APPLICATION OF ANALYTICAL HIERARCHY PROCESS FOR NARROWING DOWN NEP CANDIDATE REACTOR DESIGNS

Dasari V. Rao, Kurt Polzin†, Michael Houts†, Frank Curran‡, and Mitchell Rodriguez††*

** Los Alamos National Laboratory, Los Alamos, New Mexico*
† NASA George C. Marshall Space Flight Center, Huntsville, Alabama
‡ MZNBLUE Corp., New Market, Alabama
††Jacobs Space Exploration Group, Huntsville, Alabama

ABSTRACT

NASA is currently studying the possibility of human exploration of Mars in the late 2030s timeframe using a hybrid nuclear electric propulsion (NEP)/chemical propulsion system. Present analyses indicate such a mission could require a reactor power system in the range of 2 to 6 MW_e operating for approximately 3 years. At present, the Technology Readiness Levels (TRL) of key NEP technologies are still low, implying that significant technology development activities must be undertaken prior to making informed design selections for an integrated propulsion system. NASA's Space Nuclear Propulsion (SNP) project has recently undertaken an in-depth effort to identify NEP candidate technologies and determine the required developmental work to mature these technologies to the point where they could realistically support the design and assembly of a full NEP system. A version of the Analytical Hierarchy Process is being employed to narrow the technology candidates for SNP investment planning. This multi-attribute decision-making process relies on quantified technical inputs and the judgement of Subject Matter Experts (SMEs) to evaluate technology options against key technical and programmatic Figures-of-Merit (FOM) at the subsystem and optimized system level. The process and an example evaluation of nuclear fuel/moderator combinations in a sodium heat pipe honeycomb-geometry reactor concept are described. The evaluation and analysis methodology described in this paper can be employed for pairwise comparisons between reactor concepts and technology choices as new data become available.

INTRODUCTION

NASA consideration of Nuclear Electric Propulsion (NEP) for high-value robotic and human exploration began in the late 1950's and continues to this day. This continued interest stems from the reduced propellant mass that high specific impulse (I_{sp}) NEP systems would provide compared to conventional chemical propulsion systems. Figure 1 presents a notional NEP system which can be functionally broken into five Critical Technology Elements (CTE) – Power Generation (CTE 1), Power Conversion (CTE 2), Power Management and Distribution (CTE 3), Electric Propulsion (CTE 4), and Heat Rejection (CTE 5). Each of these sub-systems must be designed to mesh on the system level. For example, the output of the system extracting heat from the nuclear reactor (CTE 1) must be compatible with the input to the selected power converter system.

Examples of significant NASA investments in NEP include the Systems for Nuclear Auxiliary Power (SNAP) program of the 1960's,[1,2] the Space Exploration Initiative (SEI)[3,4] in the 1990's performed in conjunction with the Space Power 100 (SP-100) program,[5,6] and the Jupiter Icy Moon Orbiter (JIMO) program that became Project Prometheus in the 2000's.[7,8] Despite these sporadic investments, a major finding of a recent review by the National Academies of Science, Engineering, and Medicine (NASEM)[9] stated:

> *"As a result of low and intermittent investment over the past several decades, it is unclear if even an aggressive program would be able to develop an NEP system capable of executing the baseline mission in 2039."*

with a recommendation that:

"NASA should invigorate technology development associated with the fundamental nuclear electric propulsion (NEP) challenge, which is to scale up the operating power of each NEP subsystem and to develop an integrated NEP system suitable for the baseline mission. In addition, NASA should put in place plans for (1) demonstrating the operational reliability of an integrated NEP system over its multiyear lifetime and (2) developing a large-scale chemical propulsion system that is compatible with NEP."

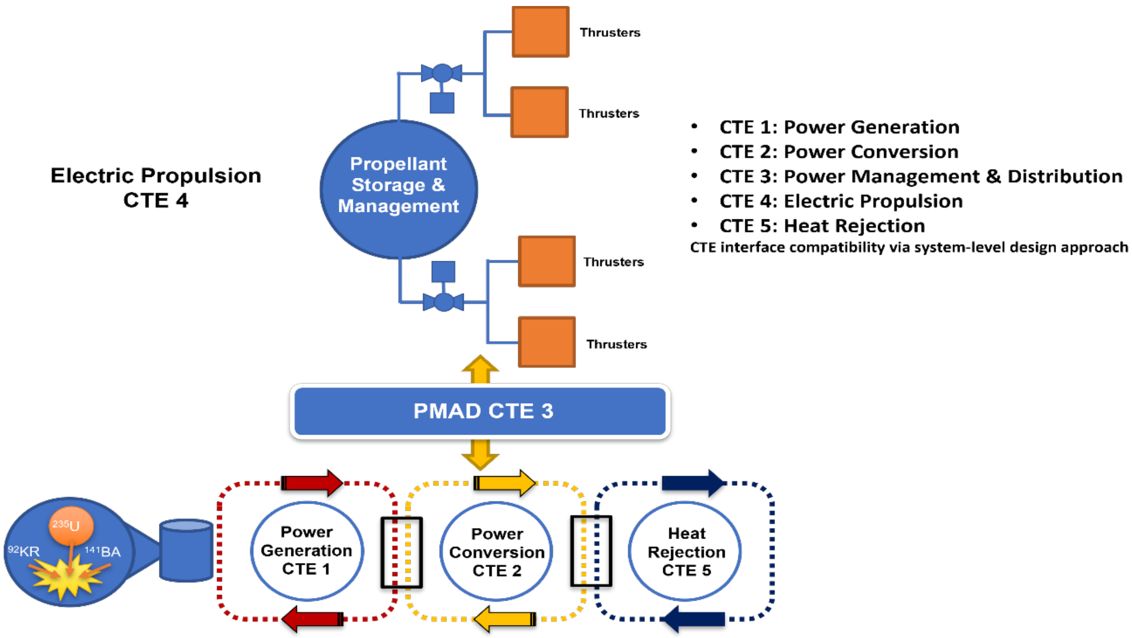

Figure 1. NEP System Functional Block Diagram.

Another recent, comprehensive report on the status of nuclear propulsion technologies performed by the NASA Engineering & Safety Center (NESC)[10] assessed a majority of candidate NEP technologies as having technology readiness levels (TRLs) of 4 and below on the component/subsystem level (with commensurately lower TRLs on the system level). That work also determined that most of the critical technologies had advancement degrees of difficulty (AD^2) ratings high enough to require parallel-path research and development (R&D) efforts on the various CTEs, carrying more than one option for each element to reduce risk to an acceptable level.

In response to these findings, NASA's Space Nuclear Propulsion (SNP) project is conducting work that 1) identifies system requirements and the most likely technology candidates based on quantitative assessments and 2) will result in an actionable plan to advance these technologies to TRL/AD^2 levels that permit reliable assessments of efficacy on the NEP system level. For this, a comprehensive technology maturation plan (TMP) is under development. The TMP rests on three legs – 1) technology identification and status assessment, 2) modeling and simulation (M&S) to evaluate technology impacts on the system level, and 3) subject matter expert (SME)-driven technology selection exercises followed by non-advocate review.

The technology identification portion of the program was accomplished through a series of technical interchange meetings (TIM) covering each of the CTE's.[11] These SNP-sponsored TIMs (held virtually) were broadly attended (typically 100+ attendees) by industry, academia, and other government agencies. These meetings provided both the state-of-the-art (SOA) for candidate technologies and insight into relevant ongoing activities for consideration in program planning.

The required M&S is ongoing and takes place on two levels. The first level consists of SME evaluations of the various technologies to develop component, element and subsystem level technology attributes required for system level analysis. The system level analysis is accomplished through an SNP-sponsored modeling effort performed by Analytical Mechanics Associates (AMA) and reviewed in Ref. [12] The AMA model is modular and provides the ability to evaluate the impacts of selected technologies on key system performance metrics (e.g., system α).

As noted above, all of the CTE candidates (and their associated technologies) are at TRL/AD[2] levels not commensurate with down-selection to exclusive development. As always, resource and time constraints require careful selection of the candidates with the most potential. For this, SNP has chosen to employ a version of the analytical hierarchy process (AHP)[13,14] for technology evaluation. As applied by SNP, the process first generates the key figures-of-merit (FOM) to describe desired system attributes. These are further decomposed into sub-elements (key attribute descriptors) as required. Comparisons are then made in a pair-wise fashion based on SME valuations to narrow the technology selection space. The process as applied by SNP is described briefly below along with a specific example of the process for the selection of an optimal reactor fuel/moderator combination.

THE PROBLEM STATEMENT AND APPROACH

Extensive NASA-sponsored R&D has been performed on nuclear fuels and moderators over the past 6 decades. Perhaps the most focused effort of relevance to NEP was in the SP-100 program.[5,6] This R&D focused on pellet-based highly enriched uranium mono-nitride (HEU-UN) fuel. UN possesses unique advantages, such as high uranium density, high thermal conductivity, and a high melting point, enabling extremely compact, high-temperature fast spectrum reactors. More recently, significant resources are supporting development of tailored UN TRi-structural ISOtropic (TRISO) micro-engineered particle (MEP) fuels for terrestrial commercial and Department of Defense (DoD) applications and for in-space NASA and DoD nuclear thermal propulsion (NTP).[15] Since fuel development is costly and time-consuming, The challenge to SNP is to identify the fuel/moderator combination that is best suited for the NEP application while keeping an eye towards progress being made in parallel by the NTP project. To start this process, SNP is examining minimum viable nuclear reactor alternatives for powering nuclear electric propulsion spacecraft.

It must be emphasized here that the work presented in this paper is meant to be taken as an illustration of how AHP can be applied to the problem of narrowing the field of candidate reactor technologies. The evaluations of different reactor concepts and candidate fuel and moderator options, including those contained in this paper, are still in-work and the data and discussion presented are not meant to imply in any way that the SNP project has selected specific technologies options and design choices for the NASA NEP application. Furthermore, it is anticipated existing AHP valuations will be revisited and evaluations of additional technology options will be performed as more data become available. Finally, while the government is presently performing these evaluations, it is hoped that during any proposal process external partners will apply AHP in the same manner to evaluate their preferred technology options relative to those investigated by the government.

For the purpose of this paper and the example valuation contained herein, it is assumed that reactor would generate fission heat sufficient for the production of 2 to 6 MW_e when coupled to a closed cycle Brayton power conversion system with a power conversion inlet temperature of 1200 K. Three full years of operation are required and high assay low enriched uranium (HALEU) is used, aligning with Space Policy Directive-6. Although a required system alpha (kg/kWe) has not been specified, it is assumed that "lighter is better" with respect to FOM/sub-element scoring. The reactor technology should also possess some measure of fault-tolerance to the level of reliability required for a human mission. Based on the results of several TIMs, a heat pipe reactor design was selected as the initial basis for comparison for the fuel/moderator evaluation described herein. Other reactor concepts (e.g., HeXe gas and liquid Li-cooled designs) will be similarly evaluated in the future. In the selected heat pipe reactor case, fission heat generated in the fuel is transferred to the heat pipes via conduction through intervening material interfaces. The reactor concept is shown in **Figure 2.** In this design, the core is divided into several smaller fuel assemblies consistent with design for manufacturing (DFM) practices. The fuel core is surrounded by a BeO reflector that houses control drums which act as thermostats for the reactor. Successive outer annular layers of metal alloys and neutron absorbers are engineered to maintain the core fuel assemblies and the reflector elements under compression. This figure also shows a base 7-assembly design for fabrication. **Figure 3** shows three variants of fuel assemblies that house the fuel, the moderator and the heat pipes – these are designed to optimize fuel and moderator performance. The configuration on the left uses larger fuel channels to take advantage of low fission densities typical of TRISO fuel compacts whereas the other two configurations use smaller (5-mm) fuel channels to avoid neutron self-shielding and to minimize temperature drop across the pellet. Similarly, the configurations on the left and middle use clad, insulated self-cooling segments of YH while configuration on the right firmly bonds BeO or Be_2C moderators to the graphite matrix to enable efficient conduction across the core. In all three designs, the fuel-to-graphite and fuel-to-moderator volume ratios were selected to assure negative reactivity feedback (~ -1 pcm/K) at operating conditions. Secondly, materials and technologies choices were made to assure sufficient thermal and structural margin to accommodate both the normal operating and severe transient conditions. Together these features assure that self-regulation, an inherent mechanism by which a reactor regulates itself, can be relied upon to simplify the control system. Industry standard CAD software was used for authoring the geometries and for configuration control. Fully coupled multi-physics analyses examined neutronic, thermal and mechanical performance of each assembly design including its ability to withstand extreme heat pipe failures (such as simultaneous failure of three adjacent heat pipes) and power conversion transients such as loss of complete heat removal over extended periods. Temperature and power density limits were selected to minimize thermal stresses during postulated off-normal operating conditions and specific mass estimates were directly obtained from the CAD models.

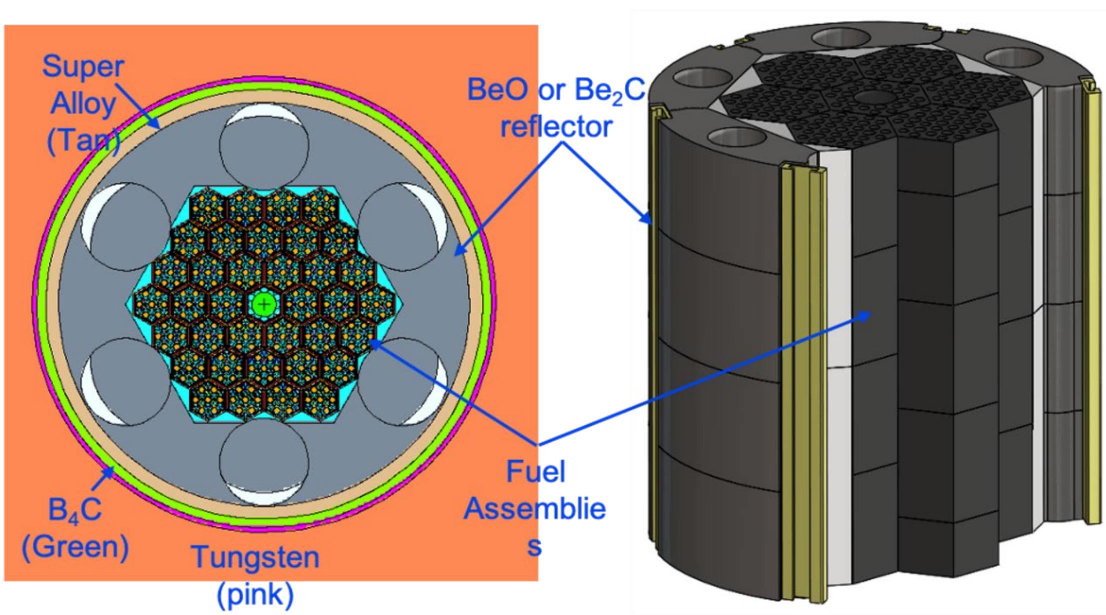

Figure 2. Cross-sectional view of a 37-assembly heat pipe reactor (left) and a notional design for a 7-assembly layout being used for design development (right).

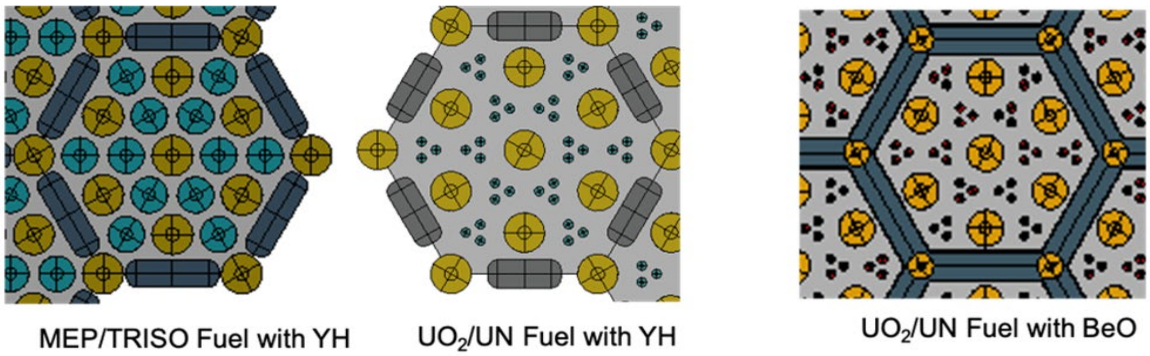

Figure 3. Cross-sectional views of tailored assembly design examples for different fuel/moderator combinations. MEP/TRISO fuel compacts are 2 cm in diameter. UN and UO_2-based fuel pellets are ~ 0.5 cm in diameter to accommodate fuel density variability. YH segments are individually insulated and BeO and Be_2C elements are configured to optimize the robustness of the structure. All fuel cases incorporate SiC_f-SiC cladding for structural support.

The rationale for the selection of the fuel/moderator combinations for evaluation along with full descriptions is provided elsewhere.[15] Briefly, alternatives selected for pairwise comparison are:

- Four candidate **fuel** forms: (1) pellets of uranium dioxide (UO_2); (2) pellets of uranium mono-nitride (UN); (3) Advanced Gas-Cooled Reactor (AGR) TRISO UCO particle fuel[16]; and (4) a more generic micro-engineered particle fuel[17]. UO_2 pellets are widely used in commercial reactors and US possesses numerous commercial ventures that can fabricate qualified UO_2 fuel[15]. UN pellets are also a well demonstrated fuel form[15], possess superior fuel density and thermal conductivity, and are well suited for space application. However they are not presently used in US and their nuclear qualification data was obtained primarily in the fast neutron energy range. Both UO_2 and UN fuels have also been demonstrated in particle form as well. Advanced Gas Cooled Reactor (AGR) TRISO fuel is a fully nuclear qualified fuel form. It uses coated 425-micron $UO_{1.6}C_{0.4}$ (a chemical variant of UO_2) with a smeared uranium density < 1 g/cc. Micro-Engineered Particle (MEP) fuel is a term used to broadly capture particle fuels that range from Bistructural ISOtroipic (BISO) to TRISO to achieve higher smeared fuel density (target: 2-3 g/cc). The Fully Ceramic Microencapsulated (FCM) TRISO is one such fuel[17]. It uses a larger UN kernel (800-micron diameter) and has undergone limited nuclear testing. Other examples are BISO particle fuels being considered for the NTP mission. These particles are typically small in size and are coated with one or two layers of carbides. However, none of the MEP fuels are qualified fuel forms; extensive characterization and irradiation campaigns would be required to qualify this fuel type. Reference [15] further describes the fuel forms being considered in this study.
- Three candidate **moderators**: (1) beryllium oxide (BeO); (2) beryllium carbide (Be_2C); and (3) yttrium hydride (YH). Of these three, BeO is most widely used in commercial industry, but possesses the least moderating power. Both Be_2C and YH have previously been used in high temperature reactors, and efforts are underway to re-capture technology base for these moderators. Reference [15] further describes the moderators being considered in this study.

THE EVALUATION PROCESS

Simply stated, the Analytic Hierarchy Process (AHP) as applied to an NEP system provides a logical approach to decision-making by breaking a complex problem into more easily understandable parts, known here as Figures-of-Merit (FOM), that can be analyzed separately by Subject Matter Experts (SME). The FOMs form the top of the hierarchy and are, in turn, broken into sub-elements as necessary to further simplify and quantify the expert assessment process. While we have not done so here, the sub-elements can be quantitatively "scored" by the SMEs for each candidate technology and these scores can subsequently be rolled-up to provide an overall score for each FOM. The FOM scores are then combined to provide a final score, or rating, of each potential technology. Both the FOMs and the sub-elements can be given weighting factors prior to combining them into the next higher level scoring.

As noted above, this is an introductory exercise for NEP and is focused on the evaluation of multiple nuclear fuel/moderator combinations for honeycomb reactor concepts incorporating sodium heat pipes (NaHP) for heat transfer to a Brayton cycle-based power conversion system.

From the outset of our efforts, SNP has focused on specific project development goals and these were stated at the outset of each TIM. These goals formed the basis for the FOMs and are paraphrased here as the identification of:

- Specific "high-confidence" candidate technologies and approaches for development,
- Key areas for potential collaboration with external partners and feasible execution strategies, and
- Technology maturation solutions that take advantage of proven capabilities and both past and ongoing investments.

SNP is looking for the best overall system solution. In addition to system mass and performance, the characteristics of manufacturability, operability, testability, robustness, cost, industry alignment, and other factors are under consideration. In the present work, SMEs selected three FOM's for the top level of the hierarchy for evaluating fuel/moderator combinations, and these were further subdivided into three sub-elements.

FOM 1: Performance describes the ability of the chosen fuel/moderator combination to meet specific performance requirements as described by the sub-elements:

Power – capability to support reactor designs in 2 to 6 MWe range over 3-to-5-year lifetime.

Power Conversion Inlet Temperature (PCIT) – capability to provide a minimum outlet temperature of 1200 K.

Subsystem α – capability to minimize reactor α. Evaluation of relative mass contributions between options.

FOM 2: Technology Readiness describes the status of the fuel/moderator ranging from "available and ready to manufacture" to "R&D required before potential application." This FOM is subdivided into the simple status of the required materials – i.e., **Fuel**, **Moderator**, and **Insulation**.

FOM 3: Design Confidence describes the level of confidence that the SMEs have in fuel/moderator combination being incorporated into an effective reactor with respect to sub-elements for **Margin**, **Robustness** and **Ground-Testability**. It recognizes the fact that reactor core designs, at the present time, are not matured to the 'nuts-and-bolts' level of detail, but possess a sufficient level of fidelity to draw insights related to performance limits. For example, full-core coupled neutronic, thermal and structural analyses that explicitly examine impacts of single, multiple random, and multiple adjacent heat pipe failures as well as partial and complete loss of cooling by the power conversion system. Additional analyses are underway to accurately characterize reactor performance during other, more severe events (e.g., collision with space debris). Also, it is expected that ongoing structural assessments may result in reinforcing core internals with refractory alloys. FOM sub-elements of **"Margin"** and **"Robustness"** capture SME's confidence in the ability of the reactor to accommodate changes from anticipated design spirals. It is noted here all ground-testing of nuclear systems is difficult, and the "**Ground-Testability**" sub-element rests mainly on whether or not Beryllium or one of its alloys is part of the active core, which complicates handling and eventual disposal.

AHP analysis can range from highly-quantitative (in which numerical scoring criteria are generated) to qualitative (where simple high, medium and low scores are attached). The process described herein was semi-quantitative – that is, subjective color scores (based on specific calculations or definitive research into status) were given to each sub-element. The scoring could have been numerical and weighted, with the scores rolled up as described above.

RESULTS AND DISCUSSION

The results of the pairwise comparisons are shown in **Table 1**. In all designs the fuel assembly relies on an equivalent of 10 heat pipes (including heatpipes located on the periphery). Each heat pipe has a 2-cm outer diameter, is charged with purified sodium, and is constructed of 0.4 mm-thick molybdenum alloy tubing and a molybdenum alloy wick (32-micron pore diameter). Molybdenum walled sodium and lithium heatpipes of this dimension and material combination have been fabricated and subjected to long-term thermal cycling testing by LANL since 1985[19-22]. Maximum thermal throughput for a 1.4-cm diameter lithium heat-pipe operating at 1500 K was measured at 35 kWt. Such a heat pipe was operated for 2 full years before the test was decommissioned[21]. In the case of sodiumcharged heat pipes, evaluations included 45,000 hours of non-nuclear testing followed by post-test examination, SAFE-30 Testing at NASA Marshall[20], and operation in both the EBR-II and the Krusty nuclear reactor cores.

Design capacity of the proposed heat pipe is 15 kW_t, which is derated to 60% of the estimated throughput of 25 kW at 1200 K. Reference [18] provides the geometric and performance details of the proposed heatpipes. An uncertainty in the modeling is the contact conductance between the heat pipe and the graphite monolith. Another uncertainty relates to heat-pipe heat exchanger performance. Early non-nuclear tests are proposed to reduce these uncertainties to the level required to decide whether or not heat pipe reactors are feasible for the NEP timeline of interest[18].

Reactor designs using UO_2 and UN fuel pellets are based on the configurations in the center and on the right in Figure 2. Both these configurations used small diameter pellets (~5 mm diameter) encapsulated in ~ 1-mm thick SiC_f-SiC cladding. Additionally, a thin graphite buffer layer located between the pellet and the clad is designed to absorb structural loads associated with fission gas release and swelling). Note that UO_2 and UN fuel pellets are well understood fuel forms and nuclear qualified SiC_f-SiC clads are being developed under funded programs as part of DOE Office of Nuclear Energy's (DOE-NE) Accident Tolerant Fuels (ATF) program[16].

Based on these preliminary analyses, it is concluded that UO_2 and UN pellet fueled reactors are likely to enable reactor masses/sizes in the range of interest for NEP. Either option could be relied upon for early missions and both have sufficient margins to accommodate design uncertainties without exceeding mass limits. As expected UO_2 designs resulted in slightly higher fuel temperatures (1375 K in extreme cases where damage to adjacent heat pipes was modeled). Such temperatures would have been a problem with traditional metal cladding, but are well within the operating range of SiC_f-SiC cladding. UN generally resulted in slightly smaller core sizes with fuel operating temperatures slightly lower than what was predicted for UO2. The UN core, however, was found to have slightly higher performance uncertainty due to lack of irradiation performance data at the operating conditions of present interest. (Note: if ongoing analyses were to demonstrate a clear advantage of UN, then required qualification of UN at the burn-up and neutron spectrum of present interest should fit within a reasonable schedule.) In recognition of this difference, the UN pellet form was assigned "meets requirement" in comparison to UO_2, which is assigned "exceeds requirement". These two options may reverse in ranking once the ATF campaign results are factored in and after further analyses are performed. A technology maturation need for both these fuel forms is a robust SiC_f-SiC clad that is in place to reinforce the design against launch loads. Note that thermal spectrum reactors are not very sensitive to slight displacement or movement of fuel pellets during launch; as such fuel channels in the graphite matrix are sufficient to assure neutronic performance. The purpose of the SiC_f-SiC cladding is primarily to carry the load and prevent damage to the graphite.

On the particle fuel front, conclusions carry much higher uncertainty due to the lack of reliable experimental data. It can be stated that the Advanced Gas-cooled Reactor (AGR) TRISO fuel design that would fulfill the minimum power (3-6 MWe), temperature (1200 K) and life-time ($\geq$3-yr) requirements resulted in unacceptably large reactor sizes (alpha) relative to the other fuel options. AGR TRISO is the primary form of TRISO that has been researched by DOE and approved by the NRC. The AGR TRISO particles are comprised of a 425-micron diameter UCO fuel kernel coated successively by layers of ~100-micron thick graphite buffer, ~35-micron thick pyrolytic-carbon, ~35-micron SiC, and ~40-micron pyrolytic-carbon layers. Particles are then mixed in a graphite matrix and compressed into cylindrical compacts. Two primary limitations of AGR-TRISO fuel pellets are: (1) the low uranium density, which resulted in the larger reactor core estimate, and (2) the low effective thermal conductivity of the pellet, resulting in the SiC coating reaching its temperature limit at higher power densities. Excessive thermalization and a large inventory of in-core burnable poisons were required to achieve and maintain criticality. These findings resulted in an evaluation of the AGR TRISO particle reactor design as “not meet requirements.”

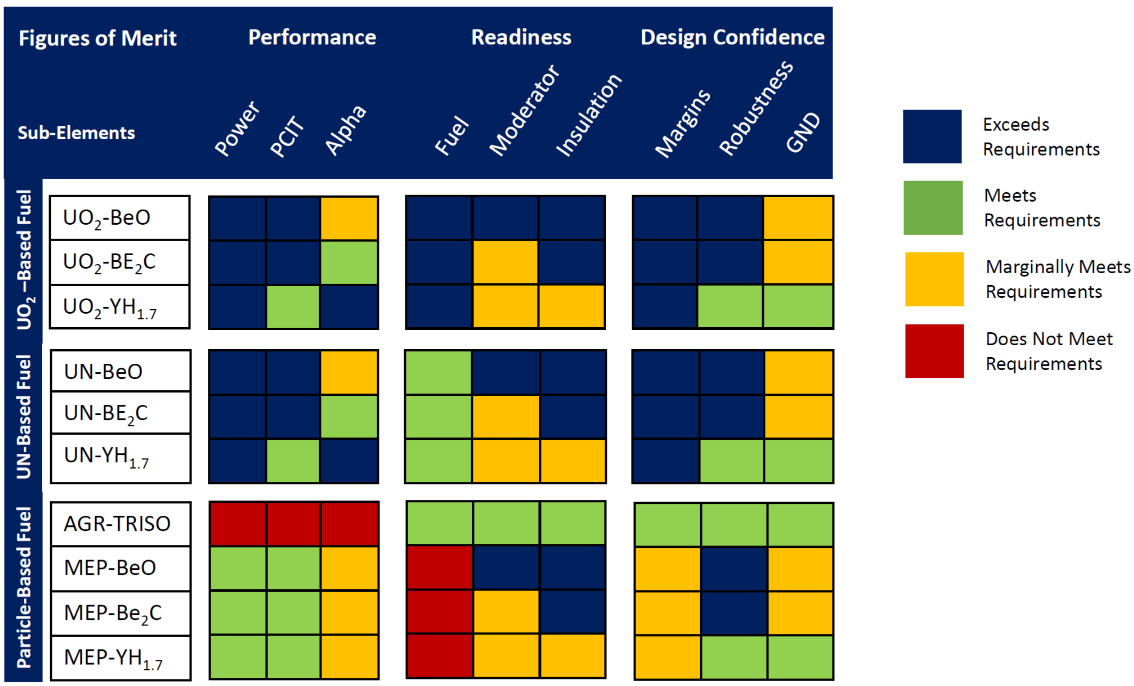

Table 1. Preliminary results of the ongoing assessment of design alternatives for a NEP nuclear reactor design. Baseline assumptions are: reactor power of 3-6 MWe; outlet temperature of 1200 K; and life time of > 3 full power equivalent years. Fuel and moderator choices were based on industry input as part of NEP TIMs.

Given recent industry interest in higher density particle fuels, micro-engineered particle (MEP) based fuels other than the AGR TRISO fuel were assessed in combination with three moderators. The SNP project is researching a form of MEP to fuel NTP reactors. It was determined that MEP fuel with a uranium density approaching 3 g/cc was required to reduce the overall mass of the reactor to the range where it was competitive with the UO_2 and UN fuels. It should be noted that MEP fuels with such large uranium packing density are currently not at the TRL/AD_2 levels required for confidence in development because they 1) have not been made in large quantities to assure quality control and 2) have not undergone neutron irradiation testing. Traditionally, higher smeared densities are achieved through both the use of larger uranium nitride fuel kernels in lieu of UCO and compression of the fuel composite particle-to-matrix volume ratio of 55%. Both these modifications fundamentally alter physical and burnup characteristics of the fuel. As such, nuclear qualification data from AGR TRISO cannot easily be scaled to understand the properties of this generic MEP based fuel. As a result, fuel readiness was assessed as “not meeting requirements..

It is recognized that there are plans in place to execute focused testing as these higher uranium density fuels are developed, and fuel readiness should be revisited if these tests are successful and as more irradiation data become available. These fuels analyzed with all three moderator combinations examined in this paper. MEP fueled heat-pipe reactors may possibly be engineered for early flight, but they require a higher volume fraction of moderator material to sustain criticality over the reactor lifetime and may experience large reactivity swing during its operation. They will have very low performance margins, which may make it difficult to accommodate design changes, such as the use of higher volumes of parasitic structural elements.

Among the moderators, $YH_{1.8}$ always resulted in the lowest weight despite the fact that those moderators needed cooling and thick thermal insulation (in all the designs, moderator 'segments' were designed with a dedicated heat pipe and were wrapped in a layer of thick insulation followed by a mm-thick cladding). As expected, use of the Be_2C moderator resulted in slightly more compact geometries compared to BeO. Be_2C and YH moderators require additional development, and due to temperature limitations owing to a loss of hydrogen at high temperatures, YH also requires additional R&D into high performance insulating materials. YH and Be_2C were therefore assigned the 'marginally meets requirements' assessment.

The final figure of merit relates to design confidence. For the alternatives considered, the only distinction is that BeO and Be_2C resulted in more robust designs in that material thermal limits were not exceeded in any of the postulated accident scenarios including prolonged loss of cooling. By contrast, YH is susceptible to hydrogen loss during prolonged loss of cooling – which could be overcome through the use of heatpipes coupled to passive radiative elements to cool the moderator material. Another difference is that Be-based moderator compounds necessitate use of Be during ground nuclear demonstration. For the case of a YH moderator, a reflector composed of MgO could be used during ground testing to avoid hazards associated with Be handling without loss of accuracy. In the case of BeO/Be_2C-moderated reactors, several tons of Be-compounds might become irradiated during ground testing

SUMMARY AND CONCLUSIONS

This paper uses an example case to present a tailorable analytical hierarchy process that can be used to narrow the technology candidates for SNP NEP design efforts and investment planning. In the example, a heatpipe-based NEP reactor design is studied and assessments of several fuel-moderator combinations are ranked relative to each other based upon a series of FOMs that permit the evaluation of the design based on how well it fulfills different selection factors. The AHP technique was used to delineate relative advantages and challenges associated with each proposed fuel-moderator combination. While preliminary rankings was performed to demonstrate the AHP technique, no down-select decisions have been made at this time. The use of the AHP process on fuel and moderator combinations serves as an illustration of how it will be used for future reactor evaluations and for other elements and subsystems that will see investments under the SNP NEP technology maturation plan.

ACKNOWLEDGMENTS

The authors wish to thank the many individuals and organizations that participated in the technical interchange meetings held over the past 6 months for each of the NEP critical technology elements. The authors would like to acknowledge modeling and simulation assistance provided by Mikaela Blood and Alexis Maldonado of Los Alamos National Laboratory. The SNP project managed by Dayna Ise funded this effort.

REFERENCES

1. Voss, S.S., **SNAP Reactor Overview**, Final Report, AFWL-TN-84-14 (Aug. 1989).
2. Wallner, L.E. and Czika, J. Jr., **Arc-jet Thrustor for Space Propulsion**, NASA TN D-2868 (June 1965).
3. Brophy, J.R. and Barnett, J.W., **Benefits of Electric Propulsion for the Space Exploration Initiative**, AIAA-90-2766 (July 1990).
4. Hogan, T., **Mars Wars: The Rise and Fall of the Space Exploration Initiative**, NASA SP-2007-4410 (May 2007).
5. Mondt, J.F., Truscello, V.C., and Marriott, A.T., **SP-100 Power Program**, *AIP Conference Proceedings* 301, 143 (July 1994).
6. Truscello, T.C., **Commercial Technologies from the SP-100 Program**, *AIP Conference Proceedings* 324, 633 (1995).
7. Ashcroft, J. and Eshelman, C., **Summary of NR Program Prometheus Efforts**, LM-05K188 (Feb. 1988).
8. **Prometheus Project Final Report**, NASA Jet Propulsion Laboratory report number 982-R120461, California Institute of Technology (Oct. 2005).
9. A Consensus Study Report of the National Academies of Sciences, Engineering and Medicine, **Space Nuclear Power for Human Mars Exploitation** (Feb. 2021).
10. NASA Engineering and Safety Center Technical Assessment Report, **Independent Assessment of the Technical Maturity of Nuclear Electric Propulsion (NEP) and Nuclear Thermal Propulsion (NTP) Systems**, NESC-RP-20-01509 (June 2020).
11. SNP Technology Interchange Meetings (TIM) – NASA Technology Memorandum in progress, notes for individual TIM's available through NASA's SNP project.
12. Duchek, M.E. *et al*., **Hybrid NEP-Chemical Vehicle and Propulsion Technology Study for Crewed Mars Mission**, presented at the 68th JANNAF Propulsion Meeting (June 2021).
13. Saaty, T.L., **How to make a decision: The Analytic Hierarchy Process**, *European Journal of Operational Research*, Volume 48, Issue 1 (Sept. 1990).
14. Wilhite, A. *et al*., **Estimating the Risk of Technology Development**, **Engineering Management Journal**, Volume 18, Number 3 (Apr. 2015).
15. Rao, D.V., Polzin, K, Houts, M., and Rodriguez, M., **Fuel and Moderator Development Strategy for NEP Systems**, JANNAF Joint Propulsion Meeting, Spacecraft Propulsion Subcommittee (June 2021).
16. Office of Nuclear Energy, **TRISO Particles: The Most Robust Nuclear Fuel on Earth**, https://www.energy.gov/ne/articles/triso-particles-most-robust-nuclear-fuel-earth (July 2019).
17. Powers, J.J., Lee, W.J., Venneri, F., Snead, L.L., Jo, C.K., Hwang, D.H., Chun, J.H., Kim, Y.M., Terrani, K.A., **Fully Ceramic Microencapsulated (FCM) Replacement Fuel for LWRs**, Oak Ridge National Laboratory (2013).
18. Rao, D.V., Polzin, K, Houts, M., and Rodriguez, M., **Affordable Development Strategy for NEP Nuclear Systems**, (*presentation only*) JANNAF Joint Propulsion Meeting, Spacecraft Propulsion Subcommittee (June 2021).
19. Merrigan, M.A, E.S. Keddy and J.T. Sena, **Transient Performance Investigation of a Space Power System Heat Pipe**, *AIAA/ASME 4th Joint Thermophysics and Heat Transfer Conference*, AIAA-86-1273 (1986)
20. Martin, J.J. and Reid, R.S., **Life Test Approach for Refractor Metal/Sodium Heat Pipes**, *AIP Conference Proceedings* 813, 108 (2006).
21. Reid, R.S., Sena, J.T., Merrigan, M.A., Elder, M.G., Martinez, A.L., **Heat Pipe Development for Advanced Energy Transport Concepts: Phase II Final Report**, LA-13549, report to Knolls Atomic Power Laboratory, Los Alamos National Laboratory (1998).
22. Woloshun, K., Merrigan, M.A., Sena, J. T., and Crtichley, E., **High Temperature Heat Pipe Experiments in Low Earth Orbit**, LA-UR-93-0197, *National Heat Transfer Conference*, Villanova, PA (1993).