Center for Space Nuclear Research (CSNR) NTP Design Team Report

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To: Omar Mireles, NASA MSFC
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　　　　Michael Eades – CSNR Visiting Technologist/The Ohio State University/NSTRF Fellow
　　　　Wesley Deason – CSNR Research Scientist
Date: 10/1/2015
Subject: This report is a summary of work done by the CSNR Nuclear Thermal Rocket (NTR) Design Team for NASA MSFC on the SCCTE project.

Background

The CSNR NTR Design Team has developed a history of collaboration with NASA Marshall Spaceflight Center (MSFC). First assembled in the summer of 2013 to explore the concept of the Low-Enriched Uranium Nuclear Thermal Rocket (LEU-NTR), the team continued research to find solutions with guiding Nuclear Thermal Propulsion (NTP) technological development and provide design and analysis support for the Space Capable Cryogenic Thermal Engine (SCCTE) Project. The work presented in this report provides a background on the cermet LEU-NTR, summarizes work complete to explore a new design space for NTRs, and describes the SCCTE core that was developed.

The Space Capable Cryogenic Engine (SCCTE) is an LEU W-UO$_2$ cermet fuel, ZrH$_{1.8}$ moderated NTP concept undertaken by NASA Marshall Space Flight Center (MSFC) and Center for Space Nuclear Research (CSNR). SCCTE is part of a larger effort at NASA to investigate NTP for a human Mars mission in the mid-2030s.

At the start of the SCCTE project, a LEU W-UO$_2$ cermet fuel NTP system was a new concept with little relevant information existing in literature. SCCTE deviates significantly from 1960's Rover/NERVA experience and previous Highly-Enriched Uranium (HEU) NTP cermet design studies. The driving factors that force SCCTE to be different from pervious NTP experience are the use of LEU in SCCTE and the high performance goals put on SCCTE. LEU was chosen for the SCCTE reactor because it has considerably less regulatory burden than HEU and the use of LEU in a NTP system could lead to a more affordable development.

In order to study LEU W-UO$_2$ cermet fuel NTP systems, CSNR has developed its Space Propulsion Optimization Code (SPOC). SPOC is a multifunctional combined neutronic and thermal hydraulic code for NTP analysis.

This report is organized to give a reader a brief, but detailed overview of the key accomplishments made in the SCCTE design effort. First a technical overview of the SCCTE reactor is presented which shows key figures and performance parameters, then cermet LEU-NTR design considerations are presented, followed by a detailed description of the trade studies completed to fully understand the LEU-NTR design space which led to the SCCTE design. More detailed design studies are then shown which include the optimization of the radial and axial power profile, sensitivity studies on thermal aspects of a flow channel, which is followed by a section on future work. Lastly, attached to the end of this report is a description of the thermal

methods used within SPOC and a summarized form of the SCCTE reactor as a concept "baseball card". This smaller report was used to communicate with NASA MSFC and provide internal record of code verification.

Technical overview of the SCCTE Reactor

The SCCTE reactor was designed with several goals in mind. These goals were motivated by trajectory analysis, experimental capabilities, political concerns, and engineering intuition. A list of many of the externally decided details is shown below.

- An outlet temperature sufficient to meet 900 s Isp
- Thrust of 35000 lbf
- Low Enriched Uranium (<20 w% $^{235}U/U$, NRC Category II)
- Tungsten-based CERMET fuel
- 1000 psi (6.9 MPa) nuclear core inlet pressure
- Traditional NTP stacked hexagon geometry
- Sufficient reactivity shut down margin
- Sufficient excess reactivity margin to stay critical as the design and analysis refines

The SCCTE reactor designed meets each one of these goals, except for 900s Isp, where it attains 894.2 seconds. This was deemed sufficiently close to the design goal. The Isp can be changed by trading some thrust for Isp, or by allowing for an increased maximum fuel temperature, which is currently set to a conservative value of 2850 K.

As mentioned in the introduction, the primary difference between the SCCTE reactor and previous NTP designs is the use of LEU fuel. To allow the use of LEU fuel in the SCCTE reactor, $ZrH_{1.8}$ is used for neutron moderation. Additionally, the tungsten fuel matrix material has been enriched to 98 atom % ^{184}W (versus natural tungsten used in other designs), reducing the parasitic absorption of thermal neutrons in the matrix material. The fuel stabilizer has also been changed to 6 mol % ThO_2 (versus Gd_2O_3 used in other designs), reducing the parasitic absorption of thermal neutrons in the fuel material. These details, as well as many others, are listed within Table 1 and depicted in Figure 1. Figure 1 is composed of several images that help visually describe the analysis done on the SCTTE reactor. Figure 1A is a lateral cross sectional view of SCCTE Monte Carlo N-Particle (MCNP) model produced by SPOC. The different colors in the figure represent different computational "materials" defined by SPOC to allow MCNP to determine desired neutronic characteristics. Figure 1B is an axial cross sectional view of the same SCCTE model, with an excerpt showing the detailed geometry features of the reactor core. This amount of detail is necessary to allow the computation of specific energy deposition values and complicated neutron streaming effects. Figure 1C shows the radial arrangement of fuel enrichments, which allow for flatter radial power deposition values (discussed in more detail toward the end of this report). Lastly, Figures 1D and 1E show relative computed core power deposition and in-element power deposition values, also discussed in more detail in later sections of this report.

Table 2 shows the enrichment pattern in the core. Combining the fuel element radial enrichment zones in Fig. 1C with this table allows for recreation of the SCCTE fuel lattice; all axial zones are equally sized. Figure 2 shows the materials dimensions of the above core sections, and Fig. 3 shows the same for the active portion of the core radial reflector. The internal shielding shown has not been optimized with respect to engine configuration, external shield, and dose

requirement; as such, it is only a representative shield. The H_2 flow path dimensions for adequate core peripheral cooling and reduction of pressure drop have also not been analyzed, meaning the dimensions of plenums are not representative of the final design. These aspects are not very important neutronically so the lack of detail is acceptable for core design.

The SCCTE reactor must be able to operate for a total time on the order of 1-3 hours with 4 power up and down transients separated by short (hours) and long (months) time. Even with large power densities, very little fuel burnup is expected. During design iteration, SCCTE-like cores were found to have acceptable excess reactivity to overcome fission product buildup for rocket burn times provided by trajectory analysis. However, large reactivity swings were observed during the second of the two burns that would require control drum movement during power operations that may lead to transient core operations. The CSNR is involved in a separate project [6] to mitigate the large reactivity swings during a burn such that the SCCTE reactor might operate on passive nuclear controls. Full burnup analysis of the current SCCTE iteration has not been performed but no issues in terms of total excess reactivity margins are expected.
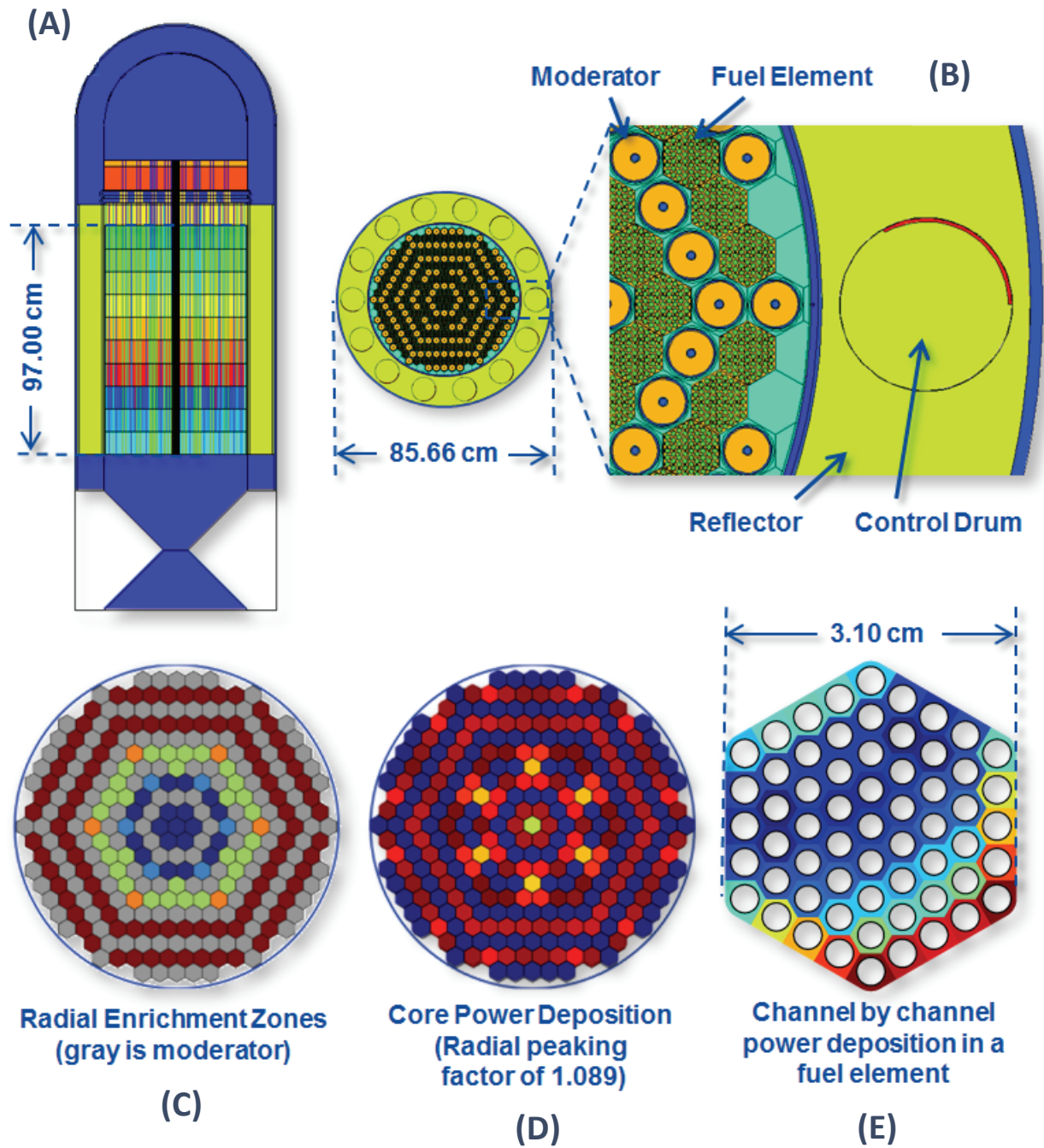
(A)

97.00 cm

85.66 cm

Moderator   Fuel Element   **(B)**

Reflector   Control Drum

Radial Enrichment Zones
(gray is moderator)

(C)

Core Power Deposition
(Radial peaking
factor of 1.089)

(D)

3.10 cm

Channel by channel
power deposition in a
fuel element

(E)

FIGURE 1: Key views of the SCCTE reactor core and analysis results.

TABLE 1: Key specifications for the SCCTE reactor.

Reactor System Mass

Fuel Mass (151 Elements) (kg)	1029.8
Tie Tubes (150 Elements) (kg)	562.5
Radial Reflector + Control Drums (kg)	618.6
Axial Reflector (kg)	61.5
Barrel+Vessel+Other Core Structure (kg)	285.3
Total Reactor Mass (Excluding Shield) (kg)	2557.6
Estimated Non-Reactor Masses (kg) [7]	2086.3
Total Estimated Engine Mass (kg)	4643.8

Key Performance Parameters

Nominal Isp (150:1 Nozzle)	894.2
Nominal Thrust (kN)	155.7 (35k lbsf)
Reactor Thrust to Weight	6.21
Engine Thrust to Weight	3.42
Reactor Power (MW)	765.6
Fuel Temperature Max (K)	2850.0

Engine System Interface Information

Interface Point	Flow Rate (kg/s)	Pressure (MPa)	Temp. (K)
Core inlet	17.75	6.93	291
Core outlet	17.75	4.72	2695.2

Fuel Details

Fuel composition	W-UO_2-ThO_2
Volume loading of Oxide (% vol.)	60.0
ThO_2 in the Oxide (% mol.)	6.0
Enrichment of ^{184}W (% atom)	98.0
Enrichment of ^{235}U (% atom)	13.13375 to 19.75
Total Enriched W (kg)	376.0
Total ^{235}U (kg)	45.9
Percent Theoretical Density (% TD)	97.0

TABLE 2: Enrichment Pattern for SCCTE

Axial \ Radial Zone	Inner				Outer
Top	0.1975	0.187625	0.1975	0.187625	0.1975
	0.1975	0.187625	0.1975	0.187625	0.1975
	0.1975	0.187625	0.1975	0.187625	0.1975
	0.1975	0.187625	0.1975	0.187625	0.1975
Unit: Atom % ^{235}U	0.1975	0.187625	0.1975	0.187625	0.1975
	0.1975	0.187625	0.1975	0.187625	0.1975
	0.158	0.1501	0.158	0.1501	0.158
	0.13825	0.1313375	0.13825	0.1313375	0.13825
	0.13825	0.1313375	0.13825	0.1313375	0.13825
Bottom	0.13825	0.1313375	0.13825	0.1313375	0.13825

m#	Material
1	Hydrogen gas
3	BeO
4	W-25w%Re
5	^{6}LiH
13	Al-2024

Units: cm

Measured Relative
to Axial Reflector
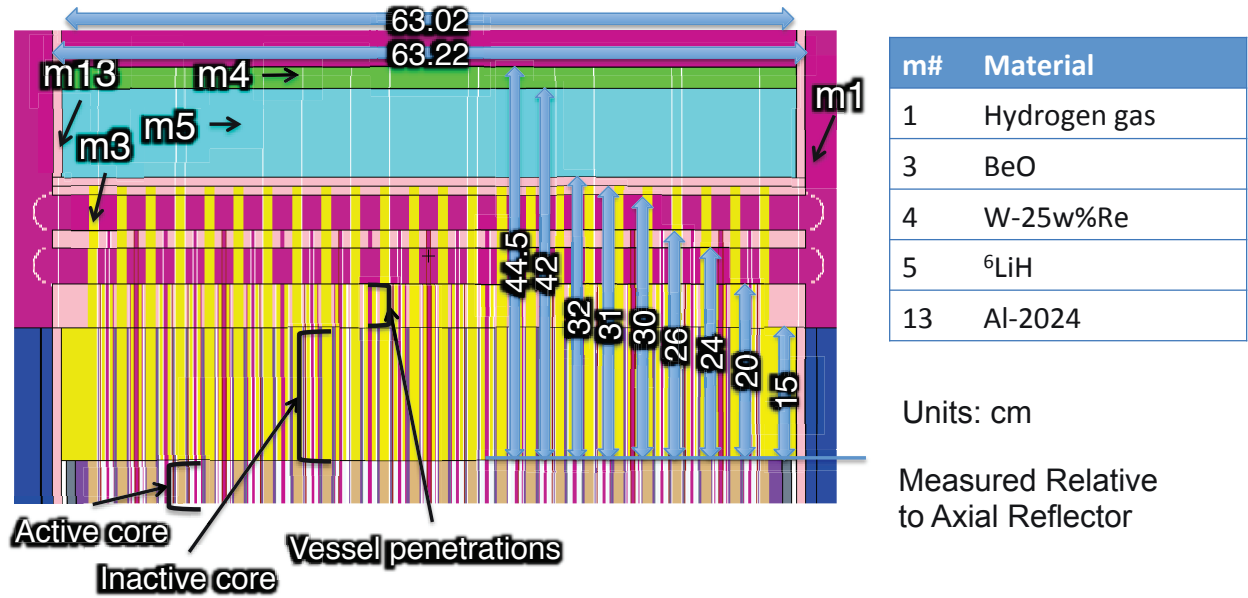
FIGURE 2: Above Core Dimensions and Materials

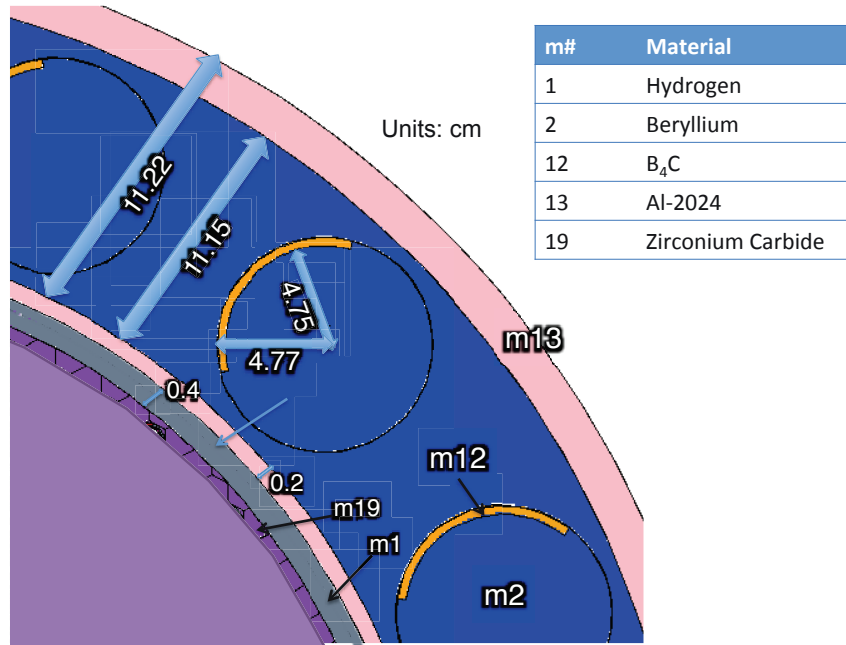| m#	Material
1	Hydrogen
2	Beryllium
12	B_4C
13	Al-2024
19	Zirconium Carbide

FIGURE 3: Radial Reflector Region Dimensions

Low Enriched Uranium Nuclear Thermal Rocket Design Considerations

The SCCTE reactor is different than most heritage NTP concepts in that it uses LEU (<20 wt% ^{235}U/U) as opposed to heritage NTP concepts which use Highly Enriched Uranium (HEU) (>20 wt% ^{235}U/U). In an effort to develop affordable NTP technology, a recent effort has emerged to investigate NTP technology that uses LEU. Research involving multi-kilogram amounts of HEU requires significant physical security, considerable regulatory oversight, and attracts political scrutiny because of the proliferation concerns related to HEU. A reason for the LEU push is that an LEU NTR will have fewer proliferation related concerns and less regulatory burden, and therefore can be significantly more affordable to develop and will have less scheduling risk.

HEU fuel was once common in research reactors. From a purely technical standpoint, HEU is a logical choice for research reactors because it allows for more compact and therefore higher neutron flux cores per MW than LEU. Proliferation concerns associated with the use of HEU have led to a steep decline in the number of HEU reactors in the world. The Reduced Enrichment for Research and Test Reactors (RERTR) and Global Threat Reduction Initiative (GTRI) programs have thus far been effective in converting the majority of HEU-fueled reactors to LEU.

An example of the success of the HEU minimization efforts can be seen with the cancelation of the Advanced Neutron Source (ANS) in 1996. The ANS was to be a next generation research reactor that would produce the world's largest continuous neutron flux. The project was canceled partially because the ANS design used HEU fuel, which was deemed to be in conflict with the United States' non-proliferation policy [1]. Testimony in Congress, at the time, stated that building a HEU reactor would damage the US's position to credibly urge other nations to not use HEU [2].

With the assumption that an HEU NTP engine would not be built in the current political climate, a sound choice is to design an LEU NTP engine. This requires large deviations from heritage designs, but because much engineering expertise from the heritage designs is not available today, the large design changes for LEU NTP can be acceptable as technical expertise must be built for any NTP that will fly, including heritage designs.

Transitioning from HEU to LEU in an NTP reactor requires study of the neutron economy. The utilization and loss of neutrons is of largest concern. In HEU systems, leakage is large and non-fission absorption is low. When transitioning to LEU fuel in the same geometry as HEU fuel, neutron leakage will decrease and non-fission absorption will increase very quickly. To mitigate this, neutron moderator material such as that used in heritage designs, ZrH, must be added to the system. Moderated neutrons with LEU fuel will increase fission-absorption events and reduce leakage.

Heritage designs often used the tie-tube as a structural material. The function of the tie-tubes in LEU systems is more for moderating rather than structural so a separate structural support mechanism must be used in these designs.

To further discuss LEU design characteristics, a fuel form must be chosen. For the SCCTE project, tungsten-cermet fuel with UO_2 in a tungsten matrix with thoria stabilizer was used. This

was chosen to be similar to heritage W-Cermet fuels with the change of GdO_3 stabilizer to ThO_2 and the removal of Re from the fuel cladding, both to reduce parasitic neutron absorption. The main driver for this fuel choice was that new manufacturing processes such as spark-plasma sintering can produce this fuel with close to theoretical material properties allowing for a material with high fuel density, good thermal conductivity, and good heat capacity. This was not found to be possible in the 60s during initial HEU cermet fuel manufacturing due to design deficiencies in the manufacturing process.

The use of LEU fuel and moderation shifts the neutron spectrum to be thermal rather than fast, which unfortunately increases absorption for many materials that were not large absorbers in the fast spectrum. Namely, the tungsten in the fuel, in natural form will not allow for a LEU reactor to be built. By enriching the W in the ^{184}W isotope to 98 atom % (90% works too), critical cermet LEU cores can be designed.

To further increase moderation, fuel element to moderator element ratios can be varied from 3 to 1 all the way to 1 to 3. It will be seen later that 1 to 1 ratios are a good choice for the 35 klbs thrust engine being designed in the SCCTE effort. Thermal hydraulic characteristics cause the neutronic design to be constrained, hence 1 to 3 fuel to moderator ratio does not beat out 1 to 1 due to the larger surface area of heat transfer per unit area in the 1 to 1 core.

To maximize fuel packing, the fuel elements should be hexagonally shaped with many coolant channels going through each element. Larger fuel elements with smaller channel size and thin fuel webbing (fuel between coolant channels) tend to allow for decreased neutron self-shielding, increasing the neutron utilization factor. Furthermore, larger fuel elements allow for larger moderator elements, which in turn allows for more moderator per unit area by increasing the moderator outer radius in the moderator tube.

The moderator tube was initialize created base on the Small Nuclear Rocket Engine tie-tube design. The tie-tube consisted of moderator and structural material. The moderator tube was created by increasing the total size of the tube to allow for more moderator, by changing Inconel components to Zircalloy to reduce neutron absorption, and substituting the outer graphite sleeve with more ZrC. The geometry of a typical moderator tube is shown in Fig. 4.

The departures from heritage NTP design discussed above are not entirely exhaustive of the changes that will be required to full design and implement an LEU-based core design. Using modern design practices and the ability to conduct multiple computer-based design iterations will significantly change the look and operation of an LEU-based core from heritage NTR designs.
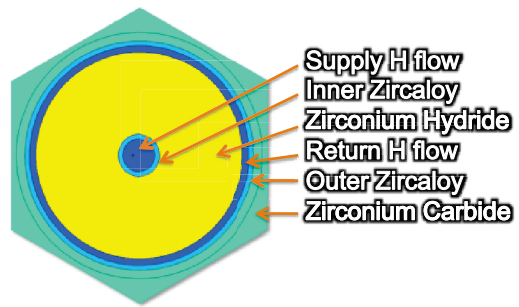
FIGURE 4: Moderator Tube Cross-Section

Trade studies

When designing a traditional new nuclear reactor, there are many initial steps and design metrics that can be used to stay near optimal reactor designs. Reactor design procedures for NTP cores differ from power reactor design in enough ways where power reactor design strategies are not always relevant in NTP core design. For example, fuel economics in terms of burnup is largely ignored in NTP design and fuel layout is generally less than optimal neutronically to allow for compact systems that can be adequately cooled. Similar to power reactor design, an NTP core should be created based on desired power levels, which translates to thrust levels for NTP design. Smaller cores will have different optimum layouts than larger cores. In order to properly understand how different variables change design metrics, a large-scale study of many NTP reactors was created.

In the study of modern nuclear thermal rocket design, the utilization of available computing resources is particularly advantageous for reducing development costs and mission costs. With currently available tools, it is possible to create and analyze thousands of complex nuclear reactor designs to a sufficient fidelity in about a week's worth of time. However, this task is not a simple one – complexities may arise in the following ways: In reducing the large number of design variables available to the engine design to a level which can be handled by the large but finite array of available computing resources; In understating the reasons for complex behaviors in engine performance; In communicating the results using the proper figures of merit which allow program managers to direct technology development. For the SCCTE reactor in particular, it was important that the design methods used gave the confidence required of a NASA-led nuclear thermal rocket design project. With good analysis, the results could be cumulative, and used to influence NTP designs many years down the road.

Methods

The primary objective of this particular study was to improve upon current methods used to design and develop NTRs. Previous methodology for modern NTR design relies heavily upon heritage designs and manual iteration. While effective at producing a working design, the manual iteration associated with design optimization is labor intensive and is less effective if the designer does not have a full understanding of the design space available. To address these concerns, a large scale design study was conducted which varied many of the important design parameters and give a full description of the design space available to the designer.

SPOC

To conduct the large scope design study discussed in the present work, the Space Propulsion Optimization Code (SPOC) was used. SPOC was developed and is maintained by the CSNR to allow the rapid design and analysis of NTR reactor cores [8]. In summary, SPOC receives input from the user needed for all aspects of the nuclear and thermal design calculations. It then generates an MCNP [3] or Serpent [4] (two similar nuclear analysis programs) input file, which can be used to determine the design's nuclear performance and operational characteristics. It then analyzes the MCNP or Serpent output files and computes the designs thermal performance using any available information. When given a desired thrust, maximum allowable fuel centerline temperature and supply hydrogen pressure, the code can determine the core power and hydrogen

mass flow rate that the NTR will be operating at for those conditions, thus producing a "working" core design. The code can be run to complete analyses on up to thousands of designs, as was done in the following large scope design study. An example of the SPOC operational interface is depicted in Figure 5.
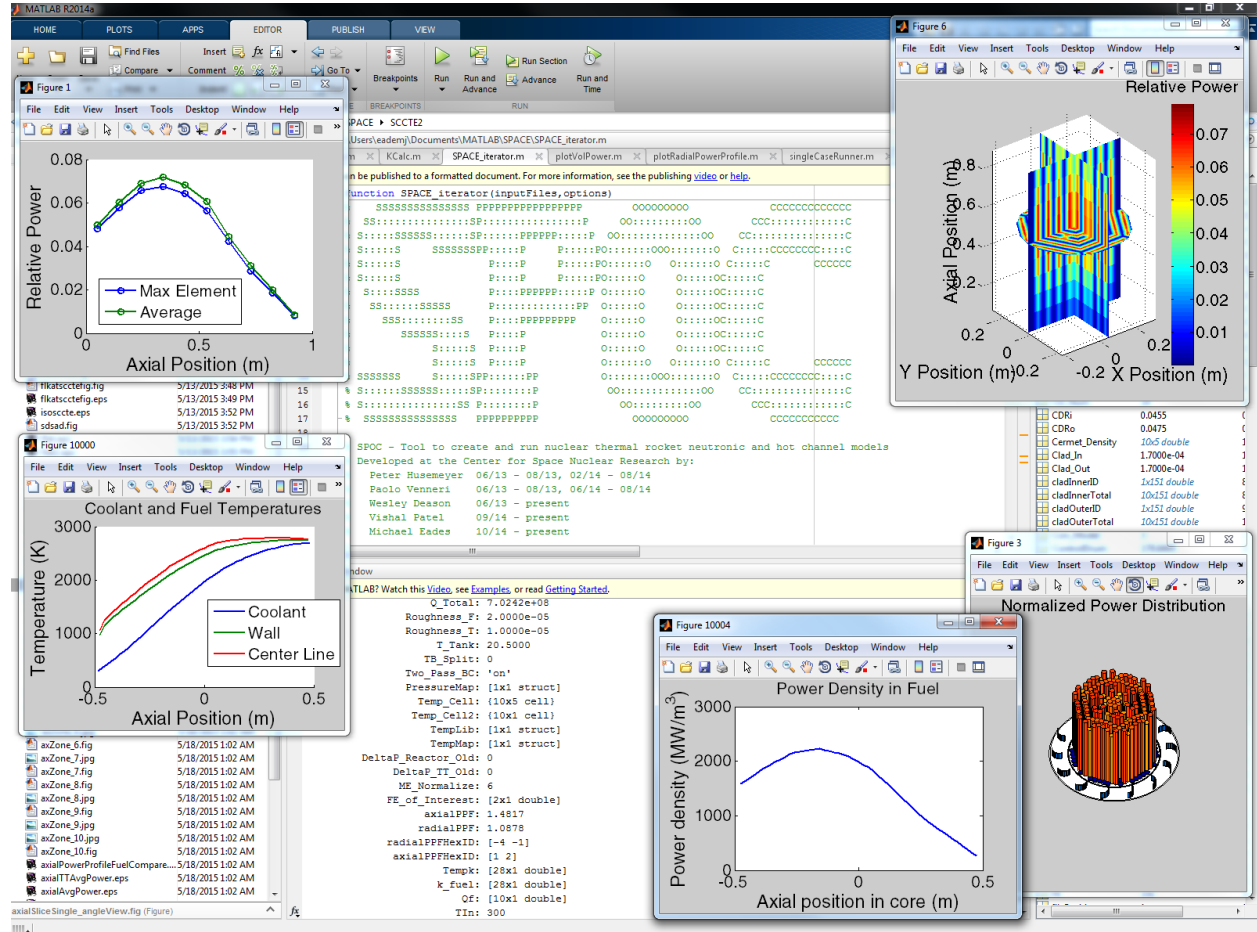

FIGURE 5: The SPOC operational interface.

Study Setup

The first step to studying an NTR design space is to determine how large of space is required. This determination governs the number of variables changed from a baseline design, how much they are varied, and at what intervals. It is also necessary for establishing driving performance parameters. In particular, the thrust required from the engine design, the fuel centerline temperature, and inlet pressures and temperatures play important roles in determining the hydrogen flow rate and reactor power for a given NTR design. From these parameters and variables, two important figures of merit, system mass and propellant mixed-mean outlet temperature, can then be calculated.

Preferable Design Dimensions to Vary

While attempting to establish the design space available for a NTR core, the number of variable dimensions to be considered is large. Unfortunately, currently available computing resources are

not infinite, meaning that preferable design dimensions must be chosen. The design dimensions varied in the present work are shown in Table 3.

TABLE 3: Varied Design Dimensions.

Dimension	Values
Fuel Channel Radius	1.0287, 1.37935, and 1.73 mm
UO$_2$ Fuel Loading	40%, 48%, 54%, 60% (% of total fuel meat composition)
Number of Coolant Channels per Fuel Element	19, 37, 61, 91, and 169 channels
Fuel Webbing Thickness	0.646 and 0.9 mm
Core Height	0.5, 0.66, 0.75, 1.0, 1.25, and 1.5 m
Core Radius	0.25, 0.30, 0.35, 0.4, and 0.5 m
Reflector Thickness	0.1, 0.15, and 0.2 m
Hexagonal Element Arrangement	3:1. 2:1, 1:1, 1:2, and 1:3 Fuel to Moderator Element Ratio

The design dimensions shown in Table 3 were assembled following lessons learned in an initial study. They represent a very broad stroke to capture as many different core combinations as possible. It was assumed that some of the design would perform very poorly (low I$_{sp}$ and high mass), though all designs chosen in the end would be "working" designs capable of producing 26,500 lbf of thrust. Initial iterations of SCCTE were at this thrust level but were later changed to 35,000 lbf for mission needs. The general results presented here are applicable to the 35,000 lbf engine design space and a similar large study was performed to create the SCCTE core.

While taking a single core design and changing only one variable at a time is one option for understanding the behavior of the system, multi-variable effects and the ability to use the design study to select a next generation design are lost. To mitigate this, the outer product of each variable was taken along with the varied values to create a large set of NTP cores to study. In other words, every combination of NTP designs possible with the given variables and values were created (54000 cores in this case). The set of variables varied is not exhaustive; any variable could in theory be varied within SPOC, even the fuel type.

Furthermore, using the current method, a design which features the most favorable combination of variables can be selected for further study without any manual iteration required. If desired, the selected design can also serve as a baseline for a finer study in which a design space around the selected design can be created to determine optimum cores near the selected design.

14

Choosing Realistic Performance Parameters and Setting Assumptions

The next step in setting up SPOC to complete the large scope design study was to choose the performance parameters that the conceptual reactors will operate under in order to provide a realistic comparison and allow for the best understanding of the NTR design space as possible. To do so, the performance parameters shown in Table 4 were chosen.

TABLE 4: Chosen Performance Parameters for Large Scale Study

Parameter	Values
Mach Number Limit in Average Channel	0.3
Target Thrust	25,000 lbf
Max Fuel Temperature	2850 K
Hot Channel Inlet Temperature	300 K
Hot Channel Inlet Pressure	10.0 MPa
k_{eff} lower limit (drums inserted halfway)	0.99

Additionally assumptions were made for the reactor materials and configurations. The baseline core design selected for the large scope design study is assumed to have the following characteristics:

- LEU (19.75% U_{235}) tungsten (98% W_{184}) cermet fuel
- $ZrH_{1.8}$ moderator
- Zircaloy moderator structure with 40% porous ZrC insulator and ZrC sleeve
- Standard hexagonally shaped fuel and moderator elements
- BeO axial reflector and Be radial reflector
- B_4C control material in crescent shape
- Aluminum pressure vessel
- A flat radial temperature peaking factor (all channels are perfectly orificed to produce the same outlet temperature
- A constant axial fuel loading and enrichment

Results

The results generated by SPOC for the large scope design study have been arranged to show the performance of each design in terms of the engine I_{sp}, engine weight, engine thrust-to-weight, and average core power density within Figures 6 to 14. It is important to note that the results only show thermal and weight effects. Any neutronic effects other than the k_{eff} lower limit of 0.99 are not discussed and will require further study for quantification. All plots are repeated with two different color codes to better visualize the dense design space.

Figures 6 to 14 show engine performance in terms of engine Isp and weight show that the maximum engine I_{sp} increases with weight. In other words, engine I_{sp} can be traded for engine weight to achieve a more optimum configuration depending on the mission it is designed for.

Figure 6 shows the effect of changing coolant channel radius from 1.0287 mm (ANL-like, similar to the small radii found in the ANL-200 design) to 1.73 mm (the coolant channel radius used in the recently fabricated CSNR 16 inch fuel segments). As can be seen, reducing the coolant channel radius has a significant effect on the NTR I_{sp} for a given system weight. This effect is shown across the board for reactor designs, which indicates a universal benefit for decreasing coolant channel radius. However, smaller coolant channels are harder to manufacture so the coolant channel size should be selected with those considerations.
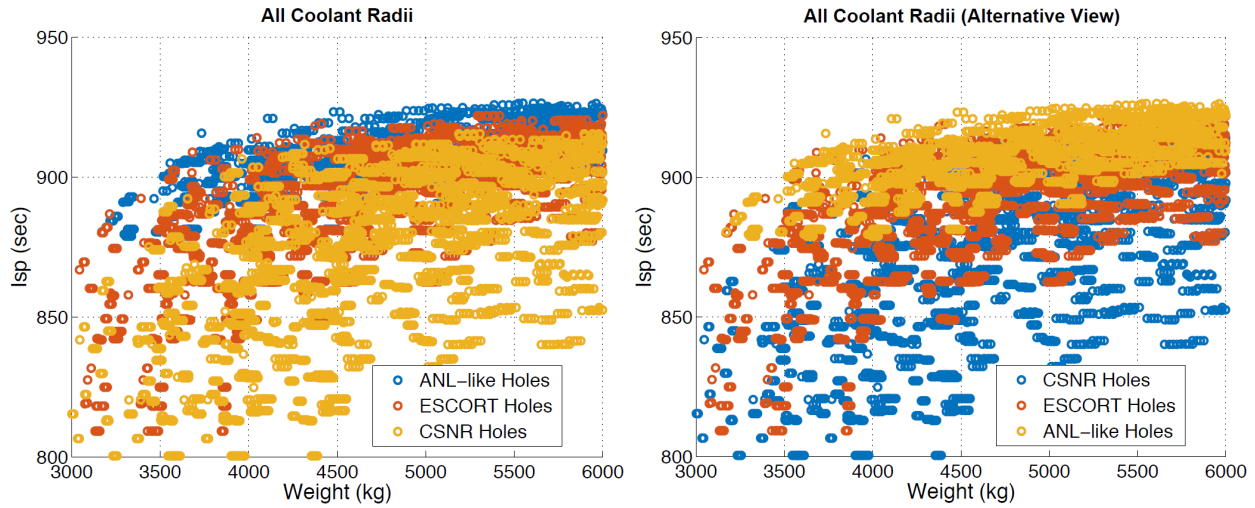

FIGURE 6: Comparison of I_{sp} vs. system weight depicting variance in coolant channel (hole) radius.

Figure 7 shows that thermal or weight performance is not affected much by fuel loading. A positive aspect of this is that fuel loading can be varied for neutronic, material, or economic reasons without worry of significantly affecting thermal performance or mass. The present work did not consider varying fuel loading throughout the core radially or axially. This may have significant effect on thermal performance and will be studied in future analyses.
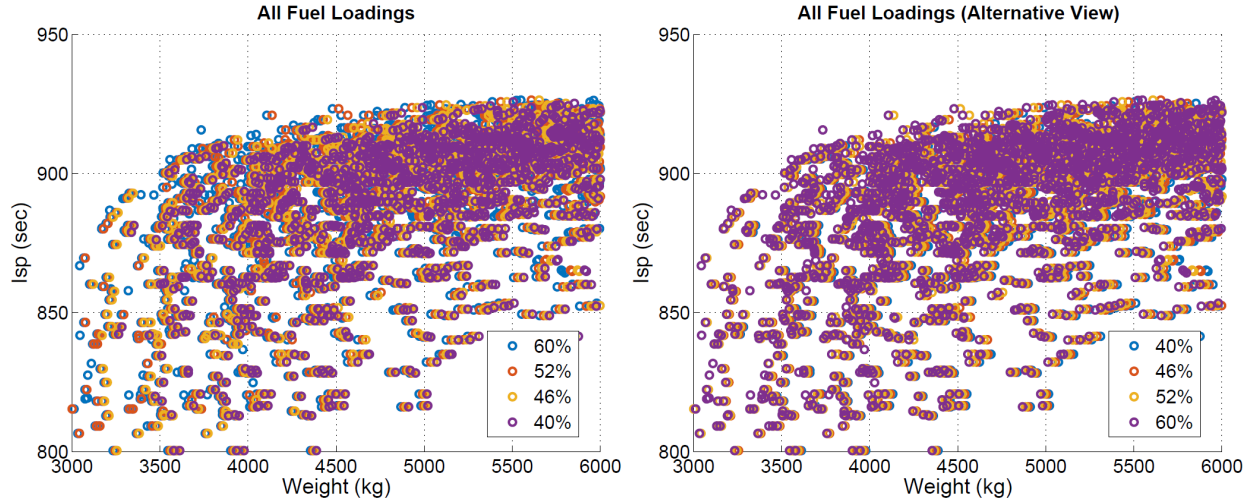

FIGURE 7: Comparison of I_{sp} vs. system weight depicting variance in fuel loading.

Figure 8 shows that, excluding several unique cases, after 61 channels (holes) diminishing returns are reached for the present core design. The increase in fuel manufacturing complexity for larger numbers of channels must be taken into account for reactor design and Figure 6 allows the judgment to be made on the trade between complexity and performance.
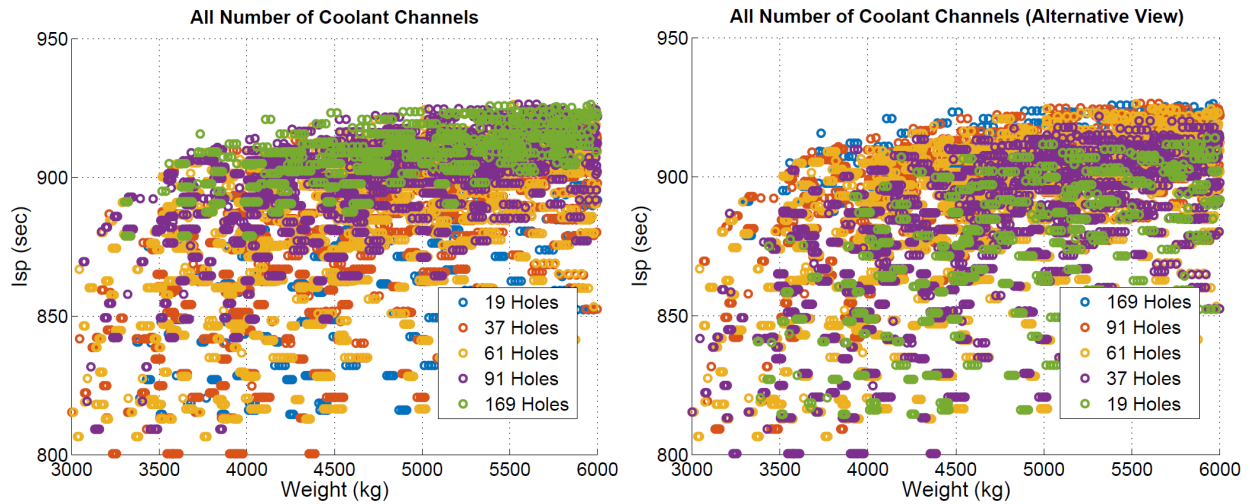

FIGURE 8: Comparison of I_{sp} vs. system weight depicting variance in the number of coolant channels (holes).

Figure 9 shows that there is a small benefit to decreasing web thickness for cores of smaller system weight. This effect becomes diminished as system mass is increased. The fuel webbing

thickness plays a large part in thermal stress determination, which was not considered in this study. Structural analysis should be used to better guide the selection of the webbing thickness.
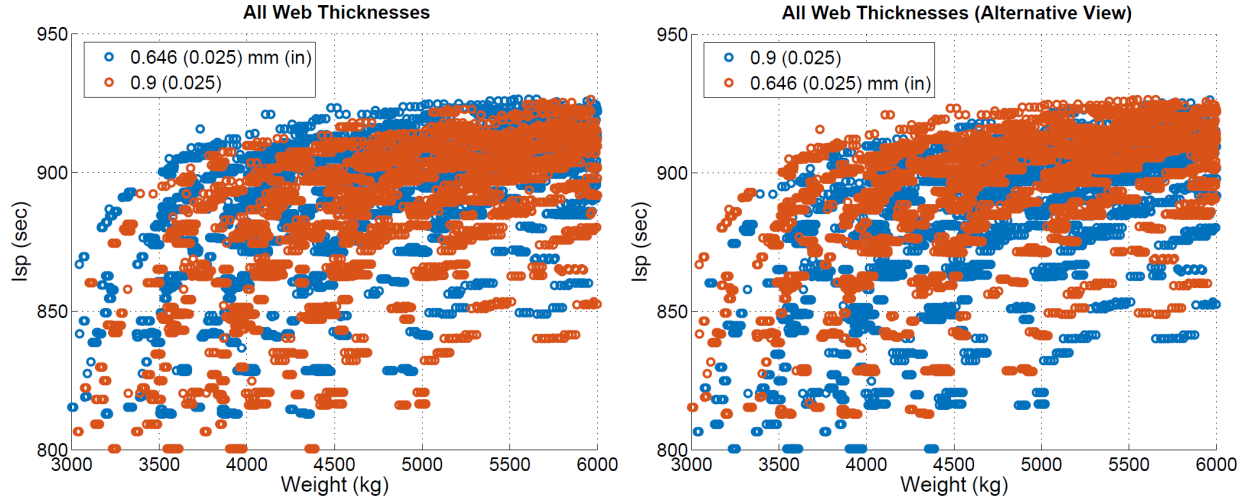

FIGURE 9: Comparison of I_{sp} vs. system weight depicting variance in fuel web thickness.

Figure 10 shows that longer cores generally have better I_{sp}, but greater system weight.
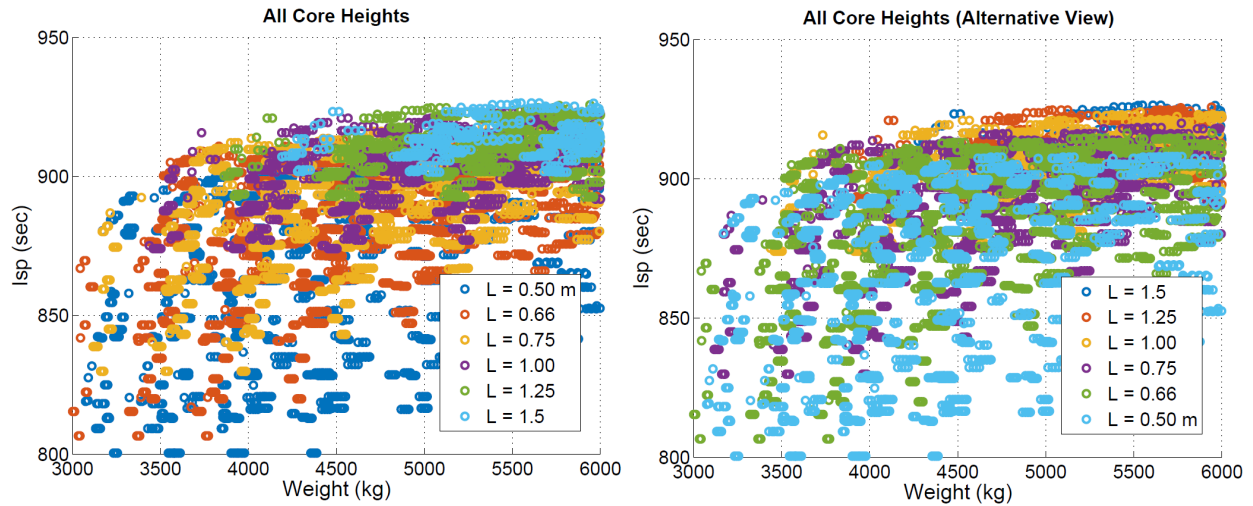

FIGURE 10: Comparison of I_{sp} vs. system weight depicting variance in core height.

Figure 11 shows that a preferred core radius exists around 0.3 m, with some outlying cases having a core radius of 0.25 m. This trend may indicate that cores with radii less than 0.25 m may need to be examined in future studies, particularly for lower thrust designs.
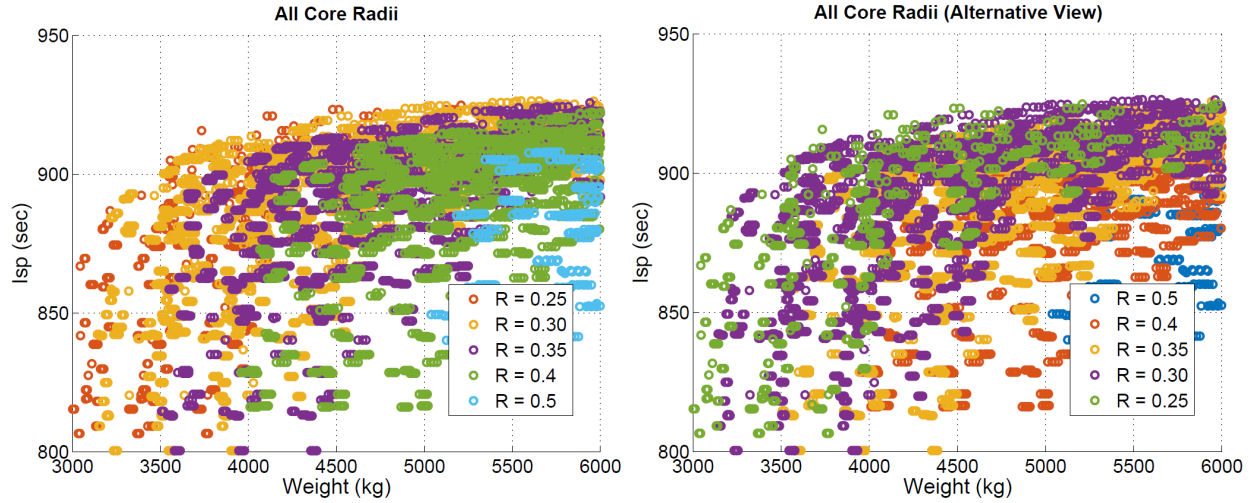

FIGURE 11: Comparison of I_{sp} vs. system weight depicting variance in core radius.

Figure 12 shows that smaller reflectors seem to be preferred across the board. It is important to note that control drum swing was not tracked in the large-scale design study. This may be a competing effect, which should be taken into account in future studies.
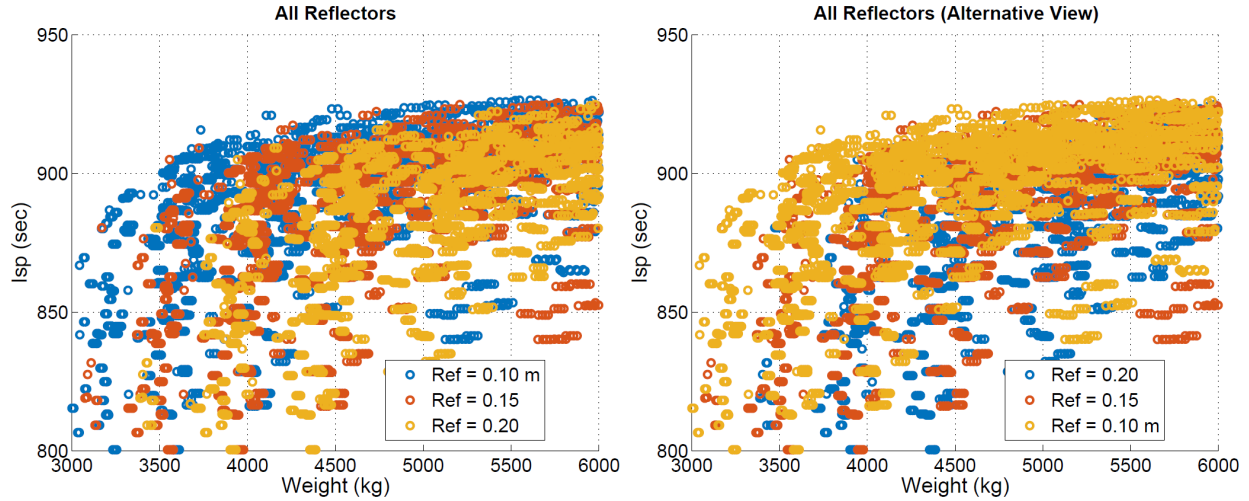
FIGURE 12: Comparison of $_{Isp}$ vs. system weight depicting variance in reflector thickness.

Figure 13 shows that no reactors with a 3:1 fuel to moderator (F:M) ratio made the k_{eff} cutoff of 0.99. There is a general trend that high engine I_{sp} NTRs with low mass have a lower F:M ratio, while a F:M ratio of 1:1 seems to universally be the preferred configuration.
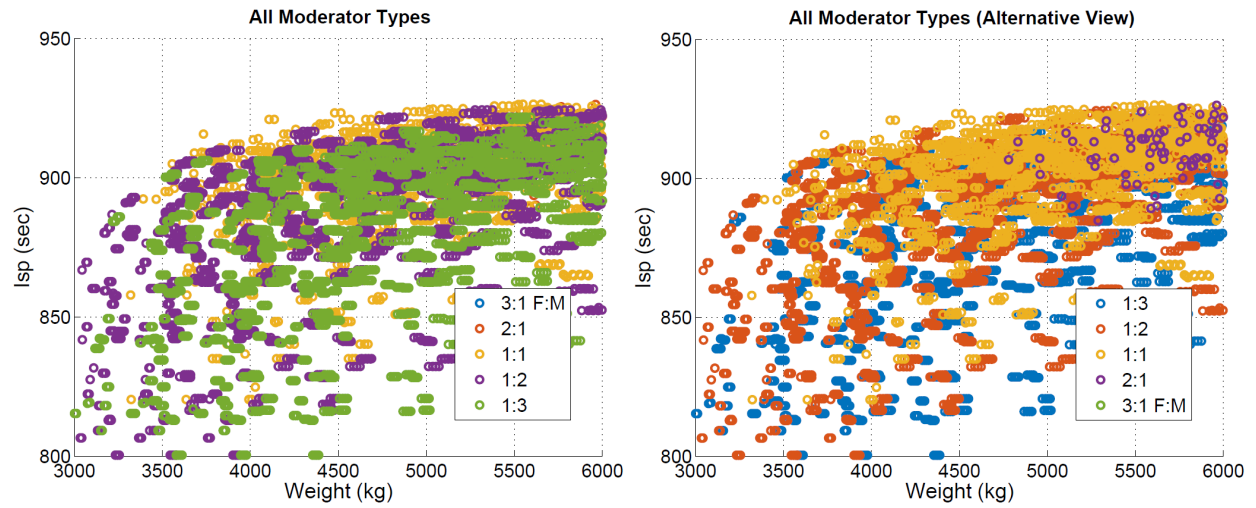

FIGURE 13: Comparison of I_{sp} vs. system weight depicting variance in hexagonal element arrangement.

Figure 14 shows that both engine thrust to weight and engine I_{sp} are correlated with average reactor power density. In general, higher engine thrust to weight means a higher average reactor power density and higher engine I_{sp} means a lower average reactor power density. While all the reactor design points shown in Figures 6-14 have a max fuel centerline temperature of 2850 K further limitations on core power density due to thermal gradients may limit the thrust-to-weight of NTR designs.
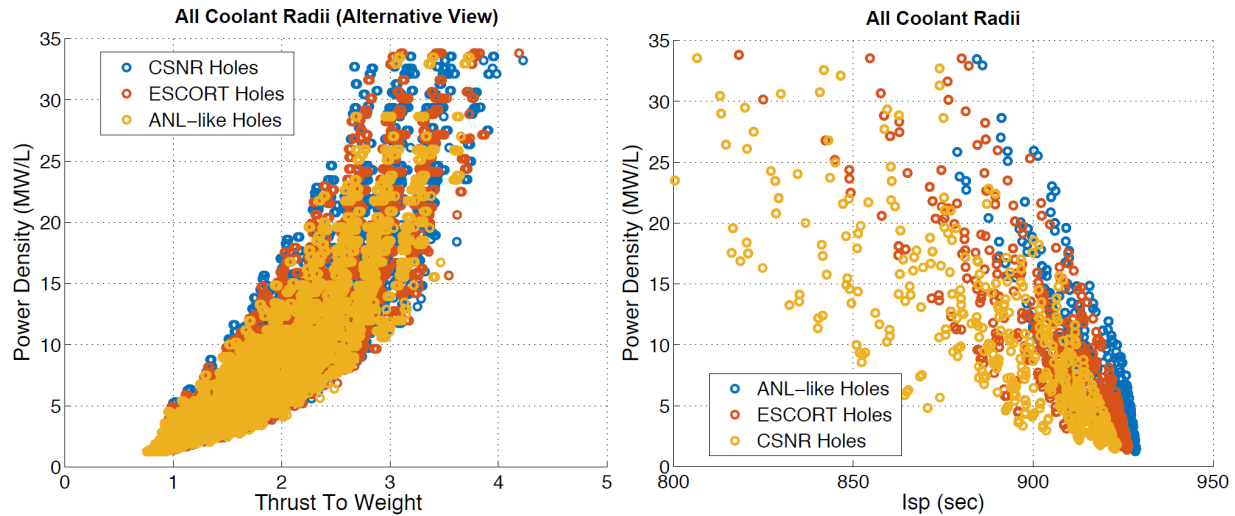

FIGURE 14: Comparison of Thrust to Weight and I_{sp} vs. Power Density depicting variance in coolant channel radius.

Radial and Axial Power Profile Optimization

In an NTP system, it is advantageous to have a radially flat power profile across the fuel elements. A flatter radial power profile is associated with less pressure drop across the core, a lower peak power density in the fuel, and possibly less ^{135}Xe worth. To flatten the radial power profile in the SCCTE reactor, variable enrichment and small change to the placement of fuel elements and tie tubes where employed. Figure 15 depicts how tie tube and fuel elements where moved and added to the core to flatten the power profile.
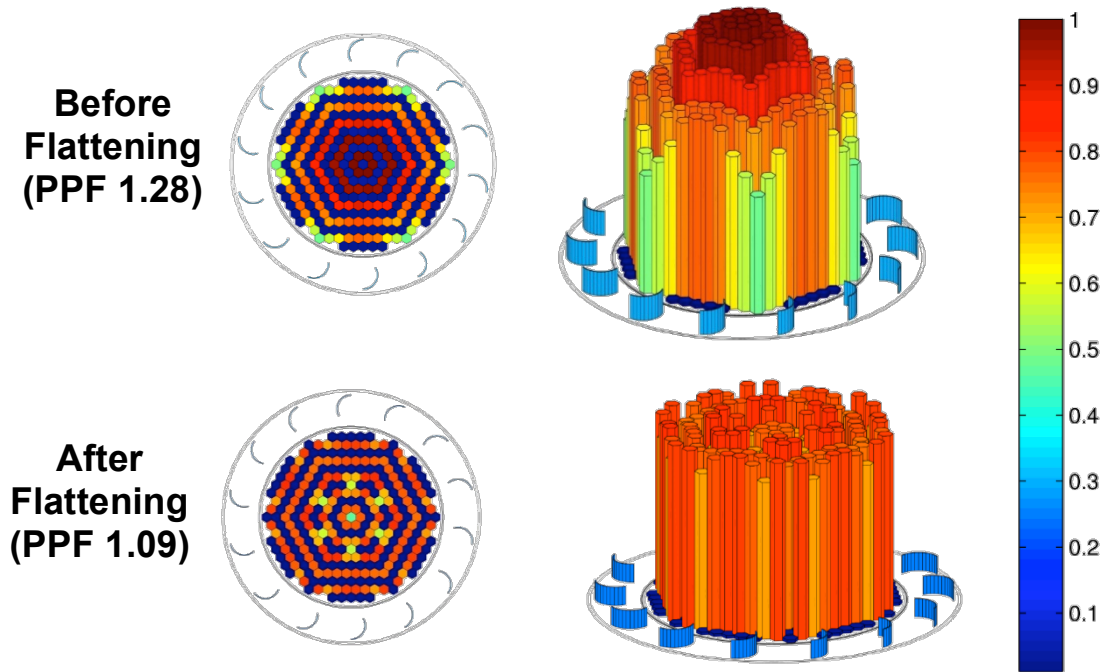

Before Flattening (PPF 1.28)

After Flattening (PPF 1.09)

FIGURE 15**:** A figure depicting the radial power flattening of the SCCTE Reactor.

A flatter power radial power profile is associated with a lower pressure drop because each channel must be flow restricted (orificed) to match flow to the power deposition in that channel. The larger the ratio between the power deposition in the hottest channel and the average channel the larger the presser drop from the required orificing.

Channel by channel flow orificing was demonstrated during the ROVER/NERVA project for reactors with ~76,000 individual channels. It is believed that channel by channel orificing can be employed in the SCCTE reactor as it only has 9211 individual channels.

To take into account potential affects of power peaking in the core, a `peaked channel factor' of 1.5 was used in reporting the pressure drop in Table 1. This essentially models another coolant channel that receives 1.5 times the average coolant channel power as well as 1.5 times the average coolant flow rate. All other parameters are reported for an average channel to ensure conservative estimates of all parameters.

Several additional seconds of I_{sp} from a NTP system can be achieved by altering the axial power profile in the fuel elements so that more power is deposited closer to the coolant entrance of a fuel element. In the SCCTE reactor, the power profile was shifted to the beginning of the fuel element via variable axial enrichment. An optimum point in the trade between I_{sp} and reactivity with axial enrichment was found by running many cases with SPOC. The point with the highest I_{sp} that met reactivity requirements was chosen. This point trades an additional 7.5 s of I_{sp} for 570 pcm of reactivity. Figure 16 shows the original un-optimized axial enrichment with the optimized axial enrichment that was found with our studies.
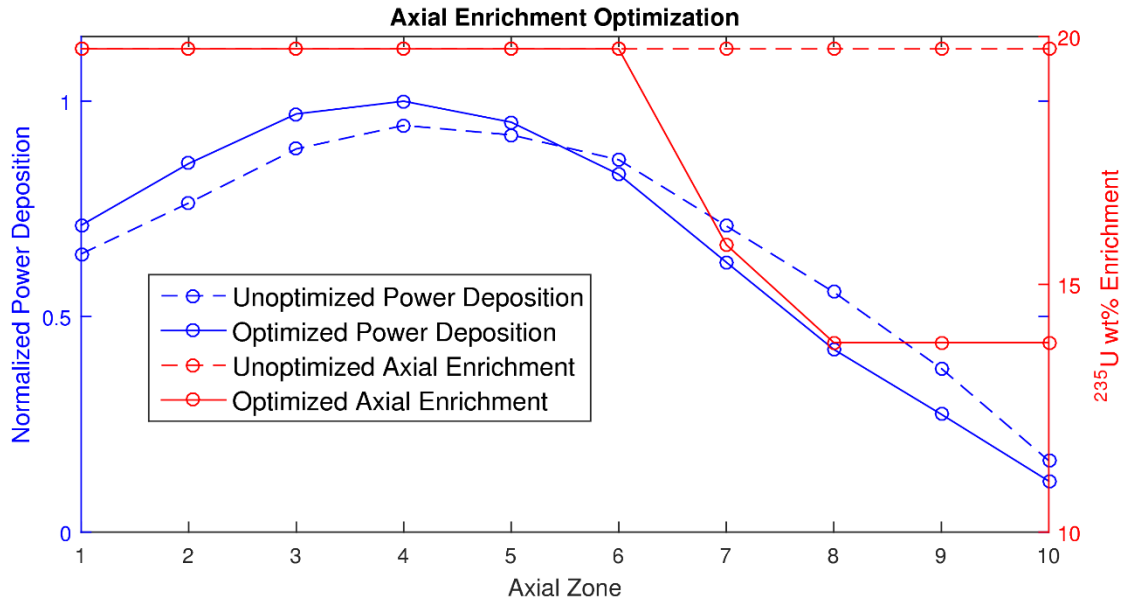

FIGURE 16: The axial enrichment and power profile for the axially optimized and un-optimized SCCTE reactor.

Thermal Sensitivity Study

The SCCTE nuclear reactor is in the initial design phase. As such, parameters that affect the system in any way should be identified so more engineering may be completed to mitigate any problems. Furthermore, due to the new fuel form being utilized, little material data is available and best estimates must be made from data for constituent materials. Likewise, the fuel has never been mass-produced so manufacturing uncertainty on a large scale is not available. Material and geometric uncertainties should be investigated in light of these ideas. SPOC was used to perform this study.

The previous neutronic and thermal sensitivity study focused on reactors that could achieve specific performance parameters. The following study focuses on a single reactor (SCCTE), and the effects of chosen parameters on the reactor's performance. The study focuses on individual parametric uncertainty by varying a single variable at a time while keeping all others at nominal values. This study presumes a single channel was created with off nominal parameters and evaluates the coolant outlet temperature and maximum fuel temperature for that channel. Volumetric power densities are kept constant when fuel geometry is varied. The pressure drop in the channel is kept at the nominal value by varying the mass flow rate.

The results of the parametric study are presented in Fig. 17. The clad thickness, channel roughness, fuel channel radius, clad conductivity, and fuel conductivity were all varied by the factor shown, up and down. The values were changed to high and low values around the nominal by various factors shown on the plots. These factors were chosen based on reasonable manufacturing and material uncertainties, however they are completely notional at this point. The blue bar graphs show how the value being measured changed when the variable was varied upwards, and the red bar shows the same when the value being varied was lowered. It can be seen that coolant channel exit temperatures are not affected by fuel and clad conductivities and geometry affects the outlet temperature substantially. The varied geometry has a large effect on the power produced in the channel because a 10% decrease in channel radius causes a 40% increase in power due to the power density remaining constant. Roughness also has a large effect; mainly in changing heat transfer coefficients.

The maximum fuel temperature is also not effected much by material conductivities, which shows at steady state, uncertainties in conductivity are not very important. The other three parameters are shown to be quite important for similar reasons as previously stated.

The assumption of constant power density should likely be re-evaluated but the actual change cannot easily be predicted with the tools at hand. In reality, more heat would be conducted away by neighboring coolant channels so these estimates are conservative. There are also neutronic effects that could change the power generation profile near the coolant channel.
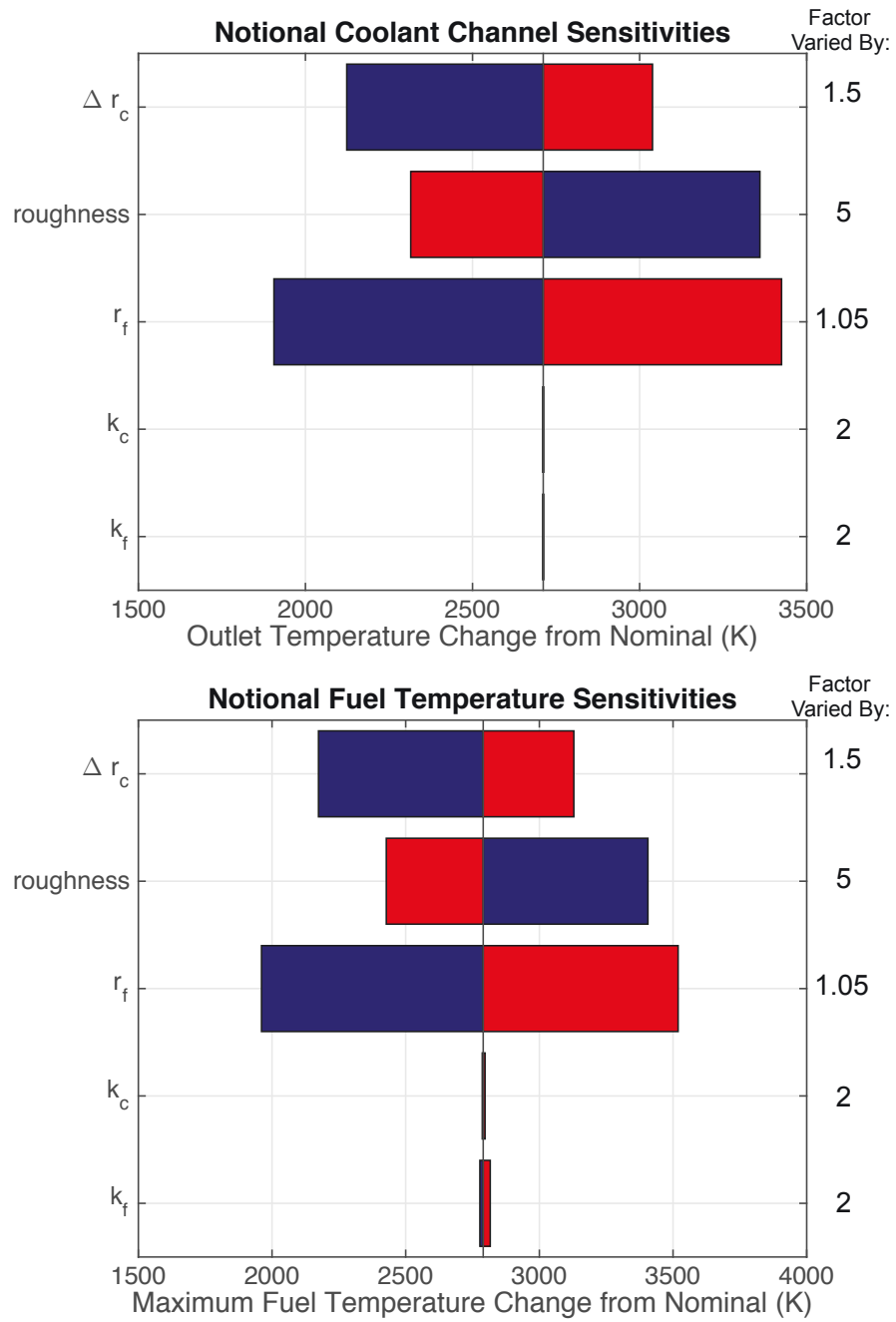
FIGURE 17: Notional Parametric Uncertainty of SCCTE Design Variables

Addressing In-Element Power Peaking

The power profile within a single fuel element of SCCTE is not flat in shape. A large power peaking exists. This in-element power peaking is a consequence of geometry and neutronics. When placing evenly spaced channels in a hexagon with a constant webbing thickness, the end result is that the outside channels are in contact with more fuel. Furthermore in a LEU cermet system, the mean free path of thermal neutrons in the fuel element is short enough that a larger number of fissions occur in outer regions of the fuel element that are adjacent to a moderator element.

It is important to understand and mitigate in-element power peaking phenomena. Large in-element power peaking factors increase the degree of orificing that is needed and may cause more stress in the fuel element.

The following technologies where identified to mitigate the in-element power peaking. Every idea has not been implemented into SCCTE because manufacturing concerns should be addressed before using the concepts in a design. A key is provided to describe modeling efforts.

Key:

* - Can be modeled in SPOC
** - Implemented in SCCTE
^ - Not currently in SPOC

- Channel by channel orificing.^
- Variable channel sizing so that the inner channels have more fuel*
- Rounding the edges of the fuel element*
- Decreasing the distance between the outer channels and the outer wall**
- Adding a neutron absorber to the corner channels*
- Increasing the heat transfer from the fuel to the moderator^
- Smaller flat to flat fuel elements*
- Modify Tietube and fuel arrangement so that power peaks along the entire outside of the fuel element as opposed just to a few faces*

Unmitigated in-element power peaking was on the order of 1.9 (Peak channel power/average channel power.). A few of the above not-SCCTE implemented concepts were used to determine their effect on element power peaking. Figure 18 shows the channel-by-channel power deposition for the peak fuel element in the SCCTE reactor after the edges where rounded, channels in the center where given a smaller diameter, and the outside wall of fuel was made to be 400 μm from the default webbing thickness of 650 μm. By employing these mitigation technologies, in-element power peaking was reduced to 1.41.

It is believed that the current pressure drop margin built into SCCTE's power balance will be able to allow for sufficient orificing to ensure all channels have the same peak fuel temperature. Multi-physics analysis will be needed to better understand in-element power peaking and to more accurately quantify the amount of orificing need. Specific phenomena not included in

current methods include conduction between channels and heat loss from outside channels to the moderator tubes touching the fuel elements.

To further mitigate possible effects caused by in element peaking, the actual peak temperature designed to was 2790 K rather than 2850 K. This 60 K margin was used in SNRE design to account for hard to quantify aspects of channel-by-channel orificing. Further study must still be completed to verify this in SCCTE.

It should be noted that in-element power peaking was a noted issue in the ROVER/NERVA work and to a lesser extent would be an issue for HEU cermet NTP systems.
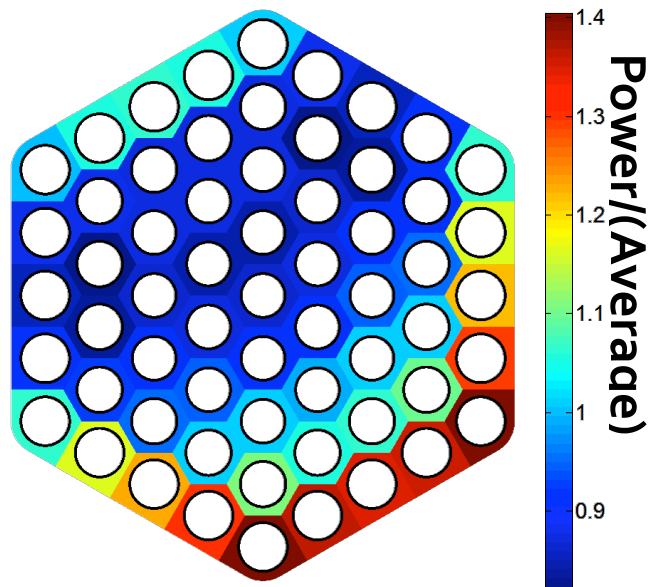

FIGURE 18: Power deposition in the peak fuel element of the SCCTE reactor.

Channel size and webbing thickness

The SCCTE coolant channels are placed on equally spaced triangular lattice. The methods described in [5] are used to model conduction between the internally heated fuel and the coolant channel wall.

The SCCTE fuel element coolant channel diameter is not much larger than the pitch between coolant channels (i.e. there is a thin fuel web between fuel channels). This low ratio of channel pitch to channel diameter (about 1.2 s/r_0 in Fig. 19) results in an azimuthally non-uniform heat flux on the coolant channel wall, which can cause a large azimuthal temperature gradient in the channel. Estimates from [5] indicate that the ratio of the minimum heat flux to maximum heat flux on a channel wall at any given horizontal cross section will be on the order of 1.5 for SCCTE. Figure 19 shows the angular dependence of heat flux on a channel wall and the ratio of channel pitch to channel diameter.

With current analysis, it cannot be stated if SCCTE's low pitch to diameter ratio is problematic but the associated phenomena merits further investigation. It is also possible that the angular dependence on the heat flux may not be compatible with our convective heat transfer models in the channel. In addition there may be unacceptable thermal stresses associated with the azimuthal temperature gradients around the fuel channel. Greater modeling and insight into the fuels mechanical properties is needed to understand the effects of a low pitch to diameter low on SCCTE's fuel.
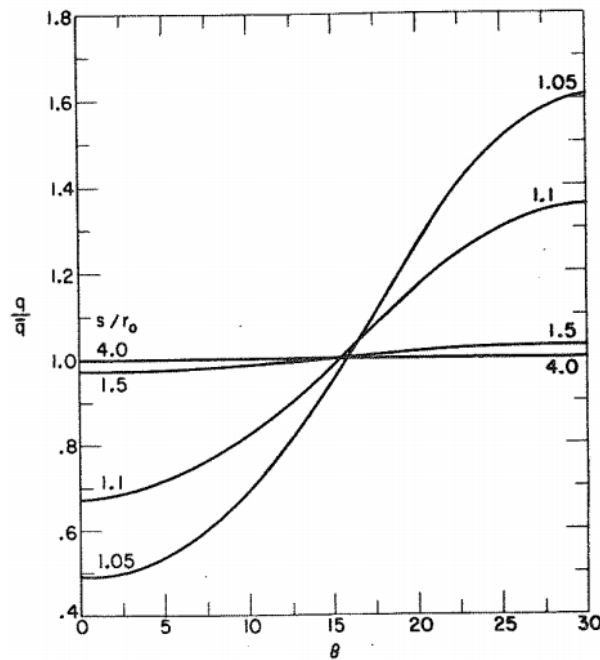

Fig. 4   Local wall heat flux distribution for triangular cooling arrangement

FIGURE 19: The angular dependence of heat flux on a coolant channel wall in a triangular lattice. s/r_o is pitch to diameter ratio. s/r_o for SCCTE is 1.21. Figure from [5]

Future Work and Conclusions

The SCCTE reactor design was shown to achieve an Isp of 894 s with 35000 klbs of thrust utilizing an LEU-W-Cermet fuel that would allow for reduced proliferation risk as well as reduced development costs. Many external design parameters were selected by NASA-MSFC and incorporated into the SCCTE design by CSNR. This allowed for an NTP reactor design that took design inputs from every aspect of the NTP design process.

A design study of the new type of NTP reactor that uses LEU W-Cermet fuel was created using the SPOC tool. A parametric sweep of many NTP design variables was performed to find general trends for NTP designs that have critical nuclear cores that are capable of being cooled with constraints of a given maximum fuel temperature and required thrust. The conclusions from this study allowed for a near optimum (with respect to many external design constraints) initial core design. This initial core design was further adjusted manually to reduce power peaking by making small adjustments to the fuel layout and creating axial and radial enrichment profiles.

Using the average SCCTE core coolant channel as a nominal case, another parametric study of coolant channel properties was created to determine effects of several parameters on material property and manufacturing uncertainties. The conclusions of this small study give insight into which parameters are important to have precise manufacturing on. The study also motivates more research into the sensitivity study in order to capture more details with a better model of the whole core, which may include heat transfer to surrounding channels as well as a power balance.

Several potential performance limiting aspects of the design such as non-spatially linear in-element power profiles were identified. Mitigating concepts were addressed and the ability to easily analyze some of these factors with current code capabilities was identified and demonstrated. Further research should also be performed here that can be incorporated into larger sensitivity studies.

It has been demonstrated that an LEU-NTP reactor can achieve performance parameters of interest, namely near 900s Isp with 35,000 klbs of thrust. Some new concepts were utilized that need technology demonstrations such as enriching W in the ^{184}W isotope, but the use of very low TRL technologies was minimized during the reactor design. It has been shown that the SCCTE reactor is a viable NTP concept that warrants further study.

The SCCTE reactor design is far from complete, but an excellent basis of study has been established. A non-exhaustive future works is listed below. This list focuses on computation aspects of the reactor design, though it should be said that experimental verification of the design is also extremely important.

- Model heat transfer to moderator-tubes to verify temperature limits are not exceeded and enough energy is extracted to drive turbomachinary
- Verify oriphiced coolant channels can adequately cool all channels given element power profile
- Complete sensitivity study by including neutronic effects

- Study and employ techniques to minimize in-element power peaking
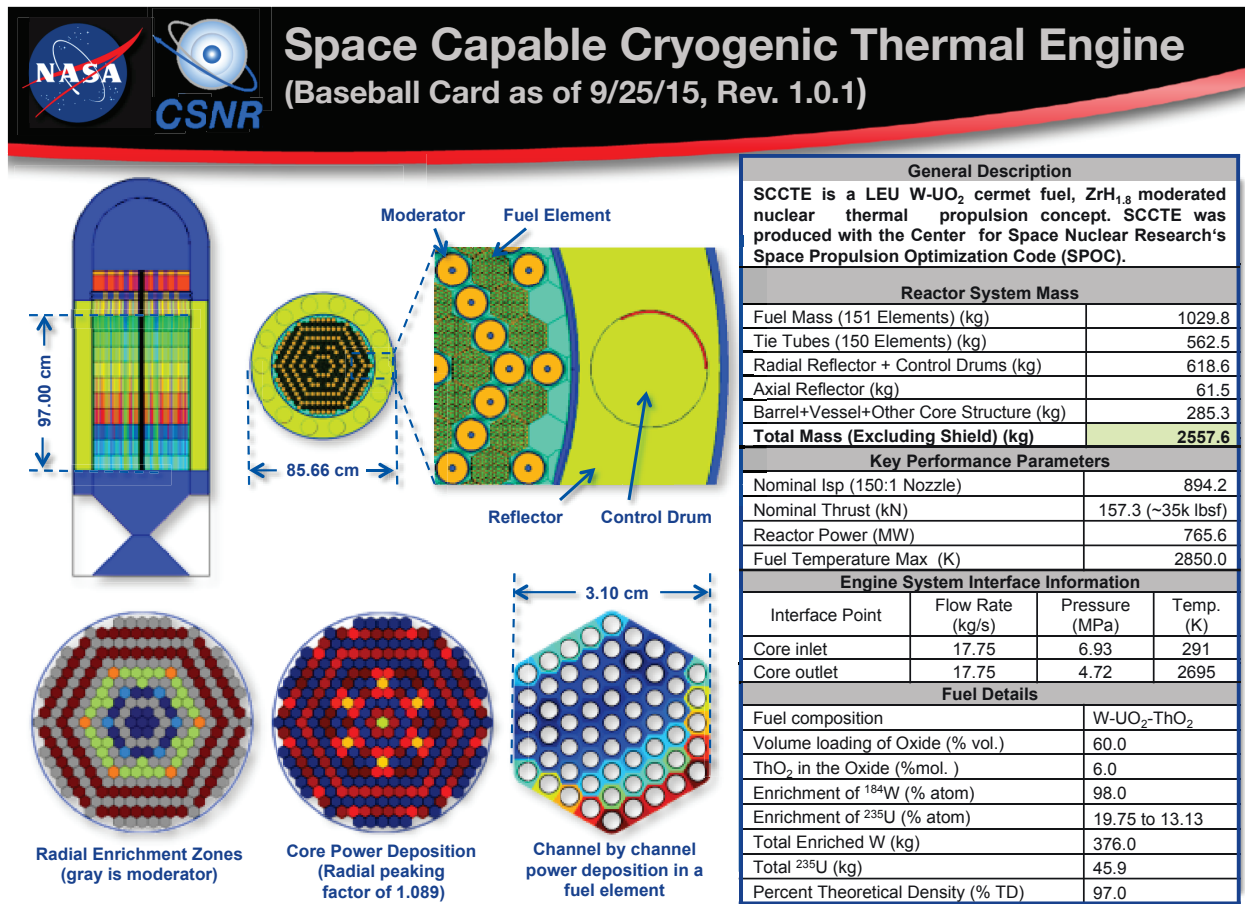- Produce New Design iteration based on what is learned

Space Capable Cryogenic Thermal Engine
(Baseball Card as of 9/25/15, Rev. 1.0.1)

General Description

SCCTE is a LEU W-UO$_2$ cermet fuel, ZrH$_{1.8}$ moderated nuclear thermal propulsion concept. SCCTE was produced with the Center for Space Nuclear Research's Space Propulsion Optimization Code (SPOC).

Reactor System Mass	
Fuel Mass (151 Elements) (kg)	1029.8
Tie Tubes (150 Elements) (kg)	562.5
Radial Reflector + Control Drums (kg)	618.6
Axial Reflector (kg)	61.5
Barrel+Vessel+Other Core Structure (kg)	285.3
Total Mass (Excluding Shield) (kg)	**2557.6**

Key Performance Parameters	
Nominal Isp (150:1 Nozzle)	894.2
Nominal Thrust (kN)	157.3 (~35k lbsf)
Reactor Power (MW)	765.6
Fuel Temperature Max (K)	2850.0

Engine System Interface Information			
Interface Point	Flow Rate (kg/s)	Pressure (MPa)	Temp. (K)
Core inlet	17.75	6.93	291
Core outlet	17.75	4.72	2695

Fuel Details	
Fuel composition	W-UO$_2$-ThO$_2$
Volume loading of Oxide (% vol.)	60.0
ThO$_2$ in the Oxide (%mol.)	6.0
Enrichment of ^{184}W (% atom)	98.0
Enrichment of ^{235}U (% atom)	19.75 to 13.13
Total Enriched W (kg)	376.0
Total ^{235}U (kg)	45.9
Percent Theoretical Density (% TD)	97.0

Moderator **Fuel Element**

Reflector **Control Drum**

97.00 cm

85.66 cm

3.10 cm

Radial Enrichment Zones (gray is moderator)

Core Power Deposition (Radial peaking factor of 1.089)

Channel by channel power deposition in a fuel element

FIGURE 20: SCCTE Baseball Card Reactor Description

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