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Los Alamos
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Thermal Stress Calculations for Heatpipe-Cooled Reactor Power Systems

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Abstract. A heatpipe-cooled fast reactor concept has been under development at Los Alamos National Laboratory for the past several years, to be used as a power source for nuclear electric propulsion (NEP) or as a planetary surface power system. The reactor core consists of an array of modules that are held together by a core lateral restraint system. Each module comprises a single heatpipe surrounded by 3-6 clad fuel pins. As part of the design development and performance assessment activities for these reactors, specialized methods and models have been developed to perform thermal and stress analyses of the core modules. The methods have been automated so that trade studies can be readily performed, looking at design options such as module size, heatpipe and clad thickness, use of sleeves to contain the fuel, material type, etc. This paper describes the methods and models that have been developed, and presents thermal and stress analysis results for a Mars surface power system and a NEP power source.

INTRODUCTION

Nuclear fission power systems are proposed for future space missions as a means of providing abundant power for propulsion and science, and for environments not suited to solar power (e.g. deep space). Design requirements for these missions include power levels of 3-100 kW(e) and mission times of 10 years or more. Further, these systems must minimize weight and size, have high reliability and, as much as possible, use proven materials and technology. One system that is being developed to meet these requirements is a heatpipe-cooled reactor using either UO_2 or UN as a fuel. In the heatpipe-cooled reactor concept, power produced in the core is transported to a secondary coolant system by the heatpipes, which provide multiple, parallel heat paths. Also, if desired, the heatpipes may be connected at the condenser end to multiple secondary coolant systems to enhance redundancy and reliability. The clad material may be stainless steel for lower temperature applications and in an oxidizing environment, and a refractory metal or alloy for higher temperature applications.

This paper addresses the stresses arising from differential thermal expansion at full power operations. Additional loads are imposed on the core modules during operation, due to the buildup of fission gas pressure and by the core restraint system. Also, prior to the reactor being placed into operation, the core is subjected to launch loads. As the design progresses, stresses produced by these loads will be calculated and combined with the thermal stresses presented in this paper.

THERMAL AND STRUCTURAL ANALYSIS METHOD

The analysis method that has been developed is shown schematically in Figure 1.

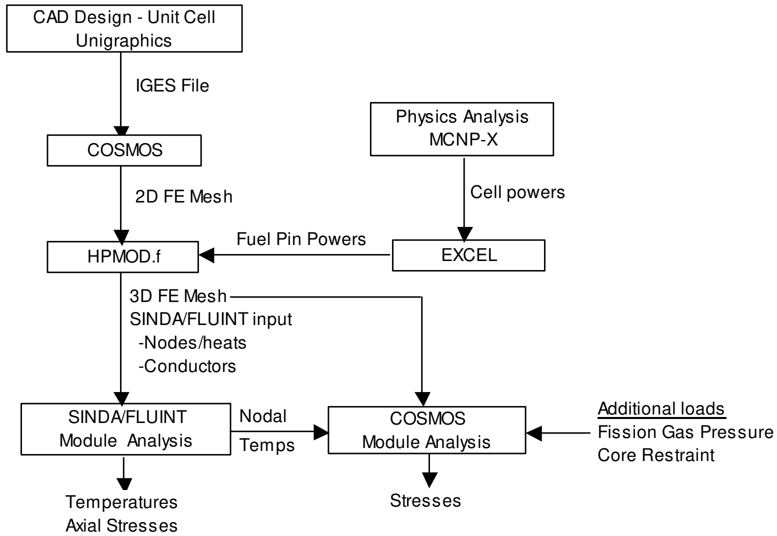

FIGURE 1. Heatpipe-Cooled Reactor Thermal/Stress Analysis.

Typically a CAD drawing of a 2D cross-section of a module (heatpipe, fuel clad and fuel only) is generated. A finite element mesh of this cross-section is then produced using COSMOS, which is input to a code written for this analysis, called HPMOD.f. The HPMOD.f code reads the finite element mesh files and adds elements for the heatpipe wick and for the heatpipe coolant that is located in the annulus between the fuel clad and wick. Options are also available to insert a sleeve between the fuel and fuel clad, and to add gap elements between the fuel and sleeve and between the sleeve and clad. The gaps may be either void (radiation heat transfer only) or filled with gas (radiation plus conduction). Also, the dimensions of the heatpipe and fuel clad (diameter, length and wall thicknesses) may be modified as desired. The 2D mesh is then converted into a 3D mesh, as shown in Figure 2.

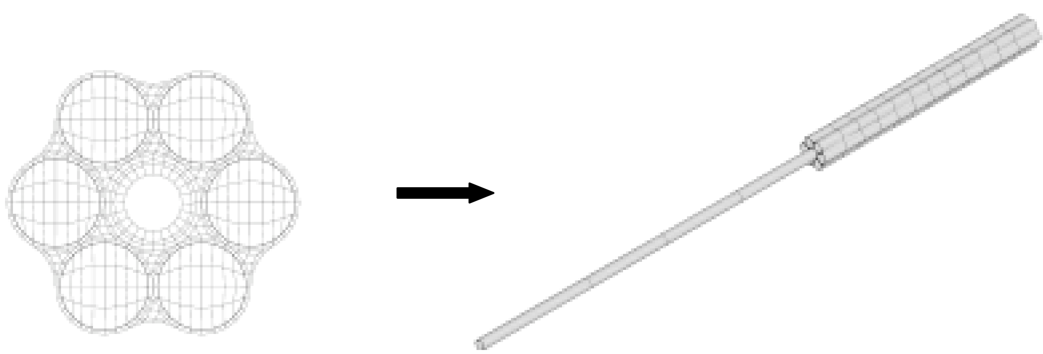

FIGURE 2. 6-Pin Heatpipe Module, 2D and 3D Meshes.

Also in HPMOD.f, node and element files are written for the 3D mesh for later use by COSMOS. The finite element mesh is converted into a finite difference nodal network; and node, conductor and power data files are generated for input to SINDA/FLUINT. The power data are taken directly from the MCNP-X physics analysis, and processed in an EXCEL spreadsheet to provide power densities for the fuel pins within each module. The following simplifications are generally made in the thermal model, in the interest of reducing the number of heat transfer branches: 1) heat transfer across the gaps and sleeve between the fuel and clad is assumed to be one-dimensional, and 2) similarly, heat transfer from cross-section to cross-section (along the length of the module) is assumed to be one-dimensional. Even with these simplifications and a relatively coarse mesh the number of nodes in this model is 13,500 and the number of conductors is 54,000.

The thermal analysis is performed using SINDA/FLUINT, and a temperature output file is generated for input to COSMOS. Both the mesh and temperature files are output by component such that non-structural materials may be eliminated for the structural analysis. A simplified axial stress analysis assuming a generalized plain condition is also performed as part of the SINDA/FLUINT analysis. Since the module thermal stresses are in large part axial, this simplified analysis provides a very good initial assessment of the module stresses (generally within 10% of the Von Mises stresses calculated in COSMOS), and these results have proved valuable in performing rapid assessments of design options as part of the conceptual design effort.

The nodal temperatures are input to COSMOS together with other loading input, and a module stress analysis is performed. As noted earlier, only the thermal loads have been considered to date. Loads imposed by the core restraint system can easily be added to the model.

MARS SURFACE REACTOR DESIGN AND ANALYSIS

Detailed descriptions of the Mars surface power system design and of the proposed heatpipe-cooled reactor are presented in companion papers to this conference (Lipinski, 2002a and Poston, 2002a). A brief description is provided here.

The Mars surface reactor design is shown in Figures 3 and 4. The core consists of thirteen 6-pin modules (heatpipe surrounded by 6 fuel pins) and six 4-pin modules (heatpipe surrounded by 4 fuel pins). The heatpipes and fuel clad are 1.59 cm (0.625") in diameter and made from SS-316. The fuel is UN and each fuel stack has a 4 cm BeO pin at each end. The fueled height is 36 cm, for a total core height of 44 cm. There are BeO pins located at each of the 6 corners of the hexagonal core array. The fuel is contained inside a 0.0127 cm (0.005") SS-316 sleeve. The gaps between the clad and sleeve and between the sleeve and fuel are filled with low-pressure helium. The core is designed to operate at 15 kW(t). The heatpipes extend 65 cm beyond the core, passing through a radiation shield and having a 20 cm condenser section at the end. The condenser section is connected to a sodium boiler, which operates at a temperature of 873 K.

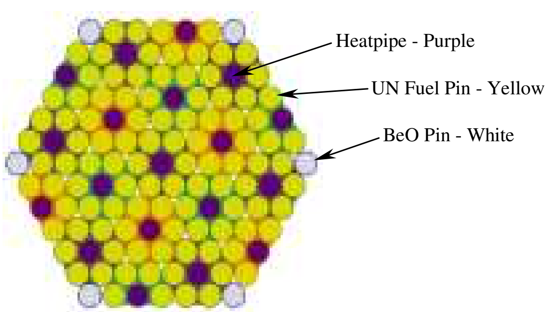

FIGURE 3. Mars Surface Reactor Layout.

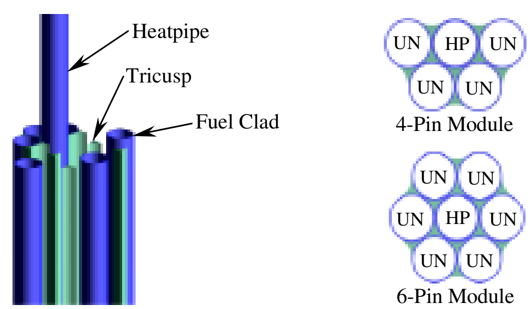

FIGURE 4. 4-Pin and 6-Pin Modules.

Initially a series of analyses was performed using the SINDA/FLUINT model, where the size of the fuel clad and heatpipes was varied from 2.38 cm (0.9375") to 1.27 cm (0.5"). Based on these results, a diameter of 1.59 cm (0.625") was selected for the Mars surface power reactor design. A core layout having 13 6-pin modules and 6 4-pin modules was developed, using this diameter of fuel clad and heatpipe. A COSMOS stress analysis was performed for this design. Summary results from the SINDA/FLUINT and COSMOS stress analyses are presented in Table 1. Temperature and stress contour plots from the COSMOS analysis are shown in Figures 5 and 6. Note that the fuel, heatpipe wick and coolant have been removed from the stress plot.

TABLE 1. Analysis Results for Mars Surface Reactor.

Result	6-Pin Modules				4-Pin Modules			
Fuel Clad Diameter (cm)	2.38	1.91	1.59	1.27	2.38	1.91	1.59	1.27
Heatpipe Diameter (cm)	2.38	1.91	1.59	1.27	2.38	1.91	1.59	1.27
Peak Clad Temp (K)	939	919	908	898	932	914	904	895
Ultimate Strength (Mpa)	283	301	309	317	290	305	312	319
Yield Strength (Mpa)	110	111	112	112	110	112	112	112
SINDA/FLUINT								
Axial Stress (Mpa)	103	70	51	35	109	76	58	42
COSMOS								
Axial Stress (Mpa)			56				62	
Von Mises Stress (Mpa)			50				59	

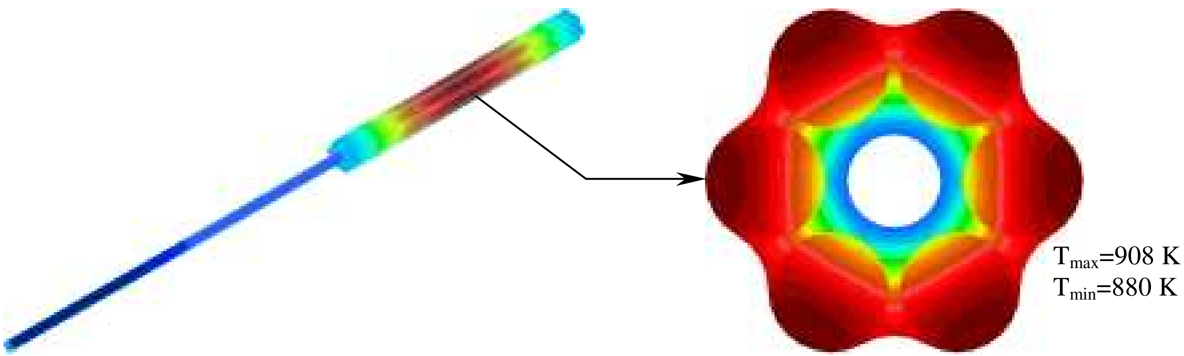

FIGURE 5. Mars 15 kW(t) Reactor Temperature Contours.

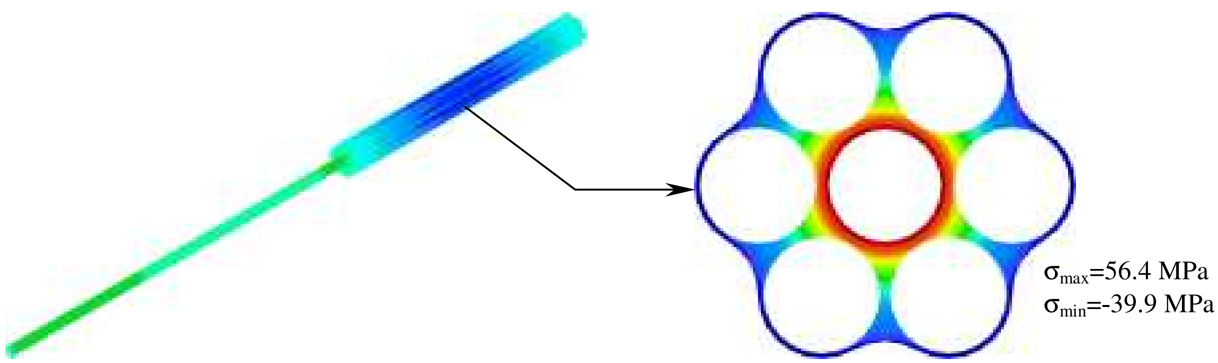

FIGURE 6. Mars 15 kW(t) Reactor Stress Contours.

The selected design produces peak secondary stresses that are about 50% of yield and 20% of ultimate strength. This is a very conservative design, especially considering fuel burnup is only about 0.04% and peak fast neutron fluence is less than $4x10^{20}$ n/cm^2 (Amiri, 2002). At these irradiation conditions, there is essentially no fission gas pressure buildup, the radiation effects in this core are negligible and the material retains its ductility. More detailed analysis needs to completed, with finer meshes and including combined loads and design basis events, but this design offers promise of enhanced performance (eg. higher core power).

As seen in Table 1, the axial stresses calculated in SINDA/FLUINT are in good agreement with those calculated using COSMOS. Further, the peak axial stresses calculated by COSMOS are about the same magnitude as the Von Mises stresses. This is due to the fact that the modules are long and slender and in-plane temperature gradients are large compared to the axial temperature gradients. As a result, in-plane stresses are low compared to axial stresses and the use of generalized plane strain proves to be a good approximation.

Stresses in the 4-pin modules are higher than those in the 6-pin modules, despite the fact that these modules are located at the core boundary and have lower power densities than the 6-pin modules in the center of the core. The reason for this is the asymmetry in the 4-pin modules produces elevated stresses in these modules. Because the 4-pin modules tend to bow, a second calculation was performed where these modules were constrained to be straight. Stresses for these cases were somewhat lower than if the modules were allowed to bow.

NEP REACTOR DESIGN AND ANALYSIS

Detailed descriptions of the NEP power source system design and of the proposed heatpipe-cooled reactor are presented in companion papers to this conference (Lipinski, 2002b and Poston, 2002b). A brief description is provided here.

The NEP reactor design is shown in Figures 7 and 8. The core consists of 127 3-pin modules, whose heatpipe and fuel pin diameters vary from 1.1-1.2 cm (0.435”-0.470”). Every other pin at the core/side reflector boundary is a BeO pin. Because this reactor is designed to operate at very high temperatures (peak coolant temperature of 1150 K or higher), refractory metals are being considered for the core structural material (both heatpipes and fuel clad). Table 2 presents four candidate designs.

TABLE 2. NEP Reactor Design Options.

Material	**UN/Nb-1Zr**	**UN/Mo**	**UO_2/Mo**	**UO_2/Mo-44Re**
Fuel	UN	UN	UO_2	UO_2
Clad/Heatpipe	Nb-1Zr	Mo	Mo	Mo-44Re
Sleeve	Re	Re	None	None
Heatpipe Wick	Nb-1Zr	Mo-44Re	Mo-44Re	Mo-44Re
Working Fluid	Na	Na	Na	Na
Dimension	**UN/Nb-1Zr**	**UN/Mo**	**UO_2/Mo**	**UO_2/Mo-44Re**
Clad/HP Diameter (cm)	1.105	1.105	1.143	1.194
HP Thickness (cm)	0.127	0.127	1.016	1.016
Clad Thickness (cm)	0.076	0.076	0.071	0.076
Sleeve Thickness (cm)	0.051	0.051		
Clad Gap (cm)	0.005	0.005	0.005	0.006
Sleeve Gap (cm)	0.003	0.003		
Fueled Height (cm)	56	50	56	55
Core Height (cm)	64	58	64	65

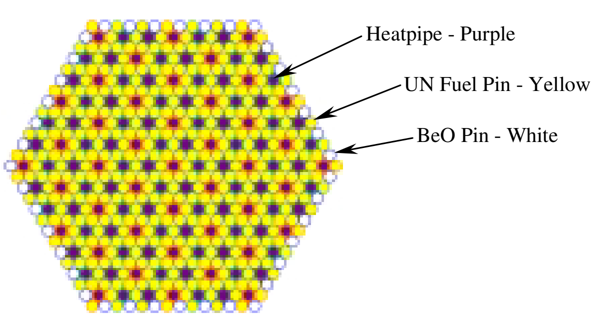

FIGURE 7. NEP Reactor Layout.

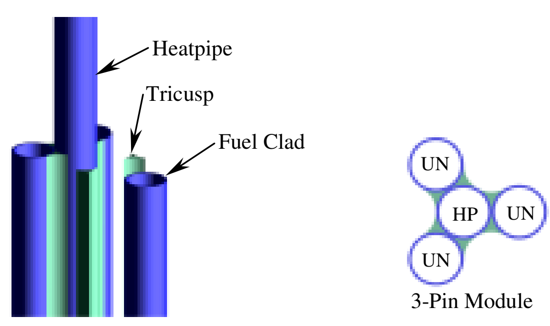

FIGURE 8. NEP Reactor, 3-Pin Module.

The gaps between the clad and sleeve and between the sleeve and fuel are filled with low-pressure helium. The core power for these designs is 400 kW(t), with the coolant temperatures of 850 K at inlet and 1150 K at exit. The heatpipes extend 54 cm beyond the core, allowing for a 50 cm condenser section. The condenser section is connected to a heat exchanger, which transfers heat to a Brayton cycle power system.

Temperature and stress contour plots for the UN-Mo design are shown in Figures 9 and 10. Contour plots for the other designs are similar. Summary results for all cases are presented in Table 3. The calculated stresses in all these designs are below both yield and ultimate strength of the materials. The Nb-1Zr clad design has the lowest stress, in large part because this material has a low modulus of elasticity and thus a low product of modulus of elasticity times coefficient of thermal expansion (E*CTE). Molybdenum has the highest thermal conductivity of the materials considered, and this is reflected in the lower temperature gradients for these cases. Stresses are higher than in Nb-1Zr because its E*CTE product is almost twice that of Nb-1Zr. The designs using UO_2 fuel have higher temperatures and stresses because the thermal conductivity of UO_2 fuel is very low, only about 1/10 that of UN at 1250 K. The lower conductivity fuel causes the heat from the fuel to be more uniformly distributed to the fuel clad (less skew towards the cold heatpipe), increasing temperature gradients in the clad. Finally the Mo-44Re design has relatively high stress because its thermal conductivity is about equal to that of Nb-1Zr (and about 2/3 that of pure molybdenum), but it has a E*CTE product about equal to that of molybdenum.

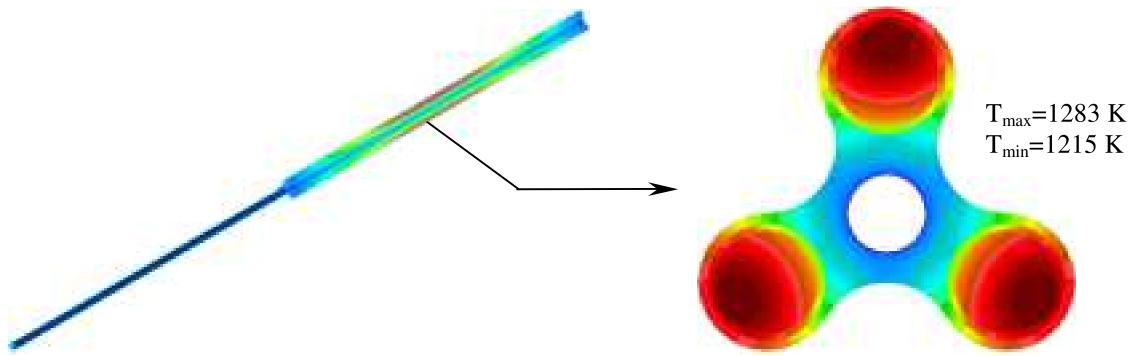

FIGURE 9. NEP 400 kW(t) Reactor Temperature Contours.

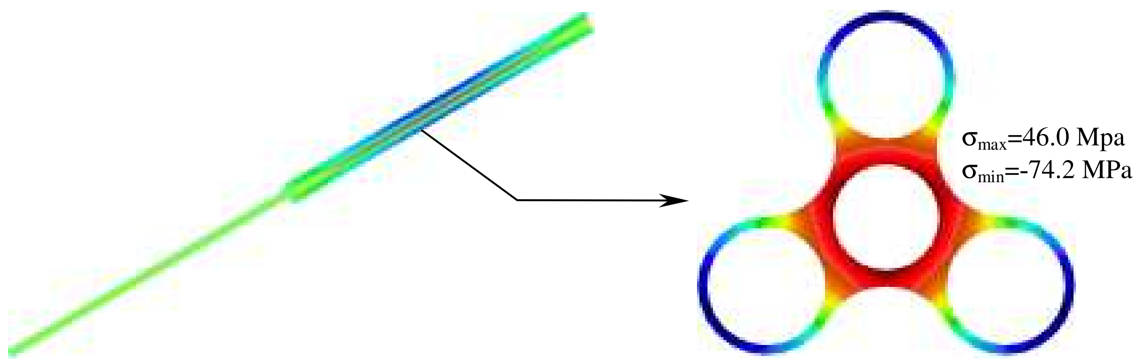

FIGURE 10. NEP 400 kW(t) Reactor Stress Contours.

TABLE 3. Analysis Results for NEP Power Source Reactor.

Result	UN/Nb-1Zr	UN/Mo	UO_2/Mo	UO_2/Mo-44Re
Peak Clad Temp (K)	1293	1283	1303	1321
Ultimate Strength (Mpa)	158	321	308	399
Yield Strength (Mpa)	68	277	265	284
Max Axial Stress (Mpa)	27.7	46.0	55.7	93.1
Min Axial Stress (Mpa)	-41.5	-74.2	-105.7	-179.4
Von Mises Stress (Mpa)	41.3	73.8	105.1	177.7

CONCLUSIONS

Design methods have been developed to perform thermal and stress analysis the heatpipe-cooled reactors envisioned for space applications. These methods have been used to perform analysis for two proposed applications, a 15 kW(t) Mars surface power reactor and a 400 kW(t) NEP power source. As the design progresses, these methods will be used to assess design options and establish reactor performance vis-à-vis design requirements. These assessments will include consideration of combined loads and design basis events (e.g. failed heatpipe, failed fuel clad, failed fuel sleeve), and will account for radiation effects on material properties and design allowables.

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