re hot frit and sintered SST cold frit • Test of computer controlled transient operation	• Two nuclear tests, PIPE1 and PIPE2, were run, with overall success • Computer control system debugged
NET	• Fuel element tests (with advanced frit material) • Cryogenic H_2 used • In ACRR reactor	• Performance limits of prototypic fuel elements (at 0.5 – 1.0 MW/l)	• One nuclear test run • Hot frit failed • Computer controlled transient operation worked very well – stopped transient when abnormalities were noted

Table 3.2.5
SNTP Advanced Fuels Evaluation Panel

Panelist	Affiliation
Dr. Walter A. Stark, Jr. - Panel Chair	Los Alamos National Laboratory
Dr. Robert Amodeo	Xerad, Inc.
Dr. Samit K. Bhattacharyya	Argonne National Laboratory
Dr. Terry D. Gulden	General Atomics
Dr. Michael J. Kania	Oak Ridge National Laboratory
Dr. Edmund K. Storms	Los Alamos National Laboratory
Dr. Steven A. Wright	Sandia National Laboratory
Dr. John Metzger - Executive Secretary	Grumman Aerospace Corporation

3.3 Russian Twisted Ribbon Fuel

The USSR undertook a significant R&D effort to develop nuclear thermal propulsion and related capabilities during the late 1960's to the late 80's. A major part of the effort was to develop high temperature fuels that had the lifetime potential needed for practical NTP systems. The reported work included fabrication of a large number of samples, non-nuclear hot hydrogen flow tests, individual fuel element tests under rapid transient conditions (IGR reactor), and prototypic NTR operation conditions (IVG-1 and RA reactor). In all, a very large amount of data were produced and an invaluable amount of experience gained in the development of such fuels (29). The program costs appear to be at least of the same order as the Rover/NERVA program. Unfortunately, despite a decade of attempts to understand quantitatively the information gathered, US experts are still uncertain about the precise nature of all of the information developed. There has been confusion on information presented to US personnel by the Russians and there has been confusion about interpretations and extrapolations made by US personnel. Because of self-interest issues on both sides, such confusing interpretations were inevitable.

The USSR groups evidently followed the NERVA program quite closely and chose to follow the mixed carbide fuel path early in the development program. In their optimized designs for NTP systems they used (U, Zr)C fuels for the low temperature portion (gas temperature ≤ 2500 K) of the reactor core. For the higher temperature portions of the core they used (U, Zr, Nb)C fuel (up to 3100 K). The carbide fuels were fabricated in a wide variety of shapes – rods, cruciforms, twisted ribbon, spherical particles, etc. The twisted ribbon shape was their preferred choice as the best compromise between large surface area and controlled geometry. Figure 3.3.1 shows a photograph of a typical twisted ribbon fuel in an assembly. They seemed to have had good success in the uniform fabrication of these materials, although no US representative appears to have actually witnessed the entire fabrication process. Work has been done with other carbides in place of Nb – most notably Ta and Hf. There are claims of 200 K higher temperature potential with Ta relative to Nb. This is a very large benefit, if the increased neutron capture cross-section of Ta (or Hf) could be tolerated in the reactor design.

The first direct discussions with the Russians on the subject was at the Third International Conference on Space Nuclear Power and Propulsion held in Semipalatinsk in 1992 (30). A special panel session on NTP fuels (co-chaired by Ivan Fedik of NPO LUCH and S.K. Bhattacharyya) spent two hours discussing various facets of the data. There were also a number of private discussions held by SNTP personnel (BNL, Aerojet, B&W and Grumman) with Russian and Kazakhstan experts since the SNTP project had decided to enter into formal arrangements with the Russian Institutes to obtain the advanced carbide fuel data. Subsequently there were de-tailed discussions at the Fourth International Conference on Space Nuclear Power and Propulsion held in LUCH (Podolsk) in 1993 (31). A NASA Lewis Research Center (now GRC) contract was awarded to INSPI (Diaz and Anghaie) to perform collaborative work with LUCH personnel (D'Yakov) to obtain firsthand knowledge of the fuels activities (32). Finally a number of private meetings were held between LUCH and Semipalatinsk personnel over the years with ANL, BNL, SNL, Industry, NASA and DOE personnel. The results of these discussions are often in conflict with each other.

Carbonitride fuels were also developed, primarily for use with ammonia propellants. These fuels were also studied for use in cermets. Table 3.3.1 shows the list of candidate fuels used in the former Soviet Union, their uranium density and peak temperature limits and the current Russian perception of technology status (33).

The overall conclusion from all of the discussions is that a very large amount of work was performed in the former Soviet Union on the development of NTP fuels. Figure 3.3.2 shows a summary sketch of the geometries of fuels fabricated and tested, for all of the compositions shown in Table 3.3.1. These fuels were subjected to numerous non-nuclear and nuclear tests. Table 3.3.2 presents a summary of the peak non-nuclear test results obtained for the various classes of fuels (33). The time at peak temperatures reported is impressive. The reported number of nuclear tests run is very large. In Table 3.3.3 a summary of the tests is compiled from various sources. The tests, which spanned 19 years, included 1550 fuel assemblies and 7 fuel core tests. Taken as a whole, the best results achieved (shown in the table) included a peak hydrogen exit temperature of 3100 K. A significant problem with this very large data base is the lack of post irradiation examination data that severely limits our ability to analyze and understand the behavior of the fuel. Some of the top Russian experts (34) have suggested that the key tests be repeated jointly with the US experts in order to ensure that reproducible, defensible results are obtained.

All indications are that despite these shortcomings, significant successes were achieved and that suggests that the binary and ternary carbides are viable for NTP applications. An attempt should be made to obtain a compilation of the data in order to have a starting point from which to initiate future development.

Nuclear Test Facilities

A discussion of the CIS NTR fuels work is not complete without a brief overview of the nuclear test facilities that were used in the development of these fuels and the engine systems.

1. Nuclear Critical Facilities (very low power)

These test facilities are used for criticality and basic core physics and safety physics parameters of a reactor. There were numerous such facilities in Russia (e.g. at Kurchatov Institute, Moscow; the IPPE in Obnisk, etc.) The Kurchatov facilities had been used in the NTP engine reactor development work. The facilities are currently in bad physical condition and many of the operations personnel reassigned. Dr. V. Pavshuk has told us several times in the recent past that these facilities could be refurbished and brought back into operation at modest costs.

2. The Impulse Graphite Reactor (IGR) at Semipalatinsk

The IGR is a pulsed reactor fueled with UO_2 or UN dispersed in a graphite matrix (exactly like the TREAT reactor at ANL-W) capable of depositing a large amount of energy in a central fueled test section for a short (seconds) time. Its energy deposition capability (5 GJ per pulse) is twice that of TREAT, largely because of the fact that its design (which followed that of TREAT) avoided the limit imposed by the peak fuel temperature limits on the zircalloy-2 clad in TREAT fuel. In addition, TREAT used a more conserveative operating philosophy (the self-limiting criteria) than does IGR. IGR also has an approved H_2 flow test rig capability.

According to Yuri Cherepnin (35), former Director General of the National Nuclear Center in Kazakhstan that ran the Semipalatinsk test site after 1991, during the period of 1962-1978, approximately 160 tests were run on IGR in support of the NTP program. His oral summary of the highlights of these tests were:

- Temperatures of exhaust H_2 – up to 3000 K
- Power densities – up to 25 MW/l
- Test duration – up to 5 s

The IGR was in the best shape of any of the former USSR test reactors I visited in 1992. It has been operational since then, but conditions have deteriorated. Vladimir Pakhnitz, the very able chief engineer, has been assigned to other duties (and has had serious health problems). Decisions on whether to use this facility will require careful evaluation of both our NTP test plans and the cost to refurbish it for our purposes. The Russians have pursued the design of a higher performance transient test reactor (MIGR) (36), but so far have had no success in obtaining support for its construction.

3. The IVG-1 Facility (Semipalatinsk)

The IVG-1 was an impressive test facility in which an ensemble of NTP fuels could be tested as part of a full core with flowing H_2. Each fuel element was individually plumbed and there was detailed monitoring to keep track of each element. There were provisions to act if any of the fuel elements failed during the test since the flowing hydrogen was exhausted into the open. No damage to fuel elements was reported in the course of the tests, which ran from 1972 – 1984 (about thirty tests were run). The facility was designed for a power level of 720 MW, but it was never operated at greater than 230 MW. It was estimated that there had been an equivalent of 23 hours of operation with hydrogen during the operating life of the facility. The maximum power that an NTP fuel element ever ran in H_2 was 10 MW (corresponds to about 3.4 MW/L).

The facility was not in operable condition in 1992 when I visited it and it has not been used since. There was an ISTC funded project that was used in 1996 to prepare a decommissioning plan for the IVG-1 and RA reactor, which suggest that further operations of these facilities was unlikely.

4. The RA Reactor (Semipalatinsk)

A prototype NTP reactor facility was constructed at the Semipalatinsk test site in the 1970's. This was called the IRGIT reactor. The reactor designs produced 4000 kg of thrust, had a design outlet gas temperature of 3000 K and a calculated ISP of 910s. The thrust to weight ratio was poor (~ 1:1) because of the heavy moderator in the design. The tests that were run ranged in reactor power from 24 MW to 272 MW. The actual exhaust temperatures were generally below 2000 K and several design problems showed up. The test program was terminated in 1980.

The reactor facility was converted to the RA reactor test facility, which was used to investigate fission product deposition data for a range of fuel types (stated to be 10), range of coatings, range of exhaust gas temperatures and several coolant gases. Tests were performed under steady state and transient conditions. A computer code was developed to predict fission product behavior and compare against experiments. However, the data as presented in the Semipalatinsk conference (30) had several internal discrepancies that need to be understood before they can be used with confidence.

As stated for the IVG-1 reactor, the RA reactor was not in operable condition when we first visited the facility in 1992. It was part of the ISTC funded project aimed at decommissioning the reactors in Semipalatinsk. Thus, the possibility of using this facility for future US NTP programs use is not good.

As an added comment on the potential use of the nuclear test facilities in Semipalatinsk, it is noted that there is a steady exodus of the Russian leader from Semipalatinsk to Moscow. Dimitri Zelenski, the Deputy Director of the National Nuclear Center of Kazakhstan, left to take a position in Moscow in 1998. Recently, Yuri Cherepnin, the Director of the National Nuclear Center, followed him to Moscow (partly for health reasons). Several Russian specialists have done the same. In view of the above, it is not clear that there remain the technical expertise to refurbish and operate the complex facilities in Semipalatinsk any more. A very careful evaluation with on-site assessments will need to be made if there is any interest in using these facilities.

Figure 3.3.1
Russian Twisted Ribbon Fuel Located in Fuel Assembly

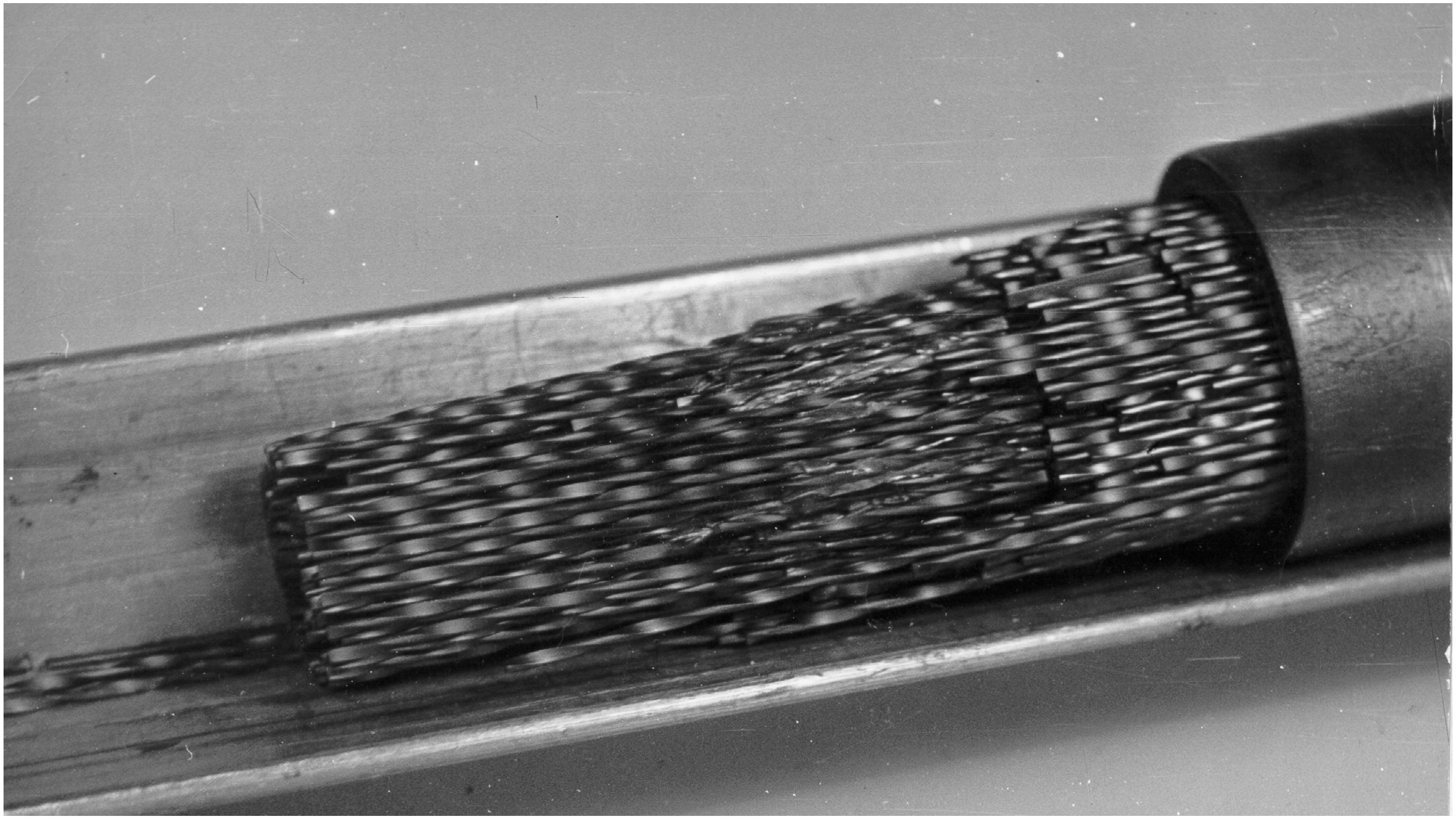
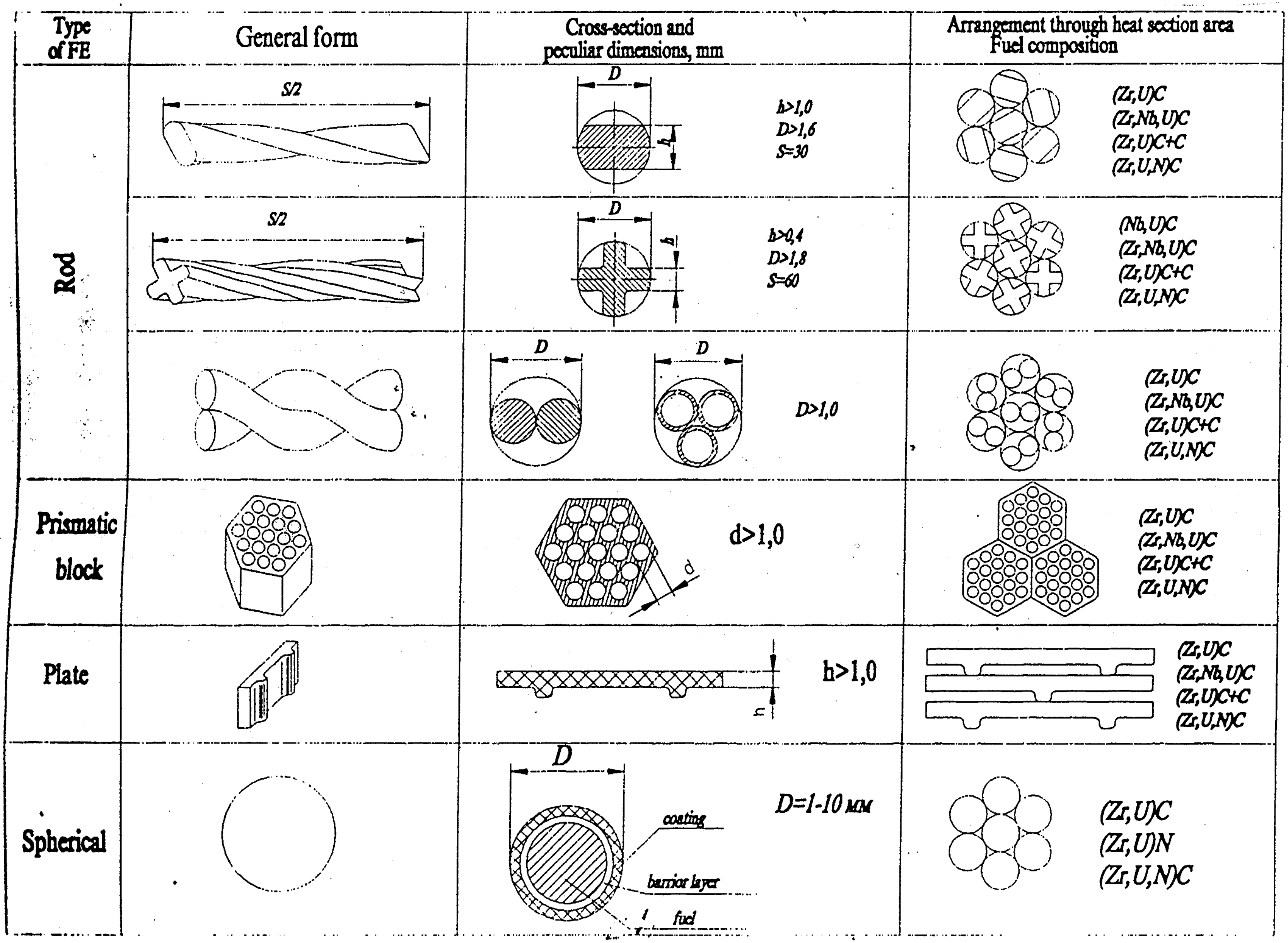


Figure 3.3.2
Fuel Geometries Used in NTP Tests



Type of FE	General form	Cross-section and peculiar dimensions, mm	Arrangement through heat section area Fuel composition
Rod	S/2	D h h>1,0 D>1,6 S=30	$(Zr,U)C$ $(Zr,Nb,U)C$ $(Zr,U)C+C$ $(Zr,U,N)C$
	S/2	D h h>0,4 D>1,8 S=60	$(Nb,U)C$ $(Zr,Nb,U)C$ $(Zr,U)C+C$ $(Zr,U,N)C$
		D D D>1,0	$(Zr,U)C$ $(Zr,Nb,U)C$ $(Zr,U)C+C$ $(Zr,U,N)C$
Prismatic block		d d>1,0	$(Zr,U)C$ $(Zr,Nb,U)C$ $(Zr,U)C+C$ $(Zr,U,N)C$
Plate		h h>1,0	$(Zr,U)C$ $(Zr,Nb,U)C$ $(Zr,U)C+C$ $(Zr,U,N)C$
Spherical		D coating barrier layer fuel D=1-10 мм	$(Zr,U)C$ $(Zr,U)N$ $(Zr,U,N)C$

Table 3.3.1
Candidate Fuels Listed by CIS

Fuel Type		Uranium Content (density g/cm^3)	Maximum Operating Temperature, K	Technology Status
Carbides	(U, Zr) C,C (U, Zr) C (U, Zr, Nb) C (U, Zr, Ta) C	≤ 2.5	2500 3300 3500 3700	Highest Intermediate
Carbonitrides	(U, Zr) C,N	6-8	3100	
Cermet (carbonitride)	(U, Zr) C,N-W	≤ 6.5	2900	Lowest

Intermediate

Table 3.3.2
Time-Temperature Limits of Non Nuclear Hot Hydrogen Testing
(V. Daragan – LUCH)

Type of FE	Test temperature, K	Max test time, hours
UC-ZrC-NbC	2800 K	200
	3500 K	0, 5
	3300 K	1, 0
UC-ZrC-TaC	3300 K	2, 0
Uzr(CN)	2800 K (H_2 + N_2)	100

Table 3.3.3
CIS NTP Fuels Summary

- Development and Test Duration — 19 years
- Total Number of Fuel Assemblies Tested — 1550
- In-reactor Testing
  - Full Core Tests
    - IVG-1 Reactor — 4
    - IRGIT Reactor — 2
    - RA Reactor — 1
  - Loop Channels
- IGR Reactor — 36 (6 modifications)
- Best Performance (for different tests)

- Hydrogen exit temperature	3100 K
- Tests duration	4000s
- Heating rate	≤1000 K/s
- Number of cycles	12
- Power cycle duration at T = 1200 K	6000 hrs.

3.4 Cermet Fuels

The idea of using tungsten as a matrix in which to imbed fissile fuel for high temperature propulsion applications came up as early as the late 1940's in view of the known high temperature capabilities of tungsten and its known compatibility with hydrogen. In essence, cermet fuels are spherical particles of UO_2 (or UN) embedded in a matrix of tungsten or molybdenum arranged in a geometry that provides the requisite strength, uranium density and coolability. UO_2 volume fractions of up to 60% were achieved by the fabrication processes developed in the US. Fuel samples with these volume fractions performed well in nuclear and non-nuclear tests. The Russians have claimed reaching up to 70% volume fraction of UO_2 in their fabrication process; however the resulting "web" thickness of the matrix might not be sufficient structurally for several designs. The fuel concept was actually developed practically during two significant programs undertaken in the 1960's in the US:

1. The GE-710 program, which was a follow up to the Aircraft Nuclear Propulsion (ANP) program.
2. The ANL Cermet Nuclear Rocket program, undertaken as an alternative to the mainline carbide fueled Rover/NERVA program.

In addition, there was ongoing work in the USSR which focused on carbonitride fuel in the 1980's. Table 3.4.1 presents a summary of the cermet programs. As shown in the Table, the work on cermet fuels, while not nearly as extensive as the Rover/NERVA program, was substantial. It established fabrication procedures for UO_2 (and for a smaller set of UN) cermets in W. The results of a larger number of non-nuclear hot hydrogen tests at temperatures up to 3000 K (with cycling) established the viability of the fuel. Irradiation tests were performed for both steady state and transient conditions and the fuel performed excellently up to 1.6 a/o burnup at 1900 K and at heating rates up to 16000 K/sec. at 2900 K. No significant failures were observed.

<u>The GE-710 Program (37)</u>

During this program the goals changed several times from an initial demanding objective of:

1. Open cycle reactor systems with exit hydrogen temperatures of 2750 K for 100 thermal cycles and a 10 burn life.
2. Closed cycle reactor systems with inert gas coolant for temperatures up to 2420 K

to a much greater focus on longer term operation (~10,000 hours) at fuel temperatures in the 2000-2250 K range. These changes affected the test program, which were conducted mostly at temperatures considerably lower than NTP requirements. However, a lot of information on failure mechanisms and materials behavior were obtained.

Table 3.4.1
CERMET Program

- GE710 Reactor Program
  - 1962-1968
  - Open and closed cycle systems considered
- ANL Cermet Nuclear Rocket Program
  - 1961-1968
  - Alternative to Rover
- Russian Program
  - 1980's
- Significant Activities
  - Fabrication of UO_2 and UN based cermet fuels
  - Large number of non-nuclear hot H_2 tests at temperatures up to 3000 K with cycling
  - Steady state irradiation tests at fuel temperatures up to 2000 K and total U burnups up to 1.6 a/o
  - Transient nuclear heating at rates up to 16000 K/s and temperature of 2900 K

Table 3.4.2 presents a summary of the significant achievements of the GE-710 program. A few major issues will be discussed here in greater detail (a good summary discussion is presented in Reference 38). Specific non-nuclear tests were performed to determine the peak temperature capabilities of the cermet fuel system in flowing hydrogen.

Table 3.4.3 presents a detailed list of the results. W-25Re clad (W-UO_2-ThO_2) fueled specimens were heated to 3270 K for one hour, 3180 K for three hours, and 3070 K for 10 hours, (with two thermal cycles) without major damage. These non-nuclear tests were conducted in flowing hydrogen to determine fuel response. Additional tests were completed using the same fuel clad with W-30Re-30Mo for 10-1000 hours at temperatures of 2970 K to 2670 K respectively and up to seven thermal cycles. None of the tests exhibited any reaction or sensitivity of any sort to flowing hydrogen at the temperatures tested.

Post-test examinations showed that the thermal stability of the W-25Re clad W-(UO_2-ThO_2) specimens was excellent with the exception of the specimens tested at 3270 K which lost their core-cladding bond. Microstructure examination did not indicate fuel melting; however a layer of fuel was completely deposited around the core at the core-cladding interface. The volatility of the fuel at 3270 K was evidently high enough that vapor phase transport to the void area occurred. Cladding of specimens tested to 3270 K was large grained, but clean and single phase with no porosity. Specimens tested at lower temperatures showed no separation at the core-cladding interface; however cracking was observed in the core matrix of all three specimens. It is suspected that this cracking or separation occurred during sectioning of the specimen for microstructural examination. Pulse-echo inspection of the fuel indicated complete bonding of the cladding after testing but before sectioning.

An alternate fuel development approach investigated during the latter portion of the 710 program involved the use of UN instead of UO_2 as the fuel bearing material in the cermet matrix. UN has several potential advantages over the oxide form. UN has the highest specific uranium content (W-72 vol.% UN is equivalent to pure UO_2), the highest thermal conductivity, the highest resistance to deformation at elevated temperatures, and reportedly more favorable resistance to high burnup at temperatures about 1470 K. The linear expansion coefficient of UN is also lower, a fact that would reduce volumetric expansion and eventual cracking in cermet fuels using tungsten. (See Figure 3.4.1). The temperature limitations imposed by the dissociation of UN will need to be considered in designs.

- **Results of Steady State Nuclear Tests**

The first series of irradiation experiments was performed in the ETR facility at INEL. As in all subsequent irradiations the fuel was either UO_2 or ThO_2 stabilized UO_2 at a volume fraction of 60%. Matrix materials were W, W-Re and Mo with Ta, Ta-10W, W-30Re-30Mo and Nb cladding. Approximately half of the fuel samples developed fission gas leaks. As shown in Figure 3.4.2, this behavior correlates well with high temperature and burnup.

Further testing was performed in the <u>L</u>ow <u>I</u>ntensity <u>T</u>est <u>R</u>eactor (LITR). This test contained samples with W and W-Re matrix material and W-30Re-30Mo and W-25Re-3Mo cladding. The results of this test are summarized in Table 3.4.4. The behavior in terms of failures was similar to that of the ETR test.

A significant improvement was obtained in a third series of tests performed in Oak Ridge Research (ORR) reactor with basically the same matrix – cladding combinations. This performance improvement, as shown in Figure 3.4.3, was achieved by reducing the density of UO_2, thus providing void space for accumulating fission products, which resulted in lower stresses on matrix cladding.

The following conclusions were drawn at the end of the GE-710 program:

1. Fuel failure modes experienced with these cermet compounds were primarily:
    - Loss of oxygen from UO_2 at high temperature followed by formation of substoichiometric UO_2, free uranium and penetration of the cladding wall with subsequent thermal cycling.
    - Volume expansion and eventually cracking of fuel matrix (W-UO_2) during extremely high temperature operation and after significant thermal cycling.
    - Void formation between the cermet fuel and the alloy "can" cladding during fabrication and early operation.
    - Fission product damage/ release after 4000/7500 h of operation at 1870-2270 K in fuel specimens sintered to high (95% or greater) theoretical densities.
    - Preferential vaporization of Mo and other lower melt point materials out of the clad at higher temperatures above 2470 K.
2. The physical mechanisms determined to cause the above failures are (in order of occurrence):
    - Transparency of tantalum and tantalum alloys (T-111) to oxygen at intermediate and high temperatures. **No tungsten alloy** clad material exhibited any oxygen transparency at any temperature tested.
    - Volume expansion (and eventually cracking) is caused by incompatibility of the coefficient of thermal expansion between tungsten and UO_2 in the fuel matrix. At high temperatures and after multiple thermal cycles, fuel particles pull apart form the tungsten matrix.
    - Void formation appears to be caused by difficulties in getting a good seal between the metal alloy "can", the internal metal alloy tubes, and the surfaces of the cermet fuel material. Insufficient permeation of alloy-clad material into the cermet during autoclaving leaves

weakness that develops into a void.

- Fission product damage to the cermet and eventually to cladding material is caused by accumulated buildup of pressure, lattice stresses, and dislocation weaknesses inherent in fuel materials under irradiation. The W-UO_2 fuel region obviously has significant strength to resist these forces for the burnup achieved during testing before damage thresholds were reached. A significant improvement in burnup fraction was obtained near the end of the program when cermet fuels sintered to lower theoretical densities (84% to 90%) were tested in reactor experiments. Almost a fact-or of 10 increase in burnup capability (fission/ cm^3) was obtained simply by giving the products additional area for expansion without exerting stresses in excess of the capability of the tungsten matrix at elevated temperature.
- Molybdenum (Mo) and other materials are commonly used as alloying agents in tungsten alloy cladding to add some ductility to an otherwise extremely brittle material. Mo is not a good candidate for high temperature applications however. Above 2470 K the vapor pressure starts getting significant. Addition of Mo to W-30Re to form W-30Re-30Mo lowers the alloy melting point to 2970 K.

These failure modes and their physical causes can be corrected using modern ceramic materials technology.

The ANL Cermet Fuel Nuclear Rocket Program (39)

The primary fuel choice for this program was UO_2 imbedded in tungsten matrix, similar to the GE-710 program choice. Gadolinia was used to stabilize the UO_2 at ANL (in contrast to the ThO_2 used for the 710 program). Several fuel fabrication procedures were attempted with mixed success. The process that gave the best results was a powder metallurgical process that produced near net shape fuels, cold isostatic pressures followed by sintering at ~ 1950 K and finally vapor deposition of cladding on the coolant channels. The deposition of uniform cladding thickness proved to be difficult in the 1960’s, but advances in CVD technology should make the process much more effective at this time. Figure 3.4.4 shows a photograph of several of the fuel samples produced for the program. Excellent uniformity was observed in the composition and the coolant channels were remarkably straight even for the long fuel specimens.

A set of non-nuclear tests were performed in two hydrogen loops. Figure 3.4.5 shows a compilation of results of representative samples cycled in hydrogen at elevated temperatures. Fuel loss (%) is plotted against time at temperatures (2770 K) and number of thermal cycles. The gadolinia stabilized fuel showed excellent retentivity at 2770K for times up to 45 hours and over ~ 180 cycles. Other test showed that flowing hydrogen (700 psi) at temperatures exceeding 2700 K had essentially no impact on fuel loss rates.

Nuclear tests on the ANL cermet samples were run in the transient test reactor TREAT. Eight specimen cermet fuels, each with seven coolant holes and vapor deposited W cladding, were tested. The duration of the tests were generally 200 to 430 msec (the normal TREAT pulsed operation), although two specimens were subjected to "flat top" transients lasting 2-3 seconds. Table 3.4.5 summarizes the results. A video recording was performed of the tests. A few failures were observed (fuel material ejected from top of the samples). These were attributed to fabrication issues, particularly coating thickness irregularities. The last two samples were subjected to multiple transients at heating rates of up to 16000 K/s. The samples showed no evidence of damage after these severe thermal and neutronic burst conditions (at temperatures up to 2870 K and power density of 30 MW/l).

In addition to the cermet fuel development system design studies for a large (2000 MW) and a small (200MW) NTP system were conducted. The detailed core designs were supported by critical experiments which were performed to obtain basic physics, safety and control parameters. No design problems were identified and it was observed that very compact designs could be achieved. It is of interest to note that even in the 1960's there were attempts made to evaluate the impacts of using ^{233}U as the fissile material. Figure 3.4.6 shows the results of parametric studies of system weight vs. thrust for ^{235}U and ^{233}U fueled systems for the hot-bleed and topping cycles respectively. Significant mass savings with the ^{233}U system are noted.

Table 3.4.6 presents a summary of the significant results of the ANL Cermet Fuel Program.

Recent Work on Cermet Fueled Systems

The potential of cermet fuels for NTP, bimodal and high power steady state (multimegawatt) systems has been recognized by a number of organizations. ANL performed a detailed conceptual design of systems using cermet fuels for various SDIO applications (the Multimegawatt Space Nuclear Power Program (40)–1980's). ANL teamed up with GE and Rocketdyne for several other systems studies using cermet fuels for NTP and multimegawatt applications. Pratt and Whitney used cermet fuels for several of their NTP systems designs (41). Several independent studies conducted by different organizations (Pratt and Whitney, B&W (42), etc.) have all led to the conclusion that cermet fuels are a promising choice for NTP as well as high power steady state systems. For the bimodal system studies conducted by the Air Force and DOE, cermet fueled systems proved to be a strong contender.

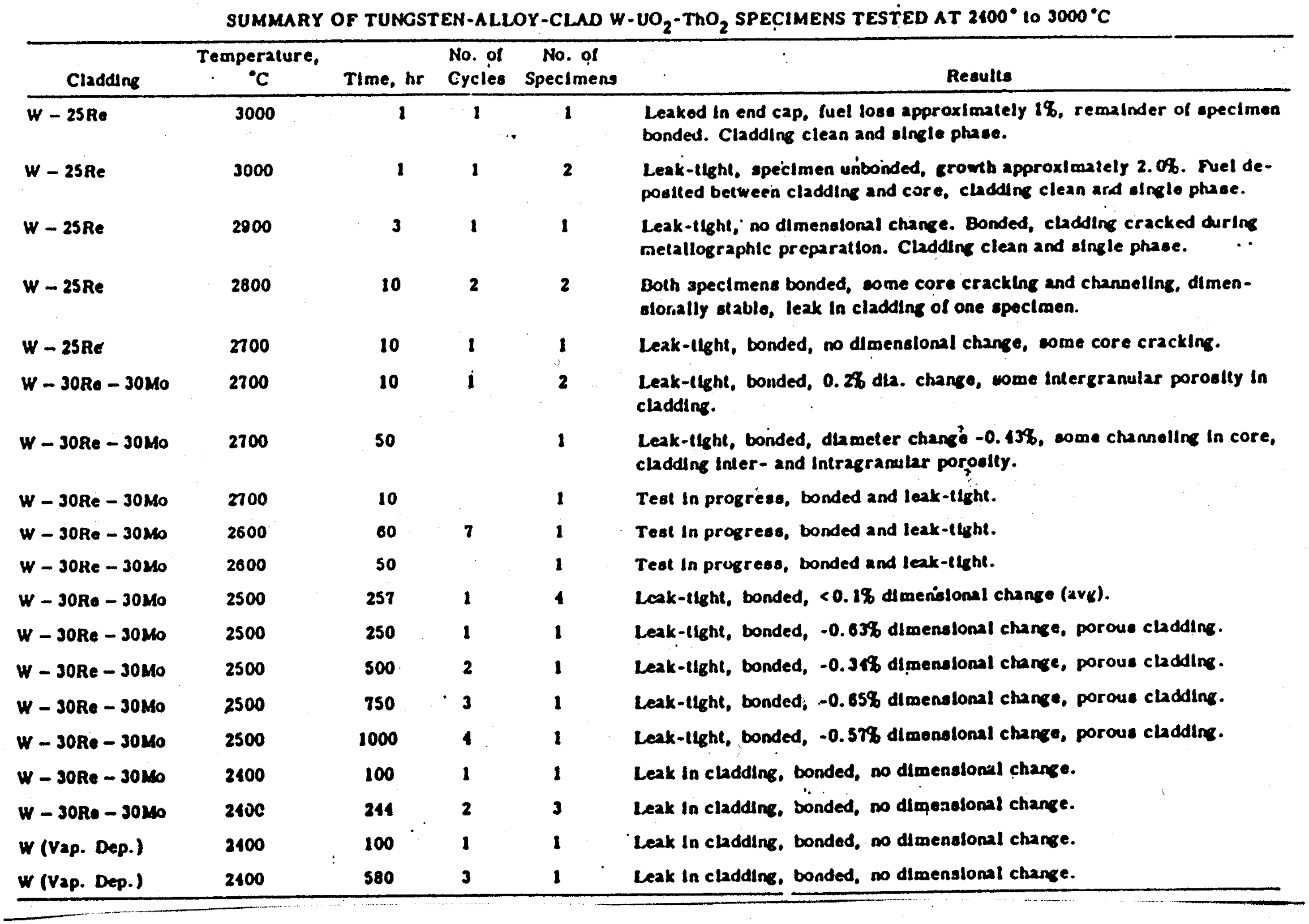
Unfortunately, resources have not so far been allocated in sufficient amounts to make significant progress in technology development since the 1960's. As part of the multimegawatt technology development program, ANL (W based) and PNL (Mo based) performed work on recapturing cermet fuel fabrication technology. Surrogate materials were used (in place of UN), but both ANL (43) and PNL (44) were able to use powder metallurgy techniques to produce excellent uniform microstructures in these surrogate fuels. This leads to confidence in the belief that fabrication of cermet fuel can be reinstated at laboratories and industry. The first step in the recapture of cermet fuel technology will be to establish fabrication methods to produce quality controlled fuel samples for testing.

Based upon currently available data, Pratt and Whitney (38) prepared performance potential curves for cermet fuels (in a manner similar to what was done for carbide fuels (Figure 3.1.5). Their analysis results are shown in Figure 3.4.7. It should be noted that there are several potential mechanisms that could limit the life of the fuel at the elevated temperatures and the actual test data are for very short times for the high temperatures. Thus the information in Figure 3.4.7 should be viewed as a very preliminary estimate at this time.

Table 3.4.2
Summary of Significant Results from GE-710 Program

Activities	**Brief Description**	**Major Conclusions**
Fuel Fabrication	1. Powder metallurgy process to fabricate UO_2 cermet hexagonal wafers (W and Mo matrix) 2. Stacking wafers in W-Re-Mo hexagonal cans with electron beam welding closure 3. Hot isostatic pressing to bond clad to matrix	Electron beam welding of total clad containment followed by hot isostatic loading to matrix gave a second barrier to fission product or fuel escape
Out-of-Pile Thermal Cycling and Life Tests	1. W-UO_2 cermet clad in W-Re was leak tight after 50 hours @ 2865 K in hydrogen 2. W-UO_2 cermet clad in W-Re-Mo was leak tight after 51 cycles over 500 hours from 2420 K to 530 K 3. 200,000 total hours of life testing in 68 samples of refractory metal clad UO_2 cermet at temperatures to 1920 K	Early cycling difficulties due to brazing closures and lack of UO_2 stoichiometric control. Electron beam welding closure and development of stoichiometric control techniques brought demonstrated success
In-pile Irradiations	20 fuel element specimens (W-UO_2 cermet clad in W-Re-Mo) tested for a total of 50,000 hours in LITR and ORR at temperatures to 1900 K and burnups of 1.4 x 10^{20} f/cc (~ 1.6 a/o uranium)	Fission product gas pressure cause of failure at higher burnups – far beyond rocket application burnups
Critical Experiments	Nine critical experiments were run to study: 1. Fast reactor critical configurations 2. Neutronic properties of uranium, tungsten, rhenium, and molybdenum 3. A number of reflector control schemes and elements	Benchmark critical series useful as experimental validation of today's analytical techniques
Reactor System Designs	1. Several designs including liquid metal and gas coolants 2. 850-870 Isp NTP design 3. 30,000 – 200,000 lb. thrust 4. Control analyses	No design problems identified

Table 3.4.3
Isothermal Cermet Material Tests in Hot Hydrogen



SUMMARY OF TUNGSTEN-ALLOY-CLAD W-UO_2-ThO_2 SPECIMENS TESTED AT 2400° to 3000°C

Cladding	Temperature, °C	Time, hr	No. of Cycles	No. of Specimens	Results
W – 25Re	3000	1	1	1	Leaked in end cap, fuel loss approximately 1%, remainder of specimen bonded. Cladding clean and single phase.
W – 25Re	3000	1	1	2	Leak-tight, specimen unbonded, growth approximately 2.0%. Fuel deposited between cladding and core, cladding clean and single phase.
W – 25Re	2900	3	1	1	Leak-tight, no dimensional change. Bonded, cladding cracked during metallographic preparation. Cladding clean and single phase.
W – 25Re	2800	10	2	2	Both specimens bonded, some core cracking and channeling, dimensionally stable, leak in cladding of one specimen.
W – 25Re	2700	10	1	1	Leak-tight, bonded, no dimensional change, some core cracking.
W – 30Re – 30Mo	2700	10	1	2	Leak-tight, bonded, 0.2% dia. change, some intergranular porosity in cladding.
W – 30Re – 30Mo	2700	50		1	Leak-tight, bonded, diameter change -0.43%, some channeling in core, cladding inter- and intragranular porosity.
W – 30Re – 30Mo	2700	10		1	Test in progress, bonded and leak-tight.
W – 30Re – 30Mo	2600	60	7	1	Test in progress, bonded and leak-tight.
W – 30Re – 30Mo	2600	50		1	Test in progress, bonded and leak-tight.
W – 30Re – 30Mo	2500	257	1	4	Leak-tight, bonded, <0.1% dimensional change (avg).
W – 30Re – 30Mo	2500	250	1	1	Leak-tight, bonded, -0.63% dimensional change, porous cladding.
W – 30Re – 30Mo	2500	500	2	1	Leak-tight, bonded, -0.34% dimensional change, porous cladding.
W – 30Re – 30Mo	2500	750	3	1	Leak-tight, bonded, -0.65% dimensional change, porous cladding.
W – 30Re – 30Mo	2500	1000	4	1	Leak-tight, bonded, -0.57% dimensional change, porous cladding.
W – 30Re – 30Mo	2400	100	1	1	Leak in cladding, bonded, no dimensional change.
W – 30Re – 30Mo	2400	244	2	3	Leak in cladding, bonded, no dimensional change.
W (Vap. Dep.)	2400	100	1	1	Leak in cladding, bonded, no dimensional change.
W (Vap. Dep.)	2400	580	3	1	Leak in cladding, bonded, no dimensional change.

Table 3.4.4
Fuel Element Specimens and Operating Conditions for Series of Six GE710 Experiments

		Specimen Details				Test Condition[a]							
		Composition							Fraction of Fission Gas Released[d], R/B		Before Fissure[e]		
Specimen No.	Test No.	Cladding[f] OD/ID	Core, Vol. %	Length cm	Time @ Temp., Hrs.	Thermal[b] Cycles	Burnup[c] $f/cm^3 \times 10^{19}$	Mean Temp., °C	Prior To Fissure, Max.	After Fissure, Max.	Time @ Temp., Hrs.	Cycles	Burnup $f/cm^3 \times 10^{19}$
16N	4	306*/306	W-54UO$_2$-6ThO$_2$	4.0	13,405	80	6.7	1150	2.9×10^{-10}	NF	NF	NF	NF
25N	5	306	W-60UO$_2$	4.0	10,597	67	1.9	1175	1.5×10^{-9}	NF	NF	NF	NF
39N	6	256	(W-3Re)-60UO2	4.3	7,903	55	4.5	1260	1.3×10^{-9}	NF	NF	NF	NF
26N	5	306	W-60UO$_2$	4.0	10,597	67	4.1	1345[g]	1.5×10^{-9}	NF	NF	NF	NF
15N	4	306*/306	W-54UO$_2$-6ThO$_2$	4.0	13,405	80	8.1	1360	2.9×10^{-10}	NF	NF	NF	NF
32N	5	306	W-60UO$_2$	4.0	10,597	67	4.4	1400[g]	1.5×10^{-9}	NF	NF	NF	NF
13N	4	306*/306	W-60UO$_2$	4.0	13,405	80	7.6	1410	2.9×10^{-10}	NF	NF	NF	NF
11-1D	2	T-111	W-54UO$_2$-6ThO$_2$	23.6	5,000	53	.12-5.4	510-1425	10^{-10}	NF	NF	NF	NF
40N	6	256	(W-3Re)-60UO$_2$	4.3	7,903	55	5.4	1425	1.3×10^{-9}	NF	NF	NF	NF
45N	6	306	W-60UO$_2$	4.3	7,903	55	4.0	1430	1.3×10^{-9}	NF	NF	NF	NF
33N	5	306	W-60UO$_2$	4.0	10,597	67	6.7	1440	1.5×10^{-9}	NF	NF	NF	NF
53N	6	256	(W-3Re)-60UO$_2$	4.3	7,903	55	6.4	1440	1.3×10^{-9}	NF	NF	NF	NF
36N	5	306	(W-3Re)-60UO$_2$	4.0	10,597	67	9.0	1455	1.5×10^{-9}	4.1×10^{-4}	7,790	47	6.6
11N	4	306	(W-3Re)-60UO$_2$	4.0	13,405	80	10.5	1455	2.9×10^{-10}	$\sim 1.7 \times 10^{-5}$	~10,000	~55	~7.8
46N	6	306	W-60UO$_2$	4.3	7,903	55	5.4	1470	1.3×10^{-9}	NF	NF	NF	NF
6N	3	306	W-54UO$_2$-6ThO$_2$	23.6	5,000	33	.76-5.0	925-1510	10^{-9}	NF	NF	NF	NF
27N	4	256/306	W-60UO$_2$	4.0	13,405	80	9.5	1510	2.9×10^{-10}	$\sim 1.7 \times 10^{-5}$	~ 8,250	~48	~5.8
52N	6	256	(W-3Re)-60UO$_2$	4.3	7,903	55	5.5	1515	1.3×10^{-9}	$\sim 1.3 \times 10^{-4}$	7,757	53	5.4
35N	5	306	(W-3Re)-60UO$_2$	4.0	10,597	67	7.4	1535	1.5×10^{-9}	NF	NF	NF	NF
20N	4	256/306	(W-3Re)-60UO$_2$	4.0	13,405	80	9.4	1535[g]	2.9×10^{-10}	$\sim 1.7 \times 10^{-5}$	~ 7,740	~45	~5.4
1N-1D	1	306	W-54UO$_2$-6ThO$_2$	4.0	3,000	40	3.5	1540	3.2×10^{-9}	NF	NF	NF	NF

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[a] Conditions common to all tests are as follows: Thermal neutron flux, n/cm-sec. (3 to 3.9 x 10^{12}); maximum heat flux at fuel element surface, 53,000 Btu/ft^2; atmosphere, helium : 5% hydrogen except for 710-2 which used pure helium; pressure 0.141g/mm^2 (200 psl), operation temperature sensed by W/W-Re thermocouples in center of specimen.

[b] When specimen temperature dropped to less than 50% of test temperature.

[c] Average burnup determined from mass spectrometer and gamma scan analysis; fission density given in terms of core (cermet) volume.

[d] Fraction of fission gas released (R/I:) based on Er $^{1:6m}$ isotope.

[e] LT710-4 specimen fissures after the first release estimated from fission gas release and radiation monitor data and leak-free service performance of previous experiments.

[f] 306 alloy composition – W-30Re-30Mo (at. %); 256 alloy composition W-25Re-30Mo (at. %); powder metallurgy process unless otherwise indicated (*).

[g] Estimated temperature; thermocouple inoperative.

Figure 3.4.1
Effect of Thermal Expansion Difference Between Fuel and Matrix on Volume in Crease During Thermal Cycling

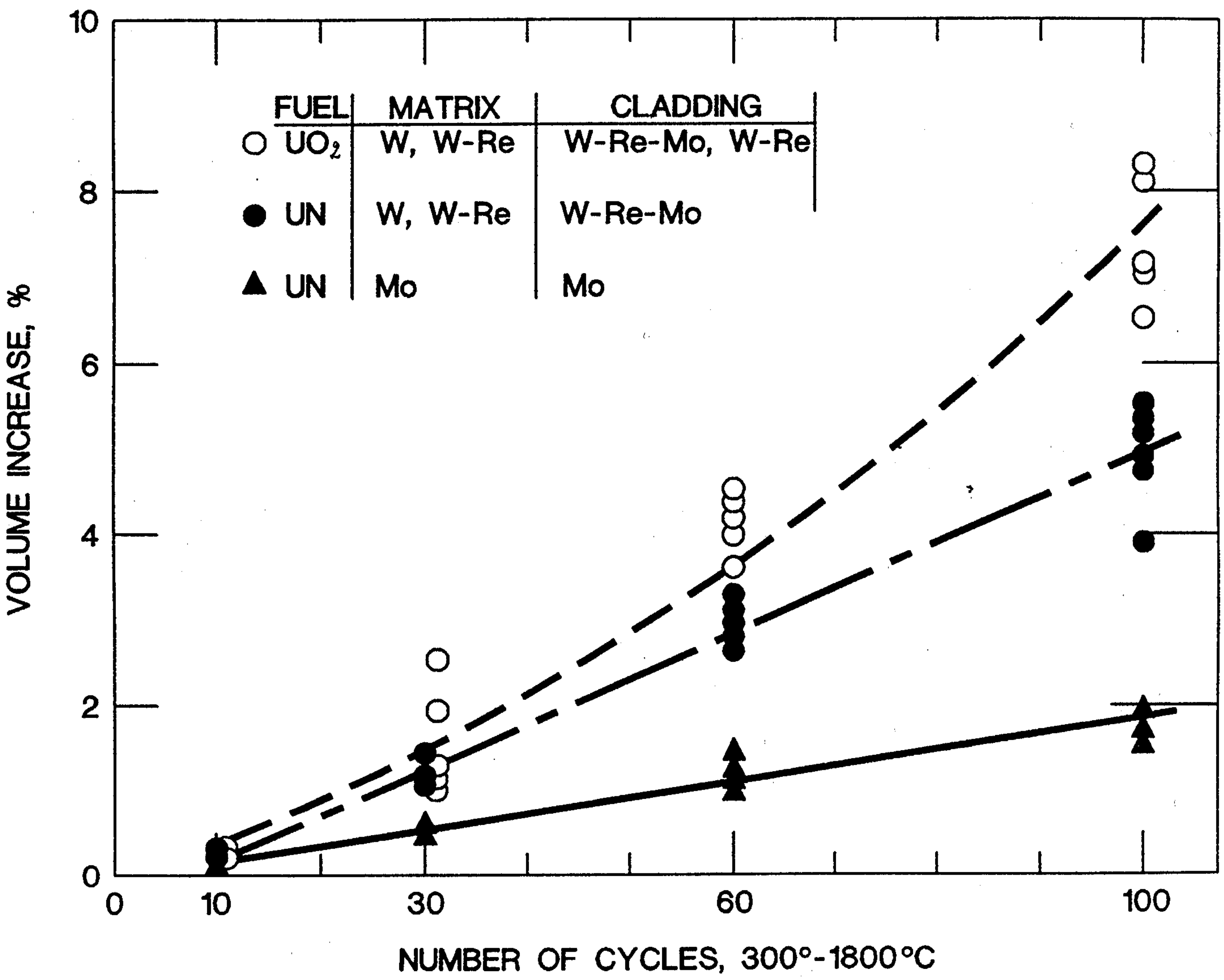


Table 3.4.5
Summary of TREAT Tests

Sample No.	Transient Duration, sec	Reactor Integrated Power, MW-sec	Maximum Recorded Surface Heating Rate, °C/sec	Maximum Recorded Surface Temperature, °C
1	0.43	164	1,700	800
1	0.3	284	3,900	1,460
2	0.3	377	5,600	1,790
3	0.2	487	8,000	2,200
4	2.1*	332	800	1,460
5	0.2	540	12,000	2,600
6	3.0	495	1,4000	2,050
7[1]	0.2	523	14,500	2,750
8[2]	0.2	532	16,000	2,750

* "Flat top" transient

[1] Sample given two additional transients of same severity.

[2] Sample given five additional transients of same severity.

Table 3.4.6
Summary of Significant Results of ANL Cermet Program

Activities	Brief Description	Major Conclusions
Fuel Fabrication	• Powder metallurgy process used to produce near net shape UO_2-W cermet fuel • Cold isostatic pressuring followed by sintering at ~1950 K • Cladding vapor deposited on coolant channels	• A large number of samples produced – various lengths (up to 50 cm)
Out-of-Pile Thermal Cycling and Life Tests	• Tests performed with UO_2-W cermets in hydrogen at temperatures up to 2770 K • Tests performed with UO2-W cermet specimens in flowing H_2 (700 psi) at 2700 K	• Minimal fuel loss at 2770 K in H_2 for up to 45 hours and under 180 cycles • No impact on fuel loss rates
In-Pile Irradiations	• Transient tests in TREAT reactor • Highest temperature reached 3023 K • Rates of 16,000 K/s	• One failure – attributed to fabrication problem • One test specimen survived eight consecutive transients at very high temperature rates and up to peak temperature of 2900 K
Critical Experiments	• Eight critical experiments performed to study - Basic physics and safety parameters of cermet fueled systems - Control systems	• Benchmark criticals provided excellent data for methods validation
Reactor Design Studies	• 2000 MW reference • 200 MW alternate • Core design and analysis • Control studies	• No design problems identified

Figure 3.4.2
Burnup as a Function of Reciprocal Absolute Temperature for Various Refractory-Metal In-pile Test Specimens, 1966

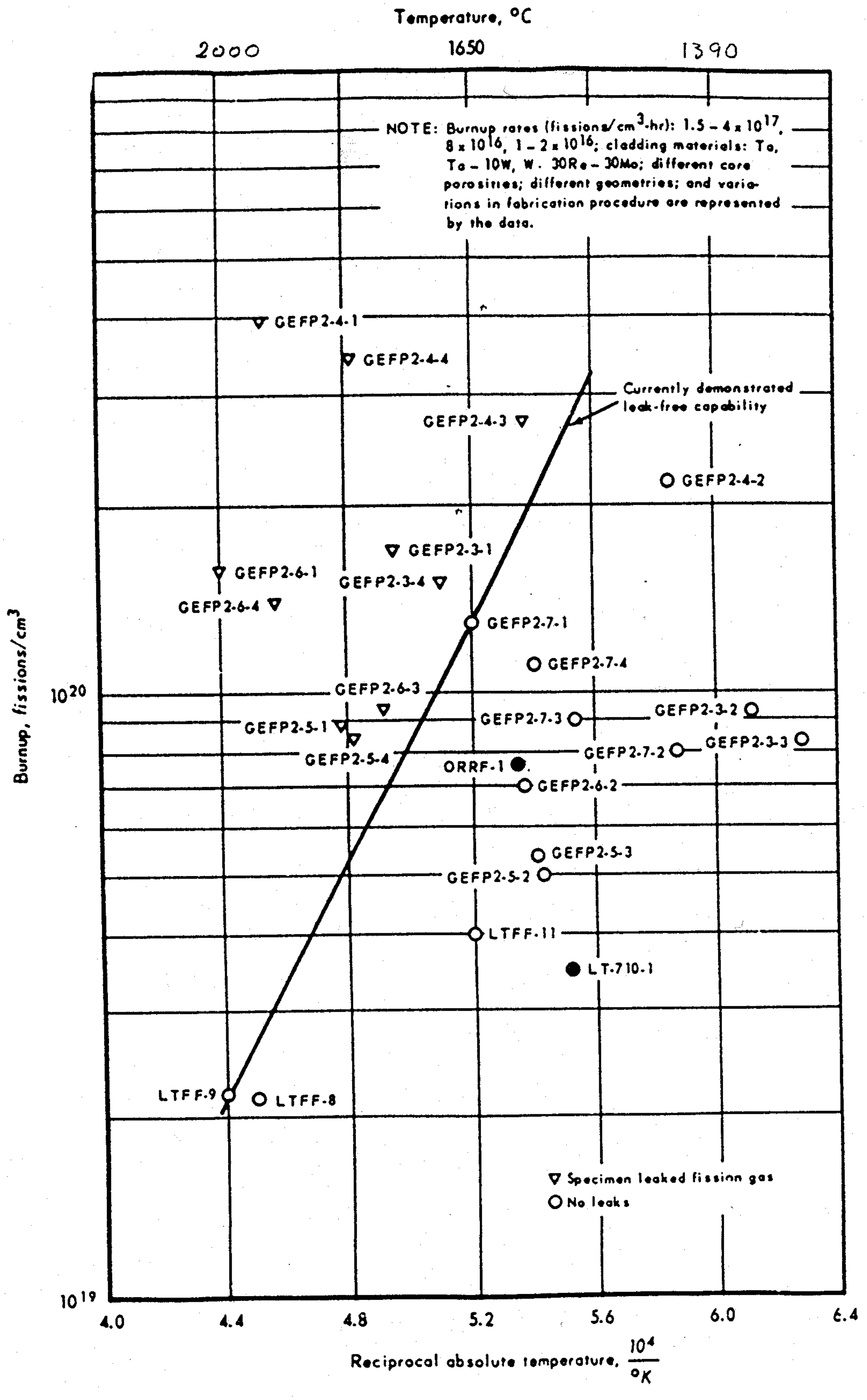
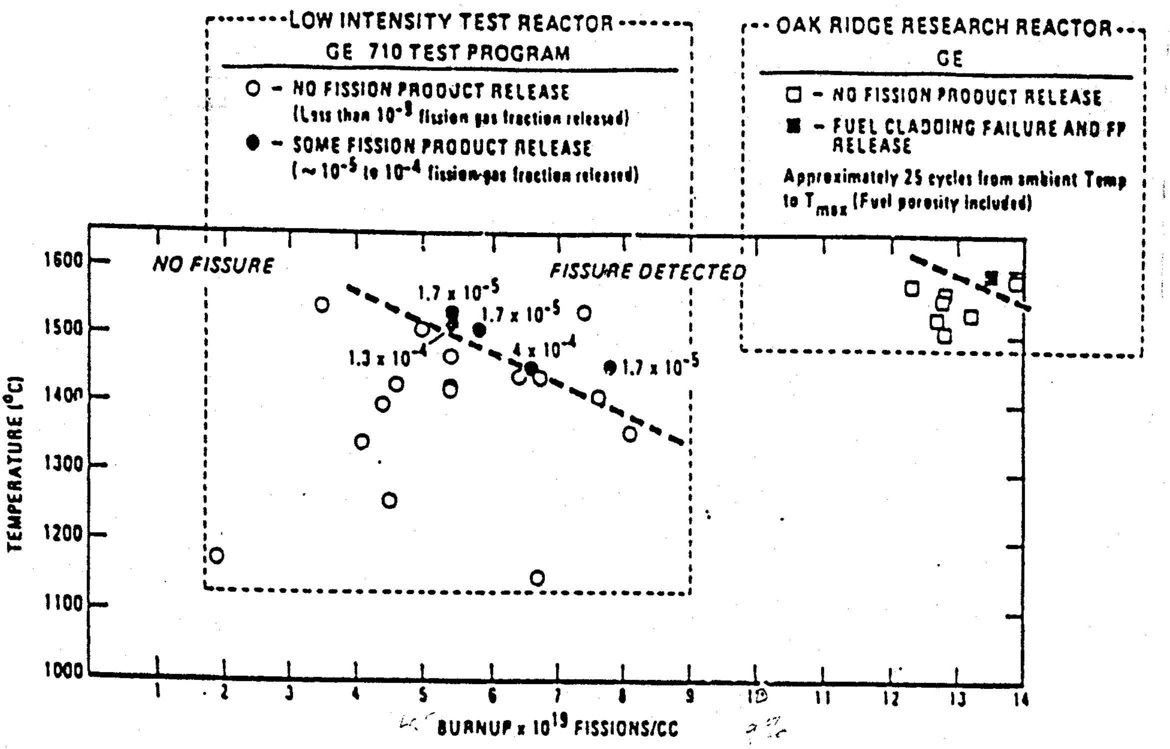


Figure 3.4.3
Relationship Between Peak Temperature and Defect Free Burnup Limit



ALL THERMAL CYCLING PINS THERMALLY CYCLED FROM 0.5 T_{max} to T_{max} DURING IRRADIATIONS – CYCLES RANGED FROM 33 to 80

Figure 3.4.4 UO_2-W Cermet Fuel Samples Prepared for the ANL Cermet Fuel Program

Figure 3.4.5
ANL Data – Temperature Cycling in Hot Hydrogen
(Material Capability at Cancellation of Program)

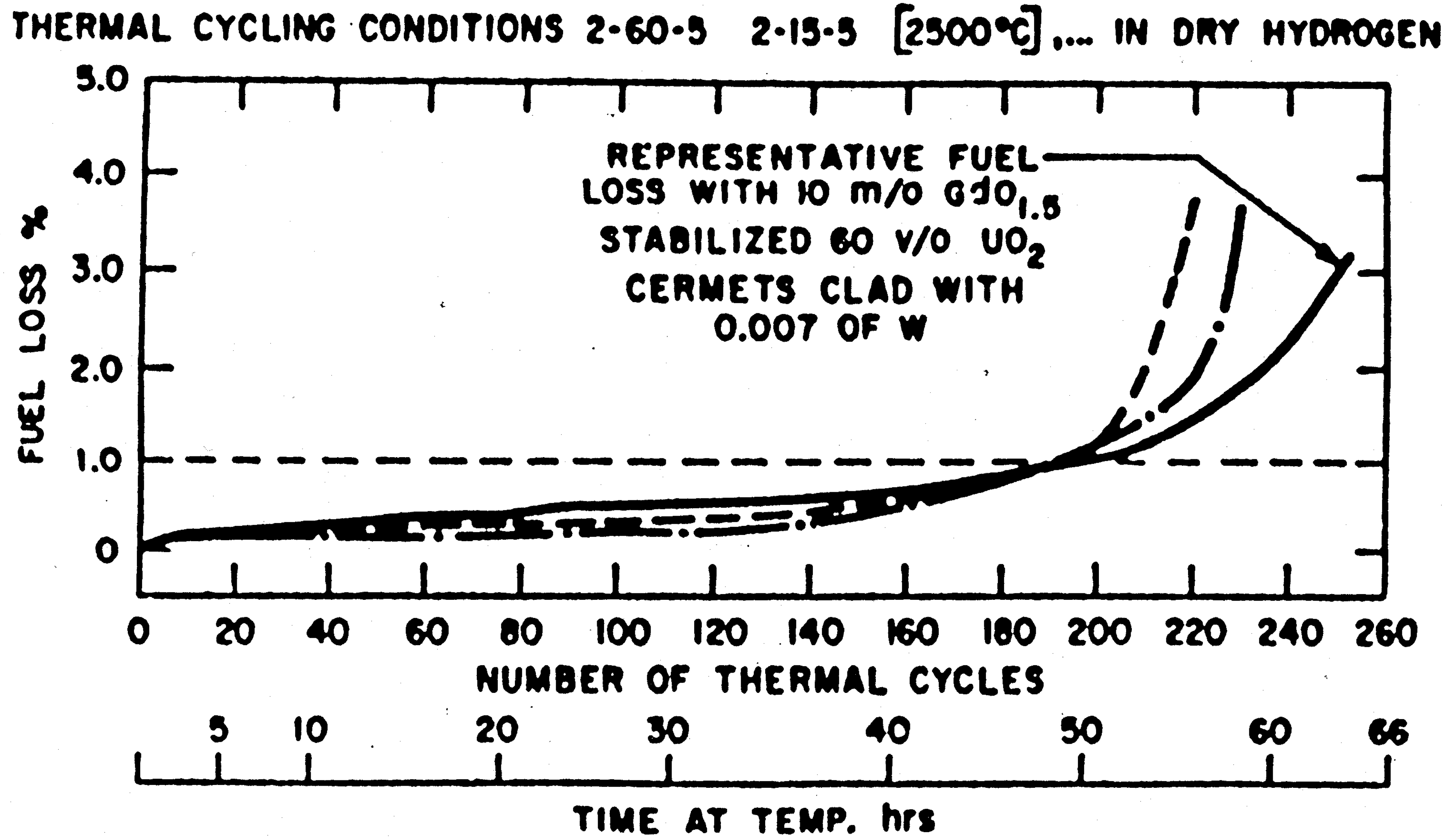
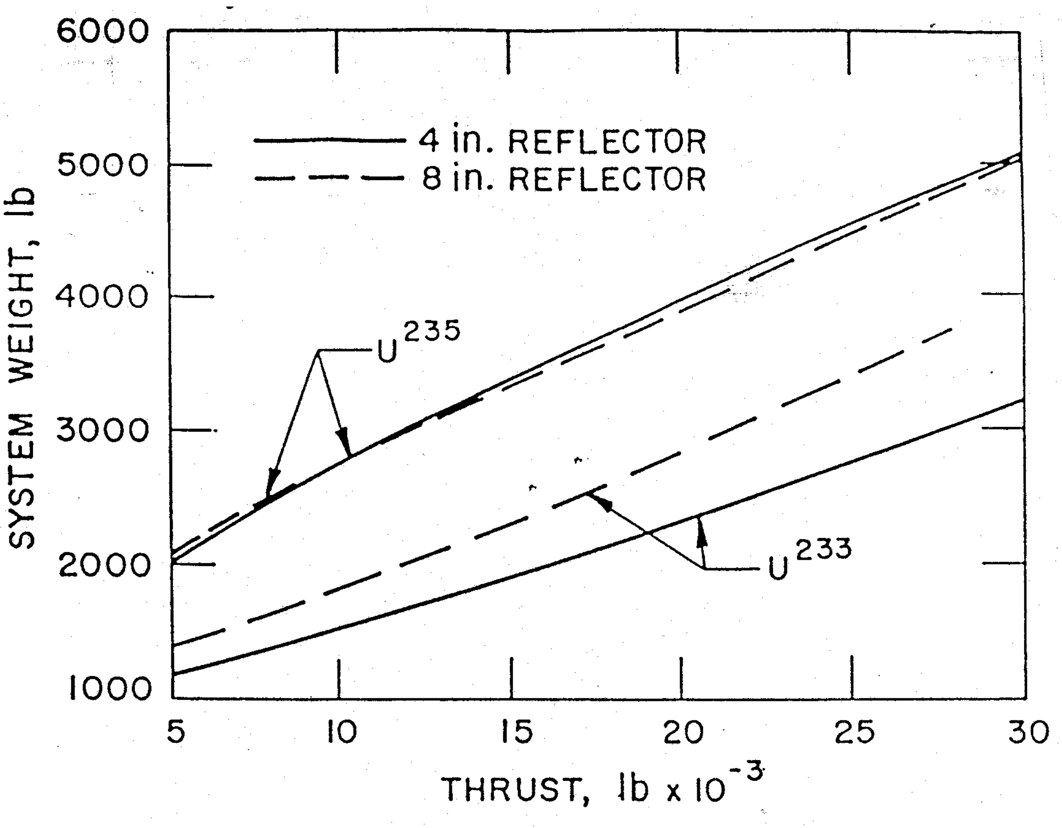
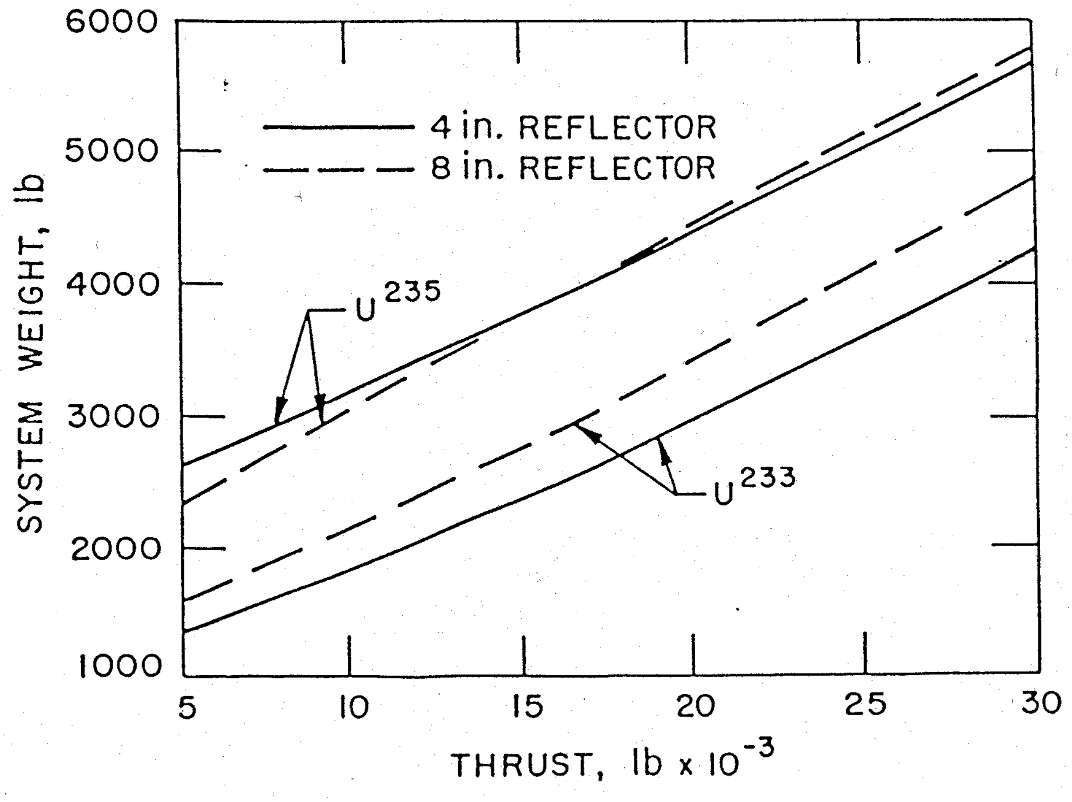


Figure 3.4.6
Calculated Results of Reduction in Weight for Hot Bleed Topping Cycle Systems with the Use of ^{233}U Fuel in Place of ^{235}U

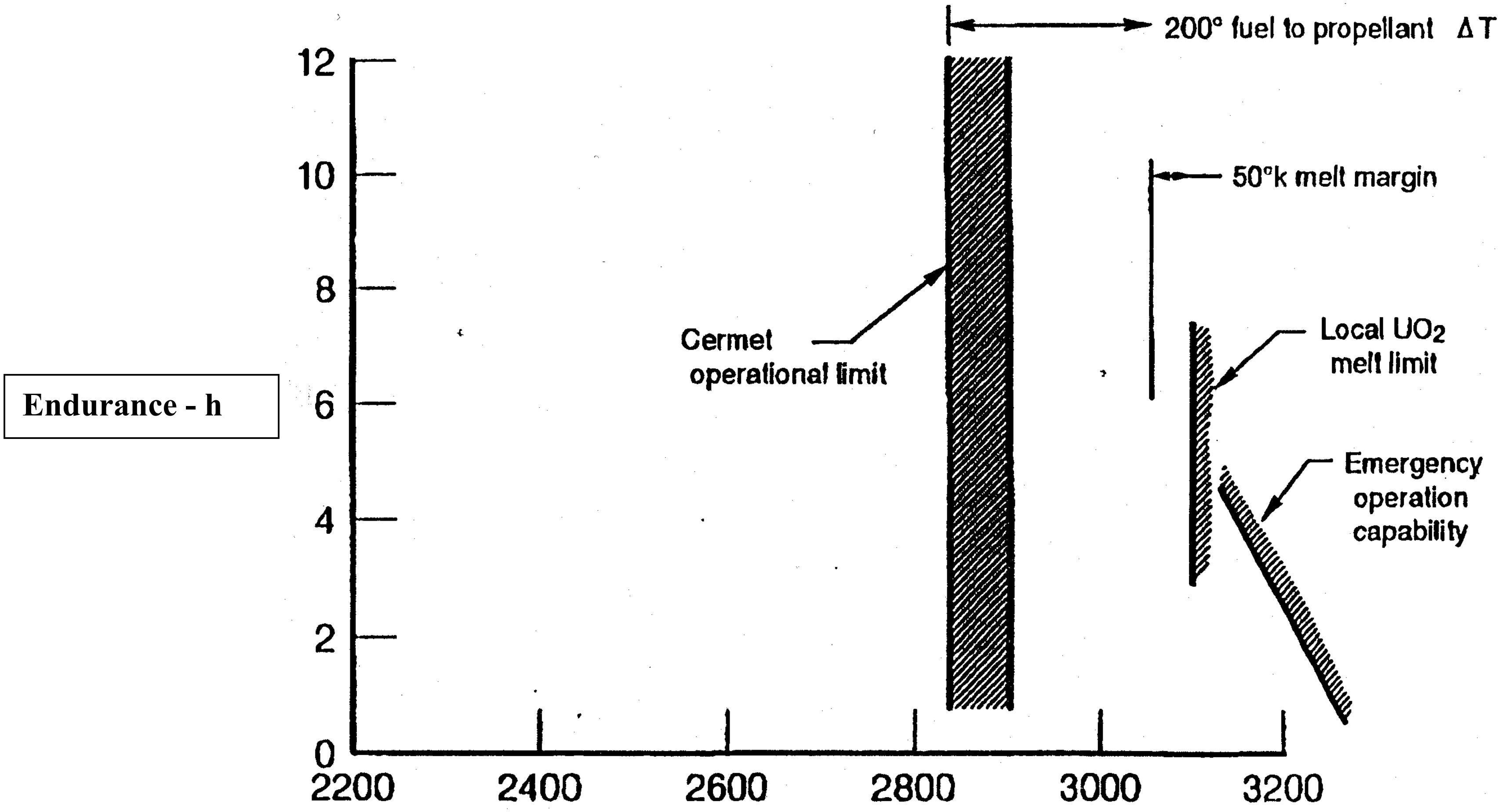


WEIGHT VS THRUST HOT BLEED CYCLE



WEIGHT VS THRUST TOPPING CYCLE

Figure 3.4.7
Cermet Operating Limits Can Be Established from Existing Data



Temperature - K

4. SUMMARY ASSESSMENT OF NTR FUELS AND RECOMMENDED PATH FORWARD

4.1 Bimodal Operations Considerations

Recently the NASA technology evaluation process(45) ranked bimodal systems higher than pure NTP systems in its technology planning. The bimodal systems incorporate both propulsion and power mode operations within its overall structure. The propulsion mode operation provides the necessary thrusts for various parts of a mission, while the power mode operation provides for housekeeping and other steady state power requirements. The steady state power requirements are modest (tens of kilowatts) and the propulsion mode operations are interspersed in time between steady state operations. The additional complexity introduced by this mode of operation from the fuels perspective is the capacity to undertake the high stress propulsion mode operation (very high temperatures for short times) with a low burnup fission product inventory built into the fuel. The fission product inventory generally weakens the fuel and there are issues of long-term creep that need to be considered when gauging the performance and lifetime of the fuel. Non-nuclear hot hydrogen testing and analyses can answer some of these questions, but a definitive answer will require additional carefully controlled in-reactor experiments where the tests are conducted with the fuel specimens that have accumulated the appropriate amount of burnup.

There are currently no fuels that have been subjected to the test conditions needed. However, the cermet fuels have been tested separately to relatively high burnup and severe transient conditions. In addition, analyses of creep stress performed by ANL in support of the DOE/USAF NEBA program indicated that for postulated typical bimodal operational conditions, the cermet fuel would easily be able to withstand the resulting stress conditions (46,47). In addition, it is our belief that the Rover/NERVA composite fuel ([U, Zr]C,C) is also a good candidate for early utilization in bimodal reactor systems.

4.2 Summary Assessment of Fuels

In Table 4.2.1, the best performance data recorded to date for each of the candidate fuels is compiled. There are several important variables beside the peak temperature that make direct comparisons difficult. It is interesting to note that the US data are thirty years old, with no new technology demonstrations. The Russian data are more recent, but suffer from poor diagnostics.

Table 4.2.2 provides a summary assessment of the NTP fuels discussed in this report. In the Rover/NERVA program the coated particle fuel was abandoned in favor of the more robust (U,Zr)C,C. The composite fuel underwent significant development in the course of the program. There is the largest amount of nuclear testing for this fuel and it survived the longest time (109 minutes) in US test history, at temperatures of interest to NTP (2550 K). There is limited steady state irradiation experience. However, for initial bimodal and NTP applications, we recommend that this fuel development be pursued.

Cermet fuel have been tested extensively in a nuclear environment at steady state and some under transient conditions. It is also a robust fuel and has the intrinsic spectral shift safety feature ensuring subcriticality under water submersion accidents. While its ultimate temperature potential is lower than that of binary or ternary carbide fuels, it is clear that cermet fuels should be developed further for bimodal systems and early NTP systems. One of the potential advantages of this fuel is that the failure modes appear to be "graceful" as opposed to the observed "catastrophic" modes for carbide fuel.

Binary and possibly ternary carbide fuels have the highest temperature potential. Early experiences with carbide fuels have not been positive. The Rover/NERVA test fuel in NF-1 cracked badly under irradiation. Reports of the Russian work have been glowing; however actual data have been very hard to get and there is a possibility that diagnostics on test fuel are not available. Thus the primary benefit of the work is the sense that the advertised performances in carbide fuel are achievable—the actual test program will most likely have to be conducted domestically or jointly with the Russians with appropriately quality controlled fuel samples. Finally, the SNTP program also came to the conclusion that binary or ternary carbides were most likely to meet their demanding requirements. It is noted that the carbide fuel have been designed and tested in various geometries. It seems reasonable to develop these fuels for the longer-term NTP applications.

4.3 Recommendations for Future Activities

Clearly all four classes of fuels discussed – the composite fuel ([U, Zr]C, C) Rover/NERA, the mixed carbide (Rover/NERVA, PBR, CIS), the Cermet (GE-710, ANL) and the Carbonitride (Russia) all have significant advantages and claims to further development and use. The constraints on budget require that no more than two be pursued in depth at this time. Thus, one needs to analyze potential mission needs and arrive at criteria to make a reasoned selection.

- Bimodal systems are of much greater interest at this time than pure NTP systems. This implies that very high specific impulse is not of near term interest. Instead the ability to operate in propulsion mode with a low built-up inventory of fission products is critical. This calls for a rugged fuel with proven fission product retention capabilities both for long term steady state operations and short-term high temperature propulsion operations.

  Advantage: 1) UO_2/W Cermet Fuels
  2)(U, Zr) C, C Composite Fuels

- There is strong emphasis on near term demonstration of capabilities of the fuels and much weaker emphasis on long term high Isp potential. This calls for fuels with a large experience/database in all or part of the bimodal and NTP operation cycle.
  Advantage: 1)(U,Zr)C,C Composite Fuel
  2)UO_2/W Cermet Fuels

- For the longer term higher performance is certainly desirable. For this the highest potential fuels should be examined. However, work on these fuels can be deferred or undertaken at a lower level in the early stages of this program. Fuels in this category are the ternary carbides (U, Nb, Zr)C with possible inclusion of Hf and Ta carbides, and the carbonitrides.

For both these longer term fuels there will be a value in engaging the Russian institutes in a joint venture to be able to access their large database. As discussed in Section 3.3, attempts to do so to date have not been very successful. The contracts with the Russians will have to be carefully constructed and monitored. If it is possible to engage them as strategic partners in the long-term goal of using bimodal/NTP systems in space, the chance for success in these joint ventures will increase.

Based upon these considerations, our recommendation is to pursue the (U, Zr)C, C composite fuel and UO_2/W cermet with most of the available resources early in the program. The more advanced mixed carbide fuels could be pursued at a more deliberate pace – possible at universities. There is also merit to engaging the Russians in a long-term program involving ternary carbide and carbonitride fuels.

In terms of actual work that needs to be undertaken for the fuels, the key strategic issues for Bimodal/NTP applications are:

- Relative weighting of recapture of NERVA technology and investment in advanced fuels for higher performance-reliability.
- Extent of cooperative activities with the Russians.
- Need and time of nuclear furnace testing or equivalent.

The key technical issues are:

- Fabricability (QA/QC)
- Adequate time/temperature capabilities
- Hydrogen compatibility
- Composition stability at temperature
- Cycling capability
- Fission product retention
- Corrosion/cracking effects
- Availability of range of test facilities needed.

A five year plan has been prepared for the technology development activity as shown in Figure 4.3.1. This plan is consistent with earlier work (48), modified by current mission need perceptions. In order to minimize costs, it is intended to use the existing hot hydrogen flow facility at BWXT (49) (with the appropriate modifications) to perform the proposed non nuclear tests. The early nuclear tests are planned to be conducted in the ATR for which a significant amount of planning has already been conducted earlier (50). Early work is proposed on the development of a comprehensive nuclear testing strategy which lays out clearly all the information needs and the approaches to obtaining them. It is also proposed to develop and maintain fuel performance codes in parallel with the conduct of experiments.

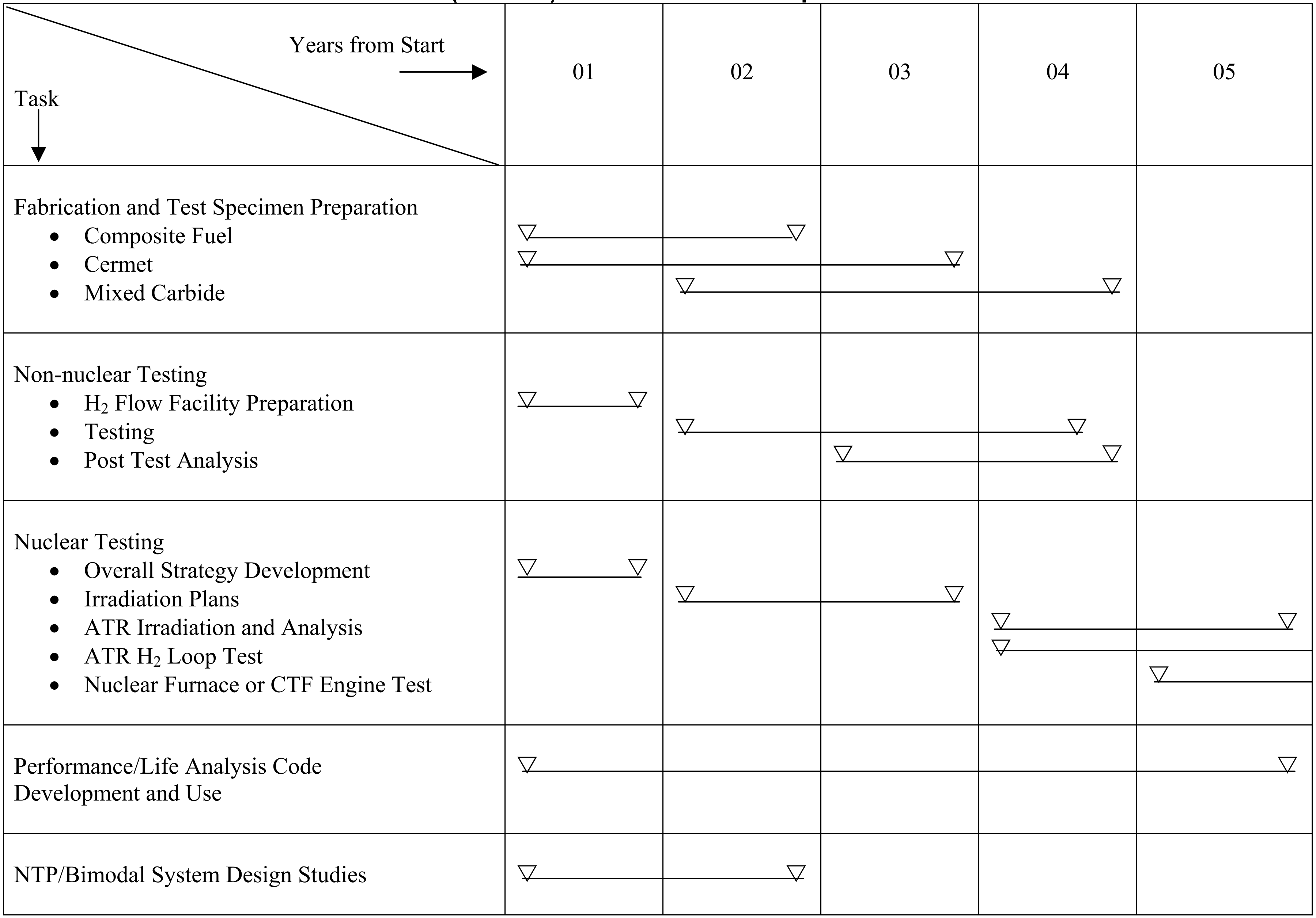
Table 4.2.1
NTP Fuel Temperature Parameters
(Test Data – Best Performance)

System	Fuel Type	Coating/ Clad	Temp. K	Lifetime	# of Cycles	Year
Rover/ PEWEE 1	Pyrolytic C-UC_2 Particle Dispersion	ZrC	2600	40 min.	1	1968
Rover/ NF-1	(UC, ZrC)C Composite (U, Zr)C Solid Solution	ZrC	2450	109 min.	6	1972
Cermet	UO_2	W	2700	0.2 s	8	1967
Russian	(U, Zr, Nb)C, C	?	3100	4000 s	?	1980's

Table 4.2.2
Summary of Assessment of Fuels

Fuel Type	Features		Recommendation
	Positive	Negative	
(U, Zr)C,C Composite Fuel			
Prismatic	• Most experience • Proven operating experience at 2550 K for 109 m in reactor.	• Limited steady state irradiation experience • Thermal shock and cracking problems	Pursue for near term applications
Mixed Carbide Fuels	Highest temperature potential		Pursue for longer term applications
Prismatic	• In reactor experience	• Extensive cracking in tests • Limited experience • Fabrication process development needed	
Particle Bed	• High T/W operation temperatures	• Limited experience • Fuel/FP loss anticipated • Mechanical Issues	Subset of above (same fuel type but geometric differences)
Twisted Ribbon	• Potential large database • Excellent high temperature performance reported	• Long effort at getting quantitative data not very successful • Uncertain mass losses	Develop Russian cooperative program
Cermet Fuels	• Robust fuel • Safety features • Steady state operating experience (Bimodal or surface power)	• Ultimate temperature potential is lower than mixed carbide	Pursue for near and intermediate term (also potential application in MMW steady state system)

Figure 4.3.1
NTP (Bimodal) Nuclear Fuels Development Schedule



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