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INTRODUCTION

The National Aeronautics and Space Administration (NASA) has set the goal of a manned mission to Mars by the year 2030 [1] and charged the National Academy of Sciences "to identify primary technical and programmatic challenges, merits, and risks for maturing space nuclear propulsion technologies of interest to a future human Mars exploration mission" [2]. One relevant technology, nuclear thermal propulsion (NTP), has notable advantages over traditional chemical rockets; most important among them is the ability to produce larger specific impulse on the order of 900s. The reduction of mission time is crucial for a manned mission to Mars to reduce the risk for the crew. Due to its higher specific impulse, NTP satisfies this need and is pursued as one technology to get humans to Mars [3, 4]. The construction of an NTP system has to negotiate several challenges laid out in Ref. [2]. One of these challenges is the need to startup the NTP system from essentially cold conditions to full power within one minute.

This paper focuses on studying the neutronics and thermal-hydraulics behavior of a simplified NTP model during prescribed reactivity insertions and mass flow rate (MFR) ramps. It is the goal of this paper to investigate startup procedures for a low enriched uranium (LEU), ceramic and metal material (CERMET), hydrogen cooled NTP system when varying reactivity insertion and MFR ramps.

In this work, the MOOSE [5] based system analysis code RELAP-7 [6] and neutronics code Griffin [7] are used to model a CERMET NTP system with a power of 250 MW. RELAP-7 provides one-dimensional thermal fluids analysis and two-dimensional solid thermal analysis that is coupled via conjugate heat transfer. Griffin provides a variety of neutron transport schemes, but in this work the point reactor kinetic equations (PRKE) are used. The feedback effects are obtained from a three-dimensional Serpent [8] model of a CERMET NTP mockup system developed at Idaho National Laboratory (INL).

To the authors' knowledge, the goals and results of the presented study are novel even though modeling and simulation (M&S) is extensively used for designing and understanding NTP systems. An approach similar to this work was reported in Ref. [9]. Zero-dimensional transient simulation capabilities for NTP systems were developed using the Modelica-based TRANSFORM library [9] and a startup transient is simulated. This study includes two-phase hydrogen flow, parts of the I&C system, and a turbine model with bypass, but it does not spatially resolve temperature distributions, include realistic feedback models, or provide guidance on a particular startup procedure. Furthermore, Krecicki and Kotlyar in Ref. [10] have performed a sweeping study of low-enriched uranium NTP design parameters with a detailed model using Serpent for neutronics, a 1.5D sub-channel code for thermal-hydraulics, and the POWER system analysis code. Their study analyzed more than 2400 core configurations with an exploratory emphasis on core design and sizing. However, this design study did not provide guidance on or consider the impact of startup procedures.

THEORY

The goal of this work is to provide a first step towards determining a suitable startup sequence for a mockup CERMET NTP system design with M&S. The current scope is limited to understanding the dependence of the average specific impulse $\bar{I}_{\text{sp}}$ defined by:

$$\bar{I}_{\text{sp}} = \frac{\int_0^{t_e} \dot{m}(t)v(t)dt}{g\int_0^{t_e} \dot{m}(t)dt}, \tag{1}$$

the maximum fuel temperature, and the time to produce a specified impulse on startup control parameters. These control parameters are the ramp rates of reactivity and MFR. In Eq. 1, $\dot{m}$ is the MFR leaving the nozzle, v is the speed at the nozzle outlet, and t_e is the time to reach a prescribed impulse (i.e., the integral of force over time).

The PRKE are used to describe the power level of the reactor [11]:

$$\begin{aligned}\frac{dp}{dt} &= \frac{\xi(t,T_f,\rho_f,\rho_m)-\beta_{\text{eff}}}{\Lambda_{\text{eff}}}p(t)+\sum_{i=1}^{6}\lambda_i C_i(t),\\ \frac{dC_i}{dt} &= \frac{\beta_{i,\text{eff}}}{\Lambda_{\text{eff}}}p(t)-\lambda_i C_i(t),\end{aligned} \tag{2}$$

where all symbols have their standard meaning, except for reactivity being denoted by ξ, power by p, average fuel temperature by T_f, and average hydrogen density in the fuel and moderator channels by ρ_f and ρ_m, respectively. The subscript "eff" indicates that parameters are effective parameters as explained in Ref. [12]. Reactivity ξ is composed of an operator-controlled reactivity insertion mimicking control drum rotation, Doppler feedback from the fuel, and density feedback effects from the hydrogen in the fuel and moderator regions:

$$\xi(t,T_f,\rho_f,\rho_m) = \xi_0(t)+\xi_{T,f}(T_f)+\xi_{\rho,f}(\rho_f)+\xi_{\rho,m}(\rho_m). \tag{3}$$

Serpent results indicated that moderator temperature has a negligible effect on reactivity (~$0.17\frac{pcm}{K}$ and temperature difference between hot and cold core is about 10 times larger for the fuel; hence moderator temperature effect is smaller by a factor of 100).

RELAP-7 solves single-phase thermal-hydraulics equations for channels and junctions and heat conduction equations for the heat structures. The main tasks of RELAP-7 are to provide feedback to the PRKE model by computing T_f, ρ_f, and ρ_m and to compute peak fuel temperatures.

SPACE NUCLEAR PROPULSION SYSTEM MODEL

A portion of the model specifications are loosely based on information presented in Ref. [4] (such as number of assemblies), but exact dimensions, power level, MFR, and fuel composition are taken from a mockup model developed at INL. The model has been used to successfully simulate 58 startup transients for the proposed reactor design. The thermal-hydraulics model is depicted in Fig. 1 with explanations provided in Table I.

The model parameters are scaled to comprise a 30-degree slice of a single fuel element of the 61 element reactor. Fluid flow is represented using one-dimensional channel components provided by RELAP-7. Conjugate heat transfer to and from heat structures are modeled using the Dittus-Boelter relationship [13], and pressure drops are calibrated to obtain 1:2 flow split between the channels c2 and c5 and a pressure ratio of 1.4 across the turbine t1 at steady-state; irreversible pressure drops in all other channels is of little importance and thus neglected. The dimensions of the heat structures have been tuned to conserve the volume of a 3D thermal model developed at INL and to reproduce cool-down curves obtained with these models for isothermal fluid temperatures. The heat resistance of hs4 has been tuned to obtain 1% of regenerative cooling under steady-state conditions. The power produced in the fuel is deposited mostly in the fuel itself with smaller portions going to the moderator and reflector. The breakdown of heat deposition is estimated using Serpent and is found in Table I. Model details are as follows:

- Steady-State power level: 341.5kW (250MW split between 61 assemblies and 12 symmetric slices).
- Steady-State MFR: $0.0084\frac{kg}{s}$ ($6.15\frac{kg}{s}$ split between 61 assemblies and 12 symmetric slices). This produces an outlet temperature of 2700K at nominal power.
- The reactivity model consists of four components. The last three are feedback relationships determined by calculating k_{eff} with several different set parameter values in the Serpent Monte Carlo code:
  - Reactivity Insertion: Tuned experimentally to produce desired final power level. This component mimics the rotation of a control drum.
  - Doppler Feedback: Feedback determined by the average temperature of the fuel $[K]$:

$$\xi_{T,f}(T_f) = -9.95 \times 10^{-4} \sqrt{T_f - 50}$$

  - Fuel Hydrogen Density Feedback: Feedback determined by density of hydrogen $[\frac{kg}{m^3}]$ flowing through channels inside the fuel:

$$\xi_{\rho,f}(\rho_f) = 5.40 \times 10^{-7}\rho_f^2 + 2.56 \times 10^{-4}\rho_f$$

  - Moderator Hydrogen Density Feedback: Feedback determined by density of hydrogen $[\frac{kg}{m^3}]$ flowing through channels inside the moderator:

$$\xi_{\rho,m}(\rho_m) = -1.52 \times 10^{-6}\rho_m^2 + 6.66 \times 10^{-4}\rho_m$$

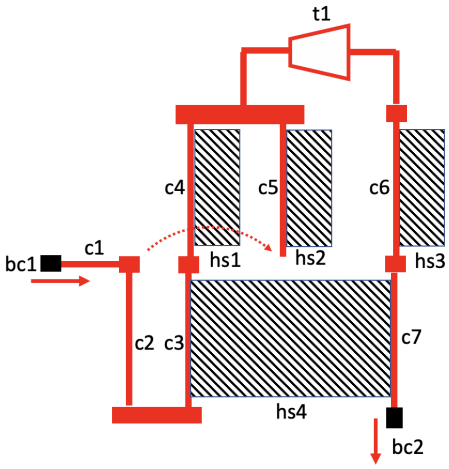

Fig. 1. Sketch of the RELAP-7/Griffin model.

RESULTS AND ANALYSIS

The results presented in this Section are preliminary. The model exhibits issues with solver convergence for several ramp duration combinations. These problems are currently under investigation. We expect the investigation to be completed before submission of the final paper.

Explanation of Test Cases

The behavior of average specific impulse, maximum fuel temperature, and time to reach a given impulse are investigated for different reactivity insertion and MFR ramps. In this preliminary study, these ramps are limited to be linear in time varying from zero reactivity and MFR at $t = 0$ to steady-state reactivity insertion and nominal MFR at the ramp end time. The outlet pressure is also linearly ramped from a value of 1 MPa to 6.9 MPa (the high non-zero initial outlet pressure is currently necessary to ensure numerical stability of the solver). The model is capable of higher-complexity ramping procedures which may be investigated in the future.

A total of 144 different combinations of reactivity and MFR ramp durations were simulated, using ramps with durations of $\Delta t = 5i$ s with $i = 1, \ldots, 12$. A total of 58 of these tests successfully finished running full two-minute transients, with the remainder failing to converge due to extremes in either of the ramping durations, that is, numerical. The simulation end time was set at two minutes as all transients were found to reach steady-state power well before this time. The smallest impulse generated by any transient was 6376Ns, thus we define t_e as the time it takes the system to reach an impulse of 6376Ns.

Analysis

Average specific impulse, overall maximum fuel temperature, and time when impulse equals 6376Ns are plotted versus MFR ramp duration for different reactivity insertion ramp durations in Figs. 2 through 4. Each line in these figures corresponds to a specific reactivity ramp duration. The general trends are:

TABLE I. Component descriptions for the NTP system model.

Shorthand	Component	Notes
bc1	Inlet Boundary Condition	Prescribed $\dot{m}_{\text{in}}(t)$, T_{in} = 50K
c1	Inlet Channel	-
c2	Duct to RGC[a]	Calibrated to ~ 1/3 of $\dot{m}^b$
c3	RGC Channel	Heated by hs4
c4	Reflector Channel	Heated by hs1
c5	Moderator Channel	Calibrated to ~ 2/3 of $\dot{m}^b$, Heated by hs2
c6	Fuel Channel	Heated by hs3
c7	Nozzle Channel	Cooled by hs4[c]
hs1	Be Reflector Heat Structure	0.8% of Power Deposition
hs2	ZrH Moderator Heat Structure	5.7% of Power Deposition
hs3	CERMET Fuel Heat Structure	93.5% of Power Deposition
hs4	Steel RGC Heat Structure	Models RGC
t1	Turbine	Calibrated Pressure Ratio of 1.4[b]
bc2	Outlet Boundary Condition	Prescribed Pressure

[a] RGC: regenerative cooling
[b] in steady-state conditions
[c] The purpose of the nozzle channel is to model RGC. The nozzle channel does not model acceleration of the fluid by converging/diverging model; this is modeled using the outlet pressure of bc2.

- $\bar{I}_{\text{sp}}$ increases as the reactivity insertion ramps shorten and, up to a point, as the MFR ramp duration increases. This behavior was expected because $\bar{I}_{\text{sp}}$ increases with core hydrogen outlet temperature. With larger core power and smaller mass flow rates early in the transient, the amount of hydrogen exiting the core at low temperatures is reduced. For fast reactivity insertions, slowing down MFR ramps can lead to a reduction in $\bar{I}_{\text{sp}}$; this behavior is discussed below.
- Maximum transient temperatures are on the order of 3200 K with deviations among different ramps of 78K. Maximum temperatures decrease as reactivity ramps become slower, which was expected. The effect of MFR ramp duration is examined below. Note that uranium nitride has a melting temperature below 3100K so for all transients at least localized melting occurs. The question if this would lead to reactor failure is beyond the scope of this work and must be answered using a more detailed reactor model.
- Faster reactivity and MFR ramps decreases the time to reach 6376Ns. This behavior was expected.

Three core reactor physics parameters explain the behavior of the core during this study:

- The core is fueled with LEU and features stronger Doppler feedback than earlier HEU designs [14]. An increase in temperature reduces reactivity almost without delay.
- The core thermal mass is small compared to the power; $\frac{p}{V\rho c_p} \approx$ 1200 K/s (V: core volume, ρ: fuel density, c_p: specific heat). At significant power levels, the core heats up very quickly unless power is balanced by a heat sink.

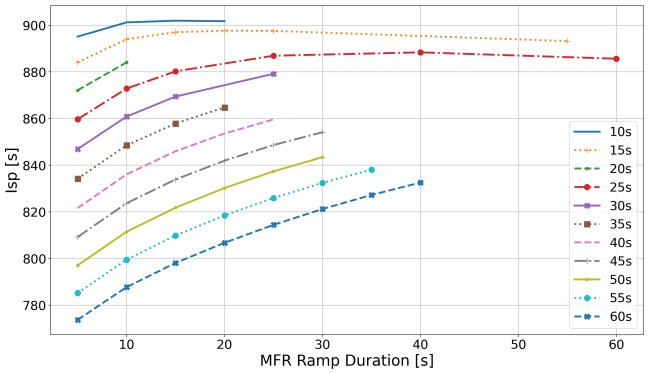

Fig. 2. Final specific impulse of the 58 tests.

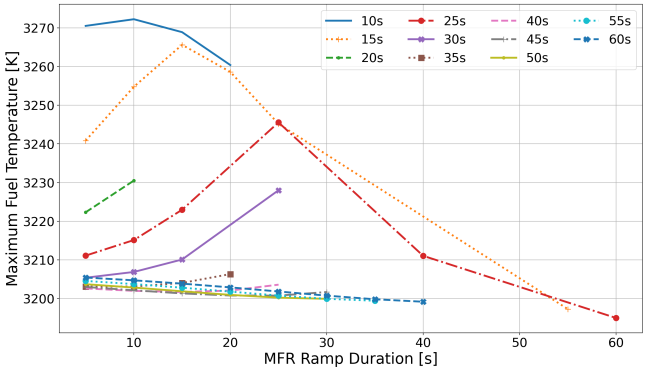

Fig. 3. Overall maximum fuel temperature of the 58 tests.

- The characteristic time scale of heat conduction is about τ = 6 seconds. The solution relaxes from a transient solution towards the steady solution by a significant amount

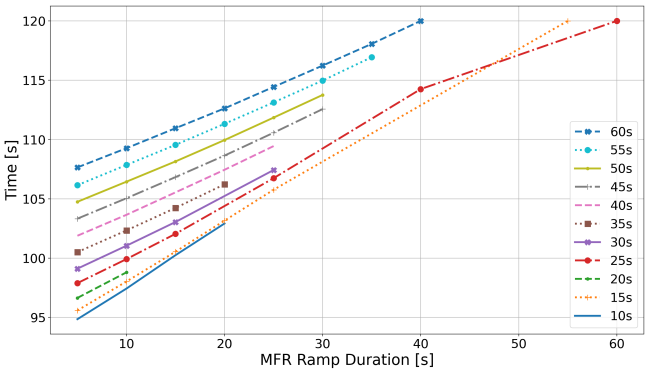

Fig. 4. Time to reach impulse of 6376Ns for the 58 tests.

within time span τ.

For slow reactivity insertions and sufficiently fast MFR ramps, the power increase is limited by the amount of inserted reactivity that is available. For fast reactivity insertions and slow MFR ramps, increasing power is frustrated by Doppler feedback. To increase power quickly, both reactivity and MFR must be ramped up at appropriate speeds. The reduction in $\bar{I}_{sp}$ for fast reactivity ramps and very slow MFR ramps is caused by slight cool down of the core (and thus hydrogen outlet temperatures) when it is Doppler feedback constrained and the MFR catches up to its nominal value.

The maximum fuel temperature during the transient is not strongly sensitive to the reactivity or MFR ramp duration, with a total range of 78K among all tests. The maximum temperature trends as predicted with reactivity ramp duration, as faster increases in power lead to larger temperature peaking. For long MFR ramp durations, hydrogen cooling limits maximum temperature, while at long durations, power is Doppler feedback-limited. At intermediate MFR ramp durations, exemplified by the 25s-25s point in Fig. 3, neither of these effects predominates to limit maximum temperature.

For an optimized startup transient, the trade off is mostly between time to desired impulse and average specific impulse because the maximum fuel temperature varies by only 2% over all studied ramp combinations. Introducing reactivity faster always benefits both figures of merit. Faster MFR ramps benefit time to desired impulse, but generally reduce specific impulse. As reactivity insertions become faster, the $\bar{I}_{sp}$ versus MFR ramp duration curves become flatter and move up towards 900s. For fast reactivity insertions, a fast MFR ramp should be chosen, as it does not hurt $\bar{I}_{sp}$, but it benefits time to desired impulse. With recommending fast reactivity and MFR ramps, practical technological constraints become central. If either ramp duration is limited in range, a more careful trade off study may be needed.

CONCLUSIONS

A nuclear thermal propulsion model has been developed by coupling the MOOSE-based codes system analysis code RELAP-7 and the neutronics code Griffin. Its dimensions are based on a higher fidelity 3D model developed at INL. The model accounts for fuel temperature and hydrogen density reactivity feedback mechanisms and models the thermal-hydraulics using a simplified 2D heat conduction/1D fluid flow system model. 144 preliminary startup transient simulation test cases were run using this model, with 58 running to completion.

Initial results indicate that to obtain large specific impulse and short time to reach a desired impulse, reactivity insertions into an NTP reactor should be performed as quickly as possible. Slower MFR ramps also improve the specific impulse of the NTP system, while faster MFR ramps improve the time to reach a desired impulse. The specific details of the optimal startup then depend on the relative importance of short startups and high fuel efficiency for a given startup scenario.

It is important to highlight the role of Doppler feedback on the startup behavior of the NTP system. For achieving a fast startup, Doppler feedback forces the operator to establish a heat sink quickly, i.e. providing sufficient hydrogen flow rate early in the transient. If the heat sink is not provided, then core temperature rises quickly and Doppler feedback offsets inserted reactivity. The NTP system becomes Doppler feedback limited.

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