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FRINK – A Code to Evaluate Space Reactor Transients

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Abstract. One of the biggest needs for space reactor design and development is detailed system modeling. Most proposed space fission systems are very different from previously operated fission power systems, and extensive testing and modeling will be required to demonstrate integrated system performance. There are also some aspects of space reactors that make them unique from most terrestrial application, and require different modeling approaches. The Fission Reactor Integrated Nuclear Kinetics (FRINK) code was developed to evaluate simplified space reactor transients (note: the term "space reactor" inherently includes planetary and lunar surface reactors). FRINK is an integrated point kinetic/thermal-hydraulic transient analysis FORTRAN code – "integrated" refers to the simultaneous solution of the thermal and neutronic equations. In its current state FRINK is a very simple system model, perhaps better referred to as a reactor model. The "system" only extends to the primary loop power removal boundary condition; however this allows the simulation of simplified transients (e.g. loss of primary heat sink, loss of flow, large reactivity insertion, etc.), which are most important in bounding early system conceptual design. FRINK could then be added to a complete system model later in the design and development process as system design matures.

Keywords: Space reactor, surface power, fission, reactor dynamics, nuclear kinetics.
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INTRODUCTION

Most proposed space fission systems are very different from previously operated fission power systems, and extensive testing and modeling will be required to demonstrate integrated system performance. FRINK[**] (Fission Reactor Integrated Nuclear Kinetics) was developed to facilitate near-term space reactor design and development by providing a simplified transient analysis tool. In its current state FRINK is a very simple system model, and the "system" only extends to the primary loop power removal boundary condition; however this allows the simulation of simplified transients (e.g. loss of primary heat sink, loss of flow, etc.). The addition of "balance of plant" components (e.g. power conversion, heat rejection, power management, etc.) would be required to make FRINK a true system model. Some good examples of complete system models can be found in El-Genk (2005) and Wright (2005), and it is possible that FRINK could be inserted as the reactor module into these or similar codes.

FRINK was initially written to evaluate small, compact reactors, specifically reactors for space power applications. Most space reactors are very compact and have a fast neutron spectrum, and thus point kinetics is generally very accurate for these systems. There are two unique aspects of highly reflected fast reactors (e.g. space reactors) that do not occur in more traditional reactors. (1) The neutron reflector has a very important impact on dynamic performance, and in some cases the temperature coefficient of the radial reflector is higher than that of the fuel. The thermal time constant of the reflector is much longer than any component in the core, which requires all reflector temperature and expansion effects to be modeled individually. (2) Reflected neutrons have a much longer fission lifespan than in-core neutrons. In effect, this creates an additional delayed neutron group, referred to as a geometric delayed neutron group. This group can have a lifespan orders of magnitude longer than neutrons that do not leave the core, and have much higher worth due to moderation.

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In addition to accommodating the effects of highly-reflected reactors, FRINK is tailored to handle other aspects unique to space reactor applications (e.g. simulation of the lunar day-night environment). In addition, the solution methodology is tailored to give optimum code performance (run-time) by evaluating only the equations and transients anticipated for space reactor simulations. FRINK supports liquid-metal and gas-cooled reactors with either open-lattice or annular flow, and heat-pipe reactors; although the thermal-hydraulic analysis of each of these options is currently rather simple. One additional value of FRINK is that the inputs are automated with output from MCNPX (Hendricks, et al., 2003), MONTEBURNS (Poston and Trellue, 1999), and MRPLOW (Amiri, et al., 2006) to fit within a streamlined space reactor design process. This allows limiting transients to be evaluated very early in the design process, as opposed to performing transient analyses late in the design process, which can result in lengthy iterations.

MODEL DESCRIPTION

This section describes the equations and assumptions that determine the "physics" of the code. The point kinetics equations are generic for any application, while the decay power, delayed neutrons, thermal-hydraulics, thermal balance, etc., are current specific to space reactor (or similar) applications. The biggest simplification within this model is that the "system" ends at the primary loop heat exchange to the power conversion system. Thus, the power transferred from the primary loop to the power conversion is specified as a boundary condition, which can be constant, a step change or a ramp change.

Point Kinetics Equations

The form of the point kinetics equations (nomenclature at end of paper) that is solved is:

$$\frac{dn(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda} n(t) + \sum_i \lambda_i C_i(t) + S(t) \tag{1}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t) \tag{2}$$

$$\rho(t) = K_0(t) + \sum_j \alpha_j(T_j)\,(T_j(t) - T_{0j}) \tag{3}$$

The relation between system power and neutron population is determined by the fraction of neutrons that cause fission, the average time it takes for these neutrons to cause fission, and the energy deposited by fission.

$$P_{fis} = \left(\frac{1.6e-13 Q_{fis}}{\Lambda}\right)\left(\frac{\rho}{\nu}\right) n \tag{4}$$

Reactivity Temperature Coefficients

Reactivity Temperature Coefficients (RTCs) are the most important parameter in most reactor dynamics calculations. FRINK offers the ability to enter any number of regions for which RTCs can be provided. In its current state, FRINK incorporates four temperature feedback mechanisms as a function of temperature.

- Fuel Meat
- Fuel Clad
- Core coolant, core structure/vessel, axial in-pin reflectors
- Radial reflector and shield (assumed that these expand freely relative to core)

Each RTC (α) is calculated with MCNPX and includes both thermal expansion and cross section effects (both Doppler and scattering). These RTCs are currently entered as constants, so their values should be evaluated within the temperature range considered. In the future it is planned to allow the input of temperature-dependent RTCs

(α(T)), plus the ability to input power (or other) feedback coefficients; however for most space reactors the RTCs are rather uniform with temperature and it is rare to have a significant power coefficient.

Delayed Neutrons

There are two types of delayed neutrons that can be modeled – (1) fission delayed neutrons and (2) geometric delayed neutrons. The delayed neutron fraction is referred to as beta, but the parameter of interest is beta-effective which indicates the reactivity worth of delayed neutrons (in the point kinetics equations β is beta-effective). In most reactors a delayed neutron has more worth than a fission neutron because they are of lower average energy – moderated neutrons have more worth in most reactors due to changes in cross sections and leakage rates.

The net fission beta-effective for a reactor is calculated with MCNPX. MONTEBURNS is used to determine the fraction of fission from each isotope (limited to Th232, U233, U235, U238, Pu239, Pu240, Pu241, Pu242). The numbers are input into FRINK, which then determines the time-constants of the delayed neutrons – the time-constant groups or bins are determined by the dominant fissile isotope and the delayed neutrons from other isotopes are force fit into these bins. All delayed neutrons are assumed to have the same energy/worth, i.e. there is no individual worth for each bin.

The need for geometric beta-eff(s) is dependent on the reactor type, and is generally only important in fast reactor systems that have very high-worth reflectors (e.g. space reactors). If the spectrum is fast and the reactor is compact, then the lambda (Λ = fission lifespan) of fission neutrons that stay within the core can be very short (<< 0.1 μs). The lifespan of neutrons that return from the reflector and subsequently cause fission can be up to several orders of magnitude longer (>> 1 μs). The longer time-constant of the reflected neutrons can slow down fast transients, therefore these neutrons can be modeled as additional delayed neutron groups by inputting the beta-effective and time-constant as determined by MCNPX. These geometric delayed (neutrons) are actually delayed fissions, but as is done with the material delayed neutrons, what is really being added to the neutron balance is multiplied by neutron worth (in this case the worth factor inherent in the geometric delayed group would be υ). In general, the overall impact of these geometric delayed neutrons is very small except for reactivity insertions that approach or exceed prompt critical (where prompt is based on the material beta).

Decay Power

Decay power is dependent on the power/time history of the reactor (i.e. the reactor state). The decay power for a given reactor state (used at beginning of a transient) can be calculated with MONTEBURNS/ORIGEN. A curve is created that represents how the power would decay over time if the reactor was shut-off (i.e. fission rate goes immediately to ~zero). The problem with simply using this curve is that fission can continue (and sometimes increase) during transients, thus decay power precursors are still being generated and decay power will not drop off as dictated by the curve. The solution to this problem is to establish a precursor, time-constant group structure for decay power in the same manner that is traditionally done for delayed neutrons. The decay power curve from ORIGEN is fit into precursor bins with time-constants ranging from tenths of seconds to tens of years. Assuming that the production rate of decay-power precursors is solely a function of fission power, then the steady-state precursor values can be calculated in the same manner as for delayed neutrons. This method does not incorporate self-shielding effects that might change the precursor production rate as a function of power (flux), and it does not properly increase the level of long-lived precursor groups that have not saturated (i.e. reached steady state levels), but for most reactor-states/transients these assumptions should be adequate.

Reactor Power Balance Calculations

For these calculations, the reactor power is defined as the total power deposited within the reactor vessel – power deposition in the reflectors and shields is factored as additional power. Power is inserted into the model via either fission or decay power. Fission power deposition is subdivided by major component (fuel, clad, structure, coolant, reflector, and shield), by inputting power fractions generated by MCNPX - the power deposition fractions are assumed to be constant and uniform within all components. It is assumed that the decay power is deposited solely within the fuel meat.

Power can be removed from the model by 3 mechanisms. The first mechanism is the power drain to the power conversion system which is currently supplied by user input, which can be a function of time with a combination of step and/or ramp functions. The second power loss mechanism is radiation from the primary loop – the user inputs what fraction of steady-state reactor power is rejected, and FRINK calculates the effective radiative surface area based on the coolant temperature and emissivity. The third power loss mechanism is heat loss from the core through the reflector and shield – this calculation is based on several geometric, thermal-bonding, and insulation inputs.
The modeling of heat transfer through the radial reflector and shielding is of increased importance for space reactors because of the large reflector RTC combined with the relatively large fraction of core power deposited in the reflector/shield (several percent). FRINK allows the emissivity and gap conductances within the reflector, drums, and shield to be individually varied to properly model the system heat loss to the ambient environment. FRINK also allows the simulation of the lunar environment over the entire day-night cycle, by allowing the surface temperature and solar insulation to be varied as a function of time.

Core Thermal-Hydraulic Calculations

Currently, FRINK thermal-hydraulic (T-H) calculations are limited to 1D core-average solutions. The fuel, clad, and core-block (if used) temperatures are calculated based on a radial 1D unit-cell, where the unit cell is the smallest symmetric section (depending on the flow configuration). The coolant is modeled as a thermal lump at one uniform temperature. This is reasonable for "slow-moving" transients with coolant flow, but more detail will be needed for rapid transients (where substantial coolant temperature changes take place in seconds) or stagnant coolant conditions (where most of the loop thermal inertia is not "seen"). The radial reflector and shield temperatures are based on a 1D radial solution for the given core, coolant and sink temperatures.

Material properties (k, Cp, ρ, CTE, ε, υ) are recalculated at every iteration as a function of temperature. The flow heat transfer coefficient, friction factor, and other parameters are calculated as a quasi-steady-state solution at each time step. The fuel/clad gap heat transfer coefficient is determined by the quasi-steady gap-size (based on the temperatures and CTEs of the fuel and clad) and the conductivity of the fill material (e.g. He). Thermal radiation across the gap is also included. There is currently no calculation of core radial delta-T (inner pin to vessel) which could be important in loss-of-flow decay heat scenario. This will be difficult to calculate because natural circulation may tend to keep temperatures much more uniform than would be the case in conduction only. In the current designs the peak coolant temperature appears more limiting than peak fuel/clad temperature, so the bulk temperature calculation is reasonable.

Structural Calculations

Vessel/loop structural integrity may be one of the limiting issues for a transient, if that transient results in a significant bulk system temperature rise. As coolant temperature increases the vapor pressure increases and the material strength decreases, which can lead to failure due to primary stresses or creep (depending on the length of the transient). FRINK calculates the hoop stress in the vessel due to vapor pressure at designated intervals (to avoid the computation time of calculating every time step), and compares it to the design limits (2/3 yield strength and 1/3 ultimate strength). FRINK also calculates the vessel creep over each interval and integrates the creep over the entire transient length. Fuel pin structural calculations are not currently performed; however, the fission gas production and release are very small in the current space reactors being evaluated, thus the pressure loads are low.

SOLUTION METHODOLOGY

The first solution step is to calculate the steady state temperatures and precursor concentrations that define the pre-transient reactor state. The point kinetics and heat transfer equations are discretized via the Crank-Nicolson method:

$$\frac{dx(t)}{dt} = F(x(t),t) \longrightarrow \frac{x(t+\Delta t) - x(t)}{\Delta t} = \theta F(x(t+\Delta t), t+\Delta t) + (1-\theta) F(x(t),t) \tag{5}$$

If θ=0, then the method is explicit, θ=1, the method is implicit. For the Crank-Nicolson method θ=0.5; this produces the smallest error and the resulting time integration is conditionally stable.

The equations are made non-linear by the ρn(t) term, which results in a $\theta^2 F(t+\Delta t)n(t+\Delta t)$ term. Instead of taking the extra time to iterate, it is assumed that the change in F is small enough that this term can be approximated as $\theta^2 F(t)n(t+\Delta t)$, which is implemented by explicitly calculating reactivity at each step based on the temperatures at the beginning of the step (this solution could be improved with a predictor-corrector for reactivity if deemed necessary).

The resulting set of equations is solved by a direct solver. The number of equations is equal to the number of precursor groups Nc, plus the number of feedback function Nf, plus one. The equations are placed in a square matrix in such a way that the set of equations can be represented as four vectors (a1, a2, a3, and b).

$$\begin{bmatrix} a_{ij} & 0 & 0 & 0 & 0 & 0 & 0 & 0 & a_{ij} \\ 0 & a_{ij} & 0 & 0 & 0 & 0 & 0 & 0 & a_{ij} \\ 0 & 0 & a_{ij} & 0 & 0 & 0 & 0 & 0 & a_{ij} \\ 0 & 0 & 0 & a_{ij} & 0 & 0 & 0 & 0 & a_{ij} \\ 0 & 0 & 0 & 0 & a_{ij} & a_{ij} & 0 & 0 & a_{ij} \\ 0 & 0 & 0 & 0 & a_{ij} & a_{ij} & a_{ij} & 0 & a_{ij} \\ 0 & 0 & 0 & 0 & 0 & a_{ij} & a_{ij} & a_{ij} & a_{ij} \\ 0 & 0 & 0 & 0 & 0 & 0 & a_{ij} & a_{ij} & a_{ij} \\ a_{ij} & a_{ij} & a_{ij} & a_{ij} & 0 & 0 & 0 & 0 & a_{ij} \end{bmatrix} \begin{bmatrix} C1 \\ C2 \\ C3 \\ C4 \\ T1 \\ T2 \\ T3 \\ T4 \\ n \end{bmatrix} = \begin{bmatrix} b_i \\ b_i \\ b_i \\ b_i \\ b_i \\ b_i \\ b_i \\ b_i \\ b_i \end{bmatrix} \tag{6}$$

A subroutine was written to solve this matrix directly with a small number of calculations. The time step adjustment is determined by the approximated numerical error, which is estimated by including the second term of the Taylor expansion in equation #1:

$$\frac{dn}{dt} = \frac{\Delta n}{\Delta t} - \frac{\Delta t^2}{6}\frac{d^3 n}{dt^3} = F(\theta, n, t) \tag{7}$$

The third derivate is approximated by:

$$\left.\frac{d^3 n}{dt^3}\right|_{i+\frac{1}{2}} = \frac{n_{i+2} - 3_{i+1} + 3n_i - n_{i-1}}{4\Delta t^3} \tag{8}$$

The relative (fractional) error in Δn at time step i +1/2 is thus:

$$\text{Error in } \Delta n \cong \frac{n_{i+2} - 3_{i+1} + 3n_i - n_{i-1}}{24(n_{i+1} - n_i)} \tag{9}$$

The code adjusts the time step so that the error is just below the specified value for the maximum allowable relative error in Δn. If a time step exceeds this error, then the code backtracks to a previous time step and resumes calculations with a smaller time step. Likewise, if calculations proceed for numerous time steps without exceeding the acceptable error then the code considers increasing the time step.

SUMMARY OF UNIQUE FEATURES IN FRINK

FRINK contains many features that make it well-suited to space reactor transient analysis. Some of the features listed below may exist individually in various codes, but as a whole FRINK is a unique design and analysis tool for evaluating limiting space reactor transients early in the design process.

Radial Reflector Reactivity Feedback

Reactors with very high radial reflector worth generally have a significant RTC associated with the radial reflector. In some space reactors, the magnitude of the reflector RTC can be equal or greater than the fuel (note that both the fuel and reflector feedback are dominated by material expansion and are negative). As a result, a change in reflector temperature will require an opposing change in fuel temperature to maintain a reactivity balance (unless active control is used) or vice versa. Therefore it is important to model reflector temperature separately from the reactor, and apply a specific reactivity coefficient to each component. The importance of this in demonstrated in Fig. 1, which shows that transient response of a NaK-cooled, SS/UO_2 surface reactor (Marcille et al., 2006) to a 50% drop in power conversion heat sink (note that the power conversion does not further respond, the heat rejection remains at 50% of nominal throughout the transient).

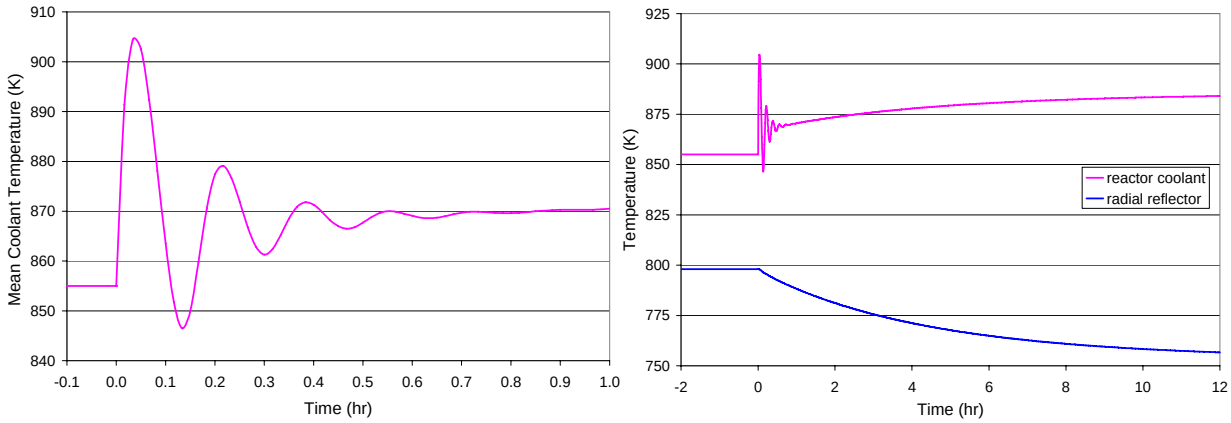

FIGURE 1. Effect of Reflector Temperature Feedback on Core Temperature for 50% Loss of Primary Heat Sink.

Both charts in Fig. 1 plot the average core coolant temperature as a function of time for the same 50% loss of primary heat sink transient. The left chart displays the transient during the first hour, which shows a somewhat typical temperature oscillation as the reactor searches for the stable reactivity point, although it is clear that the temperature is not returning to its previous steady-state value, despite the fact that the control elements have not moved. This temperature rise is even more evident on the chart on the right, which displays the transient over 12 hours. This chart also displays the temperature drop of the radial reflector, which in turn causes the subsequent core temperature rise. The reflector cooling is caused by the ~50% drop in reactor power (in response to the 50% drop in heat sink), because the reflector temperature is largely dictated by fission power deposition (neutrons and photon) as opposed to heat loss from the core vessel. In this transient the net temperature drop of the reflector is ~40 K over the course of ~12 hours. The RTC of the reflector is ¾ the value of the overall core RTC, so the core temperature must increase ~30 K to achieve the stable power/reactivity level (again assuming no control movement). A more complete system model would integrate the response of the power conversion system, instead of assuming a constant 50% removal, but it would probably not dampen this core temperature rise substantially. If this core/reflector temperature issue becomes a concern, the reactor should be designed to minimize the reflector RTC as compared to the core RTC, and to minimize the relative impact of fission power deposition on reflector temperature.

Geometric Delayed Neutron Groups

This feature is unique to very compact, fast spectrum reactors that have very high reflector worth. In these types of reactors, the neutrons that return from the reflector regions and subsequently cause fission have a much longer lifetime that those that remain within the core. FRINK allows these "geometrically delayed" neutrons to be modeled

as delayed neutron groups. To provide an example, transient calculations were performed on the previously referenced NaK-cooled, SS/UO_2 surface reactor. MCNPX was run to calculate the worth and lifetime of neutrons from various regions of the reactor. The overall average lifetime within the reference reactor is ~6 μs. When split into geometric groups, the neutrons that remain within the core region and cause fission have ~60% of the total reactivity worth and a lifetime of ~20 ns, while neutrons that return from the outer reflector and cause fission have ~1% of the reactivity worth and have a lifetime of ~200 μs (4 orders of magnitude longer). A hypothetical transient was run with FRINK for which there is an instantaneous reactivity insertion followed by a reactor scram over 5 seconds (i.e. all drums rotating to shutdown position in 5 seconds – which again is a hypothetical situation). The reactor was assumed to be at a steady-state power of 1 kWt prior to the reactivity insertion. Figure 2 shows the comparison of the peak transient power reached as a function of insertion, with and without the use of geometric delayed groups.

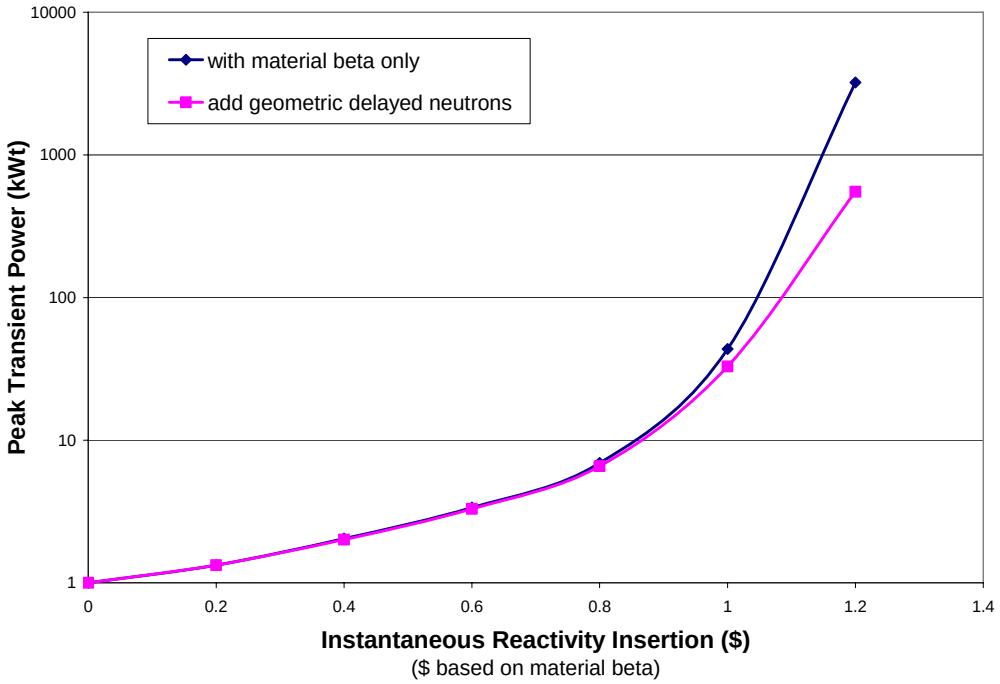

FIGURE 2. Effect of Geometric Delayed Neutron Groups for Instantaneous Reactivity Insertions.

Figure 2 shows that the effect of the geometric delayed neutron groups is only significant for very large reactivity insertions. The timescale of even the slowest geometric group (~200 μs) is still very fast relative to the timescale of temperature changes in modest transients; therefore, the time delay of these neutrons does not impact nominal transients. However, geometric delayed neutrons could be very important for evaluating postulated criticality accidents, both by increasing the magnitude of reactivity insertion that the system could survive intact as well as decreasing the energy yield of a very high (>>$1) postulated insertion.

Fission-Fraction-Weighted Delayed Neutron Bins

This function is used in many kinetics codes for terrestrial reactors, largely because plutonium and higher actinides can significantly shorten the delayed neutron fraction. For most space reactors this is not a concern because they are usually fueled with uranium and they experience little transmutation. Some more recent space reactor designs have considered reduced-enrichment fast spectrum systems, which can result in significant U238 fission fractions. The beta of U238 is nearly twice as large as for U235, therefore beta can be significantly (up to 25%) larger for these concepts than would be assumed by using that of U235. Also, in most 6-group delayed neutron data the time constants are different for various actinides. FRINK assumes the time structure of the most prevalent fission actinide, and conforms the other actinide delayed date to these bins.

Precursor Based Decay Power Model

This feature models decay power in a manner similar to delayed neutrons, with several time dependent bins into which power is added as fissions occur and power is subtracted as decay takes place over time. This is useful in all transients for which decay power eventually becomes a significant fraction of total power. In most space reactors, it takes a significant period of time for fission to "shut-down" during a loss of heat sink or flow transient (because of the relatively low adiabatic heat-up rate and feedback coefficients); therefore, decay precursors are still being produced well into the transient, and decay power will be under-predicted without a precursor model. Even more important can be transients that include a large power spike prior to fission reactor shutdown, in these cases decay power precursors created in this spike can add a significant fraction to the overall decay power profile.

Space Reactor Transient Flexibility

FRINK is very flexible with respect to transient modeling, from microsecond responses to high reactivity insertions, to very long transients occurring over several lunar day-night cycles. The FRINK input file allows unlimited time-dependent changes in reactivity, flow rate, and power removal, either as step or ramp inputs. A run can begin with a cold, subcritical reactor, or a system operating at any steady-state power. A proportional-integral-differential (PID) controller algorithm exists within FRINK that provide reactor control if desired, although the algorithm constants will generally need to be tuned for different systems (Dixon, et al., 2007). The user also has the ability to specify maximum control element movement rates for startup or nominal operation. Several parameters can be changed that impact the system heat transfer, such as component emissivities, gas or vacuum gap specification, and/or the operating/ambient environment. Finally, several computation parameters can be varied, e.g. the minimum and maximum allowable time step, the error tolerance, the number of nodes within the thermal models, etc.

Integration With a Large Suite of Space Reactor Design Tools

FRINK is part of an integrated suite of space reactor codes developed at the Los Alamos National Laboratory. The system geometry used by FRINK is specified by a file generated by MRPLOW (MRPLOW, Modular Reactor to Power Lunar Outpost Works, which is a surface reactor design tool) – this file contains dimensions, materials, and densities for the fuel-pins, core, reflector and shield. MCNPX provides reactivity data, fission lifespan, actinide fission fractions, and geometric delayed neutron, and MONTEBURNS provides decay power data. MRPLOW is also used to produce several MCNPX models with different control element positions to determine the reactivity worth of the control elements as a function of position. The results of FRINK provide feedback to the design process in several ways, e.g. if peak fuel temperatures are reached then pin size might be reduced, if peak vessel stress or creep is exceeded then vessel thickness might be increased, if reactor coolant freezes over a lunar night then more thermal resistance might be added between the reactor and shield, etc.

CONCLUSION

FRINK has been developed to evaluate simplified system transients for space and surface reactors. Several features incorporated into FRINK make it possible to evaluate some unique aspects of space reactor (or similar) systems. The modeling of the power system balance of plant (power conversion, radiator, etc.) is very simple and needs to be improved before FRINK could be considered a complete system model. As such, the code is currently being used as part of the LANL space reactor design process to evaluate limiting system transients that do not require sophisticated balance of plant models (e.g. loss of flow). FRINK has been written in great detail with respect to nuclear kinetics and reactor power balance, with the intention that it will eventually become a module within a much more sophisticated and complete system modeling tool.

NOMENCLATURE

n = neutron population (power is assumed in constant proportion with n)
C_i = precursor concentration of delayed neutron group i
λ_i = decay constant of delayed neutron group i

β = $\Sigma\beta_i$ = reactivity worth of delayed neutrons (a.k.a β_{eff})
ρ = reactivity (dk/k) (=0 if critical system)
K_0 = k_{eff} -1 at initial component temperatures (T_0) (may change due to control movement)
Λ = prompt fission neutron lifespan (s) excluding any delayed geometric groups
α_j = reactivity temperature coefficient of component j (dk/k-K)
T_j = temperature of component j (K)
S = neutron source (n/s) e.g. spontaneous fission, (α,n), spallation (accelerator or cosmic)
P_{fis} = fission power (W)
Q_{fis} = average fission energy deposition in system (MeV/fis)
υ = neutrons released per fission (n/fis)

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