CONF-980103--1

NUCLEAR MODULES FOR SPACE ELECTRIC PROPULSION

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Paper to be presented at the *Space Technology and Applications International Forum, 15th Symposium on Space Nuclear Power and Propulsion*, Albuquerque, NM, January 25-29, 1998.

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*Research sponsored by the Oak Ridge National Laboratory managed by Lockheed Martin Energy Research Corp. for the U.S. Department of Energy under contract No. DE-AC05-96OR22464.

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Abstract

The analysis of interplanetary cargo and piloted missions requires the calculations of the performances and masses of subsystems to be integrated in a final design. In a preliminary and scoping stage the designer needs to evaluate options in an iterative way by using simulations that run fast on a computer. As a consequence of a collaborative agreement between the National Aeronautic and Space Administration (NASA) and the Oak Ridge National Laboratory (ORNL), ORNL has been involved in the development of models and calculational procedures for the analysis (neutronic and thermal hydraulic) of power sources for nuclear electric propulsion.

The nuclear modules will be integrated into the whole simulation of the nuclear electric propulsion system. The vehicles use either a Brayton direct-conversion cycle, using the heated helium from a NERVA-type reactor, or a potassium Rankine cycle, with the working fluid heated on the secondary side of a heat exchanger and lithium on the primary side coming from a fast reactor.

Given a set of input conditions, the codes calculate composition, dimensions, volumes, and masses of the core, reflector, control system, pressure vessel, neutron and gamma shields, as well as the thermal hydraulic conditions of the coolant, clad and fuel. Input conditions are power, core life, pressure and temperature of the coolant at the inlet of the core, either the temperature of the coolant at the outlet of the core or the coolant mass flow and the fluences and integrated doses at the cargo area..

Using state-of-the-art neutron cross sections and transport codes, a database was created for the neutronic performance of both reactor designs. The free parameters of the models are the moderator/fuel mass ratio for the NERVA reactor and the enrichment and the pitch of the lattice for the fast reactor. Reactivity and energy balance equations are simultaneously solved to find the reactor design. Thermalhydraulic conditions are calculated by solving the one-dimensional versions of the equations of conservation of mass, energy, and momentum with compressible flow.

INTRODUCTION

As a consequence of a collaborative agreement between NASA and ORNL, we have been involved in the development of models and calculational procedures for the analysis (neutronic and thermalhydraulic) of power sources for nuclear electric propulsion.

The boundaries of the system to be modeled are the inlet and outlet plenums, and the design variables of interest are the compositions, dimensions, volumes, and masses of the core, reflector, control system, pressure vessel, and neutron and gamma shields. The thermalhydraulic conditions of the coolant are also determined. Main input variables are power, core life, burnup, pressure, and temperature of the inlet coolant and its temperature at the outlet (or the coolant flow instead).

The computer codes developed will be coupled with the calculations of the balance of plant and ion generation and acceleration to perform scoping calculations for mission analysis. It was required then to develop codes that are fast running on PCs or workstations. For that reason, although the requested parameters are quite comprehensive, the models were kept as simple as possible.

The range of input conditions specified for this task were: a power range of 10 to 50 Mw(t); operating lifetimes of 2 to 10 years; and outlet coolant temperatures ranging from 1200 to 2200 K.

TYPES OF REACTORS CONSIDERED

Two types of reactors were chosen for this analysis: the high-temperature, gas-cooled reactor of the NERVA-derivative type and the lithium-cooled, advanced fuel pin reactor, referred to as NERVA-derivative and fuel pin for short.

The elements of the core of a NERVA-derivative reactor are (a) fuel elements with ZrC cladding, a dispersion of UC-ZrC in a graphite matrix, and 19 coolant holes (diameter, 2.88 mm); and (b) support elements with ZrH_2 on an Inconel tube, with pyrolitic graphite and graphite as a thermal shield. Both elements are hexagonal with a 1.913-cm flat-to-flat distance. By changing the ratio (S/F) of support to fuel elements, the neutron spectrum can be modified considerably. The core is cooled by pumping helium, which drives a turbine in a direct Brayton cycle. The reflector is made of beryllium, which contains the control drums that rotate the control boron carbide (B_4C) sheets. The pressure vessel is located at the periphery of the reflector.

The core of the fuel pin reactor consists of bundles of rods containing uranium-nitrogen (UN) pellets. The coolant, liquid lithium, removes heat from the core to boil potassium at the secondary side of a heat exchanger which drives a turbine using a Rankine cycle. The fuel rods have a 6.4-mm outside diameter with a 0.635-mm-thick cladding of the tantalum alloy Astar. Inside the clad is a 0.122-mm-thick tungsten liner, a 0.025-mm-thick helium gap, and the UN fuel pellet, 4.786 mm in diameter. The reflector is made of BeO, which contains the control drums with the B_4C absorber. The pressure vessel is located between the core and the reflector.

Common to both reactors are the materials of the pressure vessel (Astar alloy), the neutron shielding materials [lithium hydride (LiH) in a stainless steel matrix or B_4C], and the gamma shielding material (tungsten).

NEUTRONIC CALCULATIONS

In order to find the volume of the core, a reactivity balance is performed, where the reactivity effects of the reflector, burnup, pressure vessel, structural materials, and desired reactivity at the end of life are added to the multiplication constant of the bare core. Both reactors are controlled with absorbing materials in the reflector. Thus, the excess reactivity is produced by the reflector with the bare core near critical state. Other factors like compact designs and low masses favor cylindrical shapes, with the core diameter ($2R_c$) approximately the same as the core height (H_c) (i.e., $R_c/H_c = 0.541$—the ideal shape from a neutron economy standpoint). Many of the data were then generated for a critical core of ideal shape with the transport code XSDRNPM (Greene, 1992) using a very detailed cross-section library (Ford III, 1990). A database was created for parameters used in the analysis. Criteria used in the selection of the parameters and results of the analysis are described in this section.

NERVA Derivative Database

The two free parameters of this design are the concentration of the fuel (highly enriched, 93%, ^{235}U) in the fuel element and the ratio of the number of support elements to fuel elements (S/F). A plot of the critical mass of ^{235}U and the mass of the core plus the 24-cm reflector show that beyond ~500 g of ^{235}U per liter of fuel element the total mass is not substantially reduced. It was decided to fix the fuel concentration to 500 g of ^{235}U/L; the database depends then on the ratio S/F alone. The reactivity balance is the algebraic sum of the following components of the database as a function of S/F: (a) multiplication constants, k_∞, for the infinite lattice and migration lengths, M; (b) the reactivity effects of the radial reflector; (c) the reactivity worth of the burnup, expressed as a percentage of the ^{235}U depleted (because there is a gradual change of the neutron spectra as a function of S/F, the ratio of absorption over fission rates in ^{235}U was also tabulated as a function of S/F); (d) the reactivity effects of xenon and samarium at steady state and after transients; and (e) the reactivity worth of B_4C sheets located in the control drums in the reflector.

Fuel Pin Database

The two free parameters of this design are the enrichment and the pitch of the hexagonal lattice; a possible third parameter, the diameter of the fuel rod, was fixed at 6.4 mm. The reactivity balance is the algebraic sum of the following components tabulated as a function of enrichment and the pitch of the lattice: (a) multiplication constant for the infinite lattice and critical dimensions for bare cores, (b) the reactivity worth of the BeO reflector, (c) the reactivity worth of burnup and changes in the isotopics of the actinides introduced through simple analytical models, (d) the reactivity worth of the pressure vessel located between the core and the reflector, and (e) the reactivity worth of B_4C sheets located in the reflector.

SHIELDING CALCULATIONS

The materials for the shielding are LiH in a stainless steel matrix, or B_4C, for the neutrons, and tungsten for the gammas, in a shadow shield configuration. The input parameters for the designs are acceptable values for fluences and integrated gamma doses for the entire duration of the mission at a cargo area located at z meters from the base of the reactor and with radial size r.

After its evaluation, the source term for each radiation is affected by the self shielding of the reactor materials. Removal cross sections and buildup factors are used to calculate the thickness of the shield, and the radial sizes are chosen by assuming the shield plate is at the base of the reactor and that there is no line of sight from any point of the reactor to the cargo area. Both shielding requirements were calculated together because one type of material shields the other type of radiation, though only marginally.

Neutron Shielding

For an input power, the source term of fast neutrons is calculated assuming a value of 200 Mev per fission. Self-shielding factors were calculated with simple models. Given the source term, the self-shielding, the acceptable fluence, and the removal cross sections of the LiH-ss or B_4C, the thickness of the neutron shield is fixed.

Gamma Shielding

The source term is calculated from the total number of fissions and captures—both determined by the power level. The capture reaction rate is then distributed into their material components with relative capture rates read from tables (case of NERVA derivative) or calculated with one-group capture cross sections (the case of the fuel pin). The next step is to multiply capture and fissions rates with the nine-group gamma production spectra per atom (ANL-5800, 1963). Given the source terms, the self-shielding factor and the acceptable gamma dose, the thickness of the tungsten plate is determined after using the cross sections and buildup factors from the database for the nine energy groups.

THERMALHYDRAULICS

Thermalhydraulic conditions are calculated with the approximation of 1-D equations for the conservation of mass, energy and momentum. Three steps are involved in the calculations related to the calculations of the bulk conditions of the coolant, the temperature of the wall of the channel and the temperatures within the fuel.

Thermalhydraulic variables depend on power and flow conditions. Four conditions are considered in this analysis and are described in Table 1: case 1 is the most relaxed and case 4 the most extreme; intermediate cases 2 and 3 give similar results. Note that fuel grading, cases 1 and 2, implies that reactivity penalties are not considered in the balance of reactivity.

Table 1. Power and flow conditions

Case	Power distribution		Flow distribution	Comments
	r,θ	z	r,θ	
1	Uniform	Uniform	Uniform	3-D fuel grading
2	Uniform	Cosine	Uniform	2-D (r,θ) fuel grading
3	Not uniform	Cosine	Proportional to local power	Uniform fuel, flow conditioning
4	Not uniform	Cosine	Uniform	Uniform fuel

Bulk Conditions of the Coolant

Because of the compressible flow and the high speed of the coolant, the equations of conservations have to be solved without further simplifications. When the speed of the coolant approaches sonic velocities the entire 1-D approach is no longer valid. If this condition occurs, the program prints warnings and restarts calculations relaxing inlet and power conditions. Friction factors, thermodynamics, and transport properties of the helium were taken from Melese, 1984. Considerable simplifications are possible in the case of liquid lithium coolant because the dependence of the enthalpy with the pressure is negligible as are the velocity terms.

The fuel rods are assembled in hexagonal bundles, and they are separated with helicoidal spacing wires. One result of this arrangement is that there are three types of cells with different flow fractional areas: central, lateral and at the corner. The thermalhydraulics calculations are first performed for the three cells, and then the average conditions across the bundle are calculated. For the calculation of the average pressure distribution in the bundle, we have used the Novendstern correlation (Tang, 1978), which includes the effects of the spacing wire and the different geometries of the three cells within the bundle. The temperature of the hottest point in the coolant is monitored against the saturation temperature to warn the user about potential local boiling. Thermodynamics and transport properties of lithium were taken from Jeppson, 1978.

Wall Temperature

The heat transfer coefficient for the coolant was calculated with allowances due to aerodynamics and temperature effects, including recovery factors and stagnation temperatures that allow the calculation of effective bulk coolant temperatures. Because of the excellent heat conductivity of the liquid lithium and the higher density we do not have the complications of the helium case. The correlations for the Nusselt number on pages 189, 190 of Tang, 1978 were used; the correlations depend mainly on the Peclet number, $P_e = RePr$, and are specially fitted for rods in a bundle. They are parametric in the pitch-to-diameter ratio, P/d, and in the effective ratio of the eddy diffusivity of heat to momentum, Ψ, which depends on Prandtl and Reynolds numbers and P/d.

From Wall to Fuel Temperatures

Once the wall temperature is known, the temperature distributions inside the solid fuel element are calculated by solving the 1-D (this time, the radial direction) heat conduction equation. The maximum fuel temperature is then compared with the melting temperature of the fuel to warn the user to relax the input requirements.

The coolant flows through holes in the hexagonal fuel element of the NERVA Derivative Reactor, centered in the coolant hole; and for increasing radius r, we find the wall at $r = r_0$, the fuel at $r = r_1$ and the boundary of the heat cell at $r = r_2$. Between r_0 and r_1 we have the ZrC clad, and between r_1 and r_2, a mixture of graphite and UC - ZrC; the radius of the cell, r_2, is determined by distributing the area of the mixture evenly between the nineteen holes. The temperature distribution equations are subject to the boundary conditions $T(r = r_0) = T_w$ and $(\partial T/\partial r)_{r=r_2} = 0$.

The heat conductivity *(k)* of the different materials was taken from the available literature. The temperature dependent *k(UC)* values were taken from El-Wakil, 1962. The *k* of graphite exhibits large anisotropic and irradiation effects; no intentions were made to introduce these additional complications, so average temperature-dependent values of *k* without irradiation effects were used. Graphite melts at 3889°C, above the melting temperature of the UC-ZrC mixture, which depends on the relative concentration. Melting temperatures of the UC-ZrC mixture were taken from Bussard, 1965.

Centered in the fuel rod of the Fuel Pin Reactor we have the UN fuel pin to $r = r_p$, the He gap to $r = r_c$ and the tungsten liner and astar alloy clad up to $r = r_r$. Heat conductivities were taken from (Moyers, 1987) and the heat transfer of the He gap was computed as *k(He)/* gap thickness, with *k(He)* from (Melese, 1984). This value is a simplification; very detailed discussions about these parameters can be found in (Tang, 1978). Boundary conditions are $T(r=r_r)=T_w$ and $(\partial T/\partial r)_{r=0}=0$. Melting temperatures of UN and W quoted in the program are from CRC, 1967.

ORGANIZATION OF THE CODES

In order to find a design that meets the input demands, two balance equations have to be simultaneously solved, one for the reactivity, the other for the energy. The major input variables are the reflector thickness, δ_R, the burnup, BU, the power level, *P*, the operation time, *D*, and the boundary thermalhydraulics conditions, inlet conditions of the coolant, p_i and T_i, and either its outlet temperature, T_0, or mass flow, w, and the temperature of the pressure vessel, T_{pv}.

The coolant outlet temperature or its flow is then calculated with an enthalpy balance equation neglecting the pressure drop along the reactor. If T_{pv} is not specified the code uses T_o as the temperature of the pressure vessel. T_{pv} and *D* are used then to compute σ, the maximum stress for the Astar alloy.

SELECTION AND CALCULATION OF A DESIGN

In order to write the reactivity balance equation, the worth of each component is read from tables or computed. The reactivity worth of the radial reflector, $\Delta k_{(R)}$, is read from tables. The worth of axial reflectors, if any, is computed in the same way with the additional factor of ½, which is the ratio for axial-over-radial leakages for ideally shaped cylinders.

The reactivity worth of the burnup Δk(BU) is also read from tables or computed, and the reactivity worth of the pressure vessel Δk(PV) is parameterized in terms of σ and the radius of core. Additional terms are the desired reactivity at end of life, Δk(EOL), and an estimation of the reactivity effects of structural parts, Δk(SP), which are assumed input variables.

The reactivity balance equation is then written with the dimensions of the core, R_c and H_c, as values to be determined. By specifying either ideal shape ($R_c/H_c = 0.541$) or one of the two dimensions, the dimensions of the core are determined. With R_c and H_c and the composition of the core, the total mass of fuel can then be calculated. Given this mass, the burnup and the scaling of 200 MeV per fission, the calculation of the available energy, E_a, that the reactor can release during time *D* is determined.

The shape of the available energy as a function of R_c or H_c is approximate by parabolic with a vertical assymptotic line for small R_c or H_c. In order to match the requested energy E_r = PD, the free parameters of each design are iterated until $E_a = E_r$. If this cannot be done because of contradictions between the requests and what is possible, the codes relax (under options), the input conditions.

The calculations are then continued with the computation of the number densities and masses of the core, the reflector and the pressure vessel. For the case of the fuel pin reactor, the entire process is made for P/d = 1.1, 1.2, 1.3, 1.4, and 1.5; one design is then chosen either by the user or by the code which chooses the design with the lowest total mass from those that have a pressure drop below a prescribed input value. With a design chosen, the codes proceed to calculate the geometries and masses of the neutron and gamma shieldings and the thermalhydraulic conditions for the most– and least–heated channels.

Acknowledgments

This work is the consequence of a collaborative agreement with the National Aeronautic and Space Administration (NASA), Lewis Research Center, and the Oak Ridge National Laboratory (ORNL). ORNL counterparts were B. A. Worley of the Computational Physics and Engineering Division (CPED), R. Cooper and R. Holcomb of the Engineering Technology Division (ETD), and NASA counterparts were M. Doherty and J. Gilliland from Lewis Research Center, Cleveland, Ohio. W. C. Jordan and J. C. Turner of the Computational Physics and Engineering Division calculated the database for the fuel pin reactor design.

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