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Design Parameters for a Natural Uranium UO_3- or U_3O_8-Fueled Nuclear Reactor

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EXECUTIVE SUMMARY

It is well known that natural uranium can be used to fuel both low- and high-power reactors. Historically, reactors such as the U.S. Oak Ridge Graphite Reactor and the Canadian CANDU reactors have been fueled with natural uranium in the pure metal (U) and dioxide (UO_2) forms, respectively. This report conservatively demonstrates conceptual design parameters for a low-power nuclear reactor that is fueled with 3 metric tons of natural uranium (MTU) in the form of 80% of theoretical density uranium trioxide (UO_3) powder as contained within 400 aluminum tubes/rods. The tubes/rods are 3-m long by 2.54 cm in diameter that are placed in a square pattern with 15.0-cm center-to-center spacing. To be an effective fuel, the UO_3 must be produced by thermal denitration of uranyl nitrate, $UO_2(NO_3)_2$, that has been purified via the standard solvent extraction process used prior to the production of power-reactor-grade UO_2 nuclear fuel. Additionally, the UO_3 powder must be maintained in a nearly moisture-free environment from the thermal denitration to tube/rod fabrication to avoid the evolution of moisture during low-power reactor operation. In excess of 66 metric tons of relatively pure heavy water (i.e., 0.1 wt % H_2O and 99.9 wt % D_2O) is needed as the primary coolant and neutron moderator/reflector for the reactor. Additional neutron reflection is provided by 320 metric tons of concrete. The reactor heat removal is provided by relatively low-velocity [i.e., 12.5-cm/s (5-in./s)] coolant/moderator circulation [i.e., 1 m^3/s (283 gal/s)] through the 2.88 MTU fueled core region having a limited power output of ~7.2 MW total or ~2.5 MW/MTU. The reactor power limitation is required to maintain the UO_3 tube/rod centerline temperatures to between 320°C and 450°C, above which production of triuranium octaoxide, U_3O_8, becomes pronounced. The reactor power would be regulated with primitive control rods/plates. To maintain this temperature limitation, inlet and outlet coolant/moderator temperatures are required to be 27 and 30°C, respectively. Depending upon the design of the heat-exchanging coolant loop, an additional 30 to 60 metric tons of heavy water external to the reactor core region may be needed, thereby requiring a total of 100 metric tons of heavy water for the reactor design.

The heat transfer analysis was performed assuming a uniform heat distribution throughout the core. Using a more realistic, non-uniform heat distribution analysis and increased coolant/moderator flow would permit the UO_3 reactor power to be increased substantially, to perhaps as much as 11 MW. The above conceptual design parameters for UO_3 powder fuels are also applicable to U_3O_8 powder fuels. When fueled with U_3O_8 powder and with an increased coolant flow, however, the 450°C fuel centerline temperature limitation could be substantially relaxed because the melting and decomposition temperatures of U_3O_8 are 1150 and 1300°C, respectively, versus a decomposition temperature for UO_3 between 400 and 600°C.

It is judged that material compatibilities (i.e., aluminum, minimal water, and oxygen evolution from UO_3) are of little concern for the limited reactor operation needed to create 2 kg of plutonium having less than 5 wt % ^{240}Pu [i.e., 2740 total megawatt-days (MWd) per 2.88 MTU, or 951 MWd/MTU]. Throughout this operation period, very little available excess neutron multiplication is lost from the reactor due to fission product production and no meaningful decomposition of UO_3 will occur.

The only uncertainty associated with building and operating this conceptually designed reactor is the accessibility of the necessary materials and standard industrial design/construction skills. The required materials include commercially available high-purity aluminum tubing/sheets, UO_3 or U_3O_8 powder, 99.9 wt % D_2O, high-volume/rate water pumps, heat exchangers, commercially available radiation monitoring and electronic equipment/circuits, electromechanical and pneumatic drive/controls, and.

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ABSTRACT

A recent Oak Ridge National Laboratory report provided preliminary analyses to propose alternative design parameters for a nuclear reactor that could be fueled with natural UO_3 or U_3O_8 and moderated with either heavy water or reactor-grade graphite. This report provides more specific reactor design and operating parameters for a heavy water–moderated reactor only. The basic assumptions and analytical approach are discussed together with the results of the analysis.

1. INTRODUCTION

A preliminary study was performed at ORNL[1] to determine the technical feasibility of designing a nuclear reactor using natural uranium oxide of nuclear material compositions (UO_3 or U_3O_8) suitable for fuel fabrication and nuclear reactor operation. The oxide compositions considered have the purity of uranyl nitrate source nuclear material, $UO_2(NO_3)_2$, as processed by the solvent extraction method that is used prior to the production of reactor-grade UO_2 powder or the conversion to UF_6 gas for enrichment purposes. This report provides more specific information regarding the details and limitations for nuclear fuel fabrication, reactor design, and stable operation of a heavy water–moderated nuclear reactor. This reactor is fueled with purified natural uranium, as contained in thin aluminum tubes, in a lattice configuration and in the form of UO_3 powder at 80% of theoretical density UO_3. The development of this more specific information was based upon standard and publicly available nuclear, heat transfer, and heat transfer engineering computational analytical tools.[2,3] The basic fabrication, design, and operational assumptions and analytical approaches are discussed together with the results of the analysis.

2. REACTOR CONFIGURATION AND OPERATION

The heavy-water-moderated reactor core is a 3-m (9.62-ft) cube of 400 UO_3-filled aluminum fuel tubes. Each tube is about 3-m (9.62-ft) long with an inner diameter (ID) of 2.54 cm. The 400 tubes are placed vertically on a 14.64-cm (5.76-in.) square pattern. Each tube is assumed to be vibrationally packed with UO_3 to a density of 5.83 g UO_3/cm^3, or 80% of theoretical density of UO_3. This results in approximately 9 kg UO_3 per tube or about 3.5 metric tons of UO_3 for the reactor core or 2.88 metric tons of uranium (MTU). If determined to be necessary, the tubes could be extended with an evacuated plenum region or exhaust manifold to accommodate any off gassing. The 3-m (9.62-ft) cube core region is surrounded by a 0.159-cm (1/16-in.) thick aluminum baffle/containment, which is then surrounded by a composite reflector of 0.5-m (1.64-ft) thick heavy water and 1.0-m (3.28-ft) concrete with top and bottom openings to accommodate vertical moderator/coolant flow through the reactor. An external-to-core heat exchanger is provided to remove heat from the flowing moderator/coolant.

Reactor power and coolant operating parameters were based upon heat transfer analyses that assumed uniform power densities throughout the core with the constraint that no UO_3 fuel could exceed 450^o C. It was determined that a moderator/coolant flow rate and velocity of 17,000 gal/min and 12.5 cm/s (5 in./s), respectively, at inlet and outlet temperatures of 27^{o}C and 30^{o}C, respectively, would assure that the specified fuel temperature would not be exceeded for a uniformly distributed reactor core power of 15 MW throughout the 2.88 MTU (i.e., 2.47 MW per MTU). Subsequent to the heat transfer analysis, the actual core power density distribution was determined to have an actual peak-to-average power density ratio of the core is 2.09. Because of this difference between the heat transfer analysis assumptions and the determined reactor power distribution it was judged that the design power for the reactor should be constrained to ~7.2 KW (i.e., 2.5 MW/MTU). With more realistic heat transfer modeling assumptions and the use of U_3O_8, the power rating of the reactor could be increased substantially, perhaps by a factor of 10 or more.

Though not specifically evaluated, given the low moderator/coolant flow rates, the reactor control can be managed with insertion of very primitive control rods/blades into the core.

Specific parametric information for this study is provided in the following sections.

3. MATERIALS

Reference 1 provides the background for potential nuclear material fuel fabrication uses of UO_3 or U_3O_8, as derived from the solvent extraction process that is used for purifying natural uranium in the form of uranyl nitrate, $UO_2(NO_3)_2$. That reference also provides some historic perspective on the past designs and operation of heavy-water-moderated reactors (e.g., CANDU) and graphite-moderated reactors (e.g., the Oak Ridge Graphite Reactor), which were fueled with natural uranium dioxide and metal, respectively. This follow-up report focuses on a heavy-water (D_2O)-moderated reactor design that is fueled with uranium oxides, specifically UO_3. As mentioned in Ref. 1, it is quite realistic to assume that this same D_2O–moderated reactor configuration can be fueled with other natural uranium nuclear material compositions (i.e., U_3O_8, UO_2, UC, UF_4, metal, and uranium-beryllium alloys) having similar uranium purity. The primary limitations for the use of UO_3 and UF_4 as natural uranium nuclear material fuels are their form (e.g., loose powder, pressed and/or sintered powder) and their chemical stability at elevated temperatures. As a result, either the reactor power and resulting fuel temperature must be maintained below the thermal decomposition/melting temperature of the various compounds or the fuel tubes/cladding must be vented to accommodate any gases that may evolve from decomposition. The following describes the materials that are used in this study.

The material model for this study (UO_3) presumes a nuclear reactor power sufficiently low to maintain the UO_3 well below the 450^{o}C decomposition/reduction temperature that initiates in the transformation to U_3O_8.

3.1 FUEL

The nuclear material composition for the assumed fuel fabrication is uranium trioxide (UO_3) powder such as obtained with a rotary thermal denitration kiln. In the presence of oxygen, uranyl nitrate, $UO_2(NO_3)_2$, as received from the chemical solvent extraction process that renders reactor-purity uranium,[4] is admitted to the kiln and dry UO_3 is discharged. This process of thermal denitration to UO_3 occurs at temperatures between 400 and 600^{o}C. The same denitration process of $UO_2(NO_3)_2$ at temperatures between 650 and 800^{o}C yields U_3O_8, another nuclear material composition suitable for fuel fabrication and use in the proposed reactor design considered here. It is assumed that the discharged UO_3, and incidentally produced U_3O_8, are maintained in dry ambient conditions throughout the fuel fabrication process. The assumed density is 5.83 g UO_3/cm^3, or 80% of the UO_3 theoretical density. This density value, which exceeds typical pour densities of powders (i.e., 25 to 50% of theoretical densities) is achieved via vibrational tamping of the material within the fuel tube/cladding. The total mass of UO_3 used in the core of the reactor is 3.46 metric tons, or 8.65 kg UO_3 for each of the 400 fuel tubes having 2.93-m lengths of fuel region per tube.

3.2 CLADDING

The UO_3 nuclear material powder used for the fuel fabrication is clad, or contained, within 2.54-cm-ID (1-in.-ID) commercial-grade Aluminum 1199-O metal tubing having a 0.159-cm (1/16-in.) wall thickness. The aluminum has a density of 2.7 g Al/cm^3. The total mass of aluminum tubing within the reactor core region is 425.9 kg Al, or 1.07 kg Al for each of the 400 fuel tubes of 2.928-m length within the reactor core.

3.3 CORE MODERATOR/COOLANT

Heavy water is used as the neutron moderator and coolant for the reactor. The heavy water has an assumed purity of 99.9 wt % D_2O and 0.1 wt % H_2O. The assumed density is 1.105 g/cm^3. The total mass in the reactor core is 26.9 metric tons.

3.4 CORE MODERATOR/COOLANT BAFFLE/CONTAINMENT

The 2.928-m cubed core moderator/coolant is contained within a commercial-grade Aluminum 1199-O metal sheeting having a 0.159-cm (1/16-in.) wall thickness. This baffle/containment is used to direct coolant flow past the heavy water reflector region. The aluminum has a density of 2.7 g Al/cm^3. The total mass of the aluminum baffle/containment around the reactor core region is 234.8 kg Al.

3.5 COMPOSITE REFLECTOR

The reactor core region is surrounded by and in contact with a composite reflector that comprises of an inner 0.5-m thickness of heavy water and an outer 1.0-m thickness of Magnuson concrete, as described in Ref. 2. The heavy water has an assumed purity of 99.9 wt % D_2O and 0.1 wt % H_2O. The inner reflector region has a density of 1.105 g/cm^3, and the outer reflector region has a density of 2.15 g of Magnuson concrete per cubic centimeter. The total mass of the inner reflector region is 39.3 metric tons of heavy water. The total mass of the outer reflector region is 317.6 metric tons of Magnuson concrete.

3.6 SUMMARY OF REACTOR MATERIALS

Table 3.1 provides a balance of materials for this specific design.

Table 3.1. Summary of reactor materials

Material	Composition	Thickness (cm)	Density (g/cm^3)	Mass (metric tons)
Fuel	UO_3 powder	-	5.832	3.461
Cladding	Aluminum metal tubes	0.159	2.700	0.426
Moderator/coolant	Heavy water	-	1.105	26.900
Core baffle/containment	Aluminum metal sheeting	0.159	2.700	0.235
Composite reflector				
Inner region	Heavy water	50.000	1.105	39.300
Outer region	Concrete	100.000	2.147	317.600
External to core moderator/coolant	Heavy water	-	1.105	30.000

4. NEUTRONIC ANALYSES

4.1 COMPUTATIONAL METHODS

As described in Ref. 1, the SCALE code system[2] was used to perform the nuclear analysis. The CSAS/KENO analysis sequence from SCALE was used to perform three-dimensional static physics analyses to establish the minimum natural uranium mass for a heavy-water moderated reactor configuration with sufficient excess reactivity to maintain reactor operations throughout a selected irradiation and to determine the general power distribution throughout the core. The reactor fission product and fissionable nuclear material isotopic production and decay were simulated using the SAS2H/ORIGEN-S sequence of SCALE. This sequence uses an approximate neutronic model to provide cross-sectional information for the depletion, decay, and production portion of the analysis. The SAS2H/ORIGEN-S sequence has been demonstrated to be a valid method for predicting reactor core inventories of fission products and fissionable nuclear material production within commercial and research reactors.

4.2 COMPUTATIONAL RESULTS

4.2.1 Neutron Multiplication

Using the preliminary results of Ref. 1 as a starting point, the CSAS/KENO sequence was used to determine the nearly optimum 14.64-cm (5.76-in.) tube-to-tube square-pitch arrangement within the heavy-water moderator that yielded the highest neutron multiplication factor for an infinite system (k_{inf}); see Fig. 4.1. The fuel tube diameter was then varied to determine the nearly optimum 2.54-cm (1-in.) UO_3 fuel diameter while maintaining the interstitial moderator thickness associated with the nearly optimum tube-to-tube pitch previously obtained; see Fig. 4.2. Throughout the optimization searches, the fuel was maintained at 80% of the theoretical density of UO_3, or 5.832 g UO_3/cm^3. Using these optimum values for fuel rod size and spacing, another series of CSAS/KENO calculations was performed to determine the size of the reactor lattice (i.e., the number of fuel tubes) that would provide an effective neutron multiplication factor (k_{eff}) such that the initial excess reactivity ($k_{eff} - 1$) would enable operation of the reactor by compensating for fission product and actinide build up. The resultant lattice of 400 fuel tubes (20 tubes $\times$ 20 tubes) provides an excess reactivity of ~0.063) for a uniform fuel temperature of 427^{o}C and a uniform D_2O temperature of 30^{o}C. A reduction in the fuel density to 70% of the theoretical density, or 5.103 g UO_3/cm^3, reduces the excess reactivity to ~0.040. The excess reactivity changes negligibly throughout a reactor operation equivalent to 2740 megawatt-days (MWd) per the 2.88 MTU in the core. This integrated power is sufficient to produce 1.98 kg of plutonium with < 5 wt % ^{240}Pu, a *Special nuclear material of moderate strategic significance*.[5]

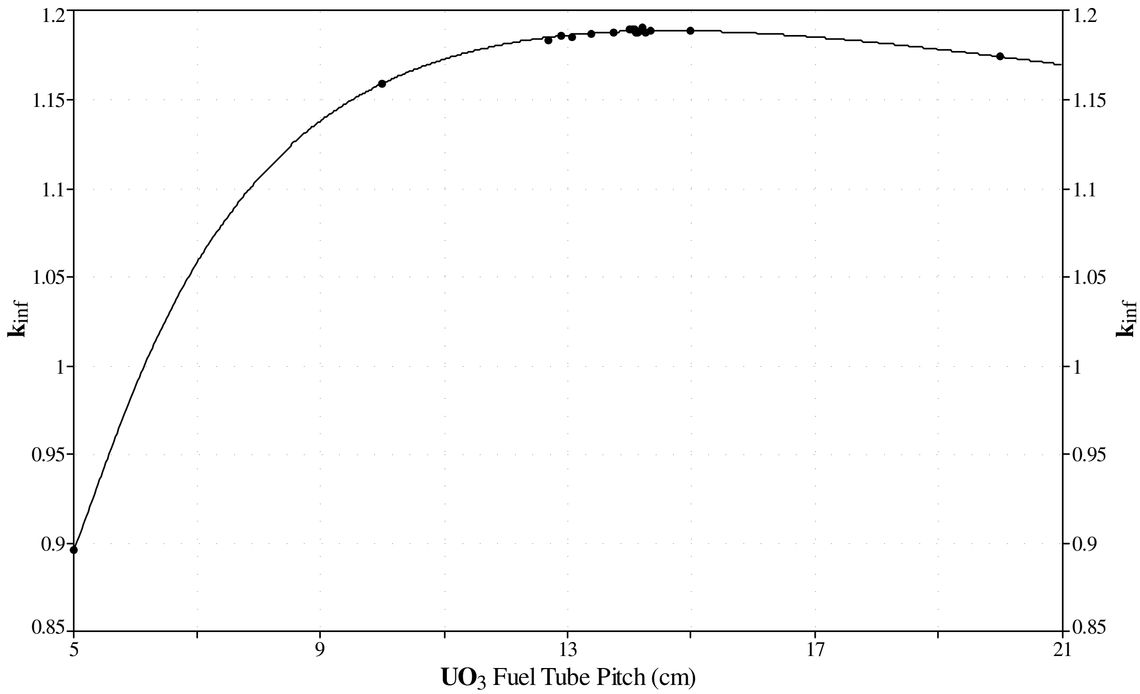

Fig. 4.1. Variation of k_{inf} with UO_3 fuel tube pitch.

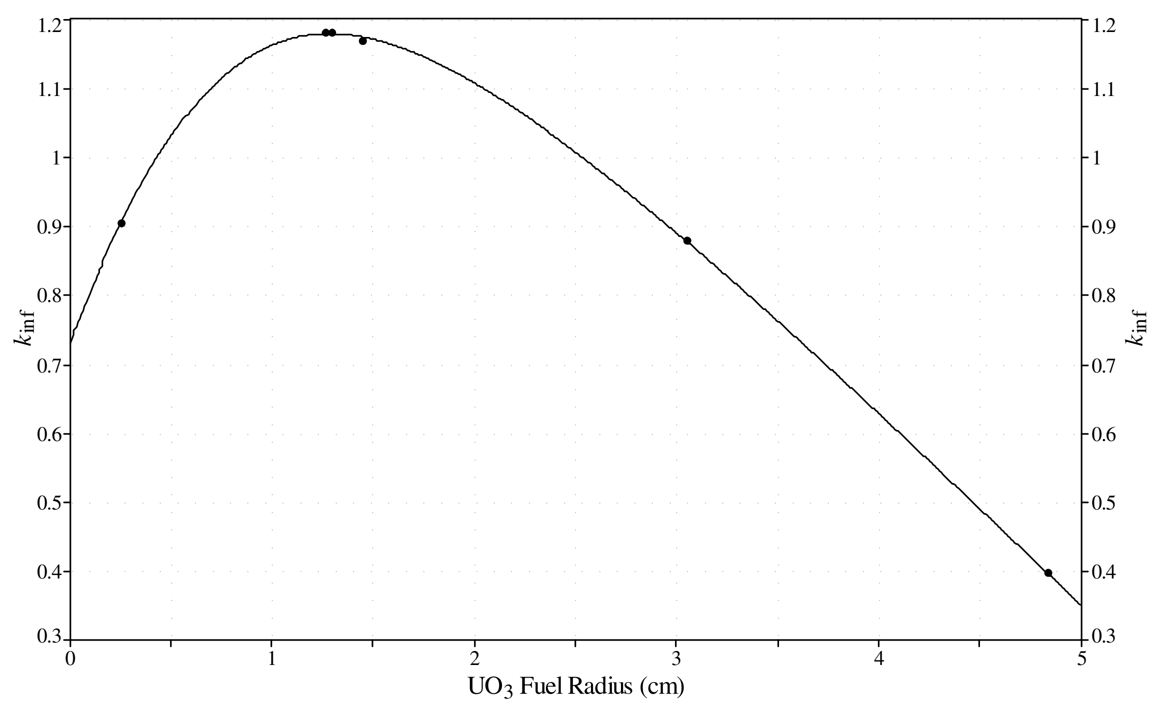

Fig. 4.2. Variation of k_{inf} with UO_3 fuel radius.

4.2.2 Fission Product Nuclide and Special Nuclear Material Production and Decay

The reactor fission product and fissionable nuclear material isotopic production and decay were simulated using the SAS2H/ORIGEN-S sequence of SCALE. It was necessary to calculate these inventories to determine the impact on the core neutron multiplication constant, k_{eff}. It was determined that after 381 days of the proposed reactor operation at ~7.2 MW (2740 MWd) with its 2.88 MTU core (951 MWd/MTU), the quantities of fission products and fissionable nuclear materials that are produced in the proposed reactor have negligible effect (<< 1% in k_{eff}) on the core reactivity. Additionally, there is sufficient excess reactivity (i.e., ~ 6% k_{eff}) to compensate for the xenon that is produced. The quantity of fission products and fissionable nuclear materials (e.g., 1.98 kg Pu) produced in the proposed reactor are equivalent to those produced in a CANDU reactor (i.e., Pickering-7) after about 1.8 days of operation at 15 MW.

4.2.3 Fission Power Density Distribution

Using CSAS/KENO, the reactor core was modeled with 8000 (20 × 20 × 20) cubic cells, 14.64 cm (5.76 in.) on a side, which contained a 14.64-cm-long (5.76-in.-long) segment of a UO_3-fueled aluminum tube, surrounded by heavy water, centered within the cubic cell. The cells were stacked to construct the 20 × 20-core array model of 2.928-m-long (9.61-ft-long) fuel tubes. Individual cell fission densities were monitored to develop the fission density distribution throughout the core. This fission density distribution essentially provides the relative fission power distribution for the reactor when it operates at any power. Figures 4.3 and 4.4 provide graphical representations of this relative power density distribution obtained through the use of KENO 3D.[6]

Figure 4.3 provides a cutaway view of the core, which depicts the most extreme variation in fission power densities from the central region to the far corners of the core. The peak power density at the center of the core divided by the minimum power density at the extreme corners of the core equals 11.2.

Figure 4.4 illustrates the horizontal power density distribution through the vertical center of the core. The peak power density at the center of the core divided by the minimum power density at the center of the fuel tubes near the vertical faces of the core boundary is 2.3. The peak-to-average core power distribution was determined to be 2.09.

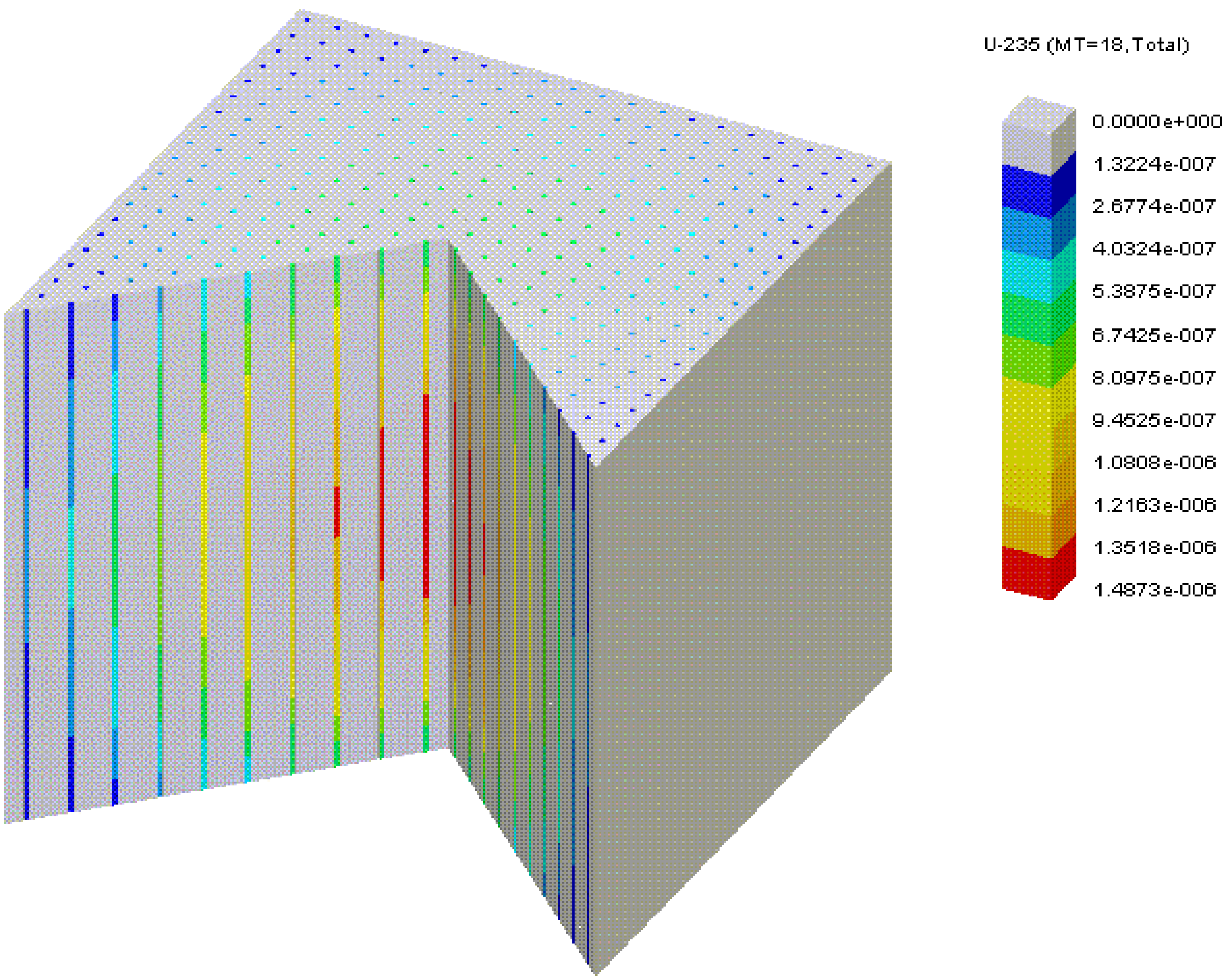

Fig. 4.3. Relative fission power density of core.

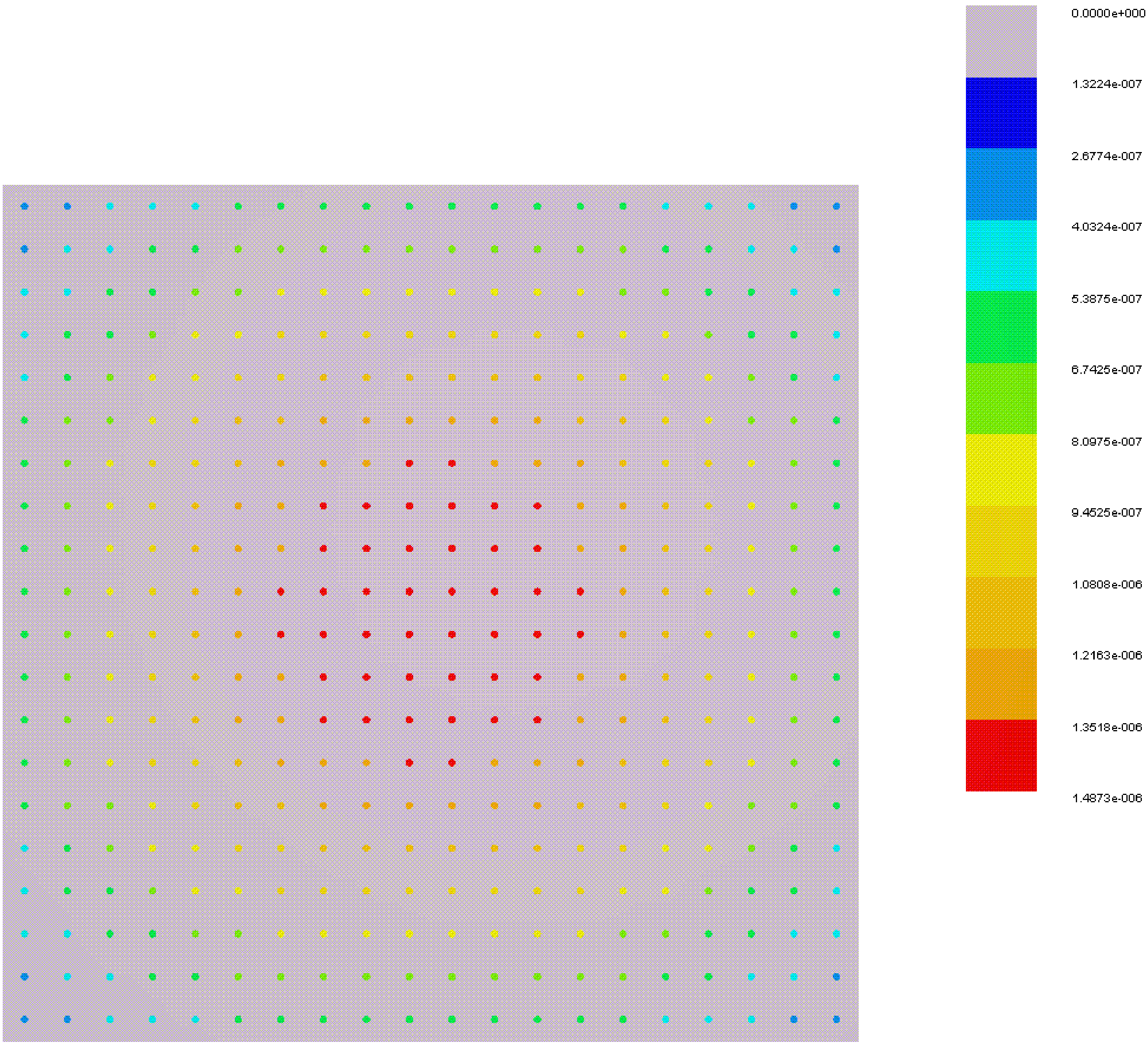

Fig. 4.4. Horizontal relative fission power density through vertical center of core.

5. HEAT TRANSFER ANALYSES

The heat transfer analyses were based upon the material and geometry specifications described in Sects. 2 and 3. However, because the fission/power density distribution was not available at the onset of these analyses, the following simplistic bounding parametric assumptions were specified:

- The fission power distribution was uniform throughout the core fuel.
- The temperature of UO_3 was not to exceed 450°C.
- The core coolant/moderator must remain single phase (not even subcooled nucleate boiling) should occur.
- The core inlet temperature of moderator/coolant was to be 27°C.
- The core outlet temperature of the moderator/coolant was not to exceed 95°C.

The core heat transfer analyses considered a matrix of core power (from 15 to 90 MW) and core fluid outlet temperatures (from 28 to 90°C). The resulting core mass flow rates are given in Table 5.1.

Table 5.1. Core flows for assumed core power and core outlet temperature matrix

Outlet temp. (°C)	Mean C_p* (J/kg•°C)	Core flow (kg/s) for assumed values of core power					
		15 MW	30 MW	45 MW	60 MW	75 MW	90 MW
28	4,226.0	3,549.46	7,098.91	10,648.37	14,197.82	17,747.28	21,296.73
30	4,225.5	1,183.29	2,366.58	3,549.88	4,733.17	5,916.46	7,099.75
40	4,222.2	273.28	546.56	819.84	1,093.12	1,366.40	1,639.68
50	4,217.2	154.65	309.29	463.94	618.58	773.23	927.88
60	4,211.4	107.93	215.86	323.80	431.73	539.66	647.59
70	4,205.3	82.95	165.90	248.86	331.81	414.76	497.71
80	4,201.2	67.37	134.73	202.10	269.46	336.83	404.20
90	4,197.1	56.73	113.46	170.19	226.91	283.64	340.37

* Heat capacity of coolant.

The resulting core inlet volumetric flow rates and core inlet velocities are presented in Tables 5.2 and 5.3.

Table 5.2. Core volumetric flow rates for assumed core power and core outlet temperature matrix

Outlet temp. (°C)	Core flow (gal/min)* for assumed values of core power					
	15 MW	30 MW	45 MW	60 MW	75 MW	90 MW
28	50,899	101,798	152,697	203,596	254,494	305,393
30	16,968	33,937	50,905	67,873	84,842	101,810
40	3,919	7,838	11,756	15,675	19,594	23,513
50	2,218	4,435	6,653	8,870	11,088	13,306
60	1,548	3,095	4,643	6,191	7,739	9,286
70	1,190	2,379	3,569	4,758	5,948	7,137
80	966	1,932	2,898	3,864	4,830	5,796
90	813	1,627	2,440	3,254	4,067	4,881

*Multiply gal/min by 3.785 to obtain liters per minute.

Table 5.3. Core inlet velocities for assumed core power and core outlet temperature matrix

Outlet temp. (°C)	Core fluid velocity (m/s) for assumed values of core power					
	15 MW	30 MW	45 MW	60 MW	75 MW	90 MW
28	0.3861	0.7722	1.1583	**1.5444**	**1.9305**	**2.3166**
30	0.1287	0.2574	0.3861	**0.5149**	**0.6436**	**0.7723**
40	0.0297	0.0595	0.0892	**0.1189**	**0.1486**	**0.1784**
50	0.0168	0.0336	0.0505	**0.0673**	**0.0841**	**0.1009**
60	0.0117	0.0235	**0.0352**	**0.0470**	**0.0587**	**0.0704**
70	**0.0090**	**0.0180**	**0.0271**	**0.0361**	**0.0451**	**0.0541**
80	**0.0073**	**0.0147**	**0.0220**	**0.0293**	**0.0366**	**0.0440**
90	**0.0062**	**0.0123**	**0.0185**	**0.0247**	**0.0309**	**0.0370**

The pin power, linear heat generation rates (LHGRs), and fuel pin surface heat flux are provided in Table 5.4.

Table 5.4. Pin power, linear heat generation rate, and surface heat flux relative to core power

Core power (MW)	Pin power (kW)	LHGR (kW/m)	LHGR (kW/ft)	Pin surface heat flux (W/m^2)
15	37.5	12.8074	3.9037	142,667.2
30	75	25.6148	7.8074	285,334.4
45	112.5	38.4221	11.7111	428,001.6
60	150	51.2295	15.6148	570,668.8
75	187.5	64.0369	19.5184	713,336.0
90	225	76.8443	23.4221	856,003.2

For these calculations it was assumed that boiling within the core is to be avoided; that is, the core coolant/moderator must remain single phase (not even subcooled nucleate boiling should occur). Given this restriction, the core-power/core-outlet-temperature combinations given in **bold** in Table 5.3 are eliminated as possible viable operating conditions.

For the remaining options, it is predicted that the surface-cooling mode is dominated by natural convection (not forced convection) on the surface of the fuel pins. The predicted fuel pin surface heat transfer coefficient will range from 1750 to 2800 W/(m^2•°C) for core powers from 15 to 45 MW.

Given the fuel pin power generation rates and the surface heat transfer coefficients, the CARTS fuel code (discussed in Ref. 3) was employed to predict the fuel centerline temperatures. At a core power of 15 MW (fuel pin LHGR of 3.9 kW/ft), the predicted fuel centerline temperature is 320°C. For core powers of 30 and 45 MW, the centerline temperatures are 663 and 1043°C, respectively. Thus, of the core powers considered here, only a core power of 15 MW yields fuel temperatures less than the 450°C disassociation temperature for the UO_3 fuel. As interpolated, the core power could likely approach 21 MW with the given assumptions.

Given the uncertainty in the correlations, the core should be operated at a low core outlet fluid temperature—preferably 30°C.

In summary, the uniformly distributed reactor power should not exceed 15 MW as cooled with a moderator/coolant flow rate of 17,000 gal/min, provided the moderator/coolant inlet and outlet temperatures are 27 and 30^oC, respectively. These constraints could be relaxed substantially with the use of U_3O_8 fuel instead of UO_3.

Because the heat transfer analysis was based upon a uniformly distributed reactor power of 15 MW and because the calculated peak-to-average fission/power distribution ratio is 2.09, it is judged that a conservative reactor power of 7.18 MW (i.e., 15 MW/2.09), with the inlet-to-outlet moderator/coolant flow and temperatures of 27 and 30^oC, respectively, will limit the temperature of the UO_3 to 320^oC, sufficiently less than 450^oC temperature initiating decomposition.

6. DISCUSSION

In summary, a heavy-water-moderated nuclear fission reactor fueled with UO_3 or U_3O_8 can be operated with natural uranium that is produced by means of a standard uranium wet chemistry solvent extraction purification process. Additionally, given the neutronic properties of other elements, it is expected that similar reactors can be operated with uranium carbide, uranium tetrafluoride, and uranium-beryllium alloys.

It is judged that material compatibilities (i.e., aluminum, minimal water, and oxygen evolution from UO_3) are of little concern for the proposed limited total reactor power generation (i.e., 2740 MWd) needed to create ~2 kg of Pu(<5 wt % ^{240}Pu) with ~3 MTU. Throughout this proposed reactor operation, very little available excess neutron multiplication is lost from the reactor due to fission product production and no meaningful decomposition of UO_3 will occur.

The only uncertainty associated with building and operating this conceptually designed reactor is the accessibility of the necessary materials and and standard industrial design/construction skills. The required materials include commercially available high-purity aluminum tubing/sheets, UO_3 or U_3O_8 powder, 99.9 wt % D_2O, high-volume/rate water pumps, heat exchangers, commercially available radiation monitoring and electronic equipment/circuits, electromechanical and pneumatic drive/controls,. Based upon the prior work of Ref. 1 and given the excess reactivity of this design (~0.06), heavy water with higher impurity levels, approaching 1 wt % H_2O, could be tolerated with an increase in core and reflector volumes.

7. REFERENCES

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