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Summary of NR Program Prometheus Efforts

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Abstract. The Naval Reactors Program led work on the development of a reactor plant system for the Prometheus space reactor program. The work centered on a 200 kWe electric reactor plant with a 15-20 year mission applicable to nuclear electric propulsion (NEP). After a review of all reactor and energy conversion alternatives, a direct gas Brayton reactor plant was selected for further development. The work performed subsequent to this selection included preliminary nuclear reactor and reactor plant design, development of instrumentation and control techniques, modeling reactor plant operational features, development and testing of core and plant material options, and development of an overall project plan. Prior to restructuring of the program, substantial progress had been made on defining reference plant operating conditions, defining reactor mechanical, thermal and nuclear performance, understanding the capabilities and uncertainties provided by material alternatives, and planning non-nuclear and nuclear system testing. The mission requirements for the envisioned NEP missions cannot be accommodated with existing reactor technologies. Therefore concurrent design, development and testing would be needed to deliver a functional reactor system. Fuel and material performance beyond the current state of the art is needed. There is very little national infrastructure available for fast reactor nuclear testing and associated materials development and testing. Surface mission requirements may be different enough to warrant different reactor design approaches and development of a generic multi-purpose reactor requires substantial sacrifice in performance capability for each mission.

Keywords: Space Reactors, Nuclear Electric Propulsion, Direct Gas Brayton, Prometheus, Naval Reactors Program

INTRODUCTION

This paper summarizes the work completed and future work recommendations of the Naval Reactors Prime Contractor Team (NRPCT) as part of the NASA Prometheus space reactor development project. The majority of the work undertaken was focused on a reactor system suitable to a deep space nuclear electric propulsion (NEP) system, with the Jupiter Icy Moons Orbiter (JIMO) as the first mission.

The NRPCT was made up of engineers and scientists from the Naval Reactors (NR) Program prime contractors: Knolls Atomic Power Laboratory (KAPL), Bettis Atomic Power Laboratory (Bettis), and Bechtel Plant Machinery Inc (BPMI). The NRPCT was initiated in March 2004 with a request from the Secretary of Energy to have the Naval Reactors program team with NASA to design a civilian space reactor for a specific class of missions. A memorandum of understanding between NASA and NR was established which defined the division of responsibilities for the JIMO/Prometheus project. The NRPCT was responsible for all aspect of the reactor plant development and integration, while NASA and its contractors were responsible for the remainder of the spacecraft, the launch vehicle, and the ground systems. The Jet Propulsion Laboratory (JPL) was assigned lead responsibility, with support from the Glenn Research Center (GRC), Marshall Space Flight Center (MSFC), and other NASA Centers. In November 2004, Northrop Grumman Space Technologies (NGST) and its industry partners won the contract to develop the JIMO spacecraft. The NRPCT placed contracts with other national laboratories, including Los Alamos, Oak Ridge, Sandia, Idaho National Laboratory, and Pacific Northwest National Laboratories for reactor plant design, reactor safety assessments, testing, and materials support. This partnership between NRPCT and NASA and DOE facilities was a key element of the development program. Based on reprioritization of missions and funding within NASA, Naval Reactors and NASA discontinued this collaboration in September 2005. An orderly closeout of NRPCT work proceeded through the remainder of 2005.

Three major reports documenting the NRPCT efforts were developed. The first report, a concept feasibility evaluation, was issued in October 2004 (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. This report 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 better developed technologies. From this evaluation, fuel and structural materials, as well as integrated system design and testing emerged as the biggest challenges. The JIMO power and calendar life requirements, pushed the reactor system operating parameters well beyond any existing reactor system or established technology basis, particularly in the area of required fuel and structural material performance.

The second report was the concept selection report (Reference 2). This report was issued within the NR program in April 2005, and issued to NASA in August 2005. This report documented a lengthy and formal process used by the NR program to evaluate the candidate integrated reactor plant concepts and select a leading approach for development. Based on the feasibility assessment results five concepts were selected for evaluation:

1) Gas reactor – Brayton energy conversion system
2) Heat pipe cooled reactor – Brayton energy conversion system
3) Liquid lithium cooled reactor – Brayton energy conversion system
4) Liquid lithium cooled reactor – Thermoelectric (TE) energy conversion system
5) Lower temperature liquid sodium reactor – Stirling energy conversion system

Based on the overall system features, including capability, reliability, deliverability, cost and safety, the direct gas reactor concept was recommended to NR headquarters for approval. NR headquarters approved the recommendation. The gas reactor system appears capable of fulfilling the mission requirements for the envisioned NEP missions, simplifies engineering development testing, and offers the fewest hurdles to development.

The final report (Reference 3) and its references summarize the work completed between the concept selection recommendation and the program closeout to better define the features and developmental issues of the direct gas Brayton system. These reports include reactor physics, thermal and mechanical design evaluations, reactor core and plant arrangements, integrated system performance estimates, evaluation of material options, instrumentation and control development, and reactor plant operating scenarios. While the majority of evaluations concerned the JIMO/NEP requirements, some evaluation was done regarding extensibility of the concept to surface missions as required in the Prometheus Project Level 1 requirements. Table 1 shows the impact of the NASA Level 2 requirements on the reactor system requirements. This final report also summarizes the plans established for developing, testing and deploying the gas Brayton system. This paper summarizes that closeout report.

The three major reports, as well as key reference reports, will be provided to NASA and DOE and stored in the Office of Scientific and Technical Information (OSTI) Science Research Connection database, administered by the Oak Ridge National Laboratory.

DESCRIPTION OF REFERENCE REACTOR PLANT

The reference gas Brayton system is depicted in Figure 1. Table 2 shows the reference reactor plant parameter list. Table 3 shows a list of envisioned materials and the concerns for each of the major components in the system. The gas-Brayton concept uses a single gas reactor attached to the forward end of the spacecraft. An inert gas (a helium and xenon mixture was assumed for initial work) is used to cool the core and transport energy around a shadow shield to the Brayton energy conversion system. The core consists of cylindrical uranium containing fuel elements arranged within the core structure. Gas flows either directly over the fuel pins 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 2. The elements consist of a fuel filler, a gas gap to accommodate differential swelling, a fission gas plenum to reduce fission gas pressure, a cladding liner to improve material compatibility, and the cladding which prevents fission gas escape. Several refractory metal alloys as well as silicon carbide were considered for the cladding. Several core configurations were also investigated. Most configurations utilized a core block which holds the fuel elements in place (Figure 3). An open lattice of fuel elements has also been investigated. The annular block arrangement allowed for simplified heat transfer and greater flexibility in block material selection but required an increased core size to minimize pressure drop. One option for this block arrangement is to use a cermet fuel system, with a refractory metal matrix and small fuel particles. This approach improves heat transfer, eliminates gas plena and leads to a smaller core and shield. However, cermet fuel fabricability and performance are not well characterized at a mass-competitive fuel loading density . Table 4 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. The table shows how the core designs change with changes in cladding material, core arrangement, coolant pressure, and required reactor power to meet the nuclear and mechanical design requirements.

The reactor vessel surrounds the core and a combination of fixed and movable reflectors surround the vessel. The movable reflector is segmented and used to maintain reactivity at the desired operating temperature over life. Instrumentation to monitor power and temperature is used to determine when to move reflector segments during reactor startup and to compensate for uranium burn-up during operation. The reactor uses at least one safety shutdown rod. Safety rods are only used during transport and launch and would be withdrawn from the core prior to initial criticality.

The average gas core exit temperature is limited to 1150 K (1610 F) in order to allow use of more conventional materials for the plant and energy conversion system and to reduce pressure loading on the fuel element cladding. 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 used by the on-board ion propulsion system, on-board computers, science and 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 unit or units 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 range of component mass estimates and overall heat balances are shown in Figure 4 and 5 respectively. Reactor and shield masses range from 3,000-5,000 kgs for a 1 MWt system. The total SNPP mass is between 7,500 and 11,000 kgs, depending on the reactor type and plant configuration.

Key Technical Findings

Reactor Design

Hundreds of parametric studies were performed to understand the range of reactor design options as a function of materials, core arrangements, and operating conditions. No reference reactor core concept was chosen prior to project closeout. A range of reactor design options was considered, including both unmoderated and moderated reactors. A fast (unmoderated) reactor spectrum offers a lower overall mass than a moderated spectrum in the range of JIMO power and lifetime requirements. The fast reactor will require several hundred kilograms of highly enriched uranium. The primary reactor physics challenge is ensuring adequate reactivity level for normal operating conditions while maintaining the reactor sub-critical during assembly, transport or launch accidents. To ensure shutdown during a water flooding, sand immersion, or compaction casualty, a combination of spectral shift poisons and in-core safety shutdown rods are envisioned.

The reactor physics uncertainties with this compact, reflector-controlled fast reactor necessitate a reactor physics qualification approach, including fundamental integral cross section testing, representative material and geometry critical testing, cold and hot critical physics experiments, and detailed physics testing on a ground prototype.

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 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 a re-entry accident. Considerably more modeling and testing would be needed to fully understand the potential of criticality and energy release on impact. Depending on the impact predictions, additional safety devices may be needed.

Reactor thermal hydraulic performance is challenging due to the low operating pressure (2-3 MPa), the properties of the helium-xenon gas mixture and the need to maintain a low reactor core pressure drop to maximize system performance. The maximum fuel element surface temperature is approximately 1300 K during normal full power operation. Sensitivity studies were performed for the range of fuel element sizes, flow configurations, and reactor materials to balance heat transfer and pressure drop through the core.

No final decision was made regarding segmented axially translating reflector sections (“sliders”) versus rotating drums for reactivity control. Control drums offer simpler control mechanism design, greater certainty in measuring control device position, and a more rugged mechanical design, but have reduced reactivity control worth relative to sliders. Sliders offer a lower mass system and a more linear differential control worth. Most of the reference reactor nuclear and mechanical design work assumed sliders.

Reactor Materials

The NRPCT developed preliminary estimates for both uranium nitride and uranium oxide fuel performance in order to judge their acceptability for NEP missions, including the expected rate of fuel swelling, the fission gas release rate, and the chemical compatibility with the fuel liner and cladding. Both uranium nitride and uranium oxide fuels were judged as feasible (in concept) to meet NEP mission requirements. Uranium nitride will lead to a lower mass reactor due to its higher fuel density and higher thermal conductivity. As part of the program, Los Alamos National Lab reproduced the uranium nitride processing approach used during the SP-100 program (Figure 6). However, demonstrating long term compatibility with the fuel cladding systems would be more challenging for the nitride system than for the more fully tested oxide fuel system. NRPCT planned to pursue the uranium oxide fuel system for the JIMO mission in order to minimize the development challenge.

All of the fuel element cladding options considered require substantial development. Nb-1Zr, the reference alloy used in past space reactor concept development (SP-100 and SNAP-50), has poor creep capability. Using Nb-1Zr cladding will reduce the allowable reactor temperature, increase the gas plenum size, and increase reactor assembly mass relative to stronger refractory alloys. Stronger refractory metal alloys such as FS-85, Ta-10W, and ASTAR-811C offer potentially better creep strength. However, niobium and tantalum alloys likely require protection, via liners and coatings, from fission product corrosion and from the low oxygen and carbon partial pressures in the gas coolant. Mo-Re alloys offer the potential of improved fuel and coolant compatibility, but suffer from phase instability and irradiation embrittlement. A silicon carbide composite clad fuel system requires more work to evaluate cladding hermeticity, toughness, and irradiation performance. Based on the developmental challenges with each system, a number of candidate materials were being carried through initial testing, including irradiation testing. Preliminary materials tests were underway at the Oak Ridge National Laboratory HFIR reactor and planned for the Japanese JOYO fast reactor.

NRPCT had an objective to use nickel superalloy pressure boundary on all parts of the reactor plant to simplify manufacturing, eliminate dissimilar material bonds at pressure boundaries, and allow for extensibility to lunar and Martian surface missions. Superalloy pressure boundaries are limited in temperature by creep capability. The reactor vessel and loop piping concept developed concentrates on maintaining the pressure boundaries at or below ~900 K (1160 F). Properties for Alloy 617 were used in reactor vessel and loop piping analysis due to the greater data pedigree compared to other wrought nickel superalloys being considered. The reactor vessel is cooled using a combination of the cooler reactor inlet gas and radiative cooling. The hot leg piping temperature is maintained below 900 K through the use of internal insulation in the piping. Compatibility between the nickel superalloy structure and core materials remained an open issue. Nickel superalloys were shown to be more compatible with silicon carbide, graphite, and molybdenum alloy core structures than with the niobium or tantalum alloys based on thermochemical analysis and review of limited test data. One potential issue that may have precluded the use of nickel superalloys for the entire pressure boundary is the higher temperature and fluence on the safety rod enclosure (thimble). Approaches for effectively cooling the thimble, or for internal safety rods were under consideration. Refractory metals were considered as a back-up approach for the hot pressure boundaries. However, refractory metals at operating temperature are intolerant of oxygen and carbon containing impurities. The use of refractory metals in the pressure boundary would require development of dissimilar material joining, increase the difficulty of earth-based testing, limit lunar and Mars surface extensibility. Potential methods to mitigate these incompatibilities using coatings were being evaluated.

Reactor Shielding

Shielding the remainder of the spacecraft from reactor radiation is a trade-off between mass, materials development, and the allowable dose to mission electronics. Neutron shielding material options evaluated included lithium hydride, water, beryllium, and boron carbide. Tungsten is considered to be the best gamma shield material based on its high density and moderate cost. Lithium hydride-based neutron shielding, which require Be/B4C for the high flux portion of the shield, provide a slightly lower overall mass than concepts without lithium hydride. However, a full Be/B4C shield is considered to be a lower cost, lower risk option. Water shielding is also mass competitive with lithium hydride. Reliability of LiH or water shielding systems would require further evaluation.

Shielding model results showed that the control drive mechanism penetrations can be made without substantial impact on the shield effectiveness. Gas pipe radiation streaming can be controlled by spiraling the piping around within the outer surface of the shield. The overall shield thickness must increase to retain the same shielding effectiveness as a reactor without large gas piping. Additional pipe shielding was not required. The effectiveness of the shielding remains adequate even for relatively high rates of fission product release (up to

10^{-3} release to birth ratio of volatile fission products) from the fuel into the gas coolant. However, the effect on the alternator would require further study.

Reactor Plant Design

As with the reactor design, parametric studies were completed to understand the range of design option available for integrating the reactor and the energy conversion system within the gas loop. Preliminary design and analysis indicates that the overall gas reactor plant performance is sensitive to the detailed design and system layout. Parasitic heat loss and pressure drop, as well as minor inefficiencies in energy conversion components can lead to large impacts on overall system performance. Similarly, the system performance is very sensitive to the operating conditions such as temperature and the uncertainty in those operating conditions. This leads to a need for careful design and developmental testing, and a strong development program to reduce performance, operational and measurement uncertainties.

Reactor plant transient analysis, performed on a range of concepts with three separate modeling tools, has not revealed any inherent instabilities in the plant response to planned and unplanned transients, either with a single Brayton or multiple units, assuming the overall reactor temperature coefficient is negative. A negative temperature coefficient is predicted for the core designs considered. The sufficiently high thermal mass of the reactor core and recuperator limit the rate of temperature change during the analyzed transients and minimize over-temperature concerns. No rapid reactor control system response is required to maintain the reactor within the desired temperature range during these transients. The most severe operational transient appears to be a loss of Brayton load and speed control, which leads to an increase in reactor power and flow and the potential for overloading the heat rejection system. The likelihood of such a Brayton control system failure must be minimized through reliable electrical system design. Figure 7 shows the plant response, assuming no control system function, to a loss of load in a two unit Brayton plant. Note that while the reactor temperature increase is relatively mild, the heat rejection temperature increases considerably, creating concern for heat rejection system materials and operability. Responding to this loss of load casualty with an increased heat rejection system flow and/or closing of control valves in the affected Brayton loop would be required to protect against overpressurizing the gas cooler in this loop.

Evaluation of Brayton system component performance, in cooperation with NASA GRC and turbomachinery vendors, has confirmed the conceptual feasibility of developing a long lived 100-200 kWe Brayton system. Similar open loop, combustion heated Brayton engine system units have operated at similar power levels in terrestrial and aerospace applications. Rotordynamic evaluations, bearing and alternator cooling studies, and recuperator and gas cooler design and performance studies all indicate such a system could be successfully developed. Known open issues include integrated system response, materials compatibility within the system, and gas leakage through welds, electrical feed-throughs, and heat exchangers. Separate effects testing and integrated system testing, as envisioned for the next steps in this program, would be necessary to better judge Brayton system performance and reliability.

Early in the design development, NRPCT investigated both direct and indirect gas cycles. The indirect gas cycle allowed for separation of the reactor coolant gas from the Brayton working fluid. This allowed for isolation of the Brayton loops and optimization of the gas composition in each loop. However, the indirect system required powerful gas circulators for the primary coolant and massive gas to gas hot heat exchangers. The direct gas system is considered to be the only practical gas reactor approach.

A decision concerning the energy conversion system redundancy was not made. A range of concepts, from a single Brayton unit to concepts with multiple operating units and spares have been considered. Figure 8 depicts the potential arrangement of one, two, three or four Brayton units. For the NEP missions, use of a single Brayton converter offers the simplest and most efficient system, eliminating piping manifolds and valves (and associated pressure drop), minimizing arrangement complexity and mass, and simplifying operation. Based on NASA-developed turbomachinery models and pressure drops from the investigated loop arrangements, the efficiency of a single loop system was expected to be 10-20% higher than that of a four loop system. This could allow for either reduction in the required reactor power and radiator area, or a reduction in maximum gas temperature. However, failure of this single Brayton would be a mission ending failure. The next simplest system that would require two operational units (each producing half power) with a single spare. If spare units are carried, using an on/off control approach for the spare unit best preserves system performance compared to operating all units at a fraction of full capacity.

The capability and performance of the heat rejection system is intimately linked to that of the reactor system. This is particularly true of a water heat rejection loop since the pressure of the water loop will be very sensitive to the operating temperature and the uncertainty in operating temperature. Analysis of the failure modes in the

Brayton system and the heat rejection system indicated the need to design the water heat rejection loops to handle operating pressures as high as 10 MPa. Lifetime degradation of radiator performance, due to changes in effective emissivity and isolated heat pipe failures will lead to a slow increase in radiator temperature. This will require a comparable increase in reactor exit temperature to compensate. NRPCT analysis confirmed NGST estimates of water heat pipe performance capability, but water heat pipe performance at high temperature and high heat flux requires continued development and demonstration.

The performance of the gas cooler, which is the interface between the reactor system and the heat rejection system, is key to the reliability of the overall plant due to its high surface area and the high heat rejection system pressure. Preliminary vendor design concept development, sponsored by the NRPCT, studied plate-fin, shell and tube, printed circuit board, and hybrid heat exchanger configurations. Mass estimates from all of these vendors are considerably higher than the initial NASA and contractor estimates.

Instrumentation and Control

A robust evaluation of candidate sensors has been completed. Technologies appear to be available for all key plant parameters including temperature, neutron flux, pressure, mechanism position, and coolant flow. Significant development remains to deliver and qualify both sensors and interface electronics for long duration space missions. Major technical challenges are lifetime, operating temperature, accumulated radiation dose, and the need to operate without recalibration. The most challenging sensor application is hot leg temperature due to the lifetime degradation of the sensors at temperature and the need to integrate the sensor with internally insulated loop piping. The most promising sensor technology for this application is judged to be sapphire fiber Bragg gratings, which have the potential for adequate stability over life.

A control system architecture has been developed, including recommended redundancy, interfaces with the spacecraft, number of sensors and board count. The control architecture provides for autonomous reactor operation and fault management with a simple system. Electronics for the control system that can 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 are a significant technical challenge and would require development of custom application-specific integrated circuits to achieve the level of radiation hardness and reliability needed, The JIMO mission schedule would have required immediate near term actions to develop these electronics.

A plan for control system software development for JIMO was completed, and selections 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.

Reactor and reactor plant startup is the most complex operation envisioned for the control system. Start-up requires tight control over the reactor through the rate of reactivity insertion, and simultaneous control of the Brayton speed through use of a start inverter and parasitic load regulator. A reference start-up procedure has been developed. This startup approach has not been fully modeled or refined.

PROGRAMMATIC CONCLUSIONS

Testing Needs

As mentioned above, the selection of a cladding materials for the reactor could not be finalized because of the uncertainties associated with all of the material options. Both in-core irradiation testing and out-of-pile creep and thermal stability tests of the candidate alloys are needed. Similarly, irradiation, creep and compatibility testing are needed for core and plant structural materials. This testing is expensive, time consuming and requires use of assets with capabilities no longer available in the United States. Specifically, it should be noted that a fuel system with sufficient capability to meet the long time (>10 years) at high temperature (up to 1800 K) is unprecedented and the consequences of widespread fission product release on radiation dose levels behind the shield, coolant chemistry, and corrosion necessitate this testing. The fast spectrum associated with the space reactor also dictates the use of a fast reactor for irradiation testing, both to mimic the prototypical conditions and to speed accumulation of fast neutron fluence in accelerated tests. Since the United States has no such facilities, the NRPCT was preparing to utilize the JOYO test reactor in Japan. Even for more modest space reactor missions, where use of more well established materials is possible, some irradiation testing will be needed. Such irradiation testing of potential reactor fuel systems is a critical and challenging long-term development need.

Fast reactor neutronic performance is very sensitive to both materials and geometry. Less conventional materials, such as tantalum, tungsten, molybdenum, and rhenium were considered. Some of these materials have never been used in meaningful quantities in a reactor before. Uncertainties with rhenium and other nuclear material cross-sections must be resolved to improve confidence in design predictions during the early design stages. Similarly, the reactor neutronic performance is sensitive to small changes in reactor geometry due to thermal expansion, fuel swelling, thermal creep and irradiation induced creep. Developing and demonstrating the ability to design for these effects dictates a need for cold and hot critical reactor physics performance testing. There are currently no operational critical facilities in the United States. Options for recreating these test facilities have been evaluated, but no conclusions on the optimal site for such tests were made.

Overall nuclear plant performance is predicted to be very sensitive to minor losses and inefficiencies in the system and the accuracy of the instrumentation. At this early stage, overall uncertainty in the system efficiency is as large as 50%. That is, the worst case estimates of temperature measurement accuracy, turbomachinery performance and loop flow resistance lead to a ~50% change in output compared to the most optimistic case. Therefore, system testing is needed to better understand and reduce these losses as well as to bring to light system engineering and control challenges and issues. Another reason to prioritize these tests early in the development process is to uncover any system chemical incompatibilities. While careful materials selection minimizes gross material incompatibilities, early testing often points out overlooked issues that can be corrected prior to larger scale testing. Early integrated system testing was planned using electrically heated reactor mockups with the thermal test module (TTM), followed by a more prototypical engineering model (EM) and a fully prototypical qualification model (QM). Figure 9 shows a representation of the system testing approach envisioned. Such integrated system testing of gas Brayton systems is a critical and challenging long term need.

Development of test hardware and test results is predicated on development of manufacturing methods for key components. NRPCT had performed initial manufacturing trials on candidate fuel and structural materials, primarily through subcontract. Small quantities of all refractory alloys under consideration were manufactured for materials testing and NRPCT was preparing to purchase production ingots of promising materials at project termination. Long term space reactor development will need to re-establish material supplies and manufacturing capabilities to ensure high quality and repeatable component performance.

A final conclusion on the need for a ground prototype reactor plant was not made during the course of the project. However, a ground prototype was a baseline NRPCT project assumption and planning was initiated in accordance with the requirements of the National Environmental Protection Act (NEPA). A prototype would likely be necessary as a means to demonstrate manufacturing and assembly, establish nuclear performance, confirm system dynamic operating characteristics, investigate lifetime effects, and uncover any additional system integration issues. All prospective sites considered in the initial prototype scoping study required considerable facility modifications.

Assembly, Test and Launch.

NRPCT developed a draft approach for prototype and flight unit construction, transport and assembly with the spacecraft. This approach, as outlined in Figure 10, minimizes the transport and safeguards requirements for the reactor cores by minimizing the number of facilities required to handle the small, highly enriched core. Considerable early planning for core fabrication, assembly, transport and testing prior to launch is needed.

Formal Engineering Design Basis

Development of a formal engineering design basis was initiated as an essential near term priority. Development of a consistent set of material properties, updated to include emerging data, was underway. The reactor and core structural design basis was being developed based on guidelines from the ASME high temperature code, foreign gas reactor structural design bases, and from past space reactor development. Reducing uncertainties while maintaining appropriate design conservatism requires careful review of data and analytical methodologies. Formal design bases were also being developed for reactor safety, reactor nuclear design, and fuel performance limits.

Formal and Timely Decision Making

Over the course of the 18 month program, the NRPCT evaluated all reasonable reactor and energy conversion technologies and selected a reference coolant and energy conversion approach that offered the best prospect of meeting NASA mission requirements within the schedule and cost constraints. The NRPCT also recommended the use of uranium oxide fuel. NRPCT was working to select reference core and plant structural materials, a reference plant arrangement and reference operating conditions in fiscal year 2006. Each of these major decisions was being made through detailed study, formal review and debate, and formal recommendation to Naval Reactors headquarters. The rapid pace of these early design decisions was needed to concentrate efforts and maintain the design and testing schedules. While some back-up materials would have been carried, in general NRPCT determined that the best prospects for successful delivery lay in devoting most resources to engineering the best system rather than maintaining multiple back-up design concepts.

Regulatory Review

For the Prometheus project, public information sessions were held by NASA and NR to initiate the process for a preliminary environmental impact statement (PEIS), as dictated by the National Environmental Protection Act (NEPA). Work was initiated to define the reactor plant options and ground based testing needs using a pre-decisional scoping approach. The schedule of follow-on design and testing activities would be guided by compliance with NEPA requirements.

Development Timeline and Cost

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 a ground based prototype, the development timeline established for the JIMO program was very aggressive. There was little or no time to recover if initial technology choices proved unworkable, and basic technology testing and validation would have to be completed in parallel with plant construction and operation, adding to program risk. To ameliorate this concern in future projects, the scope and timescale required for an engineering development, manufacturing and testing effort of this magnitude must be understood from the beginning, and appropriate funding and mission consistency must be sustained through the duration the program. Consideration of less aggressive mission requirements for initial missions could allow for initial applications with more readily available technologies, while longer term advanced technologies are pursued in parallel for follow-on missions.

The gas reactor concept was selected in part because it minimized the expected development time and cost. However, the space reactor planning estimate for the work, including a ground test reactor, was still substantial (~$3.5 billion). The development scope was driven by the functional requirements for JIMO, particularly the very long core calendar life. Development of long-lived nuclear reactors is 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 unwise and misleading.

Programmatic Leadership and Decision Making

The central programmatic challenge for this project was to develop a single integrated engineering culture from the many organizations, diverse technical cultures, and limited overlapping experience base, for a space reactor project possessing unprecedented requirements. Reactor development requires tight coordination and communication within and across many government agencies, laboratories, and industrial partners. The responsibility and authority of a central DOE agency, in this case the NR Program, to lead and manage all aspect of the space reactor power plant development and use is essential to a successful long term program. During the Prometheus effort, the NR Program managed all aspect of the nuclear work, including subcontracted development effort at other DOE Laboratories. Work performed by other National Laboratories and NASA Centers partners was essential to the progress made by the NRPCT. Table 5 summarizes the key roles played by these organizations. While the work did not proceed to completion, a centralized authority which develops the reactor design and coordinates the use of the limited national nuclear infrastructure remains the best approach for completion of new nuclear development tasks.

Extensibility to Other Missions

The NRPCT performed a limited amount of work to judge the extensibility of the gas reactor Brayton concept. The direct gas Brayton concept remains mass competitive with other reactor concepts over a range of powers between 25 kWe and 300 kWe. Figure 11 shows the approximate range of reactor coolant, fuel material and energy conversion technology applicability as a function of power and energy. At the low end of this power range, the reactor and its shield become the mass limiting items, allowing lower temperature liquid metal concepts to become more mass competitive. Linking a lower temperature liquid metal reactor with a Stirling energy conversion system maintains system efficiency at more moderate temperatures (~900 K) and allows for the use of more conventional stainless steel - uranium oxide fuel systems. However, trade studies performed on potential electric power distribution systems show a clear preference for Brayton cycle energy conversion when compared to thermoelectrics or Stirling. This advantage is due to the inherent ability of Brayton converters to produce high frequency AC power which is simpler to convert and distribute over long distances. Thermoelectric and Stirling power conditioning systems are feasible, but have substantial mass and reliability drawbacks when compared to Brayton systems at these power levels. Also, even low temperature liquid metal systems require special facilities for liquid metal processing, purification, and safe testing.

At the high end of the power range (above 300 kWe), the mass and hydraulic advantages of a high temperature liquid lithium reactor system over a gas reactor system become more pronounced. The mass advantage of a liquid lithium system would need to be balanced against its increased design and materials development, testing, and operational complexity.

One approach for a mission architecture that has been proposed is to deploy a space reactor with an easier low temperature liquid metal for a low power application and then “scale up” to a higher power capable system for follow-on missions. However, scaling up liquid metal systems, for example from NaK to lithium, from steel to refractory metals, and from TE to dynamic converters has a number of critical and varying engineering development challenges with regard to the higher power system that are not directly addressed with the low power application. In essence, such an approach would require working on two completely different systems.

Gas reactor Brayton systems also conceptually appear well suited for surface missions. The environmental conditions such as electrostatic dust and sink temperature variability bring a greater desire for conventional structural materials and non-freezing transport fluids. Surface missions may also bring with them more modest power and lifetime requirements than for JIMO or other NEP missions. Lower power levels and shorter lifetime demands reduce some of the development requirements for the gas reactor.

Notional low power and lifetime requirements for surface missions also open up the possibility of a mass competitive neutronically moderated gas reactor. Such a system could allow for conventional gas reactor fuel systems, require substantially less fuel, simplify reactor control, and reduce the need for extensive criticality testing. One approach is a water moderated gas reactor (Figure 12). In this concept, gas flows over fuel elements housed within pressure tubes within a vessel containing a mixture of light and heavy water. The water softens the neutron spectrum, enabling the use of a TRISO high temperature particle fuel system. This reactor has considerable mechanical and thermal design challenges and would require detailed concept development, based on firm mission requirements, to better judge feasibility. At the JIMO power and lifetime requirements, advanced TRISO particles, with higher than current state-of-the-art loading densities would be necessary to attain a mass-competitive design.

Future Technology Development

The scope and nature of the development challenges, together with the flexibility and extensibility of the gas reactor concept makes it well suited, in concept, for a number of space reactor missions. Therefore, any future concept development work should include the gas reactor Brayton system in its evaluation. Any continuing funding of space reactor work should include funding for non-nuclear testing of a ~100 kWe Brayton system. However, mission requirements may dictate solutions other than a gas reactor. A knowledgeable and experienced nuclear systems engineering organization should be intimately involved in early mission planning stages to provide a realistic understanding of what nuclear systems engineering and development are needed to accomplish mission objectives.

A key parameter in satisfying a large range of missions for space reactors is high temperature operation. Conventional materials, operating with reactor outlet temperatures of ~900 K, can provide only marginal capability to surface missions and NEP missions. Gas-Brayton systems extend power capability by raising coolant temperature to 1150 K or more while maintaining conventional high temperature materials for majority of the reactor plant. NRPCT concludes that the technologies for a gas-Brayton system are within reach for a first space reactor mission, either for surface power or NEP. Further improvements in power and energy density beyond the capabilities of a gas reactor system can only be achieved through widespread use of refractory metals in liquid metal systems, which will require greater development, particularly for materials compatibility and lifetime integrity. Given that there may be time prior to initiation of another large scale space reactor project, development of these key technologies and test hardware might create the potential for improved reactor power and energy density:

1) Irradiation testing of refractory metal clad uranium nitride fuel elements.
2) Thermal and irradiation creep testing, together with thermal and chemical stability testing of candidate refractory metals.

3) A high temperature (>1200 K) refractory metal liquid lithium flow loop to demonstrate electromagnetic pumping, gas separation, freeze/thaw behavior, and long term compatibility.
4) Reactor safety modeling and testing, to better understand impact dynamics and effect on reactor design.
5) Demonstration of water heat pipes for heat rejection systems with operation at greater than 550 K and greater than 10 W/cm^2 radial heat flux and development of heat pipe coolants for operation above 550 K.

CONCLUSIONS

A substantial amount of work was initiated by the NRPCT to select and develop a concept for a gas-Brayton space reactor for the Prometheus project. The three major technical reports – the feasibility study, the concept selection report, and the close-out report – have been formally issued and stored in the Office of Scientific and Technical Information (OSTI) Science Research Connection database. These reports can inform future developers of space reactors on the work completed during this phase of the Prometheus program.

REFERENCES

1) NRPCT Document "NR Program Assessment of the Design Space for the Prometheus 1 Project"
2) NRPCT Document "Space Nuclear Power Plant Concept Selection, For NR Approval"
3) NRPCT Document "Project Prometheus NRPCT Final Report"

Table 1. Design Requirements and Their Impact on Reactor Requirements

Key Level 2 Requirement	Impact on Reactor Module	Implementation
The Space Nuclear Reactor design shall utilize technologies that facilitate extensibility to surface operations.	Drives 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.	Plant Electrical Power = 200 kW Plant Thermal Power ~ 1 MW
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. 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 a 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), must demonstrate that alternatives to single point failure are not available and show sufficient robustness to mitigate risk of failure.	Where possible, 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 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 neutron flux ~10^5x and Gamma Flux 100x and must cover roughly a 12^o by 6^o 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 goal and selection criteria for design.	High temperature reactor is required to minimize overall mass

Table 2. Reference Reactor Plant Parameters

Parameter	Pre-Conceptual Base Case	Range/Options
Mission Duration	15 years	10-20 years
Core Energy Duration	15 Full Power Years	10-20 Full Power Years
Nominal Core Thermal Power	1000 kWt	0.5-1.5 MWt
Reactor Inlet Temperature	891 K (1144°F)	810-935 K (998-1220°F)
Reactor Outlet Temperature	1150 K (1610°F)	1050-1200 K (1430-1700°F)
Number of Reactor Inlet Nozzles	1	1-2
Number of Reactor Outlet Nozzles	1	1-2
Reactor Nozzle-to-Nozzle Pressure Drop (dP/P)	2.5%	2.0-4.0%
Gas Molecular Weight	31.5 g/mol (He mole fraction = 0.784)	20-45 g/mol (He mole fraction = 0.874 - 0.560)
Pipe Outer Diameter	16 cm (6.3 in)	TBD
Number of Shield Penetrations for Pipping	2	1-4
Number of Shield Penetrations for CDM	12	6-18
Payload Rx Neutron (DDD)	5E10 n/cm2	TBD
Rx Gamma (TID)	25 kRad Si Damage	TBD
Behind Shield Rx + Space Neutron (DDD)	TBD	TBD
Rx + Space Gamma (TID)	0.5 GRad Si Damage	TBD
Number of Braytons	4 (2 op at 100% capacity, 2 spare)	4 (2 op at 100% capacity, 2 spare) 3 (2 op at 100% capacity, 1 spare) 3 (3 op at 66% capacity) 2 (2 op at 50% capacity) 2 (1 op at 100% capacity, 1 spare) 1 (1 op at 100% capacity)
Brayton Shaft Speed	45000 rpm	30000-75000 rpm
Power delivered from alternator	200 kWe	195-220 kWe
Compressor Outlet Pressure	2000 kPa (290psi)	1380-4000 kPa (200-580psi)
Compressor Inlet Temperature	390 K (242°F)	350-450 K (170-350°F)
Compressor Pressure Ratio	2.0	1.8-2.2
Converter Loop Pipe Outer Diameter	10 cm (3.9 in)	10-16 cm (3.9-6.3 in)
Alternator Loop Pipe Outer Diameter	5 cm (2 in)	TBD
Isolation Valves	1 at Compressor Outlet	NONE 1 at Turbine Outlet 1 at Compressor Inlet 1 at Compressor Outlet 1 at Low Pressure Recuperator Outlet 1 Alternator Bleed Flow Outlet
Check Valves	1 at Compressor Outlet	NONE 1 at Compressor Inlet 1 at Compressor Outlet 1 at Alternator Bleed Flow Outlet
Number of Recuperators	4	1-4
High Pressure side of Recuperator dP/P	0.8%	0.5-1.5%
Low Pressure side of Recuperator dP/P	1.5%	1.0-3.0%
Recuperator effectiveness	0.92	0.86-0.94
Number of Gas Coolers	4	1-4
Gas Side of Gas Cooler dP/P	1.0%	0.5-2.0%
Pressure Drop in HRS Side of Gas Cooler	25 kPa (3.6 psi)	20-30 kPa (2.9 - 4.4psi)
Gas Cooler effectiveness	0.94	0.90-0.96
Heat Rejection System Fluid	Water	Water, NaK
Two-Sided Radiator Area with 14.5% Margin	450 m^2 (4840 ft^2)	400-650 m^2 (4300 - 7000ft^2)
Inlet Coolant Temperature	505K (449°F)	485-530 K (413-494°F)
Emmisivity	0.9	0.75-0.95
HRS Operating Pressure	7.8 MPa (1130 psi)	5-11 MPa (725 - 1595 psi)
Pressure Drop in Loop	365 kPa (53 psi)	100-400 kPa (15-58 psi)
Number of Loops	4	1-4
Number of Pumps per Loop	2	1-2

Table 3. List of Materials Envisioned for Use in the Prometheus Program

Component	Material Option	Operating Condition	Developmental Concern
Fuel	UO_2	900-1700 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
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
	B_4C		
	LiH		Neutron and gamma swelling vs. temp.
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		

Table 4. Representative Reactor Core Parameter Lists

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 5. National Laboratory and NASA Center Areas of Support and Collaboration

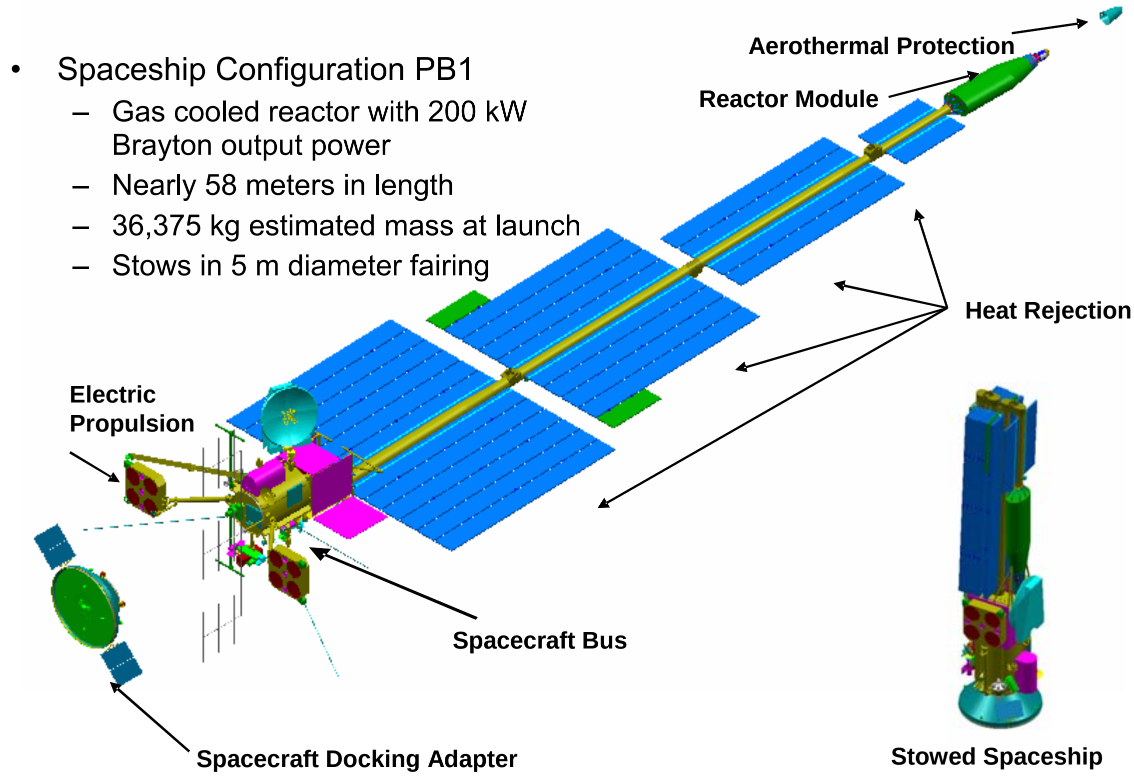
NASA Centers

- JPL
  - Program Coordination
  - Instrumentation and Control Integration
  - Environmental Effects
  - Radiation Hardening
  - Thermoelectrics Expertise
- GRC
  - Brayton Expertise
  - Stirling Expertise
  - Brayton System Testing
  - HRS Interface
  - Power Management
  - Structural Materials
- MSFC
  - Core and Loop Mock-up Testing
  - Heat Pipe Testing
  - Liquid Metal Testing
- KSC
  - ATLO Coordination
- ARC
  - Aeroshell Design

DOE Labs

- Oak Ridge
  - Shielding Development
  - Irradiation Testing (HFIR)
  - Sensor Development
  - Reactor Safety
  - Structural Materials
  - Compatibility Testing
  - Independent Cost Estimate
- Los Alamos
  - Rx design support
  - Fuel fabrication
  - Critical assembly testing (TA-18)
  - Heat Pipe Development
  - Transient Model Development
- Y-12
  - Shield and Fuel Fabrication
- INL
  - ATR Testing
  - Modeling Support
- PNNL
  - Foreign Test Reactor Support
  - Fuel Modeling
- Sandia
  - Safety Testing Guidance
  - Independent Cost Estimate
  - Gas Brayton Modeling
- Argonne
  - Fast Reactor Experience
- Lawrence Livermore
  - Reactor safety modeling
  - Fuel Science

Figure 1. Reference Spacecraft Concept (from Northrop Grumman) and Reactor Plant Concept



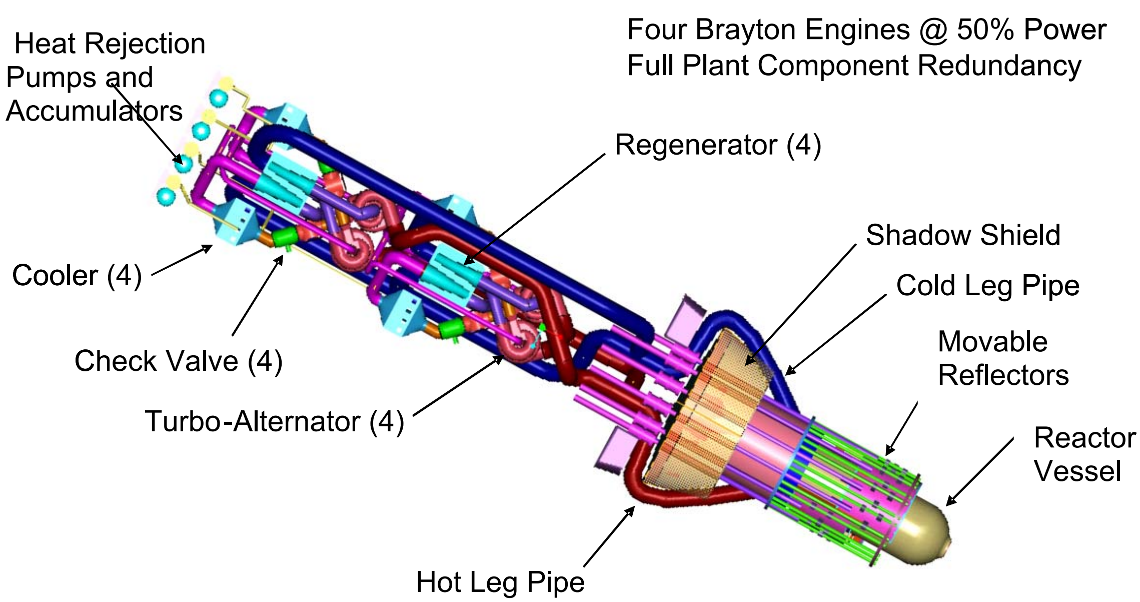


Figure 2. Reference Core and Fuel Element

Overall Reactor Configuration **Outlet Plenum Detail**

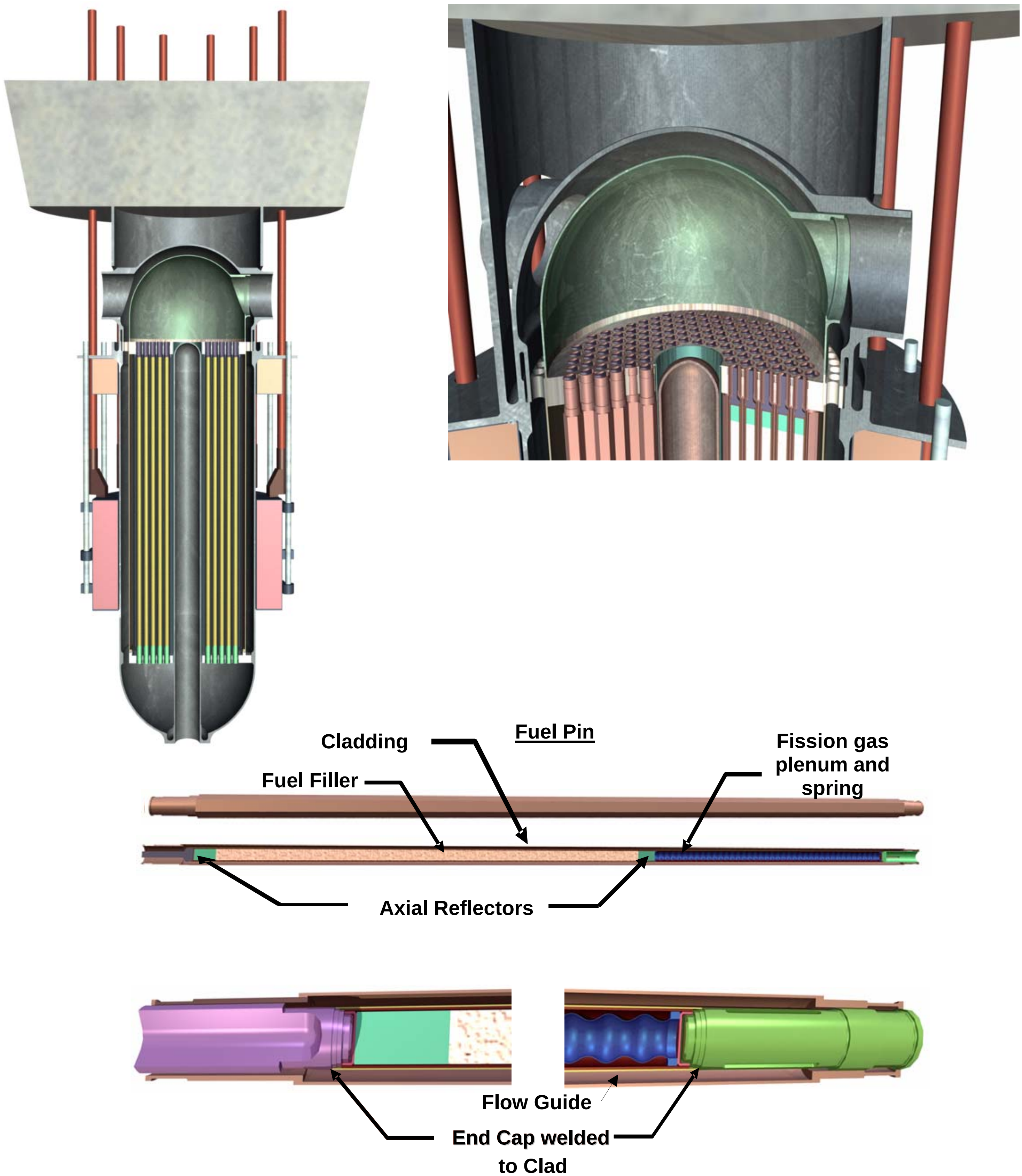
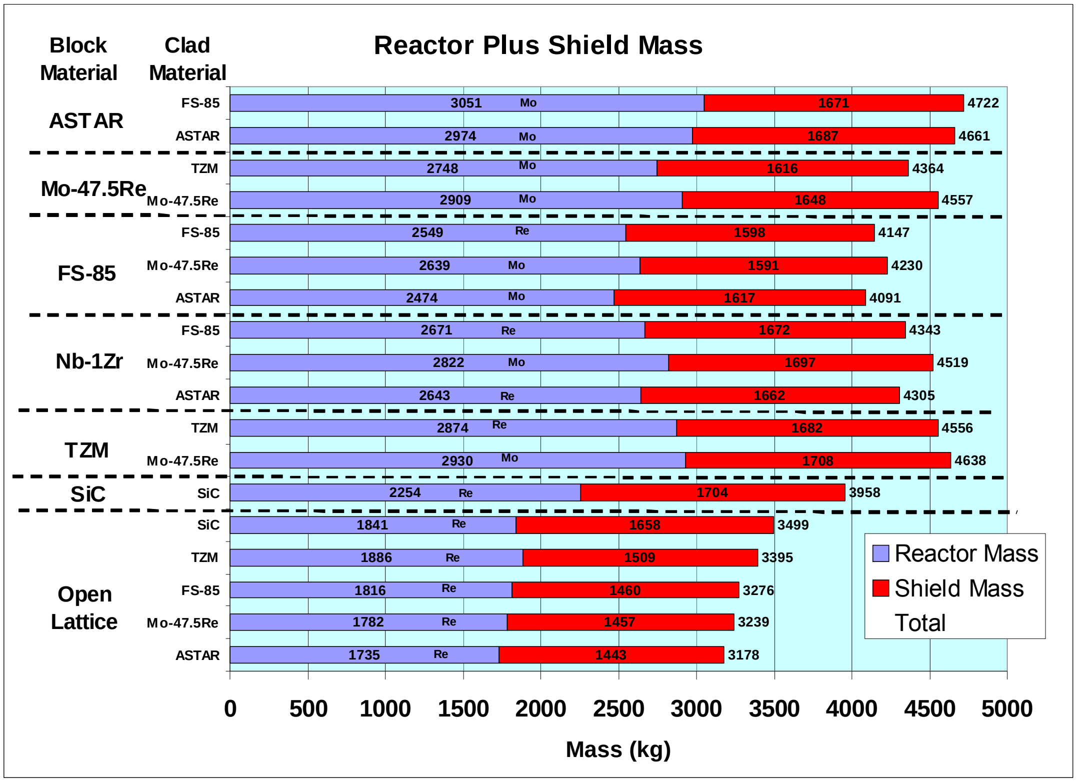


Figure 3. Reactor Core Configuration Options

PP File

Figure 4. Estimated Range of Space Nuclear Power Plant Mass as a Function of Reactor Design and Plant Arrangement. Reactor and shield mass shows masses as a function of core materials and arrangement for 1 MW reactor systems. The total SNPP mass shows the contributions to the SNPP mass, by subsystem, as a function of the number of energy conversion loops and the operating conditions. Note that the higher cycle efficiency of the simpler plant options leads to a lower required reactor power.



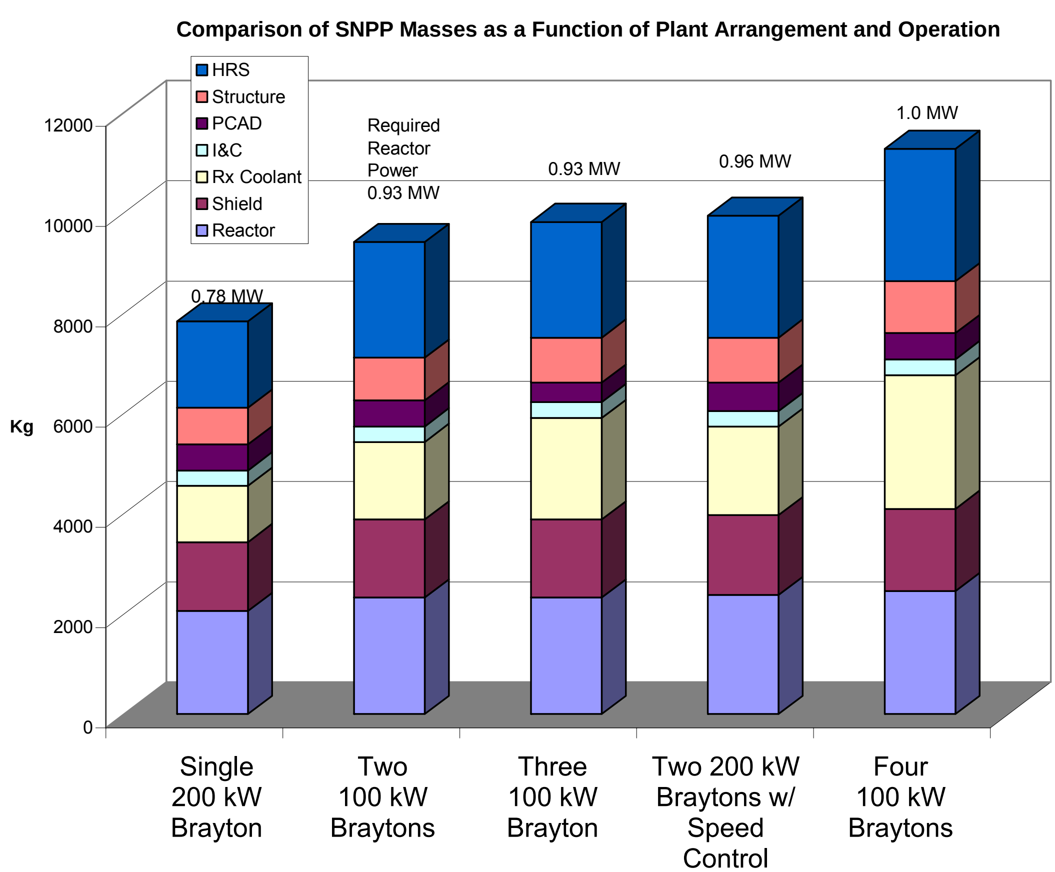
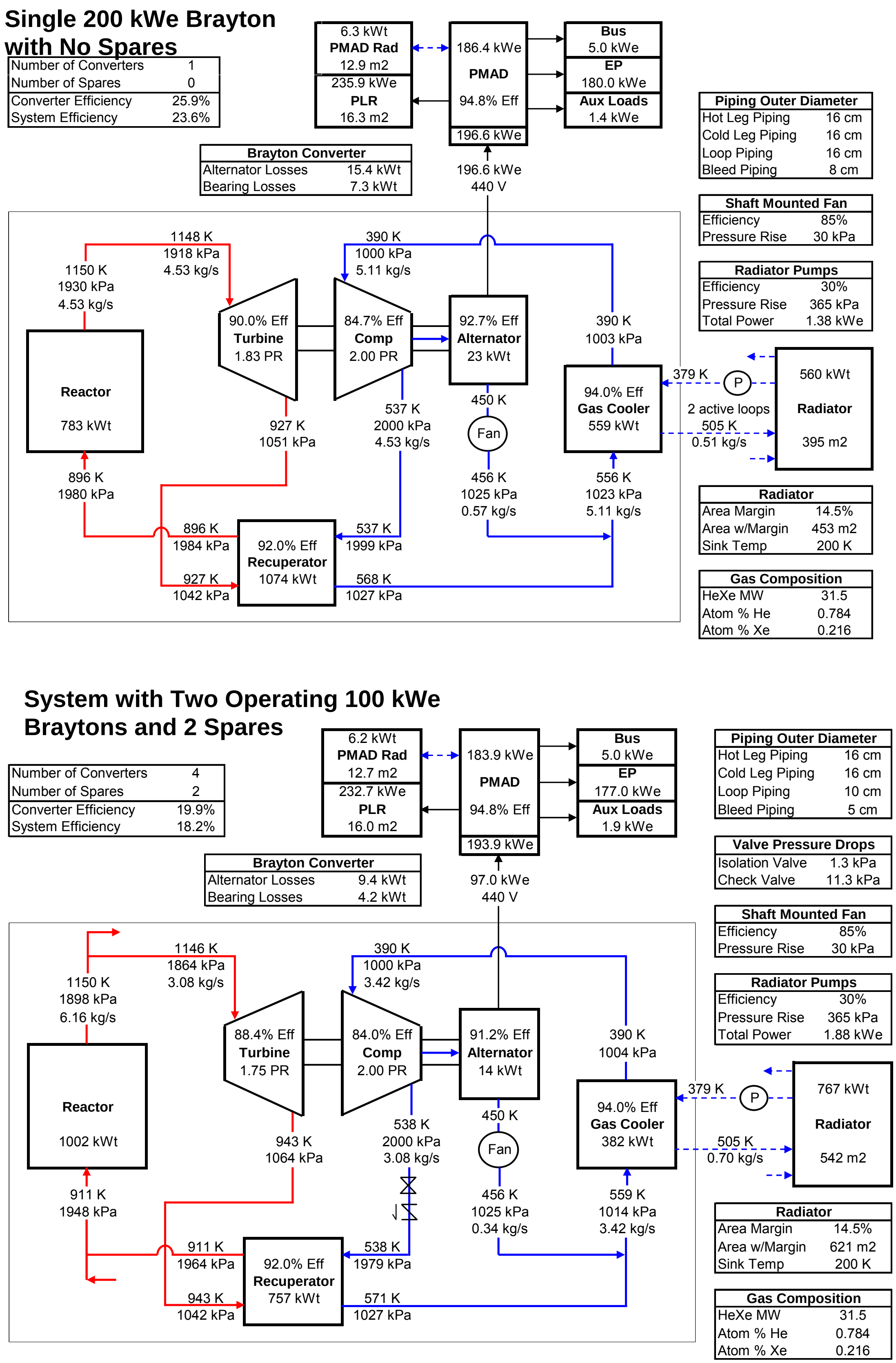


Figure 5. Reference Heat Balances for Single Brayton and Multi-Brayton System



PMAD – Power Management and Distribution
PLR – Parasitic Load Regulator and associated Resistive Load Bank Radiator
EP – Electric Propulsion
Aux Loads – Loads for Heat Rejection System Pumps and Reactor Plant Instrumentation
PMAD Rad – Radiator for rejection of heat from electrical loads

Figure 6. UN Fuel Processing Approach and Resulting UN Pellets and Microstructure

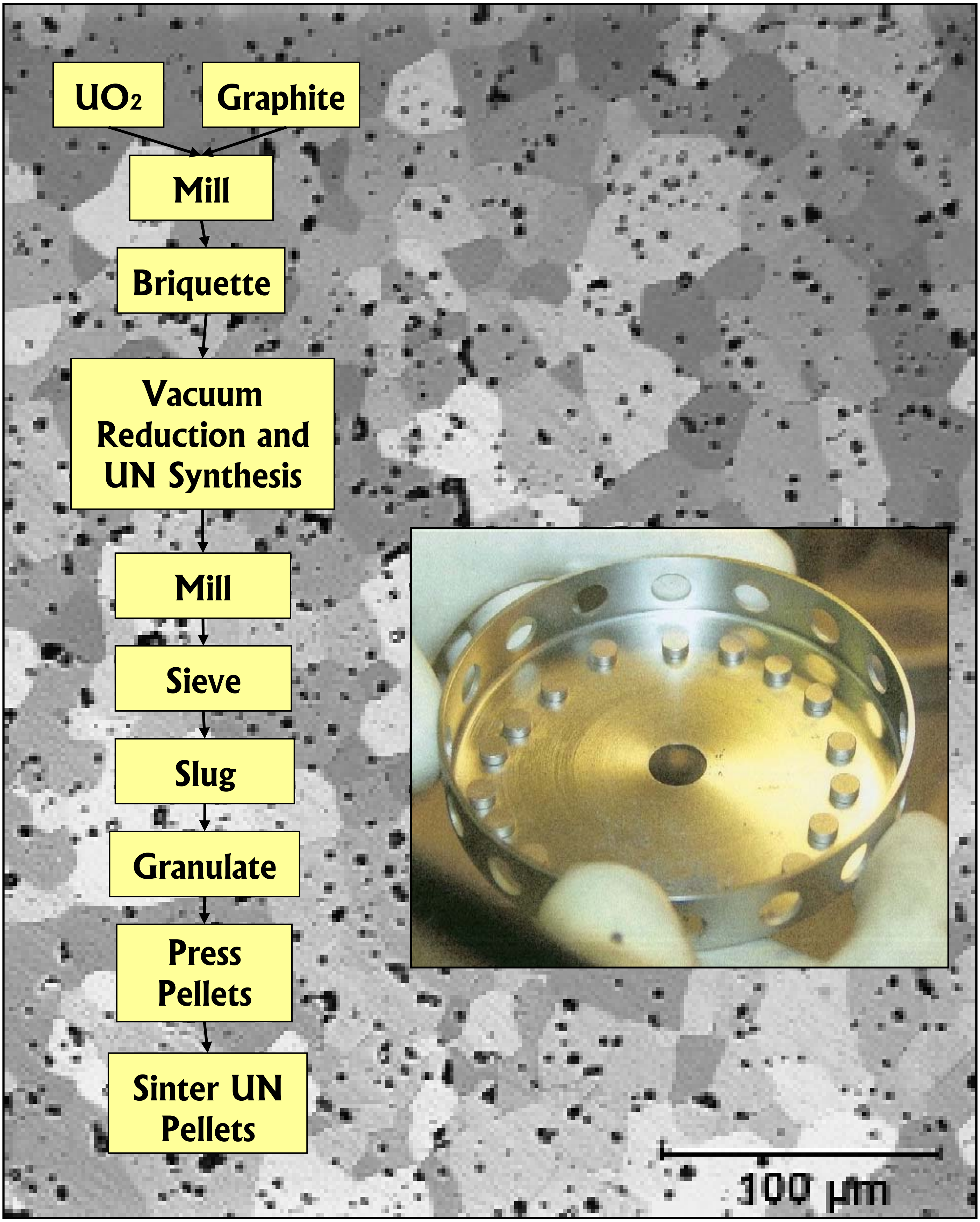
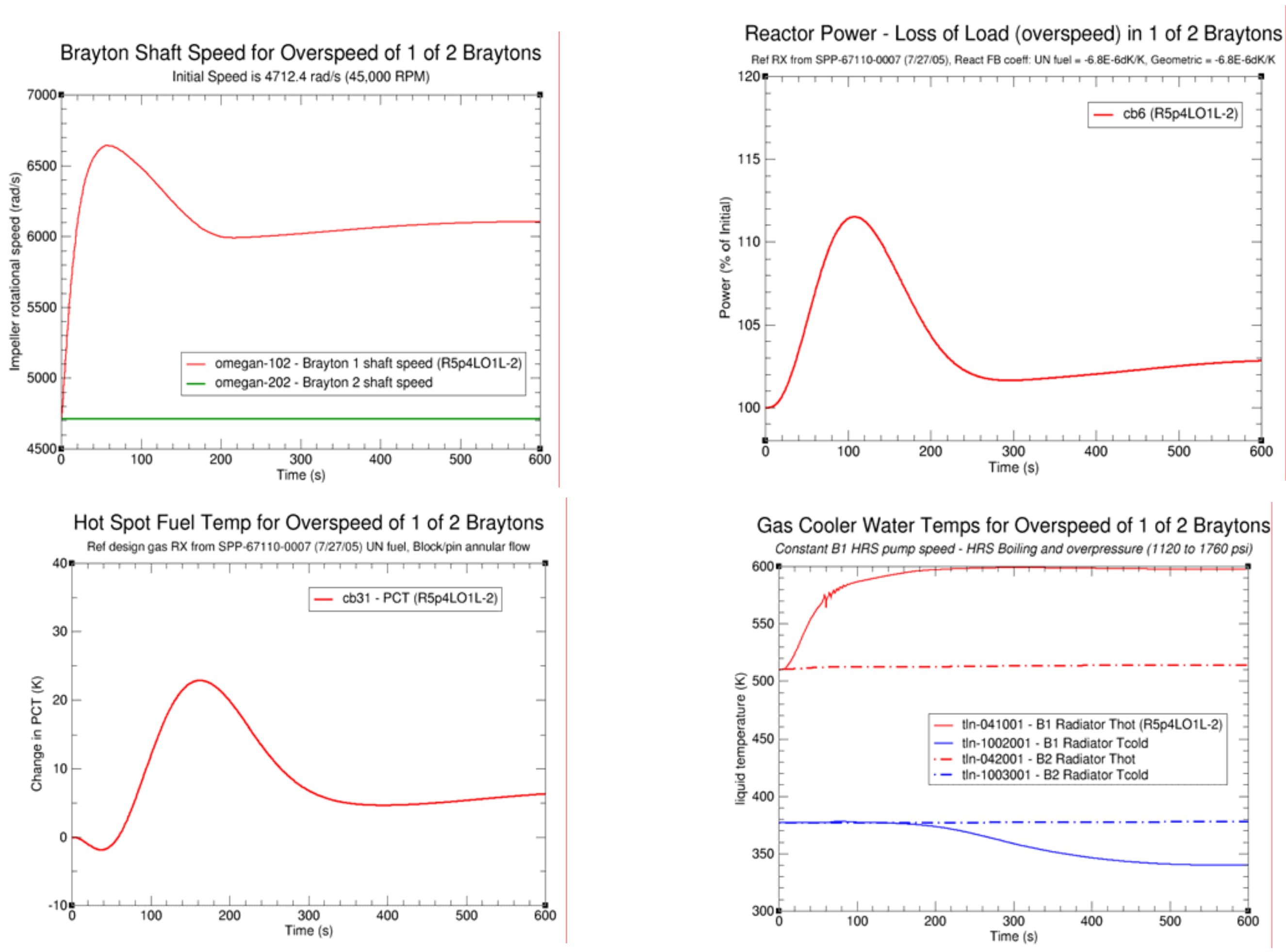


Figure 7. Reactor Plant Response to a Loss of Load Control in a Multi-Brayton Plant



The upper left plot shows the speed up of a turbine due to a loss of load on the alternator in that turbine. The speed increases until the point for the condenser work balances the turbine work. This increase in speed leads to a corresponding increase in flow which effects the entire plant. The upper right plot shows the change in power level following a loss of Brayton load in one of two operating loops. The lower left figure shows the corresponding change in maximum fuel temperature. Note that since the Brayton speed and flow rate increase during a loss of Brayton load transient, both the reactor power and peak fuel temperature rise. The lower right figure shows the change in heat rejection system coolant temperature for the two operating Brayton loops. Note that the temperature exiting the gas cooler rises sharply during this transient. This leads to an increase in pressure and the potential for boiling in this heat rejection loop and also leads to an increase in the likelihood of radiator heat pipe failures. Thus, this casualty has a greater impact on the heat rejection system than on the reactor.

Figure 8. Multiple Brayton Arrangement Options

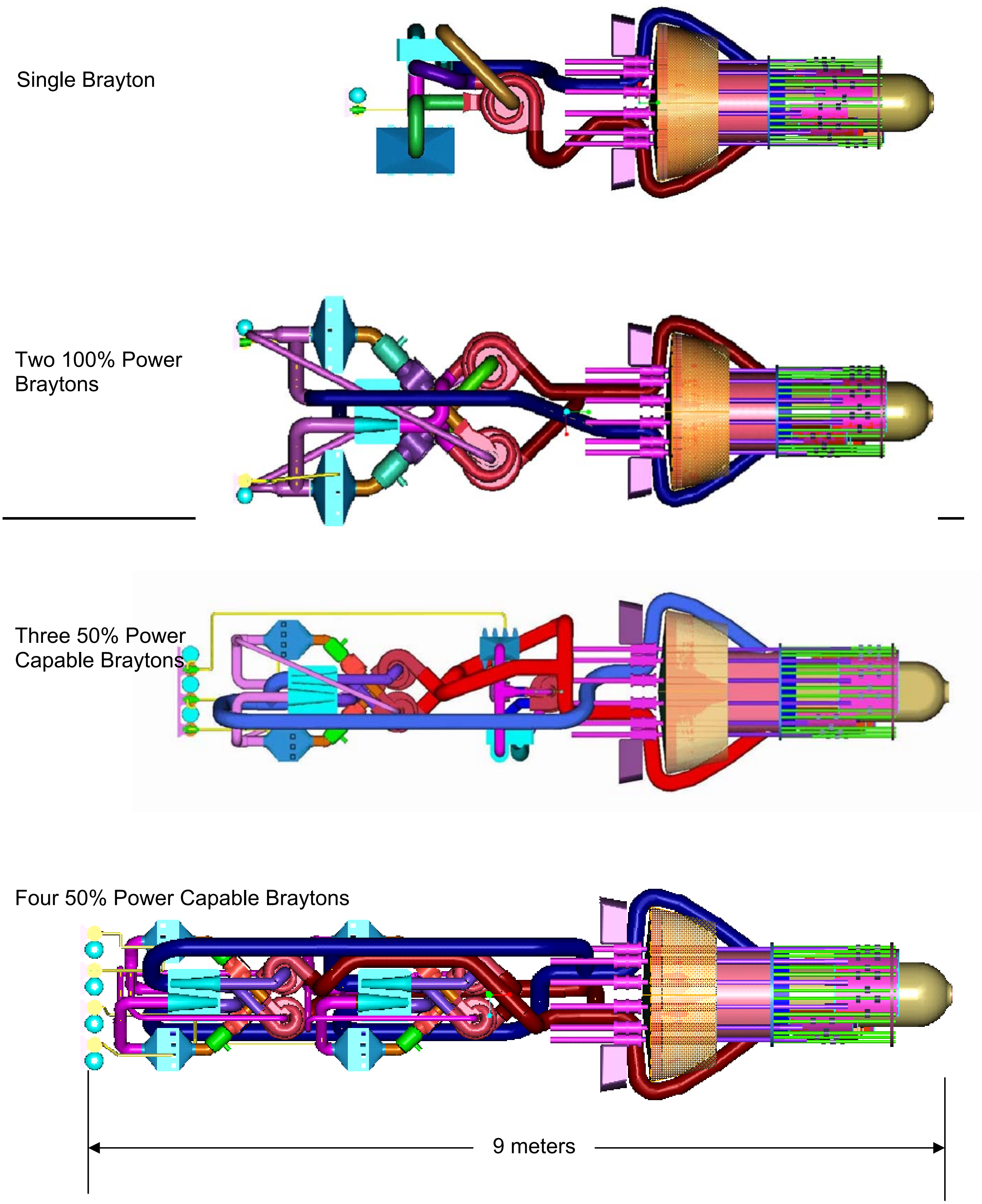
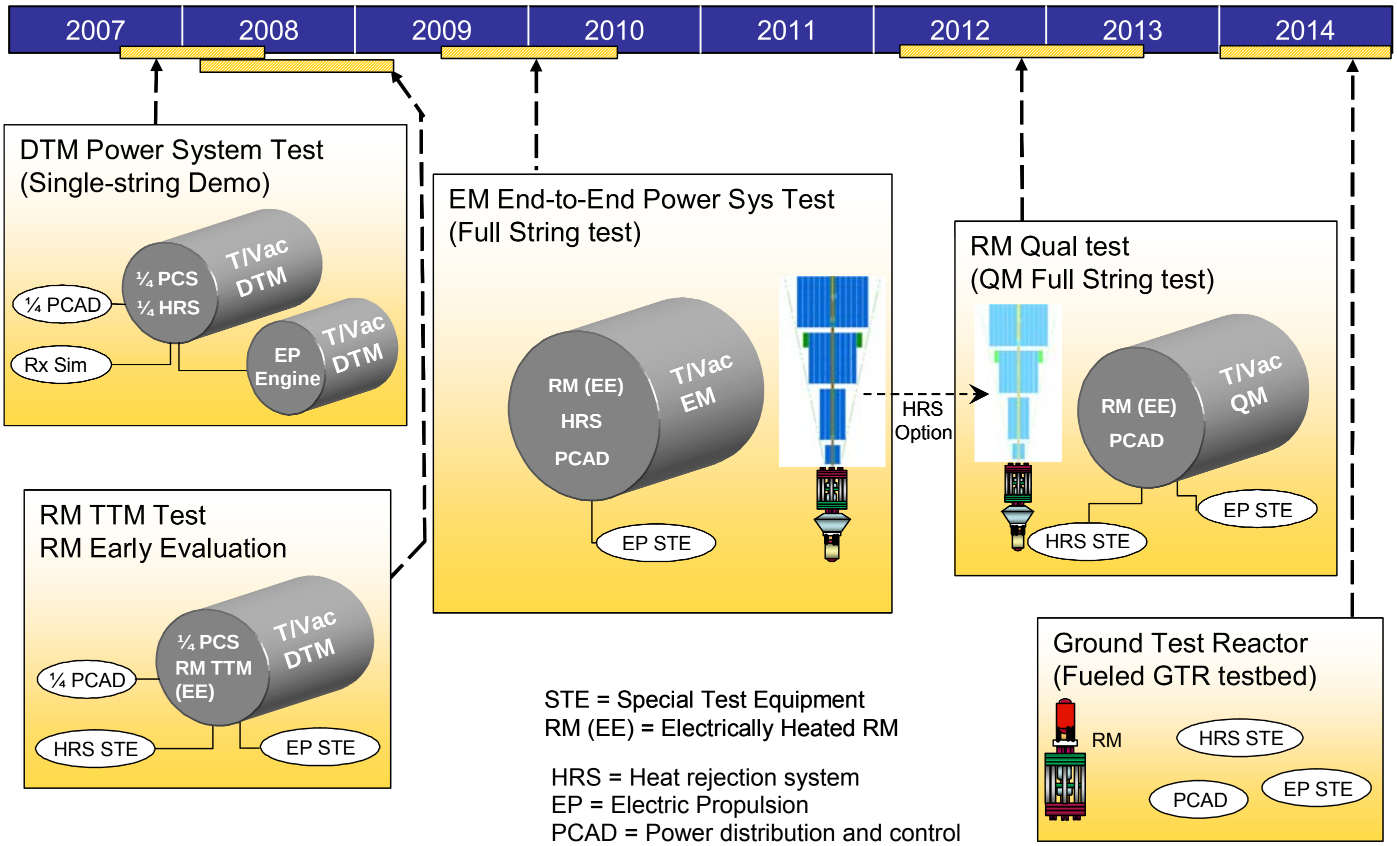
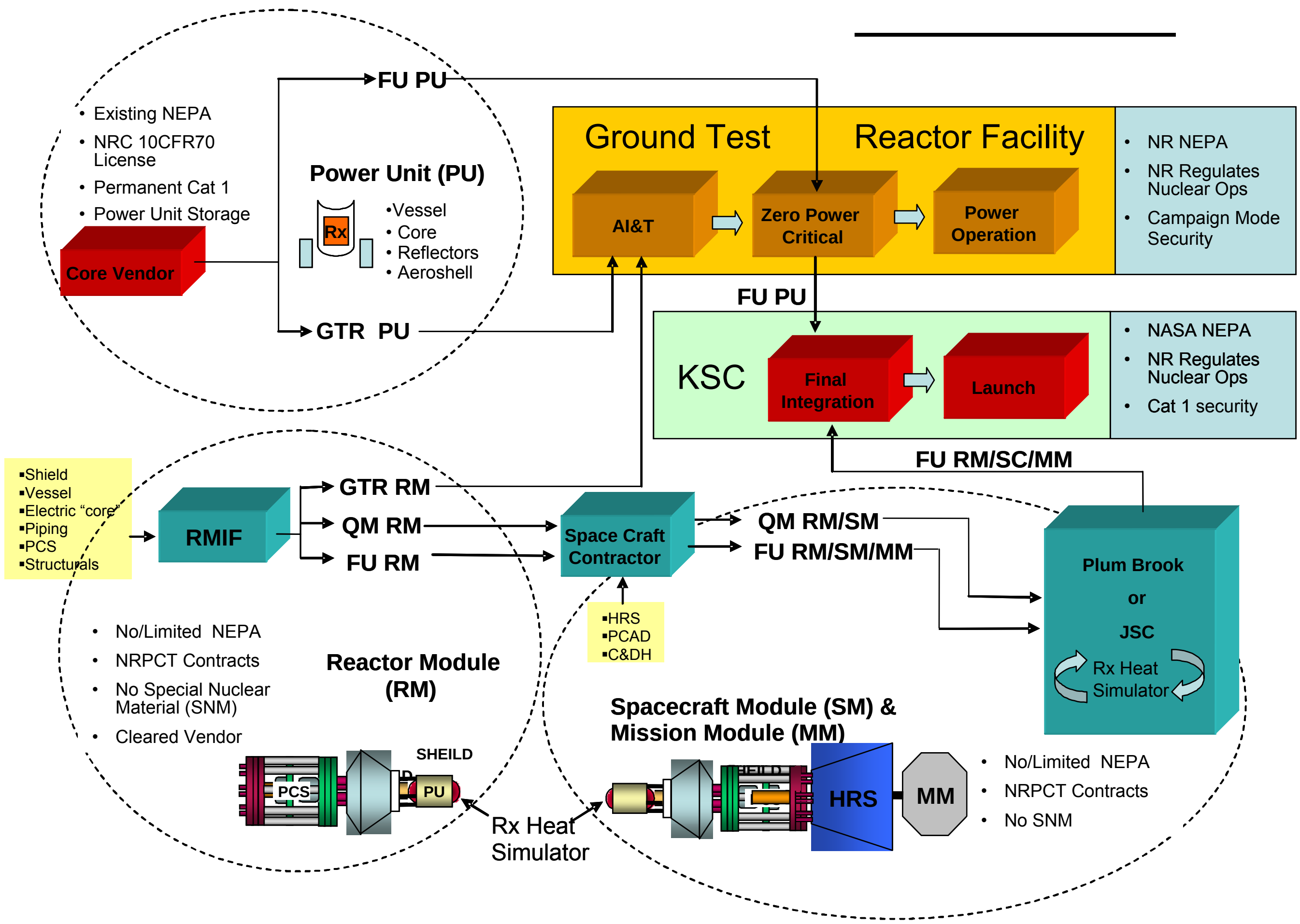


Figure 9. Approach and Timeline for System Testing



System testing proceeds from small scale components tests to the first integrated system test (the "DTM") to larger scale and more integrated testing to demonstrate system features in electrically heats tests (EM and QM). Comparable cross-section testing, cold critical and hot critical testing was planned.

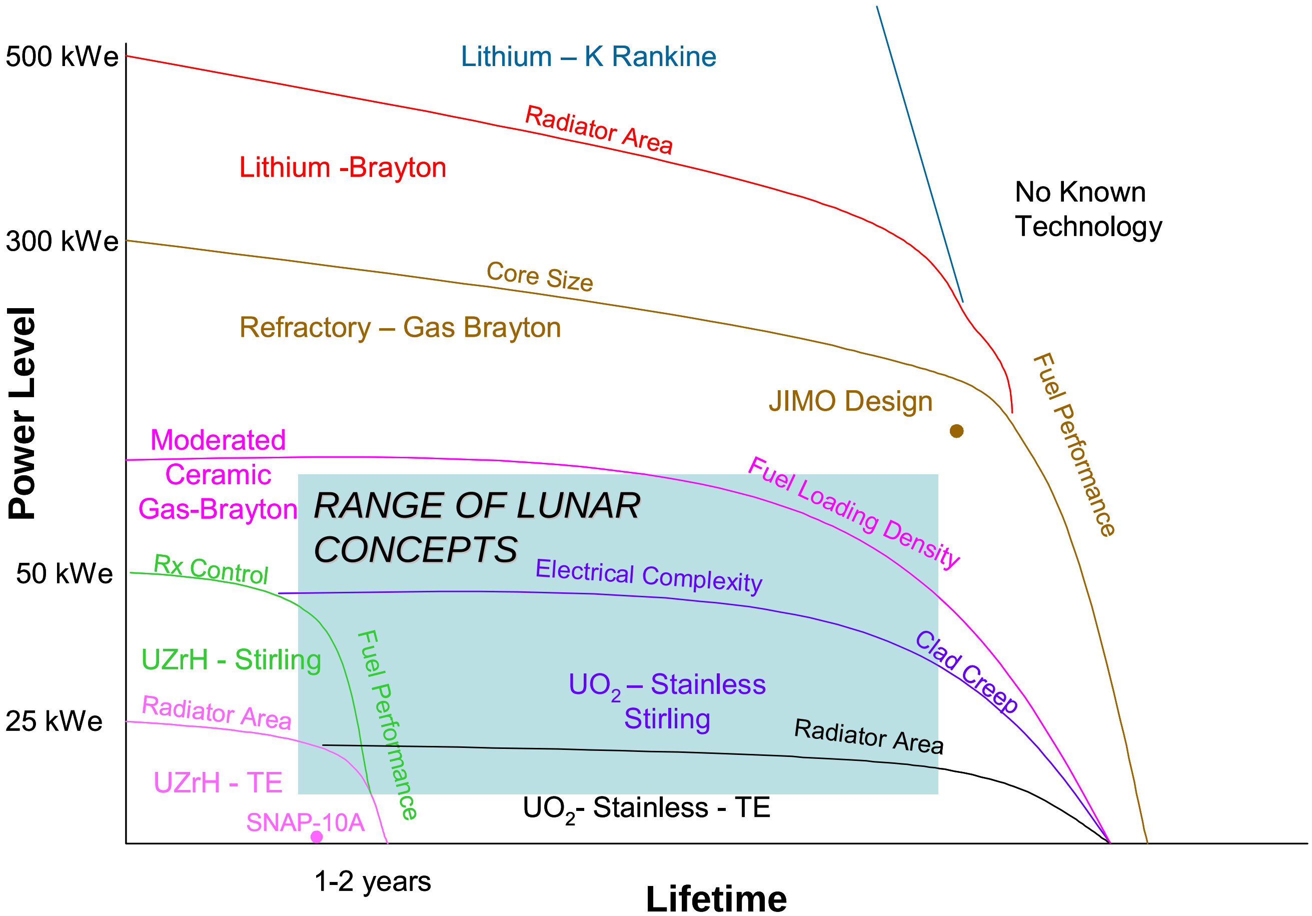
Figure 10. Assembly Test Launch Operations Plan



The goal of ATLO flow was to minimize transport of special nuclear material while assuring that flight components were tested to the maximum extent practical. Separate fabrication and testing paths were created for the nuclear materials and all other portions of the reactor module.

RMIF – Reactor module integration facility
GTR – Ground Test Reactor
EM – Engineering Module
QM – Qualification Module
FU – Flight Unit
Cat 1 – Category 1 Nuclear Security Facility

Figure 11. Reactor Technology Applicability as a Function of Power and Energy



The lower temperature, more readily available reactor technologies require greater energy conversion system development and provide limited power and lifetime capability

Figure 12. Heavy Water Moderated Gas Reactor Concept