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SPP-67110-0008
B-SE-0155

April 24, 2006

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Subject: **Prometheus Project Reactor Module Final Report, for Naval Reactors Information**

References: See Page 8

Enclosure: (1) Prometheus Project Reactor Module Final Report, Revision 1

Dear Sirs:

This letter transmits the Prometheus Project Reactor System Final Report.

**BACKGROUND**

Project Prometheus was established in 2003 with a goal of developing the first nuclear reactor-powered propulsion system for a spaceship and demonstrating that it can be operated safely and reliably for civilian deep-space exploration missions. The initial application of space fission power being evaluated was the Jupiter Icy Moons Orbiter (JIMO), a nuclear electric propulsion spaceship intended to perform deep-space scientific research.

In March 2004, the Naval Reactors Program was assigned responsibility for design and delivery of the Reactor Module for Project Prometheus. The spaceship is comprised of a multi-mission Deep Space System coupled with a mission specific Mission Module. The Deep Space System consists of two modules:

1. The Reactor Module, which includes the nuclear reactor and the energy conversion equipment to produce electric power.
2. The Spacecraft Module, which includes the spacecraft structure, radiator panels (heat rejection segment), electric power control and distribution equipment, and the ion propulsion system.

Design responsibility for the Reactor Module (with the exception of the Aeroshell reentry protection cover) was assigned to the Naval Reactors Prime Contractor Team (NRPCT) and the approval responsibility was assigned to DOE-Naval Reactors (NR). In March 2005, the NRPCT recommended and Naval Reactors approved a gas-cooled reactor with a directly coupled Brayton energy conversion system for further development.

In September 2005, NASA priorities changed and NR Program participation in Project Prometheus was terminated. Closeout activities were initiated to provide an orderly conclusion to NRPCT work and to enable the future re-start of a nuclear space reactor project. Closeout reports were generated to document the pre-conceptual design work that was in-progress.

**DESCRIPTION OF REPORT**

The Enclosure (1) Prometheus Project Reactor Module Final Report is an integrated project summary that integrates the key program perspectives and technical findings on all space reactor work conducted under NR cognizance. It describes the formal programmatic and technical information that was generated, how and where it is stored, and how future researchers can access the information. A detailed bibliography is included to serve as a roadmap that identifies the many external references that were used by the NRPCT, as well as important documents that were generated by the NRPCT.

Enclosure (1) is divided into three volumes. Volume 1 is a Program Summary that describes the scope of the Prometheus Project and NRPCT involvement as the design agent for the Reactor Module. The NRPCT approach to the program management of the project is discussed, and key perspectives are provided.

Volume 2 is a Technical Summary that describes the key design challenges of the Prometheus Reactor Module and summarizes NRPCT technical findings. This volume integrates the conclusions of the detailed design reports that have been produced as part of pre-conceptual design and project close-out. These detailed reports are referenced to provide a path for future researchers to understand the underlying technical work that is behind the conclusions discussed in the Technical Summary.

Volume 3 is a Bibliography which contains references to documents generated by the NRPCT, documents from past space reactor projects, reports produced under subcontract, and other pertinent references. Nearly 1800 references have been gathered and organized. This bibliography will be an important resource for future researchers.

Very truly yours,

MJ Wollman, Manager
SPACE POWER PROGRAM
KAPL

MJ Zika, Manager
SPACE ENGINEERING
Bettis

# Project Prometheus Reactor Module

## Final Report

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# Acknowledgement

The Naval Reactors Prime Contractor Team acknowledges the contributions made by these and other organizations to Prometheus Project Reactor Module development:

**Argonne National Laboratory**

**Barber-Nichols, Inc.**

**BWX Technologies**

**Hamilton Sundstrand**

**Idaho National Laboratory**

**Jet Propulsion Laboratory**

**Lawrence Livermore National Laboratory**

**Lockheed Martin Missiles and Fire Control**

**Los Alamos National Laboratory**

**NASA Ames Research Center**

**NASA Glenn Research Center**

**NASA Johnson Space Center**

**NASA Kennedy Space Center**

**NASA Marshall Space Flight Center**

**Northrup Grumman Power Conversion Systems**

**Northrup Grumman Space Technology**

**Oak Ridge National Laboratory**

**Pacific Northwest National Laboratory**

**Pennsylvania State University**

**Sandia National Laboratories**

**The Y-12 National Security Complex**

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## Volume 1: Program Summary



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## ACRONYMS, ABBREVIATIONS, AND SYMBOLS

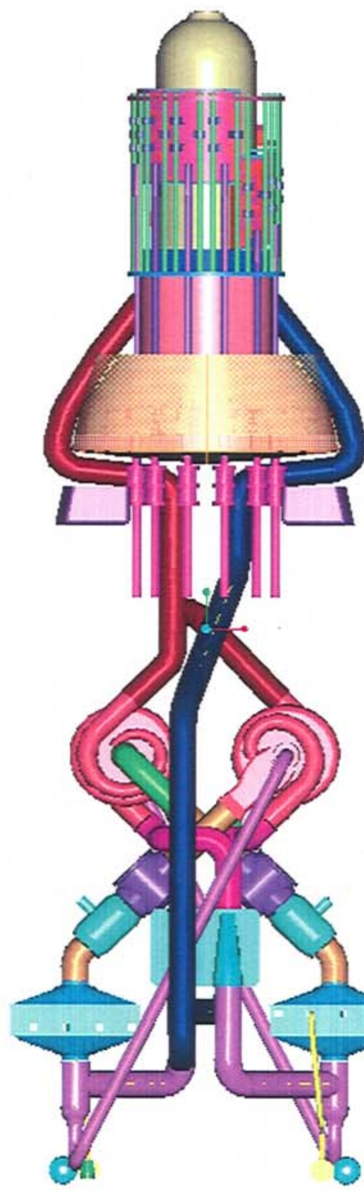
| | |
|---|---|
| AFB | Annular Flow Block |
| AGR | Advanced Gas Reactor |
| AI&T | Assemble, Integrate, and Test |
| AMTEC | Alkali Metal Thermal-to-Electric Conversion |
| ANL | Argonne National Laboratory |
| APP | Annular Plate Pin (AFB variant) |
| ARR | ATLO Readiness Review (NGST Term) |
| ATLO | Assembly, Test and Launch Operations |
| ATR | Advanced Test Reactor |
| $B_4C$ | Boron Carbide |
| Be | Beryllium |
| BeO | Beryllium Oxide |
| Bettis | Bettis Laboratory – Operated by Bechtel Corp. |
| Bi | Bismuth |
| BOL | Beginning of Life |
| BOR-60 | Russian Fast Neutron Spectrum Test Reactor |
| BPMI | Bechtel Plant Machinery, Inc. – Operated by Bechtel Corp. |
| BWXT | BWX Technologies |
| CAM | Control Account Manager |
| CCAFS | Cape Canaveral Air Force Station |
| CDM | Control Drive Mechanisms |
| CEF | Critical Experiment Facility (at Los Alamos National Laboratory) |
| CERMET | Ceramic Fuel in Metal Matrix |
| DAF | Device Assembly Facility |
| DOD | Department of Defense |
| DOE | Department of Energy |
| EBR-11 | Experimental Breeder Reactor |
| ECS | Energy Conversion System |
| EM | Engineering Model: Second-generation hardware |
| EOL | End of Life |
| EOM | End of Mission |
| EP | Electric Propulsion System (ion thrusters) |
| ERD | Environmental Requirements Document |
| FCA | Fast Critical Assembly |
| FFTF | Fast Flux Test Facility |
| FIMA | Fission of Initial Metal Atoms |
| FU | Flight Unit |
| Ga | Gallium |
| GNF | Global Nuclear Fuels |
| GRC | NASA – Glenn Research Center |
| GTR | Ground Test Reactor |
| He | Helium |
| HEU | Highly Enriched Uranium |
| HeXe | Helium Xenon |
| HFIR | High Flux Isotope Reactor |
| HRS | Heat Rejection Segment |
| HTGR | High Temperature Gas Reactor |
| HTTR | High Temperature Test Reactor |
| I&C | Instrumentation and Control |
| INL | Idaho National Laboratory |

| | |
|---|---|
| JAERI | Japan Atomic Energy Research Institute |
| JIMO | Jupiter Icy Moons Orbiter |
| JOYO | Japanese Fast Neutron Spectrum Reactor |
| JNCDI | Japan Nuclear Cycle Development Institute |
| JPL | Jet Propulsion Laboratory – Lead organization for Project Prometheus |
| JSC | NASA – Johnson Space Center |
| K | Potassium |
| KAPL | Knolls Atomic Power Laboratory – Operated by Lockheed-Martin Corp. |
| kPa | Kilo Pascal |
| KSC | NASA – Kennedy Space Center |
| kW | Kilo Watt |
| kWe | Kilo Watt Electric |
| kWt | Kilo Watt Thermal |
| LACEF | Los Alamos Critical Experiments Facility |
| LANL | Los Alamos National Laboratory |
| Li | Lithium |
| LiH | Lithium Hydride |
| M&S | Materials and Subcontracts |
| MeV | Mega electron-volts |
| MHD | Magnetohydrodynamics |
| MOA | Memorandum of Agreement |
| MOU | Memorandum of Understanding |
| MoRe | Molybdenum Rhenium |
| MOU | Memorandum of Understanding |
| MPa | Mega Pascal |
| MSFC | NASA – Marshall Space Flight Center |
| $MW_e$ | Mega Watt Electric |
| $MW_t$ | Mega Watt Thermal |
| $MW_{th}$ | Mega Watt Thermal |
| Na | Sodium |
| NaK | Sodium Potassium |
| Nb | Niobium |
| NEP | Nuclear Electric Propulsion |
| NEPA | National Environmental Policy Act |
| NFS | BWX Technologies, Nuclear Fuel Services |
| NGST | Northrop Grumman Space Technology – The spacecraft contractor |
| Ni | Nickel |
| NNPP | Naval Nuclear Propulsion Program |
| NPD | BWX Technologies, Nuclear Products Division |
| NR | Naval Reactors |
| NRC | Nuclear Regulatory Commission |
| NRPCT | Naval Reactors Prime Contractor Team (Bettis, KAPL and BPMI) |
| NTS | Nevada Test Site |
| OL | Open Lattice |
| OLC | Open Lattice Core |
| ORNL | Oak Ridge National Laboratory |
| OSTI | Office of Scientific and Technology Information |
| OUO | Official Use Only |
| P1 | Prometheus 1 – The first flight unit. |
| Pb | Lead |
| PCAD | Power Conditioning and Distribution Sub-system |
| PHENIX | French Fast Neutron Spectrum Test Reactor |

| | |
|---|---|
| PLR | Parasitic Load Radiator |
| Prometheus-1 | First flight unit |
| Prometheus-2 | Second flight unit |
| PU | Power Unit |
| RAM | Responsibility Assignment Matrix |
| RM | Reactor Module (RPU + ECS) |
| RMIC | Reactor Module Integration (and assembly) Contractor |
| RMIF | Reactor Module Integration Facility |
| RPU | Reactor Power Unit (core, vessel, safety rod, reflectors, shield, CDMs, piping) |
| SC | Spacecraft Contractor |
| SiC | Silicon Carbide |
| SM | Spacecraft Module |
| Sn | Tin |
| SNL | Sandia National Laboratory |
| SNM | Special Nuclear Material |
| SNR | Schenectady Naval Reactors |
| SNAP | 1960s US Space Nuclear Power Project |
| SNPP | Space Nuclear Propulsion Plant (RM + HRS + PCAD) |
| SPP | Space Power Program |
| SRC | Science Resource Connection |
| SRPE | Space Reactor Planning Estimate |
| SSDB | Space Structural Design Basis |
| TA-18 | Test Reactor in Los Alamos Critical Experiment Facility |
| Ta | Tantalum |
| TE | Thermoelectric |
| TI | Thermionics |
| Ti | Titanium |
| TPV | Thermophotovoltaic |
| TTM | Thermal Test Model |
| UC/UC2 | Uranium Carbide |
| UN | Uranium Nitride |
| $UO_2$ | Uranium Dioxide |
| U-SNRI | Unclassified-Space Nuclear Reactor Information |
| UZrC | Uranium Zirconium Carbide |
| V | Vanadium |
| W | Tungsten |
| WBS | Work Breakdown Structure |
| Xe | Xenon |
| ZPPR | Zero Power Physics Reactor |

# Project Prometheus Reactor Module

## Program Summary



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# 1 EXECUTIVE SUMMARY

The Naval Reactors Prime Contractor Team (NRPCT) led the development of a power plant for a civilian nuclear electric propulsion (NEP) system concept as part of the Prometheus Project. This report provides a summary of the facts, technical insights, and programmatic perspectives gained from this two-year program. The Prometheus Project experience has been extensively documented to better position the United States for future space reactor development.

Major technological and engineering challenges exist to develop a system that provides useful electric power from a nuclear fission heat source operating in deep space. General issues include meeting mission requirements in a system that has a mass low enough to launch from earth while assuring public safety and remaining safely shutdown during credible launch accidents. These challenges may be overcome in the future if there is a space mission with a compelling need for nuclear power to drive development. Past experience and notional mission requirements indicate that any useful space reactor system will be unlike past space reactors and existing terrestrial reactors.

The mission requirements of Prometheus were particularly aggressive due to the requirements for long duration, autonomous operation. The power requirements and mass constraints necessitated high operating temperatures that led to a small, fast neutron energy spectrum reactor concept. The operating temperatures within the reactor (1000-1700K) are beyond the range of conventional structural materials. High temperature materials for the reactor fuel elements and other reactor components are not well developed or characterized and have limited manufacturing experience. No existing fuel system was found that has been qualified to provide assurance of meeting the long duration Prometheus mission requirements at the necessary temperatures. In addition, the small size and high fissile uranium fuel loading in the reactor require engineered features to assure safety during a potential transport or launch accident. Delivery of a reliable Space Nuclear Power Plant will require extensive development and testing.

The United States has a limited nuclear testing infrastructure to support developing a space reactor, or other types of fundamentally new nuclear reactors. There are no fast neutron spectrum test reactors in the United States that can perform testing on structural materials, fuel materials, and integrated fuel systems to support their use in a Prometheus reactor. Fast neutron spectrum reactor test facilities exist in Japan, Russia, France, and India. Conducting fuel testing in those facilities is problematic. The United States also has very limited capability to perform nuclear physics experiments on new reactor materials. Nuclear physics tests are necessary to predict the neutronic behavior of the reactor core both to ensure the reactor will operate successfully and to demonstrate that the reactor will remain safely shut down throughout assembly, transport, and launch. In addition, construction of a ground test reactor is desirable for reactor physics qualification, lifetime testing, system integration testing, and control system qualification. Future space reactor development plans must take into consideration the state of the existing U.S. nuclear infrastructure, the challenge of utilizing international infrastructure, and the iterative nature of qualifying nuclear materials and systems.

No single organization has the expertise and facilities to deliver a nuclear fission power system for space; it is necessary to use capabilities on a national scale. The complexity, difficulty, and development risk associated with space reactor power plants demands aggressive up-front integration of all participating engineering and scientific organizations, directed by explicit, mission-specific, functional requirements. Formal lines of authority, individual responsibilities, and organizational interface management should be defined early and be captured within a detailed work breakdown structure. The critical importance and attendant effort and cost required to create and sustain such a foundation must not be underestimated for future projects, and is independent of the concept or technologies employed. This is especially true given the lack of practical experience with operating space reactors (foreign or domestic) and the degree to which scientific investigations, engineering design and risk reduction, manufacturing development, hardware fabrication and testing, mission definition and planning, regulatory processes, and other activities must be executed in a parallel and (in some instances) iterative fashion. In the case of Prometheus, the Jet Propulsion Laboratory, the Spacecraft contractor, the scientific payload community, the NRPCT, and all of the associated subcontractor partners all brought to the project different technical skills, management and administrative practices, vocabularies, policies, procedures, protocols, and information technologies, as well as engineering and scientific practices, capabilities, and experiences. These differences had to be addressed early in the project.

The development of any advanced space reactor plant will be costly and require a sustained commitment over many years. The initial schedule laid out through launch of the Prometheus 1 spaceship was about 10 years. This was an aggressive, success-based schedule that could not accommodate a significant setback in technology or materials performance. Such setbacks were likely because of the developmental nature of space nuclear power technologies. Even with this optimistic approach, the budget for developing and deploying the space reactor plant to a spaceship was estimated to be about 3 billion dollars, which included a potential ground test reactor facility and the first flight unit. Nuclear reactor development work is technically-exacting, time–consuming, and testing-intensive. This necessitates high labor and facility costs starting early in the development process. It would be inappropriate to expect that a rapid and inexpensive reactor development will meet unprecedented performance demands for space nuclear reactors.

The NRPCT selected a gas reactor directly coupled to a Brayton energy conversion system as the best prospect to meet the challenges of delivering a safe, reliable space nuclear power plant on an accelerated schedule. The Naval Reactors Program reviewed many options and concluded that the directly coupled Brayton system provided the best prospect to satisfy NASA requirements with fewer potentially disabling technology and engineering problems than liquid metal cooled reactor concepts traditionally envisioned for space nuclear power. For example, cooling the reactor with an inert gas avoided fundamental and potentially disabling problems of remotely thawing a solid metal coolant in orbit and materials degradation during long-term operation exposed to very high temperature, highly chemically reactive metal coolant.

Mission requirements drive reactor and power plant design features and therefore must be established early. An experienced reactor plant systems design and development organization should be involved early in the mission planning process to ensure that mission planning assumptions are consistent with practical nuclear technologies. While extensibility to a range of missions is a logical goal, it may not be practical to develop a common reactor system that can provide power for Lunar and Mars surface missions as well as for deep space NEP missions. Gas reactor technologies could be used for both surface and NEP missions and many basic technologies (e.g., fuel materials and core concepts) could be similar. Differences in environment, mission duration, required reliability (e.g., to support human habitation) and power requirements could lead to different designs for each specific mission. Regardless of mission, an appropriate focus for near term space reactor technology investment would be in reactor core and plant materials testing for a high temperature reactor system, so that basic materials information can be available to support a future engineering project.

This report provides (1) basic descriptions of the concepts considered by the NRPCT, (2) discussion of the engineering, techonology development, and planning that was in progress to evolve and deliver these concepts into a space nuclear power plant, and (3) references to more detailed documentation of work by the NRPCT and its subcontractors. The list of references in Volume 3 includes documents from past U.S. space nuclear reactor development projects that the NRPCT found useful for Project Prometheus. This report provides a practical summary of space nuclear power technology within the context of the specific mission challenges for Prometheus, to aid future efforts to develop nuclear reactors for operations in space.

# 2 INTRODUCTION AND BACKGROUND

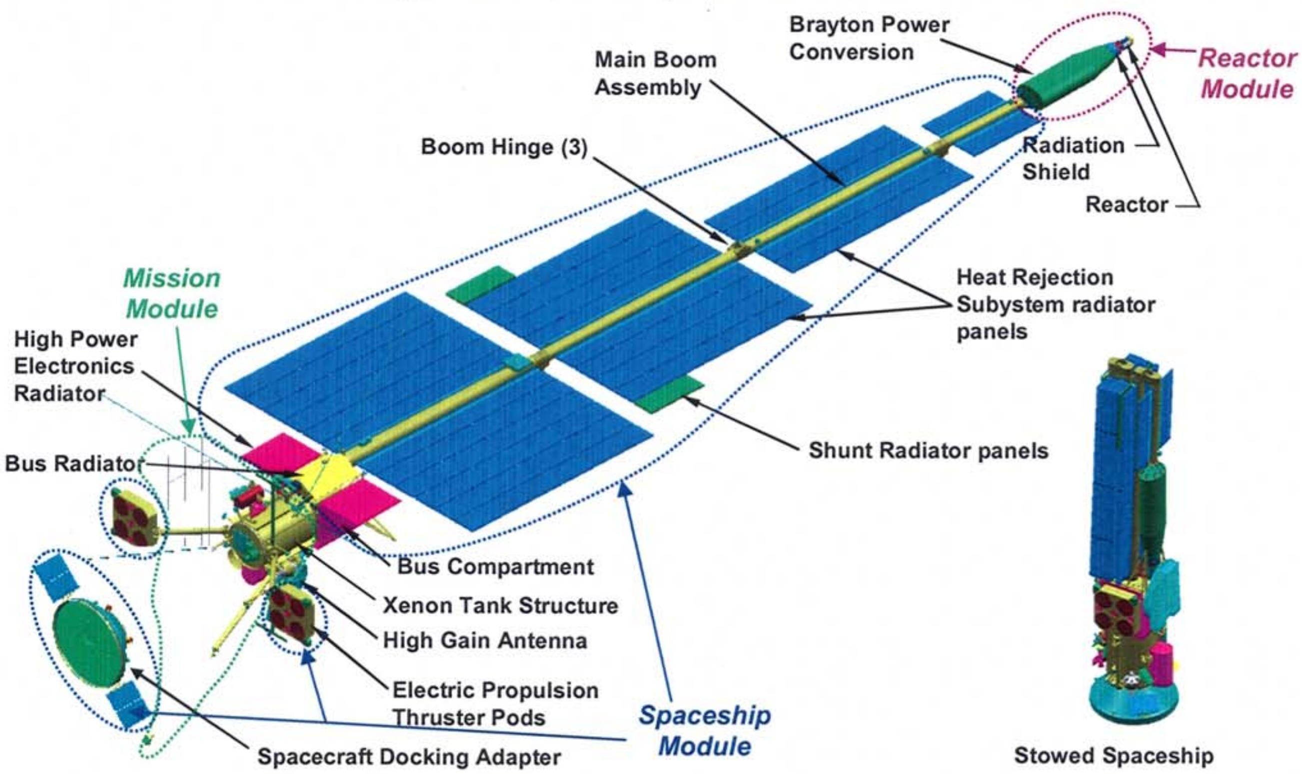
This report summarizes the work completed by the Naval Reactors Prime Contractor Team (NRPCT) as part of the National Aeronautics and Space Administration (NASA) Prometheus Project. Project Prometheus was established in 2003 with a goal of developing the first nuclear reactor-powered propulsion system for a spaceship and demonstrating that it could be operated safely and reliably for deep-space, long duration missions. The initial application of space fission power evaluated was the Jupiter Icy Moons Orbiter (JIMO).

Naval Reactors (NR) involvement in Project Prometheus began when the Secretary of Energy assigned NR the lead for development of nuclear power for civilian space exploration missions, starting with JIMO. NASA defined the Project Prometheus requirements to be a Space Nuclear Power Plant (SNPP) useful for a range of deep space exploration missions with technology extensible to exploration on the surface of the Moon or Mars.

Naval Reactors worked with NASA to establish a Memorandum of Understanding (MOU) at the agency level and a Memorandum of Agreement (MOA) at the working level. These documents formed the basis for a collaborative partnership that recognized individual NASA and NR responsibilities derived from their respective governing statutes. Specific guidelines were provided for managing SNPP development, production, and operational support activities and integrating those activities with NASA's development of the Space, Launch, and Ground Systems.

Naval Reactors and the NRPCT created the Space Power Program (SPP) to deliver the JIMO Reactor Module. The Reactor Module, the Spaceship Module and the Mission Module comprise the spaceship. Figure 2-1 shows a pre-conceptual view of the spaceship and the associated modules.

**Figure 2-1: Spaceship and Modules**

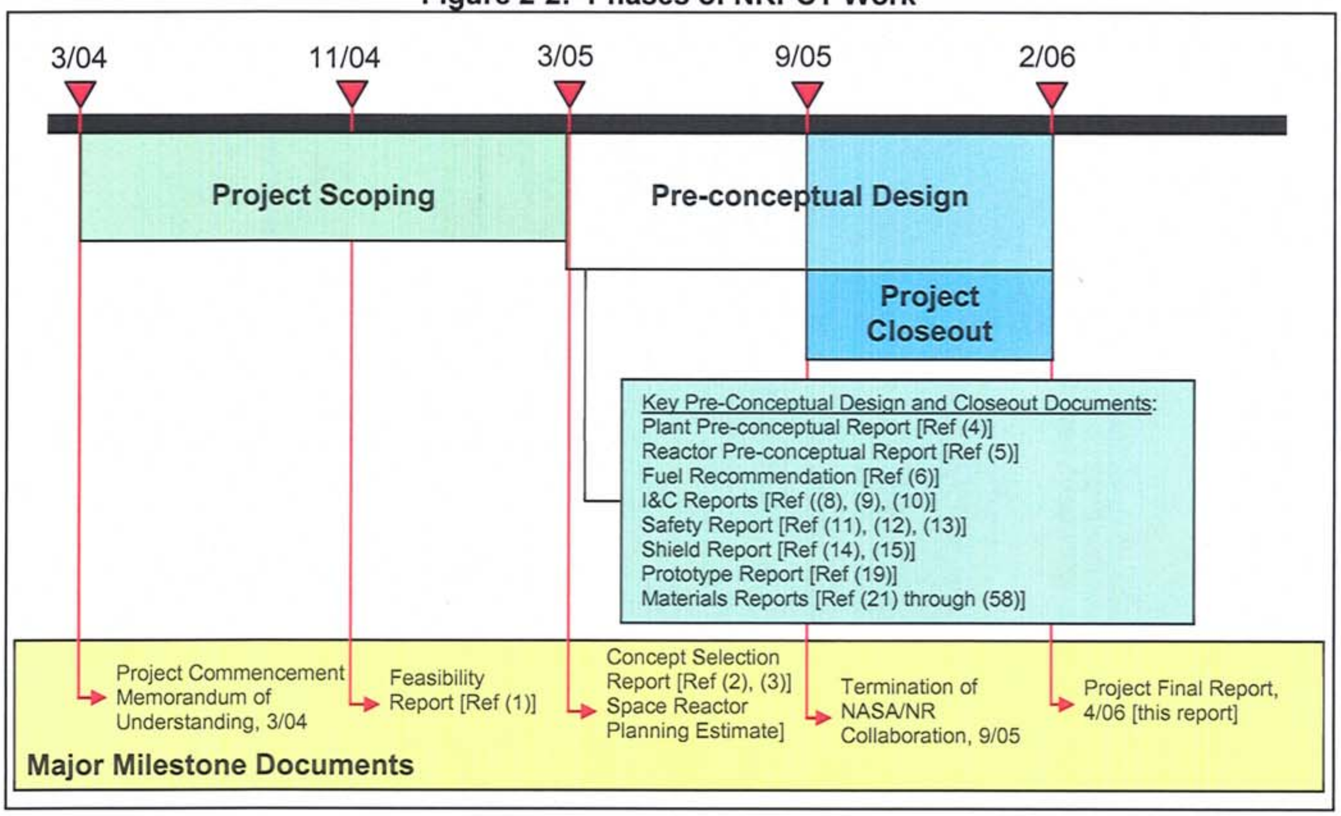


The NRPCT was made up of engineers and scientists from the NR Program prime contractors: Lockheed Martin – KAPL, Bechtel – Bettis Laboratory, and Bechtel Plant Machinery, Inc. (BPMI). The NRPCT was assigned design responsibility for the Reactor Module (with the exception of the Aeroshell reentry protection cover), while NASA and its contractors were responsible for the remainder of the spaceship, launch vehicle, and ground systems.

The NASA Jet Propulsion Laboratory (JPL) was assigned lead responsibility for the Prometheus Project, with support from the Glenn Research Center (GRC), Marshall Space Flight Center (MSFC), Kennedy Space Center (KSC), Johnson Space Center (JSC), and Ames Space Center (ASC). In November 2004, Northrop Grumman Space Technology (NGST) and its industry partners won the contract to develop the JIMO spaceship. The NRPCT drew upon existing expertise to the maximum extent practical, working with other Department of Energy (DOE) organizations, including Los Alamos National Laboratory (LANL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratories (SNL), Idaho National Laboratory (INL), Pacific Northwest National Laboratory (PNNL), Lawrence Livermore National Laboratory (LLNL), Argonne National Laboratory (ANL), and Y-12. Work on SNPP development continued through a feasibility evaluation, coolant and energy conversion selection, and pre-conceptual design work. Based on reprioritization of missions and funding within NASA, Naval Reactors and NASA mutually agreed to end this collaboration in September 2005. An orderly closeout of SPP work proceeded through early 2006.

Figure 2-2 illustrates the nature of the work accomplished by the NRPCT. The project had three major phases of work: Project Scoping, Pre-conceptual Design, and Project Close-out.

**Figure 2-2: Phases of NRPCT Work**




During the Project Scoping phase, two major technical reports were produced. The first report, a concept feasibility evaluation, was issued in November 2004 within the NR program [an unclassified version was issued in April 2006, Reference (1)]. This report reviewed the developmental challenges of a space reactor system, including the reactor design, fuel and materials performance, shielding design, primary coolant transport and compatibility, energy conversion and heat rejection options, and operational concerns. In this report, the NRPCT determined that all reactor design approaches for supporting the JIMO mission requirements needed considerable technological and engineering development and that to support the 2015 launch date, development would need to be minimized by selecting more well-developed technologies. From this evaluation, fuel and in-core materials as well as integrated plant design and testing emerged as the biggest challenges.

The second technical report was the concept selection report. This report was issued in April 2005 [an unclassified version was issued in July 2005, Reference (2)] and recommended focusing on a direct gas-Brayton concept. This report documented the formal process used to evaluate the candidate integrated reactor plant concepts and select a specific basic concept for further development. The feasibility assessment results were used to select five concepts for evaluation at this stage:

1) A direct cycle, gas-cooled reactor with a Brayton energy conversion system
2) A heat pipe-cooled reactor with a Brayton energy conversion system
3) A liquid lithium-cooled reactor with a Brayton energy conversion system
4) A liquid lithium-cooled reactor with a thermoelectric energy conversion system
5) A lower temperature, liquid metal (e.g., sodium, potassium, or NaK) cooled reactor with a Stirling energy conversion system

Each system was evaluated against criteria relevant to JIMO mission requirements, including capability, reliability, deliverability, cost, and safety. The direct gas Brayton concept (concept #1 listed above) was recommended to Naval Reactors for further development. Naval Reactors approved the recommendation via Reference (3). The gas reactor system was judged to have the best prospect to meet mission requirements within the expected development time frame. Use of an inert gas coolant simplifies engineering development testing. In addition, Brayton technology is judged to be relatively mature with an existing engineering and manufacturing base, and a Brayton system has fewer components requiring development relative to the other concepts considered. Finally, a gas Brayton system is extensible to surface missions.

The Pre-conceptual Design phase began with approval of the Direct Gas Brayton concept. The major goals of this phase were to select the nuclear fuel and fuel element clad system materials, select plant parameters to set the baseline heat balance, and select the baseline arrangement. This work was well underway when the Prometheus Project was restructured in September 2005. In this restructuring, NRPCT involvement in Prometheus through the Space Power Program was terminated. NRPCT documented the results of progress on the Pre-conceptual Design phase in a number of detailed technical reports.

This integrated project summary report provides important information and key perspectives on all space reactor work conducted under NR cognizance. It describes the formal programmatic (Volume 1) and technical (Volume 2) information that was generated, how and where it is stored, and how future researchers can access the information. A detailed bibliography (Volume 3) is included to serve as a roadmap that identifies both external references that were found to be useful by the NRPCT, as well as documents that were generated by the NRPCT. This report and its references document research, development, and engineering work performed by the NRPCT and its subcontractors at the time that NASA and NR mutually agreed to end their partnership. Most of the technical work had not progressed to the point of a recommendation to Naval Reactors to proceed forward with a specific conceptual plant design.

# 3 PROJECT SCOPE

## 3.1 Prometheus Project Goals

The goal of the Prometheus Project was to develop a spaceship for use in robotic exploration missions of the outer solar system that would combine an SNPP with electric propulsion (referred-to collectively as Nuclear Electric Propulsion (NEP)). By providing unprecedented amounts of on-board electrical power and energy, the SNPP was expected to increase spacecraft maneuverability, enable high-capability instruments, provide expanded mission design options (including successive orbits of solar system bodies), and support high-speed telecommunications to return scientific data from deep space).

Significant and challenging technology advances are required to deliver such a space nuclear power plant. Key developmental areas include the reactor, energy conversion, heat rejection, and radiation-hardened components. Additionally, the SNPP technologies (in the areas of nuclear fuel, reactor core materials and coolants, instrumentation and control, and energy conversion) were to be extensible to Lunar and Mars surface power missions.

The initial objective of the Prometheus Project would be a scientific exploration mission to the larger icy moons of Jupiter (Callisto, Ganymede, and Europa). This JIMO mission was targeted to launch in 2015 and initiate its first science orbit in 2021. Additional missions for this type of spaceship that were envisioned by NASA include:

- Saturn and its moons
- Neptune and its moons
- Comet and multi-asteroid sample return
- Kuiper Belt rendezvous

The duration of these missions ranged from 10 to 12 years for JIMO, to a maximum of 20 years for the Kuiper Belt rendezvous. These notional missions were envisioned to use the same SNPP as the JIMO mission, requiring approximately 200 kW of electric power.

## 3.2 Comparison to Past Space Reactor Projects

While there have been past studies investigating the possibility of multi-megawatt space reactors for long duration missions, the reactors that have actually been built and tested were much more modest in capability and lifetime. For example, they were designed to operate in Earth orbit and were not intended to use electric power for propulsion to escape orbit. The United States has flown just one space reactor, the SNAP-10A FS-4 reactor, in 1965. This spacecraft produced approximately 550 Watts of electric power for 43 days prior to an electrical system fault which caused a reactor shutdown. Its ground test reactor, however, operated continuously for over a year.

The Soviet Union flew 32 Bouk-RORSAT systems and 2 TOPAZ-I reactor systems over the timeframe of the early 1970s to the late 1980s. These reactors had power levels in the range of 1 to 5 kWe and operational lifetimes of one month to ~1 year. The TOPAZ-II reactor system was ground tested but never flown in space.

During the 1960s, the United States also began development of the larger scale SNAP-8 and SNAP-50 reactor systems, which were designed for higher power (up to 100 kWe for SNAP-8 and as much as 1200 kWe for SNAP-50) and longer life (up to approximately three years). Testing was done on the SNAP-8 reactor; however, these reactors were never developed through launch. During the

1980s, the United States investigated the SP-100 space reactor concept, building upon the earlier experimental work from the SNAP-50 program. The SP-100 project had a goal of producing 100 kWe with an operating lifetime of 7 years. This program was terminated before a reactor or many prototypic reactor system components were developed or tested.

The Prometheus Project was more ambitious than past space reactors due to the higher electric power level (~200 kWe), the longer calendar lifetime (up to 20 years), the greater need for system autonomy to support propulsion and science operations in deep space, and NASA's desire to maximize extensibility of the associated technologies to many types of missions. Autonomy of the reactor control system is needed due to the long delay time in communications between Earth and Jupiter and the long periods of communications blackout. The low solar incidence at Jupiter also eliminates the ability to use solar power as a back-up power supply. Thus, continuity of reactor power is essential for mission success.

The difficulties posed by increasing the required reactor lifetime are most significant to the space reactor design challenge. The necessary material performance at the high operating temperatures of the reactor system is unprecedented and requires extensive development and testing to deliver a system that would have the desired mission assurance.

## 3.3 Project Challenges

The key programmatic, engineering, and scientific challenges of the Prometheus Project stem from delivering a Space Nuclear Power Plant that:

- Relies on many as-yet undeveloped technologies
- Is required to last for up to 20 years of operation
- Will be launched in 2015

The two controlling path schedule activities that posed the greatest challenge toward meeting the targeted launch date are (1) developing Reactor Module materials that will successfully meet the JIMO mission duration and (2) establishing critical nuclear testing facilities to support the design timeline.

Materials Development Timeline

The materials within a space nuclear reactor will be most affected by radiation, temperature, and time. The specific external environments in which the reactor is expected to operate will also pose special challenges (e.g., vacuum and chemical constituent effects on high temperature reactor structural material integrity). Long-term materials performance for the reactor core and associated structures is largely dependent upon empirical testing. The mass and volume constraints on the SNPP drive the reactor design to have a higher operating temperature and higher uranium loading density than typical terrestrial reactors. Existing performance data on candidate core and structural materials are insufficient to provide a basis for reactor design, so materials irradiation tests are necessary. Since the reactor design would use a fast neutron energy spectrum, some irradiation testing on candidate structural materials must be performed in fast neutron spectrum test reactors to get a relevant understanding of materials behavior. To design, irradiate, and examine material test specimens on an expedited basis takes approximately two to three years per cycle. Once an irradiation test cycle is done, specimens are examined and improvements are generally identified, fabricated, and placed into the next two-to-three year irradiation test cycle. Because the JIMO Reactor Module design began in 2005 for a launch in 2015, only two material design and test cycles could be done before the flight reactor must be manufactured. Because the materials are developmental, there is a significant delivery risk if unresolvable test failures occur. Confidence in the lifetime of the Reactor Module would increase as additional prototypical tests are performed and as the actual 10- to 12-year JIMO mission is underway. As succeedingly longer missions are launched and proceed satisfactorily, confidence toward meeting the maximum 20-year lifetime requirement would increase.

Meeting the aggressive irradiation testing schedule was complicated by the lack of fast spectrum neutron flux reactors in the United States. While the Experimental Breeder Reactor-II (EBR-II) and the Fast Flux Test Facility (FFTF) test reactors were available in the U.S. to provide data for the SP-100 Program, these have since been decommissioned. NRPCT had planned to use the JOYO facility in Japan for fuel and structural material testing, but using the international infrastructure for nuclear development work is a challenge for an aggressive schedule. Some test work was also planned for the Advanced Test Reactor (ATR) at INL and an initial test was conducted in the High Flux Isotope Reactor (HFIR) at ORNL, but these thermal neutron spectrum reactors do not provide a suitable fast neutron flux spectrum that adequately simulates fast spectrum space reactor operating conditions.

<u>Critical Testing Facilities</u>

Three phases of critical nuclear experiments were being evaluated to support the nuclear design of a space reactor and to qualify the reactor design methods:

1. Benchmark Tests – Basic benchmark critical assemblies and fundamental cross section measurements to improve cross section accuracy.
2. Mock-up Tests – Experiments to qualify the nuclear design of the Prometheus reactor as well as the design methods.
3. Ground Test Reactor (GTR) – Prototypical reactor and power plant used for a comprehensive physics test program. A full range of tests, including cold and hot low power tests, high power tests, and transient tests were being considered for beginning-of-life and periodically through life to fully qualify both calculational models and startup and operating procedures, and to confirm lifetime trends of physics characteristics.

While numerous test reactors and reconfigurable critical facilities existed at the time of SNAP and some remained during SP-100, many of these facilities have been decommissioned. The one remaining facility suitable for material nuclear cross-section benchmarks, the Los Alamos Critical Experiment Facility (TA-18), was shut down in July 2004. At the time that Naval Reactors program involvement in Prometheus ended, no suitable domestic capability was available to perform benchmark critical experiments of space reactor materials. The lack of such data early in the design process carries a risk of a fundamental design data error that would delay delivery of the SNPP.

Similar challenges exist for mock-up tests. The Zero Power Physics Reactor (ZPPR) at INL was previously used to assemble engineering mockups for fast reactors. The ZPPR facility was placed in a cold standby status in the early 1990s, and the projected cost to restore the facility to operation was high (~$58M). Other options, such as to design and build a new facility at a DOE or NR Program site, exist but the cost and schedule impacts are not known. The only facility currently available for this work is the Fast Critical Assembly (FCA) operated by the Japan Atomic Energy Research Institute. The complexities of international operations posed additional risk to the aggressive schedule associated with Project Prometheus.

A significant programmatic challenge to meet the 2015 launch date was the siting and construction of a potential prototype GTR Facility. As documented in Reference (19), GTR operation would provide valuable support to operating a SNPP, particularly for future evolution to a manned mission. It was desirable for the prototype to begin operation at least a year before launch to provide early core operational and performance data. Preliminary NRPCT scoping indicated that it would be challenging to site and construct a prototype GTR Facility in time to support 2014 prototype operation before a 2015 launch. An environmental impact statement under the National Environmental Policy Act (NEPA) was planned to be completed for the prototype GTR.

# 4 PROJECT MANAGEMENT APPROACH

## 4.1 Planning Documents

Using the NASA/NR Memorandum of Understanding and Memorandum of Agreement as a basis, the NRPCT further elaborated the scope of the SNPP work effort in concert with JPL by defining a Work Breakdown Structure (WBS). A Responsibility Assignment Matrix (RAM) was developed to define responsibilities between the NRPCT and NASA entities. A Space Reactor Planning Estimate (SRPE) was developed to estimate resource needs.

The baseline project schedule for the JIMO mission specified a launch in 2015. Based on NRPCT's technical review of the steps for Prometheus space reactor development, a prototype GTR was under consideration. As discussed in Section 3.3, it would be difficult to complete prototype GTR siting soon enough to operate before a 2015 launch. In addition, the time available to develop and deliver reactor core materials allowed few development and irradiation test cycles, so the 2015 launch date for a 12-year mission was considered to be optimistic and success-based. Separately, JPL performed an analysis of alternative mission architectures that raised the potential for an earlier robotic mission to demonstrate NEP technology with more modest mission requirements. Based on this, the NRPCT worked with JPL on an alternative plan that would include a demonstration unit launch in 2014 and the JIMO spaceship launch in 2017. The NRPCT developed its SRPE and project plans with a contingency to enable an earlier demonstration launch as well as the JIMO mission.

### *4.1.1 Responsibility Assignment Matrix (RAM)*

The NRPCT was responsible for design and delivery of the Reactor Module (with the exception of the Aeroshell reentry protection cover) and was involved with any aspect of the spaceship that could affect the design or operation of the nuclear reactor. The division of responsibility among the NRPCT and the various NASA entities involved in Project Prometheus was detailed in the RAM. With the selection of the direct gas Brayton concept, the NRPCT was assigned design agent and approval responsibility for the power (or energy) conversion system because this concept requires the gas, which is the primary coolant, to flow directly from the reactor through Brayton loops and back to the reactor. This was consistent with the MOU, which defined the Space Reactor as "the collection of hardware consisting of a space nuclear fission reactor, its instrumentation and control system, reactor shielding, and those components in direct contact with reactor coolant, for which NR possesses both legal ownership and technical responsibilities."

There were a number of important system level considerations relative to NRPCT responsibility for power conversion. The direct coupling of the reactor to the power conversion system required a number of critical performance related issues to be managed by the NRPCT within the overall SNPP design process, such as reactor dynamics, material compatibility, reliability and redundancy trade studies, as well as an integrated system optimization (e.g., gas coolant composition, system pressure). In addition, NRPCT responsibility for the power conversion system also enabled the strategy to integrate, assemble, and test the power plant as a module and provide for nuclear material safeguards, reactor safety, mission assurance, and indemnification of contractors under the Price-Anderson Act.

The RAM denotes which entity is responsible to establish requirements, who is lead for design and design approval, and who provides design concurrence. The RAM served as the basis for developing the detailed division of responsibilities and planning for associated contracts with organizations outside the NRPCT.

### _4.1.2 SNPP Work Breakdown Structure_

Further definition of the scope for design, delivery, and operational support of the Project Prometheus SNPP, and roles and responsibilities within the NRPCT, were provided in the SNPP WBS. Figure 4-1 illustrates the SNPP WBS and its relationship to the NRPCT budget planning system and the NR organization codes. The SNPP WBS evolved to reflect the architecture and requirements flow for the SNPP and the JIMO spaceship and to align with the RAM discussed above. The SNPP WBS melded the NASA-JPL WBS construct of broadly defined systems (e.g., the Space System), hardware-centric modules, segments, and sub-systems with the Naval Reactors Program traditional nuclear power plant WBS and product structure that mainstreams system engineering, safety, and quality assurance elements within each area of the plant (e.g., Instrumentation and Control (I&C), Fluid Systems, Reactor, etc.).

**Figure 4-1: SNPP Work Breakdown Structure and NRPCT Budgeting System**

PROJECT PROMETHEUS WORK BREAKDOWN STRUCTURE AND MAPPING TO NR BUDGETING SYSTEM

| WBS | Fund Source (NASA) | Program Designator | Lead NR Technical Code | Supporting NR Technical Code | Sub-task Number | Deliverable Number | Title |
|---|---|---|---|---|---|---|---|
| 3.2.1 | | | C | | | | Project Management & System Engineering |
| 3.2.1.1 | D23 | 4 | C | C | 1 | 1 | Project Scope, Sched, Cost & Risk Mgmt |
| 3.2.1.2 | D23 | 4 | C | C | 2 | 5 | Program Mgmt |
| 3.2.1.3 | D23 | 4 | C | C | 3 | O | BPMI Labor |
| 3.2.1.4 | D23 | 4 | C | H | 1 | T | Facility Support |
| 3.2.1.5 | D23 | 4 | C | H | 2 | 1 | Financial & Planning Support |
| 3.2.1.6 | D23 | 4 | C | H | 3 | 1 | Supply Chain Mgmt Support |
| 3.2.1.7 | D23 | 4 | C | H | 4 | 1 | Human Resources Support |
| 3.2.1.8 | D23 | 4 | C | R | 1 | 1 | Radiological Controls Support |
| 3.2.1.9 | D23 | 4 | C | S | 1 | 2 | Quality Assurance Support |
| 3.2.1.10 | D23 | 4 | C | U | 1 | 1 | Regulated Materials Support |
| 3.2.1.11 | D23 | 4 | C | U | 2 | C | NEPA & Regulatory Support |
| 3.2.1.12 | D23 | 4 | C | V | 1 | 5 | Information Protection Support |
| 3.2.1.13 | D23 | 4 | C | V | 2 | A | Safeguards & Physical Security Support |
| 3.2.1.14 | D23 | 4 | C | V | 3 | A | Computer Security Support |
| 3.2.1.15 | D23 | 4 | C | V | 4 | 9 | Public Outreach |
| 3.2.1.16 | D23 | 4 | C | Y | 1 | 7 | Information Systems Support |
| 3.2.2 | | | E | | | | Integration Assembly, Test |
| 3.2.2.1 | D23 | 4 | E | E | 1 | B | Integ, Asm & Test Prometheus SNPPs |
| 3.2.2.2 | D23 | 4 | E | E | 2 | C | Integ Prometheus SNPPs with the Spacecraft |
| 3.2.2.3 | D23 | 4 | E | E | 3 | D | Integ & Asm Grnd Reactor Test Reactor Mod |
| 3.2.2.4 | D23 | 4 | E | E | 4 | E | Integ, Asm & Test the Non-Nuclear Test Units |
| 3.2.3 | | | Z | | | | Safety and Assurance |
| 3.2.3.1 | D23 | 4 | Z | Z | 1 | 1 | Ground Test Reactor Safety |
| 3.2.3.2 | D23 | 4 | Z | Z | 2 | 2 | Flight Reactor Safety |
| 3.2.3.3 | D23 | 4 | Z | S | 3 | 2 | Assurance |
| 3.2.4 | | | S | | | | Testing and Evaluation of Reactor Module Materials |
| 3.2.4.1 | D23 | 4 | S | S | 1 | A | Materials Performance Evaluation |
| 3.2.4.2 | D23 | 4 | S | S | 2 | 3 | Qual Matls & Dev Specs & Design Guidance |
| 3.2.4.2 | DRR | 4 | S | S | 2 | 3 | Qual Matls & Dev Specs & Design Guidance (Drafting) |
| 3.2.4.3 | D23 | 4 | S | S | 3 | 8 | Support Matls Testing Work |
| 3.2.4.4 | D23 | 4 | S | S | 4 | B | Fuel Methods |
| 3.2.5 | | | F | | | | Test Facilites |
| 3.2.5.1 | D23 | 4 | F | F | 1 | V | Provide Prototype Facility |
| 3.2.5.2 | D23 | 4 | F | F | 2 | 6 | Prototype Ops & Testing |
| 3.2.5.3 | D23 | 4 | F | R | 3 | Z | Provide Facility Shielding |
| 3.2.5.4 | D23 | 4 | F | V | 4 | 3 | Perform Site Selection & Establish MOAs |
| 3.2.6 | | | G | | | | Support Equipment |
| 3.2.6.1 | D23 | 4 | G | G | 1 | O | Deliver Shipping & Handling Equip & Proced |
| 3.2.6.2 | D23 | 4 | G | G | 2 | G | Deliver Prototype Disposal Canister |

| WBS | Fund Source (NASA) | Program Designator | Lead NR Technical Code | Supporting NR Technical Code | Sub-task Number | Deliverable Number | Title |
|---|---|---|---|---|---|---|---|
| 3.2.7 | | | I | | | | Reactor Segment |
| 3.2.7.1 | D23 | 4 | I | I | 1 | 4 | Reactor System Design (Nuclear) |
| 3.2.7.1 | D23 | 4 | I | I | 1 | 5 | Reactor System Design (T/H and Mechanical) |
| 3.2.7.1 | DRR | 4 | I | I | 1 | 5 | Reactor System Design (Drafting) |
| 3.2.7.1 | D23 | 4 | I | I | 1 | 9 | Reactor System Design (Rx Sys Validation) |
| 3.2.7.2 | D23 | 4 | I | I | 2 | C | Reactor Comp Dev, Design, Proc, & Test |
| 3.2.7.3 | | | NOT USED | | | | |
| 3.2.7.4 | D23 | 4 | I | I | 4 | 9 | Provide Fuel |
| 3.2.7.5 | D23 | 4 | I | I | 5 | J | Manufacture Core |
| 3.2.7.6 | D23 | 4 | I | A | 6 | M | Develop Reactor Methods |
| 3.2.8 | | | E | | | | Plant Segment |
| 3.2.8.1 | D23 | 4 | E | E | 5 | 7 | Plant Design & Integration |
| 3.2.8.1 | DRR | 4 | E | E | 5 | 7 | Plant Design & Integration (Drafting) |
| 3.2.8.2 | D23 | 4 | E | T | 1 | 5 | Component Development & Test |
| 3.2.8.3 | D23 | 4 | E | E | 7 | 2 | System Testing |
| 3.2.8.4 | D23 | 4 | E | E | 8 | C | Modeling Development |
| 3.2.8.5 | D23 | 4 | E | E | 9 | E | Ground Test Reactor Plant |
| 3.2.8.6 | D23 | 4 | E | I | 1 | 9 | Modeling Support |
| 3.2.9 | | | R | | | | Radiation Shield Segment |
| 3.2.9.1 | D23 | 4 | R | R | 1 | 3 | Radiation Shield Design |
| 3.2.9.2 | D23 | 4 | R | R | 2 | L | Struct & Therm Des & Fab of Radiation Shield |
| 3.2.10 | | | K | | | | Instrumentation & Control Segment |
| 3.2.10.1 | D23 | 4 | K | K | 1 | L | Design Inst & Control System |
| 3.2.10.1 | DRR | 4 | K | K | 1 | L | Design Inst & Control System (Drafting) |
| 3.2.10.2 | D23 | 4 | K | K | 2 | O | Dev, Des, Procure & Test Components |
| 3.2.10.3 | D23 | 4 | K | K | 3 | R | Deliver Software |
| 3.2.10.4 | D23 | 4 | K | K | 4 | S | Test Composite System |
| 3.2.10.5 | D23 | 4 | K | K | 5 | 2 | Develop Composite Test Devices |
| 3.2.10.6 | D23 | 4 | K | K | 6 | 6 | Ground Nuclear Prototype Support |
| 3.2.10.7 | D23 | 4 | K | K | 7 | 8 | Support EM Test |
| 3.2.10.8 | D23 | 4 | K | K | 8 | 9 | Support QM Test & ATLO |
| 3.5 | INCLUDED IN WBS #3.2.2.2 During Concept Development | | | | | | |
| 4.6 | | | E | | | | Launch Site Nuclear Operations |
| 4.6.1.1 | D23 | 4 | E | E | B | F | Launch Site Nuc Ops: Plan, Des, Bld Facility |
| 4.6.2.1 | D23 | 4 | E | E | C | -- | Final Integ & Test of Fueled Prometheus SNPP |
| 4.6.3 | INCLUDED IN WBS #3.2.1.10 DURING CONCEPT DEVELOPMENT | | | | | | |
| 4.6.4.1 | D23 | 4 | E | E | D | L | Support Launch Site Ops |

The NRPCT developed the WBS to highlight the importance of key elements of the project:

- The Project Management and System Engineering section of the SNPP WBS (3.2.1) fully implemented areas within the Project Management Body of Knowledge for work such as Human Resource Management, Communications Management, Quality Management, Procurement Management, etc., to ensure all the work for start-up of this unique venture was properly captured.
- To manage external interfaces, the SNPP WBS contained elements for Integration, Assembly, and Test (IA&T) of the SNPP segments (3.2.2), IA&T of the SNPP with the spaceship (3.5), and Launch Operations and Facilities (4.6).
- A portion of Safety and Mission Assurance work was included as a separate WBS element to match up with corresponding NASA-JPL activities. The NRPCT judged that with the intensity

and visibility of the Presidential Launch approval process and the challenge of supporting the requirements of the National Environmental Policy Act, a high level WBS element to ensure visibility and accountability for these efforts was prudent. Despite this, principal responsibility for safety and quality remained with the personnel responsible to design and deliver the various segments of the SNPP and did not rely on a separate quality assurance organization.

- The WBS element for the Prototype (3.2.5) brought focus to the unique facilities and regulatory approvals that would have been required to prototype a first-of-a-kind space reactor.

The second tier of the SNPP WBS was generally defined as a Control Account, and one NRPCT Manager was established as the Control Account Manager (CAM) accountable to the NRPCT Project Manager for managing scope, cost, schedule, and quality for that WBS element. For several complex elements, such as Core Manufacturing, a sub-Control Account was established to ensure focused expertise and leadership was provided to support the CAM for the higher tier WBS element. Co-CAMs were used for Materials Development, which was viewed as a "project within the project," with responsibility split between KAPL and Bettis managers at lower tiers of the WBS.

The SNPP WBS enabled a fully "projectized" organization with the primary performing organizations within the SPP (reactor engineering, primary plant, power conversion, materials, and the project office) matched directly to top tier WBS elements. Matrix-supporting organizations (e.g., servicing, shielding, reactor safety) also matched up directly to the SNPP WBS. The WBS also aligned fairly directly to the technically cognizant NR Headquarters Sections, and a lead NR Section was identified for each control account.

#### *4.1.3 Space Reactor Planning Estimate (SRPE)*

The SRPE contained estimates for Materials and Subcontracts (M&S) and Labor costs associated with the design and delivery of the Reactor Module, in accordance with the project milestone schedule provided in Figure 4-2. The SRPE was issued to: (1) communicate resource needs for the duration of the project, (2), seek approval for the RAM (discussed in Section 4.1.1), and (3) seek approval of the SRPE WBS as well as plans for Earned Value Management.

The SRPE was issued in April 2005 after approximately one year of NRPCT involvement in the Space Power Program. It was generated following the formation of collaborative relationships with NASA and DOE Laboratories and was based on an investigation of Space Program history with respect to nuclear power development, a deeper understanding of vehicle and mission requirements, and a comparison of the Prometheus Project with past Naval Nuclear Propulsion Program (NNPP) projects. The primary methods used by the NRPCT for the SRPE were:

- Primarily, analogous estimates were made based on the collective NR experience within the NRPCT as applied to specific activities like reactor or fluid systems design.
- Adjustments were made given higher complexity of Prometheus work items, levels of testing, inexperience in SNPP design, and smaller scale.

The NRPCT concluded that the initial SRPE was adequate for planning purposes, but was a work in progress that would have been supplemented in the near future by:

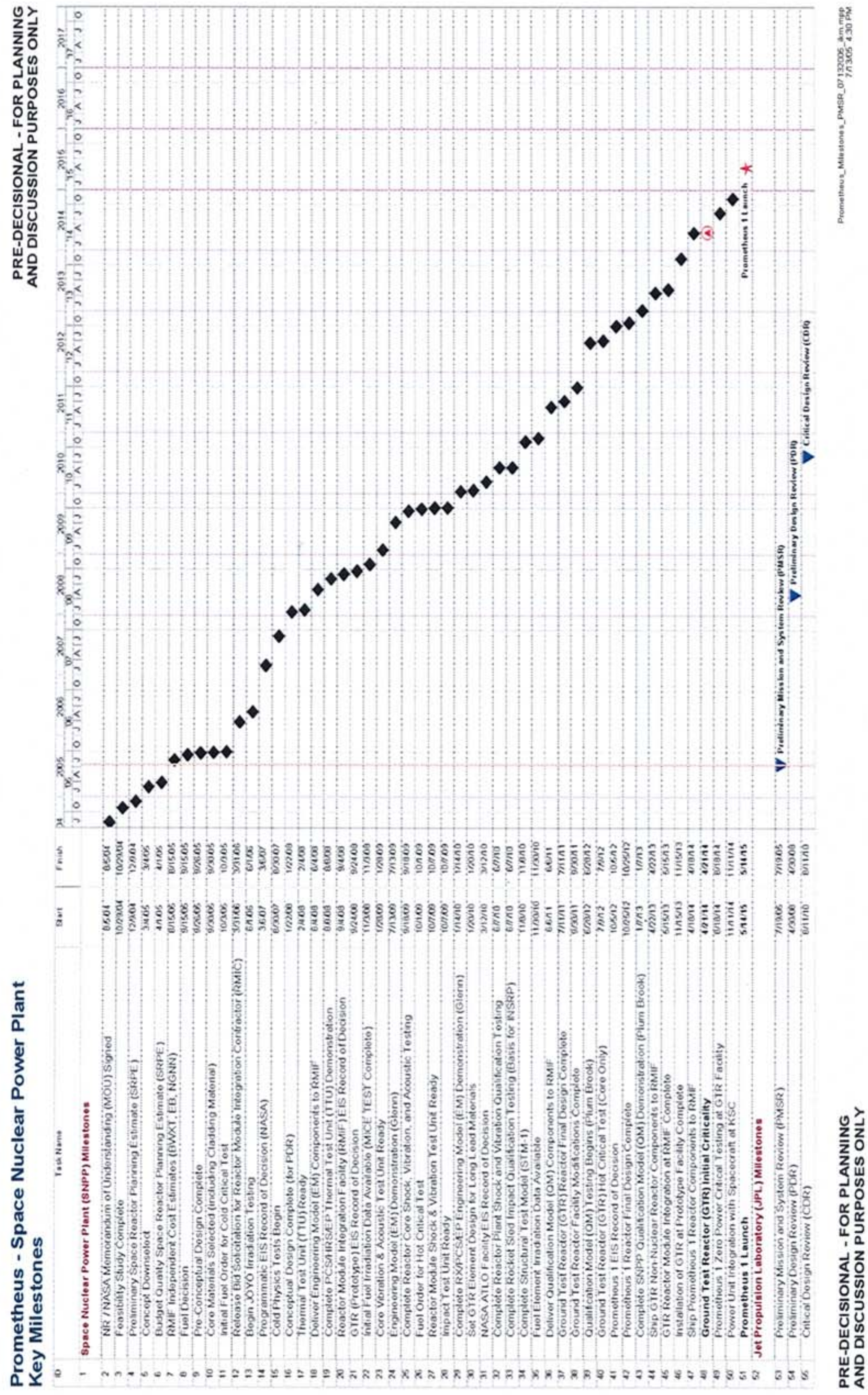
- Vendor inputs for core manufacturing.
- Input from several vendors on cost and effort for Reactor Module integration, assembly, & test.
- Independent Cost Estimates from DOE National Laboratories.
- Reconciliation of scope, resources and schedule with JPL and NGST based on the NR approved SNPP concept and implementation of the updated RAM for power conversion.
- Adjustment of near-term work plans to accommodate expected funding levels for FY07.
- Refinement of Management Reserve and Contingency funding to be included in the estimate.

The initial SRPE represented a snapshot in time. The SRPE was evolving and had not been approved by Naval Reactors at the time the project was terminated. Furthermore, NASA was reviewing the scope and extent of the first planned mission within a new mission architecture, which was expected to affect key planning assumptions. Many of the planning assumptions were being re-evaluated at the time of project termination, as discussed below.

- **Evaluation of a proposed demonstration mission.** The SRPE assumed a preliminary flight test spaceship for launch in 2014 as well as a JIMO mission for launch in late 2017 instead of only one JIMO mission for launch in 2015. This was under consideration but was not the JPL plan of record at the time of project termination. At the time of the Project Mission and Scope Review (PMSR) in July 2005, a single 2015 JIMO launch was favored. The NRPCT notes that this launch date was aggressive and required a success-based schedule. Any setbacks would negatively impact the schedule and jeopardize the launch date.
- **Inclusion of a second Ground Test Reactor.** A second GTR was included in the initial estimate. The first ground test reactor would be tested to get early information on thermal-mechanical reactivity feedback or other operational issues that could be incorporated into the final, flight unit design. The second ground test reactor would be prototypic of the flight unit. At the time of project termination, work was in progress to revise the technology development timeline to eliminate the need for a second GTR. Note that at the time of project termination, the NR program had not concluded that a GTR was necessary or where it would be sited.
- **Limited back-up plans for materials.** Based on launch dates, the NRPCT needed to finalize the primary material choices for the fuel and clad by September 2005. The NRPCT considered it wise to carry more than one candidate fuel system through material characterization and irradiation testing. The level of backup core design and core vendor manufacturing development using the alternate materials had not yet been established. The April 2005 SRPE assumed no backup plans in these areas in order to focus resources on the primary path; however, future updates to the SRPE likely would have included different assumptions.
- **Choice of primary fuel system by September 2005.** This was needed to order long lead material, build an assembly for cold critical testing, build a reactor for zero power testing, and build a ground test reactor for full power testing. $UO_2$ was selected as the primary fuel system.
- **Naval Nuclear Propulsion Program (NNPP) experience.** As a first approximation, NRPCT used past experience in NNPP projects as a basis of estimates for all elements of the project. The planning basis was under review, as some of this past experience may not be directly applicable to a space application, resulting in either over or under-estimating costs. Nevertheless, at the conceptual stage of development, these estimates were judged suitable for planning purposes.
- **Inclusion of substantial non-nuclear testing of SNPP qualification and flight models.** This was necessary to maximize assurance that the SNPP would function as intended when integrated with the deep space vehicle.
- **Inclusion of an Assembly, Integration, and Test (AIT) approach using vendors.** This integrated approach would minimize the cost, shipping complexity, regulatory effort, and cycle time for the project.

Cost estimates for M&S and Labor were $2.9B for development and launch of a preliminary flight test spaceship, including the construction of a GTR-1 facility. An additional $850M would be required for development and launch of the JIMO spaceship, including the second GTR. This rolled up to a total of ~$3.8B (with no contingency).

## Figure 4-2: Project Milestone Schedule




| ID | Task Name | Start | Finish |
|---|---|---|---|
| 1 | **Space Nuclear Power Plant (SNPP) Milestones** | | |
| 2 | NR / NASA Memorandum of Understanding (MOU) Signed | 8/5/04 | 8/5/04 |
| 3 | Feasibility Study Complete | 10/29/04 | 10/29/04 |
| 4 | Preliminary Space Reactor Planning Estimate (SRPE) | 12/9/04 | 12/9/04 |
| 5 | Concept Downselect | 3/4/05 | 3/4/05 |
| 6 | Budget Quality Space Reactor Planning Estimate (SRPE) | 4/1/05 | 4/1/05 |
| 7 | RMIF Independent Cost Estimates (BWXT, EB, NGNN) | 8/15/05 | 8/15/05 |
| 8 | Fuel Decision | 9/15/05 | 9/15/05 |
| 9 | Pre-Conceptual Design Complete | 9/26/05 | 9/26/05 |
| 10 | Core Materials Selected (including Cladding Material) | 9/30/05 | 9/30/05 |
| 11 | Initial Fuel Order for Cold Critical Test | 10/3/05 | 10/3/05 |
| 12 | Release Bid Solicitation for Reactor Module Integration Contractor (RMIC) | 3/31/06 | 3/31/06 |
| 13 | Begin JOYO Irradiation Testing | 6/1/06 | 6/1/06 |
| 14 | Programmatic EIS Record of Decision (NASA) | 3/6/07 | 3/6/07 |
| 15 | Cold Physics Tests Begin | 6/30/07 | 6/30/07 |
| 16 | Conceptual Design Complete (for PDR) | 1/22/08 | 1/22/08 |
| 17 | Thermal Test Unit (TTU) Ready | 2/4/08 | 2/4/08 |
| 18 | Deliver Engineering Model (EM) Components to RMIF | 6/4/08 | 6/4/08 |
| 19 | Complete PCS/HRS/EP Thermal Test Unit (TTU) Demonstration | 8/6/08 | 8/6/08 |
| 20 | Reactor Module Integration Facility (RMIF) EIS Record of Decision | 9/4/08 | 9/4/08 |
| 21 | GTR (Prototype) EIS Record of Decision | 9/24/08 | 9/24/08 |
| 22 | Initial Fuel Irradiation Data Available (MICE TEST Complete) | 11/3/08 | 11/3/08 |
| 23 | Core Vibration & Acoustic Test Unit Ready | 1/20/09 | 1/20/09 |
| 24 | Engineering Model (EM) Demonstration (Glenn) | 7/13/09 | 7/13/09 |
| 25 | Complete Reactor Core Shock, Vibration, and Acoustic Testing | 9/18/09 | 9/18/09 |
| 26 | Fuel Order for Hot Critical Test | 10/1/09 | 10/1/09 |
| 27 | Reactor Module Shock & Vibration Test Unit Ready | 10/7/09 | 10/7/09 |
| 28 | Impact Test Unit Ready | 10/7/09 | 10/7/09 |
| 29 | Complete RX/PCS/EP Engineering Model (EM) Demonstration (Glenn) | 1/14/10 | 1/14/10 |
| 30 | Set GTR Element Design for Long Lead Materials | 1/20/10 | 1/20/10 |
| 31 | NASA ATLO Facility EIS Record of Decision | 3/12/10 | 3/12/10 |
| 32 | Complete Reactor Plant Shock and Vibration Qualification Testing | 6/7/10 | 6/7/10 |
| 33 | Complete Rocket Sled Impact Qualification Testing (Basis for INSRP) | 6/7/10 | 6/7/10 |
| 34 | Complete Structural Test Model (STM-1) | 11/8/10 | 11/8/10 |
| 35 | Fuel Element Irradiation Data Available | 11/30/10 | 11/30/10 |
| 36 | Deliver Qualification Model (QM) Components to RMIF | 6/6/11 | 6/6/11 |
| 37 | Ground Test Reactor (GTR) Reactor Final Design Complete | 7/11/11 | 7/11/11 |
| 38 | Ground Test Reactor Facility Modifications Complete | 9/30/11 | 9/30/11 |
| 39 | Qualification Model (QM) Testing Begins (Plum Brook) | 6/20/12 | 6/20/12 |
| 40 | Ground Test Reactor (GTR) Hot Critical Test (Core Only) | 7/9/12 | 7/9/12 |
| 41 | Prometheus 1 EIS Record of Decision | 10/5/12 | 10/5/12 |
| 42 | Prometheus 1 Reactor Final Design Complete | 10/25/12 | 10/25/12 |
| 43 | Complete SNPP Qualification Model (QM) Demonstration (Plum Brook) | 1/7/13 | 1/7/13 |
| 44 | Ship GTR Non-Nuclear Reactor Components to RMIF | 4/22/13 | 4/22/13 |
| 45 | GTR Reactor Module Integration at RMIF Complete | 6/15/13 | 6/15/13 |
| 46 | Installation of GTR at Prototype Facility Complete | 11/15/13 | 11/15/13 |
| 47 | Ship Prometheus 1 Reactor Components to RMIF | 4/18/14 | 4/18/14 |
| 48 | **Ground Test Reactor (GTR) Initial Criticality** | 4/21/14 | 4/21/14 |
| 49 | Prometheus 1 Zero Power Critical Testing at GTR Facility | 8/18/14 | 8/18/14 |
| 50 | Power Unit Integration with Spacecraft at KSC | 11/11/14 | 11/11/14 |
| 51 | **Prometheus 1 Launch** | 5/14/15 | 5/14/15 |
| 52 | **Jet Propulsion Laboratory (JPL) Milestones** | | |
| 53 | Preliminary Mission and System Review (PMSR) | 7/19/05 | 7/19/05 |
| 54 | Preliminary Design Review (PDR) | 4/30/08 | 4/30/08 |
| 55 | Critical Design Review (CDR) | 8/11/10 | 8/11/10 |

## 4.2 Acquisition Strategy

The Naval Reactors Program approach was to maximize use of the existing national infrastructure within DOE and NASA. Inter-contractor purchase orders were placed with several DOE National Laboratories, and work agreements were established or being negotiated with Government-managed NASA centers (Marshall and Glenn), with others planned to follow (Ames and Kennedy). A separate purchase order or task order was planned to be placed by KAPL with NASA-JPL's operating contractor (CalTech) to enable Price Anderson indemnification and other requirements (e.g., security requirements under the Atomic Energy Act) to flow from KAPL's DOE prime contract to JPL. This process was not completed due to termination of the NASA-NR collaboration. NRPCT was also working on a purchase order with NGST for certain SNPP development activities, the contractor selected by JPL for co-design and delivery of the spaceship. The separate NRPCT contract (through KAPL) was planned to facilitate NGST support of SNPP-specific work and enable Price Anderson and other nuclear requirements to flow directly from KAPL's DOE prime contract to NGST. Negotiation of the purchase order with NGST was underway when the NASA-NR collaboration was terminated.

As further discussed in Section 4.3, the NRPCT vision was to competitively bid the contract for the role of Reactor Module Integration (and assembly) Contractor (RMIC). This acquisition strategy was in its infancy at project termination, but three potential suppliers were engaged to provide conceptual evaluations and cost estimates [References (144), (145), (146)] which would have helped the NRPCT to further develop the acquisition strategy in this area. Directly related to this was the strategy for acquisition of the reactor core and power unit. The configuration of the power unit and the activities to be performed by the core vendor versus the RMIC or other contractors had yet to be worked out at project termination. The physical similarity of some core configurations under evaluation to commercial reactor fuel systems led to interest in exploring use of a modified version of existing commercial fuel fabrication technology at a facility that is already licensed to handle highly enriched uranium. This approach has potential for reduced cycle time and cost through technology sharing and use of existing facilities infrastructure and licenses for security and handling of Special Nuclear Material. This approach had not been pursued beyond initial site visits and conceptual ideas at the time of project termination.

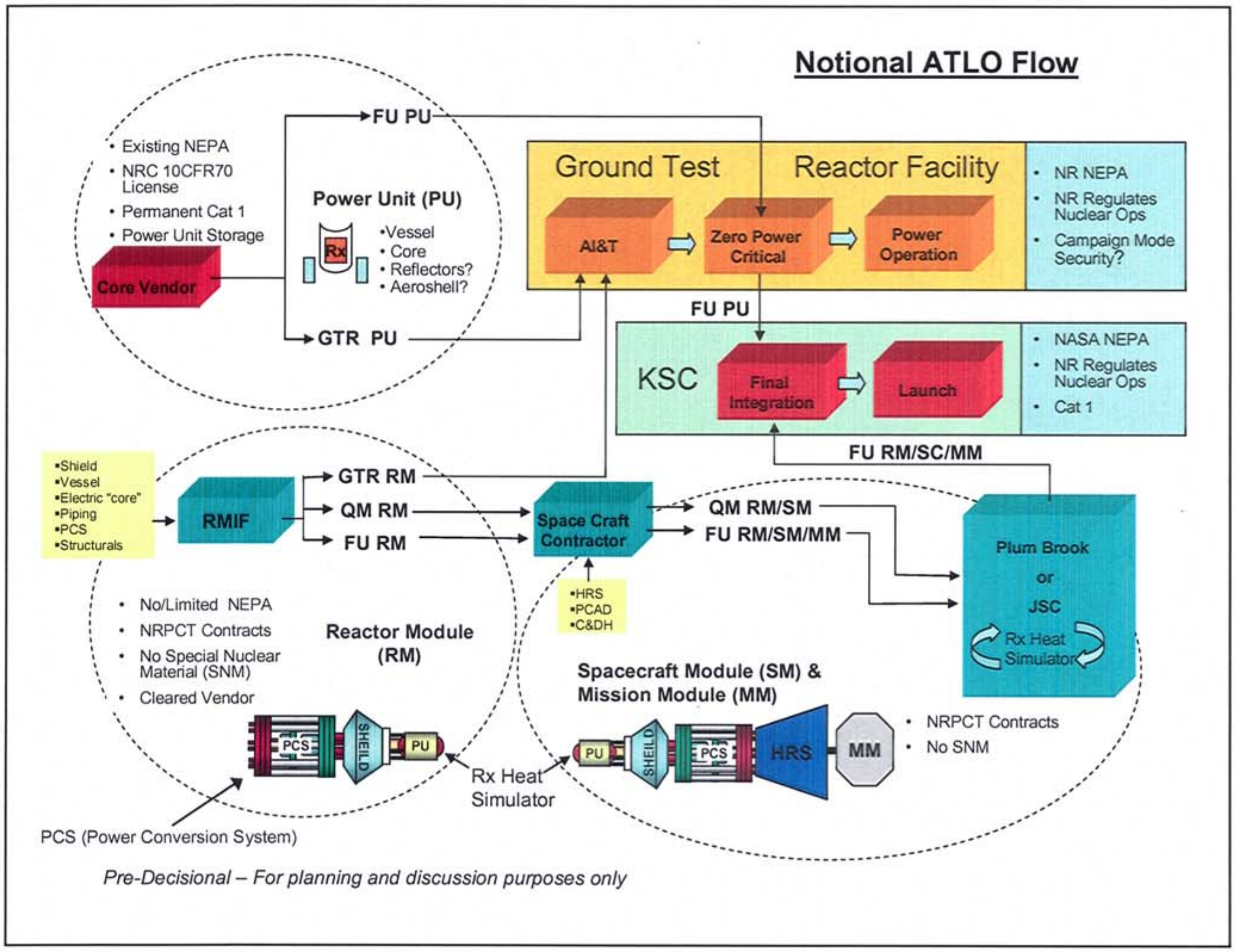
Acquisition strategies were also developed for the SNPP Reactor Instrumentation and Control Segment. I&C components (circuit boards, racks, and sensors) were to be procured by NRPCT. Software development, hardware integration, and Reactor I&C Segment testing were to be performed by the NRPCT, such that a fully tested and operational I&C system would be provided to NGST for integration with the spaceship. Interconnecting wiring and cabling would be designed and integrated with the spaceship by NGST. Sensors would be procured by BPMI and integrated with the SNPP by the RMIC.

Acquisition planning for the Reactor I&C Segment was considering use of common spaceship components whenever possible to reduce overall complexity and cost. Several components (card racks, power supplies, processor cards, etc.) would be common to the Reactor I&C, Power Conditioning and Distribution (PCAD), and Flight Computer systems to reduce both development and recurring costs. NRPCT anticipated that NGST would have supplied many of these components. A limited set of custom cards were required for the reactor I&C system for specific nuclear power plant functions. For the nuclear detector and ultrasonic cards, NR Program suppliers with existing technology were selected and teamed with corporate partners with space experience. For other cards that did not have an existing NR Program technology base, the intent was to competitively procure the cards.

## 4.3 Assembly, Test, and Launch Operations (ATLO) Strategy

The NRPCT developed a preliminary approach for integration, assembly, and test of the flight unit Reactor Module (RM) and spaceship. The relatively large size of the spaceship, combined with a significant quantity of special nuclear material, created logistical and regulatory issues (facility licensing issues, security and safeguards, and shipping) that would preclude assembling and testing the spaceship as a unit with a nuclear-fueled RM. In addition, the flight unit reactor should not operate at significant power levels before launch so as not to create a large inventory of radioactive fission products and to avoid exposing the flight reactor core to detrimental environments during testing (e.g., rapid degradation during high temperature operation in air). To satisfy these constraints, the NRPCT developed an integration, assembly, and test (IA&T) approach, shown in Figure 4-3, involving two parallel paths for the flight hardware – nuclear and non-nuclear. The paths merge as late in the assembly and test flow as possible to minimize the above concerns.

**Figure 4-3: Notional Assembly, Integration, and Test Approach**




The extent of assembly at the power unit stage had not been resolved at the time work was terminated. Instead of a fueled reactor "power unit" (PU) assembly, an unfueled reactor module would be connected to the flight unit for qualification testing. The electrically heated RM was to be received by the spaceship contractor (SC) and integrated into the flight spaceship along with the scientific mission module. The SC would then subject the flight spaceship to the necessary mechanical, thermal, and electro-magnetic environmental tests, with JPL's guiding principle to "test-as-you-fly and fly-as-you-test." This testing was to occur at both the SC integration facility and at an off-site thermal-vacuum facility (Johnson Space Center (JSC) or GRC's Plum Brook Station). The early preference was for JSC due to more favorable transportation logistics with a spaceship of this size and mass. The flight spaceship would then be shipped to Kennedy Space Center (KSC) to be mated with the fueled PU.

The nuclear path involved the assembly of the fueled PU at the core vendor and/or physics test site, zero power physics testing, and then transport to the launch site for integration with the previously tested energy conversion system and spaceship. At the time of project cancellation, the scope of the PU and options for its assembly had yet to be defined. The issues to be considered involved:

- Extent to which the PU could be assembled at a core vendor or other facility within the existing regulatory framework (i.e. 10CFR70 license from the Nuclear Regulatory Commission (NRC) or NR regulatory authority)
- Configuration of the PU during shipping with regard to nuclear safety and cost
- Configuration control requirements of reactivity control devices before and after physics testing
- Extent of the flight unit RM that should undergo environmental testing with the spaceship

The NRPCT and the SC conducted scoping studies of notional PU configurations ranging from a core cartridge to the complete Reactor Module as shown in Figure 4-4. These studies did not yield a preferred approach, but helped to define key issues that would have to be addressed.

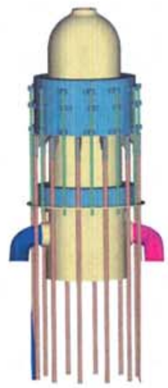
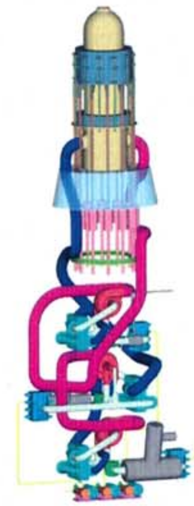
Integration of the fueled PU with the RM and spaceship was to be performed at KSC in a dedicated facility. The potential for critical physics testing to be done at KSC was also considered. Physics testing at KSC was not ruled out; however, the baseline planning assumption was to conduct physics testing at whatever site was used for physics testing of earlier test articles used for reactor development (e.g., physics test or ground prototype reactors). KSC performed a conceptual study to inventory and evaluate existing facilities at both KSC and Cape Canaveral Air Force Station (CCAFS) for suitability to support the Project Prometheus spaceship processing operations [Reference (615)]. The objective of the study also included the conceptual requirements and possible locations for a new spaceship processing facility if no existing facilities were viable candidates. This study determined that no existing buildings were suitable and provided a recommendation for a new four-building facility. The buildings were to include:

- Spaceship processing facility large enough to accommodate a fully deployed JIMO spaceship
- Power unit processing facility to house the reactor prior to integration with the spaceship
- Ground support equipment storage building
- Operations office building

Although the conceptual study was conducted with minimal input from the NRPCT, KSC recognized that the scope of requirements that would be invoked by NR to support operations involving a Category I quantity of Special Nuclear Material, would need to be developed and assessed for compatibility with existing KSC/CCAFS requirements. These requirements were expected to dictate some of the design attributes of the facility, and as such, they factored into the early phase of the spaceship processing facility plan.

The ATLO plan described above requires the RMIC to perform the assembly activities in the RMIF and potentially PU assembly operations not performed by the core vendor. It was considered likely that synergy could be gained by the RMIC also being a supporting contractor for the spaceship integration activity and the performing contractor for the pre-launch operation to mate the fueled PU to the RM. The NRPCT planned to execute a competitive bid process to select the RMIC.

**Figure 4-4: Notional Reactor Power Unit Configurations**

| Core Cartridge | Power Unit | Power Unit and Shield | Reactor Module |
|---|---|---|---|
|  |  |  |  |
| PRO<br>• Most flight hardware in system test<br>• No constraint to RM design external to vessel<br>• Few RM related industrial operations at KSC / Best final weld access<br><br>CON<br>• Constraint to reactor design internal to vessel<br>• Physics testing complexity<br>• Weld distortion of reactor vessel / reactivity control device interface<br>• Smallest package of SNM, potentially harder to safeguard | PRO<br>• More Flight Hardware included in system test: (e.g., Reactivity control devices)<br>• No constraint to RM design internal to vessel<br>• Use existing Shipping container<br><br>CON<br>• Constraint to RM design external to vessel<br>• More RM related Industrial Operations at KSC<br>• Reflectors in Shipping container with core<br>• Smaller package of SNM, potentially harder to safeguard | PRO<br>• No constraint to RM design internal to vessel, limited constraint external<br>• Fewer RM Industrial Operations at KSC<br>• Use existing Shipping container<br><br>CON<br>• Less Flight Hardware included in system test: (e.g., Reactivity control devices, shield, I&C sensors)<br>• Reflectors in shipping container with core<br>• Smaller package of SNM, potentially harder to safeguard | PRO<br>• No constraint to RM design internal or external to vessel<br>• Least RM related Industrial Operations at KSC<br>• Largest package of SNM, potentially easier to safeguard<br><br>CON<br>• Least flight hardware in system test<br>• Most complex to ship / New shipping container required<br>• Reflectors in shipping container with core |

## 4.4 Program Management Perspectives

Centralized Leadership and Program Management

The program structure established for Prometheus has the potential to be a successful model for space reactor development and deployment. A responsible SNPP lead organization possessing the authority to make technical and programmatic decisions must be established early in the project to plan, integrate, and iterate with NASA and its contractors. This is particularly important during the early spaceship design where requirements and interfaces are being defined for the organizations well as the hardware. The responsibility to integrate nuclear technical information and to differentiate among facts, opinions, and judgements from the involved parties should rest with a single lead technical organization and be based on formal technical documentation. A central and experienced nuclear system design and deployment agency, working in concert with the existing expertise and infrastructure in government (domestic and foreign), industry, and academia, is an important part of the development.

Formal Decision Process

Choosing the right reactor system technologies to meet mission requirements, maximize reliability and deliverability, and minimize risk and cost requires detailed evaluation of alternatives and a formal decision basis. The NRPCT selected a direct Gas-Brayton system for the NEP missions based on the range of mission and schedule requirements envisioned. This was done through preliminary development of multiple concepts, formal and written decision criteria, establishment of decision teams, technical input from experts, debate among those teams, review by independent consultation teams, formal recommendation to NR headquarters, and formal technical and funding approval. The rigor associated with this process surfaced a number of issues, quantified the concerns, and provided balance between performance and deliverability. A similar formal decision process was used for the fuel material recommendation and would have been used for finalizing Reactor Module architectures, selection of key core and plant materials, and operating strategies.

Development Schedule

The timeframe for development and deployment of the JIMO mission reactor system was ~11 years. Given the need for materials testing, criticality testing, system testing and likely, construction of a prototype GTR Facility, the development timeline established for the JIMO program was very aggressive. To ameliorate this concern in future projects, the scope and timescale required for an engineering development, manufacturing, and testing effort of this magnitude should be understood from the beginning, and appropriate funding and mission consistency should be committed through the duration of the program. Consideration should be given to setting less aggressive mission requirements (i.e., shorter duration) for initial missions to allow using more readily available technologies. More capable technologies could be pursued in parallel for follow-on missions through a technology development program that advances key technologies. Pursuing a technology program in parallel with initial missions would and shorten the time to deliver future projects, when identified. Examples of appropriate technology programs might include high temperature material, fuel system and shield material development and preliminary qualification

The gas reactor concept was selected in part because it minimized the expected development time and cost. However, the space reactor planning estimate ground-up cost estimate for the work was still substantial. Long-lived nuclear reactor development work is technically-exacting, time–consuming, and testing-intensive. This necessitates high labor and facility costs starting early in the development process. Expectation of a rapid and inexpensive reactor development and deployment to meet unprecedented performance requirements would be misplaced.

# 5 ASSET TRANSFER

## 5.1 NRPCT Documentation

This section describes the formal technical and programmatic information that was generated, how and where it is stored, and how future researchers can access the information. A detailed project bibliography is included in Volume 3 to serve as a roadmap that identifies both key external references that were found to be useful by the NRPCT, as well as important documents that were generated by the NRPCT or by NRPCT direction.

### *5.1.1 NRPCT Technical and Programmatic Documents*

A number of closeout reports were generated to provide an orderly closeout of NRPCT work on Project Prometheus. In addition, key technical and programmatic documents were generated during the course of the project, as shown in Figure 2-2. Section 2 of the Volume 3 Bibliography summarizes the NRPCT documentation that, taken together, provides a complete view of the NRPCT effort through pre-conceptual design. In general, this list comprises closeout documents that summarize a large amount of more detailed reports and analyses. The reference lists contained within these closeout documents provide further sources of information.

### *5.1.2 Accessibility of NRPCT Prometheus Information*

The U.S. DOE Office of Scientific and Technology Information (OSTI) will be the primary repository of NRPCT information to be accessible to appropriate agencies external to the Naval Reactors Program. The Volume 3 Bibliography denotes which NRPCT documents will be provided to OSTI. Two OSTI databases will be used.

- Unclassified, non-sensitive documents will be placed in the internet-accessible Science Resource Connection (SRC) database, which is available to DOE personnel and DOE contractors. These documents will be downloadable from the internet.

- Sensitive unclassified documents, specifically U-SNRI (Unclassified Space Nuclear Reactor Information) or OUO (Official Use Only) marked documents, will be stored in an OSTI repository that is not internet accessible. OSTI will direct requests for these sensitive documents to the Schenectady Naval Reactors Office (SNR), Security and Safeguards Division. U-SNRI requesters that are from organizations that have been officially engaged by the Government (e.g., the DOE or NASA) will need to demonstrate need-to-know and sign an information protection agreement reflecting the pertinent requirements of SN-801 [Reference (18)]. This agreement includes limiting U-SNRI access to personnel within their organization that are U.S. citizens and have an official need-to-know. If approved by SNR, OSTI will provide the requester a hardcopy of the requested document, as hardcopy distribution will eliminate information protection concerns associated with electronic transfer.

The NRPCT has created a small number of documents that are classified as CONFIDENTIAL under DOE/DOD/NASA Classification Guide for Space Reactor Power Systems, CG-SRPS-1. These documents, as denoted in the Volume 3 Bibliography, will be maintained internal to the NR Program and will not be provided to OSTI. Requests for CONFIDENTIAL documents should be directed to the Schenectady Naval Reactors Office (SNR), Security and Safeguards Division. Requesters will need to have an official security clearance, need-to-know, and meet information protection requirements as determined by SNR. Any distribution of this material will be made by SNR and may reflect redaction of CONFIDENTIAL information.

## 5.2 Equipment and Materials

Upon termination of NR's agreement with NASA, the NRPCT identified capital equipment and sensitive items with original acquisition values greater than $5,000 in condition code 7[1] or better to NASA for disposition instructions. In addition, certain other non-capital equipment and non-sensitive items that were direct funded using the NASA funding source were identified for disposition as well. In addition, the NRPCT has finalized all potential radioactive legacy issues, such as disposal or remediation at DOE laboratories. Future requests for or about any material or equipment that was transferred to NASA should be directed to the points of contact listed below.

| | |
|---|---|
| Marshall Space Flight Center (MSFC): | Mike Fazah<br>ER11/Nuclear & Advance Propulsion Brance<br>George C. Marshall Space Flight Center<br>Huntsville, AL 35812<br>Office: (256) 544-8475 |
| Glenn Research Center (GRC): | Joseph Nainiger<br>MS86-6<br>NASA Glenn Research Center<br>Cleveland, Ohio 44135<br>Office: (216) 977-7103 |
| NASA Headquarters (HQ): | Beverly Hamilton / IT Program Manager<br>2X37<br>ATTN: Receiving and Inspection<br>NASA HQ<br>300 E Street SW<br>Washington, DC 20024-3210<br>Office: (202) 358-5180 |

[1] Condition code 7 Property refers to that property which is unusable in its current condition but can be economically repaired.

# 6 PROJECT COST

On March 11, 2004, the NRPCT formally commenced Space Power Program work as directed by Naval Reactors. KAPL and Bettis were each authorized to charge 15 manyears in FY04 to begin this project. As the initial scoping and pre-conceptual design studies began, subcontracts were established with several DOE laboratories, and working relations were developed with NASA agencies. At the beginning of FY05, the Prime Contractors (KAPL, Bettis, and BPMI) began to add personnel in earnest. This staff-up, in parallel with a steady growth in subcontracted work, continued through the selection of the spaceship integration contractor (NGST), until project close-out was initiated on September 8, 2005. Subcontracts were then quickly terminated or re-scoped as appropriate, and NRPCT de-staffing began.

## 6.1 Total Project Cost (Inception-to-Date):

The total NRPCT cost for the Space Power Program (which reflects fully burdened manpower costs), from inception (March 11, 2004) through January 31, 2006, is estimated to be 302 manyears (at a fully burdened cost of $50.19M) and $42.59M of Materials and Service (M&S). These figures reflect actual costs through December 2005, combined with cost estimates for January. The inception-to-date project costs are provided by Control Account below.

**Table 6-1: Inception-to-Date Project Costs**

| Control Account | | Man Years | Labor $M | M&S $M | Total $M |
|---|---|---|---|---|---|
| 3.2.1 | Space Management & System Engineering Support | 39.4 | $ 6.49 | $ 10.97 | $ 17.46 |
| 3.2.2 | Space Plant Integration | 1.0 | 0.17 | 0.45 | 0.62 |
| 3.2.3 | Space Reactors Safety & Assurance | 3.5 | 0.59 | 2.10 | 2.69 |
| 3.2.4 | Space Materials Development & Support | 74.5 | 12.40 | 16.36 | 28.77 |
| 3.2.5 | Space Prototype Test Facilities | 8.2 | 1.38 | 0.07 | 1.45 |
| 3.2.6 | Space Reactor Servicing & Support Equipment | 0.7 | 0.11 | 0.00 | 0.11 |
| 3.2.7 | Space Reactor Engineering | 78.4 | 13.11 | 5.19 | 18.30 |
| 3.2.8 | Space Plant Engineering | 58.4 | 9.74 | 1.31 | 11.05 |
| 3.2.9 | Space Radiation Shield Engineering | 4.3 | 0.71 | 1.43 | 2.14 |
| 3.2.10 | Space Instrumentation, Control & Electrical Systems | 33.7 | 5.48 | 4.71 | 10.19 |
| | TOTAL | 302.0 | $ 50.19 | $ 42.59 | $ 92.78 |

Of the M&S costs above, a significant portion was subcontracted work at national laboratories. The final contract amounts at each DOE Laboratory are, or are expected to be, as follows:

- ORNL: $ 11.30M (estimated)
- LANL: $ 10.99M (estimated)
- PNNL: $ 1.89M (estimated)
- INL: $ 0.71M
- SNL: $ 0.50M
- Y12: $ 0.49M

The NRPCT also sponsored work at NASA-Marshall, but that work was funded directly by NASA and is therefore not included in this report.

## 6.2 Residual Project Closeout Costs (Through FY06):

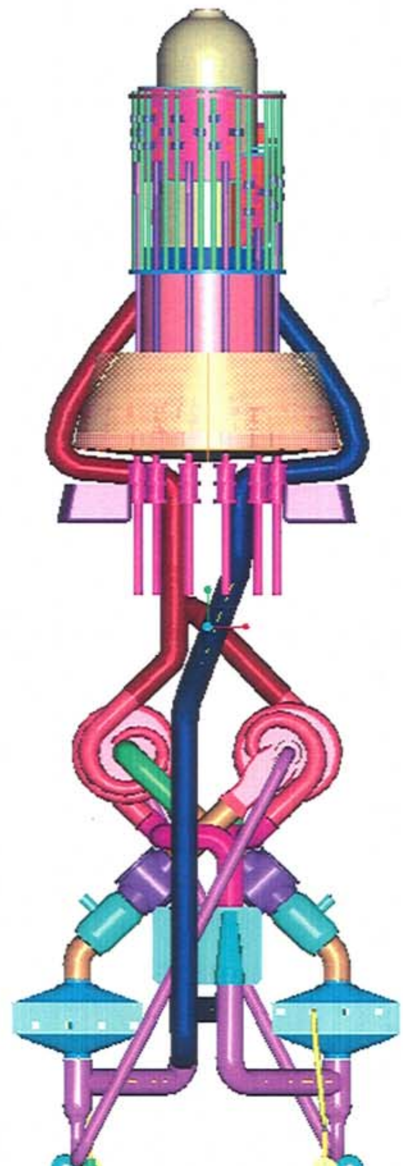
Program costs for the following residual activities are expected to be incurred:

- Close-out of subcontracts.
- Completion of information storage and archival activities.
- Distribution of the Prometheus Project Reactor Module Final Report to stakeholder organizations.
- Submittal of applicable NRPCT reports to OSTI.
- Assistance with External Queries and Requests for Information.
- Performance of final cost accounting.

Residual project close-out costs for the work above are estimated to be 8.5 MY (at a fully burdened cost of $1.39M) and ~$0.32M of M&S.

# Project Prometheus Reactor Module

## Technical Summary



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# 1 INTRODUCTION AND SUMMARY

Volume 2 of the Prometheus Project Reactor Module Final Report is a technical summary of the work accomplished by the Naval Reactors Prime Contractor Team (NRPCT). Project Prometheus was established in 2003 with a goal of developing the first nuclear-powered electric propulsion system for a spaceship and demonstrating that it can be operated safely and reliably for civilian deep-space exploration missions. The initial application of space fission power being developed was the Jupiter Icy Moons Orbiter (JIMO), a nuclear electric propulsion spaceship intended to perform deep-space scientific research.

## 1.1 Background

NRPCT involvement in Project Prometheus began in March 2004. The NRPCT was assigned responsibility for design and delivery of the Reactor Module (with the exception of the Aeroshell re-entry protection cover) for Project Prometheus, while Jet Propulsion Laboratory (JPL) and its contractors were responsible for the remainder of the spaceship, launch vehicle, and ground systems. The set of elements needed to generate and manage the power created by the reactor is referred to as the Space Nuclear Power Plant (SNPP). The SNPP includes the Reactor Module, the Heat Rejection Segment (HRS), and the Power Conditioning and Distribution (PCAD) Subsystem.

The project had three major phases of work: Project Scoping, Pre-conceptual Design, and Project Close-out. During the Project Scoping phase, two major technical reports were produced: a concept feasibility evaluation [Reference (1)] and a concept selection recommendation [Reference (2)]. These reports are described in more detail in Section 1.2. NRPCT recommended a direct cycle, gas-cooled reactor with a Brayton energy conversion system. Approval of this concept marked the start of the Pre-conceptual Design phase. The major goals of this phase were to select the fuel and clad system materials, select plant parameters to set the baseline heat balance, and select the baseline Reactor Module arrangement. This work was well underway when the NASA-NR collaboration was terminated in September 2005. NRPCT documented the results of the Pre-conceptual Design phase in a number of detailed technical reports, as shown in Figure 2-2 of Volume 1.

Section 1.3 provides a summary of the Prometheus Project requirements. The Pre-Conceptual direct gas Brayton concept is described in Section 1.5. Section 1.4 describes the technical challenges of the Prometheus Reactor Module design, and serves as an introduction for the detailed technical summaries contained in Sections 2 through 7. Section 8 discusses the mission extensibility of the direct gas Brayton concept. Finally, Section 9 provides high level conclusions and a perspective for future space reactor development endeavors.

## 1.2 Summary of Feasibility Studies and Concept Selection

The goal of the Reference (1) evaluation was to provide an initial NR Program assessment of the design space for providing a nuclear power plant to support civilian space exploration. Viable reactor, coolant, and energy conversion technologies were studied with respect to capability and technology readiness. The likelihood of meeting mission requirements (e.g., electrical power requirement, mission duration) as well as launch schedule (2015 launch date) was considered.

One of the goals of the design space assessment was to broaden the scope of potential options, including those that had been eliminated by previous Government Team studies. Specifically, lower temperature concepts were of interest. The Government Team had been working with an assumption of 1350 K for reactor exit temperature ($T_{hot}$). This was viewed to be aggressive for material systems that could be considered for this mission and delivery timeframe. The gas Brayton system and the liquid metal Stirling system, with their higher cycle efficiencies, both allow for lower reactor temperatures. The gas Brayton system being evaluated had no intermediate heat exchanger which allowed for a reactor exit temperature of 1150 K (the maximum inlet temperature for the Brayton engine using conventional materials), and the liquid metal Stirling system utilized conventional materials in the reactor and was, therefore, limited to a reactor exit temperature of ~1050 K.

The NRPCT determined that all reactor design approaches for supporting the JIMO mission requirements needed considerable development and that the minimization of such development to support the 2015 launch date required selection of the more well-developed technologies. From this evaluation, fuel and core materials as well as integrated plant design and testing emerged as the biggest challenges. The topics evaluated during the design space assessment are summarized as follows:

- Reactor types: Liquid metal-, heat pipe-, and gas-cooled reactors.
- Coolants: liquid alkali metals (lithium (Li), sodium (Na), potassium (K), and sodium-potassium eutectic (NaK)), non-alkali liquid metals (lead (Pb), lead-bismuth (Pb-Bi), gallium (Ga), and tin (Sn)), molten salts (fluorides and chlorides) and gas (helium and xenon (HeXe)).
- Fuel materials: uranium dioxide ($UO_2$), uranium mononitride (UN), and uranium carbide ($UC/UC_2$).
- Cladding and core structural materials: refractory metal alloys (niobium (Nb), tantalum (Ta), molybdenum (Mo), rhenium (Re), and tungsten (W) alloys), conventional metal alloys (stainless or ferritic steel, oxide dispersion strengthened iron alloys, nickel-base (Ni-base) superalloys, and vanadium (V) alloys), and silicon carbide (SiC) ceramics (including composite and monolithic components).
- Shielding and reflector materials: lithium hydride (LiH), beryllium (Be), beryllium oxide (BeO), boron carbide ($B_4C$), other metal carbides, water, and alternate hydrides.
- Fuel configuration: fuel pellets in bonded or unbonded cylindrical pins (with fission gas plena), dispersed particle concepts (both cermets and TRISO-like configurations), and small metal or ceramic spheres.
- Reactor neutron energy spectrum: fast and thermal.
- Reactor safety and reactivity control features: safety rods, external control devices (sliders, drums, other), in-core control rods, spectral shift poisons, and burnable poisons.
- Dynamic energy conversion systems: Brayton, Stirling, water Rankine, and potassium Rankine.
- Static energy conversion systems: thermoelectric (TE), in-core thermionics (TI), thermophotovoltaics (TPV), magnetohydrodynamics (MHD), and alkali metal thermal-to-electric conversion (AMTEC).
- Heat rejection system coolants: NaK, water, and Li pumped loops.

The NRPCT concluded that there was a design space for three reactor/coolant types (liquid metal, heat pipe, and gas) and three energy conversion types (Brayton, Stirling, and Thermoelectric). The five configurations that were evaluated in the final concept selection process were:

1) A direct cycle, gas-cooled reactor with a Brayton energy conversion system
2) A heat pipe-cooled reactor with a Brayton energy conversion system
3) A liquid lithium-cooled reactor with a Brayton energy conversion system
4) A liquid lithium-cooled reactor with a thermoelectric energy conversion system
5) A lower temperature, liquid metal- (e.g., Na, K, or NaK) cooled reactor with a Stirling energy conversion system

Based on overall system features including capability, reliability, deliverability, cost, and safety, the gas Brayton concept (concept #1 listed above) was recommended to Naval Reactors for approval in Reference (2), which also documented the formal decision process and technical basis for the decision. Naval Reactors approved the recommendation via Reference (3). The gas reactor system is likely capable of fulfilling the mission requirements for the envisioned nuclear electric propulsion (NEP) missions, would simplify engineering development testing and offer the fewest hurdles to development, and could be extensible to surface missions.

## 1.3 Project Prometheus Requirements

Following concept selection, NRPCT focused on the pre-conceptual design effort. Definition of the technical requirements and development of the methods and technologies to meet those requirements became a primary focus of the pre-conceptual design phase. Evolution and details of the driving requirements are described below.

Requirements for Project Prometheus progressed from Level 0 (the overall Exploration Requirements for NASA) which include the overall NASA mission statement, exploration requirements, and exploration objectives. The ambitious mission of orbiting and exploring the icy moons of Jupiter was developed to meet the Exploration Requirements for NASA and to support the "goal of developing the first reactor-powered spacecraft capability and demonstrating that it can be operated safely and reliably in deep space on long-duration missions" [Reference (357)]. The Level 1 JIMO Requirements [Reference (356)] issued by NASA headquarters are the Technology Development and Mission and Science requirements derived from NASA's Exploration Requirements at Level 0. The Level 1 Technology Development requirements describe the primary technical goals required to enable a deep space mission, and the Mission and Science requirements describe delivery of the spaceship to the Jovian system and operation during the science phase. The Level 1 JIMO requirements formed the starting point for development of project requirements and conceptual design efforts. Because some requirements presented in Reference (356) were still preliminary, some items are indicated as objectives or requiring further review.

The following Level 1 Technology Development requirements drive key Reactor Module requirements:

> *The JIMO Project shall develop a Deep Space Vehicle for outer solar system robotic exploration missions that combines a safe, reliable, Space Nuclear Reactor with electric propulsion.*
>
> *The Deep Space Vehicle shall have a Payload Accommodation Envelope with a mass capability of no less than 1500 kg.*
>
> *The following Space Nuclear Reactor technologies shall be developed for Lunar and Mars surface power reactors: 1) Nuclear fuel, 2) Reactor core materials and coolants, and 3) Instrumentation and Control.* (This item was indicated as an objective – minimum requirement not yet defined.)

Multiple studies and analyses were performed to develop and evolve a conceptual spaceship design to satisfy the Level 1 requirements. These studies, and the Level 1 JIMO Requirements, formed the basis for Level 2 Multi-mission (Deep Space Vehicle) and JIMO (Mission Module) requirements. The Level 2 requirements define key functional needs for the spaceship, including the Reactor Module. Numerous other Level 2 documents detail mission requirements, environments, hardware and software selection and validation requirements, safety and security requirements, science requirements, and many other aspects covering the design and validation of the JIMO mission.

The primary set of non-JIMO specific Level 2 technical requirements was collected into the Multi-mission Project Derived Requirements, Reference (358). The JIMO specific Level 2 technical requirements were given in Reference (359). The Level 2 technical requirements which drive key aspects of the Reactor Module design are listed in Table 1-1. Accompanying each requirement is a statement of the impact on the Reactor Module. Also provided, where applicable, is a description of the implementation required to meet the requirement. Electric power output of the Reactor Module is one of the most important requirements because it ultimately drives system mass, volume, and operating temperature. See Section 1 of the Plant Pre-Conceptual Design Report, Reference (4), for more detailed discussion of this requirement.

In addition to the key Level 2 requirements summarized in Table 1-1, many other important requirements were being considered in evaluating, selecting, and demonstrating design options. Some of these are listed in the JIMO Deep Space Vehicle Level 3 Key Driving Requirements, Reference (360): some in Prometheus Project Environmental Requirements Document (ERD), Reference (361): and others were still being developed. Some of the more important items for the Reactor Module are listed in Table 1-2, with references as appropriate.

Limits for several parameters such as mass and volume were still being developed in parallel with other reactor and spaceship design efforts. Although precise values were not defined, failure to address the limits indicated would result in a design which would overly burden the rest of the vehicle and could even make design of a viable spaceship untenable.

See Reference (360) for a more complete listing of the driving requirements for the Deep Space Vehicle, which includes the Reactor Module. The Plant Pre-Conceptual Design Report, Reference (4), provides additional description of requirements allocation within the project hierarchy.

**Table 1-1: Key Level 2 Requirements, Impacts, and Implementations**

| *Key Level 2 Requirement* | Impact on Reactor Module | Implementation |
|---|---|---|
| *The Space Nuclear Reactor design shall utilize technologies that facilitate extensibility to surface operations.* | Consideration in the selection of design and materials compatible with Lunar and Mars missions. | Must consider compatibility of pressure boundaries and external surfaces with surface environments. |
| *The Project shall use a Deep Space Vehicle that provides jet power greater than or equal to [130] kW of primary thrust during thrust periods.* | 200-kWe Reactor Module power output required to deliver net thruster power based on JPL orbital mechanics studies. | Plant Electrical Power ≈ 200 kWe<br>Plant Thermal Power ~ 1 MWt |
| *The Project shall design the Deep Space Vehicle to have an operating lifetime greater than or equal to [20] years.* | 20 year life is long term requirement for very deep space missions. The JIMO requirement is for 12 years. | Initial design efforts to support 15 year operational life.<br>Long term design goal is to satisfy 20 year life requirement. |
| *The Project shall use a Reactor Module that is capable of generating the maximum electrical power required by the Spaceship for cumulative minimum of [10] years, and is capable of generating the minimum required electrical power for the rest of the operating lifetime.* | This requirement permits the option of reducing power in order to conserve reactor energy or reduce pressure and temperature during non-thrust phases. This may maximize Reactor Module life for the most demanding follow-on missions to the outer solar system. | Trade studies would be required to determine if reduction of power would improve Reactor Module longevity. |
| *The Project shall comply with the Prometheus Single Point Failure Policy as documented in the Prometheus Project Policies Document 982-00057.* | Single point failure locations shall be avoided. Where this is not practical (e.g., reactor), it must be demonstrated that alternatives to single point failure are not available and sufficient robustness must be shown to mitigate risk of failure. | Where practical, redundancy would be part of the Reactor Module design. |
| *The Project shall be able to autonomously detect and correct any single fault that prevents thrusting in less than or equal to [1 hour].* (Note: Missing thrust during many of the mission phases severely jeopardizes mission success, and therefore should be prevented or minimized.) | This requirement must be considered in the design of instrumentation and control for a self-regulating plant and design for recovery from transients for which the module would be designed. | Robust and redundant system architecture for instrumentation and control.<br>Automatic recovery from transients must be considered in system design. |
| *The Spaceship shall survive without Ground System commanding for at least [50] days in the presence of a single failure.* | Must consider this, with other autonomy and single point failure requirements, in design of the control system. | Design for redundancy and robustness wherever practical. The spaceship cannot survive in deep space for more than a short time without reactor power. |
| *The Project shall assure that all Science System hardware in its deployed configuration, except approved science hardware, shall remain within the protected zone of the reactor radiation shield.* | Coordination between the shield and spaceship designs is required to assure that maximum dose levels are not exceeded. Shielding of local electronics will also be required. | Shielding sufficient to reduce payload neutron flux to 5E10 n/cm2 and payload gamma flux to 25 kRad Si damage and cover roughly a 12° by 6° cone angle. |
| *The Project shall obtain launch approval as specified in the Prometheus Launch Approval Plans.* | To meet this requirement, satisfaction of various governing safety requirements would have to be demonstrated by NRPCT and NASA. | Design features will be required to assure safety. Safety assurance must be considered during design of certain Reactor Module elements. |
| *The Spaceship total dry mass at launch shall not exceed [25,000] kg.* | Minimum module mass is a goal and a selection criteria for design. | High temperature reactor is required to minimize overall mass |

*Note: Values in [brackets] were not firm and thus subject to review.*

**Table 1-2: Additional Key Requirements**

| *Requirement* | Impact on Reactor Module |
|---|---|
| *The Spaceship launch configuration shall be compatible with a [5-m] launch vehicle payload fairing (dynamic envelope dimensions 4.5m diameter, 26m height), or smaller* (Ref (360)) | Arrangements and overall sizing must fit within allocated space inside the fairing. This mainly constrains the radiator area, which drives the heat balance design space of the Reactor Module. Preliminary studies limited radiator area to less than 450m$^2$. |
| *The Spaceship shall accommodate the solid particle mission environments defined in {ERD} with a probability of meeting end-of-mission (EOM) requirements greater than or equal to 0.99 ...* (Ref (361)) | Protection from orbital debris and micrometeoroids is required, especially of the crucial pressurized components and moving assemblies. |
| *The Spaceship shall be designed to accommodate the radiation environment specified in the Environmental Requirements Document (982-00029).* (Ref (360)) | Solar, galactic and Jovian radiation sources, coupled with reactor radiation, must be considered during electronics selection, shielding trades and material evaluation. |
| *The Spacecraft Module shall be capable of rejecting [682] kWt of heat from the Reactor Module.* (Ref (360)) | If radiator size is constrained, temperature and flow rate must be maximized to reject sufficient heat. This impacts the Reactor Module heat balance. |
| *The Prometheus flight hardware shall be designed and verified to meet applicable functional, performance, operation, and other design requirements without damage or degradation when exposed to the design environments specified herein {in ERD}.* (Ref (361)) | In addition to particle and radiation environments described above, the module must withstand launch loads and other space environments. Section 10 has more details about environments. |
| *During no-thrust periods of Science Orbits, the Deep Space System shall continuously point a Spaceship-fixed vector to commanded directions in the target-centric reference frame to within 20, 20, and 20 mrad (3 sigma) about the reference frame X, Y, and Z axes respectively…* (Ref (360)) | This requirement drives the need for positional stability of the spaceship. Counter-rotating Braytons, or alternate localized means to offset angular momentum, might be necessary to provide the needed stability. |

*Note: Values in [brackets] were not firm and thus subject to review.*

## 1.4 Technical Challenges

The Prometheus Project was aggressive from both a mission requirements and a launch schedule standpoint. The Prometheus mission is demanding due to the long mission duration (up to 20 years), high electrical power level relative to past space reactor projects (~200 kWe), harsh Jovian radiation environment (for the JIMO mission), and lack of auxiliary solar power. The SNPP that successfully meets these requirements must also fit within a launch mass and volume envelope, which was not yet specified. Four drivers – mass, lifetime, power, and reactor safety – are the primary technical challenges of the Prometheus SNPP.

In order to generate the required electrical power within a practical mass and volume envelope, the operating temperature of the SNPP was relatively high. Due to the lack of thermal conduction or convection capability in the vacuum of space, the reactor must operate at high temperature to efficiently produce the required level of electric power and to effectively reject waste heat by means of radiative heat transfer to space. Pre-conceptual design studies assumed a reactor exit coolant temperature of 1150 K, which is beyond the operating range of conventional reactor core materials. This temperature was selected to accommodate a radiator area that could fit within the fairing of an existing launch vehicle and provide a reasonble overall mass while allowing conventional materials in the Brayton engine and the primary plant piping and heat exchangers. However, temperatures in the reactor core would be much higher than coolant temperature. Therefore, non-conventional materials were under investigation for the reactor core, including refractory metal alloys and ceramics. The mechanical properties of these non-conventional materials were not well-characterized, particularly at the conditions envisioned for the JIMO mission. Developing well-qualified material properties to support a space reactor design effort requires significant lead time for testing and evaluation and would be a major challenge for the aggressive JIMO launch schedule.

The power level, reactor temperatures, and mission duration envisioned for JIMO led to focusing reactor design efforts on a small, fast neutron energy spectrum, external reflector-controlled core. A thermal spectrum core was not considered practical for the long Prometheus operating life because of: (1) practical difficulties of retaining hydrogen (in water or solid hydride) in a high temperature system (greater than 900 K), (2) the high thermal neutron absorption in many of the suitable high temperature structural materials, and (3) difficulties associated with controlling reactivity over core life. However, it is difficult to qualify materials performance for a fast neutron energy spectrum reactor because of the lack of nuclear testing and fast spectrum reactor infrastructure in the United States. The recovery and development of this capability will involve significant cost and require a long lead time. To address this shortcoming, the NRPCT investigated use of foreign test reactors including those in Japan (JOYO), Russia (BOR-60), and France (PHENIX). The complexities of international operations posed additional risk to the aggressive schedule associated with Project Prometheus.

The launch safety analysis for Prometheus reactor design efforts to date have included nominal core configurations with both dry sand reflection and flooded scenarios. However, launch accidents involving core re-entry and impact may result in a significant re-configuration of the core geometry, potentially into a much more reactive condition. Safety rod performance in such a re-configuration has not been determined. Developing a safety design basis and performing the complex mechanical and nuclear analyses and testing of a core re-configuration will be a significant challenge for continued development of a space reactor design.

For the remainder of the plant, the key challenges are the long mission duration, lack of auxiliary power, and low mass and volume constraints. These requirements can result in contradictory system design features, and important trade-offs would have been made which include aspects of redundancy, reliability, complexity, and mass. The candidate plant materials are better characterized than the reactor core materials but will still require significant testing at the JIMO plant conditions.

Another key challenge is overall system integration of the Reactor Module and integration of the Reactor Module with the spaceship. A number of scaled, non-nuclear integrated system tests would be needed to fully characterize the performance of the system and allow for component and system design optimization. The period of time to design, construct, and perform this series of tests would need to occur in parallel with materials testing, overall spaceship design, and ground test reactor (GTR) facility siting and construction. After this series of non-nuclear development tests, final testing of flight components would also be needed. This pushes the design schedule forward to facilitate flight unit integration and testing.

System integration requires close coordination with the NASA project lead (JPL) and the spaceship contractor (Northrop Grumman Space Technology (NGST)) to develop clear mission and component interface requirements and to ensure that the range of spaceship systems perform in an integrated manner. This would have required coordination of testing at DOE labs, at NGST and its vendors, at NRPCT and its vendors, and at NASA Centers. As an example, consider the power conditioning and distribution (PCAD) system. This was a system for which NGST had direct responsibility because it was part of the spaceship module. However, it interfaces with the energy conversion system and, therefore, influences reactor operation. Thus, the reactor system designer must be familiar with the interface requirements, operational characteristics and potential faults, and how the system is controlled by the space computer. The need to protect the reactor from some potential PCAD casualties may have a substantial influence on the PCAD design. Therefore, close coordination on system interface issues early in the program is a necessity.

## 1.5 Gas-cooled Reactor Brayton Concept Description

The gas-cooled reactor Brayton system is depicted in Figure 1-1. It uses a single gas-cooled reactor mounted at the forward end of the spaceship. An inert gas (a HeXe gas mixture was assumed for initial work) is used to cool the core and transport energy around or through a shadow shield to the Brayton energy conversion system. The reactor consists of a core with cylindrical fuel pin elements arranged within the core structure and a reactor vessel to direct coolant flow and provide structural support for the core and reactivity controls. Fixed and movable reflector segments are used to control the fission reaction rate by changing the fraction of neutron leakage that is reflected back in to the core. Removable, neutron-absorbing safety rods keep the reactor shutdown in the event of a launch or transport accident. Reactor control drive mechanisms are used to move the reflectors and safety rods. Gas flows directly over the fuel elements either in an open lattice array or through channels in a block into which the fuel pins are inserted.

The fuel element concept and the assembly of fuel elements within the core are shown in Figure 1-2. The elements consist of ceramic fuel pellets, a gas gap to accommodate swelling, a cladding liner to improve material compatibility, and the cladding which prevents fission gas escape. A fission gas plenum is typically situated at one end of each fuel element to accommodate the fission gas released from the pellets without producing excessive clad strain due to gas pressure accumulation. The elements are attached to the support structure at only one end to allow for differential growth between the fuel element and the structure. Several refractory metal alloys as well as silicon carbide were considered for the cladding. Several core configurations shown in Figure 1-3 were also being investigated prior to project termination. The annular flow block geometry and open lattice geometry were evaluated more than the other concepts prior to project restructuring. The annular flow block arrangement allowed for more controlled coolant distribution within the core but results in higher reactor masses as compared to the open lattice designs. The open lattice designs had the least mass but the lack of defined coolant channels makes it more challenging to control flow to specific regions of the core. Another option that was beginning to receive further evaluation at the time of project redirection was the option for a modular cermet design. The cermet fuel system consists of a refractory metal alloy matrix and small fuel particles. This approach eliminates the need for gas plena, improves conductive heat transfer, and could allow for the ability to control coolant flow to various regions of the core. However, the design is more aggressive than the other concepts considered and requires optimistic fabrication and performance design assumptions to be mass competitive.

Table 1-3 provides reactor core parameters for a number of the material and arrangement options investigated. The table compares reactor cases and their impact on mass and shielding required for simple changes in system pressure, reactor thermal power, core geometry, and cladding material. Additional information on all reactor options considered can be found in Reference (5).

The reactor vessel surrounds the core and a combination of fixed and movable reflectors surround the vessel. The reactor vessel is cooled by the incoming gas to maintain temperatures below material limits. The coolant temperature supplied to the reactor from the plant is limited depending primarily on the selection of vessel material, vessel heating rates due to gamma heating, and the configuration of structures around the vessel that may insulate the vessel. The movable reflector is segmented and used to control core reactivity to start up the reactor and maintain the desired operating temperature over life. Instrumentation is provided to monitor neutron flux, temperature, coolant pressure, and control position. Although the specific procedures for reactor control have not been determined, these measurements can be used during life to determine when to move the reflectors to compensate for uranium burn-up or to reduce reactivity during an unexpected transient condition. The reactor uses at least one safety shutdown rod to prevent inadvertent core criticality during manufacture, assembly, and transport and to prevent criticality in the event of certain accident conditions which can increase the reactivity of the core relative to the normal shutdown configuration. The safety rod(s) is required during core assembly, transport, and launch; it would be permanently withdrawn prior to startup in space, once a stable orbit was achieved. Table 1-4 provides a summary of material options and issues for major Reactor Module components.

The coolant temperature supplied to the plant from the reactor is limited to ~1150 K (1610 F) in order to allow the use of more conventional materials for the plant and energy conversion system and to reduce pressure loading on the fuel element cladding. Sample plant arrangements and heat balance results are shown in Figure 1-4 and Figure 1-5, respectively. The heat balance examples are based on the preliminary plant parameter evaluations at the time of project closeout; many additional heat balance examples are available in Reference (4). The hot gas expands through a turbine, which is connected via a common shaft to a compressor and an alternator. The alternator converts turbine power into electricity to be used by the on-board ion propulsion system, on-board computers, science and spaceship health monitoring instruments, and communications systems. After passing through the turbine, the gas passes through a regenerative heat exchanger (a recuperator) and a gas cooler. The cooled gas is then pumped back through the recuperator and to the core by the compressor. Heat is transferred from the gas cooler to the heat rejection system radiators via a pumped liquid loop (water or NaK). The high frequency, three-phase power coming from the Brayton alternator(s) is conditioned using the Power Conditioning and Distribution (PCAD) system to provide high voltage to the propulsion units and low voltage to the computers and instruments. Excess power not used by the propulsion system or the on-board electrical equipment is shed via a controllable Parasitic Load Radiator (PLR). The PLR is used as a variable load to compensate for changes in the spaceship electric load demand and to control the speed of the turbomachinery and resulting electric power frequency and voltage.

**Figure 1-1: Spaceship and Reactor Module**

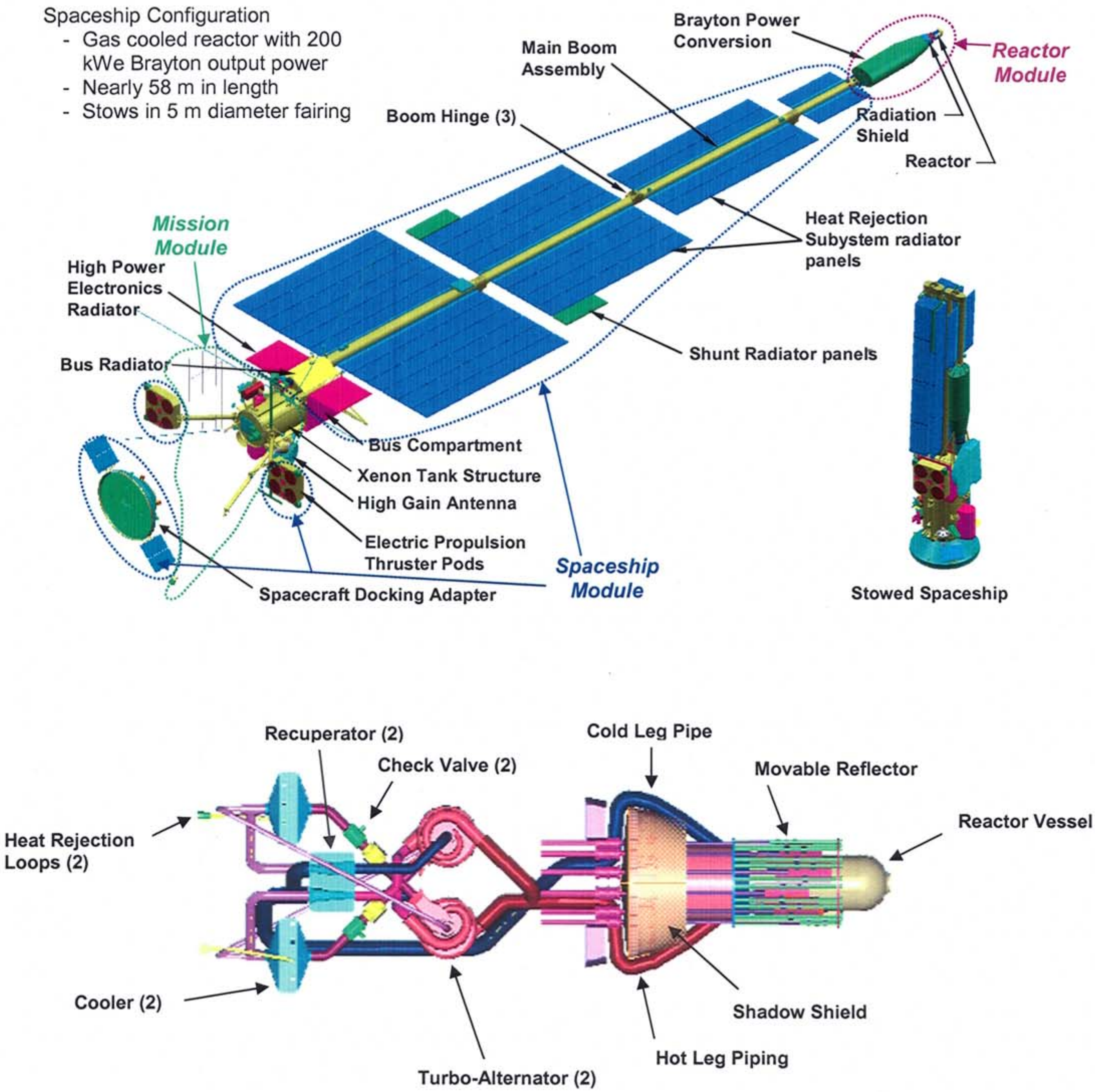


**Figure 1-2: Notional Core and Fuel Element Configurations**

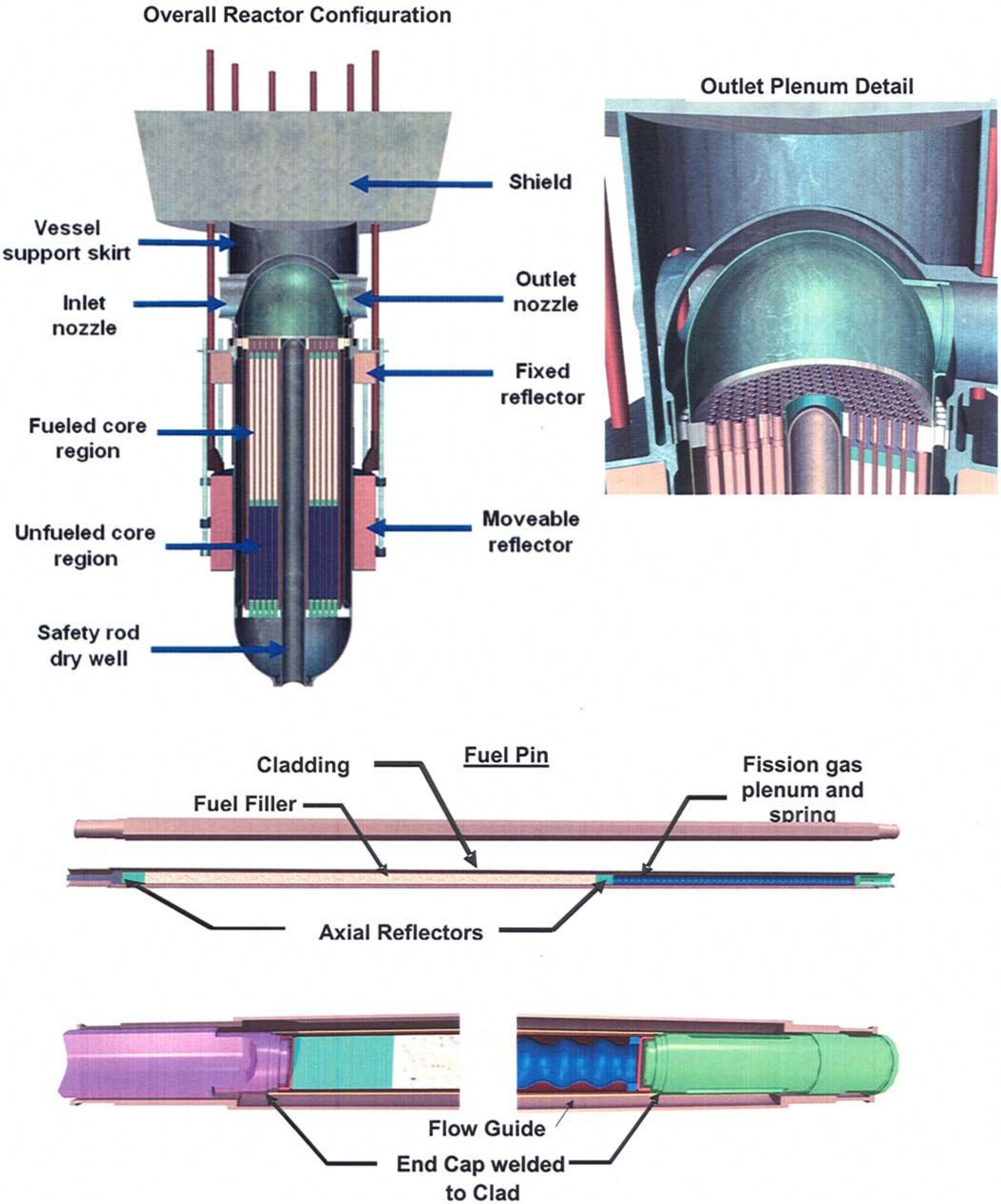
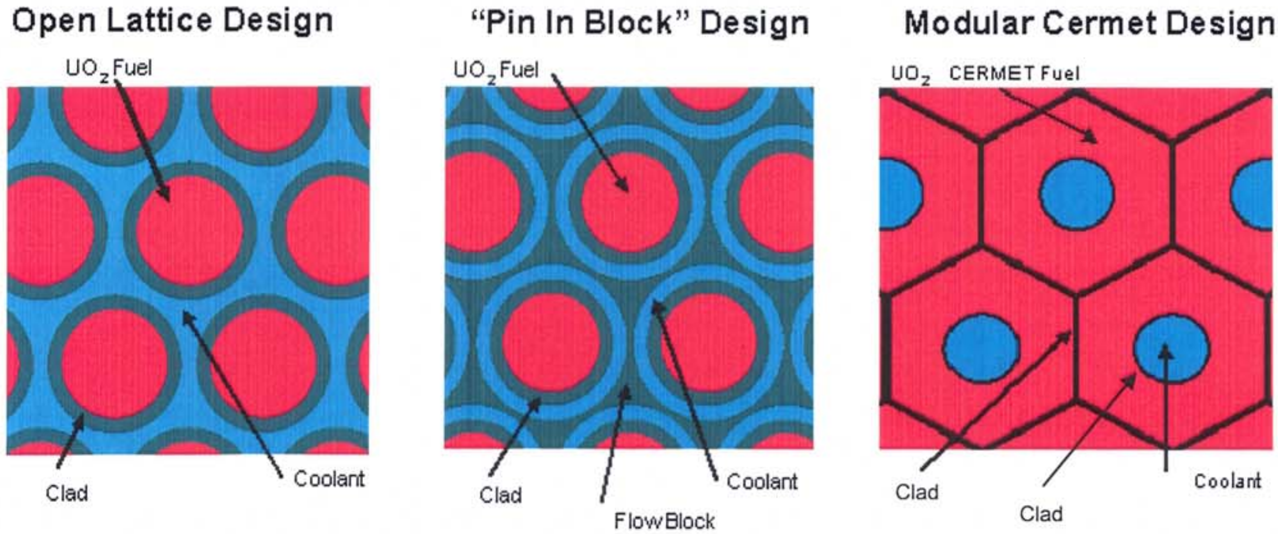


**Figure 1-3: Comparison of Primary Core Geometry Options**




**Significant Characteristics of the Open Lattice**

- The design provides the lowest mass geometry.
- Rod mechanical distortion could damage the fuel. A wire wrap or other mechanical support is needed to protect the fuel pins, although, at the expense of an increased core pressure drop.
- Initial assessments indicate axial flow will not result in flow-induced vibration issues, but testing is needed.
- The lack of defined coolant channels limits the ability to control flow to individual channels.
- The lack of a core block structure provides a limited heat sink and will likely result in the highest fuel temperature excursion during a transient.

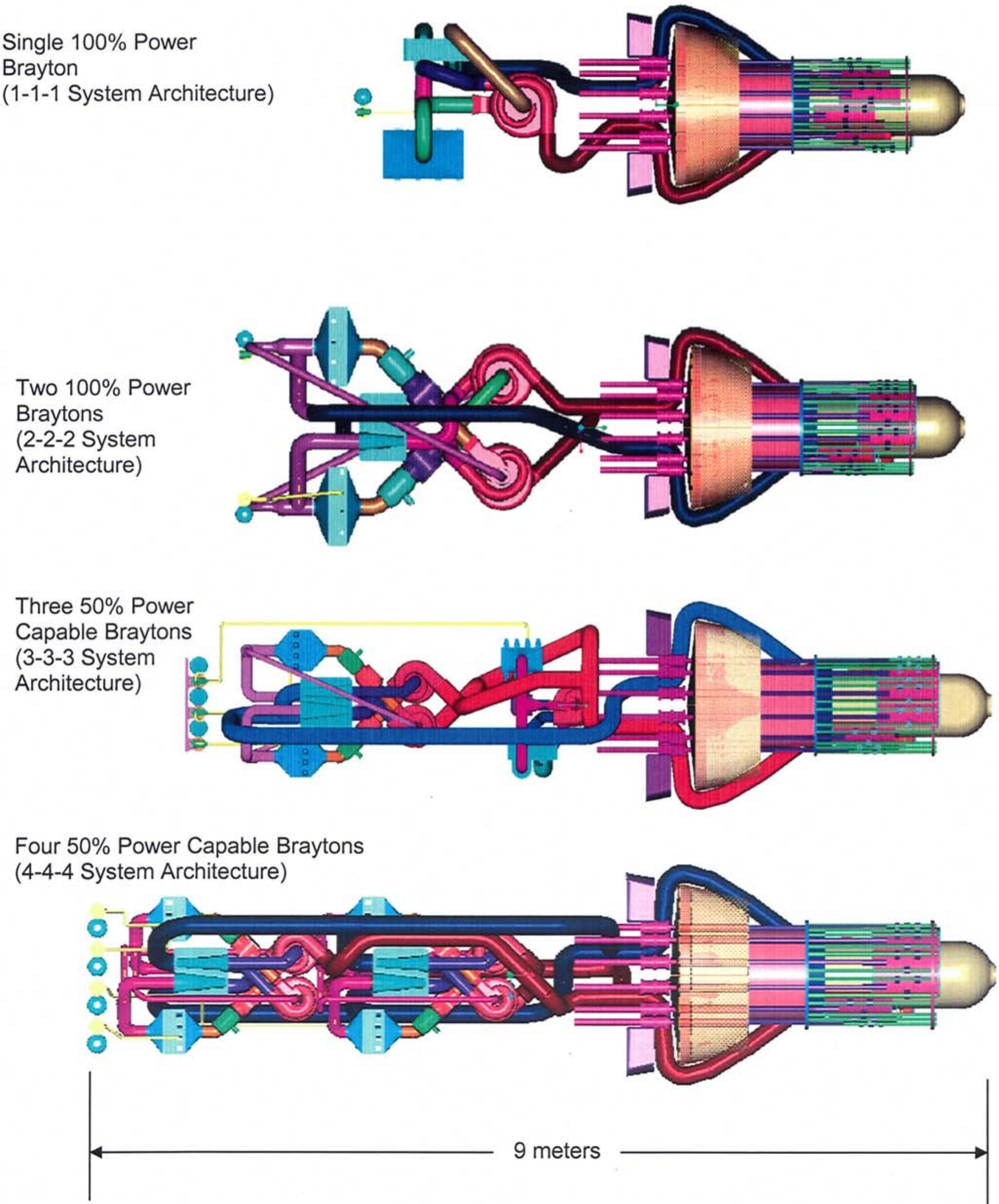
**Significant Characteristics of the Annular Flow "Pin in Block"**

- Flow is contained in each distinct channel, forcing coolant flow across each pin.
- Flow can be metered to each channel to achieve a uniform temperature rise up each channel.
- The geometry is heavy due to the mass needed for the block.
- The extra mass is separated from the fuel and is less of an effective heat sink in reducing transient fuel temperatures than designs where the fuel pin is in contact with the core block. The magnitude of this effect was not quantified at the time of project termination.
- The fuel pins need to be positioned in the channel with sufficient surrounding space to prevent overheating.
- Inlet orificing is used to distribute flow and balance the coolant temperature rise.

**Significant Characteristics of the Modular Cermet Design**

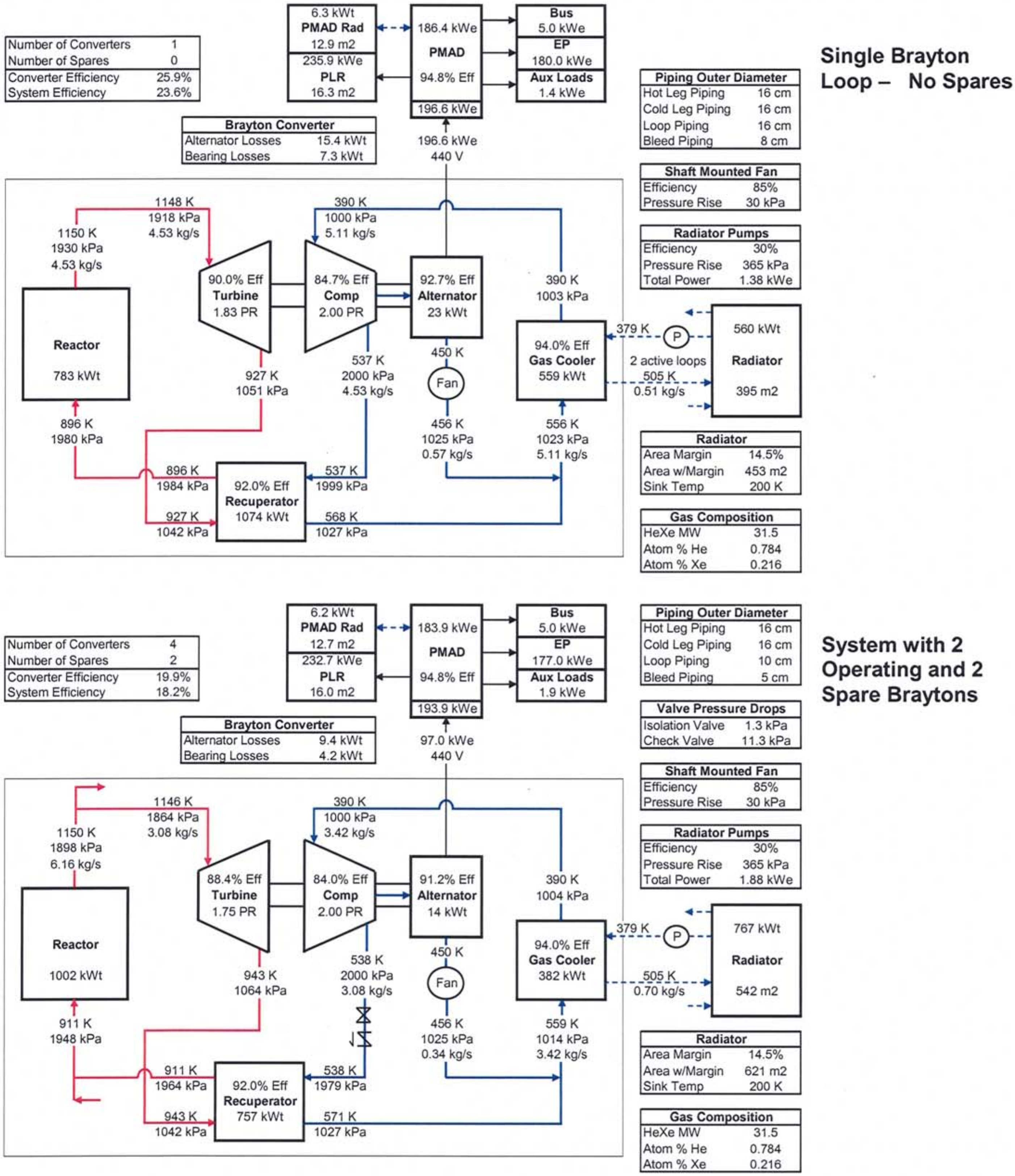
- Dedicated flow paths force coolant flow in each channel.
- Designed in modular sections for manufacturability.
- Most of the temperature drop is from the clad to coolant with no internal gap and high metal conductivity. Design is more sensitive to large uncertainties in the gas convective heat transfer coefficient.
- Concept requires optimistic fabrication and performance design assumptions to be mass competitive.
- The design considered has a very high ceramic volume fraction of 60%.
- The design provides the best use of structure to reduce transient fuel temperatures.

**Figure 1-4: Space Nuclear Power Plant System Arrangements**[2]




[2] The designation of 1-1-1 or B-R-G is B is the number of Brayton units, R is the number of recuperators sized to support the operation of one Brayton unit at 100% of rated power and G is the number of gas coolers sized to support the operation of one Brayton unit at 100% of rated power

## Figure 1-5: Example Heat Balances for Single and Multi-Brayton Architecture



## Table 1-3: Parameter List of a Subset of Reactor Case Studies

| General Parameters | Base Case | Low Power | High Pressure PCEH-129 | Open Lattice PCFD-474 | TZM | SiC |
|---|---|---|---|---|---|---|
| Power (MWth) | 1 | 0.5 | 1 | 1 | 1 | 1 |
| Full Power Years | 15 | 15 | 15 | 15 | 15 | 15 |
| Fuel Type | UO2 | UO2 | UO2 | UO2 | UO2 | UO2 |
| Fuel Form | Ceramic | Ceramic | Ceramic | Ceramic | Ceramic w/Euo3 Poison | Ceramic w/Euo3 Poison |
| Geometry | Modular Annular Flow Block | Modular Annular Flow Block | Modular Annular Flow Block | Open Lattice | Modular Annular Flow Block | Modular Annular Flow Block |
| Clad Material | Mo-47.5Re | Mo-47.5Re | Mo-47.5Re | Mo-47.5Re | TZM | SiC |
| Control Device | Sliders | Sliders | Sliders | Sliders | Sliders | Sliders |
| System Pressure (MPa) | 2 | 2 | 4 | 2 | 2 | 2 |
| Gas Composition (%He/%Xe) | 78/22 | 78/22 | 78/22 | 78/22 | 78/22 | 78/22 |
| Gas Composition (g/mol) | 31.5 | 31.5 | 31.5 | 31.5 | 31.5 | 31.5 |
| Vessel Material | Alloy-617 | Alloy-617 | Alloy-617 | Alloy-617 | Alloy-617 | Alloy-617 |
| Block Material | Mo-47.5Re | Mo-47.5Re | Mo-47.5Re | N/A | TZM | SiC |
| Shield Material | Be-B4C-W | Be-B4C-W | Be-B4C-W | Be-B4C-W | Be-B4C-W | Be-B4C-W |
| Tcold - Average @ nozzle B.E. (K) | 880 | 880 | 880 | 880 | 880 | 880 |
| Thot - Average @ nozzle B.E. (K) | 1150 | 1150 | 1150 | 1150 | 1150 | 1150 |
| **Dimensions** | | | | | | |
| Vessel Outside Diameter (cm) | 61.81 | 49.71 | 54.55 | 54.88 | 61.87 | 63.32 |
| Vessel Thickness (cm) | 0.48 | 0.38 | 0.84 | 0.43 | 0.48 | 0.49 |
| Vessel Length (cm) | 159.6 | 131.2 | 139.9 | 137.2 | 149.8 | 145.8 |
| Reflector Outside Diameter (cm) | 85.1 | 73.1 | 78.1 | 78.45 | 85.44 | 86.9 |
| Fuel Pellet OD (cm) | 1.819 | 2.238 | 1.72 | 1.476 | 1.862 | 2.057 |
| Gap Thickness (cm) | 0.022 | 0.024 | 0.022 | 0.021 | 0.022 | 0.032 |
| Clad Thickness (cm) | 0.051 | 0.051 | 0.051 | 0.135 | 0.051 | 0.102 |
| Fuel Pin OD (cm) | 1.965 | 2.388 | 1.866 | 1.788 | 2.008 | 2.325 |
| Fuel Pellet U235 Loading Density (g U235/cc) | 8.26 | 8.26 | 8.26 | 8.26 | 7.58 | 7.44 |
| Channel Thickness (MAFB) (cm) or Distance Between Pins (OL) (cm) | 0.216 | 0.188 | 0.135 | 0.145 | 0.213 | 0.234 |
| Pitch (cm) | 2.614 | 2.977 | 2.35 | 1.94 | 2.213 | 3.01 |
| Core Volume (L) | 135.6 | 79.48 | 96.53 | 88.0 | 137.1 | 145.9 |
| Number of Pins | 288 | 144 | 288 | 402 | 288 | 228 |
| Core Fuel Height (cm) | 60.8 | 55.5 | 53.3 | 49.8 | 59.9 | 61.0 |
| Gas Plenum Height (cm) | 31 | 20 | 26 | 26.5 | 22 | 15.5 |
| Number of Control Elements | 12 | 12 | 12 | 12 | 12 | 12 |
| Number of Safety Rods | 1 | 1 | 1 | 1 | 1 | 1 |
| Safety Rod Diameter (cm) | 12.72 | 9.52 | 11.00 | 11.98 | 12.95 | 15.17 |
| Shield Thickness (cm) | 66.03 | 66.71 | 67.39 | 68.07 | 66.17 | 65.91 |
| Shield Leading Edge Diameter [$D_{6degree}$] (cm) | 102.66 | 89.54 | 94.84 | 93.32 | 103.26 | 104.78 |
| Shield Cone Angle (degrees) | 6/12 | 6/12 | 6/12 | 6/12 | 6/12 | 6/12 |
| **Masses** | | | | | | |
| U235 Fuel Load (kg) | 376 | 256.8 | 294.9 | 283 | 356 | 344 |
| Reactor (kg) | 2078 | 1360 | 1731 | 1340 | 1793 | 1432 |
| Additional Reactor Components (kg) | 831 | 544 | 692 | 536 | 717 | 573 |
| Shield (kg) | 1648 | 1334 | 1520 | 1511 | 1665 | 1681 |
| Total Mass (Rx with Shield) (kg) | 4557 | 3238 | 3943 | 3387 | 4175 | 3686 |
| **Key Results** | | | | | | |
| Nuclear | | | | | | |
| Peak Burnup-B.E. (% FIMA) | 2.18 | 1.58 | 2.78 | 2.88 | 2.1 | 2.14 |
| Max Local Peaking Factor | 1.49 | 1.49 | 1.49 | 1.48 | 1.48 | 1.48 |
| Slider/Drum Worth - Most Reactive/Least Reactive Rod Out (Δρ) | 0.11 | 0.14 | 0.1 | 0.13 | 0.13 | 0.16 |
| Mechanical | | | | | | |
| Peak EOL Volumetric Fuel Swelling (%) | 4.9 | 4.1 | 4.8 | 5.1 | 4.8 | 3.8 |
| Metal - EOL Primary Membrane Von-Mises Clad Stress (MPa) | 21.8 | 25.2 | 32.4 | 13.7 | 44.4 | N/A |
| Primary Membrane Clad Creep Strain (%) | 1 | 1 | 1 | 1 | 1 | 1 |
| Primary Membrane Vessel Hoop Stress (Mpa) | 22.6 | 26.6 | 32.4 | 12.6 | 48.1 | 25.0 |
| Thermal Hydraulic | | | | | | |
| DPcore/Psystem (CORE ONLY) (%) | 1.12 | 1.06 | 1.10 | 0.99 | 1.01 | 1.00 |
| Max Surface Heat Flux (W/cm^2) | 16.6 | 14.9 | 19.9 | 15.8 | 16.3 | 17.5 |
| Max Linear Heat Generation Rate (W/cm) | 102.2 | 111.9 | 116.6 | 88.8 | 101.9 | 127.8 |
| In-Fuel Power Density Core Average (W/cc) | 26.4 | 19.1 | 33.7 | 35.1 | 25.6 | 26.0 |
| Peak Fuel Temp BOL - B.E. (K) | 1636 | 1637 | 1638 | 1622 | 1636 | 1646 |
| Peak Fuel Temp over Life - B.E. (K) | 1773 | 1771 | 1775 | 1775 | 1769 | 1720 |
| Peak Clad Temp over Life - B.E. (K) | 1275 | 1262 | 1232 | 1311 | 1274 | 1284 |

**Table 1-4: List of Materials Envisioned for Use in the Prometheus Program**

| Component | Material Option | Operating Condition | Developmental Concern |
|---|---|---|---|
| Fuel | $UO_2$ | 900-1773 K ~$10^{22}$ n/cm$^2$ | Swelling/Cracking at Low Fluence/Burn-up/burn-up rate, fission gas release rate uncertainty |
| | UN | | Fission Product Chemistry, fission gas release rate, porosity evolution |
| Fuel Cladding | Nb-1Zr | 900-1300 K ~$10^{22}$ n/cm$^2$ | Creep Capability, Radiation-Induced and Interstitial Embrittlement |
| | FS-85 | | Phase Stability, Radiation-Induced and Interstitial Embrittlement |
| | T-111 | | Phase Stability, Radiation-Induced and Interstitial Embrittlement |
| | Ta-10W | | Radiation-Induced and Interstitial Embrittlement |
| | ASTAR-811C | | Interstitial Embrittlement, Phase Stability, Fabricability |
| | Mo TZM | | Irradiation Embrittlement, Irradiated Creep Capability, Fabricability |
| | Mo-47Re | | Radiation-Induced Embrittlement, Phase Instability |
| | SiC/SiC | | Hermeticity, Fracture Toughness, Conductive Compliant Layer |
| Liner | Re, W, or W-Re | 900-1500 K ~$10^{22}$ n/cm$^2$ | Embrittlement, Hermeticity, Reaction with fuel/cladding, Neutron Poison |
| | None | | FP attack of cladding |
| Fuel Spring | W-25Re | 800-1300 K ~$10^{22}$ n/cm$^2$ | Radiation-Induced Embrittlement, Relaxation |
| | Ta alloys | | Radiation-Induced and Interstitial Embrittlement, Relaxation |
| In-Pin Axial Reflector | BeO | 900-1300 K ~$10^{22}$ n/cm$^2$ | Irradiation swelling, He Gas Release, $^6$Li Neutron Poisoning, BeO Handling Concerns |
| Core Block | Refractory Metal | 900-1200 K ~$10^{22}$ n/cm$^2$ | Fabricability, Neutron Absorption |
| | Graphite | | Fracture Toughness, C transport to refractory metal fuel |
| | Nickel Superalloy | | Irradiation Damage, C/O transport to refractory metal fuel |
| In-Core Structure | Refractory Alloys | 900-1200 K $10^{22}$ n/cm$^2$ | Fabricability, Radiation-Induced and Interstitial Embrittlement |
| Reactor Vessel | Nimonic PE-16 | Up to 900 K $10^{21}$ n/cm$^2$ | Radiation-Induced Embrittlement, Creep Capability |
| | Alloy 617 | | |
| | Haynes 230 | | |
| Safety Rod Thimble (if used) | Same as Vessel | Up to 1050 K $10^{22}$ n/cm2 | Irradiation Embrittlement, Creep Capability |
| | Refractory Metal | | Irradiation Embrittlement, Creep, Dissimilar Material Joining |
| Radial Reflector | BeO | Up to 900 K $10^{21}$ n/cm$^2$ | Irradiation Swelling and He Gas Release, $^6$Li poisoning, Be/BeO Handling Restrictions |
| | Be | | |
| Shielding | Water | Up to 500 K | Thermal Management |
| | Be | Up to 800 K | Be Handling Restrictions during manufacturing |
| | $B_4C$ | | |
| | LiH | | Neutron and gamma swelling vs. temp. and irradiation |
| Shielding and Reflector Canning | Steel or Ni Superalloy | Same Range as shielding | |
| | Titanium Alloy | | |
| Loop Piping | Alloy 617 | 300-900 K | Maintenance of internal insulation @ 900 K, Joining |
| | Haynes 230 | | Maintenance of internal insulation @ 900 K, Joining |
| Insulation | Porous Metal or ceramic | Up to 1150 K | Thermal conductivity, Loop Material Compatibility |
| | Ceramic Fiber | | Thermal conductivity, Loop Material Compatibility |
| Insulation Liner | Mo Alloy | Up to 1150 K | Fabricability, Compatibility with insulation, embrittlement |
| | Superalloy | | Compatibility with insulation |
| Turbine Casing (scroll) | In-792 | Up to 1150 K | Creep Capability, Dissimilar Materials Joining (to piping) |
| | Mar-M-247 | | |
| | Alloy 617 or Haynes 230 | Up to 900 K | Requires internal insulation |
| Turbine Wheel | In-792 | Up to 950 K | Creep capability, Carburization/Decarburization/Deoxidation |
| | Mar-M-247 | | |
| Compressor | Ti-Al-V | 400-600 K | Compatibility w/ gas loop |
| | Superalloy | | |
| Shaft | 1018 Steel | 400-900 K | |
| | Superalloy | | |
| Alternator Magnets | Sm-Co | 400-450 K | Loss of magnet strength, compatibility with gas loop |
| Electrical Insulators | Ceramic or Glass | 400-450 K | Hermeticity, compatibility with gas loop |
| Recuperator Core | Alloy 625/690 | 600-900K | Thermal Stability at Hot Side Temp, Braze Material concerns |
| | Carbon/Carbon | | Compatibility with other loop components (C transport), Fabricability |
| Cooler Core | CP Titanium | 400-550 K | Compatibility with gas and water loops |
| | Alloy 625/690 | | |

# 2 REACTOR SUMMARY

## 2.1 Issues and Challenges

As discussed in Section 1.4, the Prometheus mission requirements result in key design issues for the reactor. The electrical power requirement of ~200 kWe, coupled with a goal to minimize mass, results in a reactor exit coolant temperature of ~1150 K and a thermal power rating of ~1 MWth. The Prometheus reactor design effort was focused on a small, fast neutron energy spectrum, external reflector-controlled core as the most practical way to meet mission objectives. The major issues identified for the reactor design based on the NRPCT work performed to-date are discussed below.

Material Data and Design Bases

Most candidate reactor materials being considered (refractory metal alloys, nickel-base super alloys, and silicon carbide) were not well characterized for Prometheus reactor operating conditions at the onset of the project. NRPCT did begin early to compile available material property data; however, there were sparse data applicable to Prometheus conditions (e.g., temperature range, vacuum environment, exposure to He-Xe coolant, irradiation environment, etc.). Extensive material test programs would have been required to provide data on the wide range of properties required for design and in sufficient quantities to develop statistically satisfactory databases that provide assurance of achieving mission objectives. These test programs were being formulated and initiated early in the program. However, despite these early efforts, it was clear that the design had to proceed in parallel with material property development to achieve the program schedules. In addition to establishing a consistent set of design bases for concept comparison, there was also concern regarding the variation of confidence associated with the assumed properties for the materials under consideration.

It became evident that the lack of significant material information at the prototypical conditions, coupled with the complex design space evaluations to date, would result in increased reactor performance uncertainty. The project schedule was driving decisions before significant material information was available and before consistent design assessments could be adequately completed. While the necessary decisions would have been made, the lack of available information creates risk to the project schedule if initial judgements are shown to be wrong in subsequent testing. The risk can be mitigated by selecting materials that are less developmental, although it was found that these generally led to a more massive reactor. The characterization of the candidate materials and efforts associated with establishing common concept comparisons must be addressed early and factored into any realistic schedule to support a similar endeavor.

Testing and Testing Infrastructure

The lack of material data and information was further complicated by the lack of domestic fast reactor testing infrastructure needed to test core materials, as discussed in Volume 1, Section 3.3. There are no operating fast neutron flux test reactors in the U.S., and the NRPCT was investigating available foreign test reactors. Material irradiation testing at the JOYO facility in Japan was being pursued when the project terminated. The complexities of international operations posed additional risk to the aggressive schedule associated with Project Prometheus.

Plans for critical physics experiments were in a similar situation in that no adequate facility is currently operating within the U.S., and startup of a new or previously shut-down facility was potentially very costly and/or had significant schedule disadvantages. There are currently no operating domestic critical experiment facilities to support physics testing using more than 50 kilograms of highly enriched uranium (HEU). Two types of critical experiments were anticipated: clean benchmark experiments to qualify neutron cross sections and engineering mockups to qualify the nuclear design and nuclear design methods.

The Los Alamos National Laboratory (LANL) Critical Experiment Facility (CEF) in Test Area 18 (TA-18) was initially selected to perform a series of clean benchmark critical experiments on core structural materials. These experiments would be used to establish an initial estimate of the uncertainty in the nuclear calculations that underly all core design performance and safety estimates. One experiment was performed before the facility was shut down in July 2004; LANL was not able to resume critical testing before the termination of Project Prometheus work. At the time of project termination, domestic capability (i.e., operation of the new Critical Experiment Facility in the Device Assembly Facility (DAF) at the Nevada Test Site) to perform critical experiments with large quantities of fissile uranium fuel was not expected to support the JIMO launch schedule. The only other known operating facilities of this type are in Russia.

The Zero Power Physics Reactor (ZPPR) at the Idaho National Laboratory (INL) was previously used to assemble engineering mockups for fast reactors. The ZPPR facility was placed in a cold standby status in the 1990s, and the projected cost to restore it to operation was very expensive. Similar facilities exist in Japan, France, and Russia but were not evaluated before the project was terminated.

<u>Reactor Safety Design Basis</u>

Reactor safety and reactor fuel safeguards need to be an integral part of the reactor development process. Designing the core to ensure public safety during all phases of assembly, transport, and launch, including potential launch casualties, was a key part of the NRPCT development strategy. A key reactor safety challenge is to ensure public safety in the event of an impact following a launch accident. Preliminary structural dynamic evaluation indicates that all of the envisioned core arrangements will substantially reconfigure upon impact. Considerably more modeling and testing would be needed to fully understand the potential of criticality and energy release during impact as well as to evaluate potential design features to ensure safety. Integration of dynamic core deformation models with dynamic reactor physics calculations was being initiated at the time of project closeout.

## 2.2 Summary of Work

The primary focus of the reactor design efforts following the selection of the direct gas Brayton system included:

1. Establishing reactor design bases to facilitate consistency among the concepts under consideration.
2. Providing the basic definition of the reactor design, culminating in a Pre-Conceptual Design Report [Reference (5)] to support material development, manufacturing development, and overall system conceptual design.

Initial versions of reactor pre-conceptual design bases were submitted to Naval Reactors for approval in June 2005. These design bases began to lay the ground work for what would have become the detailed documentation of the basis and reasoning behind the various aspects of the space reactor design. In parallel with the design basis efforts, the NRPCT began evaluating the reactor design space. The goal of the reactor Pre-Conceptual Design Report was to provide definition for the reactor to support overall plant optimization studies, test development, material procurement, and manufacturing development. The report documents the results of the design studies and key conclusions and insights achieved at the termination of the NRPCT design effort. The report also describes key aspects of the Prometheus reactor pre-conceptual design studies, including:

- Reactor core material and geometry concepts
- Core reactivity control concepts
- Pressure vessel concepts
- Reactor operation assessments
- Miscellaneous design-related considerations, including:
  - Core manufacturing
  - Launch safety
  - Assembly, test, and launch operations
  - Reactor design qualification testing

A key goal of the pre-conceptual design phase was to select the reactor fuel type. This early decision was needed to avoid the additional costs associated with development of multiple fuel types over an extended period of time. Additionally, it was believed that the fuel type recommendation could be made somewhat independently of the clad/liner material recommendation. Therefore, in July 2005 the NRPCT recommended uranium dioxide ($UO_2$) as the fuel for the reactor design vice the alternative uranium mononitride (UN) fuel Reference (6). A discussion of the advantages and disadvantages of each fuel type can be found in Section 3.1.

The use of $UO_2$ results in a 300 to 500 kg mass penalty on the Reactor Module relative to similar UN concepts. The selection of the higher mass option was based on greater confidence in successfully delivering a reactor with $UO_2$ in the available time frame before the launch of JIMO. There was insufficient confidence that the issues associated with UN fuel could be resolved to support a 15-year mission, as discussed further in Section 3.1.3. UN fuel was previously proposed for the SP100 space reactor program in the late 1980s and early 1990s. While the SP-100 design had similar burnup and temperature performance goals, it had a shorter target mission duration (7 vs. 15 years) and short fuel specimen test duration (2 years). The SP-100 UN fuel system was never fully developed. There is a much more extensive operational database for $UO_2$ than for UN, although the data were obtained at non-prototypical depletion rates and temperature gradients. Subsequent to this decision, Prometheus reactor design efforts focused on the gas reactor core alternatives with the $UO_2$ fuel system.

The NRPCT core design space evaluation included assessments of a variety of reactor clad and liner materials, a number of different core geometries, two basic fuel forms, and various poisons. Core concepts were developed for each configuration and analyzed to ensure that they would meet nuclear lifetime requirements, have acceptable control swing and temperature coefficients, provide safe shutdown while flooded or surrounded by sand, provide adequate cooling to maintain fuel element surface and centerline temperatures within limits, and provide adequately low core pressure drop. External reactivity control device evaluations focused primarily on the comparison of axial sliders versus rotating drums but also included evaluations of other control device options. Evaluations for the pressure vessel and core structurals focused on assessing whether it was feasible to use conventional materials or whether refractory materials would be necessary for the design. While the primary focus of the design evaluations was on fast neutron spectrum reactors, there were also some limited evaluations performed on moderated core concepts. A description of the reactor configuration is provided in Section 1.5.

<u>Reactor Materials Alternatives</u>

The required core operating temperatures determined which materials were considered for the reactor design. While nickel-based superalloy materials can be used in the lower temperature regions of the reactor plant, such as the pressure vessel, refractory metal alloys and ceramic SiC were considered for application in the higher temperature regions of the reactor due to their strength and high temperature capability. The high temperature regions include the fuel pin liner, fuel element clad, and the core structure in block geometry concepts. A detailed discussion on reactor core material issues can be found in Section 3.2. The discussion that follows describes the use of these materials in the core design.

The refractory metals considered for use include molybdenum (Mo), tungsten (W), rhenium (Re), and a variety of alloys containing these metals or other alloys based on niobium (Nb) or tantalum (Ta). In addition to high strength, some refractory metal alloys absorb thermal neutrons and help keep the core shutdown in accident scenarios that involve flooding and surrounding the reactor with water and/or wet sand. When the fast neutron spectrum core is flooded or reflected, neutrons do not leak out of the reactor as readily and reach lower energy levels via scattering with hydrogen and oxygen atoms. This would increase the core reactivity level because the more neutrons are available to fission with U-235 and the probability of fission increases by orders of magnitude. As a result, the core could become supercritical with core flooding and/or burial upon earth impact following a launch accident. Some of the refractory metal alloys exhibit a "spectral shift" poison characteristic in which the parasitic neutron absorption cross section is relatively low at high neutron energies (e.g., during normal operation), but is much larger at lower neutron energies (e.g., for a flooded, moderated core during a launch accident). This increased neutron absorption plays an important role in meeting the accident flooded shutdown reactivity requirements for reactor cores designed with refractory metal alloys. However, it was determined that the same effect could also be achieved by dispersing rare earth poisons within the fuel system. The use of the dispersed poisons can enhance the optimization of the reactor design since the added poison can be set solely to meet shutdown requirements, unlike some refractory metal alloy concepts where mechanical performance limits may set the amount of metal and the attendant spectral shift poison exceeds shutdown requirements.

While refractory metal alloys were being considered due to their high temperature capability, there were some significant materials issues to be addressed. Among these were irradiation embrittlement, interstitial embrittlement from trace amounts of carbon, oxygen, and nitrogen transferred by the coolant, fabricability, and thermal stability of the microstructure. Past testing has indicated that chemical compatibility with fission products is generally good for pure Mo and W. However, some alloying constituents and base metals can be reactive with fission products or fuel, requiring engineered liners for implementation. The use of a liner between the fuel and clad results in a more complex fuel system that is potentially more difficult to manufacture and may be susceptible to liner failure. The liner can provide additional benefits to the reactor design such as providing spectral shift poison depending on the liner material that is selected.

Silicon carbide (SiC) was assessed as a core structural material alternative to refractory metal alloys. SiC is a low-density material with high temperature capability. This provides the potential for reactor mass savings, especially in concepts that contain a significant amount of core structural volume. Silicon carbide is very resistant to corrosion and is stable in a radiation environment. It is also expected to be compatible with the nickel-base superalloy materials envisioned for the reactor vessel and the remainder of the reactor plant. Silicon carbide has very low neutron absorption cross sections and, therefore does not exhibit the spectral shift poison characteristic of refractory metal alloys. Thus, neutron poisons must be added to the design to help meet launch accident shutdown requirements.

Two separate forms of silicon carbide were considered for use in the Prometheus reactor. The monolithic form can provide a hermetic seal to retain fission gases, but is a brittle material and design studies to assess its mechanical performance capabilities require a statistical design basis in lieu of a more traditional deterministic design basis. The composite form is high strength and not considered brittle, but the failure modes and methods required for structural evaluation are outside the experience base of the NRPCT and were still being researched at the time of termination of NRPCT involvement. In addition, it may not provide a hermetic seal. For the fuel pins in the concepts developed, monolithic and composite silicon carbide are used together to form a "duplex" clad structure to provide both hermeticity (monolithic) and strength (composite). A compliant layer of a material such as crushable sub-dense graphite may be placed between the fuel and cladding to allow for fuel swelling while still providing adequate conductivity.

While there are advantages associated with the use of silicon carbide in reactor designs, there are engineering development issues. For silicon carbide, mechanical design performance and associated analysis methods as well as fabrication and joining issues are the expected engineering challenges. Additionally, silicon carbide is susceptible to oxidation and chemical attack from some fission products (palladium) and may require a barrier layer between the fuel and clad. The use of a compliant layer and/or barrier layer between the fuel and clad results in a more complex fuel system that is potentially more difficult to manufacture and may be susceptible to barrier layer failure.

Reactor Geometry Concepts

Several core geometry concepts were assessed during the pre-conceptual design phase. Each has advantages and disadvantages. The primary concepts evaluated are shown in Figure 1-3. Two of these geometries, annular flow block and open lattice, tend to bound the design capability of the geometry alternatives that were evaluated. Additional geometries such as a modular cermet block core, a plate core, and composite block core are discussed in more detail in the Pre-Conceptual Design Report.

The open lattice configuration with open flow channels between fuel rods has the least amount of structure within the core. This results in low overall reactor mass and low flow resistance. However, this design may be more susceptible to flow induced vibration, increased uncertainty in flow distribution, and mechanical distortion of the fuel.

The annular flow block configuration places each fuel pin within a flow guide block structure. This provides a defined flow annulus around each fuel pin and a high area fraction (and volume) of structural material inside the core. The material area fraction and volume in the annular flow geometry becomes important in the design because of the extra mass and neutron absorption of this material. This design is potentially the most mechanically robust and has the least uncertainty with respect to cooling due to the well-defined flow channels that can be orificed to control flow.

The selection and optimization of reactor geometry requires a tradeoff between competing requirements for thermal, mechanical, safety, and nuclear performance. In addition, many factors such as manufacturing and design uncertainty also need to be considered.

Reactor Safety

Design of the Prometheus reactor and safety systems is crucial to ensure that the public and environment are protected from undue risk. Launch safety is of special concern and must be assured over the period from installation of the SNPP into the launch fairing to launch into a stable orbit and depending on mission specifics, any action that could result in the Reactor Module returning to Earth.

In addition to the environmental concern regarding dispersal of toxic materials in the event of a launch accident, there are two key nuclear safety challenges: (1) safeguarding the highly enriched uranium fuel in the event of re-entry following launch and (2) preventing an inadvertent criticality during a launch accident. An aeroshell was expected to be used to limit the velocity of impact and to keep the core intact and recoverable in the event of a launch accident.

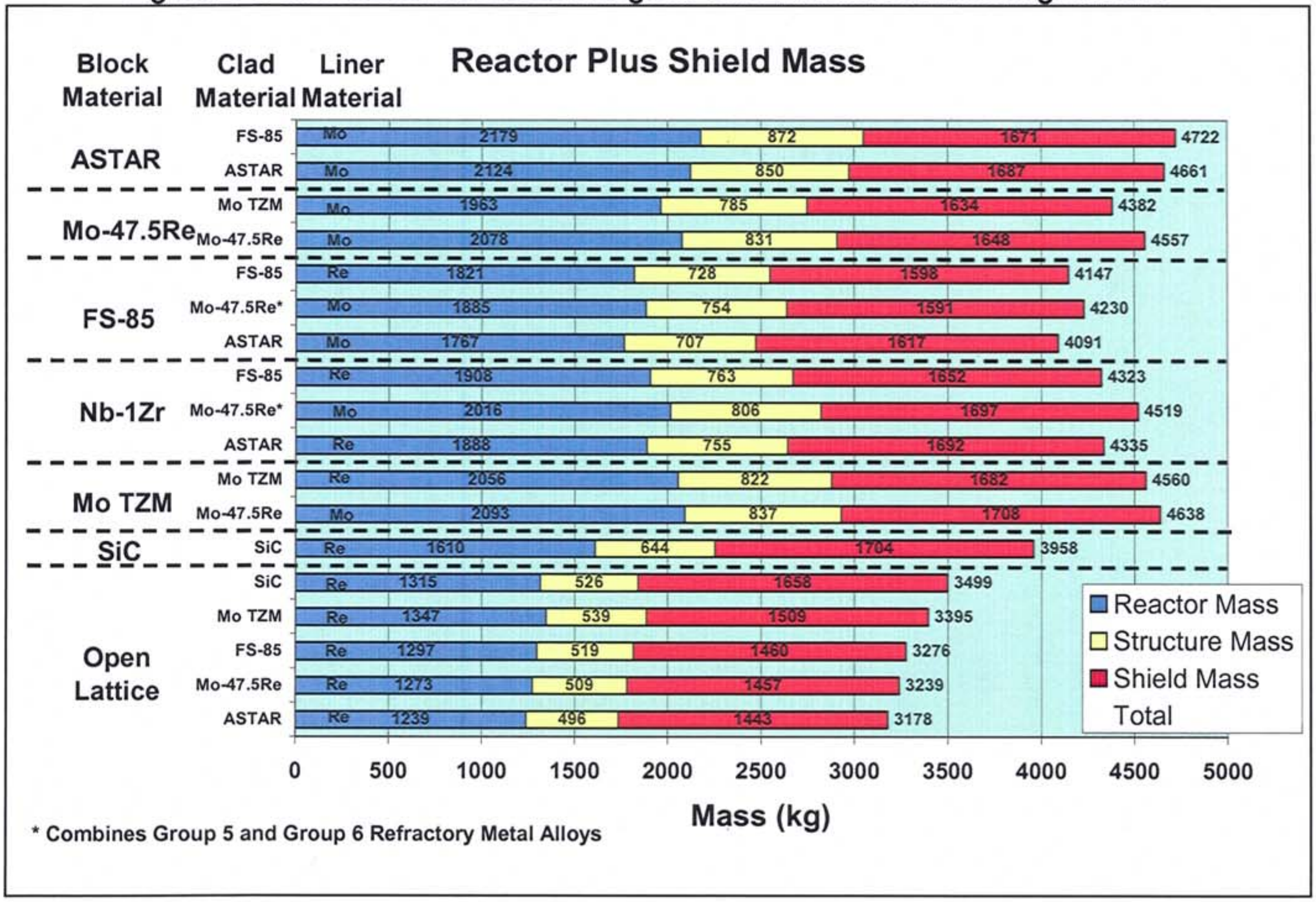
The core concept evaluations included requirements to demonstrate shutdown reactivity for both a fully water flooded and a dry sand-reflected reactor to ensure that the reactor remains subcritical before its intended startup. The shutdown requirements at this early stage of the analysis were based on past practices and engineering judgement. Additional effort was planned to define the shutdown requirements. Several engineering features were included in the design to ensure that the reactor remained safely shutdown, including spectral shift neutron absorbers and removable safety control rods. Preliminary mechanical assessments indicated that the core geometry would reconfigure during reentry and impact following a launch failure. The effects of core reconfiguration had not yet been accounted for in the core design when the project was terminated. An experimental program was envisioned to obtain data that could be used to more rigorously determine reactor performance characteristics.

## 2.3 Key Findings and Perspectives

The Reactor Pre-Conceptual Design Report [Reference (5)] summarizes the results of the NRPCT reactor engineering efforts in progress at the termination of Naval Reactors work on Project Prometheus. This report does not provide a single, recommended material or design. Results are provided, observations are stated, and issues are highlighted for future designers that may undertake a design similar to the ones evaluated.

Of the materials and design options evaluated, there was no clear and definitive choice at the time of project termination. The variety of primary options that were evaluated is shown in Figure 2-1. As shown in the figure, the reactor and shield mass ranged from 3000 kg to 5000 kg. With additional design optimization and material data, it is expected that a viable concept could be selected and would result in a reactor and shield mass near the mid-point of this range. All of the options had areas of promise and challenges with varying degrees of development and cost. A reasonable design space appears to exist with a wide range of potentially viable concepts if the design basis material properties are validated. However, not all options have equal prospect to produce a working reactor, and judgments need to be made to balance cost, risk, and potential performance to select a pre-conceptual design for further development.

**Figure 2-1: Mass Estimates for a Range of Core Materials and Configurations**




High level observations with respect to the reactor concept design space evaluations include:

- All concepts identified by NRPCT were preliminary and would require further evolution to become a workable, long-lived core design that would satisfy the Prometheus mission requirements (10 to 20 years at 200 kWe). Satisfactory materials, concepts, and engineering methods that met all design objectives had not yet been identified for all reactor components. The JIMO core operating conditions would have placed unprecedented demands on the nuclear fuel system. None of the fuel systems identified has operated at conditions and lifetimes comparable to those envisioned for JIMO. The plan to develop, test, and deliver a reactor to confidently provide the JIMO power, energy, and calendar life requirements for launch by 2015 as envisioned by NASA was a success-based plan. A significant setback in siting test facilities, completing Environmental Impact Statement requirements, materials performance, etc., could have impacted schedules, operation, or performance of the plant.

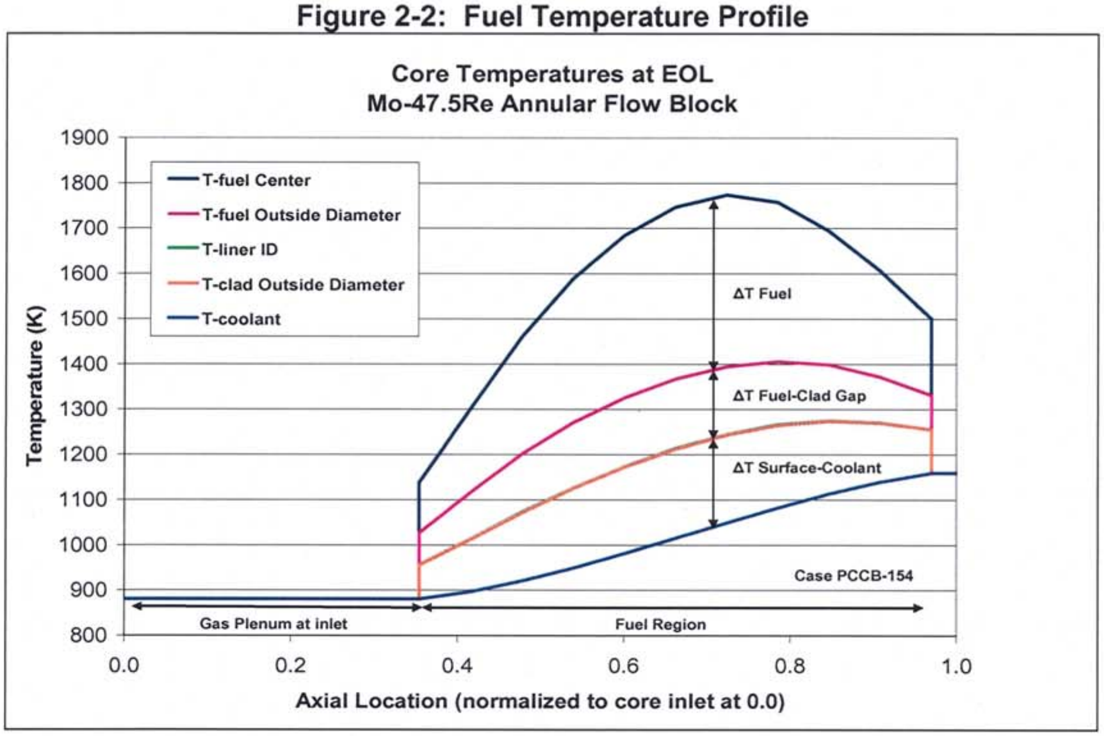
- Analytically, all of the fuel element, material, and geometry options have comparable performance characteristics, and each contains specific benefits and challenges. For example, the open lattice design concepts were typically lower mass than annular flow block concepts; the open lattice concept also has relatively more thermal hydraulic performance uncertainty and potential for flow induced vibration. Also, designs that used rare earth poisons mixed directly with the fuel are typically lighter than concepts that use a rhenium cladding liner for the same purpose; however, the use of the rare earth poisons at the levels assumed is developmental. Overall, distinctions between concepts would have had to have been drawn based on judgments about the feasibility and potential for actual performance. Data to characterize concept feasibility and performance are limited.

- The mass of the reactor and reactor radiation shield combination is very sensitive to the overall dimensions of each reactor concept (diameter, length), the configuration of the fixed reflectors and control devices (drums vs. sliders), and the proximity of the core to the shield. These factors may differentiate concepts in mass as much as variations in the mass of the reactors alone.

- No final decision was made regarding segmented, axially translating reflector sections ("sliders") versus rotating drums for reactivity control. Sliders offer a lower mass system and closer to a linear differential control worth. Control drums offer a simpler and more rugged control mechanism design, but have reduced control worth relative to sliders. Most of the pre-conceptual reactor nuclear and mechanical design work assumed sliders.

- Reactor thermal hydraulic performance is challenging due to the need to maintain a low reactor core pressure drop to maximize energy conversion system performance and to reduce the uncertainties associated with the gas convective heat transfer coefficient. The heat transfer coefficient uncertainty associated with the He-Xe mixture has a potentially large impact on the fuel element temperatures. The temperature rise in the coolant and the resultant metal temperatures are shown in Figure 2-2. The temperatures shown are for a refractory metal alloy Annular Flow Block (AFB) design targeting a peak fuel temperature of 1773 K. The low heat transfer coefficient for HeXe coolant results in a 200 K temperature rise from the coolant to the clad for typical designs. Single element and Reactor Module array tests were planned to better understand the gas convective heat transfer coefficient.

**Figure 2-2: Fuel Temperature Profile**




- The reactivity behavior of fast spectrum reactors is more sensitive to core mechanical distortions than in conventional light water reactors. Reactivity feedback characteristics are difficult to determine for fast reactor systems since they depend mostly upon subtle and small changes in the geometry of the reactor structure due to thermal expansion. Detailed and linked calculations would be needed to couple thermal, structural, and nuclear reactivity effects. These calculations would need to consider dynamic temperature distributions to account for the time lag in the reflector structure temperatures responding to reactor temperature changes. Sufficient work was not completed to adequately assess if any concepts under consideration possessed an advantage with respect to the reactivity feedback characteristics and reactor control.

- The reactor pressure vessel was assumed to be constructed of a nickel-base superalloy to mitigate compatibility issues associated with the anticipated energy conversion system materials. Sufficient work was not completed to assess whether a super-alloy would be a satisfactory pressure vessel material under the design conditions.

- A key core mechanical design hurdle is incorporation of one or more drywells for safety rods. A central drywell may be difficult to cool and may therefore have a higher temperature than the reactor vessel. In addition, the drywell has high fluence due to its proximity to the fuel. These issues increase the challenge associated with using nickel super-alloys throughout the reactor pressure boundary. In conjunction with further development of the thimble design, alternatives such as in-vessel safety control devices (eliminating the need for drywells) were being considered.

- All of the options considered have varying degrees of applicability to different missions dependent on the energy, power, lifetime, and other mission parameters. Since the project remains in the initial stages of development, a re-evaluation of the technology choices would be prudent for any significant mission changes. This re-evaluation would include reassessing the decision on the optimum reactor and energy conversion combination. The choice would be specific to the mission requirements including power level, operational requirements, cost, and schedule.

# 3 MATERIALS SUMMARY

Development, qualification and delivery of materials to support a Prometheus reactor plant were considered the most difficult technical and programmatic challenges for the Reactor Module. Generally, few materials were previously qualified in the Prometheus design space, necessitating wide ranging materials testing, analysis and modeling. Although not the only difficult testing required, irradiated materials testing was likely the longest lead time and most manpower and cost intensive area of materials development. Fuel, core structural, reflector, and shield materials all required some irradiated materials testing. Early decisions were necessary to limit the scope of the testing to be achievable despite very limited data in many cases. Other types of challenging tests were needed, including tests of the chemical compatibility of different materials in contact with the same coolant flowstream. Each of these tests could produce results that would eliminate the material from further consideration or significantly impact the design.

Generally, the materials planned to be used in a Prometheus application were to be subjected to: (1) long time (up to 20 years) at high temperatures; (2) moderate to high neutron fluences ($10^{21}$ - $10^{22}$ n/cm$^2$ at energies greater than 0.1MeV) for materials in front of the shield; (3) chemical attack from other plant materials; and (4) no ability to inspect or repair plant components during the mission. In addition, material choices must be made that limit SNPP mass.

Key NRPCT documentation on materials can be found in References (21) through (58). Final summary reports for Reactor Module materials investigations were generated for fuel materials [Reference (36)], structural materials [Reference (21)], shield materials [Reference (35)], and reflector materials [Reference (33)].

## 3.1 Fuel Materials

A dense, compact fuel system facilitates a smaller overall reactor. Fission products must be retained or managed such that there are no adverse chemical or radiological impacts on plant materials and electronics. Minimal fuel reconfiguration (e.g., the shifting of cracked fuel pellets or excessive irradiation swelling) is desirable because of potential associated effects on core reactivity, mechanical performance, and heat transfer. In addition, the fuel element components must be chemically compatible with each other and fission products. Behaviors important for reliable operation of a fuel system include the following:

- Gas release and swelling- gaseous and solid atoms are created during the fission process, increasing the volume of the fuel and pressure within a fuel element
- Total burnup- higher burnup means more fission products (solid and gaseous) and more potential for damage to fuel and clad materials
- Fission rate- higher fission rates increase the fuel operational temperature along with the mobility of atoms within the fuel, resulting in increases in fission product escape and chemical activity leading to corrosion
- Temperature and temperature gradient- larger temperature gradients enhance fission product diffusion leading to escape from the fuel and possible interaction with the liner/clad, along with increased stresses in the fuel that could enhance cracking
- Restructuring- primarily a function of temperature and fission rate, is when the fuel microstructure changes, reducing thermal conductivity, and releasing fission products
- Chemical compatibility of the system alone and with fission products- high operational temperatures along with fission product chemical activity can lead to fuel element corrosion resulting in cladding breach and release of fission products into the coolant
- Fabricability- an economical, repeatable process is desired

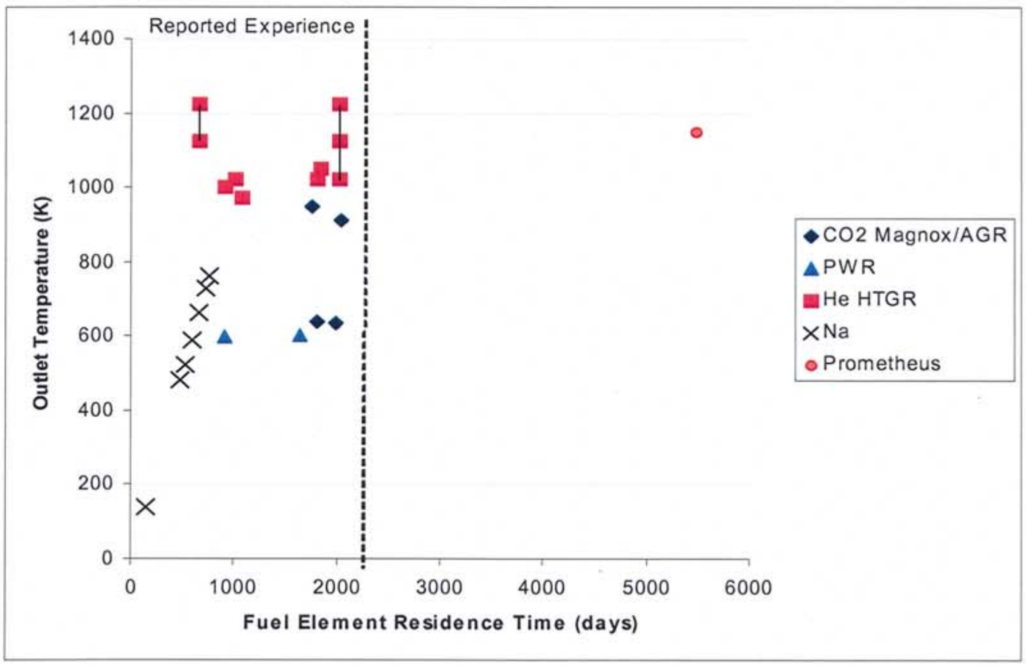
- Ability to survive launch loads and subsequent thermal cycling- a robust fuel system that does not suffer damage during launch (e.g., cracking) is desired

The current understanding of these behaviors along with the approaches to resolve open issues are described in detail in Reference (36) and are summarized below.

### *3.1.1 Issues and Challenges*

The most significant challenge for Prometheus fuel materials is the long mission time at elevated temperature and low fission rate. These conditions are significantly beyond the range of all commercial reactor designs (see Figure 3-1). Therefore, the necessary irradiation and compatibility data for Prometheus do not exist. While time often is not modeled as a primary variable compared to burnup, temperature, and fission rate, it is an important consideration when extrapolating accelerated materials testing and when drawing on the use of known correlations for predicting fuel performance parameters. In particular, compatibility, mass transport, and fuel element performance correlations generally are tested at accelerated time to assess physical phenomena such as thermal creep and diffusion. In addition, changes in restructuring behavior at low fission rates can affect gas release and swelling behaviors in some fuels and must be accounted for properly.

**Figure 3-1: Comparison of Fuel Element Residence Times**




A general comparison of commercial experience and the Prometheus concept relative to the range of burnup and fission rate illustrates two important considerations. First, the inventory of fission product is within commercial experience, and testing has taken burnup rates to significantly higher values. The concern for Prometheus relates to the ability to retain the fission products within the fuel system for the duration of the mission. Second, the lower fission rate reduces temperature gradients in the fuel and gives credence to the possibility for reduced fission product mobility and engineering solutions, such as using liners, thicker clad, or cold plenums. However, testing will still be required to resolve issues related to higher temperature and lower fission rate operation not available from commercial reactor experience.

### *3.1.2 Summary of Work*

The main goal of the pre-conceptual design phase was to select the reference fuel system, which includes the fuel material, fuel form, liner material, and clad material. The fuel material, $UO_2$, was recommended. An assessment of the other fuel system decisions was started but not completed due to the termination of NRPCT involvement in the project. This section discusses the fuel material and form; the clad material assessment is discussed in Section 3.2.

In order to select fuel material, work focused on several significant areas, including assessment of fuel performance literature and codes, evaluation of approaches for obtaining new fuel performance information, and determination of the ease of fabricability of fuel materials. Initially a variety of fuels (metallic, UZrH, $UO_2$, UN, UC/$UC_2$ and UCO) were considered; however, the choice was quickly narrowed down to UN and $UO_2$. Metallic and hydride fuel were eliminated because of lower melting temperatures and poorer dimensional and chemical stability compared to ceramics of uranium. UC, $UC_2$, and UCO were eliminated primarily due to higher swelling and gas release than UN, and less ability to retain fission products compared to UN and $UO_2$.

World experience with a variety of oxide fuel systems gave confidence in the ability to manufacture $UO_2$ fuel in a variety of configurations and correlate with available performance data. However, UN fuel had some material properties that offered potential performance advantages over $UO_2$. For example, UN demonstrated higher uranium densities (nominally 12.7 g $U^{235}/cm^3$ vs. 9.1 g $U^{235}/cm^3$ for $UO_2$) leading to a more compact core. UN also has a much higher unirradiated thermal conductivity which reduces the thermal gradient across the fuel (reduces risk of cracking) and reduces peak temperatures (lower temperatures reduce corrosion activity increasing element life). It was important to investigate UN for these potential advantages while exploring some of its key disadvantages, such as no commercial manufacturing base and a much smaller irradiation performance database. In addition, UN does not retain solid fission products as readily as $UO_2$ and is more highly reactive with potential cladding materials. Resurrecting the UN fuel fabrication processes from historical, shutdown programs such as SP-100 was a concern for the NRPCT and became a priority for determining the feasibility of using UN for a Prometheus reactor. Initial re-establishment of the manufacturing efforts was successful enough to minimize this as a concern in the fuel selection process, and performance data was gathered and compared for the final two fuel options.

Since a reactor configuration (specifically, fuel element design) was not yet selected, another factor in fuel material selection was its flexibility to support a variety of fuel forms. Fuel forms, such as pellet, CERMET (ceramic fuel in metal matrix), and TRISO (coated particle fuel) configurations were considered to meet the Prometheus conditions, and designs were compared based on weight and performance confidence. Table 3-1 summarizes the pros and cons of a variety of initial fuel configuration options and gives the initial assessment for development. Additional discussion for each is provided below:

- TRISO-coated particle fuel, successfully used in previous high-temperature reactors and in the Japanese High-Temperature Test Reactor (HTTR), is an option considered to be developed and demonstrated ("off-the-shelf") for high-temperature gas reactors. One drawback is the lower fuel loading density in TRISO particles, which increases the core size and mass. Conceptually, however, the commercial TRISO fuel systems can be modified to obtain a long life core, possibly up to a burnup of ~8% fission of initial metal atoms (FIMA). A long-life TRISO-based fuel should not be considered off-the-shelf; development and verification are required.

  TRISO fuel is considerably more attractive for a moderated core concept because a moderated core is not as dependent on high fuel loading. With TRISO geometry to contain fission products and swelling, burnup could be pushed to higher levels. Design parameters and methods for TRISO fuel forms were not in the preliminary design basis for Prometheus due to concerns of the correspondingly larger sized cores. Late in the project, concepts were being evaluated that used a moderating medium to reduce fuel loading, such that some effort on determining the extensibility of TRISO fuel and potential performance was warranted.

**Table 3-1: Pros and Cons for Fuel Options**

| Fuel option | Primary Pros | Primary Cons | Initial Decision |
|---|---|---|---|
| Uranium Dioxide ($UO_2$) Pellet | Chemical stability<br>Moderate swelling<br>Established database<br>Most stable fission products | Fission gas release<br>Pellet cracking | Investigate for fast and for moderated core option |
| Uranium Nitride (UN) Pellet | High U density<br>Fission gas retention | High swelling<br>Potential nitrogen instability<br>Rare earth fission products not generally nitrides | Investigate for fast reactor |
| Uranium Carbide (UC, $UC_2$, UCO) Pellet | High U density | High swelling<br>Potential chemical instability<br>Fission gas release<br>Least stable fission products | Do not investigate Data suggests higher risk than $UO_2$ and UN |
| Cermet $UO_2$ | Fission product retention<br>Chemical compatibility with most matrix candidates | Low fuel density<br>Fabrication | Investigate for fast reactor |
| Cermet UN | Moderate U density<br>Fission gas retention | Chemical incompatibility with most matrices | Do not investigate the DOE Office of Scientific and Technical Information database |
| Metallic | Highest U Density | Temperature limited<br>Distortion limited | Do not investigate |
| TRISO | Off the shelf<br>Demonstrated for > 3 years at desired temperatures | Plant sizing-not attractive energy density for 15 year core | Consider for Moderated core option and extensibility |
| UZrH | Prompt feedback<br>Moderation | Temperature limited<br>H Migration<br>Not attractive at high power density and high temperatures | Do not investigate the DOE Office of Scientific and Technical Information database |

- CERMET fuel elements, in which a granular ceramic fuel material is dispersed within a continuous metallic matrix, have been evaluated for space nuclear reactors since the 1950s due to several potential advantages. The advantages include higher effective thermal conductivity relative to other uranium-containing fuels, superior fission gas retention capability, enhanced mechanical stability, and minimized fission recoil damage of cladding materials. Unfortunately, there is little published work to verify or quantify the improvements in these areas for CERMET systems. Furthermore, there are also tradeoffs, primarily in the relatively low maximum attainable fuel density in a matrix, which may lead to larger core size and potentially high parasitic neutron absorption in the metal CERMET matrix.

  Initial design philosophy for CERMET fuel was to assume a hermetically-bonded cladding with no fission product release from the element and no fuel growth. It was expected that even under these ideal assumptions the system would not be attractive on a mass basis compared to pellet fuels. If a compelling concept was developed, an effort to determine realistic long-term performance would be pursued to assess the maximum loading, better define the fuel growth and assess the probability of fission product release. Using these ideal assumptions, core design analysis showed that a cermet design appears to be mass competitive. More realistic cermet properties would need to be developed to better evaluate cermet-fueled core concepts.

- Most of the reactor concepts considered used pellet fuel, because it offered the highest uranium loading. However, the major disadvantages are that the pellet has a limited barrier to fission gas release, swells under irradiation, and has a potential for cracking. Swelling and cracking can cause pellet clad interactions, and, under some circumstances, lead to fuel element failure via mechanical, thermal, or chemical means. Fission gas release was accommodated by a gas-expansion plenum within the fuel element to limit gas pressure and reduce clad stresses. The fission gas plenum increases reactor length and possibly shield mass. Annular pellet designs can accommodate fission gas and reduce swelling forces without increasing fuel element length, but at the expense of average fuel density and the aforementioned potential for reconfiguration.

With the aggressive delivery schedule for Prometheus-1, development of UN and $UO_2$ fuel manufacturing was pursued in parallel with fuel element material selection. Both pellet and CERMET options were considered but initial fabrication efforts were concentrated on pellet fuel since the majority of core design studies used pellets. Confidence was high that $UO_2$ pellet fabrication would be successful due to the large manufacturing base. In addition, with the success of LANL in fabricating UN pellets, there was confidence in UN manufacturing, albeit with stricter and more costly environmental controls. Design studies suggested that incorporating rare earth oxide spectral poisons in the pellets may be necessary to prevent criticality in the event of reactor re-entry and earth impact following a launch accident (as discussed in Section 2.2). Incorporation of rare earth oxides ($Gd_2O_3$) has been demonstrated in the open literature for $UO_2$ fuel. Thermodynamic studies were performed for UN suggesting that such incorporation is possible but additional development and experimental verification is required.

Scoping studies for fuel element manufacturing had commenced for both block cores and fuel pin designs. Common concerns include environmental controls for refractory metal alloy welds, avoiding dissimilar welds of a refractory metal alloy to a Ni-base superalloy, and criticality concerns with sub-assemblies. Manufacturing techniques were reviewed with a number of potential vendors (LANL, ORNL, BWX Technologies, Inc (BWXT), Global Nuclear Fuels (GNF), and others). Automation for the manufacturing of fuel element assemblies using $UO_2$ pellets was demonstrated by GNF. The techniques and robotic equipment to automate fuel assembly are demonstrated for PWR materials technology and could provide a cost estimate basis for producing final rod designs for Prometheus. If this technology is pursued, it would have to be adapted to handle highly enriched uranium and non-conventional materials such as refractory metals and ceramic cladding for Prometheus. The degree of automation used by GNF far exceeds that needed for Prometheus (a process line such as that used for commercial fuel can press enough pellets for a Prometheus core in a few minutes). However, the integrated welding and quality control methods at GNF are of interest. Consideration was also given to manufacturing of UN pellet fuel assemblies at LANL in the event that UN was selected. The environmental controls associated with handling UN increase the complexity of UN fuel element manufacturing when compared to $UO_2$-based elements.

Performance data for candidate materials were being gathered to support the selection of fuel and fuel system materials. A combination of approaches was envisioned to obtain additional relevant information including examination of existing irradiated material, use of computational techniques, bench-top testing, and new irradiation testing.

Obtaining fuel performance data in a timely fashion was a significant challenge due to long lead times associated with new irradiation test programs (test design, specimen fabrication, duration in reactor, and subsequent examinations) and the abbreviated schedule allotted for reactor development. To abbreviate this cycle, and potentially save costs, a search was conducted for existing materials that had been irradiated under conditions similar to Prometheus and could be examined. Several sources of fuel were investigated, including low fission rate blanket fuel from US reactors, Japan Nuclear Cycle Development Institute (JNCDI) and Japan Atomic Energy Research Institute (JAERI) fuel, Nippon Nuclear Fuel Development Company (NFD) fuel, Advanced Gas Reactor (AGR) fuel from the UK, and specimens from the SP-100 space reactor program.

While US blanket fuel from fast reactors and PWRs has fission rates of interest, the fuel came with many drawbacks (e.g., age and condition, the majority of it was mixed oxide fuel, inherent cost of regulated handling and disposal) which were judged to exceed the value of the potential data. JNC/JAERI fuel, irradiated at low fission rates (though higher burnup than Prometheus), and additional fuel owned by NFD were more promising options available for NRPCT. Examination of this fuel could provide important data, including microstructural characterization to locate the fission products, annealing studies to determine fission product release, and annealing and temperature transient studies to characterize swelling and cracking. The possibility of obtaining data from $UO_2$ AGR fuel, which has a similar burnup but a slightly higher fission rate, was considered but never pursued due to termination of NRPCT participation in the Prometheus project.

Removing the irradiated UN SP-100 fuel specimens from long-term storage for examination was considered, but not pursued due to many uncertainties about pin conditions and possible costs associated with future disposal of cask contents under newer, more restrictive regulations. However, the NRPCT also learned about JNC/JAERI UN specimens that could be obtained for examination. Although the density of these UN specimens was lower than desired for Prometheus, they could be useful for obtaining valuable irradiated thermal conductivity measurements. As with the AGR fuel, the JNC/JAERI UN was not pursued due to termination of NRPCT participation in the Prometheus project.

Computational materials techniques were also employed to predict fuel performance data for both UN and $UO_2$ fuels. These techniques were used to predict irradiated thermal conductivity of UN and were planned to help discern fundamental reasons for a difference in the gas release behaviors between UN and $UO_2$. Thermodynamic calculations were used to increase understanding about the stability of solid fission products in UN; however, due to insufficient ternary compound data, the analysis was incomplete. Additional calculations were planned to approximate the data needed for more complete thermodynamic analyses. Bench-top compatibility tests with fuel element materials, and fission products and/or simulated fuel specimens were planned.

To guide compatibility studies, fission product yields for a fast reactor prototypical of Prometheus were calculated using the RACER Monte Carlo neutron transport computer code and reported in Reference (29). While the fission product inventory is the same for UN and $UO_2$ fuels, the chemical state of the products varies greatly. Therefore the chemical reactivity of these fission products is dependent on the primary fuel. Reviews of experimental data along with thermochemical analyses to identify fission products most likely to impact the system were performed.

A preliminary set of relationships for the irradiated behavior of fuels was established for use by designers [Reference (28)]. Upper and lower bound relationships were extrapolated from the available higher burnup, faster fission rate data, however many properties need to be updated with data obtained for Prometheus operating conditions. To assist in predicting Prometheus fuel element behavior existing codes, such as FRAPCON-3 and URANUS, were evaluated. They did not readily apply to Prometheus designs because of their empirical dependence on PWR data. However, the UK code ENIGMA for AGR fuel, though not fully evaluated, may have provided a more realistic model for fission gas release because the operating conditions are closest to designs for a Prometheus reactor.

### *3.1.3 Key Findings and Perspectives*

Based on the investigations discussed above, $UO_2$ was selected as the fuel material for the Prometheus reactor. Given the aggressive delivery schedule and lack of development time, it was judged not possible to sufficiently retire the risks associated with the delivery of a UN based reactor to meet the long, high-temperature operation. UN offered too many material interactions with potential fuel element materials and did not confine a majority of the expected fission products in the fuel matrix. In addition, questions about its stability during operation and increased uncertainty in existing irradiation data were important aspects that would have needed to be addressed with irradiation testing. There is a potential risk that these issues cannot practically be resolved for long-term operation of UN at Prometheus temperatures. $UO_2$, on the other hand, presents a significantly more well-known fuel system. It has well-documented performance with several clad materials, ties up many of the fission products in oxide and intermetallic compounds, and is not expected to dissociate under Prometheus reactor conditions because of the strength of the U-O bond. The options available to engineer UN fuel elements to last the duration of the mission (e.g., utilize thick or multiple clads) posed large increases in reactor mass, with continued uncertainty in the compatibility performance. $UO_2$ therefore became the most appropriate fuel solution.

There are significantly less data available for UN than for $UO_2$ and very few data fall within the operational space of the Prometheus reactor. Furthermore, the existing data have high associated experimental error because the fuel test temperatures are calculated based on coolant temperature. Obtaining more accurate test data would be a high priority if UN were pursued. For these reasons, selected UN studies (e.g., compatibility) were to continue in the event that a $UO_2$ system did not meet mass requirements or possibly to support later missions.

Although $UO_2$ was recommended as the fuel material, it should be noted that the resurrection of the LANL SP-100 UN fabrication process is important for the consideration of UN for future space reactor projects. After being dormant for over 10 years, the SP-100 UN fabrication process was used successfully by LANL to produce SP-100 grade UN with the exception of the grain size, due to furnace limitations at the time. As a result, UN pellet fabrication concerns were not considered to be a deciding factor between UN and $UO_2$ for fuel selection. Production of $UO_2$ was still expected to be less costly and developmental than UN fuel production.

As discussed previously, several techniques were employed to obtain performance data for candidate Prometheus reactor materials. Existing irradiated fuel was evaluated and determined to be a potential source of fuel irradiation performance data, and irradiation test plans were developed to fill in the gaps in the existing performance data. Preliminary correlations for fuel materials properties were compiled for design use. Computational materials approaches were taken to predict fuel performance behavior and compatibility. Despite these efforts, however, many issues still remain about fuel performance under Prometheus conditions, including:

- Insufficient irradiation data at low fission rates, high temperatures, and low burnups to develop design relationships
- Reducing the uncertainty in the existing UN database
- Experimental verification of $UO_2$ restructuring behavior
- Determination of the best relationship for fission gas release from $UO_2$
- Alleviating uncertainty in $UO_2$ swelling calculations
- Long term chemical compatibility of fuel, fission products, and fuel system materials

If NRPCT involvement in Project Prometheus had continued, additional efforts would have focused on gathering information to alleviate these concerns.

## 3.2 Core and Plant Structural Materials

Reliable operation of components for the reactor and energy conversion plant requires well understood properties of structural materials. There are three categories of materials for structural applications in the Reactor Module: fuel cladding, core structure (e.g., core block, structural joints), and pressure boundary (e.g., reactor vessel, piping, plant components). The material classes pursued for structural applications included refractory metal alloys, nickel (Ni)-base superalloys, titanium (Ti)-base alloys and silicon carbide (SiC). Refractory metal alloys and SiC were considered primarily for fuel cladding due to their high-temperature strength and creep resistance. In addition, refractory metal alloys were considered for both core structure and the pressure vessel. Ni-base superalloys were considered for the pressure vessel and a majority of the plant components due to adequate temperature capability, an overall good balance of properties and an established industrial manufacturing base for fabricating complex components such as heat exchangers and turbines. Ti-base alloys were considered for lower temperature plant components to reduce mass.

As discussed in Reference (21), structural material behavior that is important for reliable performance includes:

- Thermal creep in multiple product forms, such as thin-walled tubing, thin sheet, thick plate and rod or bar
- Fatigue/creep-fatigue resistance
- Creep crack growth resistance
- Resistance to irradiation effects such as creep, swelling and embrittlement
- Resistance to environmental degradation from:
  - Fuel, fission products, and transmuted elements
  - Impurities inherent to the helium/xenon gas
  - Transport of impurities from other spaceship components
  - Space vacuum
- Resistance to microstructural phase instability upon thermal exposure
- Good fabricability for:
  - Thermomechanical processing of complex shapes
  - Welding and post-weld thermal annealing
  - Dissimilar metal joining

### *3.2.1 Issues and Challenges*

As discussed in Section 1.4, the Prometheus mission requirements result in key materials challenges for the Reactor Module. The electrical power requirement of ~200 kWe, coupled with a goal to minimize mass, results in a reactor exit temperature of ~1150 K that limited the reactor materials to refractory metal alloys and silicon carbide. Initial materials selection considerations assumed that the upper temperature limit of these materials would be determined by creep, chemical interactions with the coolant, and minimum ductility or toughness requirements. No single material was expected to satisfy all requirements, requiring combinations of materials and skillful engineering to achieve the final design of the fuel assembly and core structural components.

A wide variety of candidate materials were considered based on the pre-decisional operating parameters established for the reactor, shielding and energy conversion systems. Prior work on the SP-100 program focused on Nb-1Zr fuel cladding, but a review of the data revealed unsatisfactory creep properties for the JIMO mission requirements and uncertainties in the data. Compatibility between the fuel and fuel cladding was a significant issue as most Ta- and Nb-base alloys degrade in the presence of small amounts (ppm) of oxygen and nitrogen and chemical liners between the fuel and cladding would have been needed. Molybdenum and SiC may be more chemically compatible with the fuel, potentially avoiding the complications associated with the SP-100 fuel liner. Refractory metal alloys and silicon carbide exhibited more than adequate thermal creep resistance at high temperatures, while nickel-base superalloys were satisfactory for the pressure vessel at temperatures up to 900 K. Core structural materials would have to possess sufficient toughness to survive launch loads and thermal transients during operation. More conventional structural materials were advocated for the plant due to an established industrial experience with fabricating complex components such as heat exchangers and the turbines from nickel-base superalloys and stainless steels. Ti-base alloys were also considered as a means to reduce mass. All of the materials considered were susceptible to degradation by radiation damage, thermal aging and chemical interactions with the fuel, working gas impurities and space environments. Joining of similar materials is well established, but challenges with dissimilar material joints would have required extensive research, especially for use at elevated temperatures. Significant materials testing and development programs were planned to measure and quantify the material limitations for designing the space reactor system. The following is a list of key issues from References (21), (25), and (31).

*Nickel-Based Superalloys*

- Require an environment with sufficient oxidation potential to create and maintain a protective surface oxide
- Relatively low (~900 K) upper limit service temperature based on thermal creep concerns
- Grain boundary embrittlement due to radiation-induced solute segregation and/or production of helium at elevated temperatures
- Microstructural phase instability leading to degradation in mechanical properties is more likely as the number of alloy constituents increases. Ni-base superalloys tend to have more constituents than refractory metal alloys.
- Embrittlement of dissimilar metal joints upon joining and during service

*Refractory Metal Alloys*

- General concerns
    - Susceptible to radiation-induced hardening and embrittlement at T<0.3 of the melting point (800 to 1100 K, depending on the alloy).
    - Possibly susceptible to radiation-induced solute segregation and production of detrimental elements through neutron transmutation reactions that lead to phase instability and embrittlement. Transmutation of Re to Os was a possible concern.
    - Microstructural phase instability leading to degradation in mechanical properties.
    - Embrittlement of dissimilar metal joints upon joining and during service
- Alloy specific concerns
    - Ta and Nb-based alloys (e.g., ASTAR811C, Ta-10W and FS-85)
        - High susceptibility to interstitial embrittlement by carbon, oxygen and nitrogen. Impurities may be derived from start-up and other system materials such as Ni-base superalloys. Coatings and/or environmental controls may be required.
        - Ta becomes highly activated when irradiated and could hinder ground unit testing
    - Mo-base alloys (e.g., MoRe)
        - Forms a volatile oxide that is not protective at high oxygen potentials.
        - Re increases Mo ductility, but there are thermal phase stability issues with 47.5% Re. Lower Re-content alloys (e.g., Mo-41Re) are marginal for fabricability.

*Silicon Carbide Materials*

- General concerns
    - Require an environment with sufficient oxidation potential to create and maintain a protective surface oxide

- No significant NRPCT experience with analysis tools and ability to set performance limits
- Require a probabilistic design approach
- Lower thermal conductivity than metals
- Joining of SiC to dissimilar materials is not well established

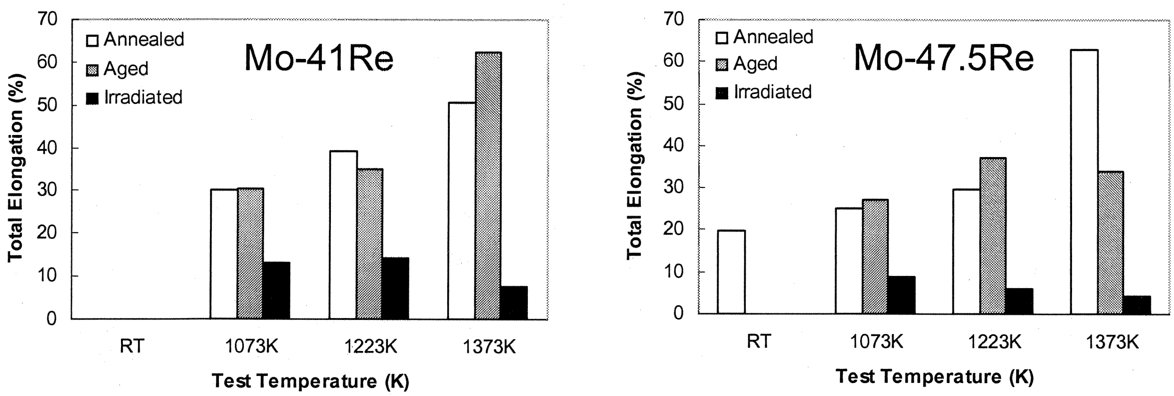
- Material form specific concerns
  - Monolithic form - Low fracture toughness and component strength will be inversely proportional to the size of the largest flaw
  - Composite form – Improved fracture toughness, but work would be needed to understand whether service loadings could result in matrix cracking
    (While the component would still have structural strength after such cracking, there is a concern that SiC particles from matrix cracking could enter the coolant and/or result in a loss of hermeticity).

### *3.2.2 Summary of Work*

Extensive experimental test programs were in progress when NRPCT's involvement with the Prometheus project was terminated. These included irradiation test programs at the High Flux Isotope Reactor (HFIR) [Reference (22)], located at ORNL, and at a Japanese fast test reactor, JOYO [Reference (32)]. The ORNL study compared thermally aged refractory metal alloys to materials irradiated at high temperatures in HFIR. Thermally aged samples of ASTAR-811C, T-111, FS-85, Mo-41Re and Mo-47.5Re were exposed for 1100 hours, equivalent to the comparable radiation samples at temperatures 25 K above the expected irradiation temperature to account for uncertainties. These exposures represented 1% of the mission thermal life and 10 to 20% of the end of life fluence. The HFIR experiments revealed irradiation embrittlement concerns for Mo-Re alloys (see Figure 3-2) and inconclusive results for FS-85 after 1% of the thermal exposure and 10% of the irradiation exposure expected for the JIMO mission. No microstructural characterization of any irradiated material or mechanical testing of Ta-based alloys T-111 and ASTAR-811C were obtained as the program was terminated prematurely. The transmutation of Re to Os occurs more readily in the thermal neutron spectrum of HFIR. The HFIR testing simulated an accelerated Re to Os transmutation effect thereby showing the potential materials behavior under long time exposure in a fast reactor. Thermal aging studies at ORNL that complemented the HFIR studies (1% of the JIMO thermal exposure) revealed a loss of ductility for Nb-based FS-85 and Nb-1Zr alloys as well as Ta-based T-111 [Reference (55)]. Initial microstructural characterization provided a preliminary explanation for the thermally-aged FS-85 behavior; embrittlement of FS-85 resulted from the precipitation of $ZrO_2$ and other Zr-rich compounds at the grain boundaries. Additionally, T-111 and Nb-1Zr exhibited an increase in precipitation along grain boundaries with an increase in thermal aging temperature. Chemical analyses of the aged materials were consistent with the proposition that the loss in ductility in these alloys was due to precipitation. These results could preclude the use of tantalum- and niobium-base alloys in lithium-cooled reactors. Removal of the reactive elements Hf and Zr could allow their use in gas-cooled reactors, but further work is needed both to confirm the effect and to validate the remedy. Finally, the most complete data were obtained for the Mo-Re alloys where the tensile ductility was reduced after irradiation, Figure 3-2. In most cases, low energy, intergranular and transgranular fracture morphology was dominant.

Irradiation testing of Ni-base superalloys was planned but not conducted. However, due to the known susceptibility of Ni-base superalloys to radiation-induced embrittlement the NRPCT was considering investigation of nuclear grade Ni-base and Co-base alloys as a method to mitigate radiation-induced embrittlement by reducing helium generation and solute segregation.

**Figure 3-2: Effect of Thermal Aging and Neutron Exposure on Mo-Re Alloy Elongation**
*Summary of total elongation for (a) Mo-41Re and (b) Mo-47.5Re after out of pile thermal aging and neutron exposure at HFIR as a function of irradiation and test temperature. Both materials exhibited radiation-induced embrittlement [Reference (22)]*




Environmental degradation programs were addressing concerns on the mass transport of impurities such as carbon and oxygen from nickel-base superalloys to refractory metal alloys that would promote mechanical property degradation of both materials [Reference (37)]. ORNL conducted thermodynamic calculations for Alloy 230, Alloy 792, Nb, Nb-1Zr, Ta, Ta-1Hf and Mo. These calculations indicated a sufficient driving force for the mass transport of carbon from Ni-base superalloys to refractory metal alloys that would most likely result in embrittlement of the refractory metal alloys and the decarburization and loss of mechanical properties of the Ni-based superalloys. These results indicated the need for more sophisticated modeling that would include reaction kinetics and system geometry. Experimental verification of potential mass transport effects was underway with three different experimental techniques. Capsule tests would have been conducted in static gas environments to screen for major effects. Once-through flowing gas loop tests would have evaluated the kinetics of elemental uptake and release under more representative conditions. Recirculating gas loop tests were being initiated at ORNL and Bettis to generate well-controlled conditions, representative of the actual system, under which extensive evaluation of dissimilar materials behavior could be performed.

Mechanical testing had focused initially on obtaining additional creep behavior on refractory metal alloys and re-establishing high vacuum test facilities [Reference (189)]. Little useful data were obtained as most of the effort was expended on establishing test capability from systems that had not been used in many years. Although only subtle differences in microstructure were noted, the initial results indicated that the creep resistance of tube material was inferior to that of heavily rolled sheet. Therefore, the results did underscore the need to obtain properties from material with the appropriate processing history.

Joining studies focused primarily on dissimilar metal joining processes of solid state joining and brazing with the intent of minimizing the formation of embrittling intermetallics at the interface. A solid-state joining feasibility study was conducted at the Edison Welding Institute that showed promising results using inertia welding to join refractory metal alloys to Ni-base superalloys. NASA Glenn Research Center began a feasibility study to better understand the kinetics of intermetallic formation between refractory metal alloys and Ni-base superalloys, but little progress was made due to program termination. Both NASA-GRC and NRPCT pursued the use of an interlayer to hinder interdiffusion between dissimilar metal alloys to prevent the formation of embrittling intermetallics [Reference (50)].

### *3.2.3 Key Findings and Perspectives*

The selection of the Direct Gas Brayton system represented a significant challenge to the development of materials due to the unique operational requirements of a high-temperature gas reactor. In general, structural material choices based on existing data would have required additional design tradeoffs to minimize lifetime risks. Materials testing and development programs were underway and planned to determine material limitations as statistically significant data were lacking, especially under prototypical conditions. The aggressive schedule limited initial materials selection to those that would require minimal development and design decisions likely would have been made with incomplete data, increasing the risk to the budget and schedule.

## 3.3 Shield Materials

Materials selections are based on providing both materials with low atomic mass to attenuate neutrons and materials with a high electron density (e.g., tungsten) to attenuate gamma radiation. The shield is a combination of these materials carefully arranged to reduce both the neutron and gamma doses with minimal overall mass. Some of the candidate materials are damaged or changed by the radiation (e.g., structural swelling and chemical degradation) and testing is necessary to characterize these effects well enough that they can be accommodated in the shield design. At the time of project restructuring, the five primary shielding materials were being considered: beryllium (Be), boron carbide ($B_4C$), tungsten (W), lithium hydride (LiH), and water ($H_2O$) with neutron absorbing material dissolved in it. Materials engineering efforts were planned to complete the materials studies for these promising shield materials.

### *3.3.1 Issues and Challenges*

At the time of program restructuring, a final low mass neutron shielding material had not been chosen. Lithium hydride and $H_2O$ were being considered, but both required irradiated materials testing. The cost and delays associated with this testing, as well as the possibility that neither material would be found acceptable, were a concern. Generally, one of these materials was considered necessary to ensure a low mass shield could be incorporated on a Prometheus reactor plant. However, even if both materials were ultimately eliminated, the remaining primary materials were judged to be sufficient to design a shield with a modest mass penalty—depending on the design. The challenge with LiH was understanding and mitigating the irradiation-induced material swelling. For $H_2O$, the challenge was corrosion of a containment vessel over the long life of the shield. One complication with water was that neutron absorbers likely would be needed in the water, e.g., lithium as LiOH, or boron as boric acid. These chemicals were expected to make the task of understanding the corrosion of a water/containment vessel much more challenging than for a pure $H_2O$ system.

No significant issues or developmental challenges were noted for Be, $B_4C$ or W, except that Be is considered toxic, and controls would need to be in place to ensure safe handling.

### *3.3.2 Summary of Work*

The NRPCT completed detailed literature reviews of LiH, and Be. In the case of LiH, the results of fifty irradiated materials tests that were conducted over the past ~50 years were reviewed and evaluated. The focus of much of this testing was irradiation-induced swelling. The NRPCT also performed quantum mechanical modeling of irradiated LiH, as well as statistical studies of the prior experiments and other research. The NRPCT developed a hypothesis as to why LiH swelled (up to 25%) in some cases, but not significantly in other cases (e.g., <2%). The NRPCT hypothesized that LiOH contamination of the LiH may have accelerated the irradiation-induced swelling. Therefore, eliminating the LiOH contamination and repeating some of the prior tests along with detailed characterization was planned, but not executed. This information is included in Reference (34).

In the case of Be, the literature review indicated that little if any irradiated materials testing would be needed for this material. Detailed studies of $B_4C$, W, and $H_2O$ were not performed. Studies of shield structural or other (e.g., insulating) materials were not performed.

### *3.3.3 Key Findings and Perspectives*

The primary materials were judged to be sufficient to design a Prometheus shield—excluding structural materials, etc. Although detailed design studies are required to accurately estimate the mass of shields, the LiH and $H_2O$ based shields were both expected to be similar in mass, and lower mass than virtually any other alternative. The foremost pre-conceptual shield concepts included: (1) a Be/$B_4C$/W/LiH shield; (2) a Be/$B_4C$/W shield; (3) and a Be/$B_4C$/$H_2O$ shield. Since the shield design and materials studies were still preliminary, alternative materials (e.g., $^{nat}B$ or $^{10}B$ metal) were still being screened, but at a low level of effort. Each of the above three shield concepts was judged to be mass competitive for a Prometheus application, and detailed design studies were needed to understand the final masses of each.

Key findings on each of the candidate materials are provided below. Further information can be found in Reference (35).

<u>Primary Gamma Shielding Materials:</u>
Tungsten was recommended as a primary gamma shielding material. Tungsten has well known shielding properties, is readily available, manufacturable, and relatively inexpensive. Tungsten is commonly used in irradiation testing as a gamma heating material, with no significant material degradation noted. It is not expected to require any materials development. A complete literature search on this material should be performed to ensure there are no gaps in the material properties that warrant testing.

<u>Primary Neutron Shielding Materials:</u>
Beryllium provides some gamma attenuation as well as neutron moderation (slowing down of fast neutrons, which is critical to shielding effectiveness) but not absorption. A beryllium slab at the front of the shield may also be beneficial as: (1) a mounting plate for the aeroshell; (2) a potential housing for neutron detectors; and/or (3) a heat removal device. Beryllium is a commonly used reflector material, with extensive irradiated materials data in the literature. From a manufacturing standpoint, large, high quality Be slabs up to ~114 cm diameter are commercially available.

Boron carbide is a high-temperature material that provides some gamma attenuation as well as neutron moderation and excellent thermal neutron absorption. Boron carbide is a commonly used as a neutron absorber, especially in light water reactor control rods. As such, it has been irradiation tested extensively. High quality plate is commercially available and relatively inexpensive, but fabrication issues were not fully investigated, and depending on the fabrication method chosen, some confirmatory irradiated materials testing may be required.

Lithium hydride provides excellent neutron moderation and absorption, and is very low density (~0.8 g/cm$^3$). ORNL shield material screening studies identified that all minimum mass shields included LiH. However, irradiation-induced swelling was identified as a key issue for LiH. The NRPCT completed a literature review and analysis (including quantum mechanical modeling) of LiH, focused on swelling [Reference (34)]. It was observed that either casting or out-gassing the LiH reduced swelling in prior tests; however, this was not conclusive. The NRPCT hypothesized that LiOH contamination is contributing to the swelling; however, further unirradiated and irradiated materials testing and engineering evaluation are required to clarify LiH swelling mechanisms.

Pure $H_2O$ is an excellent neutron moderator, but without the addition of a neutron poison (e.g., dissolved boric acid and/or lithium hydroxide) neutrons would be absorbed by H atoms releasing a gamma ray and reducing shield effectiveness. Therefore, the NRPCT and ORNL evaluated $H_2O$ systems with dissolved neutron absorbers, which are routinely used in operating reactors to control reactivity and in safety systems to ensure reactor shutdown. The differences between these systems and the space reactor shield environment must be evaluated in detail to determine the viability of $H_2O$ for use in the space shield design. For instance, no chemistry monitoring or corrections are envisioned for the spaceship. A key issue is corrosion of the containment system over the mission duration (12 years for the JIMO mission, up to 20 years for other Prometheus missions). Effort should focus on defining an overall system that provides the desired neutron attenuation characteristics, and has the lowest likelihood of having significant corrosion concerns. Radiolytic decomposition of the water is also a concern. As with LiH, irradiated materials/system testing would likely be required.

## 3.4 Reflector Materials

A large fraction of neutrons (greater than 30 percent) would leak from the small, fast spectrum nuclear reactors being considered. A neutron reflector was used to optimize the design. A combination of stationary and movable reflectors around the reactor core was being considered for reactivity control. An ideal reflector material would reflect leaking neutrons back into the core without absorbing them, would not be significantly damaged by core radiation, and would be very low mass. Figure 1-1 and Figure 1-2 in Section 1.5 show the reflector in the Reactor Module arrangement.

The two materials studied in depth by the NRPCT, which appear to have the most promise in a Prometheus type reflector application, are Be and BeO; however, toxicity concerns would necessitate safety controls (primarily during the manufacturing process). Three alternative materials, magnesia (MgO), alumina ($Al_2O_3$), and magnesium aluminate spinel ($MgAl_2O_4$) were recently identified in reflector studies as having similar reactivity control swing performance as Be and BeO in a Prometheus-type application. However, each of these alternate materials are denser than Be and BeO, and therefore are expected to increase reflector mass. They also cause a difference in the power distribution within the core and changes to the reactor kinetics parameters. Further, isotopically enriched $^{11}B_4C$ provides similar reflector performance with a comparable mass to a Be or BeO reflector. The issue with $^{11}B_4C$ is that the material must be highly enriched in $^{11}B$ and may require irradiated materials testing, which could be prohibitively expensive. Due to program restructuring, detailed literature reviews were not performed by the NRPCT for MgO, $Al_2O_3$, $MgAl_2O_4$, or $^{11}B_4C$.

### *3.4.1 Issues and Challenges*

The reflector essentially surrounds the reactor, and was expected to be approximately 10 to 12 cm thick. As such, it represents a large volume, and therefore, choosing the lowest density material is important to keeping the overall spaceship mass low. Beryllium was the lowest density material being considered (~1.85 g/cm$^3$), BeO was the next lowest (~2.9 g/cm$^3$), and the other alternatives were somewhat denser than BeO, except that $^{11}B_4C$ was ~2.5 g/cm$^3$.

Ideally, the reflector material would be strong enough to support itself; however, only Be was potentially suitable as its own structure, and that was uncertain. Therefore, structural and canning materials must be identified in conjunction with the reflector materials.

The material properties for Be are largely known, whereas gaps in the properties of BeO relative to a Prometheus application were identified. If BeO were selected, testing would need to be performed, which was likely to require significant effort and several years to complete. Of particular interest was the irradiation-induced swelling of BeO, which affects mechanical stability and would need to be accommodated to some extent in a BeO reflector design. This was a significant issue, since the literature indicated that many BeO specimens had disintegrated or fractured, likely due to swelling.

Toxicity concerns for Be and BeO were also evaluated. Generally, machining of Be-bearing materials requires significant controls; however, if the parts are fabricated, cleaned, and handled carefully, adequate controls could be established with reasonable effort.

Due to Prometheus program restructuring, the NRPCT did not study the alternative reflector materials in detail, and therefore significant testing may be required. However, it was expected that the toxicity issues would be lower with these materials.

### *3.4.2 Summary of Work*

A detailed pre-conceptual design information document was issued that provided a material property compilation for Be and BeO [Reference (30)]. Beryllium oxide specimens were planned to be irradiated in the JOYO Japanese test reactor to partially fill the material property gaps, but more testing in the High Flux Isotope Reactor (HFIR) test reactor at Oak Ridge National Laboratory (ORNL) was expected to be needed. A key issue for BeO was reducing the current commercially available grain size from ~10 to ~5 micrometers to both improve irradiated material properties (e.g., irradiation-induced swelling) and allow the use of historic irradiated materials test results in the literature (lowering the extent of required testing and therefore cost of using this material). A subcontract with Brush Ceramic Products (BCP) produced ~7 micrometer grain size BeO [Reference (33)]. The BCP subcontract also produced material properties for the current state of the art ~10 micrometer BeO material, BW-1000.

### *3.4.3 Key Findings and Perspectives*

A summary of the findings from the Be and BeO reflector materials studies is presented below.

<u>Beryllium</u>
Unirradiated and irradiated material properties are largely known and discussed in Reference (30). Beryllium is the lowest density candidate material (~1.85 g/cm$^3$) and has a melting point of ~1558K. It can be considered for structural application, but is known to irradiation embrittle at relatively low fast fluences (~7.5x10$^{20}$ n/cm$^2$ E>1 MeV). Beryllium reflectors tend to be somewhat thicker than BeO; however, the significantly lower density can make them a lower mass alternative to BeO. Detailed studies must be performed to determine the best material.

<u>Beryllium Oxide</u>
In addition to the external radial reflector, BeO was also evaluated as a fuel element axial reflector (within the fuel element) because of its very high melting point, ~2840 K. Following a detailed literature review and analysis, material property data gaps for these applications were identified [Reference (30)]. Most of the data available was from the 1950s and 1960s and correspond to different grades of BeO (i.e., impurity content, processing, grain size). The grain size of BeO strongly affects the irradiation properties (primarily swelling). A contract was placed with Brush Ceramic Products to procure BeO specimens for material property testing, including efforts to reduce their standard material (BW-1000) nominal grain size from 10 to 5 micrometers—7 micrometers was achieved after three attempts. Results of this contract are discussed in Reference (33). Specimens were planned to be tested at various fluences [13 – 51x10$^{20}$ n/cm$^2$ (E>0.1MeV)] and temperatures (850K and 1050K) in the JOYO test reactor. Irradiation testing of BeO was also planned in the HFIR at ORNL. The focus of these irradiation tests was to determine irradiation swelling, irradiated thermal conductivity and irradiated compressive strength.

A plan was developed with Brush Ceramic Products to safely handle BeO test specimens planned for irradiation testing. This plan, described in Reference (33), included loose surface contamination limits, cleaning procedures, Be detection methods and packaging/shipping requirements for BeO test specimens. A key outcome from this planning was that BeO specimens could be handled safely provided no actions are taken that would produce loose BeO particulate (e.g., no grinding, machining, etc.).

## 3.5 Irradiation Testing of Fuel, Fuel Elements, and Structural Materials

### *3.5.1 Issues and Challenges*

To support the development of a fuel system and core and plant materials as well as to generate the necessary design and performance data for Prometheus, a comprehensive irradiation test plan was developed. Based on testing needs, aggressive timing, and available test facilities, the NRPCT formulated a plan that included existing facilities and developmental facilities in the United States as well as existing foreign facilities. Most of these tests were at aggressive temperatures and required extensive test specimen and test capsule development.

### *3.5.2 Summary of Work*

<u>Fuel Performance Testing</u>

The ATR MICE Facility was to be used for conducting key fundamental performance testing of UN and $UO_2$ fuel pellet testing. Irradiation temperatures from 1400 to 1600K were planned with burn-ups up to 4 ($UO_2$) to 10 (UN) $x10^{20}$ fissions/cc pellet. Test hardware designs were being created for testing UN and $UO_2$ specimens in the MICE test facility. Achieving prototypical conditions and minimizing temperature uncertainties were key facility goals for these tests. The details of this testing can be found in Reference (23). Specific details on test specimen fabrication are provided in Reference (24). Additional testing in the MICE facility was expected to support either fundamental fuel element testing and/or additional fuel performance testing.

<u>Fuel Element Testing</u>

Facility assessments were being conducted to determine the best way of irradiation testing of prototypical fuel pins. Initial scoping efforts focused on assessing the feasibility of testing in a new or modified facility in the ATR, the proposed Materials Testing Station being developed for the LANSCE Facility at LANL, and the JOYO reactor. A high priority goal for the program was to establish a means of obtaining fuel element performance irradiation data in time to support key recommendations and decision dates. A standard facility that conducts this type of testing does not currently exist. Based on programmatic experience, the best testing option to initiate testing in the soonest practical amount of time involves use of uninstrumented drop-in-type capsules. Various facilities that were under consideration included the ATR, HFIR, and some research universities. Locations in the ATR under consideration included the I-Hole or B-Hole positions in the ATR reflector region, MICE Secondary, and finally Loop ICE. In HFIR, the RB (removable beryllium) position was under consideration.

In assessment of the ATR for instrumented facility options, a conceptual nuclear model of a facility with a modified thermal spectrum was developed which incorporated a “dry well” test configuration in an inactive (South) flux trap. This concept has been referred to as MICE/ITV hybrid and is essentially very similar to the gas test loop being developed by the Advanced Fuel Cycle Initiative (AFCI, DOE-NE), except this concept does not include booster fuel. This facility would include gross fission gas and temperature monitoring. Initial studies showed such a facility could be created to provide a modified thermal neutron spectrum which minimizes transmutation concerns for the candidate fuel system materials, with the exception of rhenium. However, preliminary thermal assessments indicate that such a facility may require a high volume flowing gas (helium) system to cool the test hardware and specimens. It is estimated that this type of facility could be available for fuel element testing in early 2009.

Another option for testing fuel elements in an instrumented facility involves creating a test vehicle that would be installed in the ATR I-Hole locations. This would be a type of drop-in test train, surrounded by booster fuel, which would also utilize gross fission gas and temperature monitoring. This test vehicle, however, would be inserted directly in the ATR process water instead of using a "dry well." The ATR process water would be in direct contact with the test hardware and should provide sufficient cooling for the test hardware. Rough estimates indicate this test vehicle could be available for testing by mid-2007. Transmutation concerns with the fuel systems test materials need to be evaluated. The ATR contractor (BEA) completed a limited conceptual feasibility study of the I-Hole facility.

In addition, the LANSCE MTS was being considered for fuel element testing and was currently deemed to be a contingency action. Concerns with the less than prototypical test environment of a spallation facility, along with the project funding and schedule uncertainty made this a back-up platform.

Finally, efforts were underway to ascertain whether fuel element testing can be conducted in the JOYO reactor. While the prototypic test environment (fast spectrum) makes this a desirable test platform, potential complications of obtaining approvals for testing fuel abroad and JOYO reactor licensing requirements lead to scheduler uncertainties that make this option less attractive than the ATR option for near term testing. Efforts were being pursued to ascertain whether conventional schedules for this type of testing could be accelerated.

<u>Core and Fuel Element Structural Materials</u>

For testing representative structural and fuel materials, two primary testing initiatives were pursued, one at the High Flux Isotope Reactor (HFIR) located at ORNL and the second at the JOYO Experimental Reactor (operated by the Japan Atomic Energy Agency, JAEA) in Japan.

The first structural materials irradiation test was the Pathfinder series in HFIR [Reference (22)]. HFIR offers a modified thermal spectrum that can be useful for testing select structural materials. The Pathfinder testing generated rapid, low-fluence information on refractory metal alloys (FS-85, ASTAR 811C, Mo-47.5Re, Mo-41Re, T-111). The post irradiation examinations (PIE) of these specimens was completed in December 2005 and summarized above.

The second structural materials test program involved the JOYO Experimental Reactor. This facility was planned for testing core, fuel system, and shielding structural materials under more prototypic fast neutron spectrum conditions [Reference (32)]. This was a unique effort which involved the NRPCT designing, assembling, and delivering test capsules to JAEA for insertion in the JOYO reactor. Specimen fabrication for the JOYO-1 test campaign had begun [References (52), (53), and (54)], and the disposition of those samples is listed in Reference (52). An assessment was underway to determine the PIE plan which included selecting where the PIE should be conducted (totally in Japan, totally in the United States, or a combination). Two additional structural materials testing campaigns in the JOYO reactor were also planned to support qualification and confirmatory test objectives. Issues related to waste and shipping, primarily associated with JOYO, as well as international testing experiences can be found in References (27) and (38).

#### *3.5.3 Key Findings*

The NRPCT developed a test strategy to understand fuel behavior and performance using an existing ATR test platform in the near term and a developmental facility for longer term testing. Alternative facilities were evaluated but were not pursued due to not meeting the required timing, technical issues with the data to be generated, or to operational concerns. A fuel test in JOYO appears feasible, but the timing would not support early design decisions.

The United States no longer has an irradiation testing infrastructure to prototypically test materials for a fast reactor. Further, some materials that experience significant transmutation in a thermal reactor cannot be evaluated effectively without a fast spectrum reactor. A review of the world's available test reactors identified the JOYO test reactor in Japan as the best facility to conduct the planned space reactor materials testing. The NRPCT successfully identified an approach to conducting material irradiation tests in the JOYO reactor and was moving ahead for a test insertion in May 2006.

# 4 SHIELDING SUMMARY

The reactor shield provides neutron and gamma radiation attenuation to reduce reactor radiation damage to the spaceship electronics at the mission module. Other components (e.g., motors, alternators, cables, reactor plant sensors, radiator materials) that are not as radiation sensitive as the electronics but are in higher radiation fields closer to the reactor must also be protected. The shield also provides structural support between the reactor and energy conversion subsystems, provides passage for reactor coolant pipes, provides support for reactor control mechanisms and shaft penetration through the shield, and provides attachment points for the aeroshell and nuclear instrumentation. Calculated radiation limits from reactor radiation for the mission module were 25 $kRad_{Si}$ (gamma) and $5x10^{10}$ $n/cm^2$ 1-MeV equivalent silicon damage (neutrons).

The reactor shield is a significant mass component of the Reactor Module, with a mass approximately equal to the reactor itself (see Section 5 for additional discussion of Reactor Module mass). Shield mass is minimized by shielding only within a very narrow cone angle as shown in Figure 4-1, thus creating a locally protected shadow region, rather than shielding around the entire reactor. A 12°/6° elliptical shadow cone was based on pre-conceptual spaceship radiator and high gain antenna configurations. A uniform 12° cylindrical cone could have been used, but would have resulted in a higher shield mass. Outside of the shadow cone, radiation levels would be very high. All equipment is kept within the shadow cone to prevent high direct reactor radiation doses and to prevent scattering reactor neutron radiation back into the shadow cone and increasing doses to other components. The reactor shield attenuates gamma and neutron radiation by about a factor of 100 and 100,000, respectively. The mission module is located approximately 50m from the core. The radiation doses are reduced by about another factor of 1000 due to distance ($\sim 1/r^2$) fall off. Other equipment, like energy conversion components and spaceship thrust mass (xenon), can also provide shielding if they can be beneficially arranged between the reactor shield and the mission module, which would reduce reactor (especially gamma) shield mass.

The shield size is a primary driver for shield mass. The shield radius is determined by projection at a 12° (6°) angle from the forward outboard most corner of the reactor or reflector as shown in Figure 4-2. The reactor diameter, reflector diameter, and reactor length are, therefore, key factors in determining shield radius and mass. Placing the shield as close to the reactor as possible minimizes the shield radius. The reactor control device types (e.g., sliders, drums, etc.) and assumptions (e.g., slider travel) also factor into the shield radius. The shield cone half-angle (12°/6°) affects shield radius and is controlled by required radiator area, antenna size, and payload distance. The shield thickness is primarily driven by reactor power, lifetime, core to payload distance, payload neutron dose limits, and shield material selection and optimization.

**Figure 4-1 Reactor Shield Shadow Cone and Basic Shield Shape**

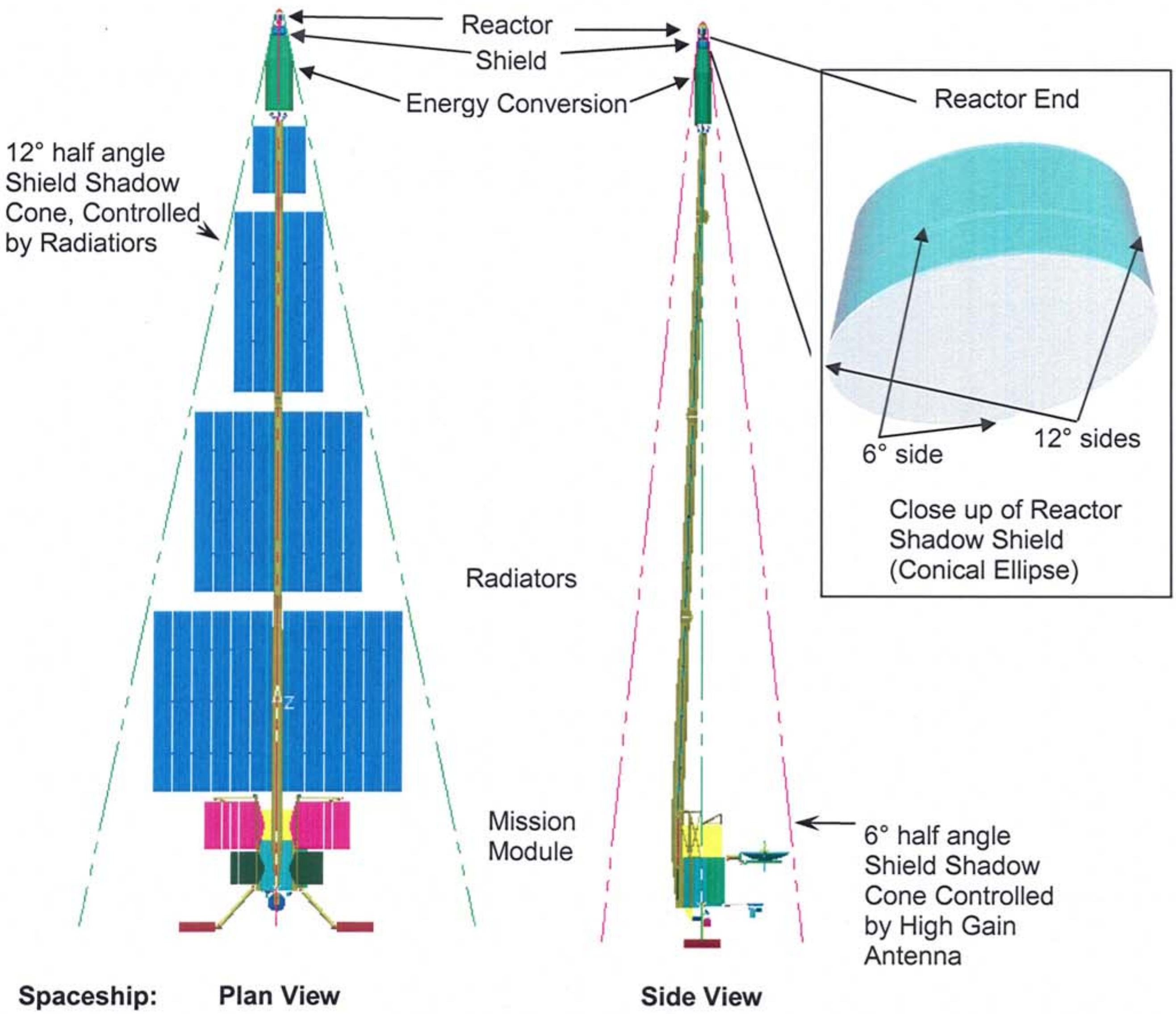
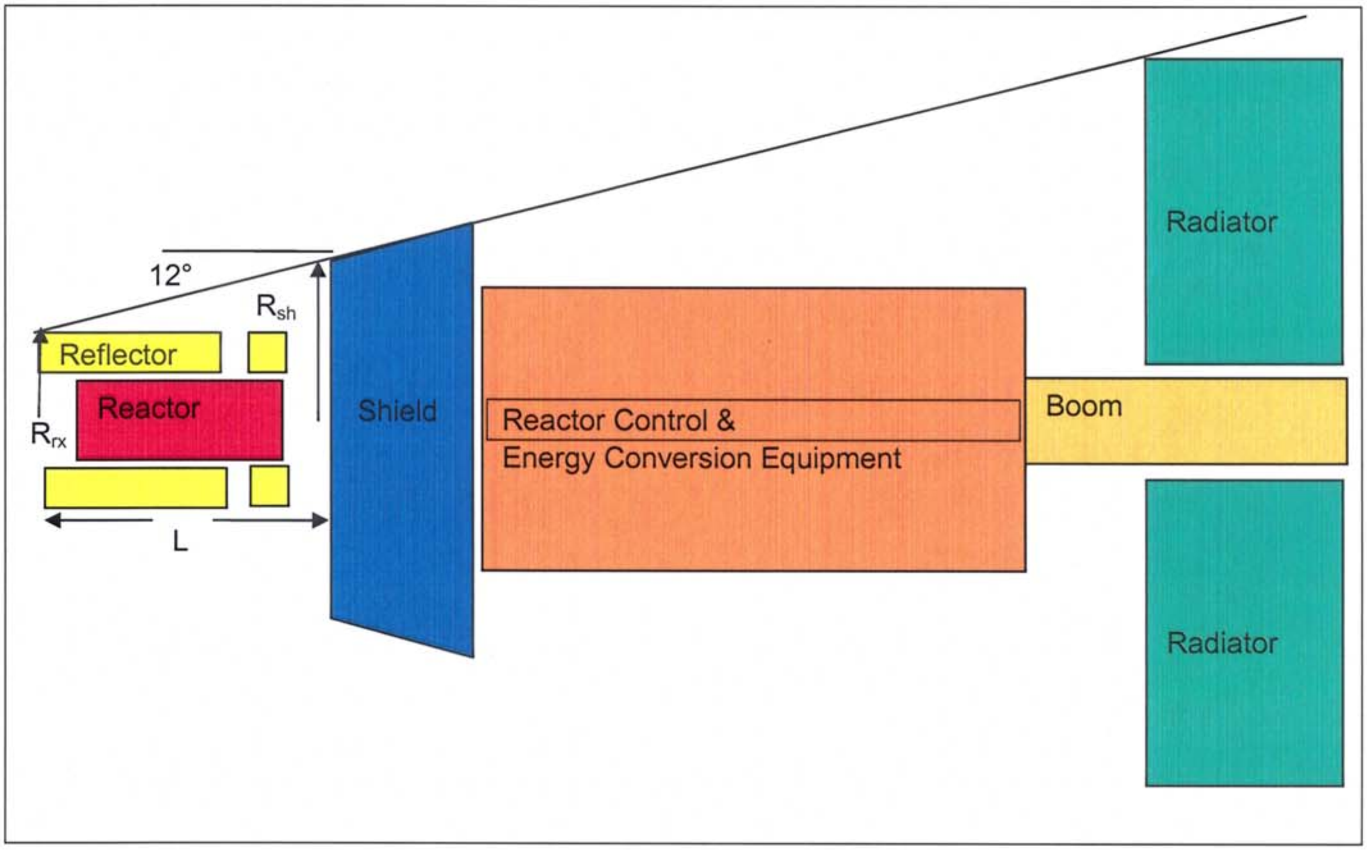


**Figure 4-2 Example (12° direction) of Shield Size Projection from the Reactor (not to scale)**




## 4.1 Issues and Challenges

Shielding the mission module from reactor radiation is a trade-off between reactor shield mass, radiation hardening capability of electronics, and local shielding from the solar system radiation environment. The challenge is to develop a "mass-efficient" reactor shield design; that is, the most effective shield while minimizing mass. The NRPCT had narrowed the slate of candidate shield materials to five primary materials [lithium hydride (LiH), beryllium (Be), boron carbide ($B_4C$), tungsten (W), and water ($H_2O$)] and one alternate material ($^{10}B$ metal). Each results in a different impact on the shield design relative to mass and complexity of design. Design issues include thermal growth, swelling, and thermal management requirements. In addition, there are other programmatic considerations such as schedule, reliability, cost and toxicity.

Shielding benchmark testing was planned to confirm shielding calculation tools and cross sections for both radiation attenuation and heat deposition. The benefits of this testing would be to reduce shield mass to confirm that the shield will meet radiation and temperature limits. Development and execution of this test needs to be early in the design process.

Another challenge to the shield design is incorporating it into the overall plant arrangement. Hot and cold leg piping must go either through or around the shield in order to allow for core cooling and energy transport. Piping through the shield results in radiation streaming paths, while piping around the shield results in neutron scatter to the payload. Similar issues exist for the reactor control device mechanism penetrations.

High gamma levels on the outside of the reactor could drive electrons from the surface which could significantly contribute to spaceship electrical charging and beta radiation that could go around the shield. Further characterization, testing, and mitigation of this should be pursued in future studies.

## 4.2 Summary of Work

Preliminary reactor shielding development focused on:

- Understanding materials sensitivities in the design
- Understanding the impact of control drive mechanism penetrations and piping cutouts on shield effectiveness
- Estimating the overall mass of the shield
- Estimating the shielding required for the ground test reactor

NRPCT and ORNL investigated dozens of shielding materials in many combinations for neutron and gamma shielding effectiveness, mass effectiveness, gas generation, heating rates, thermal requirements and capabilities, structural properties, deliverability, safety, toxicity, and cost. Results of these studies are documented in Reference (15). A simple shield mass estimating code was developed and used to study relationships among various reactor and spaceship parameters and shield mass. Preliminary mass and thermal analyses were made for several shield concepts. Investigations were made to begin optimization of large pipes and reactor control arrangements through the shield and incorporation of nuclear instrument detectors into the shield. Investigation was also underway to determine tradeoffs between nickel and stainless steel structural materials and design impacts of varying gamma limits relative to neutron limits. From these investigations, alternate shield configurations were developed. Investigations were also made of radiation levels from various reactor/piping structural materials, from the reactor following zero power critical testing, and of preliminary Ground Test Reactor shielding requirements. Scoping studies were conducted to determine the effect of fission product retention levels on the shield design and on the possibility of radiation induced reactor electrical charging. Testing of various radiation transport codes for neutron, gamma and charged particle transport was begun.

## 4.3 Key Findings and Perspectives

Based on analyses performed during the pre-conceptual design phase, the NRPCT has reached a number of technical findings:

- The diameter of the reactor and the distance between the shield and the far end of the reactor assembly have the greatest impact on shielding mass. Minimizing core volume and core-to-shield distance have a bigger impact on the overall Reactor Module mass than minimizing reactor mass at the expense of reactor volume.
- Based on shielding scoping evaluations, the leading material options include lithium hydride, water, beryllium, and boron carbide, for neutron or combined gamma/neutron shielding. Lithium hydride-based neutron shielding, which require Be/$B_4C$ for the high flux portion of the shield, provide a slightly lower overall mass than concepts without lithium hydride. However, a full Be/$B_4C$ shield is considered to be a lower cost, lower risk option. A water-based shield is mass competitive with lithium hydride if the pipes can be routed around the shield, but LiH would still be required for shield caps behind where the pipes reenter the shadow cone after the main shield. A water-based shield with pipes through the shield will be heavier than a LiH-based shield. A water-based shield may require heating to prevent freezing prior to start-up, and cooling to prevent boiling during operation. A LiH shield may require heating during low

power (coast) modes. Reliability of a vessel used to contain either a lithium hydride or water shielding system will require further evaluation. Tungsten is considered to be the best shield material for strictly gamma shielding based on its high shielding effectiveness, low toxicity, and moderate cost.

- Gas pipe streaming can be controlled by spiraling the piping through channels in the outer surface of the shield. The overall shield thickness must be increased to retain the same effectiveness as a system without large gas piping. Additional shielding on the loop piping located beyond the pipe penetrations of the main shield were considered to reduce neutron streaming, but were not necessary. Comparative studies involving routing of piping around rather than penetrating the shield had not yet been performed.
- Shielding model results showed that the control drive mechanism penetrations can be made without substantial impact on the shield effectiveness.
- The effectiveness of the shielding remains adequate even for unanticipated high rates of fission products release (up to ~$10^{-3}$ release to birth ratio of volatile fission products) from the fuel into the gas coolant. Damage to the alternator winding insulation and sensors due to accumulation of fission products will require further study and may create a more restrictive limit on fission product release.

# 5 PLANT SUMMARY

## 5.1 Issues and Challenges

Designing, building, and operating a low-mass, long-life, high-temperature power plant that operates in a hostile environment (high radiation, high vacuum, low temperature), requires no maintenance, and is highly reliable is a challenge:

- Mass and Volume – Maintaining mass and volume that are within the capability of the launch vehicle and that meet the spaceship mass, power, trajectory solution requires optimization within the SNPP as well as with the remainder of the spaceship. There are many trades that will impact mass, typically trading larger design space or less development risk for higher mass. Reliability and redundancy trade-offs will impact system complexity and mass.

- Performance and Reliability – The required design performance for major plant components (recuperator, gas cooler, Brayton turboalternator, valves, bearings, and piping) are within the bounds of current technology. However, designing these components to be leak tight for the long mission duration remains a development item. The largest turbine component design uncertainties are the scale-up of the turboalternator assembly for a single 200-kWe Brayton system and the ability of the turbine to operate for the mission lifetime in the operating environment. Further rotor dynamic evaluations, bearing development and alternator cooling studies and testing would be needed.

- Close coupling of the reactor and plant – The Brayton compressor is also the reactor coolant pump, which complicates startup and reactor plant transients. Changes in plant parameters affect both the plant and the reactor performance requiring a coordinated approach toward optimization of the overall system.

- Testing – To support the development and deployment of an SNPP, extensive material, component, and system testing had been planned. This testing would need to be a coordinated effort leading up to prototypical nuclear and non-nuclear testing (see Section 7 for additional discussion on integrated system testing).

- Materials - Selection of pressure boundary materials is challenging. Nickel-base superalloys are considered as leading candidates for the reactor vessel, loop piping, turbine, and heat exchangers. Specific concerns were thermal creep and irradiation embrittlement for the pressure vessel at operating temperatures. Mass transport between refractory metal core components and the pressure boundary components may limit plant lifetime. An alternative is the use of refractory metal alloys for pressure boundary components, but these alloys required substantial testing and development to mitigate inherent risks. Adequate protection from contamination from reactive elements such as carbon and oxygen would have been required.

- Reliability Data - Data is not and will not be available prior to launch for SNPP components. Data is available for similar components, but service life and environment is significantly different. Limited test results may be available, but not in statistically significant quantities.

## 5.2 Summary of Work

Reactor plant development work focused on a disciplined engineering approach to understand overall system characteristics and performance for a variety of system architectures (See Figure 1-4 for examples of system arrangements that were evaluated). Establishing this basic system understanding, in parallel with developing functional requirements for the Reactor Module would have led to spaceship trade studies and the selection of the Reference system architecture and heat balance. At the time of project termination, system optimization work had not yet been initiated. Trade studies would have included items such as radiator size vs. operating temperatures, mass vs. redundancy, and boom length vs. shield mass. This work is documented in Reference (4). Evaluations completed include:

- Potential system architectures and fluid schematics were developed. These evaluations included comparisons of the potential systems, preliminary location of valves (as required) and operating strategies. Preliminary system arrangements were also developed, resulting in comparative piping pressure drops in the system and identifying potential packaging issues related to the size and number of components. Design iterations relating arrangements, pressure drops, piping stress analyses, and heat balances had not yet been initiated.

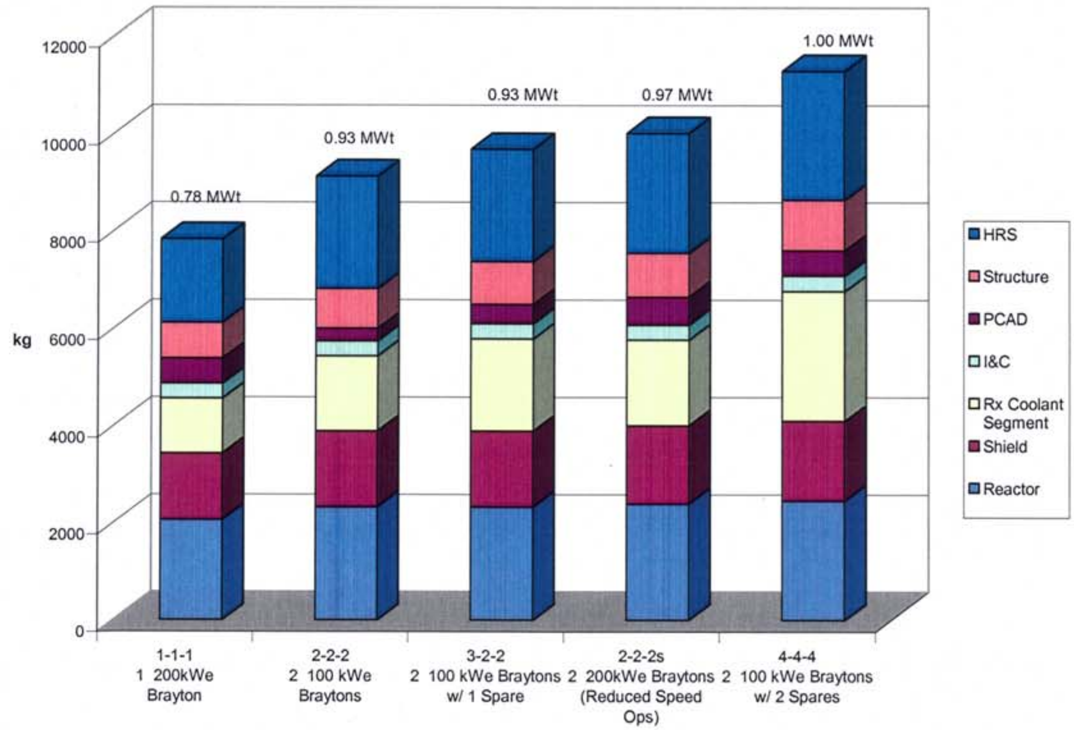
- Comparative system steady-state heat balance studies and parameter sensitivity studies were performed for a range of SNPP system architectures. These evaluations were performed to establish the range of plant operating parameters needed to satisfy the system functional requirements. The results of these studies are used to establish SNPP component sizing and operating conditions, enable comparison of candidate system architectures, and initiate trade studies. A number of sensitivity studies were performed to determine the effect of parameter variations on the plant's overall heat balance and other parameters of interest. Of particular interest was the effect of parameter variation on required reactor thermal power output, required radiator area, and reactor inlet temperature. The overall mass of the SNPP is minimized when reactor thermal power output and required radiator area are near minimum values as the mass of the reactor, the reactor radiation shield, and the heat rejection segment dominate the overall SNPP mass. Minimizing the required thermal power output of the reactor has many benefits as the mass of the fuel load, reactor structure, and reactor radiation shield decrease with required reactor thermal power output. The effect of parameter variation on reactor inlet temperature was of interest since concerns associated with material performance at high temperatures are alleviated as reactor inlet temperature is reduced. Sensitivity studies were performed for the following plant parameters: converter loop piping diameter, reactor outlet temperature, compressor inlet temperature, compressor outlet pressure, Brayton turbine and compressor efficiency and HRS heat pipe operating temperature

- Potential power plant transients were identified as a precursor to Design Events development. Transient reactor plant response analyses were performed to study the behavior of the primary plant and the heat rejection segment (HRS).

- All reactor plant components including the turboalternator, recuperator, gas coolers, piping, and valves were evaluated. The evaluations included a review of the current state of component development, independent evaluation of component performance, and initial concept design of the hot leg piping and the gas cooler.

- Overall system mass estimates and mass sensitivity studies were performed. These preliminary mass studies are for comparative purposes (since optimizations hadn't been started) and provide insight into the relative masses of each system architecture (see Figure 5-1). These results were combined with the results of a system reliability study, as shown in Figure 5-2which illustrates that system architectures consisting of more than two Brayton units have both a mass and reliability penalty.

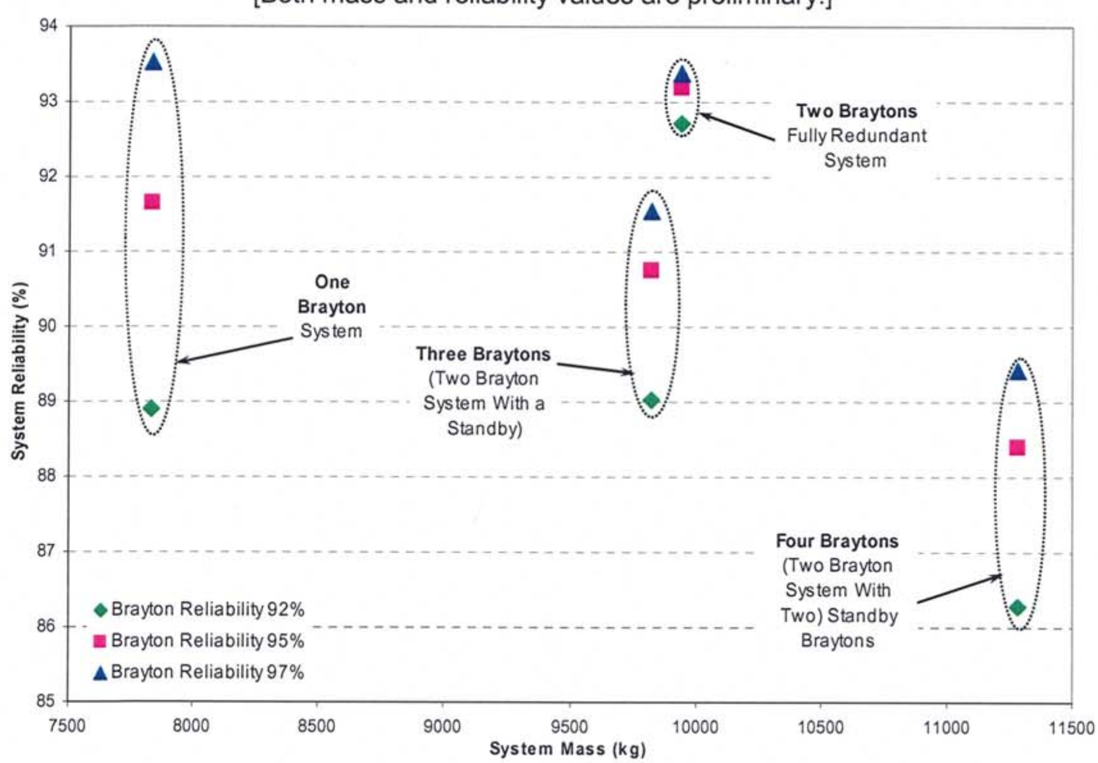
## Figure 5-1: Mass Comparison for Key System Architectures




Notes:
1. "s" designates partial load control using a combination of speed and temperature control.
2. Mass estimates do not include the Aerothermal Protection Segment or Thermal Management Subsystem masses
3. Best-estimate mass values do not include uncertainty or design growth allocations

## Figure 5-2: SNPP Reliability versus System Mass

[Both mass and reliability values are preliminary.]



- Preliminary evaluations of Reactor Module thermal management were performed to evaluate piping and component support requirements and to evaluate piping and component heat loss rates. These studies were undertaken to scope out the magnitude of these thermal related issues.
- Preliminary evaluations were performed to assess the relative reliability of various system architectures. Emphasis was placed on evaluating the reliability of systems with one, two, three, and four Brayton loops to support plant design decisions. Individual component reliabilities were based on engineering experience with very limited data. Sensitivity studies revealed the impact on overall system reliability due to a change in the reliability of a given component. This can help determine which components require the most attention during development and testing to ensure the mission is successful.
- A preliminary operational strategy was developed for the Direct Gas Brayton space nuclear power plant. General methods of plant operation and control that will allow it to most effectively meet its mission requirements within the constraints of the space application are identified.

Various system architecture configurations were evaluated with respect to performance, operating strategy, mass, and reliability. Some system architectures were being evaluated in which the PCAD system would be designed to operate producing full rated power at multiple frequencies. This capability would permit a mode of operation where each of two or three turboalternators are normally operating at less than their rated capacity and jointly produce the full rated system power by operating at a reduced turbine inlet temperature and rotational speed. In the event of failure of one turboalternator, full rated system power is restored by increasing the turbine inlet temperature and rotational speed of the remaining unit(s) such that each turboalternator operates at its full rated capacity. This concept was early in development and steady state analysis had not yet been performed.

The base case reactor vessel material for Prometheus is a Ni-based superalloy, which is cooled by the reactor inlet flow. A design temperature of ~900K is near the limit for acceptable creep allowance. Plant operation is envisioned to monitor and control several system parameters including the turbine inlet temperature, controlled by reactor reflector motion, and Brayton unit speed (shaft rpm), controlled by the Parasitic Load Radiator. Preliminary steady state heat balances show that achieving a system performance within the originally envisioned design space is challenging. Limiting the heat rejection area to 450 $m^2$, maximum heat rejection heat pipe temperature to 500 K and maximum normal reactor exit coolant temperature to 1150 K can only be achieved in plant configurations where one Brayton unit normally provides the total electrical output given the preliminary piping system arrangements and non-optimized plant conditions that have been developed thus far. Allowing some increase in converter loop piping diameter, radiator area, heat pipe temperature, or allowing less margin for reactor temperature uncertainty would be required for plant configurations where two or more Brayton units normally provide the total electrical output.

## 5.3 Key Findings and Perspectives

System Architecture Impact on Reliability and Mass

- For the reactor plant, primary concerns were minimizing mass while meeting the power output requirements and demonstrating reliable operation for the long-duration of the deep space missions. Although spaceship mass allocations were not yet established, Project Prometheus was iterating toward a solution of mass, power, and mission duration for the JIMO mission. Comparisons of the reliability of plants having from one to four Brayton loops indicate that having one or two loops would likely provide a more reliable plant than having three or four loops. These reliability analyses are comparative at this stage and cannot be used to determine the absolute value for a given plant configuration.

- Based on NRPCT recent evaluations of each component within the SNPP, the preliminary mass predictions are higher than NASA's for most parts of the plant. These findings would contribute to consideration of less redundancy (but potentially more reliability) among energy conversion loops, which NASA had baselined as four in prior spaceship studies (see Figure 5-2). A decision on the appropriate number of Brayton loops was not made during this project closeout process. This decision would have involved JPL and the spaceship designer to include overall spaceship mass trades, evaluations of alternate methods to accommodate angular momentum, parameter optimizations for each arrangement, and other aspects of spaceship and mission integration which were not done prior to project termination.

System Architecture

- A single Brayton system offers the simplest design, the least required component development, and the simplest plant operation. The single Brayton system also has the lowest mass and the highest thermal efficiency. The extra capability could not be quantified until plant arrangements and parameter optimization studies are completed. Plant parameters that would be part of an optimization include HeXe coolant mixture, turbine speed, coolant temperatures at each heat exchangers, system pressure, and compressor pressure ratio. The single Brayton concept should be considered along with other system architectures, although it would require deviation from the single point failure tolerance criteria. An additional momentum compensation system would be required for any spaceship architecture that does not have counter-rotating turbines, including the single Brayton system.

- To meet the single failure tolerance criteria, multiple Brayton units would be required, resulting in a higher mass and an overall thermal efficiency decrement relative to a single Brayton system. NRPCT and NASA experience indicates the prudence of having redundant components, even if they have demonstrated reliability, to ensure that manufacturing defects, human error, or an unexpected event does not lead to a mission ending failure. For those components where redundancy is considered impractical, exceptions to the single point failure avoidance requirement is provided. For the currently envisioned Prometheus spaceship, the reactor, the reactor coolant loop, the boom, and the xenon propellant tank all require such exceptions. Some critical elements of a single Brayton system could have redundancy built in (e.g., alternators); others, such as turbine bearings, cannot have redundancy built in. Exceptions could be provided for a single Brayton system, but the reliability of operating high-speed equipment over an extended lifetime with little testing under service conditions must be considered.

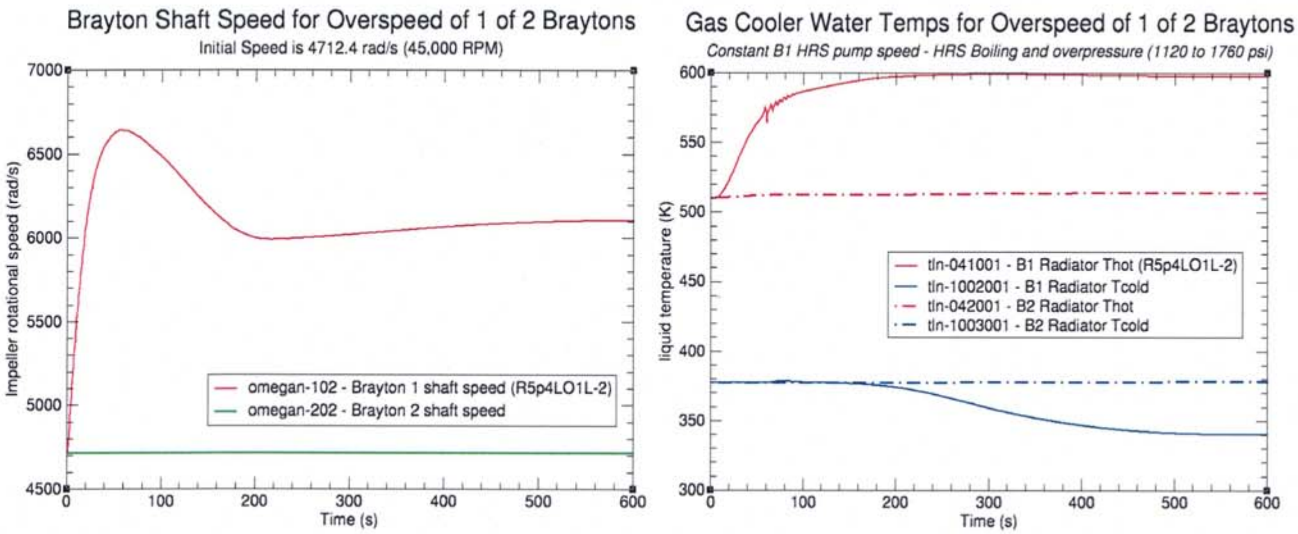
- The determination of the most reliable system layout depends on the reliability of the constituent components. A two Brayton system in which both Braytons are normally running but each could, upon failure of one Brayton, supply full power to the spaceship would probably be the most reliable system. This system would have a mass of ~2000 kg greater than the single-Brayton system. Assuming Brayton components could be developed with a demonstrated reliability above a certain breakpoint (approximately 97% using the assumed values in the NRPCT study), the most reliable system would be the single-Brayton system. However, for systems with Brayton assembly reliability less than this breakpoint, a second redundant Brayton would result in a higher overall system reliability, offsetting the impact of additional components and increased complexity. System architectures with three or more Braytons (see Figure 5-2) had lower overall reliability. This reliability was reduced because of additional welds, valves, and surface area vulnerable to leaks to space and/or micrometeoroid impact. Because the direct gas Brayton system is significantly different than any other in existence, the decision on plant redundancy will have to be made without specific data on component reliability, and the envisioned test program will not be sufficient to establish a true statistical basis.

- Actions to minimize mass had not yet been undertaken. These actions would have included optimization of the system arrangement and heat balance; trades on the reactor shield, mission module shield, and boom length; selection of the most appropriate reactor configuration; selection of materials and specification of their design bases; trimming of the reactor shadow shield configuration based on established spaceship configuration; etc.

Plant Operation and Plant Dynamics

- Reactor plant dynamic models were developed using three different modeling tools (Simulink, RELAP5-3D, and TRACE). Model results show that a multiple closed Brayton unit system is feasible with no apparent system instabilities. However, parallel operation of multiple Braytons in a closed loop has never been done and substantial testing would be needed to further demonstrate feasibility and validate the models. Preliminary model results indicate that the HRS is more vulnerable to exceeding design limits than other components in the SNPP including the reactor. The concern in a water-cooled HRS is coolant over-pressurization, leading to a potential gas cooler failure, as shown in Figure 5-3. In a NaK-cooled HRS, the coolant is not over-pressurized, but the load-carrying capacity of the heat pipes may be exceeded. Further development of the SNPP would require a control system action to prevent HRS damage.

**Figure 5-3: Complete Loss of Electrical Load and Impact on Water-cooled HRS**




- Utilization of two distinct power levels may allow for the reduction of system temperature and/or reactor power for significant portions of the mission, extending spaceship life. Two distinct power levels were envisioned for the JIMO mission: high power (thrusters operational, ~185-kWe load) and low power (thrusters secured, ~40-kWe or less load). Low power would include periods when science data is being collected and may require compensation for the angular momentum induced by the Brayton system on the spaceship. This is most efficiently accomplished by a combination of speed control and temperature reduction. Use of on/off control of Braytons and use of gas inventory control would require increased system complexity and would not significantly improve performance. Reactor material performance may be affected by lowering of reactor temperatures and must also be considered.

Design Space

- Cycle analysis shows that achieving a system performance within the originally envisioned design space is challenging. Limiting the heat rejection area to 450 $m^2$, the maximum heat rejection heat pipe temperature to 500 K, and the reactor coolant outlet temperature to 1150 K can only be achieved in plant configurations where one Brayton normally provides the total electrical output given the preliminary piping system arrangements and non-optimized plant conditions that have been developed thus far. Allowing some increase in converter loop piping diameter, radiator area, heat pipe temperature, or allowing less margin for reactor temperature uncertainty would be required for plant configurations where two or more Brayton units normally provide the total electrical output.

- A key driver to overall plant efficiency is the arrangement of the converter loop piping system and the resulting impact on piping system pressure drop. For plants with multiple converter loops, flow splits in the reactor inlet and outlet headers and inclusion of valves become significant additions to the overall piping system pressure drop. Large diameter pipe, few valves, low pressure drops through components, simple pipe runs, and large gentle bend radii are required to minimize loop pressure drop. However, these considerations will need to be balanced against the need to make the plant sufficiently flexible to accommodate thermal transients and the need to make it fit within the available volume.

- Allowances for off-design parameters (e.g., temperatures and pressures) need to be allocated early in the conceptual phase of the project to account for component degradation over service life, transient performance, casualty recovery, instrument error, operating strategy, and associated operating bands.

Material Selection

- Use of refractory metal alloys for pressure boundary components were considered, but these alloys required substantial testing and development to mitigate inherent risks. Leading concerns included irradiation embrittlement, interstitial embrittlement from the absorption of working gas impurities and the integrity of dissimilar metal joints. Adequate protection from reactive elements such as carbon and oxygen would have required vacuum facilities exceeding those necessary for a nickel-base alloy pressure boundary. Similar complications would exist for extensibility to Lunar or Mars surface applications. Irradiation embrittlement would have made thermal cycling of a ground test unit a concern.

- Nickel-base superalloys were considered as leading candidate materials for the reactor vessel, loop piping, turbine, and heat exchangers. A large property, component manufacturing and performance database exists for Ni-base superalloys with the majority of the data obtained for air-breathing turbine engine applications. However, significant testing and development was required for the Prometheus design. Specific concerns were thermal creep, irradiation embrittlement for the pressure vessel, chemical interactions with the working gas and working gas impurities and the integrity of dissimilar metal joints.

- Demonstration of dissimilar metal joining feasibility, including cast to wrought nickel-base superalloys, wrought nickel-base superalloy to titanium alloys, wrought nickel-base superalloy to various refractory metal alloys, and possibly, stainless steel to titanium alloys will be required prior to the selection and specification of materials for all major system components. If dissimilar metal joints are required, development of a sound joint for the material pairs would be required.

- A thorough understanding of environmental degradation issues relevant to the application of all candidate material types in a common system, including the implications of the heat rejection system materials, must be obtained prior to selection and specification of materials for all major system components. Issues included potential degradation from the space vacuum, fission products, impurities in the working gas and the transport of impurities from one material to another in the common gas loop.

- To support the development and deployment of an SNPP, extensive material, component, and system testing had been planned. This testing would need to be a coordinated effort leading up to the potential nuclear testing of a ground based prototype and non-nuclear testing integrated with the spaceship leading up to launch. It should be noted that no final decision on the need for a GTRF or where it should be sited had been made; preliminary engineering work had started to support the National Environmental Policy Act (NEPA) process, establish scope, and meet the schedule. At the time of termination of NRPCT participation in Project Prometheus, the functional requirements for the facility were preliminary and had not yet been reviewed against federal requirements, peer reviewed, or approved by Naval Reactors.

<u>Components</u>

- The designs for major plant components (recuperator, gas cooler, Brayton turboalternator, valves, bearings, and piping) are within the bounds of current technology assuming non-refractory materials are used. Designing these components to be leak tight for the long life of the deep space missions remains a development item.

- The largest turbine component design uncertainties are the scale-up of the turboalternator assembly for a single 200-kWe Brayton system and the ability of the turbine to operate for an extended lifetime in the service environment. Further rotor dynamic evaluations, bearing development, and alternator cooling studies and testing would be needed.

- Reactor Module thermal management must be designed to reject sufficient heat from the hot components while maintaining other components at acceptable temperatures. Thermal management is complicated by structures around the plant that are needed to provide micrometeoroid protection and by the need to provide sufficient heat before start-up to maintain components at acceptable temperatures. Identification and testing of materials, coatings, finishes, etc., will be needed to identify options that will satisfy requirements after prolonged exposure to radiation and temperature. Thermal management must be integrated with the power plant design and should be considered early in conceptual design.

- Further systems integration between the heat transfer segment and the Reactor Module is warranted to establish the design trade space available to reduce mass, spaceship size, and project risk. Further integration will affect selection of heat transport loop coolant and ducting materials that interface with the gas cooler, start-up and normal operational strategy, radiator mass versus temperature capability, heat load capability, and radiator size. Lifetime degradation in the radiator performance due to changes in effective emissivity and isolated heat pipe failures will lead to a slow increase in radiator temperature. High temperature water heat pipe performance and life capability for this application is not fully established. The relatively high temperature (500K) and evaporator surface heat flux (10 $W/cm^2$) and lifetime are beyond established water heatpipe performance.

# 6 INSTRUMENTATION AND CONTROL SUMMARY

Once the JIMO spaceship had been launched into space, real time operation of the space reactor would be performed by the reactor Instrumentation and Control (I&C) Segment. While Earth generated commands might enable certain events and operations to begin, it would be the I&C Segment that would monitor the state of the reactor plant, and provide for control of the plant through life. The Reactor I&C Segment would also be responsible to assess the condition of reactor plant equipment, respond to emergent events, (foreseen or unforeseen), ensure the reactor at all times remained within its operational envelope and that the power required for the spaceship would be available without interruption. The I&C Segment would accomplish these objectives over a lifetime of fifteen years, largely autonomously with only infrequent contact from Earth controllers.

## 6.1 Issues and Challenges

The high-temperature reactor plant, the space radiation environment of Jupiter, the 15-year mission, the lack of a backup power source aboard the spaceship, the limited remote communications with the spaceship, and the aggressive schedule to deliver prototype and flight versions of the space nuclear power plant represented significant technical and programmatic challenges for the development and delivery of the Instrumentation and Control equipment supporting the Prometheus program.

The Reactor I&C Segment would be made up of a diverse set of technologies and equipment including control element motors and sensors that are in close proximity to the reactor, electronics and computers located in the shielded vault, software that would monitor system behaviors and define its actions, and power conversion modules that would drive the reactivity control motors. The I&C Segment encompasses all of the reactor sensing, instrumentation, and control equipment aboard the spaceship as well as the reactor Ground Control station on Earth. It is through this system, that the reactor plant would remain linked to its Earth-bound observers, allowing them to evaluate its operational parameters, tune its performance, and adjust for emergent conditions or events.

<u>System</u>

The space Reactor I&C Segment would face many new and unique challenges compared to its terrestrial predecessors. It would operate a nuclear reactor in space without the direct presence of a human operator while traveling through some of the most difficult environments in the solar system. It would have to operate the reactor plant flawlessly for more than a decade to allow the spaceship to accomplish its mission and goals. For the I&C Segment to meet these challenges, significant advances would be necessary in the key technology groups supporting it. New technologies would be required for the sensors and electronics to meet the harsh environments of the reactor and space. New approaches to system architecture and control would be necessary for reliable, fool-proof control and sensing without a human-in-the-loop. New software approaches would have to be developed to provide detailed process tracking, to mitigate the impact of emergent system faults, and to allow system adjustments over the life of the plant. These challenges would have to be met in order to provide the pedigreed space reactor I&C system demanded for the Prometheus SNPP and JIMO mission.

One of the first major efforts associated with the development of a SNPP I&C Segment was the development of a preconceptual I&C Segment architecture that addressed some of its most important fundamental challenges.

The I&C Segment would have required a flexible system architecture to accomplish autonomous reactor control for the duration of the mission with no ability to upgrade or repair hardware post launch. Additionally, the system architecture design would have provided the ability for software upgrades post launch and provided for single or multiple slider (or drum) control. Moreover, the architecture would have been extensible to a ground-based prototype of the SNPP whose requirements may have differed in some significant ways from the flight unit.

The I&C system architecture would need to meet the mission's single fault requirements within tight mass and volume constraints. A single fault cannot be allowed to either shut down the reactor and cause a loss of spaceship power or cause the reactor plant to exceed its design limitations. The I&C system architecture would also have accommodated the performance of fault detection, containment, and recovery from design basis events without interaction with Ground Control.

The I&C system would have been fully integrated with the spaceship Command, Control, and Data Handling subsystem to provide for operation of the power plant as commanded by the spaceship, to fully integrate the Reactor Module's operation with the spaceship's electrical Power Conditioning and Distribution (PCAD) subsystem, and to provide for the flow of information and communications between the Reactor Module, the spaceship and Earth.

<u>Sensors</u>

Reactor plant operations demand sufficient sensors to provide for the control and monitoring of the reactor for all the modes of plant operation. These modes include:

- Startup
- Constant power operation (steady state power operation)
- Power transitions
- Casualty Operations
- Maintenance and Testing Operations
- Shutdown

These operations must be successfully performed under normal as well as off-normal conditions. To provide for all these conceivable situations a sufficiently diverse set of reactor plant sensors would have been required to ensure that the control actions taken by the I&C Segment were always consistent with the actual state of the plant and its mode of operation. Accordingly, sensors for measurement of the reactor coolant temperatures, flow(s), and pressure; neutron flux reactor power; positions of the reactivity control elements were established as the baseline set for the SNPP reactor. The highly integrated nature of the direct gas Brayton plant also demanded that parameters associated with its Brayton units and its PCAD subsystem be measured as well (e.g. Brayton unit speed, and the power, current, voltage, and frequency of its alternator, Parasitic Load Radiator (PLR), and Start Inverter).

For the SNPP sensors, the requirements would have been exceptionally demanding and the conditions in which many would have operated quite severe. The necessity of a low mass SNPP would have required that the reactor plant be operated at the highest temperature possible within the material limits of the plant to maximize its efficiency. This would lead to the need for narrow control bands whose width was primarily limited by the accuracy of the sensors and the small discrete actions of the reactor controls. Therefore, exceptional accuracy was demanded of the controlling reactor parameters for the duration of the flight. For the most desirable control parameter of the plant, reactor coolant outlet temperature, this meant sustaining a ±5 K or better accuracy while measuring a 1150 K temperature continuously for 15 years. For the reactor inlet temperature, the requirement was only slightly less daunting. Additionally, the necessity of reactor coolant system integrity to mission success drove a strong desire to limit the number of penetrations of the reactor coolant system boundary by the temperature, flow, and pressure sensing elements. Conventional sensor technologies were unable to meet these demanding requirements and goals for even a few years. Identification of advanced sensor technologies was a necessity for the SNPP program.

The environmental conditions (including radiation exposure) for the SNPP sensors, cables, and connectors would have presented additional design challenges. These factors significantly influenced considerations for sensor placement. The total radiation exposure experienced by these components would have been a combination of the radiation produced by the reactor plus that encountered throughout the mission from natural sources in space. The combined total ionizing dose (TID) forward of the reactor shield was estimated to be $5.5 \times 10^9$ rads(Si) ($4.5 \times 10^9$ from the reactor and $1 \times 10^9$ from the space environment). Just aft of the reactor shield, the combined TID was estimated to be $1.2 \times 10^9$ rads(Si) ($2 \times 10^8$ from the reactor and $1 \times 10^9$ from the space environment), showing a reduction in the reactor radiation afforded by the reactor shield. However, a design limit of $0.5 \times 10^9$ rads(Si) TID was initially established for the most sensitive elements of the I&C components located in the Reactor Module region, i.e., electrical insulation materials in the sensors, cables, and connectors. This design limit favored sensor placement aft of the reactor shield and would have required the installation of additional shielding (incorporated with the micrometeoroid protection) to attenuate the radiation from space sources.

Electronics

All sensor and computational electronics would reside aft on the spaceship in a shielded, thermally managed electronics vault. The major electronics challenge would have been to ensure reliable operation over the duration of the mission in the high-radiation environment of the JIMO mission. JPL had identified the expected contributions from the SNPP reactor (internal) and from space (external) in a set of requirements for ionizing and neutron equivalent dose. The internal SNPP reactor induced radiation allowances within the vault were set at $2.5 \times 10^4$ rad(Si) (gamma) and $5 \times 10^{10}$ n/cm$^2$ 1-MeV equivalent silicon damage (neutrons). The space environment radiation allowances within the vault were estimated to be $2.63 \times 10^5$ rads(Si) and $4.4 \times 10^{11}$ n/cm$^2$ 1-Mev equivalent neutrons. These radiation exposures assumed a baseline vault shielding provided by 1000 mils of aluminum, and yielded a combined total radiation exposure (reactor plus space sources) of $2.96 \times 10^5$ rads(Si) and $5.7 \times 10^{11}$ n/cm2 1-Mev equivalent neutrons. Applying a radiation design factor (RDF) = 2, the resulting qualification requirements for electronics in placed in the vault would have been TID = $5.92 \times 10^5$ rads(Si), and displacement damage = $11.4 \times 10^{11}$ n/cm$^2$ 1-Mev equivalent neutrons [Reference (361)]. At the time of project termination, JPL was also considering the qualification of electronic components to a TID of $1 \times 10^6$ rads(Si) as an additional project design goal.

Software

The dynamic response of a nuclear reactor plant, the requirement for continuity of power, the remoteness of spaceship operation, and long periods without direct communication with Earth all required the use of a robust software component in the Reactor Module I&C Segment. This included not only the flight software used on the spaceship for reactor control and fault protection but also the software used for telemetry and command functions in Ground Control. Communication delays and the fast response of the nuclear power plant leave no margin for error or intervention from Earth. Additionally, the software must be tolerant to single event upsets from high energy particle radiation. Thus, a very robust software development and testing program was planned for the Prometheus Project.

Controls

Terrestrial nuclear power plants employ human control and decision making for operations and benefit from periodic maintenance. In contrast, the SNPP control system for a deep space mission would have needed to provide continuous operation for the mission duration with limited human interaction and no opportunity for hardware maintenance or sensor calibration. The SNPP would have to independently operate the power plant while maintaining power production even when subject to off-nominal conditions or component failure. This capability would have been essential because it would not be possible to rely upon continuous, immediate human interaction for control or maintenance of the plant. The spaceship's journey to Jupiter would have been generally autonomous in nature with only limited periodic contact with Earth except when totally precluded by planetary occlusion.

While remote, infrequent communications with Earth mandated autonomous control of the plant, the specific frequency and nature of the controls both in normal and off-normal conditions would be driven by a number of considerations. Constant power operation requires that plant efficiency, off-nominal performance, material and component performance limitations, electrical power quality, and casualty response all be addressed.

The mass of the reactor plant must be limited to support the launch of the spaceship; yet, the reactor must also meet the electrical power demands of the spaceship throughout its journey to support continuous propulsion. The direct gas Brayton reactor plant is a tightly coupled plant whose efficiency is a strong function of reactor temperature and is significantly affected by even small variations in its value. Moreover, the efficiency variation with control band width is expected to be similarly mirrored in its off-nominal performance. This suggests that reactor parameter control bands must be maintained as small as practical.

The materials and component limitations must be accommodated by any SNPP control system. The plant systems considered for the SNPP would require that any control system maintain the plant within an operating envelope with sufficient margin to ensure component integrity and operability under both normal and abnormal conditions. The design temperature limitations for the reactor coolant system boundary materials, the operational temperature limitations on the heat rejection system radiator fluids, the limitation of reactor power to 100%, and the maintenance of Brayton alternator electrical frequency at 2250 Hz, all exemplify the type of operational limitations that must be managed by the SNPP controls.

There would have been no alternate power source aboard the JIMO spaceship once the SNPP had been brought on-line and the solar powered support system for reactor startup jettisoned. Electrical power for plant and spaceship operation would have relied entirely upon the maintenance of reactor power at a sufficient level to meet the minimum needs of the spaceship.

The controls would have been required to carryout complicated sequences of operations, some while in communication with Earth and some without Earth support. An example of the former would be the startup process for the SNPP early in the mission; an example of the latter might be the shutdown of degraded Brayton unit and the startup of a standby unit to support power continuity to the spaceship while in deep space.

## 6.2 Summary of Work

Develop a Preconceptual Reactor I&C Segment Architecture

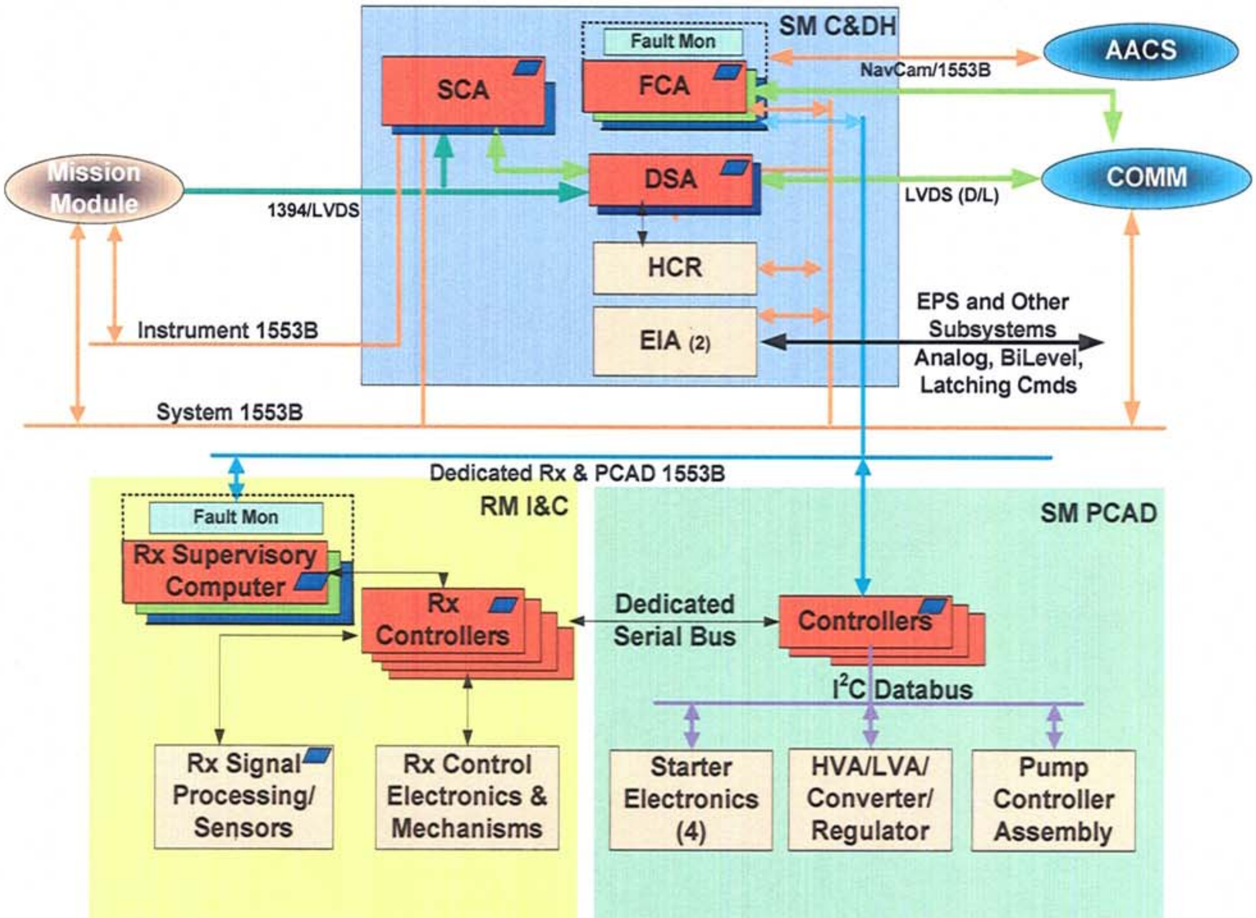
The pre-conceptual I&C Segment architecture was a key starting point for development. This architecture was intended to set the technical approach for the entire I&C Segment. It defined interfaces to other spaceship systems, defined hardware blocks for future development, and provided a basis for accurate cost and schedule estimates. Since the reactor system requirements were not known at the start of development, it was anticipated that the architecture would evolve as the design of the Reactor Module was matured. Figure 6-1 depicts the preconceptual reactor I&C architecture and its interfaces to other important spaceship subsystems.

A trade study was performed on the use of a shielded multiplexer at the forward end of the spaceship immediately behind the reactor shield. The SP-100 I&C system architecture used a multiplexer to reduce cable mass. However, the trade study showed that implementing a secondary electronics vault closer to the reactor would not be practical for the JIMO mission. The increase in mass for shielding would outweigh the increase in cabling required without multiplexing signals. Thus all electronics in the selected architecture would reside in the main spaceship electronics vault.

This architecture was selected from several different alternatives for its relative simplicity and fault tolerance. The selected architecture used a 3-layer approach consisting of a supervisor, reactor controller, and sensors/actuators. Redundancy was implemented differently in each layer to achieve overall system fault tolerance:

- Reactor Supervisor: Three Supervisor Channels, configured in a Hot/Warm/Cold or a Hot/Warm/Warm configuration, were used to provide overall system control. The Supervisor Channels manage diagnostic data, coordinate software upgrades, and communicate coincidence to the Slider Controllers.
- Reactor Controller: Four Reactor Controller Channels in a channel coincidence configuration provide sensing and control of the reactor, provide telemetry data to the Supervisor Channels, and directly communicate to the Power Conditioning and Distribution (PCAD) subsystem.
- Sensors/Actuators: The design assumed twelve independent slider control channels, each corresponding to one of the twelve (assumed) sliders used for reactivity control. The Slider Controllers apply coincidence to Reactor Controller Channel signals, control slider motion, determine slider position, and communicate errors and status information to the Reactor Controller Channels.

**Figure 6-1: Prometheus Spaceship Control Architecture**




The pre-conceptual I&C architecture could be implemented with approximately 45 circuit cards and a total mass of approximately 320 kg including sensors, cables, and electronics. Reference (9) contains specific details on the following:

- Configuration and functional description of the Supervisor Channels, Reactor Controller Channels, and Actuators.
- A breakdown of card types and sensor types with an estimate of quantity for each.
- A breakdown of mass for cards, sensors, and cables.

The salient features of the pre-conceptual I&C architecture include the following:

- Redundancy with coincidence to provide adequate reliability and fault tolerance.
- Channel independence to mitigate common mode failure.
- Diversity to mitigate common mode failure of developmental sensor technologies.
- A microprocessor-based system whose software is fully modifiable post launch to provide flexibility to respond to anomalies, spaceship degradation, or lessons learned through continued prototype operation.
- The ability to modify the control scheme from sensor inputs, control band setpoints, or complete control scheme algorithm change-out post launch.

- Modularity to scale up or down with respect to the number of required sensors, channels, and actuators (sliders/drums, safety rod(s), and isolation valves).
- Computational capacity to support system response time between tens of milliseconds to minutes, hours, or days.
- Provide a flexible framework to support autonomous reactor operation and fault management with a simple system

In preparation for the next phase of the I&C architecture development, the NRPCT engaged ORNL to survey a broad set of advanced system architectures found in safety critical applications such as commercial nuclear and non-nuclear power, aviation, and others. The ORNL study is presented in Reference (117).

Evaluate Candidate Sensor Technologies

The NRPCT worked with ORNL to identify the best sensor technologies for the SNPP and the JIMO mission. This work initially sought to identify feasible sensors for measurement of reactor coolant temperatures and flow, a single range detector for neutron flux power measurement, and the positions of the reactivity control devices. ORNL conducted a review of the sensor technologies in each of these areas and identified the leading candidates for the SNPP application for the JIMO mission [References (120) and (123)].

This initial work was followed by a series of detailed studies by ORNL and the NRPCT to further evaluate the leading sensor technologies for the SNPP reactor plant. While the initial study had considered four notional reactor concepts in its development of a slate of feasible sensor technologies (a reactor concept had not yet been selected at the time of the initial work), this second set of studies focused on the identification of sensors particularly suited to the direct gas Brayton reactor plant, including the evaluation of reactor coolant pressure sensing technologies. ORNL evaluated resistance temperature detector and thermocouple technologies for the reactor coolant temperature measurement, fission chambers for neutron flux measurement, ultrasonics for coolant flow measurement, as well as cabling for all these systems as described in References (116), (118), (119), and (121) through (123). The NRPCT studied resistance temperature detector and thermocouple methods for reactor coolant temperature measurement; ultrasonics for coolant temperature and pressure, position; electromagnetic and capacitive methods for control element position; and optical methods including pyrometry and Fiber Bragg Grating technologies for coolant temperature measurement. Their results are reported in Reference (10).

The NRPCT also initiated efforts to develop the signal processing methods and electronics for the leading candidate sensor technologies. These efforts included Application Specific Integrated Circuits (ASIC) electronic implementations to meet the radiation hardened requirements of the JIMO mission. These limited efforts are documented in References (126) through (128).

The NRPCT in Reference (10) evaluated all of the technologies reviewed during the Prometheus program by ORNL and the NRPCT and identified the leading sensor technology in each category for the SNPP. These selections are shown in Table 6-1.

**Table 6-1: Leading SNPP Reactor Parameter Sensor Technologies**

| Parameter *Technology* | Key Advantages | Key Challenges |
|---|---|---|
| Temperature<br><br>*Optical Fiber Bragg Grating* | -Potential for many sensors on a single fiber<br>-Demonstrated sensor robustness, accuracy, and low drift at high temperature<br>-Non-amplitude based signal measurement, tolerant to fiber optic transmission line degradation<br>-Small sensor mass and size | -Radiation tolerance of specialized components, including electro-optical components (diodes, amplifiers, lasers)<br>-Signal processing complexity<br>-Technology largely limited to laboratory applications |
| Flux<br><br>*Fission Chamber* | -- Mature technology, widely used in nuclear plant applications<br>- High energy neutron output pulses eases discrimination from gamma induced pulses and allows large separation (up to ~100m) between sensor and electronics. Local preamplification unnecessary.<br>- Capable of full range operation.<br>- Moderate detector bias voltage compared to other ion chamber technologies.<br>- High temperature performance (~800K) | - Handling, shipment, and storage issues associated sensor with highly enriched Uranium coating<br>- Providing startup flux sensitivity and a neutron-moderating environment within a reasonable volume<br>- Complex signal processing may be required to provide pulse counting during startup and current measurement during power operation<br>- Non-traditional materials required for fore-of-shield service at ~1000K. Performance must be demonstrated.<br>- Large size and mass required to achieve necessary sensitivity. |
| Position<br><br>*Ultrasonics* | - Magnetostrictive ultrasonic position technology commercially available.<br>- Magnetic and insulation materials exist for the magnetostrictive components (magnets, waveguides, coils) that are robust in sensor radiation and thermal environment.<br>- Flexible sensor range of measurement<br>- Relatively moderate sensor mass. | - Performance and lifetime in thermal and radiation environment must be verified.<br>- Significant susceptibility to electromagnetic interference from control element motor.<br>- Large sensor space envelope. Mass of sensor unknown. |
| Flow, Compressor Outlet, 550K<br><br>*Ultrasonics* | - Non-invasive flow measurement<br>- Widely available piezoelectric materials can be used in transducer.<br>- UT flow systems in wide usage in natural gas industry at this service temperature.<br>- Current materials and design methods may be used to design acoustic couplers and standoffs. | - Radiation and aging effects of piezoelectric transducers for long duration service must be verified. |
| Pressure<br><br>*Ultrasonics Thin-Walled Bellows* | - Non-invasive flow measurement<br>- Accurate, high resolution measurements demonstrated for low, moderate, and high system pressures.<br>- Ultrasonic transducers and mounting accessories exist for moderate temperature service less than 550K. UT sensors in wide usage in natural gas industry at this service temperature. | - Ultrasonic transducers for high temperature service above 850K in significant radiation fields do not exist. High risk research and development required.<br>- Radiation and aging effects of piezoelectric transducers for long duration service must be verified. |

<u>Software Process Planning</u>

An early task in the Reactor I&C development was to lay the foundation for a meticulous and challenging software development effort that would integrate well with parallel development efforts at JPL and NGST. NRPCT began this effort by selecting the software life cycle, design methodology, and programming language to be used in the project. These selections were documented in Reference (8).

The software life cycle provides the framework and sequence for the software requirements, design, implementation, and testing activities performed as part of the overall software development effort. NRPCT chose the Incremental software life cycle for the development of the Reactor Module flight and ground software. The Incremental life cycle is defined in a series of tasks, starting with initial requirements, architecture development, and increment planning. The tasks applied within each increment include: detailed requirements development, detailed architecture development, module design, module implementation, unit testing, integration, system test, release, and independent verification and validation. Processes were identified for each of the development tasks as well as for related tasks such as configuration management and defect tracking. The Incremental life cycle would have provided for a series of software releases (builds) that would add increasing functionality as the project matured until full software functionality was achieved in a final release. These releases, when coordinated with other spaceship software developers, would have afforded opportunities for integration with other software components earlier in the project schedule, thus allowing the performance of spaceship interface testing between the Reactor, Spaceship, and Mission Modules and the mitigation of risk in a very aggressive design/construction schedule.

The software methodology (object-oriented or structured) provides the approach to requirements development and software implementation. The structured design methodology was chosen for requirements and software implementation because of its more frequent use in space embedded applications and suitability for a real-time embedded design. The selection of a structured design methodology allows for a robust software architecture design, making use of top-down design, functional decomposition (hierarchical refinement of functionality from a coarse level of detail to a fine level of detail), and structured programming. This allows for strong modularity in the design while avoiding some of the inspection burden associated with object-oriented design methodologies.

The NRPCT evaluated several programming languages and chose C for use in the Reactor Module I&C software implementation. C is widely understood and recognized, has been used in many space applications, and has a broad experience base within the NRPCT. The selection of the C programming language complemented the choice of the structured design methodology and would have minimized the software inspection burden that might be associated with other languages.

The NRPCT also developed a draft Prometheus Reactor Module Software Development Plan (SDP) to provide mission specific definitions or project roles, deliverables, a documentation hierarchy, organizational division of responsibilities, and a description of software tools. It was envisioned that the flight software, ground software, and test beds would have each had individual development plans that would trace up to the Reactor Module SDP.

Evaluate Radiation Hardened Electronics

Radiation damage from both the reactor and the Jovian environment poses significant challenges for electronics. All sensor signal conditioning and control electronics for the Reactor Module would have resided aft on the spaceship in a shielded and thermally managed electronics vault. Although this vault would have provided significant radiation protection, it would not have eliminated all possibility of damage. One of the major challenges for the Reactor I&C designers would have been to ensure the reliable operation of this electronics suite over the duration of the JIMO mission and in the high-radiation Jovian environment.

All solid state devices (digital, analog, and power devices) are subject to various forms of radiation damage. Radiation damage is generally considered to affect electronic devices in three different ways: ionization due to energy deposition (total ionizing dose (TID)), damage to the crystalline structure of the device (displacement damage dose (DDD)), and a change in the state of a device caused by a single collision with a high energy particle (single event upset (SEU)). These damage mechanisms have different effects on electronic circuits and require differing mitigation strategies. The radiation qualification goals for the JIMO mission established by JPL for electronic components within the shielded vault were $1 \times 10^6$ rad(Si) total ionizing dose and $11.4 \times 10^{11}$ n/cm$^2$ 1-MeV equivalent displacement dose. These values were intended to provide design margin that could allow reduction in the enclosure shielding requirements and accommodate the external Jovian radiation environment and its uncertainties. By comparison, unhardened commercial electronic devices can typically survive only $5 \times 10^3$ to $10 \times 10^3$ rad(Si) TID without degradation.

While a robust radiation-hardened electronics industry exists to support current space and nuclear applications, the JIMO mission's projected radiation requirements were more severe than the advertised allowable radiation doses for most existing radiation-tolerant and radiation-hardened semiconductor devices. Therefore, the development of custom radiation-hardened electronic devices would have been required, and it would have been challenging, time consuming, and expensive.

Evaluations were underway of various electronic component and circuit board vendors and laboratories with experience in radiation tolerant electronic designs. These included BAE Systems, Honeywell, Jet Propulsion Laboratory, and two Northrop Grumman divisions. Efforts were initiated for the sensor interface circuits to engage two current NR Program vendors which also have corporate partners active in the space program and knowledgeable in radiation hardening practices.

Evaluate Controls

*SNPP Constant Power Control*

Preliminary control studies by the NRPCT had just commenced as the project came to a close. These studies evaluated simple temperature control strategies for operation of the SNPP at constant power levels and during power level transitions. These efforts focused on setpoint and narrow band control using reactor coolant outlet or average temperatures for reactor power level control and Parasitic Load Regulator (PLR) speed and voltage setpoint control for the Brayton units. These controls were applied in model simulations of the SNPP allowing the behavior of the plant and its components to be investigated in constant power, power transitions, and component casualty situations. In general, the SNPP simulations have shown a plant concept that responds in predictable, controllable, and stable manners under these various scenarios.

The models of the plant were under development at the time of project termination and their completion would have provided a better understanding of the plant behavior and system interactions. These would have in turn led to the development of initial algorithms to define the control actions needed to change or maintain the state of the plant. Model development and preliminary transient analyses are discussed in Reference (4).

The SNPP control studies have demonstrated that measurement errors associated with a control parameter, the size of the discrete control actions, and the width of the control band all have an impact on the overall sizing of major plant components. The uncertainty in the measured control parameter and the minimum plant response to a discrete control input determines the minimum width of the control band. Other considerations such as minimization of actuator operation can add to this width.

The incentive to minimize the size of the control band is to maximize plant operating efficiency and minimize the potential impact on component ratings, required core power, and overall plant mass. The upper limit of the control band for temperature is fixed at the maximum allowable (actual) plant temperature because of material limitations. The benefits of a small control band are potentially limited by other considerations such as actuator wear or controller complexity. These drivers would have to be balanced to determine an appropriately sized control band.

*SNPP Startup Control*

An important part of the effort to understand the operation of a direct gas Brayton reactor plant was the development of a notional startup sequence for the spaceship's reactor and energy converters. The NRPCT worked with JPL, NGST, Hamilton Sundstrand, and GRC to assemble a procedure that would bring the reactor from its cold shutdown condition to hot, critical operation with the Brayton units at full power supplying the spaceship's electrical buses so that the powered, interplanetary transit to Jupiter could commence.

The procedure provided in Reference (4) called for the establishment of initial reactor coolant flow by motoring one Brayton unit's alternator with power from the spaceship's solar arrays. The reactor controller would then begin the slow and deliberate process of adding reactivity to the reactor with the movement of its reflectors. Once critical, the reactor would be capable of producing useful heat, and additional reactivity would then be added to raise the reactor power level further to begin heating the gas coolant. When sufficiently high coolant temperature was reached, the Brayton unit operation would become self-sustaining without motoring and begin producing enough electrical power to transfer Brayton speed control to the PLR. With the reactor controller adjusting coolant temperature through reactivity increases and the PCAD controller limiting Brayton speed with load adjustments, a coordinated pattern of temperature and speed adjustments would then be made until the reactor coolant reached full operating temperature and the Brayton unit reached full operating speed and voltage. The second Brayton unit could then be started using on board electrical power from the first Brayton and carefully brought up to full power output. At that point, the reactor plant and energy converters would be providing rated output power and ready to support spaceship operation for the duration of the mission.

The development of this notional startup sequence was important for several reasons. It supported the development the overall mission phase profile by NGST and demonstrated that the startup could easily be accomplished within the allocated time period. It confirmed earlier assumptions on the electrical power demands for the startup and helped solidify the sizing of the solar array panels for the spaceship. More importantly, it helped define the communication paths within the spaceship which would impact the reactor controller, and it underscored the need for close coordination between the reactor controller and the PCAD controller to maintain the power plant within its operating envelope. These conclusions are reflected in the communication paths of Figure 6-1. The development of this procedure also demonstrated the inability to manually control a space nuclear reactor in real time from Ground Control due to the inherent time delays with space communication and the unpredictable interruptions in the communication path between the spaceship and Ground Control. (The procedure was based on the conclusion that all of the procedural steps would be pre-programmed in the Reactor I&C Segment and that Ground Control would only initiate an entire sequence of steps. Their execution would be accomplished autonomously, and the reactor controller would return an indication of their positive completion.) The procedure also opened discussions with JPL on communication security and the issues concerned with directing a nuclear reactor through the normal channels of the deep space communication network.

*SNPP Control Concepts*

In preparation for the next phase of SNPP control development, the NRPCT engaged ORNL to review the general control principals, methods, and challenges that should be considered in the development of the SNPP autonomous control capabilities. The ORNL study is presented in Reference (124).

## 6.3 Key Findings and Perspectives

Based on work performed during the preconceptual design phase, the NRPCT developed a number of technical findings:

- An I&C Segment architecture was developed which included recommended redundancy, interfaces with the spaceship and the number of sensors and board count. The I&C architecture provided for autonomous reactor operation and fault management with a simple system. Electronics for the control system that could withstand the Jovian radiation environment for the intended mission duration appear to be available. Computing electronics in particular are judged to be low risk based on available technologies. Analog electronics for use particularly in the sensor conditioning circuits would be a significant technical challenge and would require development of custom application-specific integrated circuits to achieve the level of radiation hardness and reliability needed. Based on the JIMO schedule, near term actions would have been necessary to initiate development of these electronics.

- An evaluation of candidate sensors has been completed. Technologies appear to be available for all key plant parameters including temperature, neutron flux, pressure, control drive mechanism position, and coolant flow. Significant development would remain to deliver and qualify sensors and interface electronics to support long duration space missions. Major technical challenges would be lifetime, operating temperature, accumulated radiation dose, and the need to operate without recalibration.

- The most challenging sensor application would be hot leg temperature due to the lifetime degradation of the sensors at high temperature and the need to integrate the sensor with internally insulated hot leg piping. The most promising sensor technology for this application was judged to be sapphire Fiber Bragg Gratings, which have the potential for adequate stability over life.

- A plan for control system software development for JIMO was completed, and selections were made for process management, interfaces with other systems, and software design. For the reactor controller, emphasis was placed on high reliability software design with simple operating structures. Algorithms for system control and autonomy are not yet developed.

- The reactor plant start-up would be the most complex operation envisioned for the Reactor I&C Segment and would require close integration with other spaceship subsystems. The reference approach involved using a solar array to power the Brayton alternator to initiate gas coolant flow. Once flow is initiated, the reactor would be brought critical by slow increases in reactivity using the reactor control devices. Once critical, the reactor would be slowly heated with reactor power until full power and temperature conditions were reached. This would require tight coordination between CDM positioning and Brayton speed control through the use of a start inverter and PLR. A notional start-up procedure was developed, although it was not fully modeled or refined.

- The start-up of a second Brayton in a multiple Brayton system would create additional control system complexity. Starting both Brayton units simultaneously might require a prohibitive amount of electrical power. Starting the second Brayton after the reactor was critical could lead to a sudden increase in reactor flow and heat removal, which in turn would lead to a sudden increase in reactor power. This would require further modeling and testing to refine the startup sequence for a second Brayton unit.

- Control studies indicate that independent control loops for both reactor temperature and turbine speed can be successfully implemented. No stability issues with plant control have been found. Direct communication between the PCAD subsystem and Reactor I&C Segment controllers was planned, as opposed to communicating data through the spaceship flight computer.

- A preliminary evaluation of spaceship radiation and thermal environments has been completed. Materials for both sensors and cabling appear to be available that will withstand the environment given careful design.

# 7 INTEGRATED SYSTEM TESTING SUMMARY

## 7.1 Issues and Challenges

As discussed in Section 1.4, obtaining test data in time to support design decisions was a key challenge for Project Prometheus. While time and budget pressures are not unusual in large test programs, this space reactor project posed unique challenges, and a robust test program was considered necessary to achieve adequate assurance of mission success. The objective of integrated system testing was to provide component and system data that could be incorporated into the ongoing design process. Individual components can be tested to some extent on a stand-alone basis, but integrated system testing inevitably identifies issues that had not been previously found. System performance testing was important because the concept of directly coupling a gas reactor to a multiple Brayton system has never been demonstrated.

In addition to system performance testing, reliability testing was also critical to the project. Some lifetime data for a 15- to 20-year mission could be accelerated, but only limited data would have been available prior to launch. Reliability is a key factor in making important decisions, such as trading off redundancy for mass.

A significant part of the integrated system testing plan included a notional Ground Test Reactor Facility (GTRF). GTR operation would provide invaluable support for SNPP design and operation, particularly for a manned mission. Siting and constructing a GTRF requires significant resources early in the project in order to deliver data in time to impact design and operational decisions. Key issues to be addressed in GTR recommendation included the benefits, where to site the facility, whether containment was required or confinement would be acceptable due to low radiological risk, and the impact on project schedule and budget.

## 7.2 Summary of Work

The test program to support development of the Direct Gas Brayton system had just been initiated at a low level of testing at the time of project termination. Significant progress was made in establishing a common integrated test plan as well as a final ATLO strategy. These plans built upon the extensive testing and deployment experiences of NRPCT, NASA Centers, DOE National Laboratories, and the spaceship subcontractor (NGST). Input from individuals directly responsible for the SP-100 and SNAP space nuclear power programs was also used in developing these plans.

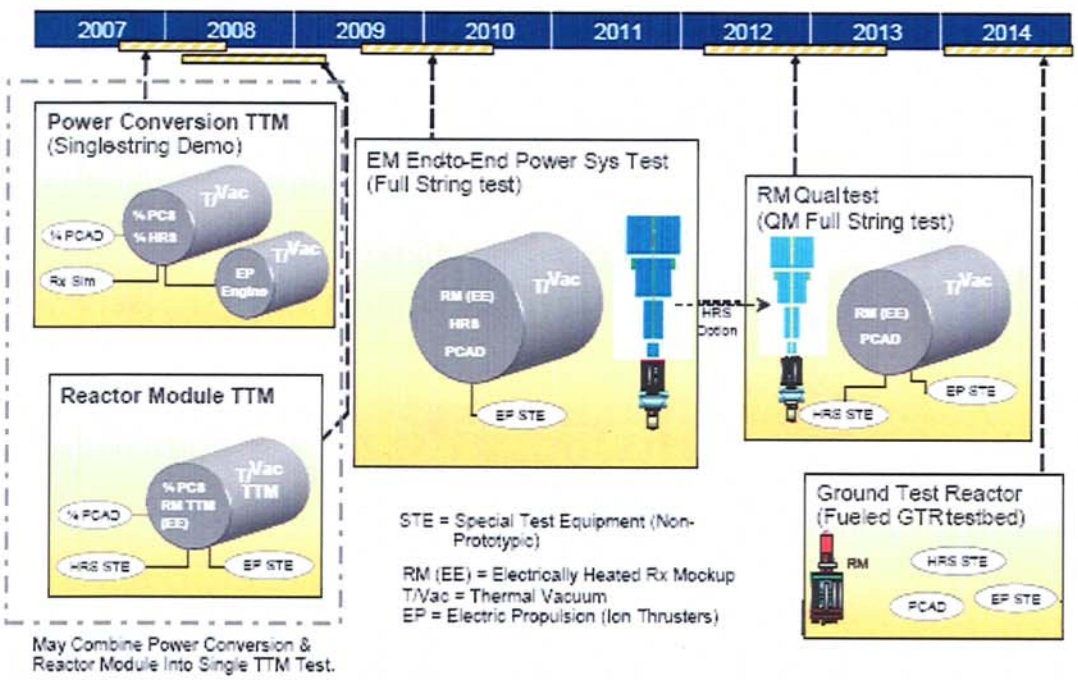
A key strategy of the Prometheus test program was to use the existing national infrastructure to the maximum extent practicable. This approach would minimize facility startup costs and lead time for obtaining data, as well as take advantage of the technical expertise that existed at the various NASA Centers, DOE National Laboratories, and vendor facilities. Test facilities internal to the NR Program would be limited to those needed for fundamental reactor research and development, as well as those necessary to provide designers with basic operational experience on Brayton system and instrumentation and control hardware. However, Reactor Instrumentation and Control would be extensively tested at NRPCT sites.

The integrated test strategy had a series of system level tests that increased in prototypicality over time, as shown in Figure 7-1. These tests were called the Thermal Test Model (TTM), Engineering Model (EM) and Qualification Model (QM). The test program culminated with the final flight unit assembly, test, and launch operations (ATLO). Such an approach was methodical and logical, but because long-term program budgets had not yet been established, schedule and plan compromises may have had to be considered as the project progressed.

In addition to TTM-EM-QM system testing, a GTR Facility (GTRF) was being investigated. The initial benefits of the GTRF are to confirm key reactor design parameters and plant system dynamics. Once the flight unit is en route, the GTRF would be used to shadow the flight unit for early indications of problems, and to perform diagnostic troubleshooting if problems are encountered. Some of the key benefits of operating a prototype would be:

- Characterize beginning of life reactivity over the operating temperature range
- Confirm worth of reactivity control devices and safety rod(s)
- Verify reactor feedback coefficients and power coefficient
- Characterize reactor power distribution
- Demonstrate the SNPP startup sequence
- Verify plant dynamic performance predictions and demonstrate plant procedures
- Testing of the flight reactor instrumentation and control system
- Ground communications/telemetry testing
- Training of the flight unit ground control system operators
- Verify throughout-life reactor operation and reactivity

**Figure 7-1: Integrated System Test Strategy**




## 7.3 Key Findings and Perspectives

Significant test programs had not yet commenced for the project prior to termination. However, some general testing perspective was gained over the course of the project:

- "Test as you fly, fly as you test" philosophy. NRPCT used the NASA/JPL philosophy of "test as you fly, fly as you test" to guide the development of the integrated test plan. This philosophy is based upon lessons learned from previous missions and the need to do everything practical to exercise the actual flight system on the ground since repair after launch would not be possible. Ideally, the complete system would be fully tested on the ground in a prototypical manner to provide greater certainty of success and help eliminate unknown issues that may not have surfaced in more isolated, separate effects test programs. While this approach makes sense, it was recognized and accepted early on that the presence of a nuclear reactor would present significant limitations in the ability to fully meet this principle. All plans formulated assumed that the fueled reactor would be integrated with the spaceship after integrated system testing was completed.

- Test early and often. The need to commence testing early in the program and to continually iterate between testing and design was recognized because of the lack of significant experience with the reactor plant concepts being considered for space applications. While this issue is related to the "test as you fly..." perspective, it is different in that it drove the desire for early fundamental testing to gain insight into the design even when the hardware being tested was not prototypical.
- High temperature vacuum testing of refractory metal alloys. If refractory metal alloys were required in the pressure boundary, the integrated system testing would have been significantly more complicated and costly to 1) provide sufficiently high vacuum conditions and, 2) provide protection for a loss of vacuum condition. This would be true for the non-nuclear test facilities, as well as the ground test reactor facility. Furthermore, the full size integrated EM and QM testing would likely have to be reduced in scope or prototypicality since the planned facilities (NASA Plum Brook or Johnson Space Center) would likely not have been capable of reliably achieving these high levels of vacuum.
- Ongoing fundamental research and development. Future space reactor projects would benefit from maintaining some level of fundamental research and development for space nuclear reactor development, so that key data that takes years to produce (such as irradiated material properties and lifetime reliability data) could be available at the start of the design effort rather than occurring in parallel.

# 8 MISSION EXTENSIBILITY

## 8.1 Issues and Challenges

As discussed in Section 1.3, extensibility to other missions was a Level 1 requirement for Project Prometheus. The specific Level 1 Development Technology requirement that discussed extensibility to surface missions was as follows:

> ***"The following Space Nuclear Reactor technologies shall be developed for Lunar and Mars surface power reactors: 1) Nuclear fuel, 2) Reactor core materials and coolants, and 3) Instrumentation and Control."*** (This item was indicated as an objective – minimum requirement not yet defined.)

At the time of project termination, this Level 1 requirement was further incorporated into a Level 2 requirement:

| Key Level 2 Requirement | Impact on Reactor Module | Implementation |
|---|---|---|
| *The Space Nuclear Reactor design shall utilize technologies that facilitate extensibility to surface operations.* | Consideration in the selection of design and materials compatible with Lunar and Mars missions. | Must consider compatibility of pressure boundaries and external surfaces with surface environments. |

Extensibility, in the context of the Prometheus Project was considered in two broad categories:

1. Other deep space, nuclear electric propulsion missions
2. Surface missions, specifically the Moon and Mars

Each of these categories of missions presented difference issues and challenges as discussed below.

<u>Extensibility to Deep Space Missions</u>

Extensibility of the JIMO design reactor to other deep space missions was considered to be relatively straightforward, since the deep space missions discussed by NASA were to be similar to the JIMO mission in terms of using nuclear electric propulsion, and the power levels were envisioned to be the same as the power level of the JIMO reactor (~200 kWe). Similarly, the radiation levels in the Jovian system were likely more limiting than the radiation levels expected for other deep space missions. The key challenge with some of the envisioned deep space missions was long mission duration. Missions of up to 20 years were being considered and this would have posed the most significant challenge to extensibility in terms of system reliability and long term materials behavior.

<u>Extensibility to Lunar and Mars Surface Missions</u>

Extensibility to surface missions to the Moon or Mars presented a different set of challenges since these missions were intended to be manned. While the requirements for such missions were not defined, it was clear that the system design and operation would likely be significantly different for manned surface missions to account for personnel and environmental protection. However, the assertion that the technology developed for JIMO would be extensible to these missions even if significant design changes were required was still considered valid. It was also unclear whether the reactor core design would require significant changes for manned surface missions.

## 8.2 Summary of Work

As part of the Reference (2) concept selection process, the NRPCT concluded that the selection of a direct gas cooled reactor coupled to a Brayton energy conversion system had the best prospect to support the envisioned JIMO mission and was extensible to other deep space missions and surface missions on the Moon or Mars. Although some features of the nuclear power plant would likely be the same, engineering and modifications would have been required for a manned surface mission vs. an unmanned deep space mission. Despite these anticipated design changes, the underlying technologies (e.g., material development, fuel, plant components) would have been extensible to a surface mission. The direct gas Brayton concept remains mass competitive with other reactor concepts over a range of powers between 25 and 300 kWe. Based on specific requirements for a surface mission (power, lifetime, schedule, etc.), a trade study may be necessary to determine if a direct gas Brayton system would be most appropriate for that application.

Although extensibility was only one of many factors in the selection of the gas Brayton system, it did play a role in avoiding the use of refractory metal alloys in the pressure boundary. While refractory metal alloys may survive the high reactor temperatures in the deep vacuum of space, they would not easily endure the lunar or Martian atmospheres without additional protection (vacuum chamber or coatings). This is because refractory metal alloys at prototypical operating temperatures are extremely susceptible to embrittlement when exposed to elements commonly found in a lunar or Martian surface environment, specifically oxygen and carbon. Such protection would significantly complicate the design and operation, and would make the deep space reactor much less extensible to surface missions. Nickel-base superalloys were primarily considered for the reactor and energy conversion pressure boundary, but no final material selections were made at the time of project termination.

Specific mission and environmental requirements for other deep space missions were not defined and no effort was expended by the Naval Reactors program in evaluating specific issues with these missions.

## 8.3 Key Findings and Perspectives

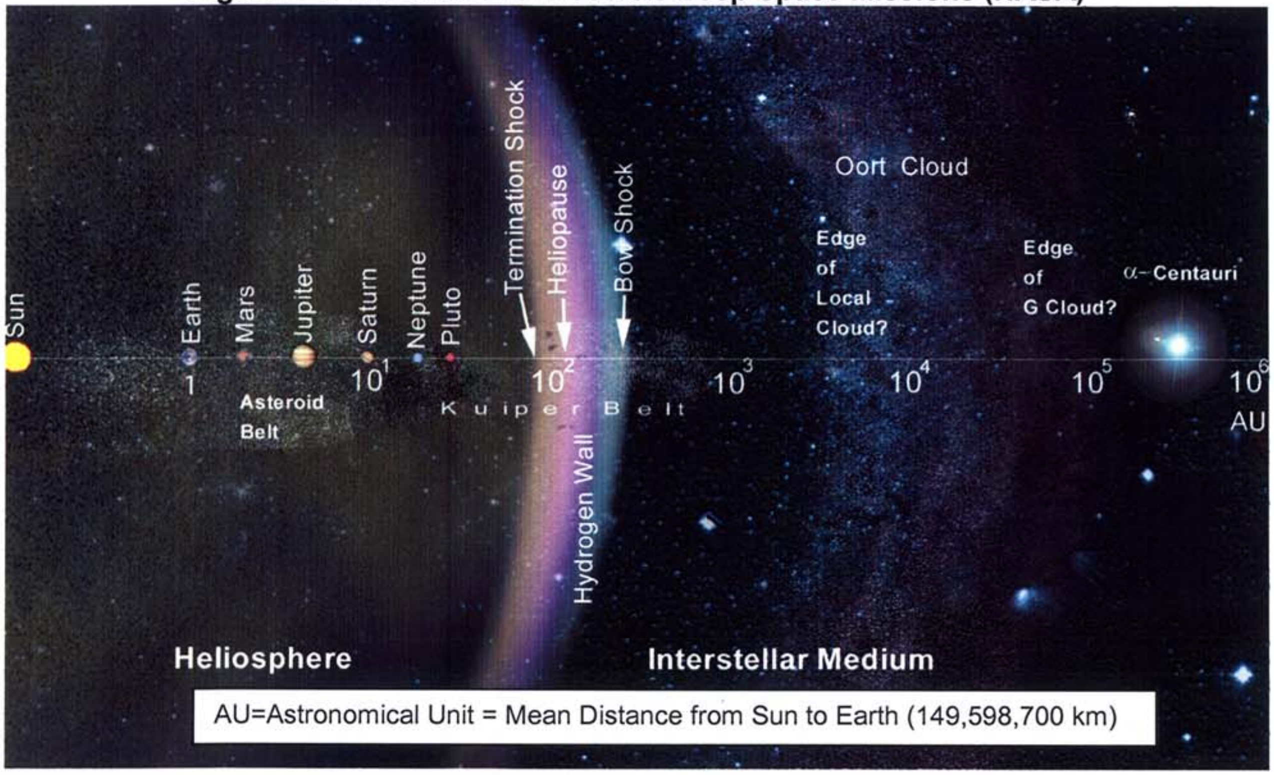
### *8.3.1 Extensibility to Other Deep Space Missions*

Some of the Level 2 requirements were established specifically for the JIMO mission and some were multi-mission, meaning that they would meet the expected requirements for all the deep space missions for the Deep Space Vehicle. For example, a target mission lifetime of 20 years was based upon supporting projected follow-on missions, even though the JIMO mission was only expected to last 10 to 12 years. Future missions that were studied by NASA include:

- Saturn and its moons
- Neptune and its moons
- Comet and multi-asteroid sample return
- Kuiper Belt rendezvous

While these possible mission concepts were only preliminary in scope, they were all estimated to be achievable within a 20 year lifetime and the power requirements established for JIMO (i.e., ~200 kWe). Other possible missions such as an "interstellar precursor" to the Heliopause (200 AU from the sun) were beyond the 20 year mission life of Prometheus requirements and not truly extensible from the JIMO design. Figure 8-1 illustrates the location of these missions relative to Earth.

**Figure 8-1: Illustration of Possible Deep Space Missions (NASA)**




The 10 years of full power operation required for JIMO were expected to encompass longer missions, as well. This was because despite the longer mission lives, there would also be longer reduced power periods between thrusting such that the number of full power years was no greater than for JIMO. Because of this, the JIMO reactor design would be extensible to other deep space missions. One key issue would be system reliability as mission life approached 20 years. A system for 20 years of operation may become more feasible as the JIMO design matured and test and flight system data were obtained and factored into subsequent designs.

### *8.3.2 Extensibility to Moon and Mars Surface Missions*

Extensibility to surface missions posed significantly greater challenges for the nuclear power plant design, particularly since missions to the Moon and Mars were expected to be manned. Specific surface missions under consideration by NASA at the time of project termination included establishment of a base camp on the Moon and an eventual manned mission to Mars. Because of the major differences between these types of surface missions and the planned JIMO mission, extensibility to surface missions was a more significant factor in the design than extensibility issues with other deep space missions. Significant differences in the design requirements would exist for a manned mission compared to an unmanned mission, such as personnel safety, reactor protection, backup power supplies, shielding, redundancy, deployment, maintenance, etc.

Significant differences between surface mission requirements and the JIMO mission may be specified such as lower power, shorter lifetime, and more power transients. At different power levels, some alternatives to the Direct Gas Brayton concept become more competitive. At the low end of this power range, the reactor and its shield become the mass limiting items, allowing a lower temperature liquid metal Stirling concept to become more mass competitive. Notional lower power and lifetime requirements for surface missions also open up the possibility of a mass competitive neutronically moderated gas reactor. A moderated reactor would require about 70-80% less highly-enriched uranium fuel than a small fast reactor, which may permit using some adaptation of a prior high temperature gas reactor fuel system for a space mission. One approach that could be considered is a water moderated gas reactor. This reactor has considerable mechanical and thermal design challenges and would require detailed concept development based on firm mission requirements to better judge feasibility. These systems would not be as compatible with deep space missions, as discussed in Reference (4).

# 9 KEY FINDINGS AND A PERSPECTIVE ON NEAR-TERM DEVELOPMENT

The work completed by the NRPCT, as summarized in this Technical Summary, lead to some overall conclusions that deserve reiteration:

1) After more in-depth concept development, the gas reactor Brayton system continues to offer the prospect to meet Prometheus mission requirements. The NRPCT judges that a direct gas Brayton is more deliverable in the near term than liquid metal cooled reactor system at comparable power levels. The thorough review of Brayton system component performance capability supports the feasibility and strategy of using Brayton systems for long duration deep space missions.

2) Material performance and current uncertainty in characterizing material performance affect selection of reactor design features for a high temperature long life plant. Obtaining improved data on fuel element cladding materials to allow for selection of a reference cladding was a top NRPCT priority. This lack of well quantified fuel system performance is a major design uncertainty.

3) Fast neutron spectrum reactor designs appear suitable for the Prometheus deep space missions. However, fast reactors bring additional challenges of increased nuclear design uncertainty and increased safety measures to ensure safety during a potential transport or launch accident, including compaction considerations. Extensive reactor physics testing, safety analysis, and safety performance testing are necessary to ensure public safety throughout reactor transport, installation, and launch.

4) Preliminary evaluation of reactor plant dynamic performance indicates that the reactor system can be started and controlled with no apparent instabilities associated with the direct coupling of the reactor core and the Brayton energy conversion system.

5) Operation of the SNPP in a low power mode during periods without thrust should be evaluated to potentially reduce overall stresses on the plant and maximize lifetime.

6) Preliminary system evaluations point to a system with one or two Brayton loops being more reliable and less massive than the four loops considered in past studies.

7) Testing of all aspects of the design concept will be needed to increase design confidence and demonstrate performance. The initially envisioned test sequence includes fundamental material testing, fuel system testing, component testing, instrumentation and control testing, physics experiments, and integrated non-nuclear system testing leading to a ground test reactor prototype.

8) The Direct Gas Brayton system technologies are extensible to a range of deep space missions, as well as manned surface missions. Design requirements for manned missions will be substantially different than those for an unmanned mission; therefore, the specific design will be different. Because a surface mission may have significantly different mission requirements such as power level and duration, alternatives to the Direct Gas Brayton system should also be evaluated.

Perspective for Future Development

The scope and nature of the development challenges, together with the flexibility and extensibility of the direct gas Brayton system makes it well suited, in concept, for deep space and surface missions. NRPCT concludes that the technologies for a gas-Brayton system are within reach for a first space reactor mission for both types of missions. Given that there may be time but limited resources for space nuclear work prior to initiation of another large scale space reactor project, work should be concentrated toward making progress on reactor fuel and material systems development and testing. Specifically, irradiation testing of fuel system materials, including fuel materials, refractory metal alloys, and ceramics as well as nickel-base superalloys should be performed to obtain this long-lead data for the next space reactor project.

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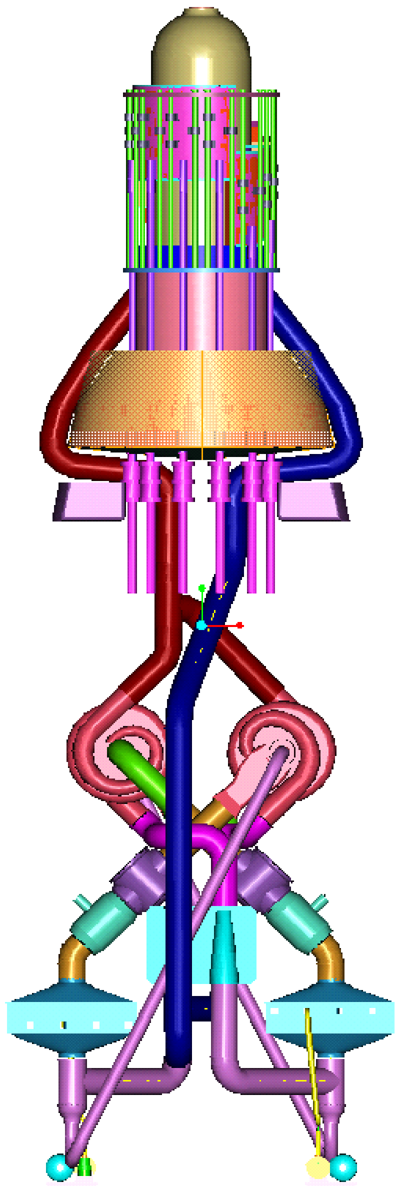
# Project Prometheus Reactor Module

## Bibliography



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# 1 INTRODUCTION

In addition to documents generated during Project Prometheus, NRPCT gathered many relevant documents from past programs and from open literature. These references have been combined into an extensive Bibliography to provide important information for a future space fission reactor program. The Bibliography contains three types of information:

- Information generated by the NRPCT during Project Prometheus
- Information generated by NRPCT subcontractors and partners during Project Prometheus
- Information generated by other sources before Project Prometheus (e.g., SP-100, SNAP Program, etc.)

The Bibliography is divided into major categories that generally follow the Reactor Module Work Breakdown Structure. Each category is subdivided as appropriate by topic.

All documents generated by the NRPCT that are listed in the Bibliography are accessible via the Internet-accessible U.S. DOE Office of Scientific and Technology Information (OSTI) Science Resource Connection (SRC) database, except those marked with a star (✱), plus (+), or cross (×) after the document description. The reports available in the SRC database are unclassified and not sensitive.

The symbols listed above denote the following types of information:
✱ Sensitive U-SNRI
+ CONFIDENTIAL
× Sensitive Official Use Only (OUO)

Sensitive unclassified documents, specifically U-SNRI and Official Use Only documents, will be stored in an OSTI repository that is not internet accessible. OSTI will direct requests for these sensitive documents to the Schenectady Naval Reactors Office (SNR), Security and Safeguards Division. Two documents are classified as CONFIDENTIAL per the DOE/DOD/NASA Classification Guide for Space Reactor Power Systems, CG-SRPS-1. CONFIDENTIAL material will be maintained internal to the NR Program and will not be provided to OSTI. See Section 5.1.2 of Volume 1 for additional access information for sensitive documents.

Bibliographic information for each NRPCT document stored at OSTI (including sensitive documents) will be entered into the OSTI SRC database to enable topical and key word searching. The keyword "NRPCT" will bring up all NRPCT documents. All other documents listed in this bibliography are available in the open literature, from DOE OSTI, and/or from the originating organizations.

# 2 NRPCT Documents

All documents listed below are available in the SRC database, except those marked with ✱, +, ×.

(1) KAPL Letter SPP-67410-0014, "Documentation of the NR Program Assessment of the Design Space for the Prometheus 1 Project," April 17, 2006. ✱

(2) KAPL Letter SPP-67110-0007 / Bettis Letter B-SE-0143, "Documentation of Space Nuclear Power Plant Concept Selection," July 27, 2005. ✱

(3) NR letter I#05-01228, "Space Nuclear Power - Reactor Coolant and Power Conversion System Concept - Approval of", dated April 20, 2005.

(4) KAPL Letter SPP-67210-0010 / Bettis Letter B-SE(SPS)-001, "Space Nuclear Power Plant Pre-Conceptual Design Report; For Information," January 27, 2006.

(5) KAPL Letter SPP-67410-0013 / Bettis Letter B-SE(RE)-0003, "Project Prometheus Space Reactor Pre-Conceptual Design Report, for Naval Reactors Information" January 27, 2006. ✱

(6) Bettis Letter B-SE(RE)-0001, "Fuel Type Recommendation for the Project Prometheus Space Nuclear Reactors, For Naval Reactors Approval," July 28, 2005. ✚

(7) KAPL Letter SPP-67410-0002, "Space Program Annual Report, for Approval," December 15, 2004.

(8) KAPL Letter SPP-67610-0007, "Prometheus Reactor I&C Software Development Methodology, for Action," July 30, 2005.

(9) KAPL Letter SPP-67610-0008, "Space Power Program, Instrumentation and Control System Architecture, Pre-conceptual Design, for Information," October 20, 2005.

(10) Bettis Letter B-SE(SPS)IC-008, "NRPCT Closeout of Prometheus Sensor Development Work for NR Information", December 21, 2005.

(11) KAPL Letter SPP-SRS-0002, "Request for Naval Reactors Comment on Proposed Prometheus Space Flight Nuclear Reactor High Tier Reactor Safety Requirements and for Naval Reactors Approval to Transmit These Requirements to JPL," April 28, 2005.

(12) KAPL Letter SPP-SRS-0007, "Reactor Safety Planning for Prometheus, for Naval Reactors Information," May 6, 2005.

(13) KAPL Letter SPP-SRS-0022, "Prometheus Space Reactor Launch Safety - Discussion of Approach, Technical Effort and Resources, for Naval Reactors 08Z Information," January 25,2006. ✱

(14) KAPL Letter SPP-67210-0009, "Summary of Prometheus Radiation Shielding Nuclear Design Analyses," January 13, 2006.

(15) KAPL Letter SPP-67210-0011, "Space Reactor Radiation Shield Design Summary, for Information" February 17, 2006.

(16) KAPL Letter SPP-SEC-0039, "Documentation of Naval Reactors Papers and Presentations for the Space Technology and International Forum (STAIF) 2006," March 2, 2006.

(17) CG-SNR-1, "DOE-NASA Classification Guide For Civilian Space Nuclear Reactors To Support Nasa Project Prometheus Missions (U)," December, 2004. ✖

(18) SN-801, "Guidelines for the Control and Protection of Unclassified Space Nuclear Reactor Information," August 2, 2005. ✖

(19) Bettis Letter B-SE(SPS)GT-005, "Space Nuclear Power Plant - Ground Test Reactor Facility Planning Closeout Report - For Information," December 13, 2005. ✖

(20) Bettis B-TM-1639, "Specifications, Pre-Experimental Predictions, and Test Plate Characterization Information for the Prometheus Critical Experiments," April 2006.

(21) KAPL Letter MDO-723-0010 / B-MT(SRME)-59, "Summary of Core, Fuel Cladding, and Plant Structural Materials Considered for the Prometheus Space Nuclear Power Plant," April 14, 2006.

(22) KAPL Letter MDO-723-0011 / Bettis Letter B-MT(SRME)-53, "Refractory Metal Irradiation Testing at Oak Ridge National Laboratory," February 23, 2006. ✖

(23) KAPL Letter MDO-723-0015 / Bettis Letter B-MT(EDT)S-028, "Multiple Irradiation Capsule Experiment (MICE)-3B Irradiation Test Of Space Fuel Specimens in the Advanced Test Reactor (ATR) - Close Out Documentation for Naval Reactors (NR) Information," January 9, 2006.

(24) KAPL Letter MDO-723-0017, "Haynes 230 Mini-Can Welding to Support Planned Irradiation Testing of Candidate Space Fuel Materials," January 17, 2006.

(25) KAPL Letter MDO-723-0018, "Initial Assessment of Environmental Barrier Coatings for the Prometheus Project," December 15, 2005.

(26) KAPL Letter MDO-723-0027 / Bettis Letter B-MT(SRME)-34, "Fuel Material Properties and Guidance to Support Pre-Conceptual Design Efforts," July 7, 2005. *

(27) KAPL Letter MDO-723-0036 / Bettis Letter B-MT(SRME)-37, "Space Nuclear Propulsion Program – Overview of Domestic and International Shipping of Irradiated Structural Materials and Handling Associated Waste, for NR Information," January 23, 2006.

(28) KAPL Letter MDO-723-0038, "Addendum to MDO-723-0027, Updated Fuel Material Properties," December 7, 2005.

(29) KAPL Letter MDO-723-0040 / Bettis Letter B-MT(SPME)-15, "Comparison of fission product yields and their impact," February 1, 2006.

(30) KAPL Letter MDO-723-0042, "Reflector and Reflector Material Properties For Project Prometheus," November 2, 2005.

(31) KAPL Letter MDO-723-0043 "Assessing the Effects of Radiation Damage on Ni-base Alloys for the Prometheus Space Reactor System," January 19, 2006.

(32) KAPL Letter MDO-723-0044 / Bettis Letter B-MT(SRME)-52, "JOYO-1 Irradiation Test Campaign Technical Close-out, For Information," January 31, 2006.

(33) KAPL Letter MDO-723-0046 / Bettis Letter B-MT(SPME)-23, "Space Reflector Materials for Prometheus Application," January 31, 2006.

(34) KAPL Letter MDO-723-0048, "The Evaluation of Lithium Hydride for Use in a Space Nuclear Reactor Shield, Including a Historical Perspective," December 9, 2005.

(35) KAPL Letter MDO-723-0049 / Bettis Letter B-MT(SPME)-22, "Space Shield Materials for Prometheus Application," January 20, 2006.

(36) KAPL Letter MDO-723-0052, "Program Summary of Prometheus-1 Fuel Development, For Information," February 15, 2006. +

(37) KAPL Letter MDO-723-0053, "Mass Transport Modeling for the Prometheus Space Nuclear Power Plant (SNPP), For Information," January 20, 2006 *

(38) KAPL Letter MDO-723-0057," Naval Reactors Prime Contractor Team (NRPCT) Experiences and Considerations with Irradiation Test Performance in an International Environment," to be issued (expected April 2006).

(39) Bettis Letter B-MT(AMSI)-17, "Testing Results of Magnetostrictive Ultrasonic Sensor Cables for Signal Loss," May 2005.

(40) Bettis Letter B-MT(AMSI)-43, "Materials for the Control Rod Drive Mechanisms," December 14, 2005.

(41) Bettis Letter B-MT(AMSI)-44, "Alternator Electrical Feedthrough Insulator Materials for Project Prometheus," January 4, 2006.

(42) Bettis Letter B-MT(SPME)-4, "Compatibility of Space Nuclear Power Plant Materials in an Inert He/Xe Working Gas Containing Reactive Impurities," January 31, 2006.

(43) Bettis Letter B-MT(SPME)-17, "On-Line Coolant Chemistry Analysis and Control," February 27, 2006.

(44) Bettis Letter B-MT(SPME)-18, "A Review of Tribological Coatings for Control Drive Mechanisms in Space Reactors," February 21, 2006.

(45) Bettis Letter B-MT(SPME)-20, "Carbon-Carbon Composites as Recuperator Material for Prometheus System," February 27, 2006.

(46) Bettis Letter B-MT(SPME)-21, "Metallic and Non-metallic Materials for the Primary Support Structure," February 21, 2006.

(47) Bettis Letter B-MT(SPME)-24, "Hot Leg Piping Materials Issues," February 27, 2006.

(48) Bettis Letter B-MT(SPME)-25, "Barrier Coatings for Refractory Metals and Superalloys," February 23, 2006
(49) Bettis Letter B-MT(SPME)-26, "Double Retort System for Materials Compatibility Testing," February 23, 2006.
(50) Bettis Letter B-MT(SRME)-45, "Summary of Dissimilar Metal Joining Trials Conducted by Edison Welding Institute," November 18, 2005.
(51) Bettis Letter B-MT(SRME)-46, "Experimental Design for Evaluation of Co-extruded Refractory Metal/Nickel Base Superalloy Joints," December 16, 2005.
(52) Bettis Letter B-MT(SRME)-49, "Closeout of JOYO-1 Specimen Fabrication Efforts," October 31, 2005.
(53) Bettis Letter B-MT(SRME)-50, "Biaxial Creep Specimen Fabrication," February 9, 2006.
(54) Bettis Letter B-MT(SRME)-51, "Processing of Refractory Metal Alloys for JOYO Irradiations," February 21, 2006. ✱
(55) Bettis Letter B-MT(SRME)-54, "Refractory Metal Mechanical Testing at Oak Ridge National Laboratory," February 23, 2006. ✕
(56) Bettis Letter B-MT(SRME)-55, "Fuel System Compatibility Issues for Prometheus-1," January 20, 2006.
(57) Bettis Letter B-MT(SRME)-56, "Modeling of Fission Gas Release in $UO_2$," January 23, 2006.
(58) Bettis Letter B-MT(SRME)-58, "Failure Analysis of Cracked FS-85 Tubing and ASTAR-811C Endcaps," February 9, 2006.

# 3 NRPCT Subcontracted Reports

The following reports were generated by DOE Laboratories, NASA Centers, and NR Program contractors / vendors under NRPCT subcontracts during Project Prometheus.

## 3.1 Reactor Module Reactor Segment

Idaho National Laboratory
(59) Olsen, D.N., et al. "Configurations and Experiments in the ZPPR-16 Power Reactor Space Benchmark Program." INL/ANL. INL/EXT-05-00555, ANL-ZPR-475, August 2005.
(60) Olsen, D.N., et al. "Experiments for the SP-100 Space Reactor in ZPPR-20." INL/ANL. INL/EXT-05-00556, ANL-ZPR-497, August 2005.
(61) Olsen, D.N., et al. "Configurations for SP-100 Space Reactor Experiments in ZPPR-20." INL/ANL. INL/EXT-05-00558, ANL-ZPR-498, August 2005.
(62) McEligot, Donald M. "Convective heat transfer and pressure drop in low-Prandtl-number gas mixtures." INL. INL/EXT-05-00779, September 26, 2005
(63) Clayton, Kevin K. and Bruce G. Schnitzler. "Prometheus Program Fuel Pin Testing Assessment–Phase 1." INEEL and Bechtel BWXT. INEEL/EXT-05-00965, November 2005.

Los Alamos National Laboratory
(64) Poston, David I. "Temperature Dependent Reactivity for 1-Mwt Gas-Cooled Prometheus Reactors." Task 2, Subtask C, Item 2. LANL. LA-CP-05-0793, July 18, 2005.
(65) Poston, David I. "LANL input for NRPCT Reactor Design Basis." Task 2, Subtask C, Item 3. LANL. LA-CP-05-1368, December 12, 2005.

(66) Poston, David I. "Detailed Description of the Nuclear and Thermal Design Processes and Codes: ALLGEN & TMSS." Task 2, Subtask D.1, Item G. LANL. LA-CP-05-1373, December 12, 2005.
(67) Sadasivan, P., Testing Plan for the Heat Pipe Reactor, LANL. LA-CP-04-0786, September 28, 2004.
(68) Reid, R.S., Report Assessing Heat Pipe Performance Capability, LA-CP-05-0044, January 19, 2005.
(69) Dale, C. "Material Property Database." LANL, LA-CP-04-0741, September 21, 2004.
(70) Dale, C. "Design Assumptions and Methodologies Used in the Feasibility Study." LANL. LA-CP-04-0938, December 17, 2004.
(71) Poston, D. "Determination and Assessment of Graded Safeguards Category and Attractiveness Level for Lower-Enrichment JIMO Nuclear Reactor Fuel Material." LANL. LA-CP-04-0713, September 16, 2004.
(72) Poston, D. "Jupiter Icy Moons Orbiter Reactor Design Studies to Support the Naval Reactors Prime Contract Team Feasibility Report." LANL. LA-CP-04-0882, November 23, 2004.
(73) Marcille, T. "JIMO Reactor Module: Hydrodynamic Analysis." LANL. LA-CP-04-0885, November 29, 2004.
(74) Poston, D. "Prometheus 1 Design Studies Report." LANL. LA-CP-05-0137, February 4, 2005.
(75) Kapernick, R. "Creep Analysis for a Liquid Metal Cooled Reactor Vessel." LANL. LA-CP-04-0756, September 23, 2004.
(76) Marcille, T. "JIMO Reactor Module: MCNP(X) Code and Cross-section Qualification Description Document." LANL. LA-CP-04-0684, August 23, 2004.
(77) Marcille, T. "JIMO Reactor Module: Critical Benchmark Experiment Review for Application to JIMO." LANL. LA-CP-04-0706, August 31, 2004.
(78) Marcille, T. "JIMO Reactor Module: ZPPR-16 and ZPPR-20 Qualification Task." LANL. LA-CP-04-0723, September 16, 2004.
(79) Reid, R. "Report Assessing Heat Pipe Performance Capability / Heat Pipe Technology Assessment." LANL, LA-CP-04-0722, September 14, 2004.
(80) Loaiza, D. "Niobium-1 Zirconium Moderated by Polyethylene and Fueled with Highly Enriched Uranium." LANL. D-5-05-067 / LA-UR-04-8751, December 16, 2004.
(81) Marcille, T.F. "SPACE05L Cross Section Library." LANL. D-5-06-066 / LANL-SPP-05-0002; Revision 5; Deliverable d.2, January 30, 2006.
(82) Poston, D. "External Control Options and Design Studies for 1MWt Gas-Cooled Prometheus Reactors." Task 2, Subtask C, Item 4. LANL. D-5-06-066 / LA-CP-06-0152, January 27, 2005.
(83) Marcille, T.F. "Nb-1%Zr Experiment Plate Dimensions." LANL. D-5-06-066 / LANL-SPP-05-0006; close-out deliverable, January 30, 2006.
(84) Loaiza, D., et al., "Hand-stacking for Rhenium Critical Experiments Fueled by Highly Enriched Uranium and Moderated by Polyethylene and Various Graphite Plate Thicknesses." LANL. LA-UR-05-9236_Draft, January 2006.
(85) Loaiza, D., et al., "Hand-stacking for Baseline Critical Experiments Fueled by Highly Enriched Uranium and Moderated by Polyethylene and Various Graphite Plate Thicknesses." LANL. LA-UR-05-9237_Draft, January 2006.
(86) Loiaza, D., et al., "Hand-stacking for Molybdenum Critical Experiments Fueled by Highly Enriched Uranium and Moderated by Polyethylene and Various Graphite Plate Thicknesses." LANL. LA-UR-05-9238_Draft, January 2006.
(87) Loiaza, D., et al., "Hand-stacking for Niobium – (1%) Zirconium Critical Experiments Fueled by Highly Enriched Uranium and Moderated by Various Graphite Plate Thicknesses." LANL. LA-UR-05-9239_Draft, January 2006.

(88) Loiaza, D. et al., "Hand-stacking for Tantalum – 2.5 wt% Tungsten Critical Experiments Fueled by Highly Enriched Uranium and Moderated by Polyethylene and Various Graphite Plate Thicknesses." LANL. LA-UR-05-9240, January 2006.

(89) Sadasivan, P. et al., “DAF Critical Experiment Estimates.” LANL. LA-CP-05-1176, September 26, 2005.

Argonne National Laboratory and Idaho National Laboratory

(90) K. N. Grimm, R. M. Lell, R. D. McKnight, and R. W. Schaefer, “ZPPR-20 Phase C: A Cylindrical Assembly of U Metal Reflected by Beryllium Oxide,” HEU-MET-FAST-075, International Handbook of Evaluated Criticality Safety Benchmark Experiments, OECD Nuclear Energy Agency, NEA/NSC/DOC(95)03, September 2005 Edition.

(91) R. M. Lell, K. N. Grimm, R. D. McKnight, and R. W. Schaefer, “ZPPR-20 Phase D: A Cylindrical Assembly of Polyethylene-Moderated U Metal Reflected by Beryllium Oxide and Polyethylene,” SUB-HEU-MET-MIXED-001, International Handbook of Evaluated Criticality Safety Benchmark Experiments, OECD Nuclear Energy Agency, NEA/NSC/DOC(95)03, September 2005 Edition.

(92) R. W. Schaefer, K. N. Grimm, R. M. Lell, and R. D. McKnight, “ZPPR-20 Phase E: A Cylindrical Assembly of U Metal Reflected by Beryllium Oxide and Sand,” SUB-HEU-MET-FAST-001, International Handbook of Evaluated Criticality Safety Benchmark Experiments, OECD Nuclear Energy Agency, NEA/NSC/DOC(95)03, September 2005 Edition.

Sandia National Laboratory

(93) Wright, Steven A. et al. “Space-Based Gas-Cooled Reactor Design, Interim Report.” Sandia National Laboratory. September 2005.

## 3.2 Reactor Module Primary Plant Segment

Los Alamos National Laboratory

(94) Lin, J.C. and R.C. Johns. "Implementation of a Brayton Model in Trace." LANL. LA-CP-05-0948, August, 2005.

Idaho National Laboratory

(95) Murray, P.E., "An Electromagnetic Annular Liquid Induction Pump Model for RELAP5-3D." INL. May 2005.

(96) Davis, C.B., et al. “Implementation of Helium-Xenon Thermodynamic Properties and Transport Properties into RELAP5-3D/ATHENA.” INEEL. R5/3D-04-10, December 2004

(97) Weaver, W.L., “Software Design, Implementation, and Verification Document R5/3D-05-02, Upgrade Compressor Model,” January 5, 2005

Marshall Space Flight Center

(98) Houts, Michael G. “Heat Exchanger Thermal and Structural Analysis.” MSFC. NP50 (05-012), March 31, 2005.

Contractor Reports on Gas Cooler Development

(99) Heatric, Dorset, UK. "Design Study Phase 1 - Final Report." DR1-H1026. September 1, 2005.

(100) Hamilton Sundstrand, Windsor Locks, CT. “Evaluation of Gas Cooler Materials of Construction and Reliability, Scoping Phase Final Report.” Bechtel Bettis Purchase Order #3007352, August 30, 2005.

(101) Holtec International, Marlton, NJ. "Gas Cooler Materials of Construction and Reliability for Space Based Nuclear Power Plant." Bechtel Bettis Purchase Order, Holtec Report HI-2053418, Project 1489, September 2005.
(102) Honeywell, Torrance, CA. "Nuclear Propulsion System Gas Cooler Scoping Phase Design Study." Bechtel Bettis Purchase Order #3007132, Honeywell Report 05-73670, October 18, 2005.

## 3.3 Reactor Module Shield Segment

Oak Ridge National Laboratory

(103) Blakeman, E. D. "Quick Mass Analysis for Reactor Shields (QMARS): A Space Reactor Shield Sizing Tool, Revision 1." ORNL. ORNL/LTR/NR-JIMO/05-04, February 2005.
(104) Bucholz, J.A. "Dose Rate Outside a Reactor Coolant Pipe Due to Activated Coolant." ORNL. ORNL/LTR/NR-JIMO/05-02, January 2005.
(105) Yugo, J.J. "Multi-layer Radiation Shield Optimization Studies and Simulations." ORNL. ORNL/LTR/NR-JIMO/05-03, February 2005.
(106) "Shielding Analysis, Methods Qualification Test Outline, Cross Section Development, And Methods Development." ORNL. ORNL-SPP-05-0042, April 26, 2005.
(107) Williams, M. L. "Application of Sensitivity and Uncertainty Analysis to Space Reactor Shielding Designs." ORNL. ORNL/LTR/NR-PROM1/05-26, September 2005.
(108) Williams, M.L. and S. Goluoglu. "Sensitivity and Uncertainty Analysis for Space Reactor Shielding Calculations." ORNL. ORNL/LTR/NR-PROM1/05-16, June, 2005.
(109) Patton, B.W. and J. O. Johnson. "Preliminary Investigations into Applicable Space Reactor Shielding Benchmark Experiments and Facilities for Methods Qualification Testing." ORNL. ORNL/LTR/NR-PROM1/05-25, September 2005.
(110) Greene, N.M., et al. "Status Report: Cross-Section Library Development to Support Shielding Analyses for Space Reactor Applications - DRAFT." ORNL. ORNL/LTR/NR-PROM1/05-33, October 2005.
(111) Lillie, R.A. "GRTUNCL3D: Improvements and Extensions - DRAFT." ORNL. ORNL/LTR/NR-PROM1/05-31, October 2005.
(112) Slater, C.O. "User's Manual for the FALSTF3D Last-Flight Estimation Computer Code - DRAFT." ORNL. ORNL/LTR/NR-PROM1/05-30, October 2005.
(113) Pevey, R.E. and R.A. Lillie. "MTT: An Analysis Tool to Link MCNP to TORT - DRAFT." ORNL. ORNL/LTR/NR-PROM1/05-32, October 2005.
(114) Johnson, J. O. "Qualification Basis for Shield Design Computational Tools: Preliminary Evaluation of Cross-Section Data." ORNL. ORNL/LTR/NR-JIMO/04-11, September 2004.
(115) Johnson, J. O. "Qualification Basis for Shield Design Computational Tools: Preliminary Evaluation of Space Reactor Shielding Benchmark Experiments." ORNL. ORNL/LTR/NR-JIMO/04-10, September 2004.
(116) Yugo, J.J. and Blakeman, E.D. "Monolithic Radiation Shield Material Studies and Simulations." ORNL. ORNL/LTR/NR-JIMO/05-23, October 2005.
(117) Bucholz, J.A., et al. "Reactor Shield Optimization in Support of the Prometheus 1 Project Preconceptual Design Analyses." ORNL. ORNL/LTR/NR-PROM1/05-27, October 2005.

## 3.4 Reactor Module Instrumentation and Control Segment

Oak Ridge National Laboratory

(118) Bell, Z.W., et al. "Fission Chamber Development." Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-17, July 2005.

(119) Bryan, W. L., et al. "Jupiter Icy Moons Orbiter Mission Reactor Instrumentation and Controls Architecture Alternatives Studies." Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-06, December 2005.
(120) Emery, M. S., et al. "Sensor Cable Conceptual Design." Draft. Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-20, August 2005.
(121) Hardy, James E., et al. "Ultrasonic Flow Meter System Conceptual Design." Draft. Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-19, August 2005.
(122) Holcomb, David E., et al. "JIMO Reactor Sensor Technology Development Plan." Oak Ridge National Laboratory. ORNL/LTR/NR-JIMO/05-01, February 2005.
(123) Kisner, Roger A., et al. "Resistance Temperature Detector Conceptual Design for Space Nuclear Power Systems." Draft. Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-11. June 2005.
(124) Kisner, Roger A., et al. "Thermocouple Conceptual Design for Space Nuclear Power Systems." Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-05, October 2005.
(125) Wilson, Jr., T. L. et al. "Prometheus Sensor Test Plan." Draft. Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-12, June 2005.
(126) Wood, Richard T., et al. "Autonomous Control for Space Nuclear Power Plants." Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-07, December 2005.
(127) Wood, Richard T., et al. "Overview of Regulations, Requirements, and Guidelines for I&C Systems in Civilian Nuclear Reactors." Oak Ridge National Laboratory. ORNL/LTR/NR-PROM1/05-13, October 2005.

<u>Contractor Reports</u>
(128) Lockheed Martin Missiles and Fire Control, Archibald, PA. "Close Out Report for Space Reactor Ultrasonic Coolant Flow Measurement Conceptual Development." Purchase Order G63996. Report Number 6506-002, December, 9, 2005.
(129) Northrop Grumman Electronic Systems, Baltimore, MD. "Space Pre-Amplifier Design Close-Out Report." Purchase Order G63998. Report Number NGG63998CLOSE, January 3, 2006.
(130) Northrop Grumman Electronic Systems, Baltimore, MD. "Space Detector Close-Out Report." Purchase Order G62000. Report Number NGG62000CLOSE, January 3, 2006.

## 3.5 Reactor Module Integrated Testing

<u>NASA Marshall Space Flight Center</u>
(131) Bragg-Sitton, S., et al., "Heater Development, Fabrication and Testing: Analysis of Fabricated Heaters." NASA Marshall Space Flight Center. ER11-05-WI1-001, March 30, 2005.
(132) Bragg-Sitton, S. and Webster, K., "Application of Simulated Reactivity Feedback in Non-Nuclear Testing of a Direct Drive Gas Cooled Reactor." NASA Marshall Space Flight Center. ER11-05-WI4c-001, October 14, 2005.
(133) Bragg-Sitton, S., et al., "Heater Development, Fabrication and Testing: Update." NASA Marshall Space Flight Center. ER11-05-WI1-002, October 15, 2005.
(134) Bryhan, A.J., et al. "Cost Estimate for a Tantalum Alloy Refractory Metal Flow Circuit Concept Based on a Stainless Steel Lithium Circuit." NASA Marshall Space Flight Center. ER11-05-WI4B-003, May 10, 2005.
(135) Godfroy, T.J., "Final Report – Documentation of Stainless Steel, Lithium Circuit Test Section Design." NASA Marshall Space Flight Center. ER11-05-WI4a-001, June 24, 2005.

(136) Godfroy, T.J., et al. "Final Report-Documentation of Status of Direct Drive Gas Cooled Reactor Simulator." NASA Marshall Space Flight Center. ER11-05-WI4C1-001, October 15, 2005.
(137) Hickman, R.R., et al., "Cost Estimate for a Mo-47.5% Re Refractory Metal Flow Circuit Concept Based on a Stainless Steel Lithium Circuit." NASA Marshall Space Flight Center. ER11-05-WI4B-001.0, March 10, 2005.
(138) Hissam, D.A. and Stewart E., "Evaluation of an Integrated Direct-Drive Gas-Cooled Reactor Simulator and Brayton Cycle, Final Report." NASA Marshall Space Flight Center. ER11-05-WI4C3-002, September 30, 2005.
(139) Majumdar, A. "Lithium Cooled Reactor System Analysis For Pressure Transducer Placement." NASA Marshall Space Flight Center. ER11-04-WI4A-003, January 2005.
(140) Martin, J.J., et al., "Design of Refractory Metal Life Test Heat Pipe and Calorimeter." NASA Marshall Space Flight Center. ER11-04-WI2-001.1, December 16, 2004.
(141) \Martin, J.J., et al., "Design of Refractory Metal Heat Pipe Life Test Environment Chamber, Cooling System, and Radio Frequency Heating System." NASA Marshall Space Flight Center. ER11-05-WI2-003, February 22, 2005.
(142) Martin, J.J., et al., "Refractory Metal Heat Pipe Life Test – Test Plan and Standard Operating Procedures." NASA Marshall Space Flight Center. ER11-05-WI2-002, dated February 11, 2005.
(143) Martin, J.J., et al., "Closeout Report for the Refractory Metal Accelerated Heat Pipe Life Test Activity." NASA Marshall Space Flight Center. ER11-05-WI2-004, June 16, 2005.
(144) Reid, R., et al., "Heat Transfer and Pressure Drop in Concentric Annular Flows of Binary Inert Gas Mixtures." NASA Marshall Space Flight Center. ER1105-WI4d-001, October 28, 2005.
(145) Steward, E., "Pressure Transducer Stand-Off Tube Length Estimates." NASA Marshall Space Flight Center. ER11-04-WI4A-002, December 16, 2004.

Naval Reactors Program Vendors

(146) EB Letter 472-SPP-4389/JHC/7.1.3, "Space Nuclear Power Plant (SNPP), Rough Order of Magnitude (ROM) Budgetary Cost Estimate for Reactor Module Integration Contractor (RMIC) Effort," September 30, 2005
(147) NGNN Letter 1143T-0052, "Rough Order of Magnitude (ROM) Estimate for Space Reactor Program; Bechtel Bettis Reactor Plant Planning Yard (RPPY) Purchase Order 2101506 (NGNN Job Order 1143T)," August 31, 2005
(148) BWXT Letter CP-054-233, "ROM Cost Estimate for Space Reactor Assembly and Facility Modifications," August 19, 2005

## 3.6 Reactor Module Safety and Mission Assurance

Oak Ridge National Laboratory (ORNL)

(149) Kerley, Gerald I. "An Equation of State for Uranium Nitride." ORNL. ORNL/LTR/NR-PROM1/05-01, April 2005.
(150) Kerley, Gerald I. "Equations of State for Space Reactor Materials." ORNL. ORNL/LTR/NR-PROM1/05-02, April 2005.
(151) Hale, R.E. "JIMO Nuclear Safety Analysis and Testing Plan Outline (DRAFT)." ORNL. ORNL/LTR/NR-JIMO/04-06, August 2004.
(152) Kim, S.H., et al. "Impact Analysis for Candidate Reactor Core Concept Designs." ORNL. ORNL/LTR/NR-PROM1/05-04, September 2005.

Sandia National Laboratory (SNL)

(153) Lenard, R. X., et al. "Analytical Approach and Codes for Space Reactor Safety Analysis, Final Report." Sandia National Laboratory. October 2005.

## 3.7 Reactor Module Materials Test and Evaluation

Los Alamos National Laboratory (LANL)

(154) Scherer, Carolynn P. "Minican Weld Progress Report as of 22 June 2005." LANL. LA-CP-05-0760, July 11, 2005.
(155) Scherer, Carolynn P. "Summary Report for LANL UN Fuel Fabrication." LANL. LA-CP-05-1394, December 21, 2005.
(156) Mason, Richard E. "Thermal Conductivity of Uranium Nitride Fuel." LANL. LA-CP-04-0828, September 27, 2004.
(157) Mason, Richard E. "Fuel Pin Performance Database." LANL. LA-CP-05-0116, February 24, 2005.
(158) Mason, Richard E., et al. "Fuel Pin Performance Database Evaluation and Analysis." LANL. LA-CP-05-0117, March 9, 2005.
(159) Sims, Jeff. "Feasibility Study for the UN Fuel Fabrication Line Project at the Los Alamos National Laboratory." DMJMH+N, Inc. 1197-T228-1904-001-04, Rev. 0, December 8, 2004.
(160) Scherer, Carolynn P. "Report on SP-100 Fuel Pellet Fabrication and Storage." LANL. LA-CP-04-0611, June 28, 2004.
(161) Scherer, Carolynn P. "Report on Fabrication Issues for 0.5"-Diameter Pellets." LANL. LA-CP-04-0612, June 28, 2004.
(162) Scherer, Carolynn P. "Steps and Costs to Receive, Analyze, and Store SP-100 UN Fuel Pellets Shipped from Y-12." LANL. LA-CP-04-0703, September 2, 2004.
(163) Scherer, Carolynn P. "Inventory Existing Equipment for UN Fuel Pellet Fabrication." LANL. LA-CP-04-0710, September 2, 2004.
(164) Scherer, Carolynn P. "LANL Facility Modifications to Produce Small-Scale Batches of Depleted UN Fuel." LANL. LA-CP-04-0717, September 8, 2004.
(165) Scherer, Carolynn P. "LANL Facility Modifications for Small-Scale Work: Manipulation of Fuel Pellets from Storage." LANL. LA-CP-04-0728, September 20, 2004.
(166) Scherer, Carolynn P. "Preliminary Cost Estimate for the Fuels and Materials Test Station for the Jupiter Icy Moons Orbiter." LANL. LA-CP-04-0734, September 30, 2004.
(167) Scherer, Carolynn P. "LANL Facility Modifications for Small-Scale Work: UN Fuel Pellet Fabrication at TA-55." LANL. LA-CP-04-0831, November 5, 2004.
(168) Wilson, Stephen P. "The Feasibility of Incorporating Spectral Shift Poisons in UN Fuel." LANL. LA-CP-04-0724, October 18, 2004.
(169) Scherer, Carolynn P. "Report on Why Cermets Were Not Selected for SP-100." LANL. LA-CP-04-0609, June 28, 2004.
(170) Scherer, Carolynn P. "Report on Identification of Cermet Systems for Fuel Applications Above 850K." LANL. LA-CP-04-0610, June 28, 2004.
(171) Scherer, Carolynn P. "Fission Product Effects on Fissile Fueled Cermets During Irradiation." LANL. LA-CP-04-0804, October 5th, 2004.
(172) Scherer, Carolynn P. "Literature Review of Thermal Performance of High-Temperature Fissile Fueled Cermet Systems." LANL. LA-CP-05-0068, January 21, 2005.
(173) Schrage, Dale L. "Tungsten Target Sizing and Cooling Calculations for JIMO FMTS: First Iteration." LANL. LA-CP-04-0743, September 30, 2004.
(174) Schrage, Dale L. "Preliminary System and Equipment Layout for JIMO FMTS." LANL. LA-CP-04-0744, September 30, 2004.
(175) Schrage, Dale L. "FMTS HEBT Preconceptual Design." LANL. LA-CP-04-0745, September 30, 2004.
(176) Schrage, Dale L. "FMTS Preconceptual Beam-Diagnostics Design." LANL. LA-CP-04-0789, September 30, 2004.
(177) Schrage, Dale L. "Cost Estimate for FMTS Annual Operations." LANL. LA-CP-04-0836, September 30, 2004.

(178) Schrage, Dale L. "Assessment of Material Irradiation Testing Issues for the LANSCE Fuels and Materials Test Station." LANL. LA-CP-04-0894, January 5, 2004.
(179) Schrage, Dale L. "Neutronics Assessment of the LANSCE Materials Test Station as an Irradiation Facility for the JIMO Space Reactor." LANL. LA-CP-04-0903, January 13, 2004.
(180) Scherer, Carolynn P. "Data Package for Run 9." LANL. LA-CP-05-1320, November 14, 2005.
(181) Scherer, Carolynn P. "Data Package for Run 7D." LANL. LA-CP-05-1321, November 14, 2005
(182) Scherer, Carolynn P. "Cost Estimate for UN Fabrication Core Capability; Option A: Redo All Gloveboxes; Option B: Keep Existing Gloveboxes." LANL. D-5-05-062 / LA-CP-04-0881, November 22, 2004.

<u>Oak Ridge National Laboratory (ORNL)</u>
(183) Snead, L.L. and A. T. Nelson. "Candidate Materials for Space Reactor Shielding Applications." ORNL. ORNL/LTR/NR-PROM1/05-15, July 2005.
(184) Pawel, S. J. "Capsule Compatibility Testing for the Prometheus Program." ORNL. ORNL/LTR/NR-PROM1/05-29, October 2005.
(185) Pawel, S.J., et al. "Assessment of Available Data on the Compatibility of Liquid Alkali Metals with a Set of Candidate Structural Alloys." ORNL. ORNL/LTR/NR-JIMO/04-04, September 2004.
(186) Tortorelli, P.F., et al. "Burden Modeling of Contaminant Transfer in Gas-Cooled Power Conversion Systems." ORNL. ORNL/LTR/NR-PROM1/05-34, November 2005.
(187) Zinkle, Steven J., et al. "Critical Assessment of Structural Materials for Space Nuclear Applications." ORNL. ORNL/LTR/NR-JIMO/04-08, September 2005.
(188) Sokolov, M.A. "Fracture Toughness Characterization of Two Mo-Re Alloys." ORNL. ORNL/LTR/NR-JIMO/04-16, February 2005.
(189) Pawel, S.J. "Flowing Gas Loop Development." ORNL. ORNL/LTR/NR-PROM1/05-28, October 2005.
(190) Wilson, D.F. "ORNL Recirculating Helium Loop and Its Application to Burden Modeling." ORNL. ORNL/LTR/NR-PROM1/05-36, October 2005.
(191) McGreevy, T. E. and C. E. Duty. "Interim Report on High Temperature Creep Testing of Refractory Metals." ORNL. ORNL/LTR/NR-PROM1/05-08, May 2005.
(192) McGreevy, T. E. and C. E. Duty. "High Temperature Creep Testing of Refractory Metals." ORNL. ORNL/LTR/NR-PROM1/05-37, November 2005.
(193) McGreevy, T. E. and C. E. Duty. "Second Interim Report on High Temperature Creep Testing of Refractory Metals." ORNL. ORNL/LTR/NR-PROM1/05-18, July 2005.
(194) Busby, J.T., et al. "Refractory Metals Irradiation Testing in HFIR." ORNL. ORNL/LTR/NR-PROM1/05-24, September 2005.
(195) Busby, J.T. "The Compatibility of ZrC with Refractory Metal Alloys." ORNL. ORNL/LTR/NR-PROM1/05-14, June 2005.
(196) Busby, J.T., et al. "Effects of Neutron Irradiation on Refractory Metal Alloys." ORNL. ORNL/LTR/NR-PROM1/05-38, December 2005.
(197) Leonard, K.J., et al. "Microstructural Characterization of Annealed and Aged Refractory Alloys." ORNL. ORNL/LTR/NR-PROM1/05-35, November 2005.
(198) Busby, J.T., et al. "Characterization of Annealed and Aged Refractory Alloys – A First Progress Report." ORNL. ORNL/LTR/NR-PROM1/05-22, August 2005.
(199) Robertson, J.P. "Report on Evaluation of the Feasibility of Shipping Irradiated Structural Materials from the JOYO Test Reactor Located in Oarai, Japan." ORNL. ORNL/LTR/NR-PROM1/05-10, June 2005.
(200) Katoh, Y. and L.L. Snead. "Preparation of Silicon Carbide Irradiation Creep Specimens and Holders." ORNL. ORNL/LTR/NR-PROM1/05-39, October 2005.

(201) Hoelzer, D.T., et al. "Critical Review of Solubility, Diffusivity, and Metallurgical Effects in Group V and VI Metals." ORNL. ORNL/LTR/NR-JIMO/04-01, September 2004.
(202) McGreevy, T.E. "Outstanding Issues Concerning the Mechanical Behavior of Nb-1Zr Alloys." ORNL. ORNL/LTR/NR-JIMO/04-02, September 2005.
(203) Hoelzer, D.T. and S. A. Speakman. "Interim Report on the Recovery and Recrystallization Behavior of Nb-1Zr." ORNL. ORNL/LTR/NR-JIMO/04-07, September 2005.
(204) Snead, L.L., et al. "Development of Shielding Material Design Performance Database and Summary of Alternative to LiH-Based Designs." ORNL. ORNL/LTR/NR-JIMO/04-03, December 2005.
(205) Zinkle, S.J., et al. "Historical Basis for Selection of Nb-1Zr Cladding for Space Reactor Applications." ORNL. ORNL/LTR/NR-JIMO/04-05, October 2004.
(206) Bucholz, J.A., E.D. Blakeman, and J.O. Johnson. “Radiation Transport Analyses to Support the Reactor Shield Materials Selection.” ORNL. ORNL/LTR/NR-PROM1/05-21, August 2005.
(207) Snead, L.L., and A.T. Neslon. “Candidate Materials for Space Reactor Shielding Applications.” ORNL. ORNL/LTR/NR-PROM 1/05-15, July 2005.
(208) Sneade, L.L. “Development of Shielding Material Design Performance Database and Summary of Alternative to LiH-Based Designs.” ORNL. ORNL/LTR/NR-JIMO/04-03, December 2005.

Pacific Northwest National Laboratory (PNNL)

(209) Fabrication of Parts for Structural Materials Irradiation Rigs (SMIRs) for JOYO Irradiation. PNNL, Battelle. SRM-SOW-001, Rev. 1, April 21, 2005.
(210) Delucchi, T.A., et al. "Biaxial Creep Specimen Electron Beam and Laser Seal Welding Demonstration Report." PNNL. PNNL-15537, Rev. 0, December, 2005.
(211) Delucchi, T.A., et al. "Qualification Test Plan for Biaxial Creep Specimen Electron Beam and Laser Seal Welding. PNNL. SRM-PLAN-005, Rev. 0., August, 2005.
(212) Geelhood, Ken. "Material Thermal Properties for ASTAR-811C." PNNL. PNNL-15440, September, 2005.
(213) Geelhood, Ken. "Material Thermal Properties for BeO." PNNL. PNNL-15494, October, 2005.
(214) Geelhood, Ken. "Material Thermal Properties for FS-85." PNNL. PNNL-15458, October, 2005.
(215) Geelhood, Ken. "Material Thermal Properties for Haynes 230." PNNL. PNNL-15493, October, 2005.
(216) Geelhood, Ken. "Material Thermal Properties for Inconel 617." PNNL. PNNL-15495, October, 2005.
(217) Geelhood, Ken. "Material Thermal Properties for Mo-47Re." PNNL. PNNL-15441, September, 2005.
(218) Geelhood, Ken. "Material Thermal Properties for Nimonic PE-16." PNNL. PNNL-15496, October, 2005.
(219) Wootan, David. "Heating Rates and Neuton Flux for JOYO-1 Irradiation Tests - Revision 2." PNNL. PNNL-15457, September, 2005.
(220) Geelhood, Ken. "Material Thermal Properties for Rhenium." PNNL. PNNL-15459, October, 2005.
(221) Geelhood, Ken. "Material Thermal Properties for SiC." PNNL. PNNL-15497, October, 2005.
(222) Geelhood, Ken. "Material Thermal Properties for SiC/SiC." PNNL. PNNL-15492, October, 2005.
(223) Geelhood, Ken. "Material Thermal Properties for W-25Re." PNNL. PNNL-15461, October, 2005.

(224) Geelhood, Ken. "Material Thermal Properties for Tungsten." PNNL. PNNL-15460, October, 2005.
(225) Senor, D. and Painter, C. "Space Reactor Materials Irradiation Testing Project Summary Report: Irradiation of Advanced Structural Materials in JOYO to Support the Development of a Space Fission Reactor." PNNL. PNNL-15610, April 2006.
(226) Senor, D.J., Paxton, DM, Prichard, AW, "Feasibility Study of Overseas High Fluence Irradiation Test in Support of Project Prometheus: Final Assessment of Candidate Reactor Facilities," PNNL. PNNL-15124, March, 2005.
(227) Wootan, David. "Heating Rates and Neutron Flux for JOYO-1 Irradiation Tests – Revision 1." PNNL. SRM-LRPT-002, July 2005.
(228) Wootan, David. "Adjacent Fuel Effects for JOYO-1 Irradiation Tests." PNNL. SRM-LRPT-005, July 2005.
(229) Paxton, Dean. "Manufacturing and Quality Plan for Fabrication of Biaxial Creep Specimens." PNNL. SRM-PLAN-004, August 2005.
(230) Geelhood, Ken, et al. "Development and Applicability of Fuel Models in FRAPCON and FRAPTRAN." PNNL. PNNL-SPP-05-0006, May 18-19, 2005.
(231) Wootan, David. "Review of NRPCT Interpretation ofJNC Supplied Gamma Heating Rates." PNNL. SRM-LRPT-001, May 2005.

Y-12
(232) "Y-12 Support of Knolls Atomic Power Laboratory Jupiter Icy Moons Orbiter Project." BWXT Y-12 L.L.C. SEPRD-TPLT-08-V1-0. September 2004.

Contractor Reports
(233) QuesTek Innovations LLC, Evanston, IL. "Application of Computational Materials Design to Assess and Develop Structural Materials for Space Nuclear Applications, Final Report. KAPL Purchase Order PL00111933 AG. Jan. 2005.
(234) Scoles, Stephen. "IV-256 "Space Fuel" Final Report." BWX Technologies Inc., Nuclear Products Division. BWED-05-135 Revision 01, September 27, 2005.

# 4 Reactor Module Reactor Segment

## 4.1 Heat Transfer

(235) AiResearch, A Los Angeles Division, "Heat Transfer Testing of Low Prandtl-Number Mixtures in a Plate-Fin Surface." 90-63268, April 12, 1990.
(236) Churchill, S.W., "Comprehensive correlating equations for heat, mass and momentum transfer in fully developed flow in smooth tubes." Ind, Engng Chem. Fundam. 16(1), 1977. pp 109-116,.
(237) Dale, C.B., "Design Assumptions and Methodologies Used in the Feasibility Studies." LA-CP-04-0938, December 17, 2004.
(238) Giacobbe, F.W., "Heat transfer capability of selected binary gaseous mixtures relative to helium and hydrogen." App. Thermal Engng, Vol. 18 No. 3-4.,1998. pp 199-206,.
(239) Gordeev, S., Heinzel, V., Slobodtchouk, V., "Features of convective heat transfer in heated helium channel flow." Int. J. Heat Mass Transfer 48, 2005. pp 3363-3380.
(240) Kays, W.M., 1966. *Convective Heat and Mass Transfer*. McGraw-Hill, New York.
(241) Kays, W.M., Crawford, M.E., 1993. *Convective Heat and Mass Transfer*. McGraw-Hill, New York.
(242) McEligot, D.M. and Jackson, J.D., "Deterioration criteria for convective heat transfer in gas flow through non-circular ducts." Nuclear Engineering and Design 232, 2004. pp 327-333.

(243) McEligot, D.M. and McCreery, G.E., “Scaling Studies and Conceptual Experiment Designs for NGNP CFD Assessment.” Nov. 30, 2004. Available from INEEL/EXT-04-02502.

(244) McEligot, D.M. and Taylor, M.F., “ The turbulent Prandtl number in the near-wall region for low-Prandtl-number gas mixtures.” Int. J. Heat Mass Transfer. Vol. 39, No. 6, pp 1287-1295. 0017-9310(95)00146-8.

(245) Mikielewicz, D.P., Shehata, A.M., Jackson, J.D., McEligot, D.M., “Temperature, velocity and mean turbulence structure in strongly heated internal gas flows.” Comparison of numerical predictions with data, Int. J. Heat Mass Transfer 45, 2002. pp 4333-4352.

(246) Petukhov, B.S., “Heat transfer and friction in turbulent pipe flow with variable physical properties.” Adv. Heat Transfer 6, 1970. pp 503-564.

(247) Pickett, P.E., “Heat and momentum transfer to internal turbulent flow of helium-argon mixtures in circular tubes.” MSE Report, Aero. Mech. Engng Dept., Univ. Arizona, 1976. Available from NTIS as AD-A167290.

(248) Pierce, B.L., “The Influence of Recent Heat Transfer Data on Gas Mixtures (He-Ar, $H_2$-$CO_2$) on Closed Cycle Gas Turbines.” Transactions of the ASME. Vol. 103, Jan. 1981. pp 114-117.

(249) Richards, A.H., Spall, R.E., McEligot, D.M., “Numerical simulation of a strongly heated gas flowing upward in a vertical tube using a $\kappa$-$\epsilon$ model.” ASME Summer Heat Transfer Conference, July 17-22, 2005. Available from ASME as HT2005-72329.

(250) Serksnis, A.W., “Turbulent flow of hydrogen-carbon dioxide mixtures in heated tubes.” MSE Report, Aero. Mech. Engng Dept., Univ. Arizona, 1977. Available from NTIS as AD-A062442

(251) Shashkov, A.G. and Abramenko, T.N., “Thermal Conductivity of Gas Mixtures.” Inzhenerno-Fizicheskii Zhurnal, Vol. 19, No. 4, Oct. 1970. pp 761-762.

(252) Shehata, A.M. and McEligot, D.M., “Mean structure in the viscous layer of strongly-heated internal gas flows. Measurements.” Int. J. Heat Mass Transfer 41, 1998. pp 4297-4313.

(253) Shehata, A.M. and McEligot, D.M., “Mean structure in the viscous layer of strongly-heated internal gas flows.” Tech. report INEL-95/0223, Idaho National Engineering Laboratory, 1995.

(254) Sleicher, C.A. and Rouse, M.W., “A convenient correlation for heat transfer to constant and variable property fluids in turbulent pipe flow.” Int. J. Heat Mass Transfer 18, 1975. pp 677-683.

(255) Spall, R.E., Richards, A., McEligot, D.M., “An assessment of $\kappa$-$\omega$ and $\upsilon^2$-f turbulence models fro strongly heated internal gas flows.” Taylor & Francis Inc., Num. Heat Transfer, Part A, 45:831-849, 2004. Available from ISSN: 1040-7782.

(256) Symolon, P.D., “A heat transfer correlation for turbulent mixed convection in a vertical tube with flow development effects.” ASME-HTD-Vol. 346, Aug. 1997.

(257) Taylor, M.F., Bauer, K.E., McEligot, D.M., “Internal forced convection to low-Prandtl number gas mixtures.” Int. J. Heat Mass Transfer 31, 1988. pp 13-25.

(258) Taylor, M.F., Bauer, K. E., McEligot, D.M., “Internal forced convection to low-Prandtl-number gas mixtures.” Interium technical report, ONR Contract N00014-75-C-0694, Aero. Mech. Engng Dept., Univ. Arizona (1984). Available from NTIS as ADA148932

(259) Todreas, N.E., Kazimi, M.S., 1993. *Nuclear Systems 1, Thermal Hydraulic Fundamentals*, pp 443-444, Taylor & Francis.

(260) Vanco, M.R., “Analytical comparison of relative heat-transfer coefficients and pressure drops of inert gases and their binary mixtures.” NASA TN D-2677.

(261) Symolon, P., et al., “Mixed Convection Heat Transfer Experiments in Smooth and Rough Vertical Tubes.” Lockheed Martin. Schenectady, NY.

## 4.2 Silicon Carbide

(262) Lee, et al., "Study on the Mechanical Properties and Thermal Conductivity of Silicon-Carbide, Zirconia and Magnesia Aluminate-based Inert Matrix Nuclear Fuel Material after Cyclic Thermal Shock." J. Nuclear Materials 319, 2003. pp 15-23 from Korean Atomic Energy Research Institute.

(263) Purdue University & Framatom. "The Polymer Impregnation and Pyrolysis Method for Producing Enhanced Concductivity LWR Fuels. " Proceedings of 2004 International Meeting on LWR Fuel Perfomrance, Orlando, FL, September 2004, paper 1028.

(264) Lee J., et al., "Fabrication of short-fiber reinforced SiC composites by polycarbosilane infiltration," Journal of European Ceramic Society vol. 24, 2004. pp 24-31.

(265) Dong, et al., "Polymer impregnations and pyrolysis (PIP) method for the preparation of laminated woven fabric/mullite matrix composites with pseudo ductility." J. European Ceramic Socity vol. 24, 2004. pp 53-64.

(266) Verral, et al.,"Silicon Carbide as an inert-matrix for a thermal reactor fuel." Journal of Nuclear Materials vol. 274. 1999. pp 54-60.

(267) Lewinsohn, et al., "Silicon carbide based materials for joining silicon carbide composites for fusion energy applications." J. Nuclear Materials, vol 307-311. 2002. pp1232-1236.

(268) Allen, et al. "The high temperature reactions between uranium dioxide and silicon carbide." Central Electricity Generating Board, Berkley Nuclear Laboratories, Gloucestershire, UK, 1970.

(269) Katoh, et al., "Properties and radiation effects in high temperature pyrolyzed PIP SiC/SiC." J. Nuclear Materials, vol. 289. 2001. pp 42-47.

(270) Kotani, et al. "Effect of SiC partical dispersion on microstructure of polymer derived SiC/SiC composite. " Material Science and Engineering vol. A357. 2003. pp 376-385.

(271) MIT, Center for Advanced Nuclear Energy Systems – Modified FRAPCON for modeling UO2 with silicon carbide composite cladding

(272) Nuclear Safety Research at Juelich Germany – Nuclear Safety Research Program – investigating the advantages of advanced fuel elements for LWR – additional encapsulation of uranium dioxide pellets with silicon carbide – (www.fz-juelich.de/scientific-report-2004/index.php?item=37&lang=en).

(273) Feinroth, et al., NERI project by Gamma Engineering on SiC composite clad fuel for LWR, "Progress in Developing an Impermeable, High Temperature Ceramic Composite for Advanced Reactor Clad and Structural Applications."

(274) Yoichiro, et al. Japan. "PWR using HTGR fuel concept with cladding for ultimate safety." Advanced Reactors with Innovative Fuels (ARWIF) 2001 Conference UK

(275) Young, et al. "Application of Ceramic Nuclear Fuel Materials for Innovative Fuels and Fuel Cycles."

(276) Newton T.D., et al. "A Helium Cooled Particle Reactor for Fuel Sustainability." Winfrith labs UK.

(277) Heinisch, et al. "Displacement damage in silicon carbide irradiated in fission reactors." Journal of Nuclear Materials 327. April 2004. pp 175-181.

(278) Chu, Z.Y., et al. "Formation mechanism of interior micro flaws in the preparation of polymer derived silicon carbide." Key Engineering Materials, May 2004. pp 280-283.

(279) Lippmann, et al. "Laser joining of silicon carbide –a new technology for ultra high temperature resistant joints." Nuclear Engineering and Design 23, June 2004. pp 151-161.

(280) "Feasibility Study of Supercritical Light Water Cooled Reactors for Electric Power Production." DOE, DOE/SF/22533, Feb. 13, 2005.

## 4.3 Other Reactor Design

(281) Rockwell International, Space Reactor Electric Systems: Subsystem Technology Assessment, ESG-DOE-13398, dated March, 1983.

(282) J.A. Horak and M. Kangilaski, "Effects of irradiation of the tensile properties of rhenium," in B.D. Bryskin Rhenium and Rhenium Alloys, TMS Symposium Proceedings, The Minerals, Metals & Materials Society, Warrendale, PA 1997, pg 535-549.

(283) Lessmann and Gold, "determination of Weldability and Elevated Temperature Stability of Refractory Metal Alloys," NASA CR – 1608 September 1970.

(284) Gold and Begley, "Investigation of High Temperature fracture of T-111 and Astar 811C," NASA CR-72859 April 1971.

(285) Wiffen, "Effects of Irradiation of Properties of Refractory Alloys with Emphasis on Space Power Reactor Applications," Space Power Conference Proceedings, 1995.

(286) Bianco, Luther, Mueller, "Results of Biaxial Creep testing of High temperature Superalloy and refractory metal Alloys, Interim report," November 21, 1997.

(287) Grossbeck, et al., Trans SP-100 program sessions Jan. 1987 as reported ORNL/LTR/JIMO/04-05, February 2004.

(288) Wiffen, "Defects and defect Clusters in BCC Metals and Alloys," in Nuclear Metallurgy Vol. 18.ed Arsenalult 1973 as reported in ORNL/LTR/JIMO/04-05 dated February 2004

(289) Lessmann, "Determination of Weldability and Elevated Temperature Stability of Refractory Metal Alloys," NASA CR-1607, August 1970.

(290) Horak, Booker, "Evaluated Creep Data for Selected Refractory Metal Alloys with Potential Multimegawatt Space Power Applications," ORNL/TM- 11163, August 1990.

(291) Sheffler, Sawyer & Steigerwald, "Creep Behavior of Refractory Alloys in Ultrahigh Vacuum, NASA SP-245 June 26, 1969 page 75.

(292) Sheffler, Sawyer & Steigerwald. "Mechanical Behavior of Tantalum-Base T-111 Alloy at Elevated Temperature, NASA CR-1436, 1969.

(293) Povirk, G.L. and L.L. Rishel, CME-7227-0062/B-MT(ME)-466, Molybdenum Fuel Element Modeling, April 15, 2004.

(294) Fuels for Space Nuclear Power and Propulsion: 1983-1993, R.B. Matthews, R.E. Barrs, T.C. Blair, D.B. Butt, R.E. Mason, W.A. Stark, E.K. Storms, and T.C. Wallace, Los Alamos National Laboratory, Nuclear Fuels Technology Group, Los Alamos, NM 87545.

(295) Materials Properties for Uranium Nitride (UN), GE Aerospace, SP-100 Programs, GE Aerospace document number 23A3182, 1988.

(296) Some Scoping Experiments for a Space Reactor, C.A. Alexander and J.S. Ogden, Task 128, Battelle, Columbus Laboratories, 1983.

(297) Irradiation Effects on Fuels for Space Reactors, W.A. Ranken and A.W. Cronenberg, Los Alamos National Laboratory, LA-UR-84-2651, 1984.

(298) Oggianu, S.M. and M.S. Kazimi, "A review of Properties of Advanced Nuclear Fuels," Feb 2000, MIT-NFC-TR-021.

(299) Ma, B.M. "Nuclear Reactor Materials and Applications," Van Nostrand Reinhold Company, 1983.

(300) Sinizer, D.I., et al. "Irradiation Behavior of Uranium Carbide Fuels," Atomics International report # NAA-SR-7248, Aug 1962.

(301) Butt, D.P. and K.M. Chidester, "Uranium Nitride Properties: A Survey of Current Literature (Draft)," 2004, LA-CP-04-0257.

(302) Bowles, K.J. and R.E. Gluyas, "Evaluation of Refractory-Metal-Clad Uranium Nitride and Uranium Dioxide Fuel Pins After Irradiation for Times Up to 01,450 Hours at 990°C," NASA-TN-D-7891, June 1975.

(303) Thomas, J.K., et al. "Empirical Modeling of Uranium Nitride Fuel" and "Evaluation of Multimegawatt Fuel Designs," to satisfy DOE contracts #(-X6D-1879G-1 and #9-X6D-1878G-1, May 1987 (may have document # DOE/NE/37960-TI).
(304) FRAPCON-2 manual, revision Feb 1980.
(305) H. Bailly, ed. The Nuclear Fuel of Pressurized Water Reactors and Fast Reactors design and behavior., CEA, 1999.
(306) Tong, L.S. and J. Weisman, Thermal Analysis of Pressurized Water Reactors, ANS, 1970, pg 61.
(307) Storms, E.K. "An Equation Which Describes Fission Gas Release From UN Reactor Fuel," Journal of Nuclear Materials, V158 (1988), pp 119-129.
(308) Ross, SB, et al. "Uranium Nitride Fuel Swelling Correlation ," Journal of Nuclear Materials, V.170 (1990), pp. 169-177.
(309) Burdi, GF, et al. "SNAP Technology Handbook Volume III, Refractory Fuels and Claddings," NAA-SR-8617, Vol III, April, 1965.
(310) EL-Genk, M.S., S.B. Ross and R.B. Matthews, "Uranium Nitride Fuel Swelling and Thermal Conductivity Correlations," Transactions, Fourth Symposium on Space Nuclear Power Systems, Albuquerque, NM, January 1987.
(311) Babcock & Wilcox, SPND Detector Response in the CX, ASE-58-3001813-00, November 1992.
(312) Babcock & Wilcox, CX Peek-A-Boo Reactivity Worth Around Center Stalk, ASE-58-3001803-00, November 1992.
(313) Babcock & Wilcox, Neutron Lifetime in the CX, ASE-58-3001802-00, November 1992.
(314) Babcock & Wilcox, Radial Fission Power Distribution in the Critical Experiment (CX) Fuel Stalks, ASE-58-3001806-00, June 1992.
(315) Gary Hoovler, CX Documentation Notes, Babcock & Wilcox, September 24, 1993.
(316) Pratt & Whitney, Middletown, CT, SNAP-50 Critical Experiments and Analysis, PWAC-487, September 1965.
(317) Mondt, Jack F., et al., SP-100 Technical Summary Report, Volume I- Executive Summary, JPL. JPL D-11818-Vol.1, September 1994.
(318) Mondt, Jack F., et al., SP-100 Technical Summary Report, Volume II – Technical Report, JPL. JPL D-11818-Vol. 2, September 1994.
(319) Murphy, T.L., et al., Final Disassembly and Examination of the ML-1 Reactor Core, AEC. IDO-17190, June 1966.
(320) Anderson, R.V., et al., Space Reactor Electric Systems: Subsystem Technology Assessment, Rockwell International, ESG-DOE-13398, March 1983.
(321) Batey, W. and Bagley, K.Q. "Fuel/clad reactions in irradiated oxide fuel pins." BNESJ Vol. 13, 1974, pp.49-61.
(322) Board. "A review of stainless steel properties for fast reactor fuel elements." BNESJ Vol. 11, 1972, pp. 237-257.
(323) Kurata, Y., et al., "Creep Properties of Base Metal and Welded Join of Hastelloy XR Produced for High-Temperature Engineering Test Reactor in Simulated Primary Coolant Helium." Journal of Nuclear Science and Technology, Vol. 36, No. 12, December 1999, pp. 1160-1166.
(324) Brussalis, W. "Stress Analysis; NRX-A Outer Reflector Assembly." Westinghouse Electric Corporation, WANL-TME-496, August, 1963.
(325) Bird, R. Byron, Warren E. Stewart, and Edwin N. Lightfoot, "Transport Phenomena", New York, NY. John Wiley & Sons, Inc., 1960.
(326) M. Dalle Donne, E. Meerwald, "Heat transfer and friction coefficients for turbulent flow of air in smooth annuli at high temperature", International Journal of Heat and Mass Transfer, Vol. 16 (1973), pp. 787-809
(327) Y. S. Tang, R. D. Coffield Jr., R. A. Markley, "Thermal Analysis of Liquid Metal Fast Breeder Reactors", American Nuclear Society, 1978, pp.131-132

(328) Blevins, Robert D., "Applied Fluid Dynamics Handbook", Van Nostrand Reinhold Company, Inc., 1984.
(329) M.F. Taylor, K.E. Bauer, and D. M. McEligot, "Internal forced convection to low-Prandtl number gas mixtures", International Journal of Heat and Mass Transfer, Vol. 31 (1988), pp. 13-25.
(330) D. M. McEligot, P. E. Murray, and G. W. Johnsen, "Convective heat transfer and pressure drop in low-Prandtl-number gas mixtures", INL/EXT-05-00779, September 26, 2005.
(331) G. Melese, R. Katz, "Thermal and Flow Design of Helium-Cooled Reactors", American Nuclear Society, 1984, pp.86-88.
(332) "Fundamentals of Heat and Mass Transfer (4th Edition)"; Frank P. Incropera and David P. DeWitt; Published by John Wiley & Sons, Inc. – 1996
(333) P. E. Pickett, "Heat and momentum transfer to internal turbulent flow of helium-argon mixtures in circular tubes, MSE Report, Aero. Mech Engng Dept., Univ. Arizona (1976). Available from NTIS as AD-A167290.
(334) Moran, Michael J., and Howard N. Shapiro. "Fundamentals of Engineering Thermodynamics (4th ed.)" New York, NY. John Wiley & Sons. 2000.
(335) Bejan, Adrian "Heat Transfer"; Published by John Wiley & Sons, Inc. – 1993, New York
(336) Jupiter Icy Moons Orbiter Environmental Requirements Document 982-00029,Rev.0, March 2, 2004, John Casani JIMO Project Manager, JIMO Project Library at http://jimo-lib.jpl.nasa.gov/
(337) OSTI ID 17078, SP-100 Ground Engineering System Project Annual Technical Progress Report for October 1990 – September 1991, dated February 12, 1992
(338) Ott, Karl O. and Winfred A Bezella. *Introductory Nuclear Reactor Statics*. American Nuclear Society, La Grange Park, IL, 1983. pp 164-182.
(339) "HTGR Graphite Design Handbook," General Atomics, San Diego, CA. September 1988.
(340) JPL letter 982-00056, Rev 1, "Prometheus Project, Project Requirements Document, Preliminary," dated December 15, 2004.
(341) OSTI ID 17456, SP-100 Power System Qualification Program. Annual Technical Progress Report for October 1991 – September 1992, dated February 1, 1993
(342) D. E. Holcomb, et al., "JIMO Reactor Sensor Technology Development Plan," Feburary 2005
(343) Prometheus Project Environmental Requirements Document 982-00029,Rev.0, July 6, 2005, Kin Fung Man Environments Manager, Prometheus Project, Prometheus Project Library at http://prometheus-project-lib.jpl.nasa.gov.
(344) A. M. Wahl, "Mechanical Springs (2nd Edition)," McGraw-Hill Book Company, Inc. 1963.
(345) W. C. Young, "Roark's Formulas for Stress & Strain (6th Edition)," McGraw-Hill, Inc. 1989.
(346) G. Schuster, "Stress Relaxation of Fuel-Pin Springs," Martin Marietta Company, San Jose Operations, November 2, 1992.
(347) R. C. Juvinall and K. M. Marshek, "Fundamentals of Machine Component Design (3rd Edition)"; John Wiley & Sons, Inc. 2000.
(348) Anghaie, Samim (2004), "Overview of Prismatic Fuel Development and Test Data." Paper presented at STAIF, Albuquerque, New Mexico February 8-11, 2004.
(349) GEMP-600 "710 High Temperature Gas Reactor Program Study Report, General Electric; Cincinnati, Ohio." US Atomic Energy Commission, Contract At(40-1)-2847. Dated 1967.
(350) T.R. Allen, J.T. Busby, G.S. Was and E.A. Kenik, "On the Mechanism of Radiation-Induced Segregation in Austenitic Fe-Cr-Ni Alloys", Journal of Nuclear Materials, vol. 255 (1998) pages 44-58.

(351) R. R. Smith, R. G. Matlock, F. D. McGinnis, M. Novick, and F. W. Thalgott, "An Analysis of the Stability of the EBR-I Marks I to III, and Conclusions Pertinent to the Design of Fast Reactors," SM-18/49. Proceedings of the Seminar on the Physics of Fast and Intermediate Reactors, Volume III, pp. 43-91, Vienna, August, 1961

(352) R. E. Rice, R. N. Curran, F. D. McGinnis, M. Novick, and F. W. Thalgott, "EBR-I, MARK-III Design Report," Argonne National Laboratory Repot ANL-5836, March, 1958

(353) F. W. Thalgott, J. F. Boland, R. O . Brittan, J. C. Carter, F. E. McGinnis, M. Novick, D. Okrent, H. A. Sandmeierer, R. R. Smith, and R. E. Rice, "Stability studies on EBR-I," P/1845, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Volume 12, pp. 242-266, Geneva, September, 1958

(354) R. D. Coffield, Jr., J. V. Miller, and R. A. Markley, "Phenomenological Models for Reactivity Feedback Calculations of LMR Inherent Safety Transient Type Test," Proceedings of the International Topical Meeting on Sagety of Next Generation Power Reactors, American Nuclear Society, pp 887-895, Seattle, May 1-5, 1988

(355) Gunduz, G. and Uslu, I. "Powder Characteristics and Microstructure of Uranium Dioxide and Uranium Dioxide-Gadolinium Oxide Fuel" Journal of Nuclear Materials. V231, 1996. pp.113-120.

(356) Fracture Mechanics, Fundamentals and Applications, 2nd Edition T.L. Anderson, p.627.

(357) *Nuclides and Isotopes: Chart of the Nuclides (16th Edition)*, Revised by E.M. Baum, H.D. Knox, and T.R. Miller, Published by KAPL, Inc., 2002.

### 4.4 Manufacturing

(358) IS-151500-0247, BWXT Information Submittal, "Manufacturing Assessment of Annular Flow Block Concept", August 19, 2005

(359) IS-151500-0250, BWXT Information Submittal, "Manufacturing Assessment of Various Core Concepts", September 09, 2005

(360) CP-054-233, BWXT Letter, ROM Cost Estimate for Space Reactor Assembly and Facility", August, 19, 2005

# 5 Reactor Module Primary Plant Segment

### 5.1 Prometheus Requirements

(361) NASA Document OExS-RQ-0-0003, "Baseline, Level 1 Jupiter Icy Moons Orbiter (JIMO) Requirements", dated May 18, 2004

(362) NASA Document SA-0001, "Baseline, Level 0 Exploration Requirements for the National Aeronautics and Space Administration", dated May 4, 2004

(363) JPL Document 982-00115, "Revision 2, Prometheus Project Multi-mission Project Derived Requirements", dated July 15, 2005

(364) JPL Document 982-00116, "Revision 2, Prometheus Project JIMO Project Derived Requirements", dated July 15, 2005

(365) JPL Document 982-00098, "Revision 0, Jupiter Icy Moons Orbiter Deep Space Vehicle Level 3 Key Driving Requirements", dated July 12, 2005

(366) JPL Document 982-00029, "Revision 0, Prometheus Project Environmental Requirements Document", dated July 6, 2005

## 5.2 Historical Space Reactors


(367) Mondt, Jack F., et al. "SP-100 TECHNICAL SUMMARY REPORT." Jet Propulsion Laboratory. JPL D-11818, Vol. I - Executive Summary, dated September 1994.

(368) Buksa, J., et al. "SP-100 TECHNICAL SUMMARY REPORT." Jet Propulsion Laboratory. JPL D-11818. Vol. II, Technical Report, September 1994.

(369) "SP-100 TECHNICAL SUMMARY REPORT" Jet Propulsion Lab and California Institute of Technology, (JPL D-11818). Vol. III, Bibliography, September 1994.

(370) Voss, Susan S. "SNAP REACTOR OVERVIEW." Air Force Weapons Laboratory, Kirtland Air Force Base. AFWL-TN-84-14 - Final Report, August, 1984.

(371) Eisenhawer, S.W. "Thermal Environment for the SP-100 Nuclear Reactor During Space Transportation System Launch-Phase Accidents." Los Alamos National Laboratory. LA-UR-85-3761-Revised.


## 5.3 Brayton Component Development


(372) AFRL-PR-WP-TR-2005-2018, "Integrated Power Unit (IPU) – Advanced Development Industry Version", December 2004.

(373) Allied Signal, Garrett Fluid Systems Division, "Space Power 1991 and Beyond" p.49 1991.

(374) AiResearch Letter APS-5334-R, "Final Report – The Design and Fabrication of the Brayton Rotating Unit (BRU)", March 1971.

(375) AiResearch Mfg. Company, Phoenix, Arizona. "Executive Summary: Mini-BRU/BIPS 1300 Watt-e Dynamic Power Conversion System Development." Contract #NAS3-18517, NASA-CR-159440, October 1978.

(376) AiResearch Mfg. Company, Phoenix, Arizona. "Final Report: Analysis, Design, Fabrication and Testing of the Mini-Brayton Rotating Unit (Mini-BRU) Volume 1 - Text and Tables." Contract #NAS3-18517, NASA-CR-159441, October 1978.

(377) AiResearch Mfg. Company, Phoenix, Arizona. "Final Report: Analysis, Design, Fabrication and Testing of the Mini-Brayton Rotating Unit (Mini-BRU) Volume 2 - Figures and Drawings." Contract #NAS3-18517, NASA-CR-159441, October 1978.

(378) Barrett, M., Performance and Mass Modeling Subtleties in Closed-Brayton-Cycle Space Power Systems, Proc. of 2005 AIAA-IECEC conf.

(379) Beremand, Donald G., et al. "Experimental Performance Characteristics of Three Identical Brayton Rotating Units." NASA Lewis Research Center. NASA TM X-52826, September 1970.

(380) Bowman, R.,et al. "Evaluation of Candidate Materials for a High Temperature Stirling Convertor Heater Head." NASA Glenn Research Center. NASA TM-2003-212734, December 2003.

(381) Brayton Reports, Studies & Contracts. (Comprehensive List provided by NASA Glenn Research Center).

(382) Deyo, J.N., et al. "Experimental Performance of a 2-15 Kilowatt Brayton Power System in the Space Power Facility Using Krypton." NASA TM X-52750, January 1970.

(383) DOT/FAA Letter – Technical Standard Order (TSO) C77b, "Gas Turbine Auxiliary Units," December 2000.

(384) Dunn, James H. "Post Test Inspection of Three Brayton Rotating Units." NASA Lewis Research Center. NASA TM X-67841, August 1971.

(385) Dunn, James. "Inspection of Two Brayton Rotating Units after Extensive Endurance Testing." NASA Lewis Research Center. NASA TM X-73569, December 1976.

(386) Edkin, R. A., et al. "Automated Endurance Testing of a Brayton Power Conversion System." NASA Lewis Research Center. NASA TM X-67830, August 1971.

(387) Gayda, J. and T.P. Gabb. "Two Dimensional Viscoelastic Stress Analysis of a Prototypical JIMO Turbine Wheel." NASA/TM-2005-213650, June 2005.

(388) Halsey, DG, Dolwing, RS, Nguyen, DC, Barrett, MJ, "Closed Brayton Cycle Engine Starter/Generator Cooling, AIAA 2005, IECEC.

(389) Hamilton Sundstrand Letter HSPS2322190600, "Auxiliary and Ground Power Systems Reference Guide."

(390) Kattus, J.R., "Ni-Base Alloys, MAR-M-247", in Aerospace Structural Metals Handbook, Purdue Research Foundation, West Lafayette, Indiana, 1999, Code 4218, pp 1-6.

(391) Kerwin, Paul T. "Analysis of a 35- to 150- Kilowatt Brayton Power-Conversion Module for use with an Advanced Nuclear Reactor." NASA Lewis Research Center. NASA TN D-6525, September 1971.

(392) Klann, John L. "2 to 10 Kilowatt Solar or Radioisotope Brayton Power System." NASA Lewis Research Center. NASA TM X-52438, August 1968.

(393) Klann, John L., et al. "Performance of the Electrically-Heated 2 to 15 KWe Brayton Power System." NASA Lewis Research Center. NASA TM X-52824, September 1970.

(394) Kofskey, Milton G. and William J. Nusbaum. "Effects of Specific Speed in Experimental Performance of a Radial-Inflow Turbine." NASA Lewis Research Center. NASA TN D-6605, February 1972.

(395) "Final Report – Analysis, Design, Fabrication, and Testing of the Mini-Brayton Rotating Unit (MINI-BRU)." NASA CR-159441, October 1978.

(396) NGST Report 04.1F102.JM.006, "Task 2 Conceptual Design Summary Report, Jupiter Icy Moons Orbiter (JIMO) Phase A Study, JPL Contract N. 120530." August 2004.

(397) Mason, Lee and Barrett, Michael. "Preliminary Comments on Potential Brayton Turboalternator Failure Modes." February, 2005.

(398) Nusbaum, William J. and Charles A. Wasserbauer. "Experimental Performance of a 4.59-Inch Radial-Inflow Turbine over a range of Reynolds Number." NASA Lewis Research Center. NASA TN D-3835, February 1967.

(399) Postlethwait, M., et al. "Comparison of Direct and Indirect Gas Reactor Brayton Systems for Nuclear Electric Space Propulsion." Proceedings of the Space Nuclear Conference 2005. Paper #1177, June 2005.

(400) Rohlik, H. E., "Analytical determination of Radial-Inflow Turbine Design Geometry for Maximum Efficiency." NASA TN D-4384, 1968.

(401) Valerino, Alfred S., et al. "Preliminary Performance of a Brayton-Cycle-Power-System Gas Loop Operating with Krypton over a Turbine Inlet Temperature Range of 1200°F to 1600°F." NASA Lewis Research Center. NASA TM X-52769, March 1970.

(402) Weigel, Carl, et al. "Overall Performance in Argon of 4.25-Inch Sweptback-Bladed Centrifugal Compressor." NASA Lewis Research Center. NASA TM X-2129, November 1970.

(403) Wright, S.A., Fuller, R., Lipinski, R.J., Nichols, K., Brown, N., "Operational Results of a Closed Brayton Cycle Test Loop," Space Technology and Applications International Forum – STAIF 2005, CP746, February 2005

(404) Wong, Robert Y., et al. "Turbine and Compressor Performance of a Brayton Rotating Unit During Hot Closed-Loop Operation." NASA Lewis Research Center. NASA TM X-2350, September 1971.

(405) Wood, H. J., "Current Technology of Radial-Inflow Turbines for Compressible Fluids", J. of Eng. For Power. Trans. Am. Soc. Mech. Engrs., 85 (1963).

(406) "CCEP Match Attempt of Lee's JIMO TB 2.5 Brayton Design Point," Informal NASA memo dated 10/27/2004.

(407) "CCEP Turbomachinery Off-Design Calculations", Informal NASA memo.

## 5.4 Alternator Development

(408) Blackmore, E.W. "Radiation Effects of Protons on Samarium-Cobalt Permanent magnets", IEEE Tran. Nucl. Sci., NS-32,3669 (1985).

(409) Brown, R.D., et al. "Radiation Effects on Samarium-Cobalt Permanent Magnets", Los Alamos National Laboratory Report, LA-9437-MS (1982).

(410) Coninckx, F., et al. "Radiation effects on Rare-Earth Cobalt Permanent Magnets" Pre'vessin, February 1983, European Organization for Nuclear Research.

(411) Job PK, et al. "Radiation induced demagnetization of Nd-Fe-B Permanent magnets, LS-290, November 2000, DOE Sponsored.

(412) Mason, LM. "Experimental data for two different alternator configurations in a Solar Brayton Power System." NASA Lewis Research Center. NASA-TM 107509, IEC-97481, 1997.

(413) Stone GC, et al. Electrical Insulation for Rotating Machines, 2004 IEEE.

## 5.5 Gas Foil Bearing Development

(414) Agrawal, Giri L., "Foil Air/Gas Bearing Technology ~ An Overview", ASME Publication 97-GT-347, dated June 1997.

(415) Bauman, Steve. "An Oil-Free Thrust Foil Bearing Facility Design, Calibration, and Operation." NASA Glenn Research Center. NASA/TM-2005-213568, March 2005.

(416) Chen, H.M., et al. "Application of Foil Bearings to Helium Turbo-compressor", Proceedings of 30th Annual Turbo-machinery Conference, pp. 103-113.

(417) DellaCorte, Christopher. "Future Issues and Approaches to Health Monitoring and Failure Prevention for Oil-Free Gas Turbines." NASA Glenn Research Center. NASA/TM-2004-213058, April 2004.

(418) DellaCorte, Christopher, et al. "Performance and Durability of High Temperature Foil Air Bearings for Oil-Free Turbomachinery." NASA/TM-2000-209187/REV1, ARL-TR-2202, March 2000.

(419) DellaCorte, C., et al. "Evaluation of Advanced Solid Lubricant Coatings for Foil Air Bearings Operating at 25 and 500 °C," NASA/TM-1998-206619, 1998.

(420) Dellacorte, Christopher, et al. "Advanced Rotor Support Technologies for Closed Brayton Cycle Turbines." 3rd International Energy Conversion Conference. AIAA 2005-5513, August 2005.

(421) Dellacorte, Christopher and Mark J. Valco. "Load Capacity Estimation of Foil Air Journal Bearings for Oil-Free Turbomachinery Applications." NASA Glenn Research Center. NASA/TM-2000-209782, ARL-TR-2334, October 2000.

## 5.6 Recuperator Development

(422) AiResearch Manufacturing Company of Arizona, "Executive Summary – Mini-BRU/BIPS 1300 $W_e$ Dynamic Power Conversion System Development", NASA Contractor Report 159440, NASA Glenn Reseach Center, Cleveland, OH, October 1978.

(423) Barrett, M. J. and Reid, B. M. "System mass variation and entropy generation in 100-kWe closed-Brayton-cycle space power systems", Proceedings of Space Technology and Applications International Forum (STAIF – 2004), pp. 445 – 452, 2004.

(424) Barrett, Michael J. "Performance Expectations of Closed-Brayton-Cycle Heat Exchangers in 100-kWe Nuclear Space Power Systems." NASA Glenn Research Center. AIAA 2003-5956, August 2003.

(425) Dunn, J. H. "Inspection of Two Brayton Rotating Units After Extensive Endurance Testing", NASA Technical Memorandum X-73569, NASA Glenn Research Center, December 1976.

(426) Garrett Fluid Systems, Space Power 1991 and Beyond, 1991.

(427) Graham, L.W. Journal of Nuclear Materials, vol. 171, pp. 76-83, 1990.

(428) Hamilton Sundstrand, Windsor Locks, CT. "Evaluation of Gas Cooler Materials of Construction and Reliability." Bechtel Bettis Purchase Order #3007352, July 13, 2005.

(429) Johnson, P. K. and Mason, L. S. "Design and Off-Design Performance of 100 kWe-Class Brayton Power Conversion Systems." Proceedings of Space Technology and Applications International Forum (STAIF – 2005), pp. 711 – 718, 2005.

(430) Kays, W. M. and London, A. L. Compact Heat Exchangers Third Edition. New York, McGraw-Hill, 1984.

(431) Killackey, J. J., et al. "Brayton-Cycle Heat Exchanger Technology Program." Prepared by AiResearch Manufacturing Company of California, NASA Contractor Report 135158, NASA Lewis Research Center, August 1976.

(432) Killackey, J. J., et al. "Design and Fabrication of the Mini-Brayton Recuperator Final Report." Prepared by AiResearch Manufacturing Company of California, NASA Contractor Report 159429, NASA Lewis Research Center, April 1978.

(433) Manglik, R. M. and Bergles, A. E. "Heat Transfer and Pressure Drop Correlations for the Rectangular Offset Strip Fin Compact Heat Exchanger." Experimental Thermal and Fluid Science, pp. 171 – 180.

(434) Massalski, T.B., et al. Binary Alloy Phase Diagrams, 2nd ed., ASM International, 1990.

(435) McEligot, D. M., et al. "Convective heat transfer and pressure drop in low-Prandtl-number gas mixtures" Idaho National Laboratory. DOE Office of Naval Reactors Contract DE-AC07-05ID14517, September 26, 2005.

(436) NGST Report 04.1F102.JM.006, "Task 2 Conceptual Design Summary Report, Jupiter Icy Moons Orbiter (JIMO) Phase A Study, JPL Contract N. 120530" dated 8/13/2004.

(437) Shah, R. K. and Sekulic, D. P., Fundamentals of Heat Exchanger Design. New York, John Wiley, 2003.

(438) Smithells, C.J. Metals Reference Book, 4th ed., Plenum Publishing Corporation, 1967.

(439) Staudt, J. E. "Design Study of an MGR Direct Brayton-Cycle Power Plant." PHD Thesis, Massachussetts Institute of Technology, May 1987.

(440) Taylor, M. F., et al. "Internal forced convection to low-Prandtl-number gas mixtures." International Journal of Heat and Mass Transfer. Vol. 31, pp. 13 – 25, 1988.

(441) Von Arx, A. V. and I. Ceyhan. "Laminar Heat Transfer for Low Prandtl Number Gases." 8th Symposium Space Nuclear Power Systems, Albuquerque, AIP Conference Proceeding, pp. 719 – 722, 1991.

(442) Warren, M.R. High Temperature Technology. Vol. 4, pp. 119-130, 1986.

## 5.7 Plant Systems

(443) Kerwin, Paul. "Analysis of a 35- to 150-Kilowatt Brayton Power-Conversion Module for Use with an Advanced Nuclear Reactor." NASA Lewis Research Center. NASA TN D-6525, September 1971.

(444) NASA-JPL Letter 05-113, "Prometheus Level 2 Requirements Applicable to the SNPP Sizing", dated May 31, 2005.

(445) JPL Document 982-00101, Draft, Rev. 0, "Project Engineering Plan", dated July 15, 2005.

(446) NGST document SDRL SE-002-001 Draft Rev -, "NGST Prometheus Spacecraft Module and Subcontractor-provided Reactor Module Segment Design Description", dated May 16, 2005.

(447) JPL Document 982-00007, "JIMO Shield Trade Study: Reactor and Natural Space Radiation," E. Blakeman (ORNL), J. Johnson (ORNL), I. Jun (JPL, Lead), W. McAlpine (JPL), I. Remec (ORNL), J. Yugo (ORNL), April 2004.

(448) Mason, Lee. "A Power Conversion Concept for the Jupiter Icy Moons Orbiter." NASA Glenn Research Center. AIAA-2003-6007, August 2003.

(449) Wong, Robert Y. "Effect of Operating Parameters on Net Power Output of a 2- to 10-Kilowatt Brayton Rotating Unit." NASA Lewis Research Center. NASA TN D-5815, May 1970.

## 5.8 Alternative Energy Conversion Systems - Stirling

(450) Backhaus, Scott and Robert S. Reid. "A Self-Circulating Heat Exchanger for Use In Stirling and Thermoacoustic-Stirling Engines." LA-UR #04-5506.

(451) Bowman, Randy, et al. "Evaluation of Candidate Materials for a High-Temperature Stirling Convertor Heater Head." NASA Glenn Research Center. NASA/TM-2003-212734, December 2003.

(452) Brickhaus, S. and GW Swift. "A Thermoacoustic-Stirling Heat Engine: Detailed Study." 2000 Acoustical Society of America. February, 2000.

(453) Dochat "SPDE/SPRE Final Summary Report", NASA Contractor Report 187086, 1993– Prepared for Lewis Research Center.

(454) Dochat, George R., et al. "Free-Piston Stirling Engine System Considerations for Various Space Power Applications." CONF-910116, 1991 American Institute of Physics.

(455) Dochat, George R. and James E. Dudenhoefer. "Performance Results of the Stirling Power Converter." CONF 940101, 1994 American Institute of Physics.

(456) Furlong, Richard and Richard Shaltens. "Technology Assessment of DOE's 55-We Stirling Technology Demonstrator Convertor (TDC)." AIAA-2000-3018.

(457) General Electric. "SP100, Stirling Engine System Studies." JPL Contract 956473, February 1985.

(458) Kristalinski, Alex, et al. "Radioisotope Powered Missions to Martian Moons Using Electric Propulsion."

(459) Lewandowski, Edward and Timothy Regan. "Overview of the GRC Stirling Convertor System Dynamic Model." 2nd International Energy Conversion Engineering Conference. AIAA 2004-5671, August 2004.

(460) Mechanical Technology Inc., Latham, NY. "Stirling Space Engine Program, Final Report." NASA/CR-1999-209164, August 1999.

(461) Mechanical Technology Incorporated, Latham, New York. "Space Power Free-Piston Stirling Engine Scaling Study." Contract NAS3-25148, NASA CR-182218, MTI 89TR6, October 1989.

(462) Mechanical Technology Inc., Latham, New York. "Stirling Space Engine Program, Vol 1 - Final Report." NASA/CR-1999-209164/VOL1, August 1999.

(463) Mechanical Technology Inc., Latham, New York. "Stirling Space Engine Program, Vol 2 - Appendixes A,B,C and D." NASA/CR-1999-209164/VOL2 August 1999.

(464) Mechanical Technology Inc., Latham, New York. "SPDE/SPRE Final Summary Report." Contract NAS3-23883, NASA CR 187086, September 1993.

(465) Qiu, Songgang, et al. "Developing a Free-Piston Stirling Convertor for Advanced Radioisotope Space Power Systems." Space Technology and Applications International Forum - STAIF 2002.

(466) Regan, Timothy. "Free-Piston Stirling Convertor Controller Development at NASA Glenn Research Center." NASA/CR-2004-213038.

(467) Sargent, Noel B. "The Electromagnetic Compatibility (EMC) Design Challenge for Scientific Spacecraft Powered by a Stirling Power Convertor." Proceedings of IECEC '01: 36th Intersociety Energy Conversion Engineering Conference. IECEC2001-CT-37, August 2001.

(468) Schreiber, J. and L Thieme, "Accomplishments of the NASA GRC Stirling Technology Development Project", AIAA 2004 conference.

(469) Schreiber, Jeffrey and Lanny Thieme. "Accomplishments of the NASA CRC Stirling Technology Development Project." 2nd International Energy Conversion Engineering Conference. AIAA 2004-5517, August 2004.

(470) Schreiber, Jeffrey and Robert Skupinski. "Accomplishments in Free-Piston Stirling Tests at NASA GRC." Space Technology and Applications International Forum - STAIF 2002.

(471) Schreiber, Jeffrey G. and Lanny G. Thieme. "Update on the Stirling Convertor Testing and Technology Development at NASA GRC." Proceedings of IECEC '01: 36th Intersociety Energy Conversion Engineering Conference. IECEC2001-CT-26, August 2001.

(472) Schreiber, Jeffrey G. "Assessment of the Free-Piston Stirling Convertor as a Long Life Power Convertor for Space." AIAA-2000-3021.

(473) Schmitz, Paul, et al. "Preliminary SP - 100/Stirling Heat Exchanger Designs." NASA TM-106444, January, 1994.

(474) Schmitz, Paul, et al. "Preliminary SP-100/Stirling Heat Exchanger Designs." CONF 940101 1994 American Institute of Physics.

(475) SP100 Stirling Engine System Studies, Final Briefing. JPL Contract 956473, February 1985.

(476) Thieme, Lanny G., et al. "Technology Development for a Stirling Radioisotope Power System for Deep Space Missions." 1999-01-2454.

(477) Thieme, Lanny G., et al. "Stirling Technology Development at NASA GRC." Space Technology and Applications International Forum - STAIF 2002.

(478) White, Maurice A., et al. "Status of an Advanced Radioisotope Space Power System Using Free-Piston Stirling Technology." 33rd Intersociety Energy Conversion Engineering Conference. IECEC-98-417, August 1998.

## 5.9 Alternative Energy Conversion Systems - Liquid Metal

(479) Atomics International Handbook NAA-SR-8617. "SNAP Technology Handbook Volume I Liquid Metals." August 1964.

(480) Bazinet, G.D., et al. "Corrosion Behavior of Materials Selected for FMIT Lithium System." Nuclear Technology/Fusion. Vol. 4, September 1983.

(481) Burrow, G.C., et al. "Corrosion Inhibition Experiments in Liquid Lithium." Journal of Nuclear Materials. Vol. 103 and 104, 1981. pp. 657-662.

(482) Chopra, O.K., and D.L. Smith. "Effects of Lithium Environment on the Fatigue Properties of Ferritic and Austenitic Steels." Journal of Nuclear Materials. Vol. 103 and 104, 1981. pp 651-656.

(483) Chopra, O.K., and D.L. Smith. "Corrosion of Ferrous Alloys in a Flowing Lithium Environment. " Journal of Nuclear Materials. Vol. 133 and 134, 1985. pp 861-866.

(484) Chopra, O.K., and D.L. Smith. "Influence of a Flowing Lithium Environment on the Fatigue and Tensile Properties of Type 316 Stainless Steel." Journal of Nuclear Materials. Vol. 122 and 123, 1984. pp 1213-1218.

(485) Elliot, David G. " Two-Fluid Magnetohydrodynamic Cycle for Nuclear-Electric Power Conversion, ARS Journal June 1962, pp 924-928.

(486) Elliot, David G. "Performance Capabilities of Liquid Metal MHD Induction Generators" from IAEA conference on "Electricity from MHD – 1968."

(487) Fabris, Gracio. "Review of Two-phase Flow Liquid Metal MHD and Turbine Energy Conversion Concepts for Space Applications." Space Nuclear Power Systems, 1990.

(488) Hammon, D.L., et al. "The Influence of Cyclic Loading on the Lithium Corrosion Behavior of Reactor Materials." Journal of Nuclear Materials. Vol. 103 and 104, 1981. pp. 663-668.

(489) Katsuta, H., et al. "Hydrogen Release through Niobium Contacting with Liquid Lithium." Journal of Nuclear Materials. Vol. 61, 1976. pp 324-325.

(490) Liquid Metals Handbook (by the Committee on Basic Properties of Liquid Metals, Office On Naval Research, Department of the Navy), second edition, January 1954.

(491) Morse, Frederick H. " Survey of Liquid Metal Magnetohydrodynamic Energy Conversion Cycles." Energy Conversion. Vol. 10, 1970. pp 155-176.

(492) Petrick, Michael, et al. "Analytical And Experimental Studies of Single and Two-Phase Liquid Metal Faraday Generators." Argonne National Lab Report. AFAPL-TR-68-93, August 1968.

(493) Petrick, Michael and Kung-You Lee. "Performance Characteristics of a Liquid Metal MHD Generator." Argonne National Laboratory Report. ANL-6870, July 1964.

(494) Tortorelli, P.F., and O.K. Chopra. "Corrosion and Compatibility Considerations of Liquid metals for Fusion Reactor Applications." Journal of Nuclear Materials, Volumes 103 and 104, 1981. pp. 621-632.

(495) Tortorelli, P.F. "Dissolution Kinetics of Steels Exposed in Lead-Lithium and Lithium Environments." Journal of Nuclear Materials. Vol. 191-194, 1992. pp. 965-969.

## 5.10 Alternative Energy Conversion Systems - Thermoelectric

(496) Noon, E.L., et al. "Restoration of a Computer Model for Predicting Long Term Thermoelectric Generator Performance." JPL Report # D-29697, August 2004.

(497) Nolas, George, et al. "Semiconductor Clathrates: A Phonon Glass Electron Crystal Material with Potential for Thermoelectric Applications."

(498) Uher, Citrad. "Skutterudites: Prospective Novel Thermoelectrics," Recent Trends in Thermoelectric Materials Research, Volume I, AP of Semiconductors and Semimetals. Vol. 69, 2001.

(499) MacDonald, D.K. "Thermoelectricity: an introduction to the principles," 1962.

(500) Heikes R.R., and R. W. Ure Jr. "Thermoelectricity," pp. 5-6, by International Publishers, NY, 1961.

## 5.11 China HTR / Japan HTTR/HTGR Development Efforts

(501) Lohnert, G. "The Chinese High Temperature Reactor HTR-10, The First Inherently Safe Generation IV Nuclear Power System." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 218, October 2002.

(502) Hada, K., et al. "Developments of Metallic Materials and a High-Temperature Structural Design Code for the HTTR." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(503) Hada, K., et al. "Application of New Design Methodologies to Very High-Temperature Metallic components of the HTTR. Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(504) Huang, Z.Y., et al. "Design and Experiment of Hot Gas Duct for the HTR-10." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 218, 2002.

(505) Iyoku, T., et al. "Graphite Core Structures and their Structural Design Criteria in the HTTR." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(506) "Materials Behavior in HTGR Environments." U.S. Nuclear Regulatory Commission. Argonne National Laboratory. NUREG/CR-6824, ANL-02/37.

(507) Neumann, G., et al. "The HTR Module Pressure Vessel Unit, Design Criteria and Safety Philosophy." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(508) Ogawa, T., et al. "Release of Short-Lived Noble Gases from HTGR Fuel with Failed Coated Fuel Particles and Contaminated Matrix." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(509) Okubo, M., et al. "Structural Integrity Evaluation of a Helically-Coiled He/He Intermediate Heat Exchanger."

(510) "Present Status of HTGR Research & Development." Japan Atomic Energy Research Institute. March 1990.

(511) Saito, Shinzo, et al. "Present Status of the High Temperature Engineering Test Reactor (HTTR)." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(512) Schubert, F., et al. "Structural Design Criteria for HTR - A Summary Report." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(513) Shindo, M., et al. "Safety Characteristics of the High Temperature Engineering Test Reactor." Nuclear Engineering and Design. An International Journal devoted to the Thermal, Mechanical, Materials, and Structural Aspects of Nuclear Fission Energy. Volume 132, December 1991.

(514) Wang, C., et al. "Design of a Power Conversion System for an Indirect Cycle, Helium Cooled Pebble Bed Reactor System." NERI Project DE-FG03-00SF22171.

(515) Yan, X., et al. "Design and Development of GTHTR300." HTR2002, the 1st International Topical Meeting on HTR Technology. April 2002.
(516) "5.2.5 High Temperature Structural Design Guideline." Design of High Temperature Engineering Test Reactor (HTTR) JAER1 1332.
(517) Hada, K. "A Proposal to Develop a High Temperature Structural Design Guideline for HTGR Components."

## 5.12 Sublimation

(518) Whittenberger, J.D. "Effect of Long-Term 1093 K Exposure to Air or Vacuum on the Structure of Several Wrought Superalloys." Journal of Materials Engineering and Performance. Volume 2(5), October 1993, pp 745-757.
(519) Whittenberger, J.D. "Tensile Properties and Structure of Several Superalloys after Long-Term Exposure to LiF and Vacuum at 1173 K." Journal of Materials Engineering and Performance. Volume 4(6), December 1995, pp 657-673.

## 5.13 Space Structural Design Basis (SSDB)

(520) Robinson, E.L., "Effect of Temperature Variations on the Long Time Strength of Steels," Transactions of the ASME, Vol. 74, 1952, pp 777-780
(521) Bree, J., "Elastic-Plastic Behavior of Thin Tubes Subjected to Internal Pressure and Intermittent High-Heat Fluxes with Applications to Fast Nuclear-Reactor Fuel Elements," *Journal of Strain Analysis*, Vol. 2, No. 3, 1967
(522) O'Donnell, W.J., and Porowski, J.S., "Upper Bounds for Accumulated Strains Due to Creep Ratcheting," Trans. ASME, Journal of Pressure Vessel Technology, Vo. 96, 1974, p. 150-154
(523) Mankins, W.L., Hosier, J.C., and Bassford, T.H., "Microstructure and Phase Stability of INCONEL Alloy 617," Metallurgical Transactions, Vol. 5, 1974, pp. 2579-2590
(524) Jakub, M.T., "New Rules for Construction of Section III, Class 1 Components for Elevated Temperature Service," Journal of Pressure Vessel Technology, Vol. 98, Series J, No. 3, August 1976, pp. 214-222.
(525) Jetter, R.I., "Elevated Temperature Design – Development and Implementation of Code Case 1592," Journal of Pressure Vessel Technology, Vol. 98, Series J, No. 3, August 1976, pp. 222-229.
(526) Berman, I, and Gupta, G.D., "Buckling Rules for Nuclear Components," Journal of Pressure Vessel Technology, Vol. 98, Series J, No. 3, August 1976, pp. 229-231.
(527) Barnaby, J., Barton, P.J., Boothby, R.M., Fraser, A.S., and Slattery, S.F., "The Post-Irradiation Mechanical Properties of AISI Type 316 Steel and Nimonic PE16 Alloy", International Conference: Radiation Effects in Breeder Reactor Structural Materials, The Metallurgical Society of AIME, 1977, pp. 159-175
(528) Griffin, D.S., "Design Limits for Creep Buckling of Structural Components," Creep in Structures, 3rd IUTAM Symposium, Leicester, UK, September 1980, pp. 331-348.
(529) Kirchhofer, H., Schubert, F., and Nickel, H., "Precipitation Behavior of Ni-Cr-22Fe-18Mo (Hastelloy-X) and Ni-Cr-22Co-12Mo (INCONEL-617) after isothermal aging," Nuclear Technology, Vol. 66, 1984, pp. 139-148
(530) Cook, R.H., "Creep Properties of INCONEL-617 in Air and Helium at 800 to 1000C," Nuclear Technology, Vol. 66, 1984, pp. 283-288
(531) Ennis, P.J., Mohr, K.P., and Schuste, H., "Effect of Caburizing Service Environments on the Mechanical Properties of High Temperature Alloys," Nuclear Technology, Vol. 66, 1984, p. 363
(532) Lindgren, J.R., "Irradiation Effects on High-Temperature Gas-Cooled Reactor Structural Materials", Nuclear Technology, Vol. 66, 1984, pp. 607-618

(533) Robinson, D.N., “Constitutive Relationships for Anisotropic High Temperature Alloys,” Nuclear Engineering and Design, Vol. 83, 1984, pp. 389-396
(534) Kimball, O.F., “Thermal Stability and Environmental Compatibility of INCONEL 617,” High Temperature Metallic Materials for Gas-Cooled Reactors,” Proceedings of a specialists meeting held in Cracow, June 20-23, 1988. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-18, 1988, pp. 66-72
(535) Breitling, H., Dietz, W., and Penkalla, H.J., “Evaluation of Mechanical Properties of the Alloy NiCr22Co12Mo (Alloy 617) for Heat Exchanging Components of HTGR,” Proceedings of a specialists meeting held in Cracow, June 20-23, 1988. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-18, 1988
(536) Becht IV, C., “Behavior of Pressure-Induced Discontinuity Stresses at Elevated Temperatures,” Trans. ASME Journal of Pressure Vessel Technology, Vol. 111, August 1989, pp. 322-325.
(537) Sartory, W.K., "Effect of peak thermal strain on simplified ratchetting analysis procedures," PVP – Vol. 163, The American Society of Mechanical Engineers, 1989, pp. 31-38
(538) Corum, J.M., "Evaluation of weldment creep and fatigue strength-reduction factors for elevated temperature design," PVP Vol. 163, The American Society of Mechanical Engineers, 1989, pp. 9-17
(539) Huchtemann, B., “The Effect of Alloy Chemistry on Creep Behavior in a Helium Environment with Low Oxygen Partial Pressure, “Materials Science and Engineering, Vol. A121, 1989, p. 623
(540) Dyson, B., "Use of CDM (Creep Damage Mechanics) in materials modeling and component creep life prediction," Journal of Pressure Vessel Technology, Vol. 122, August 2000, pp. 281-295
(541) Corum, J.M., and Blass, J.J., “Rules for Design of Alloy 617 Nuclear Components to Very High Temperatures,” PVP-Vol. 215, The American Society of Mechanical Engineers, 1991, pp. 147-153
(542) Severud, L.K., "Creep-fatigue assessment methods using elastic analysis results and adjustments," Journal of Pressure Vessel Technology, Vol. 113, February 1991, pp. 34-40
(543) Marriott, D.L., “Current trends in high temperature design,” International Journal of Pressure Vessel Piping, Vol. 50, 1992, pp 13-35
(544) Hada, K., and Baba, O., “Structural Design Code for Very High Temperature Cooled Nuclear Reactor Cooling Components,” PVP-Vol. 262, The American Society of Mechanical Engineers, 1993, pp. 1-8
(545) Schubert, F., Breitbach, G., and Nickel, H., “German Structural Design Rule KTA 3221 for Metallic HTR-Components,” PVP-Vol. 262, The American Society of Mechanical Engineers, 1993, pp. 9-18
(546) Ohno, N., "Recent progress in constitutive modeling for ratchetting," Material Science Research International, Vol. 3, No. 1, 1997, pp. 1-9
(547) Jetter, R.I., "An alternate approach to evaluation of creep-fatigue damage for high temperature structural design criteria," Book No. H01146-1998, The American Society of Mechanical Engineers, 1998, pp. 199-205
(548) Takahashi, Y., “Advancement of High Temperature Structural Design Method for Fast Reactor Components, Part I: Creep-Fatigue Damage Evaluation Method for 316FR,” PVP Vol. 365, The American Society of Mechanical Engineers, 1998, pp. 159-166
(549) Sawai, T., Shimakawa, T., Nakayama, Y., and Takahashi, Y., "Creep-fatigue tests and analyses for Y-piece models under thermal stress," PVP Vol. 365, The American Society of Mechanical Engineers, 1998, pp. 249-256

(550) Prager, M., "The omega method - an engineering approach of life assessment," Journal of Pressure Vessel Technology, Vol. 122, August 2000, pp. 273-280

(551) Jetter, R.I., “Subsection NH – Class 1 Components in Elevated Temperature Service,” in *Companion Guide to the ASME Boiler & Pressure Vessel Code, Volume 1*, ASME Press, New York, 2002, pp. 369-404

(552) Miura, N., Nakayama, Y., and Takahashi, Y., "Development of flaw evaluation guideline for FBR components," Nuclear Engineering and Design, Vol. 212, 2002, pp. 13-19

(553) Carter, P., Marriott, D.L., and Swindeman, M.J., "Cyclic analysis for high temperature design," PVP Vol. 472, 2004, pp. 61-68

(554) Ainsworth, R.A., “Use of Advanced Creep Data for Assessment of Plant Life and Safety,” Materials at High Temperatures, Vol. 21, No. 1, 2004, pp. 11-15

(555) Swindeman, R.W., Swindeman, M.J., and Wen, W., “A Brief Review of Models Representing Creep of Alloy 617”, ASME Proceedings of PVP2000, PVP2005-71784, The American Society of Mechanical Engineers, 2005, pp. 1-8

(556) Carter, P., and Marriott, D.L., “Developments in Cyclic Analysis and High Temperature Design“, ASME Proceedings of PVP2005, PVP2005-71510, The American Society of Mechanical Engineers, 2005, pp. 1-7

(557) ASME Boiler and Pressure Vessel Code. Section III, Division 1 Subsection NH, Class 1 Components in Elevated Temperature Service, The American Society of Mechanical Engineers, New York, N.Y., 2004

(558) ASME Boiler and Pressure Vessel Code. Section III, Division 1 Subsection NB, Class 1 Components, The American Society of Mechanical Engineers, New York, N.Y., 2004

(559) ASME Boiler and Pressure Vessel Code, Section II, Part D – Properties (Metric), American Society of Mechanical Engineers, New York, N.Y., 2004

(560) ASME Boiler and Pressure Vessel Code, Code Cases – Nuclear Service, Code Case N-201-4, Class CS Components in elevated Temperature Service; The American Society of Mechanical Engineers.

(561) ASME Boiler and Pressure Vessel Code Section III, Division 1, Criteria for Design of Elevated Temperature Class 1 Components; The American Society of Mechanical Engineers.

(562) ASME Boiler and Pressure Vessel Code Section III, Division 1, Criteria of the ASME Boiler and Pressure Vessel Code for Design by Analysis; The American Society of Mechanical Engineers.

(563) Nuclear Standard, Requirements for Design of Class 1 Elevated Temperature Nuclear System Components (Supplement to ASME Code Case N-47), NE F 9-4T, Department of Energy, Nuclear Energy Program, June 1986

(564) Nuclear Standard, Guidelines and Procedures for Design of Class 1 Elevated Temperature Nuclear System Components, NE F 9-5T, Department of Energy, Nuclear Energy Program, September 1986

(565) Structural Design Criteria Document (SVS-11510): Structural Design Criteria for the SP100 Space Power Reactor Power System, General Electric Company, Astro-Space Division, Philadelphia, May 1988

(566) Structural Design Criteria Document (SVS-11510): Design Criteria for SP100 Reactor Nb-1Zr Structures, General Electric Company, Astro-Space Division, Philadelphia, September 1987

(567) RCC-MR, Design and Construction Rules for Mechanical Components of FBR Nuclear Islands, Second Edition, AFCEN, France, 2002

(568) R5 Procedure, Issue 3, Assessment Procedure for the High Temperature Response of Structures, British Energy, U.K., 2003

(569) Ryder, R.H., and Dahms, C.F., “Design Criteria for Dissimilar Metal Welds,” WRC Bulletin 350, Welding Research Council, January 1990

(570) Dhalla, A.K., "Recommended Practices in Elevated Temperature Design: A Compendium of Breeder Reactor Experiences (1970-1987), Volume I – Current State and Future Directions," WRC Bulletin 362, Welding Research Council, April 1991
(571) Dhalla, A.K., "Recommended Practices in Elevated Temperature Design: A Compendium of Breeder Reactor Experiences (1970-1987), Volume II – Preliminary Design and Simplified Methods," WRC Bulletin 363, Welding Research Council, May 1991
(572) Dhalla, A.K., "Recommended Practices in Elevated Temperature Design: A Compendium of Breeder Reactor Experiences (1970-1987), Volume III – Inelastic Analysis," WRC Bulletin 365, Welding Research Council, July 1991
(573) Dhalla, A.K., "Recommended Practices in Elevated Temperature Design: A Compendium of Breeder Reactor Experiences (1970-1987), Volume IV–Special Topics," WRC Bulletin 366, Welding Research Council, August 1991
(574) Yukawa, S., "Review and Evaluation of the Toughness of Austenitic Steels and Nickel Alloys After Long-Term Elevated Temperature Exposures," WRC Bulletin 378, Welding Research Council, January 1993
(575) NASA Technical Standard, Fracture Control Requirements for Payloads Using the Space Shuttle, NASA-STD-5003, October, 1996
(576) American National Standard, Space Systems – Metallic Pressure Vessels, Pressurized Structures, and Pressure Components, ANSI/AIAA S-080-1998, American Institute of Aeronautics and Astronautics, 1999
(577) International Space Station Program, Structural Design and Verification Requirements, SSP 30559 Revision C, September, 2000
(578) NASA Technical Handbook, Fracture Control Implementation Handbook for Payloads, Experiments, and Similar Hardware, NASA-HDBK-5010, NASA, May, 2005
(579) Metallic Materials Properties Development and Standardization (MMPDS), formerly MIL-HDBK-5J, DOT/FAA/AR-MMPDS-01, U.S. Department of Transportation, FAA, 2003
(580) ASM Handbook, Volume 2, Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, 1991
(581) INCONEL Alloy 617 Datasheet, Publication No. SMC-029, Special Metals, March 2005
(582) Porowski J.S., and O'Donnell, W.J., "More efficient creep ratcheting bounds," ORNL/Sub-7322/1, Oak Ridge National Laboratory, October 1978
(583) Huddleston, R.L., "An assessment of the adequacy of selected creep-fatigue models for design of Alloy 800H steel components," ORNL/NPR-90/31, Oak Ridge National Laboratory, 1990
(584) Ainsworth, R.A., Ruggles, M.B., and Takahashi, Y., "High temperature flaw assessment procedure," ORNL-6641, Oak Ridge National Laboratory, October, 1990
(585) Ruggles, M.B., Takahashi, Y., and Ainsworth, R.A., "High temperature flaw assessment procedure - final report," ORNL 6677, Oak Ridge National Laboratory, August 1991
(586) "Dissimilar-weld failure analysis and development, prediction of damage in service (PODIS)", EPRI Report EPRI CS-4252, Volume 1-7, 1998
(587) Shah, V.N., et al." Review and Assessment of Codes and Procedures for HTGR Components," NUREG/CR-6816, Nuclear Regulatory Commission, June 2003
(588) Natesan, K., et al. "Materials Behavior in HTGR Environments," Argonne National Laboratory, NUREG/CR-6824 and ANL-02/37, July 2003
(589) "Design Features and Technology Uncertainties for the Next Generation Nuclear Plant," by Independent Technology Review Group, INEEL/EXT-04-01816, Idaho National Engineering and Environmental Laboratory, June 2004
(590) Zinkle, S.J., et al. "Critical Assessment of Structural Materials for Space Nuclear Applications," ORNL/LTR/NR-JIMO/04-08, October 2004

(591) Corum, J.M., and McGreevy, T.E., "R&D Plan for Development of High-Temperature Structural Design Technology for Generation IV Reactor Systems," Oak Ridge National Laboratory, ORNL/TM-2004/309, December 2004
(592) Corum, J.M., and McGreevy, T.E., "R&D Plan for Development of High-Temperature Structural Design Technology for Generation IV Reactor Systems," ORNL/TM-2004/309, December 2004
(593) Jetter, R.I., and McGreevy, T.E., "Simplified Design Criteria for Very High Temperature Applications in Generation IV Reactors", ORNL/TM-2004/308, Rev.1, December 2004.
(594) "Next Generation Nuclear Plant Research and Development Program Plan," Idaho National Laboratory, Oak Ridge National Laboratory, and Argonne National Laboratory, INEEL EXT-05-02581, January 2005

# 6 Reactor Module Shield Segment

(595) JPL Document 982-00007, "JIMO Shield Trade Study: Reactor and Natural Space Radiation," E. Blakeman (ORNL), J. Johnson (ORNL), I. Jun (JPL, Lead), W. McAlpine (JPL), I. Remec (ORNL), J. Yugo (ORNL), April 2004.

# 7 Reactor Module Instrumentation and Control

(596) Barnes, Charles, et al. "Recent Photonics Activities Under the NASA Electronic Parts and Packaging (NEPP) Program." Jet Propulsion Laboratory and NASA Goddard Space Flight Center.
(597) Fielder, Robert S. "Test and Evaluation of Fiber Optic Sensors for High Radiation Space Nuclear Power Applications." Presented at the International Conference of Advances. June 2004.
(598) Fielder, Robert S., et al. "High-Temperature Fiber Optic Sensors, an Enabling Technology for Nuclear Reactor Applications." Proceedings of ICAPP '04. 4298, June 2004.
(599) Grobnic, Dan, et al. "Sapphire Fiber Bragg Grating Sensor Made Using Femtosecond Laser Radiation for Ultrahigh Temperature Applications." IEEE Photonics Technology Letters, Vol. 16, No. 11, November 2004.
(600) Jensen, Fredrik, et al. "Development of a Distributed Monitoring System for Temperature and Coolant Leakage." University of Tokyo, Japan Atomic Energy Research Institute and Hitachi Cable Ltd.
(601) Kimura, Atsushi, et al. "Application of a Raman Distributed Temperature Sensor to the Experimental Fast Reactor JOYO with Correction Techniques." Measurement Science and Technology. Vol. 12, 966-973.
(602) Klemer, Daniel, et al. "Test and Evaluation of Fiber Optic Sensors for High-Radiaiton Space Nuclear Power Applications." Proceedings of ICAPP '04. 4292, June 2004.
(603) Lavin, Milton. "Software Development Requirements." Jet Propulsion Laboratory, D-23713, DocID 57653, September 2003.
(604) Ng, Daniel. "A Self Calibrating Emissivity and/or Transmissivity Independent Multiwavelength Pyrometer." NASA Lewis Research Center. 107149, January 1996.
(605) Ng, Daniel. "Application of the Self Calibrating Emissivity and/or Transmissivity Independent Multiwavelength Pyrometer in an Intense Ambient Radiation Environment." NASA Lewis Research Center. 107151, January 1996.
(606) Ng, Daniel. "Application of the Self Calibrating Emissivity and/or Transmissivity Independent Multiwavelength Pyrometer to Measure the Temperatures of Tungsten and Refractory Material Surfaces." NASA Lewis Research Center. 107152, January, 1996.

(607) Pluta, Philip R. and Smith, Michael A. "SP-100, A Flexible Technology for Space Power From 10's to 100's of KWe." General Electric, SP-100 Programs, Astro-Space Division.

(608) Ruddy, F. H. "Silicon Carbide Semiconductor Radiation Detectors." Presentation at Ohio State University (Power Point Slides). October 2000.

(609) Ruddy, F. H., et al. "Nuclear Reactor Power Monitoring Using Silicon Carbide Semiconductor Radiation Detectors." Nuclear Technology. Vol. 140, No. 2, November 2002. pp 198-208.

(610) Ruddy, F. H., et al. "Silicon Carbide Semiconductor Neutron Detectors." Fourth American Nuclear Society International Topical Meeting on Nuclear Plant Instrumentation, Controls and Human-Machine Interface Technologies (NPIC&HMIT 2004), Columbus, Ohio. September 2004.

(611) Ruddy, F. H., et al. "The Fast Neutron Response of Silicon Carbide Semiconductor Radiation Detectors." Nuclear Science Symposium Conference Record, 2004 IEEE. Vol. 7, October 16-22, 2004. pp 4575-4579.

(612) Seshadri, S., et al. "Demonstration of an SiC Neutron Detector for High-Radiation Environments." IEEE Transactions on Electron Devices. Vol 46, No. 3, 1988. pp 567-571.

(613) Shepard, N. F., et al. "Design and Performance Characteristics for Low Power Space Reactor Systems." General Electric Company, Astro-Space Division.

(614) Shtessel, Yuri B. "Enhanced Sliding Mode Control of the Space Nuclear Reactor System." Proceedings of the 34th Conference on Decision & Control. TP10, December, 1995.

(615) Shtessel, Yuri B. "Sliding Mode Control of the Space Nuclear Reactor System." IEEE Transactions on Aerospace and Electronic Systems. V34, April 1998.

(616) Stinson-Bagby, Kelly L. and Fielder, Robert S. "Fiber Bragg Gratings for High-Temperature Thermal Characterization" Proceedings of ICAPP '04. 4299, June 2004.

(617) Vicente, F. A. "Space Nuclear Power Applied to Electric Propulsion." General Electric Company, Astro-Space Division.

(618) Watanabe, Kenichi, et al. "Development of Failed Fuel Detection and Location Technique Using Resonance Ionization Mass Spectrometry." Journal of Nuclear Science and Technology. Vol. 38, No. 10, October 2001. pp 844-849.

(619) Wood, Richard, et al. "I&C Requirements, Operations and Technology Base." Los Alamos National Laboratory and Oak Ridge National Laboratory. JIMO Reactor Module Peer Review, January 2004.

# 8 Reactor Module Integrated Testing

(620) NASA Kennedy Space Center Letter, KS-VA-7384, "Prometheus Project Spacecraft Processing Facility Conceptual Study," 6/17/05

# 9 Reactor Module Safety and Mission Assurance

(621) Bartram, B.W. and Weitzberg, A. "Radiological Risk Analysis of Potential SP-100 Space Mission Scenarios." August 19, 1988.

(622) Bartram, Bart W. and David K. Dougherty. "A Long Term Radiological Risk Model for Plutonium-Fueled and Fission Reactor Space Nuclear System", DOE/ET/32079--2, DE87 010466, June 12, 1987.

(623) Damon, Dennis R."ACCIDENT INITIATING EVENTS FOR SP-100 REFERENCE FLIGHT SYSTEM PRELIMINARY SAFETY ANALYSIS." GE Space Nuclear Engineering & Technology. PIR#: U-1T26-SP-100-402, February 11, 1988.

(624) Damon, D.R., et al. "SP-100 MISSION RISK ANALYSIS." GE Aerospace. GESR-00849, Vol. I - Main Report, August 1989.

(625) Damon, D.R., et al. "SP-100 MISSION RISK ANALYSIS." GE Aerospace. GESR-00849, Vol. II - Appendices, August 1989.Damon, Dennis R. "SP100 NAT RISK ANALYSIS INITIATING EVENT DOCUMENT." GE Space Nuclear Engineering & Technology. PIR#: U-1T26-SP-100-960, March 29, 1991.

(626) White, GW and Pupek, CP. "Hazard Analysis Summary For The SP-100 Space Reactor Power System (SRPS) WBS (J.4.4)." GE Space Division. PIR#: U-1335-SP-100-488. April 27, 1988.

(627) Wheeler, Timothy A. "Cassini Spacecraft Uncertainty Analysis Data and Methodology Review and Update." Vol. 1: Updated Parameter Uncertainty Models for the Consequence Analysis. Sandia National Laboratories. SAND2000-2719/1, November 2000.

(628) Wyss, Gregory D. "Cassini Spacecraft Uncertainty Analysis Data and Methodology Review and Update." Vol 2: A Technical Description of the Sampling Methods Employed in the Cassini Uncertainty Analysis. Sandia National Laboratories. SAND2000-1764/2, July 2000.

(629) Thornton, Marcia Lt Col, USAF. "Safety  Evaluation Report (SER)." United States Air Force. July 31, 1997.

(630) Kastenberg, W.E. and Richard Wilson. "Risk of nuclear powered space probes", December 19, 2003.

(631) Frank, Michael V, P.E., Ph.D. "THE DECISION TO LAUNCH: A NUCLEAR RISK ASSESSMENT OF THE CASSINI MISSION." Safety Factor Associates, Inc.

(632) Carter, R.D., et al. "Criticality Handbook." ARH-600 Vol. I, June 30, 1968.

(633) Carter, R.D., et al. "Criticality Handbook." ARH-600 Vol. II, May 23, 1969.

(634) Carter, R.D., et al. "Criticality Handbook." ARH-600 Vol. III, September 1, 1971.

(635) Marshall, Albert C. "SPACE REACTOR SAFETY, 1985-1995 LESSONS LEARNED." Sandia National Laboratories. SAN095-2362C, CONF-960109-4.

(636) Kehler, C Robert, et al. "Eastern and Western Ranges" (EWR), 127-1, Dec. 31, 1999.

(637) Vesely, W.E., et al. "Fault Tree Handbook", NUREG-0492, January 1981.

(638) Swain, A.D. and H.E. Guttmann. 'Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications - Final Report." NUREG/CR-1278F, SAND80-0200, August 1983.

(639) Schoenecker, Lee, et al. "Interagency and Intergovernmental Coordination for Environmental Planning." United States Air Force Instruction 32-7060, March 25, 1994.

(640) Reinertson, Kenneth L. "The Environmental Impact Analysis Process" – United States Air Force Instruction 32-7060, January 24, 1995.

(641) "NASA Procedures and Guidelines." QS/Safety & Risk Management Division. NPG 8715.3, January 24, 2000 - January 24, 2006.

(642) "30$^{TH}$ Space Wing Safety Vandenberg AFB CA – Range Safety Operations Requirements (RSOR)." September 16, 2002.

# 10 Reactor Module Ground Test Reactor

(643) Cooper, R. H., et al. "Proposal for the SP-100 Ground Engineering System Reactor Test." Oak Ridge National Laboratory. April 1985.

(644) Spellman, D. J. and G.T. Mays. "Evaluation of DOE Facilities for Potential Use in JIMO for Ground Nuclear Test." Oak Ridge National Laboratory. January 2004.

(645) Bazinet, G., et al. "Jupiter Icy Moons Orbiter Reactor Test Assembly: Ground Test Facility Requirements – Preliminary requirements for a Space Reactor Ground Test Facility Based on the SP-100 Ground Engineering System Test Site Experience." Pacific Northwest National Laboratory. October 31, 2003.

(646) Wiborg, J. C. "Initial Cost Estimate for the Jupiter Icy Moons Orbiter Reactor Assembly Ground Test Facility – An Estimated Cost Range for a Space Reactor Ground Test Facility Based on the SP-100 Ground Engineering System Test Site Experience." Pacific Northwest National Laboratory. August 29, 2003.

# 11 Reactor Module Materials Test and Evaluation

## 11.1 Ni-base Superalloys Joining

(647) Fairbanks, N. 1975. "High temperature braze for superalloys." Interim Technical Report, General Electric Company Cincinnati, OH April 1- September 30.

(648) Fairbanks, N. 1975. "High temperature braze for superalloys." Interim Technical Report, General Electric Company Cincinnati, OH April 1- September 30.

(649) Discus, D.L., Buckley, J.D. 1972. "Effects of High-Temperature Brazing and Thermal of Cycling on Mechanical Properties of Hastelloy X." NASA TM X-2704, December.

(650) Lugscheider, E. and Aachen, D.K. 1991. "New low-melting nickel-based high-temperature brazing alloys." Welding and Cutting, 4: E84-E86.

(651) Dicus, D.L. and Buckley, J.D. no date listed. "The effects of high-temperature brazing and thermal cycling on the mechanical properties of Hastelloy X." NASA Langley Report L-8376.

(652) Dini, J.W.. "The role of electroplated coatings in metal joining." Welding Journal. May 1996, pp 47-49.

(653) Ernst, S.C. "Weldability studies of Haynes 230 alloy." Welding Research Supplement. April 1994. pp 80-89.

(654) Dolby, R. no date listed. "Trends in materials joining science and technology in Western Europe", ASM Conference on Welding and Joining Science and Technology, TWI, Abington Hall, Cambridge, UK: 666-676

(655) Partridge, P.G. and Ward-Close, C.M.. "Diffusion Bonding of advanced materials." Metals and Materials. Volume 5, No. 6: June 1989, pp 334-339.

(656) Bienvenu, Y., et al. "Diffusion bonding of Ni-base superalloys to manufacture turbine components with a graded microstructure." FGM 94, Proceedings of the 3rd International Symposium on Structural and Functional Gradient Materials, Swiss Federal Institute of Technology, Lausanne, Switzerland; 10-12 Oct1995.: pp 487-494

(657) Sun, Z. "Joining dissimilar material combinations: materials and processes." International Journal of Materials and Product Technology, Volume 10, No. 1-2. 1995. pp 16-26.

(658) Wlodek, S.T., and R.D. Field. "The Effects of Long Time Exposure on Alloy 718." International Symposium on Superalloys 718, 625, 706 and Derivatives. 1994.

(659) Moore, T.J. "Evaluation of a Brayton Cycle Recuperator After 21,000 Hours of Ground Testing." NASA-Lewis, NASA-TM-709091, February 1979.

(660) Kelly, T.J.. "Weld Discontinuities and Imperfections." Properties of Nickel-Alloy Welds. 1997. pp 329-352.

(661) Nishimoto, K. “Weldability and high temperature performance of welded joints of heat-resisting alloys.” Welding International, Volume 19, No. 5. 2005. pp 349-356.

## 11.2 Titanium Joining

(662) Kuo, M., Albright, C.E., Baeslack III, W.A.. “Dissimilar friction welding of titanium alloys to alloy 718.”, EWI Research Report MR9302, February 1993.
(663) Kuo, M. 1994. "Dissimilar Friction Welding of Titanium Alloys to Alloy 718." Dissertation.
(664) Berger, D.D.. “Vacuum brazing titanium to Inconel.”, Welding Journal, November 1995. pp 35-38.
(665) No authors listed.. “Designing Brazed Joints.” Metal Heat Treating. November / December 1994. pp 23-27.
(666) Seretsky, J., Ryba, E.R.. “Laser Welding of Dissimilar Metals-Titanium to Nickel.” Welding Journal, Research Supplement, Vol. 55. 1973 pp 208s-211s.
(667) Aleman, B., Gutierrez, I., Urcola, J.J., “Interface Microstructures in the Diffusion Bonding of a Titanium Alloy Ti6242 to an Inconel 625”, 1995
(668) Ells, C.E., Taylor, G.F., and Mansey, R.C.. “Joining Nickel to Titanium.” Titanium Science and Technology. Vol. 4. 1973. pp 2527-2534.
(669) Davis, J.R., et al. . “Properties and Selection: Nonferrous Alloys and Special-Purpose Materials.” ASM Handbook, Formerly Tenth Edition, Metals Handbook, V2 R669.02 ASMINT C1.
(670) Peacock, D. 1998. “Connecting Titanium to Other Metals.” Materials Performance (USA), Vol. 37, No.8, August: 68-69.
(671) Partridge P.G., Ward-Close C.M.. “Application of Diffusion Bonding to Advanced Materials.” Near Net Shape Manufacture. 1989. pp 250-256
(672) Ghosh M., et al. “Influence of Interface Microstructure on the Strength of the Transition Joint between Ti-6Al-4V and Stainless Steel.” Metallurgical and Materials Transactions A, Vol. 36A. 2005, pp 1891-1899.
(673) Fuji A., et al. “Improving Tensile Strength and Bend Ductility of Titanium/AISI 304L Stainless Steel Friction Welds.” Materials Science and Technology, Vol. 8, No. 3. 1992. pp 219-235.
(674) Sheela G., et al. “Joining of Passive Dissimilar Metals- Ti, SS, and AL: An Emerging Application of Electrodeposition.” Transactions of the Institute of Metal Finishing, Vol. 82, January-March 2004. pp. 24-28.
(675) Chen, C.C. “An Investigation of Diffusion Bonding of Titanium to Stainless Steel,” Titanium ’80: Science and Technology. . Vol. 4. 1980. pp. 2379-2388.
(676) Aleman, B., Gutierrez, I., and Urcola, J.J “Interface Microstructures in the Diffusion Bonding of a Titanium Alloy Ti6242 to an Inconel 625.”1995.
(677) Destefani, James D. “Introduction to Titanium and Titanium Alloys.” ASM Handbook. Vol 2, 1990.
(678) Lampman, Steven R. “Wrought Titanium and Titanium Alloys.” ASM Handbook. Vol 2, 1990.

## 11.3 Nickel Alloys to Stainless Steel

(679) Brosilow, R., Kiser, S. “Better Fillers for Nuclear Welds.” Welding Design and Fabrication. Vol. 66, No. 12, December 1993. pp 32-33.
(680) Lopez, B., Gutierrez, I., Urcola, J.J.. “Microstructural Analysis of Steel-Nickel Alloy Clad Interfaces,” Materials Science and Technology, Vol. 12, No. 1, January 1996. pp 45-55.

(681) Dupont, J.N., et al. "Microstructural Evolution and Weldability of Dissimilar Welds between a Super Austenitic Stainless Steel and Nickel-Based Alloys." Welding Journal, Vol. 82, no. 6, June 2003. pp 125S-135S.

## 11.4 Silicon Carbide Joining

(682) Jackson, M.R., Mehan, R.L., Davis, A.M., and Hall, E.L.. "Solid state SiC/Ni alloy reaction," Metallurgical Transaction A Volume 14A, March 1983. pp 355-364.
(683) McDermid, J.R. and Drew, R.A.L.. "Thermodynamic brazing alloy design for joining silicon carbide." Journal of the American Ceramic Society, Volume 74, No. 8. 1996. pp. 1855-1860.
(684) McDermid, J.R., Pugh, M.D., and Drew, R.A.L. "The interaction of reaction-bonded Silicon Carbide and Inconel 600 with a nickel-based brazing alloy." Metallurgical Transactions A, Volume 20A, September. pp 1803-1810.
(685) Riccardi, B., et al.. "Low Activation Brazing Materials and Techniques for SiCf/SiC Composites." Journal of Nuclear Materials. 2002. pp 307-311.
(686) Hiraoka, Y., and Nishikawa, S. "Joining of Single-crystalline Molybdenum and Carbon-ceramics by Using Palladium and Palladium-Silver Alloy as Brazing Metal." International Journal of Refractory Metals and Hard Materials, Volume 14, 1996. pp 311-317.
(687) Rabin, B.H. "Joining of Silicon Carbide/Silicon Carbide Composites and Dense Silicon Carbide Using Combustion Reactions in the Titanium-Carbon-Nickel System." Journal of the American Ceramic Society, Volume 75, No. 1, 1992, pp 131-135.
(688) Hall, E.L., et al. "Chemistry and distribution of phases produced by solid state SiC/NiCrAl reaction." Metallurgical Transaction A, Volume 14A, May 1983. pp 781-790.
(689) Lewinsohn, C.A., et al. "Silicon carbide-based materials for joining silicon carbide composites for fusion energy applications." Journal of Nuclear Materials, Volume 307-311: 2002. pp 1232-1236.
(690) Larker, R., et al. "Diffusion bonding reactions between a SiC/SiC composite and two superalloys during joining by hot isostatic pressing.", Acta Metallurgica Materialia, Volume 40, No. 11, 1992. pp 3129-3139.
(691) Moore, T.J.. "Feasibility Study of the Welding SiC." Journal of the American Ceramic Society, Volume 68 Number 6, 1985. pp C-151-C-153.
(692) Li, S., et al. "Interdiffusion involved in SHS welding of SiC ceramic to itself and to Ni-based Superalloy." International Journal of Refractory Metals and Hard Materials, Volume 18: 2000. pp 33-37.
(693) Li, J., and Xiao, P. "Fabrication and characterization of silicon carbide/Superalloy interfaces." Journal of the European Ceramic Society, Volume 24, 2004. pp 2149-2156.
(694) Li, S., Zhou, et al. "Joining of SiC ceramic to Ni-based superalloy with functionally gradient material fillers and a tungsten intermediate layer." Journal of Materials Science, Volume 38: 2003. pp 4065-4070.
(695) Li, S., et al. "Joining of SiC ceramic to Ni-based superalloy with Cu intermediate layer." Key Engineering Materials, Vol. 217, 2002. pp 101-110.

## 11.5 Tribology

(696) Ying, T.N., et al. "Tribology of Si-based Ceramics: Wear Mechanisms", Tribology Transactions, Volume 40, 1997. pp. 685-693.
(697) Kasiarova, M., et al. "Wear and Creep Characteristics of a Carbon-Derived Si3N4/SiC Micro/Nanocomposite." Source not listed. No date listed.

(698) Arnell, R.D., et al. "Tribology; Principles and Design Applications." Springer-Verlag. 1991.
(699) Holmberg, K., Matthews, A. "Tribology of Engineered Surfaces." Chapter 7,MEP Book – Wear – Materials, Mechanisms and Practices, submitted June 29 2004.
(700) Kubart, T., et al. "Temperature Dependence of Tribological Properties of MoS2 and MoSe2 Coatings." Surface and Coating Technology. V193. 2005. pp. 230-233.
(701) Doll, G., Evans, R., Ribaudo, C. "Improving the Performance of Rolling Contact Bearings with Tribological Coatings." Surface Engineering in Materials Science. VIII. 2005. pp. 153-162.
(702) Miyoshi, K. "Aerospace Mechanisms and Tribology Technology: Case Studies." NASA Glenn Research Center. NASA/TM-1999-107249, Aug. 1999.
(703) Jones, W., Jansen, M. "Lubrication for Space Applications." Sest, Inc. and University of Toledo. NASA/CR-2005-213424, Jan. 2005.
(704) Miyoshi, K., Wheeler, D., Zabinski, J. "Surface Chemistry, Friction, and Wear Properties of Untreated and Laser-Annealed Surfaces of Pulsed-Laser-Deposited WS2 Coatings." Lewis Research Center and Wright Laboratory. NASA Technical Memorandum 107342, Dec. 1996.
(705) DellaCorte, C., et al. "A System Approach to the Solid Lubrication of Foil Air Bearings for Oil-Free Turbomachinery." Glenn Research Center and ARL. NASA/TM-2002-211482, Oct. 2002.
(706) DellaCorte, C., Edmonds, B.J. "Preliminary Evaluation of PS300: A New Self-Lubricating High Temperature Composite Coating for Use to 800°C." Lewis Research Center. DOE/NASA/50306-8 and NASA TM-107056, Nov. 1995.
(707) DellaCorte, C. "The Effects of Substrate Material and Thermal Processing Atmosphere on the Strength of PS304: A High Temperature Solid Lubricant Coating." Glenn Research Center. NASA/TM-2002-211483, Sep. 2002.
(708) Stanford, M., et al. "Thermal Effects on a Low Cr Modification of PS304 Solid Lubricant Coating." Glenn Research Center and Cleveland State University. NASA/TM-2004-213111, June 2004.
(709) Sliney, H. "Sliding Contact PM212 Bearings for Service to 700°C©." Tribology Transactions. V49. 1997. pp. 579-588.
(710) Balić, E., Blanchet, T. "Thrust-washer tribological evaluation of PS304 coating against Rene 41." Wear. V259. 2005. pp. 876-881.
(711) Sliney, H., Waters, W., Soltis, R. "MS212-A Homogeneous Sputtered Solid Lubricant Coating for Use to 800°C." Lewis Research Center and OMNI Corporation. NASA Technical Memorandum 107437, May 1997.
(712) Sliney, H. "Sliding Contact PM212 Bearings for Service to 700°C©." Tribology Transactions. V49. 1997. pp. 579-588.
(713) Liu, G., Mengelle, C. "CrC•CaF2-A PVD Stratified Coating for Friction and Wear Under Heavy Load at High Temperature." Metals and Manufacturing Processes. V17. 2002. pp. 37-44.
(714) Johnson, R. "Coatings for Fast Breeder Reactor Components." Thin Solid Films. V118. pp. 31-47. 1984.
(715) Taylor, D., et al. "Sol-Gel Derived Nickel Titanate for Tribological Coatings." Sol-Gel Commercialization and Applications. pp. 121-126.
(716) Schuster, G., et al. "Measurement of Static and Dynamic Friction Coefficients for SP-100 Tribological Coatings in a High-Temperature/High-Vacuum Environment." American Institute of Physics. 940101. 1994.
(717) Krumpiegl, T., et al. "Amorphous Carbon Coatings and Their Tribological Behavior at High Temperatures and in High Vacuum." Surface and Coatings Technology. V120-121. 1999. pp. 555-560.

(718) Habig, K. "Fundamentals of the Tribological Behavior of Diamond, Diamond-Like-Carbon and Cubic Boron Nitride Coatings." Surface and Coatings Technology. V76-77. 1995. pp. 540-547.

## 11.6 Refractory Metal Alloys Joining

(719) Nieh, T.G. "Solid-state diffusion bonding of tungsten-25rhenium alloy." Journal of Materials Science, Volume 21: 1986. pp 2327-2334.
(720) Masumoto, H., et al. "Joining of molybdenum and titanium." Welding International, Volume 7, 1993 pp 845-849.
(721) Lison, R. and Stelzer, J.F. "Diffusion welding of reactive and refractory metals to stainless steel." Welding Research Supplement. October 1979. pp 306-314.
(722) Paprocki, S.J., et al. "Solid-Phase Bonding of Columbium." Columbium Metallurgy, Metallurgical Society Conference, Volume 10, Bolton Landing, NY, June 9-10. 1960.
(723) Leong, K.H. "Laser welding of refractory metals.", Journal of Laser Applications, Volume 13, No. 5. October 2001. pp 199-203.
(724) Raab, B. and Schock, A. "Electron beam welding of tungsten to tungsten/rhenium and tungsten/rhenium to niobium." Presented at the 1970 Thermionic Conversion Specialists Conference, October 26-29, 1970, Miami, Florida. pp 7-11
(725) Lessmann, G.G. and Gold, R.E. "The weldability of tungsten base alloys." Welding Research Supplement, December 1969. pp 528-542.
(726) Cadden, C.H., and Odegard Jr., B.C. "Refractory metal joining for first wall applications." Journal of Nuclear Materials, Volume 283-287, 2000. pp 1253-1257.
(727) Kirby, L.J. and Fullam, H.T. "Reaction of selected metal-metal couples at 1100°C." Battelle-Northwest Letter BNWL-421, June1967.
(728) Santella, M. "Joining issues for refractory alloys.", ANS—Space Nuclear Conference 2005 Powerpoint presentation, San Diego, CA, June 5-8 2005
(729) Arcella, F.G. 1974. "Interdiffusion Behavior of Tungsten or Rhenium and Group V and VI Elements and Alloys of the Periodic Table Part I." Westinghouse Astronuclear Laboratory (WANL) /NASA-Lewis, NASA-CR-134490 Part 1, WANL-M-FR-74-005, September.
(730) Arcella, F.G. "Interdiffusion Behavior of Tungsten or Rhenium and Group V and VI Elements and Alloys of the Periodic Table Part IIA (Appendices) Final Report." Westinghouse Astronuclear Laboratory Laboratory (WANL)/NASA-CR-134526, Part II, WANL-M-FR-74-005, September 1974
(731) Buckman Jr., R.W., Goodspeed, R.C. "Evaluation of Refractory/Austenitic Bimetal Combinations Final Technical Report."Westinghouse Astronculear Laboratory(WANL), NASA – Lewis, WANL-PR-(EE)-004, August. 1969
(732) Stoner, D.R. "Evaluation of Tantalum/316 Stainless Steel Transition Joints." Westinghouse Astronuclear Laboratory (WANL) /NASA – Lewis, NASA CR 121111, WANL-M-FR-72-006, December 1972.
(733) Camerson, H.M. "Thermal Cycling Test on a 3-Inch-Diameter Columbium – 1 Percent Zirconium to 316 Stainless Steel Transition Joint.", NASA-Lewis, NASA Technical Memorandum – NASA TM X-2118, November 1970
(734) Schwartz, M.M. "The Fabrication of Dissimilar Metal Joints Containing Reactive and Refractory Metals." WRC Bulletin 210, October 1975
(735) Thompson, S.R., Marble, J.D., EkVail, R.A. "Development of Optimum Fabrication Techniques For Brazed Ta/Type 316 SS Tubular Transition Joints." NASA-Lewis, CR-72736, GESP-521. No date listed.
(736) Stoner, D.A. "Joining Refractory/Austenitic Bimetal Tubing", Westinghouse Astronuclear Laboratory (WANL)/NASA-Lewis, WANL-PR-(ZZ)-001 NASA-CR-72275. no date listed

(737) Lessman, G.G., Gold, R.E. "Determination of Weldability and Elevated Temperature Stability of Refractory Metal Alloys." Westinghouse Astronuclear Laboratory NASA-CR-1607 (v.)2 c.1, September 1970.
(738) Ferry, P.B., Page, J.P. "Metallurgical Study and Niobium/Type 316 Stainless Steel Duplex Tubing." NAA-SR-11191 Metals, Ceramics and Materials, January 5 1966.
(739) Kass, J.N., Stoner, D.R. "Evaluation of Tantalum/316 Stainless Steel Bimetallic Tubing." NASA CR-1575, May 1970.
(740) Wiffen, F.W. "Joining Issues For Interfaces Between Dissimilar Metals." Critical Assessment of Structural Materials for Space Nuclear Applications, October 1 2004. pp 179 – 190.
(741) Lessman, G.G, Gold, R.E. "Determination of the Weldability and Elevated Temperature Stability of Refractory Metal Alloys." Westinghouse Astronuclear Laboratory (WANL)/NASA – Lewis, WANL-PR-(P)-017, October 1969.
(742) Gold, R.E., Lessman, G.G. "Influence of Restraint and Thermal Exposure on Welds in T-III and ASTAR-811C." Westinghouse Aeronautics Laboratory (WANL)/NASA-Lewis, NASA CR-72858, WANL-PR(VVV)-001, March 1971.
(743) Lessman, G.G., Gold, R.E. "Determination of Weldability and Elevated Temperature Stability of Refractory Metal Alloys V-Weldability of Tungsten Base Alloys." Westinghouse Astronuclear Laboratory/NASA-Lewis, NASA Cr-1611, September 1970.
(744) Lessman, G.G. "Welding of Refractory Alloys." Refractory Alloy Technology for Space Nuclear Power Applications, CONF-8308130 (DE4001745), January 1984. pp 145-167.
(745) Lessman, G.G. "Determination of Weldability and Elevated Temperature Stability of Refractory Metal Alloys." Westinghouse Astronuclear Laboratory NASA CR-1607, August 1970

## 11.7 Cast to Wrought Ni-base Superalloy Joining

(746) Zhang, Li, et al. "Autogenous welding of Hastelloy X to Mar-M 247 by laser." Journal of Materials Processing Technology, Volume 70, 1997. pp 285-292.
(747) Schmitt-Thomas, K.G. and Siede, R. "Flywheel friction welding of nickel base superalloys of different hot ductility." Quality and Reliability in Welding, Proceedings, International Conference, Hangzhou (Hangchow), China, Session D, Paper D8; September 6-8 1984
(748) General Atomic, Co. "Transition piece for joining together tubular pieces." UK Patent Application, No. 8113511, May 1 1981.
(749) Weiss, C.D., Moen, L.J., and Hallett, W.M. "Design considerations in inertia welding at turbocharger and gas turbine component.", American Society of Mechanical Engineers (ASME) Gas Turbine Conference and Products show Paper 71-GT-21, Houston, TX, March 28-April 1 1971.
(750) No author listed. "Inconel Welding Electrode 117." Alloy Digest, Volume Ni-300, July 1984. pp 2.
(751) Hoppin, G.S. III and Danesi, W.P. "Manufacturing Processes for Long-Life Gas Turbines." Journal of Metals, Volume 38, No. 7, July 1986. pp 20-23.
(752) Harrison, W.H., Kelly, T.J., and Weimer, M.J. "Method of Restoration of Mechanical Properties of Cast Inconel 718 for Serviced Aircraft Components." General Electric Co., US Patent Application Publication, Appl. No. 10/029,365, Dec. 20, 2001, Pub. No. US 2003/0116242 A1, June 26, 2003
(753) Kelly, T.J. "Properties of Nickel-Alloy Welds," Weld Discontinuities and Imperfections. 1997. pp. 329-352.

## 11.8 Miscellaneous Joining

(754) Winiowski, A. “Application of diffusion brazing to the bonding of metals and alloys.” Welding International, Volume 13, No. 11, 1999. pp 870-874.
(755) Lin, R.Y., et al.. “The infrared infiltration and joining of advanced materials.” Journal of Metals, March1994. pp 26-30.
(756) Baeslack III, W.A. and Froes, F.H. “Joining similar and dissimilar advanced engineered materials.” Journal of Materials, March 1995. pp 13-15.
(757) Pang, J.W.L., et al... “Effects of tooling on the residual stress distribution in an inertia weld.” Materials Science and Engineering A, Volume 356, 2003. 405-413.
(758) Ochi, H., et al. “Friction welding using insert metal”, Welding Journal, March, pp 36-39.
(759) Gale, W.F. and Butts, D.A. “Transient Liquid Phase Boding – Overview”, Science and Technology of Welding and Joining, Vol. 9, No. 4, 2004. pp 1-18.
(760) Mountford, J.A. “Titanium meeting the challenge of the new millennium.” Corrosion 2001, Paper 01329. 2001.
(761) Kunitomi, K., Takeda, T., Oiré, T., Iwata, K. “Development of Compact Heat Exchanger with Diffusion Welding.” Japan Atomic Research Institute, Sumitomo Precision Products Co. Ltd, Japan. X9642788.
(762) Lessman, G.G., Gold, G.E.. “The Varestraint Test For Refractory Metals.” NASA-CR-72828, WANL-PR(VVV)-002, November 1970. pp 42.
(763) Christopher, J.D. “Machining Refractory Alloys – An Overview.” Refractory Alloy Technology for Space Nuclear Power Applications CONF-8308130 (DE84001745), January 1984, pp 130-145.
(764) Conway, J.B. “Mechanical and Physical Properties of Refractory Metals and Alloys.” Refractory Alloy Technology For Space Nuclear Power Applications CONF-8308130 (DE84001745), January 1984.
(765) Hosking, F.M. “Sodium compatibility of refractory metal alloy-type 304L stainless steel joints.” Welding Research Supplement, July 1995. pp 181-190.
(766) Huang, X., Richards, N.L., and Chaturvedi, M.C. “Effect of grain size on the weldability of cast alloy 718.” Materials and Manufacturing Processes, Volume 19, No. 2. 2004. pp 285-311.
(767) Postnikov, N.S., et al. “Properties of welded joints between cast and wrought Al alloys.” Russian Castings Production (Translated from Liteinoe Proizvodstvo, 1974, (10), pp. 10-13), pp. 425-427.
(768) Zhao, J.C. “Reliability of the diffusion-multiple approach for phase diagram mapping.”, Journal of Materials Science, Volume 39, 2004. pp 3913-3925.
(769) Krotz, P.D., Davis, W.M., and Wisner, D.L. “Brazing dissimilar metals.” NASA Tech Briefs, Volume 20, No. 3, March 1996. pp 91-92.
(770) Trego, L. “Fasteners for aerospace structures.” Aerospace Engineering, November 1991. pp 19-23.
(771) Stoner, D.R. and Lessmann, G.G. “Measurement and control of weld chamber atmospheres.” Technical Paper for Presentation at the AWS 46th Annual Meeting, Chicago, IL, April 26-30 1965
(772) Zhao, J.C. “The diffusion-multiple approach to designing alloys.” Annual Reviews of Materials Research, Volume 35: 2005. pp 51-73.
(773) Lee, W.B., Yeon, Y.M., and Jung, S.B. “Mechanical properties related to the dominant microstructure in the weld zone of dissimilar formed Al alloy joints by friction stir welding.” Journal of Materials Science, Volume 38: 2004. pp 4183-4191.
(774) Avery, R.E. “Pay attention to dissimilar-metal welds.” Chemical Engineering Progress, May 1991, pp 70-75.
(775) Hawthorne, J.R., Reed, J.R., and Sprague, J.A. “Fracture resistance of two ferritic stainless steels after intermediate temperature irradiation.” Effects of Radiation on Materials: Twelfth International Symposium, ASTM STP 870, F.A. Garner and J.S.

Perrin, Eds., American Society for Testing and Materials, Philadelphia, 1985, pp 580-604.
(776) Wang, Z., Xu, B., and Ye, C. “Study of the martensite structure at the weld interface and the fracture toughness of dissimilar metal joints.” Welding Research Supplement, August 1993. pp 397-402.
(777) Lison, R. “Welding and brazing of materials on the basis of intermetallic phases.” Welding and Cutting, Volume 52, No. 2, 2000. pp E22-E27.
(778) Krishnardula, et al. “Joining of ferritic oxide dispersion strengthened alloys.” International Symposium of Research Students (ISRS) on Material Science and Engineering, December 2004. pp 20-22.
(779) Walters, R.P. and Covino Jr., B.S. “Evaluation of high-temperature diffusion barriers for the Pt-Mo system.” Metallurgical Transaction A, Volume 19A, September 1988. pp 2163-2170.
(780) Klueh, R.L. “Elevated- Temperature Ferritic and Martensitic Steels for Space Nuclear Reactor Service.” Critical Assessment of Structural Materials for Space Nuclear Applications – Final Draft, ORNL/LTR-JIMO/04-08, October 1 2004. pp 17-46.
(781) Igata, N. “Ferritic/Martensitic Dual-Phase Steel as Fusion Reactor Materials.” Journal of Nuclear Materials, Volume 133 & 134, 1985. pp 141-148.
(782) Zirker, L.R., et al. “Fabrication of Oxide Dispersion Strengthened Ferritic Clad Fuel Pins.” International Conference on Fast Reactors and Related Fuel Cycles, Kyoto, Japan, October 28-31, 1991.
(783) Irvine, Duncan. “Metal Joining: Developments in Metal Joining Techniques.” Discovery: The Science and Technology Journal of AWE. V15, February 2004. pp. 24-33.

## 11.9 Helium Gas Corrosion

(784) T. Noda, M. Okada and R. Watanabe, “The Compatibility of Candidate First Wall Metallic Materials with Impure Helium”, Journal of Nuclear Materials, V85&86, 1979. pp 329-333.
(785) W.J. Quadakkers and H. Schuster, “Corrosion of High Temperature Alloys in the Primary Circuit Helium of High Temperature Gas Cooled Reactors. – Part I: Theoretical Background”, Werkstoffe und Korrosion, V36, 1985. pp 141-150.
(786) W.J. Quadakkers, “Corrosion of High Temperature Alloys in the Primary Circuit Helium of High Temperature Gas Cooled Reactors. – Part II: Experimental Results”, Werkstoffe und Korrosion, V36, 1985. pp 335-347.
(787) W.J. Quadakkers and H. Schuster, “Thermodynamic and Kinetic Aspects of the Corrosion of High-Temperature Alloys in High-Temperature Gas-Cooled Reactor Helium”, Nuclear Technology, V66, 1984. pp 383-391.
(788) H.G.A. Bates, “The Corrosion Behavior of High-Temperature Alloys During Exposure for Times up to 10 000 h in Prototype Nuclear Process Helium at 700 to 900C”, Nuclear Technology, V66, 1984. pp 415-428.
(789) M. Cappelaere, M. Perrot and J. Sannier, “Behavior of Metallic Materials Between 550 and 870C in High Temperature Gas-Cooled Reactor Helium Under Pressures of 2 and 50 Bar”, Nuclear Technology, V66, 1984. pp 465-478.
(790) M.R. Warren, “Experimental Work on Alloy Development for High-Temperature Gas-Cooled Reactor Systems”, Nuclear Technology, V66, 1984. pp 102-116.
(791) W.R. Johnson, L.D. Thompson and T.A. Lechtenberg, “Design of Wrought Nickel-base Alloys for Advanced High-Temperature Gas-Cooled Reactor Applications”, Nuclear Technology, V66, 1984. pp 88-101.
(792) A.V. Dean and P.J. Ennis, “The Development of High-Strength Alloys Resistant to Corrosion in Impure Helium”, Nuclear Technology, V66, 1984. pp 117-123.

(793) A.C. Lingenfelter, "Inconel-618E: An Alloy Developed for High-Temperature Gas-Cooled Reactor Service", Nuclear Technology, V66, 1984. pp 63-68.

(794) R. Tanaka and T. Kondo, "Research and Development on Heat-Resistant Alloys for Nuclear Process Heating in Japan," Nuclear Technology, V66, 1984. pp 75-87.

(795) L.W. Graham, "High Temperature Corrosion in Impure Helium Environments", High Temperature Technology, V3, N1, 1985. pp 3-14.

(796) K.G.E. Brenner and L.W. Graham, "The Development and Application of a Unified Corrosion Model for High-Temperature Gas-Cooled Reactor Systems", Nuclear Technology, V66, 1984. pp 404-414.

(797) L.W. Graham, "Corrosion of Metallic Materials in HTR-Helium Environments", Journal of Nuclear Materials, V171, 1990. pp 76-83.

(798) L.W. Graham, K.G.E. Grenner and K. Krompholz, "The Behaviour of High Temperature Alloys During Exposure in Impure Helium", High Temperature Metallic Materials for Gas-Cooled Reactors. Vienna, Austria, May 4-6, 1981. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-4, paper K.

(799) W.R. Johnson and G.Y. Lai, "Interaction of Metals with Primary Coolant Impurities: Comparison of Steam-Cycle and Advanced HTGRs", High Temperature Metallic Materials for Gas-Cooled Reactors. Vienna, Austria, May 4-6, 1981. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-4, paper J.

(800) D.W. McKee and R.G. Frank, "Corrosion Behavior of Experimental Alloys in Controlled Purity Helium", High Temperature Metallic Materials for Gas-Cooled Reactors. Vienna, Austria, May 4-6, 1981. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-4, paper I.

(801) R.H. Cook, R. Exner and L.W. Graham, "Post-Service Examination of a 10 MW Helium-Helium Heat Exchanger and Comparison with Long Term Behaviour in Laboratory Tests", High Temperature Metallic Materials for Gas-Cooled Reactors. Cracow, June 20-23, 1988. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-18. pp 129-136.

(802) O.F. Kimball, "Thermal Stability and Environmental Compatibility of Inconel 617", High Temperature Metallic Materials for Gas-Cooled Reactors. Cracow, June 20-23, 1988. International Atomic Energy Agency, Vienna, Austria, International Working Group on Gas-Cooled Reactors, IWGGCR-18. pp 65-72.

(803) H. Nickel, F. Schubert and H. Schuster, "Evaluation of Alloys for Advanced High-Temperature Reactor Systems", Nuclear Engineering and Design, V78, 1984. pp 251-265.

(804) C.M. Scheuermann, T.J. Moore and D.R. Wheeler, "Preliminary Study of Niobium Alloy Contamination by Transport Through Helium", NASA-TM-88952.

(805) K. Natesan, A. Purohit and S.W. Tam, "Materials Behavior in HTGR Environments", NUREG/CR-6824 and ANL-02/327, published July 2003.

(806) Kimball, O.F. and Plumblee, D.E. "Gas/Metal Interaction Studies in Simulated HTGR Helium." General Electric Company, Schenectady, NY. HTGR-85-064, June 1985.

(807) Warren, M.R. "Rapid Decarburization and Carburization in High Temperature Alloys in Impure Helium Environments." High Temperature Technology. Vol. 4, pp. 119-130, 1986.

(808) Graham, L.W. "Corrosion of Metallic Materials in HTR-Helium Environments." Journal of Nuclear Materials. Vol. 171, pp. 76-83, 1990.

(809) Kimball, O.F. "Thermal Stability and Environmental Compatibility of Inconel 617." High Temperature Metallic Materials for Gas-Cooled Reactors. IWGGCR-18, June, 1988.

(810) Quadakkers, W.J. and H. Schuster. "Corrosion of High Temperature Alloys in the Primary Circuit Helium of High Temperature Gas Cooled Reactors. Part I: Theoretical Background." Werkstoffe und Korrosion. Vol. 36, pp. 141-150, 1985.

(811) Quadakkers, W.J. "Corrosion of High Temperature Alloys in the Primary Circuit Helium of High Temperature Gas Cooled Reactors. Part II: Experimental Results." Werkstoffe und Korrosion. Vol. 36, pp. 335-347, 1985.
(812) Graham, L.W. "High Temperature Corrosion in Impure Helium Environments." High Temperature Technology. Vol. 3, pp. 3-14, 1985.
(813) Perez, F.J. and N.M. Ghoniem. "Chemical compatibility of SiC composite structures with fusion reactor helium coolant at high temperatures." Fusion Engineering and Design. Vol. 22, pp. 415-426, 1993.
(814) Christ, H.J., Schwanke, D., Uihlein, Th., and Sockel, H.G. "Mechanisms of High-Temperature Corrosion in Helium Containing Small Amounts of Impurities. I. Theoretical and Experimental Characterization of the Gas Phase" Oxidation of Metals. V30, 1988. pp. 1–26.

## 11.10 Vacuum Evaporation

(815) J.D. Whittenberger, "Tensile Properties and Structure of Several Superalloys after Long-Term Exposure to LiF and Vacuum at 1173K", Journal of Materials Engineering and Performance, V4(6), 1995. pp 657-673.
(816) J.D. Whittenberger, "Effect of Long-Term 1093K Exposure to Air or Vacuum on the Structure of Several Wrought Superalloys", Journal of Materials Engineering and Performance, V2(5), 1993. pp 745-758.
(817) D.T. Bourgette and J.E. McCoy, "A Study of the Vaporization and Creep-Rupture Behavior of Type 316 Stainless Steel", Transactions of the AIME, V59, 1966. pp 324-399.
(818) M. Shindo and T. Kondo, "Evaporation Behavior of Hastelloy-X Alloys in Simulated Very High Temperature Reactor Environments", Nuclear Technology, V66, 1984. pp 429-438.
(819) D.T. Bourgette, "Vaporization Phenomena of Haynes Alloy No. 25 to 1150C", ORNL-TM-1786, May 1967.
(820) D.T. Bourgette, "Evaporation of Iron-, Nickel-, and Cobalt-base Alloys at 760 to 980C in High Vacuums", ORNL-3677, November 1964.

## 11.11 Stress And Strain Levels For Biaxial Creep Specimens

(821) TA Gabriel, BL Bishop, FW Wiffen, " Calculated Irradiation Response of Materials using Fission Reactor Spectra", Oak Ridge National Laboratory, ORNL/TM-6361, August 1979.
(822) FW Wiffen, "Effect of Irradiation on Properties of Refractory Alloys with Emphasis on Space Power Reactor Applications", Proceedings Symp. On Refractory Alloy Technology for Space Nuclear Power Applications, CONF-8308130, ORNL, August 1983.
(823) M.M. Paxton, B.A. Chin, E.R. Gilbert and R.E. Nygren, "Comparison of the In-Reactor Creep of Selected Ferritic, Solid Solution Strengthened, and Precipitation Hardened Commercial Alloys", Journal of Nuclear Materials, 80 (1979) 144-151.
(824) K.E. Moore, R.G. Brengle, T.G. Parker, "Hastelloy X Cladding Materials Evaluation", SNAP Reactor, SNAP Program C-92b, AI-AEC-13083 (1973).
(825) J.F. Bates and R.W. Powell, "Irradiation-Induced Swelling in Commercial Alloys", Journal of Nuclear Materials, 102 (1981) pages 200-213.
(826) R.M. Boothby, "The microstructure of fast neutron irradiated Nimonic PE16", Journal of Nuclear Materials, vol. 230 (1996) pages 148-157.
(827) Nimonic alloy PE16, Special Metals Publication Number SMC0102 (2004) page 19.
(828) W.K. Appleby, D.W. Sandusky and U.E. Wolff, "Swelling Resistance of a High Nickel Alloy", Journal of Nuclear Materials, 43 (1972) pages 213-218.

(829) Metcalfe, A.G. and A.R. Stetson. "Interactions in Coated Refractory Metal Systems." Refractory Metal Alloys: Metallurgy and Technology. April, 1968.

(830) R.M. Boothby, "Modelling Grain Boundary Cavity Growth in Irradiated Nimonic PE16", Journal of Nuclear Materials 171 (1990) 215-222.

## 11.12 Refractory Metal Properties

(831) S.G. Frykman and S.C. Daniels, "Bibliography of Refractory Alloys for Space Nuclear Power Applications", ORNL/M-257/R1, issued August 1990.

(832) Creep-Rupture Data for the Refractory Metals to High Temperatures, J.B. Conway and P.N. Flagella, Gordon and Breach, Science Publishers, Inc., New York, NY, 1971.

(833) L.A. Horak and L.K. Egner, "Creep Properties of Nb-1Zr and Nb-1Zr-0.1C", ORNL-6809, published December 1994.

(834) Refractory Alloy Technology For Space Nuclear Power Applications, ed. R.H. Cooper Jr. and E.E. Hoffman, CONF-8308130, Office of Scientific and Technical Information, United States Department of Energy, January 1984.

(835) Armstrong, P. E., & Brown, H. L. "Dynamic Young's modulus measurements above 1000 °C on some pure polycrystalline metals and commercial graphites." AIME Trans., 230, 1964962-966.

(836) Bryan, W. A. "High-temperature strength stability of three forms of chemically vapor deposited tungsten." Journal of Vacuum Science Technology. 1974

(837) Buckman, R. W. (2004). Estimating Time to 1% Creep for ASTAR-811C and Mo-47.5Re Alloy (RMT3001728-06, dated November 8, 2004).

(838) Buckman, R.W. "Alloying of Refractory Metals," ASM International. 1988.

(839) Busby, J. T. "Update on tensile testing of refractory alloys" (dated July 25, 2005): Oak Ridge National Laboratory.

(840) Busby, J. T, et al. "Molybdenum-rhenium alloys for spacecraft reactor applications." Paper presented at the Space Nuclear Conference 2005, San Diego, CA.

(841) Carlen, J.-C. "Mill products and fabricated components in rhenium metal and rhenium rich alloys". Paper presented at the International Symposium on Rhenium and Rhenium Alloys, Orlando, FL. 1997

(842) Davis, J. "Pure Tungsten - Density, ITER Material Properties Handbook" Retrieved August 4, 2005, from http://aries.ucsd.edu/LIB/PROPS/ITER/AM01/AM01-3304.html

(843) Development of Dispersion Strengthened Tantalum Base Alloy. (Contract NAS 3-2542): Westinghouse Astronuclear Laboratory.

(844) El-Genk, M. S., & Tournier, J.-M. "A review of refractory metal alloys and mechanically alloyed-oxide dispersion strengthened steels for space nuclear power systems." Journal of Nuclear Materials, 340, 2005. pp. 93-112.

(845) Fischer, B., Freund, D., Carlen, J.-C., & Leonhardt, T. "Manufacture and properties of molybdenum-rhenium alloys." Paper presented at the PM2TEC 2000: 2000 International Conference on Powder Metallurgy & Particulate Materials, New York, NY. (2000, May 30 - June 3, 2000).

(846) Garner. "Swelling, irradiation creep and growth of pure rhenium irradiated with fast neutrons at 1030-1330°C." Journal of Nuclear Materials. 2000. pp. 283-287.

(847) Grossbeck, M. L., & Wiffen, F. W. "Swelling and Tensile Properties of EBR-II-Irradiated Tantalum Alloys for Space Reactor Applications." Space Nuclear Power Systems. 1985

(848) Jehn, H. A., & Schulze, K. K. "High-temperature gas-metal reactions of molybdenum and its alloys. In K." H. Miska (Ed.), Physical metallurgy and technology of molybdenum and its alloys (pp. 107-117). Greenwich, CT: AMAX Specialty Metals Corp1985.

(849) Klar, E.,et al. (Eds.). Metals Handbook Ninth Edition (Vol. 7: Powder Metallurgy). Metals Park, OH: American Society for Metals. 1984

(850) Lambert, J. B. (1990). Refractory Metals and Alloys. In Metals Handbook (10 ed., Vol. 2, pp. 557): ASM.

(851) Leonhardt, T., Carlen, J.-C., Buck, M., Brinkman, C. R., Ren, W., & Stevens, C. O. "Investigation of Mechanical Properties and Microstructures of Various Molybdenum-Rhenium Alloys". Paper presented at the Space Technology and Applications International Forum STAIF 1999.

(852) Lessmann, G. G., Gold, R. E., Arcella, F. G., & Reed, F. (1969). Material Considerations for Design of the Potassium Turboalternator (KTA) (WANL-TME-1889): Westinghouse Astronuclear Laboratory.

(853) Marschall, C. W., & Holden, F. C. " VIII: Fracture Toughness of Refractory Metals and Alloys". In R. W. Fountain, J. Malt & L. S. Richardson (Eds.), High Temperature Refractory Metals (Vol. 34, pp. 138). New York, NY: Gordon and Breach Science Publishers, Inc. 1964

(854) Matolich, J. "Swelling in Neutron Irradiated Tungsten and Tungsten-25 Percent Rhenium." Scripta Met., 1974. pp. 8.

(855) Paxton, M. M. "Nb-1Zr Pressurized Tube Creep Correlation at SP-100 Service Conditions" (WHC-SP-1014, dated September 1993): Westinghouse Hanford Company.

(856) Pionke, L. J., & Davis, J. W. "Technical Assessment of Niobium Alloys Data Base for Fusion Reactor Applications" (C00-4247-2, dated Aug. 1979): U.S. Department of Energy.

(857) Rhenium Alloys Inc. "Datasheet: Mechanical properties - Rhenium and rhenium alloys". Retrieved August 3, 2005, 2005, from http://www.rhenium.com/Properties/mechanical_properties.htm

(858) Rhenium Alloys Inc. "Datasheet: Mo-47.5 Re Molybdenum Rhenium Alloy, Annealed." Retrieved July 7, 2005, 2005, from http://www.matweb.com/search/SpecificMaterialPrint.asp?bassnum=MRMo31

(859) Rhenium Alloys Inc. "Datasheet: W-25 Re Tungsten Rhenium Alloy, Annealed." Retrieved July 7, 2005, 2005, from http://www.matweb.com/search/SpecificMaterialPrint.asp?bassnum=MWRe04

(860) Rhenium Alloys Inc. "Molybdenum-Rhenium Datasheet". Elyria, OH.

(861) Schmidt, F. F., & Ogden, H. R. "The Engineering Properties of Tantalum and Tantalum Alloys" (DMIC Report 189). Columbus, OH: Defense Metals Information Center. 1963

(862) Senor, D. J., & Horak, J. A. "Material property correlations for W-25Re, Mo-50Re and ASTAR-811C": Oak Ridge National Laboratory, unpublished report. 1989

(863) Sessler, J. G., & Weiss, V. (Eds.). Aerospace Structural Metals Handbook (Vol. IIA, Non-Ferrous Heat Resistant Alloys). West Lafayette, IN: CINDAS/USAF CRDA Handbooks Operation, Purdue University. 1967

(864) Sheffer, K. D., & Ebert, R. R. (1973). "Generation of Long Time Creep Data on Refractory Alloys at Elevated Temperatures" (NASA-CR-134481).

(865) Stephenson, R. L. The creep-rupture properties of some refractory metal alloys, II. The properties of the niobium-base alloys FS-85 and Cb-752 and their response to heat treatment. Journal of the Less-Common Metals, 15 1968, pp. 403-414.

(866) "Technical Publication on Nb and Nb-alloy Products". (2004). Wah Chang.

(867) Tietz, T. E., & Wilson, J. W. (). "Behavior and Properties of Refractory Metals:" Stanford University Press. 1965.

(868) Titran, R. H., & Hall, R. W. "High-temperature Creep Behavior of a Columbium Alloy, FS-85" (NASA TN D-2885). Washington, D.C.: National Aeronautics and Space Administration. 1965

(869) Torti, M. L. "Physical Properties and Fabrication Techniques for the Tantalum 10% Tungsten Alloy." Paper presented at the Metallurgical Society Conference, High Temperature Materials II, Cleveland, OH. , April 26-27, 1961

(870) Touloukian, Y. S. (Ed.). (1967). Thermophysical Properties of High Temperature Solid Materials (Vol. 1: Elements). New York: The Macmillan Company.
(871) Touloukian, Y. S. (Ed.). (1967). Thermophysical Properties of High Temperature Solid Materials (Vol. 2: Nonferrous Alloys, Part I: Nonferrous Binary Alloys). New York: The Macmillan Company.
(872) Touloukian, Y. S. (Ed.). (1970). Thermophysical Properties of Matter (Vol. 13: Thermal Expansion).
(873) Touloukian, Y. S. (Ed.). (1970). Thermophysical Properties of Matter (Vol. 7 & 8: Thermal Radiative Properties).
(874) Touloukian, Y. S., & Buyco, E. H. (Eds.). (1970). Thermophysical Properties of Matter (Vol. 4: Specific Heat, Metallic Elements and Alloys).
(875) Touloukian, Y. S., Powell, R. W., Ho, C. Y., & Klemens, P. G. (Eds.). (1970). Thermophysical Properties of Matter (Vol. 1: Thermal Conductivity, Metallic Elements and Alloys).
(876) Wiffen, F. W. "Effects of Irradiation on Properties of Refractory Alloys with Emphasis on Space Power Reactor Applications". Paper presented at the Symposium on Refractory Alloy Technology for Space Nuclear Power Applications, CONF-8308130, Oak Ridge, TN. (1983, August).
(877) Yih, S. W. H. (1979). Tungsten (W). In W. H. Cubberly (Ed.), ASM Handbook (9 ed., Vol. 2, pp. 713, 816-821). Metals Park, OH: American Society for Metals.

## 11.13 Coatings for Protection of Tantalum

(878) Raghunathan, S., et al. "Characterization of Ir/Re Duplex Coatings on Ta/10W Deposited by Pulsed Electrode Surfacing (PES)." Elevated Temperature Coatings: Science and Technology I (TMS), 1995.
(879) Kallup, C. and S.V. Castner. "Protecting Tantalum Alloy at 3500°F in Air." Refractory Metals and Alloys III: Applied Aspects. December, 1963.
(880) Falco, J.J. and M. Levy. "Alleviation of the Silicide Pest in a Coating for the Protection of Refractory Metals Against High-Temperature Oxidation." Journal of the Less-Common Metals. Vol. 20, pp. 291-297, 1970.
(881) Packer, C.M. and R.A. Perkins. "Development of a Fused Slurry Slicide Coating for the Protection of Tantalum Alloys." Journal of the Less-Common Metals. Vol. 37, pp. 361-378, 1974.
(882) Dzyadykevich, Y.V. and L.I. Kytskay. "Improving the Oxidation Protection of Niobium and Tantalum by the Use of Multilayer Coatings." JOM, Vol. 49, pp. 30-31, 1997.
(883) Wirkus, C.D. and D.R. Wilder. "Oxide grain growth and the Er2O3-Ta interface." Materials Science and Engineering. Vol. 30, pp. 89-91, 1977.
(884) Berkowitz-Mattuck, J.B. "Mechanisms of Oxidation of Ta-10W Alloy Coated with Tungsten Disilicide." Journal of the Electrochemical Society. Vol. 116, pp. 700-709, 1969.
(885) Hallowell, J.B., et al. "Silicide Coatings for Tantalum and Tantalum-Base Alloys." Refractory Metals and Alloys III: Applied Aspects. December, 1963.
(886) Levy, M. and J.J. Falco. "Oxidation Behavior of a Complex Disilicide/Tantalum-10 Tungsten Alloy System at Temperatures of 1700°F (927°C) to 2700°F (1482°C)." Journal of the Less-Common Metals. Vol. 27, pp. 143-162, 1972.
(887) International Harvester Company, San Diego, CA. "Development of Protective Coatings for Tantalum Base Alloys, Task II: Development of Technology Applicable to Coatings Used in the 3000 to 4000 F Temperature Range." Air Force Contract #AF 33(657)-11259, January 1965.
(888) General Telephone & Electronics Laboratories, Inc. "High Temperature Oxidation Resistant Coatings for Tantalum Base Alloys." Air Force Contract #AF 33(657)-7339, May 1962.

## 11.14 Iridium and Iridium/Rhenium coatings and CVD

(889) Reed, B.D. and S.J. Schneider. "Testing of Wrought Iridium/Chemical Vapor Deposition Rhenium Rocket." NASA Lewis Research Center. NASA/TM-107452, December 1996.

(890) Reed, B.D., et al. "Iridium-Coated Rhenium Radiation-Cooled Rockets." NASA Lewis Research Center. NASA/TM-107453, February 1997.

(891) Sink, D.A. "First Wall Coating Candidates for ICF Reactor Chambers Using Dry Wall Protection Only." Nuclear Technology/Fusion. Vol. 4, pp. 712-717, 1983.

(892) Ohriner, E.K. "Rhenium and Iridium." Rhenium and Rhenium Alloys (TMS), February 1997.

(893) Sayre, E.D., et al. "Development of Bonded Rhenium/Niobium-1%Zirconium Tubing for the SP100 Space Nuclear Reactor." Rhenium and Rhenium Alloys (TMS), February 1997.

(894) Tuffias, R.H., et al. "A History of Rhenium in High-Performance Bipropellant Rocket Engines." Rhenium and Rhenium Alloys (TMS), February 1997.

(895) Tuffias, R.H., et al. "State-Of-The-Art Fabrication Processes for Iridium/Rhenium Thrust Chambers." Rhenium and Rhenium Alloys (TMS), February 1997.

(896) Fang, C.S., et al. "Re/Ir/W and Os/Ir/W Alloy Coatings on Impregnated Tungsten Cathodes." Applied Surface Science, Vol. 33-34, pp. 1189-1199, 1988.

(897) Sherman, A.J., et al. "The Properties and Applications of Rhenium Produced by CVD." JOM, Vol. 43, pp 20-23, 1991.

(898) King, H.C., et al. "Chemical Vapor Infiltration of Rhenium." Chemical Vapor Deposition. Vol. 9, pp. 59-63, 2003.

(899) Ogura, Y., et al. "Metal Chloride Reduction Chemical Vapor Deposition for Ta, Mo and Ir Films." Japanese Journal of Applied Physics. Vol. 43, pp. L56-L59, 2004.

(900) Kim, K.T., et al. "Chemical Vapor Deposition (CVD) of Rhenium." Materials Letters. Vol. 12, pp. 43-46, 1991.

(901) Union Carbide Corporation, Parma, OH. "High Temperature Protective Coatings for Refractory Metals: Progress Report No. 3." NASA contract #NASw-1030, October 1965.

(902) Union Carbide Corporation, Parma, OH. "High Temperature Protective Coatings for Refractory Metals: Progress Report No. 1." NASA contract #NASw-1030, February 1965.

(903) Arnoult, W.J. and R.B. McLellan. "The Solubility of Carbon in Rhodium, Ruthenium, Iridium and Rhenium." Scripta Metallurgica. Vol. 6, pp. 1013-1018, 1972.

## 11.15 Protective Coatings for Ni-base Superalloys

(904) Goward, G.W., et al. "Progress in coatings for gas turbine airfoils." Surface and Coatings Technology. Vol. 108-109, pp. 73-79, 1998.

(905) Gedwill, M.A., et al. "A New Diffusion-Inhibited Oxidation-Resistant Coating for Superalloys." Thin Solid Films. Vol. 95, pp. 65-72, 1982.

(906) Itzhak, D., et al. "Silicon-Containing Coatings Produced by a Chemical Vapour Deposition Method on Nickel-Based Superalloys." Thin Solid Films. Vol. 73, pp. 379-384, 1980.

(907) Lindblad, N.R. "A Review of the Behavior of Aluminide-Coated Superalloys." Oxidation of Metals. Vol. 1, pp. 143-170, 1969.

(908) Godlewski, K. and Godlewska, E. "Effect of Chromium on the Protective Properties of Aluminide Coatings." Oxidation of Metals. Vol. 26, pp. 125-138, 1986.

(909) Jedlinski, J., et al. "The Influence of Implanted Yttrium and Cerium on the Protective Properties of a $\beta$-NiAl Coating on a Nickel-base Superalloy." Materials Science and Engineering. Vol. A121, pp. 539-543, 1989.

(910) Rhys-Jones, T.N. "Coatings for Blade and Vane Applications in Gas Turbines." Corrosion Science. Vol. 29, pp. 623-646, 1989.
(911) Srinivasan, V. "High-Temperature Corrosion and Erosion in Gas Turbine Engines – Where Do We Stand?" JOM. Vol. 46, pp. 34, 1994.
(912) Di Maggio, R., et al. "ZrO2-CeO2 Films as Protective Coatings Against Dry and Wet Corrosion of Metallic Alloys." Surface and Coatings Technology. Vol. 89, pp. 292-298, 1997.
(913) Cheruvu, N.S., et al. "The In-Service Degradation of Corrosion-Resistant Coatings." JOM. Vol. 48, pp. 34-38, 1996.
(914) Brady, M.P., et al. "Alloy Design Strategies for Promoting Protective Oxide-Scale Formation." JOM. Vol. 52, pp. 16-21, 2000.
(915) Xiang, Z.D. and P.K. Datta. "Formation of Hf- and W-Modified Aluminide Coatings on Nickel-base Superalloys by the Pack Cementation Process." Materials Science and Engineering. Vol. A363, pp. 185-192, 2003.
(916) Wang, Q.M., et al. "Hot Corrosion Behavior of AIP NiCoCrAlY(SiB) Coatings on Nickel Base Superalloys." Surface and Coatings Technology. Vol. 186, pp. 389-397, 2004.
(917) Khajavi, M.R., et al. "Aluminide Coatings for Nickel Based Superalloys." Surface Engineering. Vol. 20, pp. 261-265, 2004.
(918) Sidhu, T.S., et al. "Hot Corrosion of Some Superalloys and Role of High-Velocity Oxy-Fuel Spray Coatings – A Review." Surface and Coatings Technology. Vol 198, pp. 441-446, 2005.
(919) Haynes, J.A., et al. "High-Temperature Diffusion Barriers for Protective Coatings." Surface and Coatings Technology. Vol. 188-189, pp. 153-157, 2004.
(920) Benoist, J., et al. "Microstructure of Pt-Modified Aluminide Coatings on Ni-Based Superalloys." Surface and Coatings Technology. Vol. 182, pp. 14-23, 2004.
(921) Shikama, T., et al. "Silicon Oxide Coatings as Protection Against Corrosion." Thin Solid Films. Vol. 145, pp. 89-98, 1986.
(922) Tortorelli, P.F. and M.P. Brady. "Alloy Design Approaches for High-Temperature Oxidation Resistance." JOM. Vol. 52, pp. 15, 2000.
(923) Amano, T., et al. "Oxidation of Alumina-Forming Alloys with Small Amounts of Sulfur and Reactive Elements (Y,Hf) at 1273 K." Transactions of the Materials Research Society of Japan. Vol. 27, pp. 735-738, 2002.
(924) Amano, T., et al. "Oxide Adherence of Fe-20Cr-4Al Alloys with Small Amounts of Sulfur and Reactive Elements (Y,Hf)." Materials at High Temperatures. Vol. 17, pp. 117-124, 2000.
(925) Allam, I.M., et al. "The Role of Active Elements and Oxide Dispersions in the Development of Oxidation-Resistant Alloys and Coatings." LBL. LBL-9516;CONF-781093-3, 1979.
(926) Nicholls, J.R. "Designing Oxidation-Resistant Coatings." JOM. Vol. 52, pp. 28-35, 2000.
(927) Smialek, J.L. "Maintaining Adhesion of Protective Al2O3 Scales." JOM. Vol. 52, pp. 22-25, 2000.
(928) Birks, N., et al. "Forming Continuous Alumina Scales to Protect Superalloys." JOM. Vol. 46, pp. 42-46, 1994.
(929) Conner, J.A. and W.B. Connor. "Ranking Protective Coatings: Laboratory vs. Field Experience." JOM. Vol. 46, pp. 35-38, 1994.
(930) Haynes, J.A., et al. "Comparison of Thermal Expansion and Oxidation Behavior of Various High-Temperature Coating Materials and Superalloys." Materials at High Temperature. Vol. 21, pp. 87-94, 2004.

## 11.16 Protective coatings for Silicon Carbide

(931) More, K.L., et al. "Observations of Accelerated Silicon Carbide Recession by Oxidation at High Water-Vapor Pressures." Journal of the American Ceramic Society. Vol. 83, pp. 211-213, 2000.
(932) Lee, K.N. and R.A. Miller. "Development and Environmental Durability of Mullite and Mullite/YSZ Dual Layer Coatings for SiC and Si3N4 Ceramics." Surface and Coatings Technology. Vol. 86-87, pp. 142-148, 1996.
(933) Lee, K.N. and R.A. Miller. "Oxidation Behavior of Mullite-Coated SiC and SiC/SiC Composites under Thermal Cycling between Room Temperature and 1200°-1400°C." Journal of the American Ceramic Society. Vol. 79, pp. 620-626, 1996.
(934) Choy, K.L., et al. "Effect of TiB2, TiC and TiN Protective Coatings on Tensile Strength and Fracture Behaviour of SiC Monofilament Fibres." Composites. Vol. 26, pp. 531-539, 1995.
(935) Sarin, V., et al. "Corrosion Protection of SiC-Based Ceramics with CVD Mullite Coatings." ORNL. ORNL/Sub/94-SS110/01, 1996.
(936) Xu, Y., et al. "Oxidation Behavior and Mechanical Properties of C/SiC Composites with Si-MoSi2 Oxidation Protection Coating." Journal of Materials Science. Vol. 34, pp. 6009-6014, 1999.
(937) Russell, L.C., et al. "Effects of Mullite/YSZ Coatings on the Performance of SiC/SiC Composite Combustion Liners." Ceramic Engineering and Science Proceedings. Vol. 21, pp. 243-250, 2000.
(938) Varadarajan, S., et al. "Mullite Interfacial Coatings for SiC Fibers." Surface and Coatings Technology. Vol. 139, pp. 153-160, 2001.
(939) Lee, K.N. "Current Status of Environmental Barrier Coatings for Si-Based Ceramics." Surface and Coatings Technology. Vol. 133-134, pp. 1-7, 2000.
(940) Zhu, D., et al. "Advanced Environmental Barrier Coatings Development for Si-Based Ceramics." NASA Glenn Research Center. NASA/TM-2005-213444. March 2005.
(941) Vlasova, M., et al. "SiC Particles Coated by Chromium Silicides." Journal of Materials Synthesis and Processing. Vol. 10, pp. 67-74, 2002.
(942) Zhu, D., et al. "Thermal Conductivity and Thermal Gradient Cyclic Behavior of Refractory Silicate Coatings on SiC/SiC Ceramic Matrix Composites." NASA Glenn Research Center. NASA/TM-2001-210824. April 2001.
(943) More, K.L., et al. "Evaluating Environmental Barrier Coatings on Ceramic Matrix Composites After Engine and Laboratory Exposures." Proceedings of GT2002 ASME Turbo Expo. GT-2002-30630, June 2002.
(944) More, K.L., et al. "Verification of an EBC's Protective Capability by First-Stage Evaluation in a High-Temperature, High-Pressure Furnace." Proceedings of GT2003 ASME Turbo Expo. GT-2003-38923, June 2003.
(945) Lee, K.N., et al. "Rare Earth Silicate Environmental Barrier Coatings for SiC/SiC Composites and Si3N4 Ceramics." Journal of the European Ceramic Society. Vol. 25, pp. 1705-1715, 2005.
(946) Lee, K.N., et al. "Upper Temperature Limit of Environmental Barrier Coatings Based on Mullite and BSAS." Journal of the American Ceramic Society. Vol. 86, pp. 1299-1306, 2003.

## 11.17 Protective Coatings for Niobium (Columbium)

(947) Jones, K.D., et al. "Reaction Layer Structure of Silicide Coatings on Niobium Alloys." Refractory Metals: Extraction, Processing and Applications. February, 1991.
(948) Pettyjohn, R.R. "The Evaluation of Oxidation Protective Coatings for the Columbium Alloy FS-82." Refractory Metals and Alloys III: Applied Aspects. December, 1963.

(949) Liu, Y., et al. "Processing and Oxidation Behavior of Nb-Si-B Intermetallics." Contract # W-7405-Eng-82, September, 2004.
(950) Stein, B.A. and W.B. Lisagor. "Preliminary Results of a Study of 12 Oxidation-Resistant Coatings for Cb-10Ti-5Zr Columbium-Alloy Sheet" NASA TMX-51973, August, 1964.
(951) McDonnell Douglas Astronautics Company, St. Louis, MO. "Fused Slurry Silicide Coatings for Columbium Alloy Reentry Heat Shields. Volume II: Experimental and Coating Process Details." NASA CR 134483, August 1973.
(952) Carlson, R.G. "Oxidation Resistance of Aluminum Dip Coated (Aldico) Columbium Alloys." Columbium Metallurgy. June, 1960.
(953) Moore, V.S. and A.R. Stetson. "Progress in the Coating of Refractory Metal Foil." Refractory Metals and Alloys III: Applied Aspects. December, 1963.
(954) Levine, S.R. and S.J. Grisaffe. "Exploration of Alloy Surface and Slurry Modification to Improve Oxidation Life of Fused Silicide Coated Niobium Alloys." NASA Lewis Research Center, NASA/TM X-68052, May 1972.
(955) Rummler, D.R., et al. "Coated Refractory Metals for High-Temperature Structural Components." Refractory Metals and Alloys III: Applied Aspects. December, 1963.
(956) Levine, S.R. and S.J. Grisaffe. "Coated Columbium Thermal Protection Systems – An Assessment of Technological Readiness." NASA Lewis Research Center, NASA/TM X-2858, August 1973.

## 11.18 Thermodynamic and Kinetic Properties of Iridiuim

(957) Jehn, H. "High Temperature Behaviour of Platinum Group Metals in Oxidizing Atmospheres." Journal of the Less-Common Metals. Vol. 100, pp. 321-339, 1984.
(958) Krier, C.A. and R.I. Jaffee. "Oxidation of the Platinum-Group Metals." Journal of the Less-Common Metals. Vol. 5, pp. 411-431, 1963.
(959) Olivei, A. "Methods for Studying Oxygen-Platinum Metals Interactions." Journal of the Less-Common Metals. Vol. 29. pp. 11-23, 1972.
(960) Wimber, R.T. and H.G. Kraus. "Oxidation of Iridium." Metallurgical Transactions. Vol. 5, pp. 1565-1572, 1974.
(961) Wimber, R.T., et al. "Kinetics of Evaporation/Oxidation of Iridium." Metallurgical Transactions A. Vol. 8A, pp. 193-199, 1977.
(962) Fromm, E. "Reduction of Metal Evaporation Losses by Inert Gas Atmospheres." Metallurgical Transactions A. Vol. 9A, pp. 1835-1838, 1978.
(963) Bell, W.E. and M. Tagami. "Study of Gaseous Oxides, Chloride, and Oxychloride of Iridium." The Journal of Physical Chemistry. Vol. 70, pp. 640-646, 1966.
(964) Kulikov, I.S. "Oxydation des Iridiums." Zeitschrift fur Metallkunde. Vol. 72, pp. 525-529, 1981.
(965) Alcock, C.B. "The Gaseous Oxides of the Platinum Metals." Platinum Metals Review. Vol. 5, pp. 134-139, 1961.
(966) Jehn, H., et al. "Iridium Losses During Oxidation." Platinum Metals Review. Vol. 22, pp. 92-97, 1978.

## 11.19 Protective Coatings for Molybdenum

(967) Analytical Services and Materials, Inc., Hampton, VA. "Oxidation and Emittance Studies of Coated Mo-Re." NASA CR 201753, October, 1997.
(968) Clark, R.K and T.A. Wallace. "Oxidation Performance of Platinum-Clad Mo-47Re Alloy." NASA Langley Research Center, NASA/TM-4559, June 1994.
(969) Walters, R.P. and Covino, B.S. "Evaluation of High-Temperature Diffusion Barriers for the Pt-Mo System." Metallurgical Transactions A. Vol. 19A, pp. 2163-2170, 1988.

(970) Couch, D.E., et al. "Protection of Molybdenum from Oxidation at Elevated Temperatures." Journal of the Electrochemical Society. Vol. 105, pp. 450-456, 1958.
(971) Passmore, E.M., et al. "Selection of Diffusion Barriers for Use in Protective Coatings Systems for Tungsten and Molybdenum." Refractory Metals and Alloys III: Applied Aspects. December, 1963.
(972) Meyer, M.K. and M. Akinc. "Oxidation Behavior of Boron-Modified Mo5Si3 at 800°-1300°C." Journal of the American Ceramic Society. Vol. 79, pp. 938-944, 1996.
(973) Torri, P. "Oxidation of Nanocrystalline Mo-Si-N and Nanolayered Mo-Si-N/SiC Coatings." Journal of Materials Research. Vol. 14, pp. 3552-3558, 1999.
(974) Tang, J.E., et al. "An Investigation of the Microstructure in the Pest Oxide of a MoSi2-Based Composite." Ceramic Engineering and Science Proceedings. Vol. 21, pp. 477-484, 2000.
(975) Govindarajan, S., et al. "Synthesis and Characterization of a Diffusion Barrier Layer for Molybdenum." Journal of Advanced Materials. Vol. 31, pp. 23-33, 1999.

## 11.20 Corrosion of Silicon Carbide

(976) More, K.L., et al. "Observations of Accelerated Silicon Carbide Recession by Oxidation at High Water-Vapor Pressures." Journal of the American Ceramic Society. Vol. 83, pp. 211-213, 2000.
(977) Tortorelli, P.F. and K.L. More. "Effects of High Water-Vapor Pressure on Oxidation of SiC at 1200 °C." Journal of the American Ceramic Society. Vol. 86, pp. 1249-1255, 2003.
(978) More, K.L., et al. "Exposure of Ceramics and Ceramic Matrix Composites in Simulated and Actual Combustor Environments." Proceedings of 1999 ASME Turbo Expo: Power for Land, Sea, & Air. 99-GT-292, June 1999.
(979) Smialek, J.L., et al. "SiC and Si3N4 Recession Due to SiO2 Scale Volatility Under Combustor Conditions." NASA Glenn Research Center, NASA/TP-1999-208696, July 1999.
(980) Nguyen, Q.N., et al. "Oxidation of Ultra High Temperature Ceramics in Water Vapor." NASA Glenn Research Center, NASA/TM-2004-212923, April 2004.
(981) More, K.L., et al. "Effects of High Water Vapor Pressures on the Oxidation of SiC-Based Fiber-Reinforced Composites." Materials Science Forum. Vol. 369-372, pp. 385-394, 2001.
(982) Lie, L.N., et al. "Thermal Oxidation of Silicides." Journal of Applied Physics. Vol.56, pp. 2127-2132, 1984.
(983) d'Heurle, F.M., "Diffusion-Reaction: The Oxidation of Silicides in Electronics and Elsewhere." Journal De Physique III. Vol. 5, pp. 1707-1728, 1995.
(984) Radhakrishnan, R., et al. "The Reactive Processing of Silicides." JOM. Vol. 49, pp. 41-45, 1997.

## 11.21 Silicon Carbide Papers

(985) Snead, LL and SJ Zinkle. "Structural relaxation in amorphous silicon carbide." Nuclear Instruments and Methods in Physics Research. V B 191, 2002. pp 497-503.
(986) Snead, LL and TD Burchell. "Stored Energy In Irradiated Silicon Carbide." Oak Ridge National Laboratory.
(987) Nikolaenko, VA and AV Subbotin. "Radiation Annealing of Silicon Carbide Irradiated in a Bor-60 Reactor." Atomic Energy V97 No. 4. 2004. pp 701-706.
(988) Senn, RL., et al. "Design Operation, and Initial Results from a Series of Graphite Creep Irradiation Experiments." Journal of Nuclear Materials. V65. pp 96-106.
(989) Lebedev, IG., et al. "Radiation-Induced Change in the Properties of Isotropic Structural Graphite." Atomic Energy. V93 No. 1. 2002. pp 589-594.

(990) Maruyama, T., et al. “Change in physical properties of high density isotropic graphites irradiated in the “JOYO” fast reactor.” Journal of Nuclear Materials. V225, 1995. pp 267-272.

(991) Kondo, S., “History and Perspective of Fast Breeder Reactor Development in Japan.” Energy. V23 No. 7/8. 1998. pp 619-627

(992) Suzuki, T., et al. “Recovery Behavior in Neutron Irradiated β-SiC.” Journal of Nuclear Materials. V149, 1987. pp 334-340.

(993) Lewinsohn, C.A., et al. “Irradiation-enhanced creep in SiC data summary and planned experiments.” Journal of Nuclear Materials. V253, 1998, pp 36-46.

(994) Zhu, S., et al. “Tensile Creep Behavior of a SiC-Fiber/SiC Composite at Elevated Temperatures.” Composites Science and Technology. V57, 1997. pp 1629-1637.

(995) Morscher, G.N., et al. “Comparison of Bend Stress Relaxation and Tensile Creep of CVD SiC Fibers.” Journal of the American Ceramic. Society. V78 (12), 1995. pp 3244-3252.

(996) Zhu, Z. and P. Jung. “Dimensional changes of A12O3 and SiC, proton irradiated under tensile stress.” Journal of Nuclear Materials. V212-215, 1994. pp 1081-1086.

(997) El-Azab, A. and NM Ghoniem. “Phenomenological Inelastic Constitutive Equations for SiC and SiC Fibers Under Irradiation.” Fusion Technology. V26 1994. pp 1250-1264.

(998) DiCarlo, J.A. “Creep of Chemically Vapor Deposited SiC Fibers.” Journal of Materials Science. V21 1986. pp 217-224.

(999) Carter, Jr., C.H., et al. “Kinetics and Mechanisms of High-Temperature Creep in Silicon Carbide: II, Chemically Vapor Deposited.” Journal of the American Ceramic Society. V67, No. 11 1984. pp 733-740.

(1000) Carter, J.R., C.H., et al. “Kinetics and Mechanisms of High-Temperature Creep in Silicon carbide: I, Reaction-Bonded.” Journal of the American Ceramic Society. V67, No. 6 1984. pp 409-417.

(1001) Veltri, Richard D., et al. “Chemical Vapor Deposited SiC Matrix Composites.” Journal of American Ceramic Society. V72(3), 1989. pp 478-480

(1002) Riccardi, B., et al. “Issues and advances in SiCf/SiC composites development for fusion reactors.” Journal of Nuclear Materials. V329-333, 2004. pp 56-65

(1003) Hollenberg, G.W., et al. “The effect of irradiation on the stability and properties of monolithic silicon carbide and SiCf/SiC composites up to 25 dpa.” Journal of Nuclear Materials. V219, 1995. pp 70-86.

(1004) Lewinsohn, C.A., et al. “Time-dependent failure mechanisms in silicon carbide composites for fusion energy applications.” Journal of Nuclear Materials. V283-287, 2000. pp 584-587.

(1005) Veltri, Richard D., et al. “Chemical Vapor Deposited SiC Matrix Composites.” Journal American Ceramic Society. V72 (3), 1989. pp 478-480.

(1006) Riccardi, B., et al. “Issues and advances in SiCf/SiC composites development for fusion reactors.” Journal of Nuclear Materials. V329-333, 2004. pp 54-65.

(1007) Hollenberg, G.W., et al. “The effect of irradiation on the stability and properties of monolithic silicon carbide and SiCf/SiC composites up to 25 dpa.” Journal of Nuclear materials. V219, 1955. pp 70-86.

(1008) Lewinsohn, C.A., et al. “Time-dependent failure mechanisms in silicon carbide composites for fusion energy applications.” The Journal of Nuclear Materials. V283-287, 2000. pp 584-587.

(1009) Scholz, R., G.E. Youngblood. “Irradiation creep of advanced silicon carbide fibers.” Journal of nuclear Materials. V283-287 2000. pp 372-375.

(1010) Lewinsohn, Charles A., et al. “High-Temperature Creep and Microstructural Evolution of Chemically Vapor-Deposited Silicon Carbide Fibers.” Journal American Ceramic Society. V82 (2) 1999. pp 407-413.

(1011) Katoh, Yutai, et al. "Microstructural Development in Cubic Silicon Carbide during Irradiation at Elevated Temperatures." Journal of Nuclear Materials Special Issue for 2005 TMS Meeting.
(1012) Shoji, Satoko, et al. "Creep Behavior of SiC Fibers at High Temperatures Using the BSR Method." Key Engineering Materials. V247 2003. pp 187-190.
(1013) Price, RJ. "Properties of Silicon Carbide For Nuclear Fuel Particle Coatings." Nuclear Technology. V35, 1977. pp 320-336.
(1014) Holmes, John W. "A Technique for Tensile Fatigue and Creep Testing of Fiber-Reinforced Ceramics." Journal of Composite Material. V26 No 61, 1992. pp 916-933.
(1015) Nogami, S., et al. "Analysis of possible deformation mechanisms in helium-ion irradiated SiC." Journal of Nuclear Materials. V307-311, 2002. pp 1178-1182.
(1016) Pan, Yi and Joao L. Baptista. "Chemical Instability of Silicon Carbide in the Presence of Transition Metals." Journal American Ceramic Society. V79 (8), 1996. pp 2017-2026.
(1017) Snead, LL, et al. "Measurement of the effect of radiation damage to ceramic composite interfacial strength." Journal of Nuclear Materials. V191-194, 1992. pp 566-570.
(1018) Matthews, RB. "Irradiation Damage in Reaction-Bonded Silicon Carbide." Journal of Nuclear Materials. V51 1974. pp 203-208.
(1019) El-Azaz, A. and NM Ghoniem. "Phenomenological Inelastic Constitutive Equations for SiC and SiC Fibers under Irradiation." Fusion Technology. V26, 1994. pp 1250-1264.
(1020) Huang, Hanchen and Nasr Ghoniem. "A swelling model for stoichiometric SiC at temperatures below 1000°C under neutron irradiation." Journal of Nuclear Materials. V250 1997. pp 192-199.
(1021) Snead, LL, et al. "Radiation induced microstructure and mechanical property evolution of SiC/C/SiC composite materials." Journal of Nuclear Materials. V191-194, 1992. pp 560-565.
(1022) Snead, Lance L. "Development of Silicon Carbide Composites for Fusion." Fusion Technology. V24, 1993. pp 65-82.
(1023) Scholz, R. "Light ion irradiation creep of SiC fibers in torsion." Journal of Nuclear Materials. V258-263, 1998. pp 1533-1539.
(1024) Chuang, Tze-jer, et al. "Steady-State Creep Behavior of Si-SiC C-Rings." Journal of American Ceramic Society. V74(10), 1991. pp 2531-2537.
(1025) Begley, MR, et al. "Time Dependent Crack Growth in Ceramic Matrix Composites with Creeping Fibers." Acta Metallurgica Material. V43, No 11, 1995. pp 3927-3936.
(1026) Morscher, Gregory N. and James A. DiCarlo. "A Simple Test for Thermomechanical Evaluation of Ceramic Fibers." Journal of American Ceramic Society. V75(1), 1992. pp 136-140.
(1027) Scholz, R. " Deuteron irradiation creep of chemically vapor deposited silicon carbide fibers." Journal of Nuclear Materials. V254m 1888. pp 74-77.
(1028) Yun, Hee Mann and James A. DiCarlo. "Comparison of the Tensile, Creep, and Rupture Strength Properties of Stoichiometric SiC Fibers." The American Ceramic Society. V20(3), 1999. pp 259-272.
(1029) Raj, R. and MF Ashby "On Grain Boundary Sliding and Diffusional Creep." Metallurgical Transactions. "V2, April 1971-1113.
(1030) Serizawa, H., et al. "High Temperature Properties and Creep Resistance of Near-Stoichiometric SiC Fibers." Ceramic Engineering & Science Procedures. V20 (4), 1999. pp 443-449.
(1031) Zhu, S., et al . "Monotonic tension, fatigue and creep behavior of SiC-fiber-reinforced SiC-matrix composites: a review." Composites Science and Technology. V59, 1999. pp 833-851.
(1032) Youngblood, G.E., et al. "Technique For Measuring Irradiation Creep In Polycrystalline SiC Fibers." Fusion National Semiannual Progress Report. June 1996.

(1033) Popper, P. "The Preparation of Dense Self-Bonded Silicon Carbide." Special Ceramics. 1960, pp 209-219.
(1034) Adkins, Carol. "Zirconium Carbide CVD and Diffusion Coatings." Sandia National Laboratories. T54.2-01, February, 1991.
(1035) Gulf General Atomic Inc. John Jay Hopkins Laboratory for Pure and Applied Science, San Diego, CA. "Structure and Properties of Pyrolytic Silicon Carbide." US Atomic Energy Commission Contract No. AT(04-3)-167, Project Agreement No. 17, February, 1969.
(1036) General Electric Company, Breeder Reactor Development Operation, Sunnyvale, CA. "Fuel Swelling-Fast Reactor Mixed-Oxide Fuels." Contract No. AT(04-3)-189, Project Agreement No. 10.
(1037) Ashford, JP. "The Bursting Strength of Self-Bonded Silicon Carbide Tubes and the Influence of Mechanical Damage." Proceedings of the Fourth Symposium on Special Ceramics held by British Ceramics Research Assoc., July, 1967. pp 173-189.
(1038) Scholz, R., H. Pasic. "Light Ion Irradiation Creep of SCS-6 Silicon Carbide Fibers in the Temperature Range 450-1100°C." Materials Research Society Symposium. V540. 1999.
(1039) Chubb, W., et al. The Relationship of Mechanical Properties to the Swelling of Oxide Fuels at High Temperatures." Proceedings of the Conference on Fast Reactor Fuel Element Technology. April 13-15, 1971.

## 11.22 Cesium and Tellurium Compatibility

(1040) Pulham, R.J. and M.W. Richards, "Chemical reactions between caesium, tellurium and oxygen with fast breeder reactor cladding alloys, Part II – the corrosion by caesium-oxygen mixtures," Journal of Nuclear Materials, V172 Issue 1, June 1990, pp 47-53.
(1041) Aitken EA, et al. 1974 "Out of pile investigations of fission product-cladding reactions in fast reactor fuel pins." In Behavior and Chemical State of Irradiated Ceramic Fuels, edited by C.N. Welsh. Austria: IAEA
(1042) Hofmann P and O. Gotzmann, 1974 "Chemical interactions of fission products with stainless steel claddings." In Behavior and Chemical State of Irradiated Ceramic Fuels, edited by C.N. Welsh. Austria: IAEA
(1043) Ohse, R.W. and M. Schleachter, 1974 "The role of cesium in chemical interaction of austenitic stainless steels with uranium plutonium oxide fuel." In Behavior and Chemical State of Irradiated Ceramic Fuels, edited by C.N. Welsh. Austria: IAEA
(1044) Bose DK, Sundaresan M, Tangri RP, Kalyanaraman R, "Some thermomechanical studies of cesium urinate, molybdate and chromate" Journal of Nuclear Materials, V130, 1985, pp 122-125.
(1045) Rosa, F and P.S. Maiya, "Degradation in the biaxial stress-to-failure properties of type 316 stainless steel exposed to cesium oxides," Journal of Nuclear Materials, V56, 1975, pp 136-144.
(1046) Maiya, P.S. and D.E. Bush, "Grain Boundary Penetration of Austenitic Stainless Steel by Cesium Oxide," Journal of Nuclear Materials, V44, 1972, pp 96-98.
(1047) Antill J.E., et al. "Corrosion of Stainless Steel in the Presence of Cesium" Journal of Nuclear Materials, V56, 1975, pp 47-60.
(1048) Pulham, R.J., et al. 1995 "Caesium and its mixtures: Their Chemical Reactions with Alloys of Transition Metals Used to Clad Reactor Fuels." In Liquid Metal Systems: materials behavior and physical chemistry in liquid metal systems, edited by HU Borgstedt. Plenum Press, New York
(1049) Pulham, R.J., et al. 1988. "Reactions between caesium / tellurium mixtures and advanced clad alloys under a constant pressure of oxygen." In Materials for Nuclear Reactor Core Applications Volume 2, Thomas Telford House, London

(1050) Khanna, A.S. and J.B. Gnanamoorthy, "The Interaction of Liquid Tellurium and Tellurium Oxide with Stainless Steels." Transactions of the Indian Institute of Metals, Volume 39, No. 2, April 1986, pp 137-146

(1051) Arima, T., et al. "Reaction of modified SS316 with tellurium under low oxygen potentials." Corrosion Science, Volume 45, 2003, pp 1757-1766

(1052) Maiya, P.S. and D.S. Bush, "Grain Boundary Penetration of Type 316 Stainless Steel Exposed to Cesium or Cesium and Tellurium." Metallurgical Transactions A, Volume 6A, Feb 1975 pp409 -415

(1053) Faghri, Amir, 1998. Heat Pipe Science and Technology. Washington DC: Taylor and Francis

## 11.23 Haynes 230 Cracking

(1054) Seelay, R.R., Srivastava, S.K. "The Effect of Long-Term Thermal Exposure in the Mechanical Properties of Four Modern High-Temperature Nickel-Based Alloys" Long Term Stability of High Temperature Materials (as held at the 1999 TMS Annual Meeting) San Diego, CA, 1999, pp 93-108.

(1055) Lu, Y., Chen, L., et al. "Elevated temperature crack growth behavior of nickel base Haynes 230 alloy at 927°C", Proceeding of the TMS Fall Meeting, 2002, p123-133.

(1056) Lu, Y., et al. "Fracture modes of Haynes 230 alloy during fatigue-crack-growth at room temperature and elevated temperatures," Materials Science and Engineering A, Volume 397, 2005, pp. 122-131.

## 11.24 Silicon Carbide Composite Testing

(1057) Iseki, T., "Effects of Neutron Irradiation and Subsequent Annealing on Strength and toughness of SiC Ceramics." Journal of Nuclear Materials. V170, 1990. pp 95-100.

(1058) DiCarlo, James A., "Creep Limitations of Current Polycrystalline Ceramic Fibers." NASA Lewis Research Center Composites Science and Technology. V51, 1994. pp 213-222.

(1059) Wong, J., et al. "The threshold energy for defect production in SiC: a molecular dynamics study." Journal of Nuclear Materials. V212-215, 1994. pp 143-147.

(1060) Sargent, P.M., and M.F. Ashby. "Deformation-Mechanism Maps for Silicon Carbide." Script METALLURGICA. V17, 1983. pp 951-957.

(1061) Youngblood, G.E. "Irradiation Creep of Advanced SIC-Based Fibers." Ceramic Engineering & Science Procedure. V19, No 4, 1998. pp 341-346.

(1062) Li, J. "Atomistic Modeling of Finite-Temperature Properties of Crystalline □-SiC: II. Thermal Conductivity and Effects of Point Defects." Department of Nuclear Engineering Massachusetts Institute of Technology. October 1997.

(1063) Youngblood, G.E., et al. "Creep Behavior For Advanced Polycrystalline SIC Fibers." Fusion Materials Semi-annual Progress Report. DOE/ER-0313/22, June 1997. pp 81-86.

(1064) D'Eye, RWM., "The Development of Silicon Carbide Clad Fuel Pins for Advanced Gas-Cooled Reactors." United Kingdom Atomic Energy Authority, Reactor Fuel Element Laboratories. SM-111/35

(1065) Kennedy, P. and Shennan, J.V. "Refel® Silicon Carbide; The Development of a Ceramic for a Nuclear Engineering Application." TRG REPORT 2627(s)

## 11.25 Material Irradiation Effects Papers

(1066) Olander, D.R. "Fundamental Aspects of Nuclear Reactor Fuel Elements." TID-26711-P1, UC Berkeley, 1976.

(1067) Garner, F.A., et al. "Swelling, irradiation creep and growth of pure rhenium irradiated with fast neutrons at 1030-1330oC." Journal of Nuclear Materials. V283&287, 2000. pp 380-385.
(1068) Heinisch, H.L. "Correlation of mechanical property changes in neutron-irradiated pressure vessel steels on the basis of spectral effects." Journal of Nuclear Materials. V178, 1991. pp 19-26.
(1069) Brager, H.R. and F.A. Garner. "Microstructural and microchemical comparisons of AISI 316 irradiated in HFIR and EBR-II." Journal of Nuclear Materials. V117, 1983. pp159-176.
(1070) Makenas, B.J. "Swelling of 316 stainless steel and D9 cladding in FFTF." Radiation-Induced Changes in Microstructure: 13th International Symposium (Part I), ASTM STP 955, pp 146-153, 1987.
(1071) Neustroev, V.S., et al. "Temperature-shift of void swelling observed in annealed Fe-18Cr-10Ni-Ti stainless steel irradiated in the reflector region of BOR-60." Effects of Radiation on Materials: 19th International Symposium, ASTM STP 1366, pp 792-800, 2000.
(1072) Garner, F.A., and J.F. Stubbins. "Saturation of swelling in neutron-irradiated molybdenum and its dependence on irradiation temperature and starting microstructural state." Journal of Nuclear Materials. V212-215, 1994. pp 1298-1302.
(1073) Laidler, Homes, and Bennett, "US Programs on Reference and Advanced Cladding/Duct Materials." International Conference; Radiation Effects in Breeder Reactor Structural Materials. Scottsdale, AZ, pp 41-52, 1977.
(1074) LeNaour, F., et al. "Swelling and Microstructure of Neutron Irradiated Inconel 706." Materials for Nuclear Reactor Core Application. BNES, London, pp. 211-217, 1987.
(1075) Sherwood, D.J., et al. "Development of materials for fast reactor fuel assemblies." Nuclear Technology. V78, 1987.
(1076) Fujiwara, M. et al. "Development of Modified Type 316 stainless steel for fast breeder reactor fuel cladding tubes." Radiation-Induced Changes in Microstructure: 13th International Symposium (Part I), ASTM STP 955, pp. 127-145, 1987.
(1077) Johnston, W.G., et al. "The depth distribution of void swelling produced by 5MeV Ni ions." Journal of Nuclear Materials. V62, 1976. pp 167-180.
(1078) Itaki, T., et al. "Effects of alloying elements on mechanical and swelling properties of type 316 stainless steel and advanced materials for fast reactor application." Materials for nuclear reactor core application. BNES, London, pp. 203-210, 1987.
(1079) Garner, F. A., Toloczko, M. B., Sencer, B. H., "Comparison of swelling and irradiation creep behavior of fccaustenitic and bcc-ferritic/martensitic alloys at high neutron exposure." Journal of Nuclear Materials. V276, 2000. pp 123-142.
(1080) Katoh, Y., et al. "Swelling and dislocation evolution simple ferritic alloys irradiated to high fluence in FFTF/MOTA." Journal of Nuclear Materials. V225, 1995. pp 154-162.
(1081) Kittel, J.H., et al. "History of fast reactor fuel development." Journal of Nuclear Materials, V204, 1993. pp 1-13.
(1082) Garner, FA, "Application of high fluence fast reactor data to fusion-relevant materials problems." Journal of Nuclear Materials. V133&134, 1985. pp. 113-118.
(1083) Tokiwai, M., et al. "Development of new ferritic steels as cladding material for metallic fuel fast breeder reactor." Journal of Nuclear Materials. V204, 1993. pp 1-13.
(1084) Ukai, S., and M. Fujiwara, "Perspective of ODS alloys application in nuclear environments." Journal of Nuclear Materials. V307-311, 2. pp 749-757.
(1085) Agrawal et al, Philosophical Magazine. V33, 1976. pp. 343-355.
(1086) Wiffen, F.W., Proceedings of the International Conference on Radiation-Induced Voids in Metals. pp. 386-396, 1971.
(1087) Bartlet, A.F., et al. Proceedings of the International Conference on Radiation Effects and Tritium Technology for Fusion Reactors, CNF-750989, p. I 122, 1976.

(1088) Jang, H., and J. Moteff, Proceedings of the International Conference on Radiation Effects and Tritium Technology for Fusion Reactors, CNF-750989, p. I 106, 1976.
(1089) Bajaj, R., et al. "An Investigation of the Irradiation Swelling Mechanisms in Refractory Metals at High Temperatures." Westinghouse Electric Corporation report WAES-TR-89-0010 prepared for Air Force Office of Scientific Research, June 1989.
(1090) Chakin, V., Kazakov, V., "Radiation embrittlement of low-alloyed Mo alloys." Journal of Nuclear Materials. V233-237, 1996. pp 570-572.
(1091) Cockeram, B.V., et al. "Tensile Properties and Fracture mode of a wrought ODS molybdenum sheet following Fast Neutron Irradiation at temperatures ranging from 300°C to 1000°C." presented at TMS Annual Meeting, San Francisco, CA, February, 2005.
(1092) Hasegawa, A., et al. "Influence of heat-treatment on tensile behavior of neutron irradiated molybdenum", Journal of Nuclear Materials. V233-237, 1996. pp 565-569.
(1093) Hasegawa, A., et al. "Tensile behavior and microstructure of neutron-irradiated Mo-5%Re alloy." Journal of Nuclear Materials. V233-237, 1996. pp 259-266.
(1094) Kurishita, H., et al. "Development of Mo alloys with improved resistance to embrittlement by recrystallization and irradiation." Journal of Nuclear Materials. V233-237, 1996. pp 557-564.
(1095) Wiffen, FW. "The Effect of Temperature on the Microstructure of Neutron-Irradiated Tantalum." ANS Transactions. V14, 1971. pp 603.
(1096) Bates, JF and AL Pitner. "Volume Changes in Irradiated Tantalum." ANS Transactions. V14, 1971. pp 603-604.
(1097) Muroga, T., et al. "Correlation of Hardening and Microstructure of Tantalum Irradiated with Heavy Ions." Effects of Radiation on Materials: 19th International Symposium, ASTM 1366, pp. 1186-1196., 2000.
(1098) Youngblood, GE, et al, "Effects of irradiation and post-irradiation annealing on the thermal conductivity/diffusivity of monolithic SiC and f-SiC/SiC composites." Journal of Nuclear Materials. V329-333, 2004. pp 507-512.
(1099) Riccardi, B., et al. "Issues and advances in SiCf/SiC composites development for fusion reactors." Journal of Nuclear Materials. V329-333, 2004. pp. 56-65.
(1100) Post-processing of output from the Bettis computer code, RCPL1, "A Program to Prepare Neutron and Photon Cross-Section Libraries from RCP01." WAPD-TM-1268, AV Dralle, et al. August 1978.
(1101) Garner, FA, et al. "The influence of starting state on neutron induced density changes observed in Nb-1Zr and Mo-41Re at high exposures." Journal of Nuclear Materials. V212-215, 1994. pp 426-430.
(1102) FAX from Eric Pitcher (LANL) to Gary Carpenter. Pitcher, Eric, Los Alamos National Laboratory, FAX dated 11/22/2004.
(1103) Garner, FA."Review of LANL Proposal Concerning the Fuels and materials Test Station (FMTS) for Application to JIMO Irradiation Needs." NRPCT commissioned review, February, 2005.
(1104) Singh, BN, "Impacts of damage production and accumulation on materials performance in irradiation environment." Journal of Nuclear Materials. V258-263, 1998. pp 18-29.
(1105) Stoller, RE, et al. "Primary formation in bcc iron", Journal of Nuclear Materials. V251, 1997. pp 49-60.
(1106) Maloy, SA. "A comparison of Expected Materials Effects in the Materials Test Station to those Predicted for the JIMO Fast Reactor." LANL Draft Report to NRPCT, 2004.
(1107) Draft report: Pitcher, Eric, "Neutronics Assessment of the LANSCE Materials Test Station as an Irradiation Facility for the JIMO Space Reactor." Los Alamos National Laboratory, FAX dated 11/19/2004.

(1108) Farrell, K. and TS Byun. "Tensile properties of ferritic/martensitic steels irradiated in HFIR, and comparison with spallation irradiation data." Journal of Nuclear Materials. V318, 2003. pp 274-282.
(1109) Maloy, SA, et al. "Comparison of Fission Neutron and Proton/Spallation Neutron Irradiation Effects on the Tensile behavior of Type 316 and 304 Stainless Steel." IWSMT-5, Charleston, SC, May 2002.
(1110) Schroeder, H., and P. Batfalsky. "The dependence of the high temperature mechanical properties of austenitic stainless steels on implanted helium." Journal of Nuclear Materials. V117, 1983. pp 287-294.
(1111) Kurtz, RJ, et al. "Recent progress on development of vanadium alloys for fusion." Journal of Nuclear Materials. V329-333, 2004. pp 47-55.
(1112) Chuto, T., et al. "Creep rupture properties of helium implanted V-4Cr-4Ti alloy NIFS-HEAT-2." Journal of Nuclear Materials. V329-333, 2004.
(1113) Fukumoto, K.,et al. "Swelling behavior of V-Fe binary and V-Fe-Ti ternary alloys." Journal of Nuclear Materials. V258-263, 1998. pp 1431-1436.
(1114) Odette, GR, et al. DOE/ER-0313/35, Fusion Materials Semi-Annual Progress Report, pp 80-90, December 31, 2003.
(1115) Aono, Y.,et al. "Effects of Fission and Fusion Neutron Irradiations on Mechanical Properties and Defect Recovery Behavior in a-Iron." Effects of Radiation on Materials: 16th International Symposium, ASTM 1175, Arvind S. Kumar, David S. Gelles, Randy K. Nanstad, and Edward A Little, Eds., American Society for Testing and Materials, Philadelphia, pp. 130-143, 1994.
(1116) Odette, GR, et al, "Fission-Fusion Correlations for Swelling and Microstructure in Stainless Steels: Effect of the helium to Displacement Per Atom Ratio." Journal of Nuclear Materials. V130 & 104, 1981. pp 1289-1304.
(1117) Lackner, K. et al. "Long-term fusion strategy in Europe." Journal of Nuclear Materials. V301-311, 2002. pp 10-20.
(1118) Taguchi, T., et al. "Effect of simultaneous ion irradiation on microstructural change in SiC/SiC composites at high temperature." Journal of Nuclear Materials. V307-311, 2002. pp 1135-1140.
(1119) Hasegawa, A., et al. "Study of hydrogen effects on microstructural development of SiC bas materials under simultaneous irradiation with He- and Si-ion irradiation conditions." Journal of Nuclear Materials. V329-333, 2004 pp 582-586.
(1120) Barmore, W., et al. "Effects of 14 MeV neutron irradiation on creep of nickel and niobium." Journal of Nuclear Materials. V117, 1983. pp 258-263.
(1121) Trinkaus, H., and H. Ullmaier, H. "Conditions for effects of radiation pulsing." Journal of Nuclear Materials. V307-311, 2002. pp 1705-1709.
(1122) Hickman, BS. "Radiation Effects in Beryllium and Beryllium Oxide." Studies in Radiation Effects, Series A Physical and Chemical Vol. 1, 1966.
(1123) Barabash V., et al. "Neutron irradiation effects on plasma facing materials." Journal of Nuclear Materials. V283-287, 2000. pp 138-146.
(1124) Hickman, BS and AW Pryor, Journal of Nuclear Materials. V14, 1964. pp 96-110.
(1125) Collins, CG. Journal of Nuclear Materials. V14, 1964. pp 69-86.
(1126) Plumlee, DE. "BeO Performance Lessons Learned." SP-100 Program, Martin Marietta, March 2, 1994, MMAS-PIR-1203, DOE/SF/16006-T1216.
(1127) KAPL Letter MDO-722-2341, "Post-Irradiation Examination of TP-343/TP-361 Beryllium Oxide and Yttrium Hydride Specimens, For Information," March 1999.
(1128) Keilholtz, GW, et al. "Properties of Mg, Al and BeO Compacts Irradiated toFast-Neutron Doses Greater than 1021 n/cm2 at 150, 800 and 1100C." Conference Proceedings. Nuclear Applications of Nonfissionable Ceramics, Washington, D.C., May 9-11, 1966.
(1129) Pryor, AW et al. Journal of Nuclear Materials. V14, 1964. p 220.
(1130) Snead, LL and SJ Zinkle. "Use of Beryllium and Beryllium Oxide in Space Reactors."

(1131) General Electric Company, Nuclear Materials and Propulsion Operation, Report GEMP 334A (1965).
(1132) Elston, J. and CE. Labbe. Journal of Nuclear Materials. V4, 1961. p 143.
(1133) Dombrowski, DE et al. "Thermomechanical Properties of Beryllium." Brush Wellman Report TR-1182, February 20, 1995.
(1134) DOE Fundamentals Handbook, Nuclear Physics and Reactor Theory, DOE-HDBK-1019/1-93, January 1993.
(1135) Khomutov, A. et al. "Beryllium for fusion application – recent results." Journal of Nuclear Materials. V307-311, 2002. pp 630-637.
(1136) Brush Wellman Inc., Elmore, OH 43416
(1137) Abeln, SP et al. "Elevated Temperature Stress Strain Behavior of Beryllium Powder Product." Conference Proceedings 2nd IEA International Workshop on Beryllium Technology for Fusion, Jackson Lake Lodge, WY, September 6-8, 1995, pp. 57-105.
(1138) Foos, RA. "Microalloying Relationships in Beryllium." Brush Wellman Report BBC-TR-456, March 23, 1970.
(1139) Scaffidi-Argentina, F., et al. "Beryllium R&D for Fusion Applications." Fusion Eng. and Design. V51-52, 2000. pp. 23-41.
(1140) Gelles, DS, et al. "Radiation effects in beryllium used for plasma protection." Journal of Nuclear Materials. V212-215, 1994. pp 29-38.
(1141) Dalle Donne,M., et al. "Modelling of swelling and tritium release in irradiated beryllium." Journal of Nuclear Materials. V212-215, 1994. pp 954-960.
(1142) Baldwin, DL and MC Billone, "Diffusion/Desorption of Tritium from Irradiated Beryllium", Journal of Nuclear Materials. V212-215, 1994. pp 948-953.
(1143) Beeston, JM, et al. "Comparison of Compression Properties and Swelling of Beryllium Irradiated at Various Temperatures." Journal of Nuclear Materials. V122-123, 1984. pp 802-809.
(1144) E. Koonen, "Study on Irradiation Effects and Swelling of Irradiated Beryllium." CEN/SCK Report, Reactor Safety Analysis BR2 Department (1989).
(1145) Rabaglino, E., et al. "Study of the microstructure of neutron irradiated beryllium for the validation of the ANFIBE code." Fusion Eng. and Design. V61-62, 2002. pp 769-773.
(1146) Snead, LL. "Low-temperature low-dose neutron irradiation effects on beryllium." Journal of Nuclear Materials. V326, 2004. pp 114-124.
(1147) Moons, F., et al. "Neutron irradiated beryllium: tensile strength and swelling." Journal of Nuclear Materials. V233-237, 1996. pp 823-827.
(1148) Syslov, DN, et al. "Influence of high dose neutron irradiation on thermal conductivity of beryllium." Journal of Nuclear Materials. V307-311, 2002. pp 664-667.
(1149) Pitcher, E., "The Proposed Materials Test Station at LANSCE." LANL presentation 9-8-2003.
(1150) Zinkle, SJ, "Comments on "Comparison of Expected Materials Effects in the Materials Test Station to those Predicted for the Prometheus Fast Reactor." by S. A. Maloy", draft report from ORNL delivered 12-6-04.
(1151) Was, GS, "Review of article "Comparison of Expected Materials Effects in the Materials Test Station to those Predicted for the Prometheus Fast Reactor." by S. A. Maloy", article review received 1-27-05.
(1152) Trinkaus, H., Singh, B. N., "Helium accumulation in metals during irradiation-Where do we stand?" Journal of Nuclear Materials. V323 (2-3), 2003. pp 229.
(1153) Cottrell, GA and Baker, LJ, "Structural materials for fusion and spallation sources," Journal of Nuclear Materials. V318, 2003. pp 260.
(1154) Tietz, TE and Wilson, JW, "Behavior and Properties of Refractory Metals." Stanford University Press, Stanford, CA, (1965).
(1155) Loomis, BA, et al. "Effects of neutron irradiation and hydrogen on ductile-brittle transition temperatures of V-Cr-Ti alloys." Journal of Nuclear Materials. V212-215, 799 (1994).

(1156) Sekimura, N., et al. "Synergistic effects of hydrogen and helium on microstructural evolution in vanadium alloys by triple ion beam irradiation," Journal of Nuclear Materials V283-287, 2000. pp 224.
(1157) Gelles, DS,et al. "Recent results for the ferritics isotopic tailoring (FIST) experiment.," Journal of Nuclear Materials. V307-311, 2002.pp 212. (2002).
(1158) Doran, DG, "Report of the LASREF Evaluation Committee," Report No. PNL-SA-18584, (1990).
(1159) Ghoniem, N.M., and Kulcinski, "A Critical Assessment of the Effect of Pulsed Irradiation on the Microstructure, Swelling, and Creep in Materials," Nuclear Technology/Fusion, 2, 165 (1982).
(1160) Lee, EH,et al. "Effects of pulsed dual-ion irradiation on phase transformations and microstructure in Ti-modified austenitic alloys." Journal of Nuclear Materials. V117, 1983. pp 123.
(1161) Brimhall, JL, et al. "Void growth in a pulsed irradiation environment." Journal of Nuclear Materials. V117, 1983. pp 118.
(1162) Dai, Y., et al. "The Second SINQ Target Irradiation Program, STIP-I." Journal of Nuclear Materials. V343, 2005. pp 33.
(1163) Dai, Y. and Bauer, GS, "Status of the first SINQ irradiation experiment, STIP-I."
(1164) Journal of Nuclear Materials. V296, 2001. pp 43.
(1165) Jia, X. and Dai, Y., "Microstructure in martensitic steels T91 and F82H after irradiation in SINQ target-3." Journal of Nuclear Materials. V318, 2003. pp 207.
(1166) Ozturk, Z. and Wechsler, MS, "Effects of high energy protons on the mechanical properties of Fe-2.25Cr-1Mo and Fe-12Cr-1Mo steels," Tr. J. of Engr. And Environ. Sci., V22, 1998. pp197.
(1167) Wechsler, MS and Sommer, WF, "Simulation radiation damage." Journal of Nuclear Materials. V122, 1984. pp 1078.
(1168) Was, GS, et al. "Emulation of neutron irradiation effects with protons: validation of principle." Journal of Nuclear Materials. V300, 2002. pp 198.
(1169) Henry, J., et al. "Tensile properties of 9Cr-1Mo martensitic steel irradiated with high energy protons and neutrons." Journal of Nuclear Materials. V318, 2003. pp 215.
(1170) Haines, J., "Overview of Potential Irradiation Facility Options at SNS," presentation given on ORNL tour 1/21/05.
(1171) DOE/IG-0666, US DOE Audit Report on The Los Alamos Neutron Science Center, November 2004.
(1172) KAPL Letter ARP-AC-1300-DPB, Informational Memo: "Evaluation of an Accelerator Driven Sub-critical Reactor System." October, 2005.
(1173) "Preliminary Cost Estimate for the Fuels and Materials Test Station for the Jupiter Icy Moons Orbiter." Los Alamos National Laboratory report: LA-CP-04-0734, October, 2004.

## 11.26 Beryllium

(1174) Abeln, S.P., M.C. Mataya, and R. Field. "Elevated Temperature Stress Strain Behavior of Beryllium Powder Product." Proceedings 2nd IEA International Workshop on Beryllium Technology for Fusion. September, 1995.
(1175) Barabash, V., et al. "Neutron irradiation effects on plasma facing materials." Journal of Nuclear Materials. V283-287, 2000. pp 138-146.
(1176) Beeston, J.M. "Beryllium metal as a neutron moderator and reflector material." Nuclear Engineering and Design. V14, 1970. pp 445-474.
(1177) Billone, M.C. "Recommended design correlations for S-65 beryllium." Proceedings 2nd IEA International Workshop on Beryllium Technology for Fusion. September, 1995.
(1178) Brush Wellman specification, S-65 Structural Grade Beryllium Block, Rev C, July 1, 1987.

(1179) Brush Wellman specification, S-65H Structural Grade Beryllium Block, Rev A, October 5, 1998.
(1180) Brush Wellman specification, S-200F Standard Grade Beryllium Block, Rev A, April 1, 1987.
(1181) Brush Wellman specification, S-200FH Grade Beryllium, Rev B, December 12, 1990.
(1182) Chaouadi, R., F. Moons, and J.L. Puzzolante. “Tensile and fracture toughness test results of neutron irradiated beryllium.” Proceedings of the 3rd IEA International Workshop on Beryllium Technology for Fusion. October, 1997.
(1183) Dalle Donne, M., et al. “Modeling of swelling and tritium release in irradiated beryllium.” Journal of Nuclear Materials. V212-215, 1994. pp 954-960.
(1184) Dombrowski, D.E., E. Deksnis, and M.A. Pick. “Thermomechanical Properties of Beryllium.” Brush Wellman Report TR-1182, February, 1995.
(1185) Gelles, D.S., et al. “Radiation effects in beryllium used for plasma protection.” Journal of Nuclear Materials. V212-215, 1994. pp 29-38.
(1186) Goods, S.H., and D.E. Dombrowski. “Mechanical properties of S-65C grade beryllium at elevated temperatures.” Proceedings of the 3rd IEA International Workshop on Beryllium Technology for Fusion. October, 1997.
(1187) Hickman, B.S. “Radiation Effects in Beryllium and Beryllium Oxide.” Studies in Radiation Effects, Series A Physical and Chemical. V1, 1966 pp 73-97.
(1188) Kangilaski, M. "The Effect of Neutron Radiation on Structural Materials." Radiation Effects Information Center, Report 45, June, 1967.
(1189) Khomutov, A., et al. “Beryllium for Fusion Application – Recent Results.” Journal of Nuclear Materials. V307-311, 2002. pp 630-637.
(1190) Manly, W.D. "Report of a Technical Evaluation Panel on the Use of Beryllium for ITER Plasma Facing Material and Blanket Breeder Material." Sandia Report, SAND95-1693, August, 1995.
(1191) Moons, F. “Beryllium Characterization: Tensile tests on neutron irradiated and reference beryllium.” SCK CEN Progress report FT/Mol/96-03 ITER Task 23, February, 1996.
(1192) Moons, F., et al. “Neutron Irradiated Beryllium: Tensile Strength and Swelling”, Journal of Nuclear Materials. V233-237, 1996. pp 823-827.
(1193) Rabaglino, E., et al. “Study of the microstructure of neutron irradiated beryllium for the validation of the ANFIBE code.” Fusion Engineering and Design. V61-62, 2002. pp 769-773.
(1194) Scaffidi-Argentina, F., et al. “Beryllium R&D for fusion applications.” Fusion Eng. and Design. V51-52, 2000. pp 23-41.
(1195) Smith, M.F., et al. “Thermomechanical testing of beryllium for limiters in ISX-B and JET” Fusion Technology. V8, 1985. pp 1174-1183.
(1196) Snead, L.L. “Low-temperature low-dose irradiation effects on beryllium.” Journal of Nuclear Materials. V326, 2004. pp 114-124.
(1197) Syslov, D.N., V.P. Chakin, and R.N. Latypov. “Influence of high dose neutron irradiation on thermal conductivity of beryllium.” Journal of Nuclear Materials. V307-311, 2002. pp 664-667
(1198) Stonehouse, A. James, and James M. Marder. “Beryllium.” ASM Handbook. Vol 2,1990.

## 11.27 Beryllium Oxide

(1199) Alvis, J.M. "SPACE-BEO: A Predictive Code for BeO Irradiation Performance." General Electric Report, MMAS-PIR-705, April, 1989.
(1200) Brush Ceramic Products specification, Dry Pressed Ceramics as-fired or machined CDDP-10, Rev F, August 6, 2001.

(1201) Busboom, H. "Material Properties for Beryllium Oxide (BeO)." General Electric specification 23A3186, 1989.
(1202) Brush Ceramic Products specification, Isopressed Ceramics CDI-20, Rev E, August 6, 2001.
(1203) Collins, C.G. "Radiation effects in BeO." Journal of Nuclear Materials. V14, 1964. pp 69-86.
(1204) Elston, J., and C. Labbe. Journal of Nuclear Materials. V4, 1961. pp 143-164.
(1205) Francl, J., and W.D. Kingery. "Thermal Conductivity: IX, Experimental investigation of effect of porosity on thermal conductivity." Journal of the American Ceramic Society. V37, 1954. pp 99-107.
(1206) Hickman, B.S., and A.W. Pryor. "The effect of neutron irradiation on beryllium oxide." Journal of Nuclear Materials. V14 1964. pp 96-110.
(1207) Hickman, B.S. "Radiation Effects in Beryllium and Beryllium Oxide." Studies in Radiation Effects, Series A Physical and Chemical. V1, 1966 pp 98-116.
(1208) Keilholtz, G.W., J.E. Lee, Jr. and R.E. Moore. "Irradiation Damage to Sintered Beryllium Oxide as a Function of Fast-Neutron Dose and Flux at 110, 650 and 1100°C." Nuclear Science and Engineering. V26, 1966. pp 329-338.
(1209) Paxton, D.M. "Irradiation and Examination of Beryllium Oxide Pellets from the SP-100 Special Purpose Materials Test", WHC-SP-1007 1993.
(1210) Plumlee, D.E. "BeO Irradiation Induced Swelling/Microcracking Design Margin Study." General Electric Report, MMAS-PIR-947, May, 1991.
(1211) Plumlee, D.E. "BeO Performance Lessons Learned." SP-100 Program, Martin Marietta, March, 1994.
(1212) Pryor, A.W., R.J. Tainsh, and G.K. White. "Thermal conductivity at low temperature of neutron irradiated BeO." Journal of Nuclear Materials. V14, 1964. pp 208-219.
(1213) Ryshkewitch, E. "Beryllium Oxide Ceramics Processes, Properties and Applications." National Beryllia Corporation Technical Report, AFML-TR-65-378, May, 1966.
(1214) Saul, A.M. "Available Information on BeO." Atomics International, NAA-SR-Memo-1767, January, 1957.
(1215) Snead, L.L., and S.J. Zinkle. "Use of beryllium and beryllium oxide in space reactors." ORNL report.
(1216) Touloukian, Y.S., et al. 1970. Thermophysical Properties of Matter V2: Thermal Conductivity. p 123.
(1217) Touloukian, Y.S., et al. 1970. Thermophysical Properties of Matter V5: Specific Heat. p 45.
(1218) Touloukian, Y.S., et al. 1970. Thermophysical Properties of Matter V8: Emissivity. p 201.
(1219) Touloukian, Y.S., et al. 1970. Thermophysical Properties of Matter V13: Coefficient of Thermal Expansion. p 195.
(1220) Walker, D.G., R.M Mayer, and B.S. Hickman. "X-ray diffraction studies of irradiated beryllium oxide." Journal of Nuclear Materials. V14, 1964. pp 147-158.

## 11.28 Boron Carbide ($B_4C$)

(1221) "Materials Properties for Boron Carbide (B4C)." General Electric Document, 23A3183, August, 1988.

## 11.29 LiH Atomistic and Mechanistic Modeling of Irradiation Induced Swelling of LiH

(1222) Bowman, R.C., Jr., et al. "Density of Trapped Gas in Heavily-Irradiated Lithium Hydride." Journal of Nuclear Materials. V154, 1988. pp 318 -331.
(1223) Blöchl, P.E. "Projector Augmented-Wave Method." Physical Review: B. V50, 1994. p 17953-17979.

(1224) Disney, R.K. "Status Report on the SP-100 Program Investigations of Irradiation-Induced Swelling of Lithium Hydride Shield Materials." SP-GES-93-0005, 1993.
(1225) Jónsson, H., G. Mills, and K. W. Jacobsen. 1998. "Nudged Elastic Band Method for Finding Minimum Energy Paths of Transitions", in Classical and Quantum Dynamics in Condensed Phase Simulations, edited by B.J. Berne, G. Ciccotti, and D. F. Coker. World Scientific.
(1226) Kresse, G., and J. Furthmuller. "Efficiency of Ab-Initio Total Energy Calculations for Metals and Semiconductors using a Plane-Wave Basis Set" Computational Materials Science. V6, 1996. pp 15-50.
(1227) Kresse, G., and J. Furthmuller. "Efficiency of Ab-Initio Total Energy Calculations using a Plane-Wave Basis Set" Physical Review, B. V54, 1996. pp 11169-11186.
(1228) Kresse, G., and D. Joubert. "From Altra-Soft pseudopotentials to the Projector Augmented-Wave Method." Physical Review: B. V59, 1999. pp 1758-1775.
(1229) Olander, D.R. "Fundamental Aspects of Nuclear Reactor Fuel Elements." Technical Information Center, 1976. p 203.
(1230) Perdew, J.P., and Y. Wang. "Accurate and Simple Representation of Electron Gas Correlation Energy." Physical Review: B. V45, 1992. pp 13244-13249.
(1231) Perdew, J.P., K. Burke, and M. Ernzerhof. "Generalized Wave Gradient Approximation Made Simple." Physical Review Letters, V77, 1996. pp 3865-3868.
(1232) Perdew, J.P., K. Burke, and M. Ernzerhof. "Generalized Wave Gradient Approximation Made Simple." Physical Review Letters, V78, 1997. p 1396.
(1233) Pretzel, F.E., et al. Los Alamos Laboratory Report, LA-2463, 1961.
(1234) Souers, P.C., et al. "NMR and Electron Microscopy Studies on Irradiated Lithium Hydride." Journal of Physical Chemistry: Solids. V31, 1970. p 1461.
(1235) Souers, P.C., et al. "Pulsed Nuclear Magnetic Resonance of Gamma-Irradiated Lithium Hydride." Journal of Physical Chemistry: Solids. V30, 1969. pp 2649-2656.
(1236) Tufts University, Medford, MA. "A Survey Report on Lithium Hydride." Contract AT(30-1)1410, NYO-9470, 1960.
(1237) Wang, Y., and J.P. Perdew. "Correlation Hole of the Spin-Polarized Electron Gas, with Exact Small-Wave-Vector and High-Density Scaling." Physical Review: B. V44, 1991. pp 13298-13307.
(1238) Welch, F.H. "Lithium Hydride Technology: III. Properties of Lithium Hydride for SNAP Shielding Applications." NAA-SR-9400, May, 1967.

## 11.30 LiH Containment System

(1239) Evolved Expendable Launch Vehicle Standard Interface Specification. Version 6, September 5, 2000: EELV Standard Interface Working Group.
(1240) Mueller, W.M., J.P. Blackedge, and G.G. Libowitz. 1968. Metal Hydrides. New York: Academic Press.
(1241) Welch, F.H. "Lithium Hydride Technology I. Properties of Lithium Hydride and Corrosion Studies." Atomics International Document, NAA-SR-9400, 1964.

## 11.31 LiH Early (Pre- SP-100) Irradiation Experiments

(1242) Aitken, E.A., and D.L. Henry. "Radiation Damage to Lithium Hydride." General Electric Document, APEX-323, 1956.
(1243) Bauer, P., F.H. Welch, and E.P. Kilb. "Irradiation Testing of XMA-1A Shield Materials." General Electric Document, DC 59-5-81, 1959.
(1244) Hamill, C.W., and F.B. Waldrop. "Shielding Studies: Neutron Irradiation Damage to a Lithium Hydride Compact." Union Carbide Corporation, Y-12 Plant Document Y-1454, 1964.

(1245) Minushkin, B. “Final Report: Lithium Hydride Radiation Stability.” Nuclear Development Corporation of America Document, NDA-23, 1956.
(1246) Reagan, M.P. “Radiation Testing of Shield Specimen: Test LTHx-5 and 6.” General Electric Document, DC 58-1-211, 1958.
(1247) Souers, P.C., et al. “Pulsed Nuclear Magnetic Resonance of Gamma-Irradiated Lithium Hydride.” Journal of Physical Chemical: Solids. V30, 1969. pp 2649-2656.
(1248) Welch, F.H. “Lithium Hydride Technology: III. Properties of Lithium Hydride for SNAP Shielding Applications.” Atomics International Document, NAA-SR-9400 V3, 1967.
(1249) Hughes, D.J., and R.G. Sache, “Experimental Nuclear Physics Division and Theoretical Nuclear Physic Division Report for April, May and June 1947.” ANL-4010.

## 11.32 LiH Material Properties

(1250) Dinwiddie, R.B., et al. “The Effect of Gamma Irradiation on the Thermophyscial Properties of LiH + 2% Li.” ORNL Document, ORNL/TM-12766, September, 1994.
(1251) Disney, R.K. "Status Report on the SP-100 Program Investigations of Irradiation-Induced Swelling of Lithium Hydride Shield Materials." Westinghouse Electric Corporation, February 25, 1993.
(1252) “Identification and Analysis of SP-100 Reactor Shield Information, Fabrication Considerations and Resource Assessment of Space Reactor Shielding.” Y-12 Document, DOE NE-50, February, 2004.
(1253) Lundberg, L.B. “Mechanical Properties of LiH Part I – Compression Tests.” Atomics International Document, NAA-SR-MEMO-7991, 1962.
(1254) Mel’nikova, T.N., and K.A. Yakimovich. “Thermal Properties of Lithium Hydride and Its Isotopic Modifications in a Crystalline State.” High Temperature. V18, 1980. pp 250-256.
(1255) Mueller, W.M., J.P. Blackedge, and G.G. Libowitz. 1968. Metal Hydrides. New York: Academic Press.
(1256) Pretzel, F.E., et al. “Radiation Effects on Lithium Hydride.” Los Alamos Scientific Laboratory Document, LA-2463, 1961.
(1257) Smith, R. L., and J.W. Miser. Lewis Research Center, Cleveland, OH. "Compilation of the Properties of Lithium Hydride." NASA Technical Memorandum, X-483, January, 1963.
(1258) Souers, P.C., et al. 1969. “Pulsed Nuclear Magnetic Resonance of Gamma-Irradiated Lithium Hydride.” Journal of Physical Chemical: Solids. V30, 1969. pp 2649-2656.
(1259) Terry, R.E. “Lithium Hydride Debris Shields for Plasma Radiation Studies.” Naval Research Laboratory Document NRL/MR/6720--96-7868, 1996.
(1260) Welch, F.H. “Lithium Hydride Technology: I. Properties of Lithium Hydride and Corrosion Studies.” Atomics International Document, NAA-SR-9400, 1964.Welch, F.H. “Lithium Hydride Technology: III. Properties of Lithium Hydride for SNAP Shielding Applications.” Atomics International Document NAA-SR-9400 V3, 1973.
(1261) Zalkin, A. “The Thermal Expansion of LiH.” USAEC Report URCL-4239, 1953.

## 11.33 LiH Meta Analysis of Early (Pre- SP-100) Irradiation Experiments

(1262) Glass, G.V. “Primary, Secondary, and Meta-Analysis of Research.” Educational Researcher. V5, 1976. pp 3-8.
(1263) Welch, F.H. “Lithium Hydride Technology: III. Properties of Lithium Hydride for SNAP Shielding Applications.” Atomics International Document, NAA-SR-9400 V3, 1967.

## 11.34 A Perspective on the Use of Lithium Hydride In a Space Nuclear Reactor Shield

(1264) Disney, R.K. "Status Report on the SP-100 Program Investigations of Irradiation-Induced Swelling of Lithium Hydride Shield Materials." SP-GES-93-0005, 1993.
(1265) Welch, F.H. "Lithium Hydride Technology: III. Properties of Lithium Hydride For SNAP Shielding Applications." NAA-SR-9400, May, 1967.

## 11.35 LiH SP-100 Irradiation Experiments

(1266) Disney, R.K. "Status Report on the SP-100 Program Investigation of Irradiation-Induced Swelling of Lithium Hydride Shield Materials." Westinghouse Electric Corporation Document, SP-GES-93-0005, 1993.
(1267) "Documentation Supporting ATR SP-100 Irradiation Experiments." ORNL/LTR-NR-JIMO/04-03 Appendix 3, 2004.
(1268) Souers, P.C., et al. "Pulsed Nuclear Magnetic Resonance of Gamma-Irradiated Lithium Hydride." Journal of Physical Chemistry: Solids. V30, 1969. pp 2649-2656.

## 11.36 Summary of LiH Material Performance, Issues in a Space Nuclear Reactor Shield

(1269) Angelo, J.A., Jr., and D. Burden. 1985. Space Nuclear Power. Florida: Orbit Book Company, Inc.
(1270) Aftergood, S. "Background on Space Nuclear Power." Science & Global Security. V1, 1989. pp 93-107.
(1271) Aitken, E.A., and D.L. Henry. "Radiation Damage to Lithium Hydride." General Electric Document, APEX-323, 1956.
(1272) "Annual Technical Progress Report October 1990 Through September 1991." DOE Document, DOE/NBM-1091, 1992.
(1273) Baxter, W.G., and F.H. Welch. "Shield Materials." General Electric Document APEX-915, 1962.
(1274) Bowman, R.C., Jr., et al. "Density of Trapped Gas in Heavily-Irradiated Lithium Hydride." Journal of Nuclear Materials. V154, 1988. pp 318-331.
(1275) Brasier, J.E., Sr. "INEL LIH Irradiation Testing, SP-100 Final Report." February, 1995.
(1276) Bauer, P., F.H. Welch, and E.P. Kilb. "Irradiation Testing of XMA-1A Shield Materials." General Electric Document, DC 59-5-81, 1959.
(1277) Dinwiddie, R.B., et al. "The Effect of Gamma Irradiation on the Thermophysical Properties of LiH + 2% Li." ORNL Document, ORNL/TM-12766, September, 1994.
(1278) Disney, R.K. "Status Report on the SP-100 Program Investigations of Irradiation-Induced Swelling of Lithium Hydride Shield Materials." Westinghouse Electric Corporation document, SP-GES-93-0005, February, 1993.
(1279) Hamill, C.W., and F.B. Waldrop. "Shielding Studies: Neutron Irradiation Damage to a Lithium Hydride Compact. Union Carbide Corporation." Y-12 Plant Document, Y-1454, 1964.
(1280) "Identification and Analysis of SP-100 Reactor Shield Information, Fabrication Considerations and Resource Assessment of Space Reactor Shielding." Y-12 Document, DOE NE-50, February, 2004.
(1281) Minushkin, B. "Final Report: Lithium Hydride Radiation Stability." Nuclear Development Corporation of America Document, NDA-23, 1956.
(1282) Oak Ridge National Laboratory Document, ORNL/LTR-NR-JIMO/04-03, Appendix 2.
(1283) Pretzel, F.E., et al. "Properties of Lithium Hydride-V Vacancy Formation, Cavitation, and Lithium Precipitation in Irradiated Lithium Hydride. Journal of Applied Physics. V33, 1962. p 510.
(1284) Pretzel, F.E., et al. "Radiation Effects on Lithium Hydride." Los Alamos Scientific Laboratory Document, LA-2463, 1961.

(1285) Reagan, M.P. "Radiation Testing of Shield Specimen: Test LTHx-5 and 6." General Electric Document, DC 58-1-211, 1958.
(1286) Schmidt, G.L. "SNAP 10A Test Program." Air Force Weapons Laboratory Document, 1988.
(1287) Smith, R.L., and J.W. Miser. Lewis Research Center, Cleveland, OH. "Compilation of the Properties of Lithium Hydride." NASA Technical Memorandum, X-483, January, 1963.
(1288) Souers, P.C., et al. "NMR and Electron Microscopy Studies on Irradiated Lithium Hydride." Journal of Physical Chemistry: Solids. V31, 1970. pp 1461-1466.
(1289) Souers, P.C., et al. "Pulsed Nuclear Magnetic Resonance of Gamma-Irradiated Lithium Hydride." Journal of Physical Chemistry: Solids. V30, 1969. pp 2649-2656.
(1290) Terry, R.E. "Lithium Hydride Debris Shields for Plasma Radiation Studies." Naval Research Laboratory Document, NRL/MR/6720--96-7868, 1996.
(1291) Welch, F.H. "Lithium Hydride Technology I. Properties of Lithium Hydride and Corrosion Studies." Atomics International Document, NAA-SR-9400, 1964.
(1292) Welch, F.H. "Lithium Hydride Technology II. Fabrication of Shadow Shields for SNAP Applications." Atomics International Document, NAA-SR-9400, V2, 1965.
(1293) Welch, F.H. "Lithium Hydride Technology: III. Properties of Lithium Hydride for SNAP Shielding Applications." Atomics International Document, NAA-SR-9400 V3, 1967.
(1294) Welch, F.H. "Lithium Hydride Technology: IV. A Novel Neutron-Gamma Material for SNAP Shielding Applications." Atomics International Document, NAA-SR-9400 V4, 1967.
(1295) Welch, F.H. "Lithium Hydride Technology: V. Testing and Examination of SNAP Shadow Shields." Atomics International Document, NAA-SR-9400 V5, 1967.

## 11.37 LiH X-ray Irradiation Testing of Lithium Hydride

(1296) Bushberg, et al. 2001. The Essential Physics of Medical Imaging, 2nd Ed. Lippincott, Williams, and Wilkins. p 103.

## 11.38 $UO_2$ Restructuring

(1297) Bagger, C., M. Morgensen, and C.T. Walker. "Temperature Measurements in High Burnup UO2 Nuclear Fuel: Implications For Thermal Conductivity, Grain Growth and Gas Release." Journal of Nuclear Materials. V211, 1994. pp 11-29.
(1298) Bernard, L.C., J.L. Jacoud, and P. Vesco. "An Efficient Model For the Analysis of Fission Gas Release." Journal of Nuclear Materials. 302, 2002. pp 125-134.
(1299) Clement, C.F. "The Movement of Lenticular Pores in UO2 Nuclear Fuel Elements." Journal of Nuclear Materials. V68, 1977. pp 63-68.
(1300) Cunningham, M.E., M.D Freshley, and D.D. Lanning. "Development and Characterization of the Rim Region in High Burnup UO2 Fuel Pellets." Journal of Nuclear Materials. V188, 1992. pp 19-27.
(1301) Hastings, I.J. "Effect of Initial Grain Size on Fission Gas Release From Irradiated UO2 Fuel." Communications of the American Ceramic Society. V66, 1983. pp c-150 - c-151.
(1302) Holden, R.B. Ceramic fuel elements. 1966.
(1303) Hoppe, N, P. Verbeek. Restructuring and Relocation in Oxide Fuels in Fuel-Pin Modeling and Analysis. pp 175-177.
(1304) Kawamata, H., et al. "Migration Rate of Lenticular Voids in UO2 Under the Influence of Temperature Gradient." Journal of Nuclear Materials. V68, 1977. pp 48-53.
(1305) Kinoshita, M. "Towards the Mathematical Model of Rim Structure Formation." Journal of Nuclear Materials. V248, 1997. pp 185-190.

(1306) Kinoshita, M., et al. "Temperature and Fission Rate Effects on the Rim Structure Formation in a UO2 Fuel With a Burnup of 7.9%." Journal of Nuclear Materials. V252, 1998. pp 71-78.

(1307) Lucuta, P.G., Hj. Matzke, and I.J.J. Hastings. "A Pragmatic Approach To Modelling Thermal Conductivity of Irradiated UO2 Fuel: Review and Recommendation." Journal of Nuclear Materials. V232, 1996. pp 166-180.

(1308) Matzke, Hj., and J. Spino. "Formation of the Rim Structure in High Burnup Fuel." Journal of Nuclear Materials. V248, 1997. pp 170-179.

(1309) Manzel, R., and R. Eberie. Proceedings of the International Topical Meeting on LWR Fuel Performance. April, 1991.

(1310) Nichols, F.A. "On the Thermal Gradient Migration of Lenticular Voids." Journal of Nuclear Materials. V84, 1979. pp 319-326.

(1311) Nichols, F.A. "Transport Phenomena in Nuclear Fuels Under Severe Temperature Gradients." Journal of Nuclear Materials. V84, 1979. pp 1-25.

(1312) Nogita, K., et al. "Effect of Grain Size on Recrystallization in High Burnup Fuel Pellets." Journal of Nuclear Materials. V248, 1997. pp 196-203.

(1313) Nogita, K., and K. Una. "High Resolution TEM of High Burnup UO2 Fuel." Journal of Nuclear Materials. V250, 1997. pp 244-249.

(1314) Nogita, K., and K. Une. "Irradiation-Induced Recrystallization in High Burnup UO2 Fuel." Journal of Nuclear Materials. V226, 1995. pp 302-310.

(1315) Olander, D.R. 1976. Fundamental Aspects of Nuclear Reactor Fuel Elements. Technical Information Center, Office of Public Affairs, Energy Research and Development Administration.

(1316) Oldfield, W., and A.J. Markworth. "Theory of Bubble Migration Applied to Irradiated Materials." Materials Science and Engineering. V4, 1969. pp 353-366.

(1317) Pati, S.R., A.M. Garde, and L.J. Clink. Proceedings of the International Topical Meeting on LWR Fuel Performance. April, 1988.

(1318) Ray, I.L.F, H. Thiele, and Hj. Matzke. "Transmission Electron Microscopy Study of Fission Product Behavior in High Burnup UO2." Journal of Nuclear Materials. V188, 1992. pp 90-95.

(1319) Rest., J. "Application of a Mechanical Model for Radiation-Induced Amorphization and Crystallization of Uranium Silicide to Recrystallization of UO2." Journal of Nuclear Materials V326, 1997. pp 180-184.

(1320) Rest, J. "A Model For the Influence of Microstructure, Precipitate Pinning and Fission Gas Behavior on Irradiation-Induced Recrystallization of Nuclear Fuels." Journal of Nuclear Materials. V326, 2004. pp 175-184.

(1321) Rest, J., and G.L. Hofman. "An Alternative Explanation of Evidence That Xenon Depletion, Pore Formation, and Grain Subdivision Begin at Different Local Burnup." Journal of Nuclear Materials. V277, 2000. pp 231-238.

(1322) Rest, J., and G.L. Hofman. "Dynamics of Irradiation Induced Grain Subdivision and Swelling in U3Si2 and UO2 Fuels." Journal of Nuclear Materials. V210, 1994. p 187.

(1323) Roberts, J.T.A., and B.J. Wrona. "Crack Healing in UO2." Journal of the American Ceramic Society. V56, 1973. pp 297-299.

(1324) Ronchi, C., and C. Sari. "Properties of Lenticular Pores in UO2, (U, Pu)O2 and PuO2." Journal of Nuclear Materials. V50, 1974. pp 91-97.

(1325) Ronchi, C., et al. "Effect of Burn-Up on the Thermal Conductivity of Uranium Dioxide up to 100.00 MWdt-1." Journal of Nuclear Materials. V327, 2004. pp 58-76.

(1326) Sens, P.F. "The Kinetics of Pore Movement in UO2 Fuel Rods." Journal of Nuclear Materials. V43, 1972. pp 293-307.

(1327) Spino, J., D. Papaioannou, and J.-P. Glatz. "Comments on the Threshold Porosity for Fission Gas Release in High Burn-Up Fuels." Journal of Nuclear Materials. V328, 2004. pp 67-70.

(1328) Spino, J., K. Vennix, and M. Coquerelle. "Detailed Characterization of the Rim Microstructure in PWR Fuels in the Burn-Up Range 40-67 GWd/tM." Journal of Nuclear Materials. V231, 1996. pp 179-190.
(1329) Thomas, L.E., C.E. Beyer, and L.A. Charlot. "Microstructure Analysis of LWR Spent Fuels at High Burnup." Journal of Nuclear Materials. V188, 1992. pp 80-89.
(1330) Turnbull, J.A. "The Effect of Grain Size on the Swelling and Gas Release Properties of UO2 During Irradiation." Journal of Nuclear Materials. V50, 1974. pp 62-68.
(1331) Une, K., et al. "Rim Structure Formation and High Burnup Fuel Behavior of Large-Grained UO2 Fuels." Journal of Nuclear Materials. V278, 2000. pp 54-63.
(1332) Walker, C.T., C. Bagger, and M. Morgensen. "Observations on the Release of Cesium from UO2 Fuel." Journal of Nuclear Materials. V240, 1996. pp 32-42.
(1333) Walker, C.T., and M. Conquerelle. Proceedings of the International Topical Meeting on LWR Fuel Performance. April, 1991.
(1334) Walker, C.T., et al. "Concerning the Microstructure Changes that Occur at the Surface of UO2 Pellet on Irradiation to High Burnup." Journal of Nuclear Materials. V188, 1992. pp 73-79.
(1335) Spino, J, et al., "High Burn-up rim structure: evidences that xenon depletion, pore formation, and grain subdivision start at different local burn-ups", Journal of Nuclear Materials. V256, 1998. pp 189-196.

## 11.39 $UO_2$ Thermal Conductivity

(1336) Carrol, J.C., R.A. Gomme, and N.A. Leech. "Thermal Diffusivity Measurements on Unirradiated Archive Fuel, and Fuel Irradiated in the Halden IFA-558 Experiment." Proceedings HWR-345 Enlarged HPG Meeting on High Burnup Fuel Performance, Safety and Reliability and Degradation of In-Core Materials and Water Chemistry Effects and Man-Machine Systems Research. HWR-345-13, 1994.
(1337) Harding, J.H., and D.G. Martin. "Recommendation for the Thermal Conductivity of UO2." Journal of Nuclear Materials. V166, 1989. pp 223-226.
(1338) Kittel, C. 1996. Introduction to Solid State Physics. New York: John Wiley and Sons, Inc.
(1339) Kleykamp, H. "The Chemical State of the Fission Products in Oxide Fuels." Journal of Nuclear Materials. V131, 1985. pp 221-246.
(1340) Lanning, D.D., C.E. Beyer, and K.J. Geelhood. NUREG-/CR-6534, Volume 4, PNNL-11513, May, 2005.
(1341) Lucuta, P.G., Hj. Matzke, and I.J. J. Hastings. "A Pragmatic Approach to Modelling Thermal Conductivity of Irradiated UO2 Fuel: Review and Recommendation." Journal of Nuclear Materials. V232, 1996. pp 166-180.
(1342) Ohira, K., and N. Itagaki. "Thermal Conductivity Measurements of High Burnup UO2 Pellet and a Benchmark Calculation of Fuel Center Temperature. " Proceedings of the ANS International Topical Meetings on LWR Fuel Performance. March 2-6, 1997.
(1343) Ronchi, C., et al. "Effect of Burnup on the thermal conductivity of uranium dioxide up to 100.00 MWDt-1." Journal of Nuclear Materials. V327, 2004. pp 58-76.
(1344) Inoue, M. "Thermal conductivity of uranium – plutonium oxide fuel for fast reactors," Journal of Nuclear Materials. V282, 2000. pp 186-195.

## 11.40 Fuel Performance Codes

(1345) ME Cunningham and CE Beyer, "GT2R2: an Updated Version of GAPCON-THERMAL-2, NUREG/CR-3907", PNL-5178, Pacific Northwest National Laboratory (1984).

(1346) Veshchunov, M.S. and Dubourg, R. “Numerical simulation of fission product release under accidental conditions with the MFPR code.” Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE) Moscow, Russia.
(1347) D.T. Hagrman, ed. “SCDAP/RELAP5/MOD3.1 Code Manual Volume IV: MATPRO-A Library of Materials Properties for Light Water Reactor Accident Analysis”. Idaho National Engineering Laboratory, November 1993.
(1348) International Atomic Energy Agency. “Validation of Fast Reactor Thermomechanical and Thermohydraulic Codes.” IAEA-TECDOC-1318, November, 2002.
(1349) Lassmann, K., et al. “A New Data-Condensation Method Based on Multidimensional Minimisation.” Kerntechnik, V69 No. 1-2, pp 21-25, 2004.
(1350) M Suzuki, et al. “Analysis of MOX fuel behavior in reduced-moderation water reactor by fuel performance code FEMAXI-RM.” Nuclear Engineering and Design. V227, 2004, p 19-27.
(1351) H Mikami, et al. “Fission product release inventory analysis code for high-temperature gas-cooled reactor during accident: HTCORE.” Japan Atomic Energy Research Institute, JAERI-M 88-256, December 1988.
(1352) “The SAS4A/SASSYS-1 LMR Analysis Code System Volume 3.” Argonne National Laboratory, ANL-FRA-1996-3. August 1996.

## 11.41 Uranium Nitride

(1353) MA DeCrescente, MS Freed, and SD Caplow, “Uranium Nitride Fuel Development, SNAP-50”, PWAC 488, Pratt & Whitney Aircraft, Oct 1965.
(1354) SC Weaver, RL Senn, JL Scott, and BH Montgomery, “Effects of Irradiation on Uranium Nitride under Space Reactor Conditions”, ORNL4461, Oct 1969.
(1355) T Muromura and H Tagawa, “Formation of Uranium Mononitride by the Reaction of Uranium Dioxide with Carbon in Ammonia and a Mixture of Hydrogen and Nitrogen—I. Synthesis of High Purity UN”, Journal of Nuclear Materials, 71, 1977. pp 65-72.
(1356) MA DeCrescente, MS Freed, SD Caplow, “Uranium Nitride Fuel Development SNAP-50”, USAEC Report PWAC 488, Pratt & Whitney Aircraft, October 1965; USAEC Report PWAC-479, Pratt & Whitney Aircraft, September 1965.
(1357) PA Vozzella and MA DeCrescente, “Thermodynamic Properties of UN”.
(1358) DL Keller, “Development of Uranium Mononitride”, BMI-X-10083 Quarterly Progress Report for January-March 1964, Battelle Memorial Institute.
(1359) DL Keller, “Development of Uranium Mononitride”, BMI-X-10073 Quarterly Progress Report for October-December 1963, Battelle Memorial Institute.
(1360) BJ Makenas, Westinghouse Hanford Letter 8957290, “Photographs And Mosaic of sample 059-C”, December 14, 1989.
(1361) MH Fassler, FJ Huegel, and MA DeCrescente, “The Compressive Creep Strength of UC and UN”, Pratt & Whitney Aircraft – CANEL, September 30, 1965.
(1362) Cockeram, D.J. “SNAP 2, 8 and 10A Reactor Programs Progress Report.” Presented at the AIAA Third Biennial Aerospace Power Systems Conference, Philadelphia, PA, Sept 1-4, 1964.
(1363) WJ Carmack, et al. “Internal Gelation as Applied to the Production of Uranium Nitride Space Nuclear Fuel.” AIP Conference Proceedings. V699, No. 1, 2004. p 327-331.
(1364) Hales, J.W. “SP-100 Irradiation Test Plan”, Westinghouse Hanford Company. October 1987.
(1365) Makenas, B.J., Hales, J.W. and Karnesky, R.A. “Postirradiation Examination Data from SP-1 at 3 at .% Burnup”, Westinghouse Hanford Company. January 1989.
(1366) Makenas, B.J., Hales, J.W. and Karnesky, R.A. “The First Interim Examination of the SP-3 Test (U).” Westinghouse Hanford Company. September 1989.

(1367) Makenas, B.J. "Postirradiation Examination of the SP-3R Test (U)." Westinghouse Hanford Company. April 1990.
(1368) Makenas, B.J. "Postirradiation Examination of the FSP-1 Test (U.)" Westinghouse Hanford Company. September 1991.
(1369) Makenas, B.J. "Postirradiation Examination Report for the SP-3RR Test (U)." Westinghouse Hanford Company. June 1993.
(1370) Makenas, B.J. "Roadmap for SP-100 Fuel Pin Irradiation Testing." Westinghouse Hanford Company, WHC-SP-1083, February 1994.
(1371) Paxton, D.M. and Makenas, B.J. "Postirradiation Examination Report for the SP-3RR Test (U)." Westinghouse Hanford Company, WHC-93-00010. June 1993.
(1372) Makenas, B.J. "Postirradiation Examination of the SP-3R Test (U)." Westinghouse Hanford Company, WHC90-00017. April 1990.
(1373) Cockeram, D.J. "SNAP 2, 8 and 10A Reactor Programs Progress Report." Proceedings of the Space Power Systems Conference, Santa Monica, CA. September 1962.
(1374) TJ Sturiale and MA DeCrescente, "Self-Diffusion of Nitrogen on Uranium Mononitride", USAEC Report PWAC-477, Pratt & Whitney Aircraft, September 1965.
(1375) MA DeCrescente, MS Freed, SD Caplow, "Uranium Nitride Fuel Development SNAP-50", USAEC Report PWAC 488, Pratt & Whitney Aircraft, October 1965; USAEC Report PWAC-479, Pratt & Whitney Aircraft, September 1965.
(1376) MH Fassler, FJ Huegel, and MA DeCrescente, "The Compressive Creep Strength of UC and UN", Pratt & Whitney Aircraft – CANEL, September 30, 1965.
(1377) DL Keller, "Development of Uranium Mononitride", BMI-X-10083 Quarterly Progress Report for January-March 1964, Battelle Memorial Institute.
(1378) DL Keller, "Development of Uranium Mononitride", BMI-X-10073 Quarterly Progress Report for October-December 1963, Battelle Memorial Institute
(1379) BJ Makenas, Westinghouse Hanford Letter 8957290, "Photographs and Mosaic of sample 059-C", December 14, 1989.
(1380) JB Holt and MY Almassy, "Nitrogen Diffusion in Uranium Nitride as Measured by Alpha Particle Activation of N15", Journal of the American Ceramic Society, Volume 52, 1969. pp 631-635.
(1381) Demuth, S.F. "SP-100 Space Reactor Design." Progress in Nuclear Energy. V42, No. 3. 2003. pp 323-359.
(1382) Ebbinghaus, B, Choi, J and Meier, T. "A Modified Nitride-Based Fuel for Long Core Life and Proliferation Resistance." Lawrence Livermore National Laboratory.
(1383) J Collins and R Hunt. "Preparation of Uranium Nitride Kernels for TRISO Coating." Oak Ridge National Laboratory.
(1384) BJ Beer. "Memo: Fabrication of Uranium Nitride Fuel for SP-100." Los Alamos National Laboratory. 1988.
(1385) JL Collins, et al. "Effects of Process Variables on Reaction Mechanisms Responsible for ADUN Hydrolysis, Precipitation, and Gelation in the Internal Gelation Gel-Sphere Process." Oak Ridge National Laboratory. ORNL/TM-8818. 1984.
(1386) Russian Science Center Kurchatov Institute. "Mononitride Fuel for Fast Reactors." UDC 621.039.526, Atomnaya Energiya. V95, No. 3. 2003. pp. 208-221.
(1387) T Muromura and H Tagawa. "Mechanism and Kinetics for the Formation of Uranium Mononitride by the Reaction of Uranium Dioxide with Carbon and Nitrogen." Journal of the American Ceramic Society. V61, No. 1-2, 1978. p 30-35.
(1388) JM Cleveland, et al. "A New, Low Temperature Synthesis of Plutonium and Uranium Nitrides." Nuclear Technology. V25, 1974. p 541-545.
(1389) N OI, S Hirayama, et al. "Preparation of High Density Uranium Nitride and Uranium Carbide Fuel Pellets." Journal of Nuclear Science and Technology. V9, No. 9, 1972. p 521-527.

(1390) F Anselin. "Study of Uranium Nitrides and Their Solid Solutions." Journal of Nuclear Materials. V10, No. 4, 1963. p 301-320.
(1391) RB Matthews, KM Chidester, CW Hoth, RE Mason and RL Petty. "Fabrication and Testing of Uranium Nitride Fuel for Space Power Reactors." Journal of Nuclear Materials. V151, 1988. pp 334-344.
(1392) RD Shoup. "Process Variables in the Preparation of UN Microspheres." Journal of the American Ceramic Society. V60, No. 7-8, 1977. p 332-335.
(1393) SK Mukerjee, et al. "Kinetics of the Carbothermic Synthesis of Uranium Mononitride Microspheres." Journal of Nuclear Materials. V185, 1991. p 39-49.
(1394) G Ledergerber, et al. "Preparation of Uranium Nitride in the Form of Microspheres." Journal of Nuclear Materials. V 188, 1992. p 28-35.
(1395) VJ Tennery, et al. "Sintering of UN as a Function of Temperature and N2 Pressure." Journal of the American Ceramic Society. V 54, No. 7, 1971. p 327-331.
(1396) BD Rogozkin, et al. "Thermochemical Stability, Radiation Testing, Fabrication, and Reprocessing of Mononitride Fuel." Atomic Energy. V95, No. 6, 2003. p 835-844.
(1397) RF Hilbert, et al. "High-Temperature Irradiation Behavior of UN, UC, UO2 Fuels Compared." Transactions of the American Nuclear Society. V13, 1970. p 102-103.
(1398) Matthews, R.B. "Ceramic Fuel Development for Space Reactors." Nuclear Materials Technology Division, Los Alamos National Laboratory. Ceramic Bulletin, V1, 1992. pp 96-101.
(1399) Matthews, R. Bruce. "Irradiation Performance of Nitride Fuels." Specialist Conference on Space Nuclear Power and Propulsion Technologies – Materials and Fuels. September, 1992.
(1400) Matthews, RB. "Fuels for Space Nuclear Reactors." Materials Science and Technology 1985 Review. pp 51-56.
(1401) Makenas, Bruce J. et al. "SP-100 Pin Performance: Results From Irradiation Testing." 1994 SP Proceedings of the 11th Symposium on Space Nuclear Power Propulsion. pp 403-412.
(1402) Hayes, SL. et al. "Material Property Correlations for Uranium Mononitride III. Transport Properties." Journal of Nuclear Materials, V171, pp 289-299, 1990.
(1403) Pratt & Whitney Aircraft. "PWAC-498 Fabrication of Low Density Uranium Nitride for High Fission Gas Release." Contract AT(30-1)-2789, September, 1965.
(1404) Pratt & Whitney Aircraft. "PWAC-484 Fuel and Shield Component Development – SNAP-50." Contract AT(30-1)-2789, October, 1965.
(1405) Bugl, J. "Uranium Mononitride - A New Reactor Fuel." Nucleonics, V22 No. 9, pp 66-70, September, 1964.
(1406) JG Slaby, et al. "Examination of T-111 clad uranium nitride fuel pins irradiated up to 13,000 hours at a clad temperature of 990°C." NASA Lewis Research Center, NASA TM X-2950, 1973.
(1407) BD Rogozkin, et al. "Mononitride Fuel for Fast Reactors." Atomic Energy. V95, No 3, 2003, p 624-636.
(1408) AA Bauer. "Nitride Fuels: Properties and Potentials." Reactor Technology. V15, No 2, 1972.
(1409) VJ Tennery, et al. "Synthesis, Characterization, and Fabrication of UN." Oak Ridge National Laboratory/ NASA Lewis Research Center, ORNL-4608/NASA CR-72764, December 1970.
(1410) RE Mason, et al. "Uranium Nitride Fuel Fabrication for SP-100 Reactors." From ElGenk, ed. Space Nuclear Power Systems 1985.
(1411) LM Zabudko, et al. "Nitride Fuel for Advanced Fast Sodium Reactors." Presented at Global 2003, New Orleans, LA, Nobember 2003.
(1412) RR Metroka. "Fabrication of Uranium Mononitride Compacts." NASA Lewis Research Center, NASA TN D-5876, July 1970.

(1413) RB Matthews, et al. "Fuels for space nuclear power and propulsion: 1983-1993." Los Alamos National Laboratory.
(1414) RB Holden. Ceramic Fuel Elements. Gordon and Breach Science Publishers, New York. 1966.
(1415) J Katz and E Rabinowitch. The Chemistry of Uranium. "Chapter 10 – Uranium Compounds with Elements of Group Va." 1951.
(1416) KM Chidester, et al. "Uranium Nitride Oxidation and Pyrophoricity." Los Alamos National Laboratory, March 19, 1992.
(1417) ML Thompson. "SP-100 Specification: Uranium Nitride Fuel Pellet" General Electric Aerospace, Specification Number 23A3123, Revision 6. June 21, 1990.

## 11.42 UN Fission Gas Release

(1418) MB Weinstein, TA Kirchgessner, TN Tambling, "Fission Gas Release from Uranium Nitride at High Fission Rate Density", NASA TN D-7171, 1973.
(1419) WHC88-00310 Appendix A. "Gamma Scans of Encapsulated SP-1 Fuel Pins." Pp 43-60.
(1420) Vaidyanathan, Swaminathan and Donald E. Plumlee. "A Model for Fission Gas Release from UN Fuel." Proceedings of Conf 930103; American Institute of Physics.
(1421) Storms, EK, "An Equation Which Describes Fission Gas Release From UN Reactor Fuel." Journal of Nuclear Materials. V158, 1988. pp 119-129.
(1422) Weinstein, M.B. and H.W. Davison. "A Fission Gas Release Correlation for Uranium Nitride Fuel Pins." NASA Technical Note. NASA TN D-7401, October 1973.
(1423) K Tanaka, et al. "Fission Gas Release and Swelling in Uranium-Plutonium Mixed Nitride Fuels." Journal of Nuclear Materials. V327, 2004. p 77-87.

## 11.43 UN Swelling

(1424) SB Ross, MS El-Genk, and RB Matthews, "Uranium Nitride Fuel Swelling Correlation," Journal of Nuclear Materials, V170,1990, pp. 169-177.
(1425) K Tanaka, et al. "Fission Gas Release and Swelling in Uranium-Plutonium Mixed Nitride Fuels." Journal of Nuclear Materials. V327, 2004. p 77-87.
(1426) Rogozkin, BD, et al. "Mononitride Fuel for Fast Reactors." Atomic Energy. V95, No. 3, 2003. pp 624-636.
(1427) Zaabudko, LM., et al. "Nitride Fuel for Advanced Fast Sodium Reactors." Global 2003 November, 2003.
(1428) El-Genk, Mohamed S., et al. "Uranium Nitride Fuel Swelling and Thermal Conductivity Correlations." 4th Symposium on Space Nuclear Power Systems. January, 1988.

## 11.44 UN Thermal Conductivity

(1429) Ross, Steven B. "Thermal Conductivity Correlation for Uranium Nitride Fuel Between 10 and 1923 K." Journal of Nuclear Materials. V151, 1988. pp 313-317.

## 11.45 Uranium Dioxide

(1430) DJ Clough, "Creep Properties of Oxide and Carbide Fuels Under Irradiation", Journal of Nuclear Materials, Volume 65, 1977. pp. 24-36.
(1431) Speer, Michael A., "LWR Fuel Pin Performance during Burnup." Department of Nuclear Engineering, University of California.
(1432) Lassmann, K., et al. "The Radial Distribution of Plutonium in High Burnup UO2 Fuels." Journal of Nuclear Materials, V208, pp 223-231, 1994.

(1433) Amoretti, G., et al. "5f-Electron States in Uranium Dioxide Investigated Using High-Resolution Neutron Spectroscopy." The American Physical Society, V40 No. 3, pp 1856-1870, July, 1989.
(1434) Robins, R.G., et al. "The Crystal Habit of Uranium Dioxide." Journal of Nuclear Materials, V5 No. 2, pp 262-263, 1962.
(1435) J Belle. UO2: Properties and Nuclear Applications. US Atomic Energy Commission, 1961.
(1436) ASTM Standard C 696-99. "Standard Test Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets."
(1437) ASTM Standard C 1430-00. "Standard Test Method for Determination of Uranium, Oxygen to Uranium (O/U), and Oxygen to Metal (O/M) in Sintered Uranium Dioxide and Gadolina-Uranium Dioxide Pellets by Atmospheric Equilibration."
(1438) ASTM Standard C 753-04. "Standard Specification for Nuclear-Grade, Sinterable Uranium Dioxide Powder."
(1439) ASTM Standard C 776-00. "Standard Specification for Sintered Uranium Dioxide Pellets."
(1440) ASTM Standard C 1287-03. "Standard Test Method for Determination of Impurities in Nuclear Grade Uranium Compounds by Inductively Coupled Plasma Mass Spectrometry."

## 11.46 $UO_2$ Fission Gas Release

(1441) J Spino, et al. "Comments on the Threshold Porosity for Fission Gas Release in High Burn-up Fuels", Journal of Nuclear Materials. V328, 2004. pp 67-70.
(1442) CT Walker, et al., "Observations on the Release of Cesium From UO2 Fuel", Journal of Nuclear Materials. V240, 1996. pp 32-42.
(1443) Zimmermann, "Investigations on Swelling and Fission Gas Behavior in Uranium Dioxide", Journal of Nuclear Materials. V75, 1978. pp 154-161.
(1444) Bernard, L.D., Jacoud, J.L. and Vesco, P. "An efficient model for the analysis of fission gas release." Journal of Nuclear Materials. V302, 2002. pp. 125-134.
(1445) Sakurai, et al. "Fission Gas Release and Related Behaviors of BWR Fuel under Steady and Transient Conditions." Dec. 2004.
(1446) Turnbull, J.A., Menut, P. and Sartori, E. "A Review of Fission Gas Release Data within the NEA/IAEA IFPE Database."
(1447) Friskney, C.A. and Turnbull, J.A. "The Characteristics of Fission Gas Release From Uranium Dioxide During Irradiation", Journal of Nuclear Materials. V79, 1979. pp. 184-198.
(1448) Friskney, C.A., et al. "The Diffusion Coefficients of Gaseous and Volatile Species During the Irradiation of Uranium Dioxide." Journal of Nuclear Materials. V107, 1982. pp. 168-184.
(1449) Hastings, Ian J. "Effect of Initial Grain Size on Fission Gas Release from Irradiated UO2 Fuel." Atomic Energy of Canada Limited. September 1983.
(1450) Bernard, L.C., Jacoud, J.L. and Vesco, P. "An efficient model for the analysis of fission gas release." January 2002.
(1451) A.H. Booth. "A Method of Calculating Fission Gas Diffusion from UO2 Fuel and Its Application to the X-2-f Loop Test". Atomic Energy of Canada Limited. CRDC-721, 1957.
(1452) A.H. Booth and G.T. Rymer. "Determination of the Diffusion Constant of Fission Xenon in UO2 Crystals and Sintered Compacts". Atomic Energy of Canada Limited. AECL-692, 1958.
(1453) White, R.J., "The Development of Grain-Face Porosity in Irradiated Oxide Fuel." Journal of Nuclear Materials, V325, pp 61-77, 2004.

(1454) Hargreaves, R., et al. "A Quantitive Model for Fission Gas Release and Swelling in Irradiated Uranium Dioxide." J. Br. Nucl. Energy Soc., V15 No. 4, pp 311-318, October, 1976.
(1455) Chernikov, A.S., et al. "Fission Product Diffusion in Fuel Element Materials for HTGR." I.V. Kurchatov Institute of Atomic Energy, Moscow, pp 170-181.
(1456) Rondinella, V.V., et al. "Radiation Damage and Simulated Fission Product Effects on the Properties of Inert Matrix Materials." Progress in Nuclear Energy, V38 No. 3-4, pp 291-294, 2001.
(1457) K Une, et al. "Effects of Additives and the Oxygen Potential on the Fission Gas Diffusion in UO2 Fuel." Journal of Nuclear Materials. V 150, 1987, p 93-99.
(1458) JDB Lambert, et al. "Perfromance of Mixed-Oxide Fuel Elements – ANL Experience." Argonne National Laboratory.
(1459) Y Harada and S Doi. "Irradiation Behavior of Large Grain UO2 Fuel Rod by Active Powder." Journal of Nuclear Science and Technology. V35, No 6, 1998.
(1460) D Ohai. "Large Grain Size UO2 Sintered Pellets Obtaining Used for Burnup Extension." Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology, Prague, Czech Republic, August 2003.
(1461) PT Sawbridge, et al. "The irradiation performance of magnesia-doped UO2 fuel." Journal of Nuclear Materials, V95, 1980, p 119-128.
(1462) BM Rosenbaum and G Allen. "Calculation of internal pressures in the fuel tube of a nuclear reactor." NASA Lewis Flight Propulsion Laboratory, July 1952.
(1463) H Assmann, et al. "Letter to the Editors: Doping with Niobia – Beneficial or Not?" Journal of Nuclear Materials. V 98, 1981, p 216-220.
(1464) JR Lindgren, et al. "Irradiation testing and development of fast breeder reactor (U,Pu)O2 fuel rods."
(1465) BF Rubin and KJ Perry. "The irradiation of urania-plutonia fuel to 125,000 MWd/Te in a thermal flux." General Electric Co, GEAP-5671, March 1969.
(1466) CN Craig, et al. "Steady State Performance of PuO2-UO2 Fast reactor Fuels." General Electric Co.
(1467) JH Norman, et al. "Spheres: Diffusion-Controlled Fission Product Release and Absorption." Advances in Chemistry, Series 93: Radionuclides in Environment, 1990.
(1468) H Kleykamp. "The chemical state of the fission products in oxide fuels." Journal of Nuclear Materials. V131, 1985, p 221-246.
(1469) K Verfondern and D Müller. "Modeling of Fission Product Release Behavior from HTR Spherical Fuel Elements Under Accident Conditions."
(1470) JA Turnbull, et al. "The diffusion coefficients of gaseous and volatile species during the irradiation of uranium dioxide." Journal of Nuclear Materials. V107, 1982, p 168-184.
(1471) RM Carroll and PE Reagan. "Techniques for in-pile measurements of fission-gas release." Nuclear Science and Engineering. V 21, 1965, p 141-146.
(1472) RM Carroll, et al. "Release of fission gas during fissioning of UO2." Journal of the American Ceramic Society. V48, No 2, 1965.
(1473) L Väth. "Letter to the Editors: Approximate Treatment of the Grain-Boundary Loss Term in Fission Gas Release Models." Journal of Nuclear Materials. V99, 1981, p 324-326.
(1474) K Verfondern, et al. "Methods and data for HTGR fuel performance and radionuclide release modeling during normal operation and accidents for safety analyses." Research Center Jülich mbH-ISR, June 1992.
(1475) Forsberg, K. and A.R. Massih, "Diffusion Theory of Fission Gas Migration in IrradiatedNuclear Fuel UO2". Journal of Nuclear Materials, 1985. pp. 135.
(1476) Berna, G.A., et al., FRAPCON-3: A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup, Vol. 2 PNNL-11513. 1997, Pacific Northwest National Laboratory.

(1477) Speight, M.V., "A Calculation of the Migration of Fission Gas in Material Exhibiting Precipitation and Re-solution of Gas Atoms Under Irradiation." Nuclear Science and Engineering, 1969. 37: p. 180-185.
(1478) Ham, F.S., "Theory of Diffusion -Limited Precipitation." Journal of Physics and Chemistry of Solids, volume 6. 1958. p. 335-351.
(1479) Baker, C., "The Fission Gas Bubble Distribution in Uranium Dioxide from High Temperature Irradiated SGHWR Fuel Pins." Journal of Nuclear Materials, 1977. 66: p. 283-291.
(1480) White, R.J. and M.O. Tucker, "A New Fission-Gas Release Model." Journal of Nuclear Materials, Volume 118, 1983. p. 1-38.
(1481) Turnbull, J.A., "The Effect of Grain Size on the Swelling and Gas Release Properties of UO2 During Irradiation." Journal of Nuclear Materials, Volume 50, 1974. p. 62-68.
(1482) Dowling, D.M., R.J. White, and M.O. Tucker, "The Effect of Irradiation -Induced Re-Solution on Fission Gas Release." Journal of Nuclear Materials, V 110 1982. p. 37-46.
(1483) Reynolds, G.L., W.B. Beere, and P.T. Sawbridge, "The Effect of Fission Products on the Ratio of Grain-Boundary Energy to Surface Energy in Irradiated Uranium Dioxide." Journal of Nuclear Materials, V41 1971. p. 112.
(1484) Lanning, D.D., C.E. Beyer, and C.L. Painter, "FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Applications, Vol. 4" PNNL-11513. 1997, Pacific Northwest National Laboratory.
(1485) Forsberg, K., F. Lindstrom, and A.R. Massih. "Modeling of Some High Burnup Phenomena in Nuclear Fuel, paper 2.5. in Technical Committee Meeting on Water Reactor Fuel Element Modeling at High Burnup, and Experimental Support." 1994. Windermere, England, IWGFPT/41: International Atomic Energy Agency, Vienna, Austria.
(1486) Carrol, R.M., O. Sisman, and R.B. Perez, "The Effects of Fission Density on Fission-Gas Release." Nuclear Science and Engineering, V32 1968. p. 430.
(1487) Carrol, R.M. and O. Sisman, "In-Pile Fission-Gas Release from Single Crystal UO2." Nuclear Science and Engineering, V 21, 1965. p. 147-158.
(1488) Turnbull, J.A., "A Review of Irradiation Induced Re-solution in Oxide Fuel." Radiation Effects, V53, 1980. p. 243.
(1489) Nichols, F.A., "Transport Phenomena in Nuclear Fuels." Journal of Nuclear Materials, V84, 1979. p. 1-25.
(1490) Forsberg, K. and A.R. Massih. "Theory of Fission Gas Release During Grain Growth." in Transactions SMiRT V16. 2001.
(1491) Friskney, C.A. and J.A. Turnbull, CEGB Report No. RD/B/N4217. 1980.
(1492) Turnbull, J.A., et al., CEGB Report No. RD/B/N4892. 1980.
(1493) Turnbull, J.A. and C.A. Friskney, "The Relationship Between Microstructure and the Release of Unstable Fission Products During High Temperature Irradiation of Uranium Dioxide." Journal of Nuclear Materials V71, 1978. p. 238.

## 11.47 $UO_2$ Swelling

(1494) Chubb, W., Hilbert, R.F., Storhok, V.W. and Keller, D.L. "The Relationship of Mechanical Properties to the Swelling of Oxide Fuels at High Temperatures." Battelle Memorial Institute.
(1495) Nelson, R.C., Baumgartner, J.A., Perry, K.J., Zebroski, E.L. "Irradiation Induced Swelling Rates of PuO2-UO2 Fuel with Strong Radial Restraint." General Electric Co., GEAP 13686. March 1971.
(1496) Assmann, H and Manzel, R. "The Matrix Swelling Rate of UO2." Journal of Nuclear Materials. V68, 1977. pp. 360-364.

(1497) Bellamy, R.G. and Rich, J.B. "Grain-Boundary Gas Release and Swelling in High Burn-up Uranium Dioxide", Atomic Energy Research Establishment, Harwell, Didcot, Berks, UK. April 1969.
(1498) Koo, Y., Lee, B., Cheon, J., and Sohn, D. "Pore pressure and swelling in the rim region of LWR high burnup UO2 fuel." Journal of Nuclear Materials. V295, 2001. pp. 213-220.
(1499) Duncan, R.N., Cantley, D.A., Perry, K.J., and Nelson, R.C. "Fuel Swelling - Fast Reactor Mixed-Oxide Fuels" General Electric Co., Breeder Reactor Dev. Operation.
(1500) Paraschiv, MC., et al., "On the nuclear oxide fuel densification, swelling and thermal re-sintering." Journal of Nuclear Materials. V302, 2002. pp 109-124.
(1501) NUREG/CR-6534, Vol. 1 "Modifications to the MATPRO Model for Fuel Swelling."
(1502) JA Turnbull. "The effect of grain size on the swelling and gas release properties of UO2 during irradiation." Journal of Nuclear Materials. V50, 1974, p 62-68.
(1503) W Chubb, et al. "Factors affecting the swelling of nuclear fuels at high temperatures." Nuclear Technology. V18, 1973, p 231-255.

## 11.48 $UO_2$ Thermal Conductivity

(1504) Amaya, M. et al. "Thermal Conductivities of Irradiated U02 and (U, Gd) 02 Pellets." Nupec. Sponsored by the Ministry of International Trade and Industry (MITI).
(1505) Fink, JK. "Thermal Conductivity and Thermal Diffusivity of Solid UO2," Preliminary Recommendation, Version 1. July 1999.
(1506) Ronchi, C. et al. "Effect of burn-up on the thermal conductivity of uranium dioxide up to 100.000 MWdt-1." Journal of Nuclear Materials. V327, 2004. pp 58-76.
(1507) Lemehov, SE. et al. "Modelling thermal conductivity and self-irradiation effects in mixed oxide fuels." Journal of Nuclear Materials. V320, 2003. pp 66-76.
(1508) Hobson, IC. et al. Effect of porosity and stoichiometry on the thermal conductivity of uranium dioxide." J. Phys. D: Appl. Phys. V7, 1974. pp 1003-1015

## 11.49 $UO_2$ Thermal Expansion

(1509) Martin, DG. "The Thermal Expansion of Solid UO2 and (U, Pu) Mixed Oxides – A Review and Recommendations." Journal of Nuclear Materials. V152, 1988. pp 94-101.

## 11.50 General Fuel Fabrication

(1510) JL Collins, et al. "The Basic Chemistry Involved in the Internal Gelation Method of Precipitating Uranium as Determined by pH Measurement." Radiochemica Acta. V42, 1987. p 121-134.
(1511) CM King, et al. "New Insights into Uranium (VI) Sol-Gel Processing." Westinghouse Savannah River Co. WSRC-MS-90-116. 1990.
(1512) MH Lloyd, et al. "Crystal Habit and Phase Attribution of U (VI) Oxides in a Gelation Process." Journal of Inorganic Nuclear Chemistry. V38, 1976. p 1141-1147.
(1513) HP Alder, et al. "Advanced Fuel for Fast Breeder Reactors Produced by Gelation Methods." Swiss Federal Institute for Reactor Research, Würenlingen, Switzerland. 1988.
(1514) CM King, et al. "Magnetic Resonance as a Structural Probe of a Uranium (VI) Sol-Gel Process." Westinghouse Savannah River Co. WRSC-MS-89-1285. 1989.
(1515) C Ganguly. "Sol-Gel Microsphere-Pelletization Process for Fabrication of Conventional and Advanced Ceramic Nuclear Fuels." Metals Materials and Processes. V1, No. 4, 1990. p 253-274.
(1516) G Pautasso, et al. "Investigation of the Reaction UO2+x + PuO2 + C + N2 by Thermogravimetry." Journal of Nuclear Materials. V158, 1988. p 12-18.

(1517) P Bardelle and D Warin. “Mechanism and Kinetics of the Uranium-Plutonium Mononitride Synthesis.” Journal of Nuclear Materials. V188, 1992. p 36-42.
(1518) T Nakagawa, et al. “Formation of Uranium and Cerium Nitrides by the Reaction of Carbides with NH3 and N2/H2 Stream.” Journal of Nuclear Materials. V247, 1997. p 127-130.

## 11.51 Compatibility/Fission Product Interactions

(1519) M Kangilaski, EO Fromm, DH Lozier, VW Storhok, and JE Gates, “High Temperature Irradiation of Niobium – 1 w/o Zirconium Clad UO2”, BMI-1730, June 28, 1965.
(1520) RL Pearson, TB Lindemer, EC Beahm, “Simulated Fission Product-SiC Interaction in TRISO-Coated LEU or MEU HTGR Fuel Particles”, ORNL/TM-6991, November 1980.
(1521) TN Tiegs, “Fission Product Pd-SiC Interaction in Irradiated Coated-Particle Fuels”, Nuclear Technology. V57, 1982. pp 380-398.
(1522) K Minato, T Ogawa, K Sawa, H Sekino, T Koya, I Kitagawa, A Ishikawa, T Tomita, E Oeda, “ZrC-Coated Particle Fuel for High Temperature Gas-Cooled Reactors”, JAERI (unpublished).
(1523) DS Dutt, CM Cox, and MK Millhollen. “Performance of Refractory Alloy Clad Fuel Pins.” HEDL-SA-3188, 1984.
(1524) EK Storms and DG Czechowicz, Journal of Nuclear Materials. V166, 1989.
(1525) RA Karnesky, “PIE Data from the SP-2 Fuel Pins”, WHC-SP-0341, 1988.
(1526) KR Thoms, “Design, Fabrication, and Operation of Capsules for the Irradiation Testing of Candidate Advanced Space Reactor Fuel Pins”, ORNL-TM-4825/NASA-CR-134592, April 1975.
(1527) DR Cuneo, EL Long Jr, A Jostons, and TN Washburn, “Examination of Irradiated Uranium Nitride Fuel Clad with Tungsten-Rhenium or T-111 Alloy”, ORNL-TM-3895, October 1972.
(1528) GK Watson, “Evaluation of Tantalum Alloy Clad Uranium Mononitride Fuel Specimens from 7500 Hour, 1040°C Pumped-Lithium Loop Test”, NASA-TN-D-7619, April 1974.
(1529) RA Karnesky, “Postirradiation Examination Data from the SP-2 Fuel Pins”, WHC-SP-0341, 1988.
(1530) JB Holt and MY Almassy. “Nitrogen Diffusion in Uranium Nitride as Measured by Alpha Particle Activation of N15.” Journal of the American Ceramic Society. V52, 1969. pp 631-635.
(1531) TJ Sturiale and MA DeCrescente, “Self-Diffusion of Nitrogen on Uranium Mononitride”, USAEC Report PWAC-477, Pratt & Whitney Aircraft, September 1965.
(1532) BJ Makenas, JW Hales, RA Karnesky, “Post-Irradiation Data from the SP-1 Test at 3 Atom Percent Burnup”, LANL Report WHC-SP-1050, Los Alamos National Laboratory, September 1993.
(1533) BJ Makenas, “Post-Irradiation Data of the SP-3R Test ”, LANL Report WHC-SP-1052, Los Alamos National Laboratory, August 1993.
(1534) McDonald, G.E. “Irradiation of TZM-Uranium Dioxide Fuel Pin AT 1700K.” NASA Lewis Research Center, NASA TM X-2755, July 1973.
(1535) SC Weaver, RL Senn, JL Scott, and BH Montgomery, “Effects of Irradiation on Uranium Nitride under Space Reactor Conditions”, ORNL4461, Oct 1969.
(1536) TD Gulden and H Nickel. “Preface: Coated Particle Fuels.” Nuclear Technology, V35, 1977, pp 206-213.
(1537) Nickel, H. et al. “Concluding Remarks Coated Particle Fuels.” Nuclear Technology, V35, 1977, pp 567-573.
(1538) Storms, E.K. “Final Report of Material Compatibility Study (5.4.4).” 2157B.
(1539) K. Minato, et al. “Advanced coatings for HGTR fuel particles against corrosion of SiC layer”. Journal of Nuclear Materials V246, 1997, p 215-222.

(1540) K. Minato, et al. “Fission Product Palladium-Silicon Carbide Interaction in HGTR Fuel Particles”. Journal of Nuclear Materials. V172, 1990, p 184-196.
(1541) D'Eye. "The Development of SiC Clad Fuel Pins for Advanced Gas-Cooled Reactors" SM-111/35
(1542) H. Grübmeier, et al. "Silicon carbide corrosion in high-temperature gas-cooled reactor fuel particles". Nuclear Technology. V35, 1977, p 413-427.
(1543) RJ Lauf. “Out-of-reactor studies of fission product-silicon carbide interactions in HTGR fuel particles.” Journal of Nuclear Materials. V120, 1984, p 6-30.
(1544) JS Chen, et al. “Stability of rhenium thin films on single crystal (001) $\beta$-SiC.” Journal of Applied Physics. V75, No 2, 1994, p 897-901.
(1545) DC Fee, et al. “Chemical interaction between fuel and cladding in stainless steel clad fast reactor fuels.” Argonne National Laboratory, ANL-75-53, October 1975.
(1546) W Batey and KQ Bagley. “Fuel/clad reactions in irradiated oxide fuel pins.” BNSEJ. V13, 1974, p 49-61.
(1547) AS Chernikov, et al. “Behavior of HTGR coated fuel particles in high-temperature tests.” Energy. V 16, No 1/2, 1991, p 295-308.
(1548) GE McDonald. “Irradiation of TZM-Uranium Dioxide Fuel Pin at 1700K.” NASA Lewis Research Center, NASA TM X-2755, July 1973.
(1549) KJ Bowles and RE Gluyas. “Evaluation of refractory metal clad uranium nitride and uranium dioxide fuel pins after irradiation for times up to 10,450 hours at 990˚C”. NASA report TN D-7891.
(1550) H Nabielek, et al. “The performance of high-temperature reactor fuel particles at extreme temperatures.” Nuclear Technology. V84, 1989, p 62-81.

## 11.52 Pellet-Cladding Interactions

(1551) Jernkvist, Lars Olof. “A Model for Predicting Pellet-Cladding Interaction-Induced Fuel Rod Failure.” Nuclear Engineering and Design, V156, pp 393-399, 1995.
(1552) Massih, A.R., et al. “Analyses of Pellet-Cladding Mechanical Interaction Behaviour of Different ABB Atom Fuel Rod Designs.” Nuclear Engineering and Design, V156, pp 383-391, 1995.
(1553) Bernaudat, C. “Mechanical Behaviour Modelling of Fractured Nuclear Fuel Pellets.” Nuclear Engineering and Design, V156, pp 373-381, 1995.

## 11.53 SP-100 reports – Limited Access

(1554) E Storms. “Compatibility Issues in the SP-100 Fuel System.” Los Alamos National Laboratory, LA-11468-MS, September 1989.
(1555) EK Storms an DG Czechowicz. “Compatibility studies of high temperature fuels and cladding.” Los Alamos National Laboratory, LA-11571-MS.
(1556) DE Plumlee. “Fuel Pin Behavior Lessons Learned.” Martin Marietta SP-100 Program Information Release, PIR 1202, March 2, 1994.
(1557) RB Matthews, et al. “Uranium Nitride fuel development for the SP-100 ground engineering system project.” Los Alamos National Laboratory, LA-10869-MS, July 1987.
(1558) E Storms. “Internal Memo: Fuel composition specification.” Los Alamos National Laboratory, April 20, 1987.
(1559) RB Matthews, et al. “Fabrication and testing of uranium nitride fuel for space power reactors.” Los Alamos National Laboratory, LA-10636, June 1986.
(1560) E Storms and R Mason. “Internal Memo: Liquid formation in fuel pins during reactor tests.” Los Alamos National Laboratory, April 1, 1986.
(1561) RE Mason and RB Matthews. “Compatibility in Space Reactor Fuel Systems.” Los Alamos National Laboratory, LA-11071-MS, March 1988.

(1562) RB Matthews. "Compatibility testing of fuel and cladding components." Internal communication, Los Alamos National Laboratory, February 20, 1987.

## 11.54 Cermets/Dispersion Fuel

(1563) WE Gurwell, et al. "Molybdenum-Base Cermet Fuel Development." Proceedings of the 5th Symposium on Space Nuclear Power Systems, Albuquerque, NM, January 11, 1988, CONF-880122-7.
(1564) Brengle, R.G., Harty, R.B., and Bhattacharyya, S.K. "The Promise and Challenges of Cermet Fueled Nuclear Thermal Propulsion Reactors."Proceedings of the AIAA/SAE/ASME/ASEE 29th Joint Propulsion Conference and Exhibit, June 28-30, 1993. Monterey, CA. AIAA-93-2111.
(1565) Downar, T.J., Revankar, S.T., Solomon, A.A., McDeavitt, S.M., and Kim, T.K. "Thoria-Based Cermet Nuclear Fuel: Neutronics Fuel Design and Fuel Cycle Analysis." Presented at 10th International Conference on Nuclear Engineering, Arlington, VA, April 2002. ICONE10-22305.
(1566) Hofman, G.L. "High Density Dispersion Fuel." Argonne National Laboratory. September 19, 1996.
(1567) Trybus, C.L., Meyer, M.K., Clark, C.R., Wiencek, T.C., and McGann, D.J. "Design and Fabrication of High Density Uranium Dispersion Fuels", CONF-9710102, November 18, 1997.
(1568) Hayes, S.L., Trybus, C.L., and M.K. Meyer. "Irradiation Testing of High Density Uranium Alloy Dispersion Fuels." CONF-9710102. November 4, 1997.
(1569) Bretscher, M.M., Matos, J.E. and Snelgrove, J.L., "Relative Neutronic Performance of Proposed High-Density Dispersion Fuels in Water-Moderated and D2O-Reflected Research Reactors." Presented at the 1996 International Meeting on Reduced Enrichment for Research and Test Reactors, Oct. 7-10, 1996, Seoul, Korea, CONF-9610205--1.
(1570) DA Seifert, et al. "Fission Product Behavior within two W-UO2 Cermet Fuel Elements Irradiated in a Temperature Gradient." August 1968.
(1571) "Thermal Conductivity of Coated Particle Uranium Dioxide Tungsten Cermets" - General Electric Co. Aug 1968.
(1572) Collins, J.F. and Flagella, P.N. "Fabrication and Measurement of Properties of Mo-UO2 Cermets." General Electric Co., July 7, 1967.
(1573) Barner, J.O., Coomes, E.P., Williford, R.E., and Neimark, L.A. "Cermet Fuels for Space Power Systems." November 1985.
(1574) Keller, D.L., Cunningham, G.W., Murr, W.E., Fromm, E.O., and Lozier, D.E. "High-Temperature Irradiation Test of UO2 Cermet Fuels" Battelle Memorial Institute, BMI-1608. January 7, 1963.
(1575) Tummala, R.R. and Friedberg, A.L. "Thermal Expansion of Cermets Containing Uranium Dioxide." University of Illinois, Oct. 21, 1968.
(1576) Weimar, P., Thummler, F., and Bumm, H. "UO2-Cermets with Idealized Structure by Particle Coating and Isostatic Hot Pressing." December 1968.
(1577) "Development status of metallic, dispersion and non-oxide advanced and alternative fuels for power and research reactors." IAEA-TECDOC-1374, September 2003.
(1578) Anderson, R.C., et al. "Status Report on Irradiation Studies of Mo 40 v/o UO2 Cermet Fuel." Los Alamos National Laboratory, June 1965.
(1579) Schuyler, D.R. and Gregory, T.G. "The Metallographic Preparation and Microscopic Examination of a Molybdenum Uranium-Dioxide Cermet Fuel." Los Alamos National Laboratory.
(1580) AJ Patrick. "Gas-Collection Technique for Measurement of Fission-Gas Retention in Irradiated MO-UO2." Los Alamos National Laboratory, May 19, 1967.

(1581) Feldman, B.C., "The Processing and Testing of Mo-UO2 Fuel Elements." Los Alamos National Laboratory, Aug. 23, 1973.
(1582) WA Ranken and WH Reichelt. "Behavior of Tungsten-Clad MO-UO/SUB 2/Fuel Under Neutron Irradiation at High Temperature." Los Alamos National Laboratory, 1970.
(1583) Takkunen, Phillip D. "Fabrication of Cermets of Uranium Nitride and Tungsten or Molybdenum from Mixed Powders and from Coated Particles." National Aeronautics and Space Administration. NASA TN D-5136, April, 1969.
(1584) Collins, J.F., et al. "Evaluations of Uranium Mononitride Cermet Fuel." GEMP-659, October, 1968.
(1585) Pratt & Whitney Aircraft. "A Uranium Mononitride-Columbium Cermet Fuel: Status Report." Contract AT(30-1)-2789, TIM No. 698, July, 1961.
(1586) Weber, C.E. "Progress on Dispersion Elements." Progress in Nuclear Energy, Series V, V2, pp 295-362, 1959.
(1587) WH Lenz, et al. "The Processing and Testing of Mo-UO2 Fuel Elements." Los Alamos National Laboratory, LA-2429. December 1960.
(1588) AS Chernikov, et al. "Fuel Elements Based on Spherical Fuel Pellets with a Protective Coating for Enhanced-Safety Reactors." Atomic Energy. V 87, No 6, 1999.
(1589) WA Ranken, et al. "Retention of Fission Gases in the UO2 Phase of Mo-UO2 Cermets Irradiated at High Temperatures." Proceedings of the Second International Conference on Thermionic Electrical Power Generation, Stresa, Italy, May 1968.
(1590) RG Brengle, et al. "The Promise and Challenges of Cermet Fueled Nuclear Thermal Propulsion Reactors." AIAA/SAE/ASME/ASEE 29th Joint Propulsion Conference and Exhibit, Monterey, CA, June 1993.
(1591) J Rest, et al. "Analysis of the swelling behavior of U-alloys." Proceedings of the 20th International Meeting on Reduced Enrichment for Research and Test Reactors, Jackson Hole, WY, October 1997.
(1592) "Thermal conductivity of coated particle uranium dioxide tungsten cermets." General Electric Co. August 1968.
(1593) H Schneider and D Schonwald. "The deposition of molybdenum on ZrO2 and UO2 spheres from the gas phase." Monsanto Research Company, MLM-1508, May 1968.
(1594) P Weimar and H Zimmermann. "Oxide-Cermet Fuel Rods: Production, Properties, and Irradiation Behavior." Institut für Material-und Festkörperforschung. August 1973.
(1595) S. Maxwell, et al. "$\gamma$ Lattice Parameters in U-Mo-Nb Alloys." Journal of Nuclear Materials. V 11, No 1, 1964, p 119-120.
(1596) JL Snelgrove, et al. "Development of very high density fuels by the RERTR program." Presented at the 1996 International Meeting on Reduced Enrichment for Research and Test Reactors, Seoul, Korea, October 1996.
(1597) MK Meyer, et al. "Selection and Microstructures of High Density Uranium Alloys." Presented at the International Meeting on Reduced Enrichment for Research and Test Reactors, Jackson Hole, WY, October 1997.
(1598) DL Newsom and JF Collins. "Irradiation testing and evaluation of refractory metal fuel elements of 710 fast reactor program." General Electric Co., October 1969.

## 11.55 SIMFUEL

(1599) Lucuta, PG, Palmer, BJ, Matzke, HJ, Hartwig, DS, "Preparation and Characterization of SIMFUEL: Simulated CANDU High-burnup Nuclear Fuel", 2nd International Conference on CANDU Fuel, 1989, pp 132-146
(1600) Verrall, RA, Matzke, HJ, Hastings, IJ, Ray, ILF, Rose, DH, "Fission Gas Mobility in UO2, Simulating A Burnup of 30 MWd/kg-U", 2nd International Conference on CANDU Fuel, 1989, pp 172-186

(1601) Sengupta, AK, et al. "Some Important Properties of Simulated UO2 Fuel", Bhabha Atomic Research Center (BARC), Government of India, BARC/1999/E/008, April, 1999
(1602) Lucuta, PG, Matzke, HJ, Hastings, IJ, "A Pragmatic Approach to Modeling Thermal Conductivity of Irradiated UO2 Fuel: Review and Recommendations", Journal of Nuclear Materials, Vol. 232, 1996, pp 166-180
(1603) Lucuta, PG, Matzke, HJ, Verrall, "Modeling of UO2-based SIMFUEL Thermal Conductivity: The Effect of the Burnup", Journal of Nuclear Materials, Vol. 217, 1994, pp 279-286
(1604) Lucuta, PG, Matzke, HJ, Verrall, RA, Tasman, HA "Thermal Conductivity of SIMFUEL", Journal of Nuclear Materials, Vol. 188, 1992, pp 198-204
(1605) Lucuta, PG, Verrall, RA, Matzke, HJ, Palmer, BJ, "Microstructural Features of SIMFUEL—Simulated High-Burnup UO2-Based Nuclear Fuel", Journal of Nuclear Materials, Vol. 178, 1991, pp 48-60
(1606) Lucuta, PG, Matzke, HJ, Verrall, RA, "Thermal Conductivity of Hyperstoichiometric SIMFUEL", Journal of Nuclear Materials, Vol. 223, 1995, pp 51-60
(1607) Verrall, RA, Lucuta, PG, "Specific Heat Measurements of UO2 and SIMFUEL", Journal of Nuclear Materials, Vol. 228, 1996, pp 251-253
(1608) Matzke, HJ, Lucuta, PG, Verrall, RA, "Formation and Behavior of Barium Silicate in UO2-based SIMFUEL", Journal of Nuclear Materials, Vol. 185, 1991, pp 292-296
(1609) Matzke, HJ, Verrall, RA, "Release of Volatile Fission Products From ThO2 with a Simulated Burnup of 4 At%", Journal of Nuclear Materials, Vol. 182, 1991, pp 261-264
(1610) Muromura, T, Adachi, T, Takeishi, H, Yoshida, Z, Yamamoto, T, Ueno, K, "Metallic Phases Precipitated in UO2 Fuel, Part I. Phases in Simulated Fuel", Journal of Nuclear Materials, Vol. 151, 1988, pp 318-326
(1611) Adachi, T, Muromura, T, Takeishi, H, Yamamoto, T, "Metallic Phases Precipitated in UO2 Fuel, Part II. Insoluble Residue in Simulated Fuel", Journal of Nuclear Materials, Vol. 160, 1988, pp 81-87
(1612) Bevilacqua, AM, Matzke, HJ, "Mechanical Properties of UO2 and SIMFUEL to High Temperature" (translated from Spanish), downloaded from Argentinean government website www.cab.cnea.gov.ar/AATN99/Actas/Docs/ R2C54.pdf
(1613) Amaya, M, Hirai, M, Sakurai, H, Ito, K, Sasaki, M, Nomata, T, Kamimura, K, Iwasaki, R, "Thermal Conductivities of Irradiated UO2 and (U,Gd)O2 Pellets", downloaded from Japanese Nupec Organization website www.nupec.or.jp/database/paper/paper-_12/p12_fuel/R12-07-05.htm

## 11.56 $UO_2$ Database Development

(1614) Chantoin, P.M., Sartori, E., and Turnbull, J.A. "The Compilation of a Public Domain Database on Nuclear Fuel Performance for the Purpose of Code Development and Validation."
(1615) Turnbull, JA. "Review of Nuclear Fuel Experimental Data." Nuclear Energy Agency Organization for Economic Co-Operation and Development. January, 1995.
(1616) NEA Nuclear Science Committee Task Force. "Scientific Issues in Fuel Behavior." Nuclear Energy Agency Organization for Economic Co-Operation and Development. January 1995
(1617) Menut, P. et al. "The Public Domain Database on Nuclear Fuel Performance Experiments (IFPE) for the Purpose of Code Development and Validation." International Fuel Performance Experiments Database. Status 31, August, 2004.

## 11.57 General Overviews of Reactor and Fuel Systems

(1618) Cox, C.M., Dutt, D.S. and Karnesky R.A. "Fuel Systems for Compact Fast Space Reactors", Chapter 36, Ref., 367, File ACS-9, 1984.
(1619) DL Keller, "Annual Report: Progress on Development of Materials and Technology for Advanced Reactors During July 1968-June 1969", USAEC Report BMI-1868, Battelle Memorial Institute, pp A15-A19.
(1620) DL Keller, "Annual Report: Progress on Development of Fuels and Technology for Advanced Reactors During July 1970-June 1971", USAEC Report BMI-1918, Battelle Memorial Institute, pp A5-A9.
(1621) Koenig, D.R. "Experience Gained from the Space Nuclear Rocket Program (Rover)." Los Alamos National Laboratory, LA-10062-H, UC-33, 1986.
(1622) Lindgren, J.R., Flynn, P.W., N.L. Baldwin, Fitts, R.B. and Longest, Jr., A.W. "Irradiation Testing and Development of Fast Breeder Reactor (U, Pu)O2 Fuel Rods."
(1623) Matzke, Hj. "Fuel Research and Basic Aspects of Fuel In-Pile Performance." Journal of Nuclear Materials, V166, pp 120-131, 1989.
(1624) SM Oggianu and MS Kazimi. "A Review of Properties of Advanced Nuclear Fuels." MIT-NFC-TR-021, February, 2000.
(1625) Anghaie, Dr. Samim. "Overview of Prismatic Fuel Development and Test Data." - Presented at STAIF 2004.
(1626) SF Demuth. "SP-100 Space Reactor Design." Progress in Nuclear Energy, V42, No 3, 2003, p. 323-359.
(1627) "Development status of metallic, dispersion, and non-oxide advanced and alternative fuels for power and research reactors." International Atomic Energy Agency, IAEA-TECDOC-1374. September 2003.
(1628) JT Walton. "An Overview of Tested and Analyzed NTP Concepts." AIAA 91-3503, AIAA/NASA/OAI Conference on Advanced SEI Technologies, Cleveland, OH, September 1991.
(1629) FG Reshetnikov. "Status of the development and production of uranium-plutonium fuel for fast reactors." Atomic Energy. V91, No 6, 2001, p 998-1002.
(1630) AV Zrodnikov, et al. "SVBR-75/100 multipurpose modular low-power fast reactor with led-bismuth coolant." Atomic Energy. V97, No 2, 2004, p 528-533.
(1631) H Blank, et al. "Dense Fuels in Europe." Journal of Nuclear Materials, V166, 1989, p 95-104.
(1632) DR Olander. Fundamental Aspects of Nuclear Reactor Fuel Elements. 1976.
(1633) JAG Holmes. "Design of Oxide Fuel for Fast Reactors." IAEA-SM-173/54.
(1634) WH Childs. "Thermophysical Properties of Selected Space-Related Materials, Vol. 1 and 2." TOR-0081(6435-02)-1.

## 11.58 PIE of Existing Fuel (Blanket Fuel)

(1635) J.C. Clayton, B.F. Kammenzind, P. Senio, J. Sherman. "End-of-Life Destructive Examination of Zircaloy Maximum Depletion Blanket Fuel Plates from the Shipping port PWR Core 2". Presented at the 6th International Symposium on Environmental Degradation of Materials in Nuclear Power Systems-Water Reactors; San Diego, CA; August 1-5, 1993.
(1636) J.C. Clayton. "The Shipping port Pressurized Water Reactor and Light Water Breeder Reactor". Presented at the 25th Central Regional Meeting of the American Chemical Society; Pittsburgh, PA; October 4-6, 1993.
(1637) W.J. Babyak, L.B. Freeman, and H.F. Rabb, Jr. "LWBR: A successful demonstration completed." Nuclear News. V31, No 12, 1988, p 114-116.
(1638) F.O. VonPlinsky. "Metallographic Examination of PWR Seed 2 Blanket 1 Fuel Rods." NRFE-DR-186, January 1962.

(1639) R.K. Steunenberg and L. Burris. "From test tube to pilot plant, a 50 year history of the Chemical Technology Division at Argonne National Laboratory." Argonne National Laboratory. ANL-00/16. September 18, 2000.
(1640) J.O Dittmer, et al. "FFTF Metal Fuel Pin Fabrication." Materials Science and Technology: General.

## 11.59 Miscellaneous Fuel

(1641) Meyer, M.K., Trybus, C.L., Hofman, G.L., Frank, S.M./, and Wiencek, T.C. "Selection and Microstructures of High Density Uranium Alloys." CONF-9710102, Nov. 4, 1997.
(1642) Hofman, G.L., Trybus, C.L., Snelgrove, J.L.,and Wiencek, T.C. "Development of Very-High-Density Fuels By The RERTR Program." Presented at the 1996 International Meeting on Reduced Enrichment for Research and Test Reactors, Oct. 7-10, Seoul, Korea. Dec. 9, 1996.
(1643) Ford, I.J. "Rupture and Fragmentation of Pressurized Pipes and Fast Reactor Fuel Pins." Nuclear Engineering and Design, V156, pp 401-410, 1995.
(1644) FW Wiffen. "Effects of Irradiation on Properties of Refractory Alloys with Emphasis on Space Power Reactor Applications." Proceedings of the Symposium on Refractory Alloy Technology for Space Nuclear Power Applications, Oak Ridge, TN, August 10-11, 1983. January 1984.
(1645) N Saunders, et al. "Feasibility Study of a Tungsten Water-Moderated Nuclear Rocket" NASA Lewis Research Center.
(1646) H Feinroth, et al. "Progress in developing an impermeable, high temperature ceramic composite for advanced reactor clad and structural applications."
(1647) F Lee and J Moteff. "Anomalous creep behavior of neutron-irradiated molybdenum." University of Cincinnati.
(1648) RE Zielinski. "A summary report of the compressive creep properties of irradiated and unirradiated molybdenum." Monsanto Research Corporation, MLM-2406, April 1977.
(1649) WA Maxwell. "Preliminary investigation of plate-type molybdenum disilicide fuel elements for an air-cycle nuclear reactor." NASA Lewis Flight Propulsion Laboratory, NACA RM E52L18, March 1953.
(1650) LK Mansur, et al. "Radiation effects on microstructures and properties of irradiated materials." Oak Ridge National Laboratory.
(1651) AH Bremser and HH Moeller. "High Temperature Fuel/Emitter System for Advanced TFEs." Babcock and Wilcox Company Contract # CRD-1322 for Government Contract # DE-FG01-95NE32186, March 1996.
(1652) DT Goodin. "Accident Condition Performance of Fuels for High-Temperature Gas-Cooled Reactors." Journal of the American Ceramic Society. V65, No 5, 1982.
(1653) P MacDonald and P Martin. "Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels." International Nuclear Energy Research Initiative (INERI) Proposal for Collaborative Work with the French CEA, September 2001.
(1654) D Mosedale, et al. "Cladding and structural materials for fuel components and assemblies." IAEA-SM-173/55.

## 11.60 Alternator Electrical Feedthrough Insulator Materials

(1655) NGST Report 04.1F102.JM.006, "Task 2 Conceptual Design Summary Report, Jupiter Icy Moons Orbiter (JIMO) Phase A Study, JPL Contract N. 120530," Aug. 13, 2004.
(1656) CeramTec North America Corporation, company website, http://www.ceramaseal.com, viewed Sep. 2005.
(1657) Latronics Corporation, company website, http://www.latronicscorp.com, viewed Nov. 2005.

(1658) MDC Vacuum Products Corp., company website, http://www.mdcvacuum.com, viewed Nov. 2005.
(1659) Solid Sealing Technology, Inc., company website, http://www.solidsealing.com, viewed Nov. 2005.
(1660) Conax Buffalo Technologies, company website, http://www.conaxbuffalo.com, viewed Nov. 2005.
(1661) Tomsia, A. P., J. A. Pask, and R. E. Loehman, "Glass/Metal and Glass-Ceramic/MetalSeals," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, pp. 493-501, 1991.
(1662) Kingery, W. D., H. K. Bowen, and D. R. Uhlmann, Introduction to Ceramics, Second Ed., John Wiley & Sons, pp. 91-124, 1976.
(1663) Watkins, R. D., "Types of Ceramic Joining and Their Uses," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, pp. 478-481, 1991.
(1664) Loehman, R. E., and A. P. Tomsia, "Joining of Ceramics," Ceramic Bulletin, vol. 67 [2], 1988. pp. 375- 380.
(1665) Borom, M. P., A. M. Turkalo, and R. H. Doremus, "Strength and Microstructure in Lithium Disilicate Glass-Ceramics," Journal of the American Ceramic Society, vol. 58 [9-10], 1975. pp. 85-391, 1975.
(1666) Everhart, J. L., Engineering Properties of Nickel and Nickel Alloys, Plenum Press, pp. 8-31 and 58-81, 1971.
(1667) "Metallurgy, Processing, and Properties of Superalloys," ASM Specialty Handbook: Heat- Resistant Materials, ed. J. R. Davis, ASM International, 1997 .pp. 221-254,
(1668) Mankins, W. L., and S. Lamb, "Nickel and Nickel Alloys," ASM Handbook Volume 2: Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, ASM International, 1990. pp. 428-445.
(1669) Robie, R. R., B. S. Hemingway, and J. R. Fisher, Thermodynamic Properties of Minerals and Related Substances at 298.15 K and 1 Bar (105 Pascals) Pressure and at Higher Temperatures, U.S. Geological Survey Bulletin 1452, 1979.
(1670) Callister, W. D. Jr., Materials Science and Engineering: An Introduction, Sixth Edition, John Wiley & Sons, Inc., 2003. pp. 737-770.
(1671) Kirk-Othmer Encyclopedia of Chemical Technology, Third Edition, Volumes 5 and 11, John Wiley & Sons, 1979.
(1672) "Properties of Pure Metals," ASM Handbook Volume 2 – Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, ASM International, 1990. pp. 1099-1198,
(1673) Schott Technical Glasses Physical and Technical Properties, corporate literature, Dec. 2002.
(1674) High Temp Metals, Kovar® product technical data sheet, www.hightempmetals.com/
(1675) techdata/hitempKovardata.php, viewed Nov. 2005.
(1676) Morey, G. W., The Properties of Glass, Second Edition, Reinhold Publishing Corporation, p. 276, 1954.
(1677) Insaco, Inc., Zirconia PSZ product data sheet, www.insaco.com/MatPages/zirconiapsz.asp, viewed Nov. 2005.
(1678) H. Cross Company, Alloy 42 and 52 product data sheet, www.hcrosscompany.com/metals/alloy4252.htm, viewed Dec. 2005.
(1679) Special Metals, NILO® and NILOMAG® Nickel-Iron Alloys product data sheet, www.specialmetals.com, viewed Dec. 2005.
(1680) Goodfellow Alumel®, Chromel®, and Constantan® product data sheets, www.goodfellow.com, viewed Dec. 2005.
(1681) CRC Handbook of Chemistry and Physics, 83rd Edition, ed. D. R. Lide, CRC Press, 2002.
(1682) Cengel, Y., and R. Turner, Fundamentals of Thermal-Fluid Sciences, McGraw-Hill, 2001.

(1683) Shelby, J. E., W. C. Lacourse, and A. G. Clare, "Engineering Properties of Oxide Glasses and Other Inorganic Glasses," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991. pp. 845-857.
(1684) Partridge, J. H., Glass-to-Metal Seals, p. 216, The Society of Glass Technology, 1949.
(1685) Jost, W., Diffusion in Solids, Liquids, and Gases, Academic Press Inc., Publishers, 1952. pp. 287-288, 300,
(1686) Shelby, J. E., "Effect of Phase Separation on Helium Migration in Sodium Silicate Glasses," Journal of the American Ceramic Society, Vol. 56 [5], 1973.263-266.
(1687) Zinkle, Steve, "Overview of Radiation Effects in Polymers, Semiconductors, and Ceramic Insulators," Oak Ridge National Laboratory, presented Nov. 10, 2004.
(1688) Beauchamp, E. K., and S. N. Burchett, "Techniques of Seal Design," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991. pp. 532-541.
(1689) Mizuhara, H., and T. Oyama, "Ceramic/Metal Seals," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991 pp. 502-510.
(1690) M. M. Schwartz, Ceramic Joining, ASM International, 1991.
(1691) Moorhead, A. J., and H.-E. Kim, "Joining Oxide Ceramics," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991. pp. 511-522.
(1692) Suganuma, K., "Joining Non-Oxide Ceramics," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991 pp. 523-531.
(1693) Murari, A., and A. Barzon, "Ultra High Vacuum Properties of Some Engineering Polymers," IEEE Transactions on Dielectrics and Electrical Insulation, vol. 11 [4], 613-619, Aug. 2004.
(1694) ASTM Designation E 479-91, "Standard Guide for Preparation of a Leak Testing Specification," ASTM International, 2000.
(1695) ASTM Designation E 493-97, "Standard Test Methods for Leaks Using the Mass Spectrometer Leak Detector in the Inside-Out Testing Mode," ASTM International, 1997.
(1696) ASTM Designation E 1603-99, "Standard Test Methods for Leakage Measurement Using the Mass Spectrometer Leak Detector or Residual Gas Analyzer in the Hood Mode," ASTM International, 1999.
(1697) Nondestructive Testing Handbook: Leak Testing, 3rd Edition, ed. Patrick Moore, American Society for Nondestructive Testing Inc., 1998.
(1698) Henkel, D., and A. Pense, Structure and Properties of Engineering Materials, Fifth Edition, McGraw-Hill, 2002. pp. 416-423.
(1699) Morgan Advanced Ceramics, company website, http://www.alberox.com, viewed Sept. 2005.
(1700) Crawford, Bob, Conax Buffalo Technologies, phone conversation, Aug. 2005.
(1701) Schott North America, company website, http://www.us.schott.com, viewed Dec. 2005.
(1702) Tekna Seal, LLC, company website, http://www.teknaseal.com, viewed Dec. 2005.
(1703) Comus International, company website, http://www.assemtech.co.uk/glass.asp, viewed Dec. 2005.
(1704) IJ Research, Inc., company website, http://www.ijresearch.com/index.php, viewed Dec. 2005.
(1705) Accratronics Seals Corp., company website, http://www.accratronics.com, viewed Dec. 2005.
(1706) Hermetic Seal Technology, Inc., company website, http://www.glass-to-metal.com, viewed Dec. 2005.
(1707) Dash Connector Technology, company website, http://www.dashconnector.com/maininfo.htm, viewed Dec. 2005
(1708) Green, D. J., An Introduction to the Mechanical Properties of Ceramics, Cambridge University Press, 1988. pp. 210-231.

(1709) Chambers, R. S., F. P. Gerstle, and S. L. Monroe, "Viscoelastic Effects in a Phosphate Glass-Metal Seal," Journal of the American Ceramic Society, vol. 72 [6], 929-932, 1989.
(1710) Pask, J. A., and A. P. Tomsia, "Wetting, Surface Energies, Adhesion, and Interface Reaction Thermodynamics," in Engineered Materials Handbook Volume 4: Ceramics and Glasses," ASM International, 1991. pp. 482-492.
(1711) Hokanson, H. A., S. L. Rogers, and W. I. Kern, "Electron Beam Welding of Alumina," Ceramic Industry, vol. 81 [2], 1963. pp. 44-47.

## 11.61 Radiation Hardness of Optical Fibers for Application to Space Reactor Pyrometry

(1712) Brichard, et al., "Radiation-Hardening Techniques of Dedicated Optical Fibres Used in Plasma Diagnostic Systems in ITER", Journal of Nuclear Materials 329-333 (2004), 1456-1460.
(1713) Brichard, et al., "Dependence of the POR and NBOHC Defects as Function of the Dose in Hydrogen-Treated and Untreated KU1 Glass Fibers", IEEE Transactions on Nuclear Science 50 (2003), 2024-2029.
(1714) Brichard, et al., "Origin of the Radiation-Induced OH Vibration Band in Polymer-Coated Optical Fibers Irradiated in a Nuclear Fission Reactor", IEEE Transactions on Nuclear Science 49 (2002), 2852-2856.
(1715) Brichard, Borgermans, Fernandez Fernandez, Lammens, and Decreton, "Radiation Effect in Silica Optical Fiber Exposed to Intense Mixed Neutron- Gamma Radiation Field", IEEE Trans Nuc Sci, vol 48, No 6, Dec 2001, pp. 2069-2073.
(1716) Conley, Lenahan, Wallace, and Cole, "Quantitative Model of Radiation Induced Charge Trapping in SiO2", IEEE Transactions on Nuclear Science 44 (1997), 1804-1809.
(1717) Deparis, Griscom, Megret, Decreton, and Blondel, "Influence of the cladding thickness on the evolution of the NBOHC band in optical fibers exposed to gamma radiation", J Non-Cryst Sol 216 (1997), 124-128.
(1718) Devine, "The Structure of SiO2, its Defects and Radiation Hardness", IEEE Trans Nuc Sci vol 41 no 3, June 1994, pp. 452-459.
(1719) Evans and Sigel, "Radiation Resistant Fiber Optic Materials and Waveguides", IEEE Trans Nuc Sci NS-22, no 6, Dec 1975, pp. 2462-2467.
(1720) Evans and Sigel, "Permanent and Transient Radiation Induced Losses in Optical Fibers", IEEE Trans Nuc Sci vol NS-21, Dec 1974, pp. 113-118.
(1721) Friebele, Sigel and Gingerich, "Radiation Response of Fiber Optic Waveguides in the 0.4 to 1.7u Region", IEEE Trans Nuc Sci NS-25, no 6, Dec 1978, pp. 1261-1266.
(1722) Griscom "□-ray-induced visible/infrared optical absorption bands in pure and F-doped silica-core fibers: are they due to self-trapped holes?" J Non- Cryst Sol 349 (2004), pp. 139-147.
(1723) Griscom and Mizuguchi, "Determination of the visible range optical absorption spectrum of peroxy radicals in gamma-irradiated fused silica", J Non-Cryst Sol 239 (1998), pp. 66-77.
(1724) Griscom, "$\gamma$ and fission-reactor radiation effects on the visible-range transparency of aluminum-jacketed, all-silica optical fibers", JApP vol 80, no 4, 15 Aug 1996, pp. 2142-2155.
(1725) Griscom, "Radiation hardening of pure-silica-core optical fibers by ultrahigh- dose $\gamma$−ray pre-irradiation", unknown IEEE publication, 1996. Updated version of JApP 77 1995.
(1726) Hayashi, et al., "Photobleaching Effects on Radiation-Induced Loss for Silica Glass Image Fiber", Influence of Radiation on Material Properties, 1986.

(1727) Henschel, Kohn, and Weinand, "Radiation Hardening of Pure Silica Optical Fibres by High Pressure Hydrogen Treatment", unknown IEEE publication, 2001.
(1728) Henschel, Kohn, Schmidt, Kirchhof, and Unger, "Radiation-induced loss of Rare Earth dope silica fibres", IEEE Trans Nuc Sci vol 45 no 3, June 1998, pp. 1552-1557.
(1729) Henschel, Kohn, Lennartz, Metzger, Schmidt, Rosenkrantz, Glessner, and Siebert, "Comparison between fast neutron and gamma irradiation of optical fibres", IEEE Trans Nuc Sci vol 45 no 3, June 1998, pp. 1543-1551.
(1730) Henschel, Kohn and Schmidt, "Radiation hardening of optical fibre links by photobleaching with light of shorter wavelength" (1996), 0-7803-3093-5/96; reference uncertain.
(1731) Holmes-Siedle and Adams, Handbook of Radiation Effects. Oxford University Press, New York, 2004. [Kakuta, 1999] Kakuta, Yamagishi, Itoh, Shikama, and Urakami, "Application of Optical
(1732) Fibers to Instrumentation System in Advanced Nuclear Power Reactors", 7th Intl. Conf. on Nuclear Engineering, Tokyo, Japan, April 19-23, 1999, ICONE-7128.
(1733) Kakuta, et al., "Behavior of Optical Fibers Under Heavy Irradiation", Fusion Engineering and Design 41 (1998), 201-205.
(1734) Lu, Schotz, Vydra, and Fabricant, "Optical Fiber for UV-IR Broadband Spectroscopy", SPIE Conference on Optical Astronomical Instrumentation, Kona, HI, March 1998, SPIE Vol. 3355, pp. 884-891.
(1735) Ott, "Radiation Effects on Commercially Available Optical Fiber: Database Summary", publication of Code 562, NASA GSFC.
(1736) Schulman and Compton, Color Centers in Solids. Pergamon Press, New York, 1962.
(1737) Tomashuk, et al., "$\gamma$-Radiation-Induced Absorption in Pure-Silica-CoreFibers in the Visible Spectral Region: the Effect of H2-Loading", unknown IEEE publication, 1998.
(1738) Tsai and Griscom, "Experimental Evidence for Excitonic Mechanism of Defect Generation in High-Purity Silica", Phys Rev Lett vol 67, no 18, 28 Oct 1991, pp. 2517-2520.
(1739) van Lint and Holmes-Siedle, "Radiation effects in electronics", in R.A. Meyers (ed.), Encyclopedia of physical science and technology, 3rd ed. Academic Press, New York.
(1740) Weeks and Lell, "Relation Between E' Centers and Hydroxyl Bonds in Silica", JApP vol 35, no. 6, June 1964, pp. 1932-1938.
(1741) Zabezhailov, et al., "The Role of Fluorine-Doped Cladding in Radiation- Induced Absorbtion of Silica Optical Fibers", IEEE Transactions on Nuclear Science 49 (2002) 1410-1413.

## 11.62 Hot Leg Piping

(1742) A.T. Chapman, J.K. Cochran, T.R. Ford, S.D. Furlong, and D.L. McElroy, "Reduction of High Temperature Thermal Conductivity of Thin-Wall Ceramic Spheres," Insulation Materials: Testing and Applications, 2, ASTM STP 1116, 1991. pp. 464-475.

## 11.63 Carbon-Carbon Composites

(1743) Policelli, F.J., and Vicario, A. "Space and Missile Systems." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 911-914.
(1744) Kim, R.Y. "Fatigue Strength." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 436-444.
(1745) Katscher, W. and Moormann, R. "Graphite Corrosion Under Severe HTR Accident Conditions." IAEA Specialists' Meeting on Graphite Component Structural Design, Tokai-mura, Japan. JAERI-M 86-192, September 1986, pp. 182 – 188.

(1746) Norton, B. "Cost Drivers in Design and Manufacture of Composite Structures." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 419-427.

(1747) Hart-Smith, L.J. "Joints." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 479-495.

(1748) Kennel, E.B., and Deutchman, A.H., "Joining Carbon Composite Fins to Metal Heat Pipes Using Ion Beam Techniques," Proceedings of Intersociety Energy Conversion Engineering Conference, Vol. 2 (1992) pp. 323-328.

(1749) Pierre, G.R. "Basic Thermodynamic Considerations Related to Oxidation Protection of Structural Carbon/Carbon Composites." Research into Structural Carbons, Ohio State University, Columbus, OH, pp. 409 – 438.

(1750) Barrett, M. J. "Performance Expectations of Closed-Brayton-Cycle Heat Exchangers in 100-kWe Nuclear Space Power Systems." NASA Glenn Research Center. NASA/TM – 2003- 21259, 2003.

(1751) Barrett, M.J. and Johnson, P.K. "Carbon-Carbon Recuperators in Closed-Brayton-Cycle Space Power Systems," 2nd International Energy Conversion Engineering Conference. AIAA 2004 – 5652, August 2004.

(1752) Stevenson, R.D. and Vrable D.L. "Development of an Intermediate Temperature Carbon-Carbon Heat Exchanger for Aircraft Applications." 44th International SAMPE Symposium. V44, 1999. pp. 1888-1897.

(1753) Kearns, K.M., Anderson, D.P., and Watts, R. "Brazing of Carbon-Carbon for an Aircraft Heat Exchanger." 44th International SAMPE Symposium. V44, 1999. pp. 1898 – 1908.

(1754) Policelli, F.J., and Vicario, A. "Space and Missile Systems." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 911-914.

(1755) Norton, B. "Cost Drivers in Design and Manufacture of Composite Structures." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 419-427.

(1756) Kennel, E.B., and Deutchman, A.H., "Joining Carbon Composite Fins to Metal Heat Pipes Using Ion Beam Techniques," Proceedings of Intersociety Energy Conversion Engineering Conference, Vol. 2 (1992) pp. 323-328.

(1757) Materials Resources International. "WideGap Joining: Powder Preform Technology." [Online] Available October 2001: http://www.materialsresources.com/widegap/widegap.htm.

(1758) Natesan, K., Purohit, A., and Tam, W. "Materials Behavior in HTGR Environments." Argonne National Laboratory. NUREG /CR-6824 ANL 02/37, February 2003.

(1759) Katscher, W. and Moormann, R. "Graphite Corrosion Under Severe HTR Accident Conditions." IAEA Specialists' Meeting on Graphite Component Structural Design, Tokai-mura, Japan. JAERI-M 86-192, September 1986, pp. 182 – 188.

(1760) Nomura, S., Imai, H, Fujii, K., Shindo, M. "Relation Between Gasification Rates and Gas Desorption Behavior with Metallic Impurities of Carbon and Graphite Materials for the HTTR." JAERI. Tokai-mura, Japan, pp. 159 – 168.

(1761) Pierre, G.R. "Basic Thermodynamic Considerations Related to Oxidation Protection of Structural Carbon/Carbon Composites." Research into Structural Carbons, Ohio State University, Columbus, OH, pp. 409 – 438.

(1762) Perez, F.J. and Ghoniem, N.M., "Chemical Compatibility of SiC Composite Structures with Fusion Reactor Helium Coolant at High Temperatures." Fusion Engineering and Design. V22, 1993. pp. 415 – 426.

(1763) Jacobson, N.S., Fox, D.S., and Opila, E.J., "High Temperature Oxidation of Ceramic Matrix Composites." Pure & Applied Chemistry. V70, 1998. pp. 493 – 500.

(1764) Savage, G. Carbon-Carbon Composites, 1st ed. Chapman & Hall, 1993.

(1765) Diefendorf, R. "Carbon/Graphite Fibers." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 49-53.
(1766) Diefendorf, R. "Continuous Carbon Fiber Reinforced Carbon Matrix Composites." In Composites, Engineered Materials Handbook Vol. 1, edited by Dostal, C.A. et al (1987). ASM International: Metals Park, OH. pp. 816-818.

## 11.64 Waste and Shipping

(1767) Atomic Energy Act of 1954, as amended
(1768) 40 CFR 261 "Identification and Listing of Hazardous Waste."
(1769) USDOT RAMREG-001-98, "Radioactive Material Regulations Review."
(1770) DOE Order 435.1 "Radioactive Waste Management."
(1771) DOE/ID-10381, Revision 21, January 2005, "Idaho National Engineering and Environmental Laboratory Waste Acceptance Criteria."

**CONCURRENCE/DESIGN CHECK FORM FOR DOCUMENT NO.** SPP-67110-0008 / B-SE-0155 **Date:** April 24, 2006

**DOCUMENT TITLE:** Prometheus Project Reactor Module Final Report, for Naval Reactors Information

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