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L'ÉNERGIE ATOMIQUE DU CANADA LIMITÉE

THE SLOWPOKE-2 REACTOR WITH LOW ENRICHMENT URANIUM OXIDE FUEL

Emploi d'un combustible en oxyde d'uranium faiblement enrichi dans le réacteur SLOWPOKE-2

B.M. TOWNES and J.W. HILBORN

Presented at the Canadian Nuclear Society 1985 Annual Conference,
Ottawa, 1985 June 3-4

Chalk River Nuclear Laboratories

Laboratoires nucléaires de Chalk River

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Résumé

Le coeur d'un réacteur SLOWPOKE-2 contient moins de 1 kg d'uranium fortement enrichi et le risque de prolifération est très faible. Cependant, pour supprimer toute crainte de prolifération, un nouveau coeur alimenté par de l'uranium faiblement enrichi a été conçu. Ce coeur contient environ 180 éléments combustibles apparentés à l'élément d'UO_2 gainé de Zircaloy-4 employé dans les réacteurs CANDU, mais leur diamètre extérieur est plus petit. Les caractéristiques physiques de ce nouveau coeur de réacteur donnent au SLOWPOKE-2 une sûreté inhérente dans toutes les conditions concevables, de sorte que le concept de sécurité qui permet un fonctionnement sans surveillance n'est pas affecté.

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ABSTRACT

A SLOWPOKE-2 reactor core contains less than 1 kg of highly enriched uranium (HEU*) and the proliferation risk is very low. However, to overcome proliferation concerns a new low enrichment uranium (LEU**) fuelled reactor core has been designed. This core contains approximately 180 fuel elements based on the Zircaloy-4 clad UO_2 CANDU fuel element, but with a smaller outside diameter. The physics characteristics of this new reactor core ensure the inherent safety of the reactor under all conceivable conditions and thus the basic SLOWPOKE safety philosophy which permits unattended operation is not affected.

INTRODUCTION

SLOWPOKE-2 is a 20 kW pool-type research reactor which, due to inherent safety characteristics, is licensed to operate unattended (1). There are currently seven SLOWPOKE-2 units in operation, six in Canada and one at the University of the West Indies in Jamaica. The reactor is used as a neutron source, primarily for neutron activation analysis and short-lived isotope production.

In general, for a small research reactor the maximum value for the ratio of thermal neutron flux to fission power is obtained when the core fissile content is a minimum. Thus, in the SLOWPOKE-2 concept where neutron flux is at a premium, HEU fuel with its small U-238 reactivity load is the obvious first choice. Since a suitable material in the form of an aluminum clad fuel element made of an uranium-aluminum alloy was already in use in the NRX and NRU research reactors, this material was selected for the original SLOWPOKE-1 prototype and all subsequent SLOWPOKE-2 reactors.

These existing reactors are fuelled with less than 1 kg of HEU fuel contained in approximately 300 fuel elements. HEU fuel is viewed as a potential source of weapons material and the supply of HEU to research reactors is becoming more restrictive. Since enrichments of less than 20% are internationally recognized as a fully adequate isotopic barrier to weapons manufacture, the possibility of using 20% enriched uranium instead of 93%, has been under consideration, for SLOWPOKE-2, at Chalk River Nuclear Laboratories since 1979. The result is a new LEU core design which will maintain the SLOWPOKE-2 reactor as a viable product, both for new reactor installations and for replacement cores in existing reactors.

In order to minimize the effects of the change to a new fuel and to facilitate its use for replacement cores in existing reactors, the overall SLOWPOKE-2 reactor geometry has been retained and only the fuel cage and fuel elements have been altered. The fuel element is based on the dependable Zircaloy-4 clad UO_2 CANDU fuel element, but with a smaller outside diameter (5.25 mm), similar to that of the uranium-aluminum fuel element used in the current SLOWPOKE-2 reactors. The fuel cage is manufactured out of Zircaloy-4 material to avoid corrosion problems which might have occurred if the current aluminum cage were used. The first LEU core will be installed this year in a new SLOWPOKE-2 facility at the Royal Military College in Kingston, Ontario.

SLOWPOKE-2: GENERAL DESCRIPTION

SLOWPOKE, an acronym for Safe Low Power Critical Experiment, is a pool-type reactor developed by Atomic Energy of Canada Limited as a neutron source for isotope production and neutron activation analysis. Low cost, inherent safety and simplicity of operation were primary considerations. The reactor provides a usable thermal neutron flux of 10^{12} $n.cm^{-2}.s^{-1}$ at approximately 20 kW thermal power. The prototype SLOWPOKE-1 was commissioned at CRNL in 1970. The first commercial unit, SLOWPOKE-2, was installed in 1971. These reactors are licensed to operate without conventional automatic shutdown devices and without an operator in attendance. The basic design specifications are shown in Table 1. Figure 1 shows the SLOWPOKE-2 reactor assembly.

TABLE 1: SLOWPOKE-2 DESIGN SPECIFICATIONS

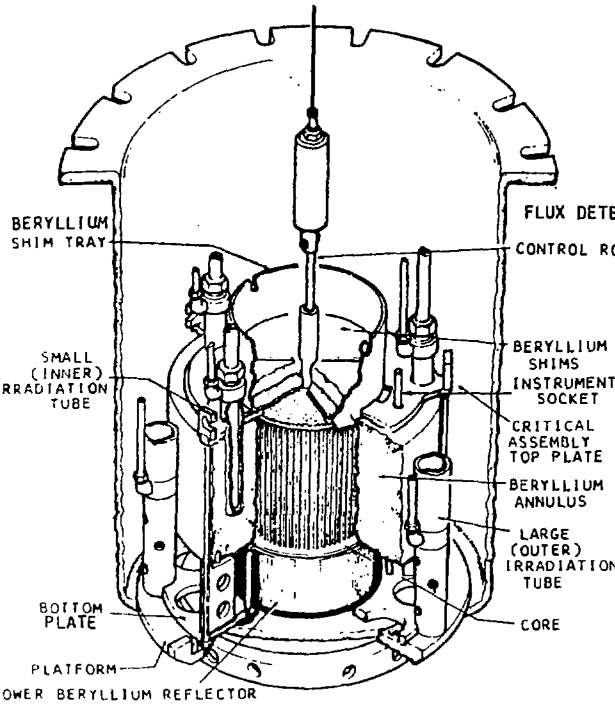
REACTOR	
Pool Diameter	2.5 m
Pool Depth	6.1 m
Container Diameter	0.6 m
Container Height	5.3 m
Core Diameter	22.0 cm
Core Height	22.0 cm
Fission Power	20.0 kW

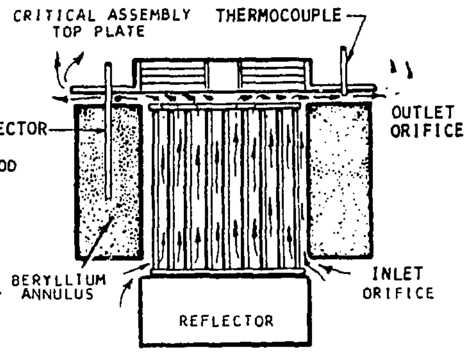
IRRADIATION FACILITIES

		INNER	OUTER
Thermal Flux	$n.cm^{-2}.s^{-1}$	10^{12}	5.8×10^{11}
Diameter	cm	1.6	2.9
Length	cm	5.4	5.4
Volume	cm^3	7	27

* HEU 93 wt% U235 in U

** LEU < 20 wt% U235 in U





COOLANT FLOW BY NATURAL CONVECTION

FIGURE 1: SLOWPOKE-2 CRITICAL ASSEMBLY

SLOWPOKE-2 cores originally contained 820 g of U235 in the form of 295 elements of an aluminum clad HEU-aluminum alloy. The latest SLOWPOKE-2 reactor, which was installed in 1984 at the Atomic Energy of Canada Radiochemical Company, Kanata facility near Ottawa, has a 317 element long life core and contains 875 g of U235 (2). The cylindrical reactor core is surrounded by 10 cm thick beryllium reflectors on the side and bottom. Long term reactivity compensation is effected by adding thin beryllium plates to a shim tray on top of the core. The reactor core and beryllium reflectors are supported inside a cylindrical aluminum water-tight reactor container suspended in the reactor pool, thereby providing double containment for the core water.

SLOWPOKE-2 has five sample sites in the beryllium radial reflector and five more sites in the water surrounding this reflector. Irradiation capsules are transferred to and from the reactor using a compressed gas system in tubes extending from the loading station to the sample site.

The core is cooled by natural convection of the coolant-moderator water. Coolant heat passes through the wall of the container to the pool where it is removed by means of a cooling coil connected to the local water supply.

Inherent reactor safety is guaranteed by a combination of the negative temperature and void coefficients of the undermoderated core, a limited maximum excess reactivity of 0.0034 $\delta k/k$, restricted user access to the reactor core, and administrative control of samples added.

The core of the SLOWPOKE reactor is designed to have negative temperature and void coefficients of reactivity, so that heating or boiling of the coolant-moderator causes the reactivity to decrease. A consequence of this self-regulating characteristic is an upper limit on the equilibrium power equal to the heat removal capacity of the cooling system. A more important consequence of the negative temperature and void coefficient is the inherent protection against reactivity transients caused by loss-of-regulation. The reactor is designed so that the power and temperature transients, resulting from the most severe reactivity transients, are safely limited by the rapid increase in the fuel and moderator temperatures and the production of sub-cooled voids.

Automatic control of the reactor is exercised by a single motor-driven cadmium absorber rod which moves along the central axis of the core through a hole in the top reflector. The control rod motor is activated by a signal from a self-powered neutron detector located in the beryllium side reflector. If the control system fails, the maximum credible reactivity insertion will result in a power transient limited to safe levels by the inherent negative feedback characteristics. If a fault develops in the automatic regulating system, the reactor can be shutdown manually by inserting cadmium filled capsules in one or more of the irradiation sites.

LEU OXIDE FUEL ELEMENT

Low enrichment (19.9 wt% U235 in U) UO_2 pellet fuel has been chosen to replace the uranium-aluminum alloy fuel used previously in SLOWPOKE-2. Ceramic grade UO_2 powder is pressed and sintered to produce cylindrical pellets with a nominal density of 10.6 Mg/m^3. The UO_2 pellets are enclosed in Zircaloy-4 sheaths with welded end caps. Larger diameter fuel elements, manufactured in this manner, have been used in CANDU reactors for many years operating reliably at maximum surface heat fluxes approximately 30 times greater than in SLOWPOKE-2, and at much higher coolant temperatures. The UO_2 fuel elements for SLOWPOKE-2 are being fabricated to AECL specifications by Westinghouse Canada Limited.

Predictions, using the ELESIM computer code (3) have shown that the cladding wall thickness and low coolant pressure ensure that a fuel/sheath gap exists at all times during normal operation. Sheath failure due to stress corrosion cracking or fatigue should, therefore, be of no concern at normal power levels. Calculations have also shown that activity release to the element free volume is negligible due to low operating temperatures, leading to negligible release of fission products should a defect occur. Failure of elements due to coolant-side corrosion is unlikely, since Zircaloy-2 fuel sheathing has operated at temperatures over the range 250°C - 300°C, in the NPD reactor, for greater than twenty calendar years with a typical coolant-side oxide thickness of 0.03 mm.

REACTOR PHYSICS

Reactor physics studies for the SLOWPOKE-2 reactor have been done using either whole core one dimensional calculations in 18 neutron groups using the CRNL version of the WIMS code (4), or two dimensional (R-Z) reactor calculations using condensed 4 group WIMS lattice parameters in the finite difference diffusion theory code CITATION (5). The latter method will be referred to as WC.

Core Loading

About 20% more U235 in total is required to produce the same LEU core reactivity as in the HEU SLOWPOKE-2 core. This is due to the additional U238 parasitic neutron absorptions in LEU fuel. However, because UO_2 has a higher uranium density than the uranium-aluminum fuel which it replaces, the number of elements needed to provide a given reactivity is less in the LEU core than in the corresponding HEU core. WC calculations indicate that 180 LEU elements will give the same initial reactivity as the latest 317 element long life HEU core recently installed at Kanata. A comparison between these two cores is shown in Table 2.

A set of preliminary WIMS calculations for the LEU reactor was run in order to determine an optimum fuel loading pattern in terms of core reactivity, sample site flux/reactor power ratio and the control rod worth. Fuel loadings using both the current hexagonal lattice with 342 possible fuel locations and lattices designed to give a uniform hexagonal distribution with about 200 possible fuel sites were investigated. The results indicated that in general the control rod worth and the sample site flux/power ratio can be increased at the expense of core reactivity by reducing the number of occupied fuel sites close to the central absorber location and/or the beryllium radial reflector. Results for regular arrays with fewer fuel sites tended to be less reactive, because the resulting increase in lattice pitch implied that fewer sites were located close to the beryllium radial reflector than could be achieved using the 342 site arrangement. Therefore, the 342 site cage has been retained and a compromise between reactivity and an acceptable sample site flux to power ratio results in the layout shown in Figure 2 for a 180 element fuel loading.

TABLE 2: COMPARISON BETWEEN HEU AND LEU CORES

	HEU	LEU
Fuel Material	U/Al	UO_2
Enrichment (%U235)	93	19.9
Fuel OD (mm)	4.22	4.19
Sheath Material	Al	Zr-4
Sheath OD (mm)	5.23	5.25
Fuel Length (mm)	220	227
No. of Elements	317	180
Total Uranium (kg)	0.94	5.22
Total U235 (kg)	0.88	1.05
Power Form Factor*	1.33	1.44
Avg.Surface Heat Flux**(W/cm^2)	1.74	2.9
Peak Surface Heat Flux**(W/cm^2)	2.31	4.2
Cool Temp.Coeff.(mk/°C)	-0.18	-0.13
Fuel Temp.Coeff.(μk/°C)	-0.50	-8.50

* Peak Power/Average Power
** at 20 kW

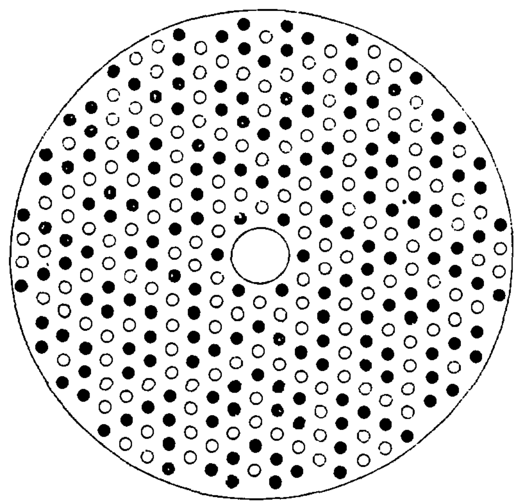


FIGURE 2: 342 SITE FUEL CAGE
PROPOSED 180 ELEMENT ARRANGEMENT

Change in Fuel Length

Because there is more uranium in the LEU core, the core is "blacker" to neutrons and there is less neutron leakage than from an HEU core. As a result the top beryllium reflector, which is used for long term reactivity control, will be worth about 15% less in the LEU core. In the new LEU core the top reflector worth has been increased by about 15% to give the same value as in the current HEU design by reducing the water gap between the top of the fuel and the base of the top reflector support tray. This has been done by increasing the LEU fuel stack length to 227 mm compared to 220 mm in the HEU core, and allowing the LEU fuel elements to protrude above the fuel cage top plate. The net result is an LEU reactor core which has a predicted lifetime of at least 40 kW.a similar to that of the long life HEU core installed at Kanata.

Power and Flux Distributions

Power and neutron flux distributions have been derived from WC calculations. As an example, the axial power and thermal flux (< 0.625 eV) distributions at a radius of 4.7 cm are shown in Figure 3 for the start-of-life core with no beryllium top reflector plates and the control absorber out. Note the asymmetry produced by the lack of a top beryllium reflector and the flux peaking which occurs above and below the reactor core. The power distribution is shown in terms of the variation in the ratio of the point power density relative to the average core power density. Plotted in this way, the power form factor can be read directly off the curve.

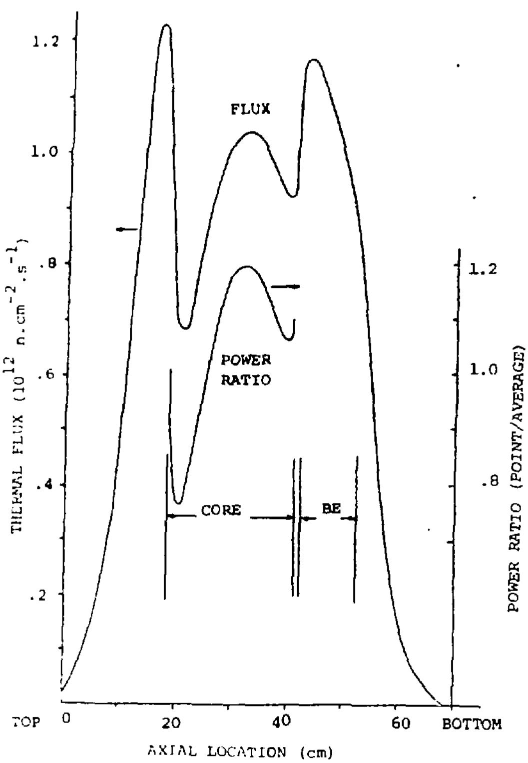


FIGURE 3: AXIAL DISTRIBUTION

When the control absorber is out, the thermal flux peaks in the central water zone and the peak fission power occurs in the six central fuel elements just below mid-core height. In this situation the power form factor (maximum/average) has its largest value of 1.44.

The calculated core thermal power required to provide a sample site thermal flux of 10^{12} $n.cm^{-2}.s^{-1}$ is 19.6 kW, corresponding to a value of 18.9 kW for the 317 element Kanata HEU core.

Reactivity Coefficients

The reactivity feedback effects due to temperature changes were calculated by treating the three variables: fuel temperature, coolant temperature and coolant void, independently. A series of WIMS calculations were done for fuel temperatures varying from 10°C to 500°C, coolant temperatures (density) from 10°C (.9997 g/cm^3) to 90°C (.9653 g/cm^3) and coolant voids from 0 to 20%.The corresponding 4 group parameters were used in CITATION calculations and the reactivity change in each case relative to a reference state (uniform temperature 20°C) was calculated. These data were fitted to polynominals for use in the thermal hydraulics code SPORTS (6) (see section on transients).

The WC results for the HEU/Al core were also calculated and are compared with experimental values in Table 3. The results show that the calculations somewhat underestimate the measured values and will therefore, lead to conservative estimates of power transients since the actual negative reactivity feedback is greater than the calculated value used in the transient simulations.

TABLE 3: CORE TEMPERATURE REACTIVITY CHANGES

CORE TEMP. RANGE (°C)	REACTIVITY CHANGE (mk)		
	HEU		LEU
	Expt.	WC	WC
20 - 40	-2.8	-2.70	-1.70
20 - 60	-7.3	-6.38	-4.29
20 - 90	-15.0	-13.30	-9.69

Results for the LEU case, also shown in Table 3 for comparison, are about 30% less than the HEU values, primarily due to the lower neutron leakage from the "blacker" LEU core. Therefore, the steady state power following a reactivity insertion will be somewhat higher in the LEU core relative to the comparable HEU case.

Control Rod and Auxiliary Shutdown System Worths

WIMS calculations were used to determine the control rod worth and the cadmium capsule auxiliary shutdown system worth relative to the corresponding values in the HEU core. The calculated LEU core reactivity worths are 96% and 89%, respectively, of the HEU control rod worth and auxiliary shutdown system worth.

The reactivity worth of the control rod is set by a limit on the depth of insertion and there is sufficient flexibility to allow the LEU control rod to provide the same 5 mk reactivity worth as is available in the HEU core.

Each cadmium capsule has a worth of -1.4 mk in the HEU core and a predicted value of -1.2 mk in the LEU core. The minimum combined capsule worth is -5.6 mk in the HEU core assuming a cadmium capsule in each of 4 empty inner sites. The corresponding predicted value of -4.8 mk in the LEU core, is adequate for any potential abnormal reactivity additions.

Long Term Reactivity Changes

During reactor operation U235 is consumed and fission products build up, resulting in a gradual reduction in reactivity with time. Compensation for this reactivity loss is effected by periodically increasing the thickness of the top beryllium reflector by the addition of beryllium plates of small known reactivity worths. As the beryllium thickness increases the incremental reactivity effect of additional beryllium plates decreases. As a result the useful thickness of the top beryllium reflector is approximately 10 cm. The total worth of a 10 cm thick top beryllium reflector is estimated from WC calculations to be 20 mk, similar to that for the HEU core. This provides sufficient reactivity for at least 40 kW.a of operation with the LEU core, corresponding to at least 20 calendar years of typical operation.

THERMALHYDRAULICS

Heat Production and Transfer Characteristics

Although the LEU fuel has similar external dimensions to the HEU fuel, the average surface heat flux for a given total reactor power is higher for the LEU fuel by a factor of about 1.7, due to the smaller number of elements in the LEU core. The power form factor of 1.44 in the LEU core compared to 1.33 in the HEU core, results in a peak surface heat flux which is about 1.9 times higher in the LEU core than in the 317 element HEU core.

The time constants which govern the transient heat removal behaviour are also somewhat longer for the LEU oxide compared to the HEU/Al metal fuel. In the high thermal conductivity metal fuel, the main thermal resistance between fuel and coolant occurs at the sheath coolant interface, whose properties, therefore, determine the time constant. In the LEU fuel the lower conductivity UO_2 has a temperature gradient within the fuel, and the situation is further complicated by the fuel to sheath heat transfer coefficient, which varies from about 1 $kW/(m^2.k)$ to 35 $kW/(m^2.k)$, depending on whether the fuel contacts the sheath or not.

Changes in the average fuel temperature can be approximately described in terms of a simple time constant:

$$\tau = \frac{\rho c a^2}{8\lambda} + \frac{\rho c a}{2h}$$

where ρ = fuel density
c = fuel mean specific heat
a = fuel radius
λ = fuel thermal conductivity
h = surface heat transfer coefficient
= $(1/h_o + 1/h_f)^{-1}$

where h_o and h_f are the sheath-coolant and fuel-sheath heat transfer coefficients. In high conductivity U/Al alloy the first term in the expression for τ is negligible and $\tau \sim \rho ca/2h$. For UO_2 both terms are important. Values for the fuel time constant τ for typical values of the surface heat transfer coefficients are listed in Table 4 for both fuel types. Although the oxide fuel has a longer response time (1 - 4 s) than the U/Al fuel (0.5 - 2.5 s) this has only a small effect on the prompt power peak (see transient section) and no effect on the time taken to reach the limiting steady power. In the case of a 6 mk transient for example the prompt peak occurs about 15 s into the transient and the steady state power is reached after about 100 s.

TABLE 4: FUEL TIME CONSTANTS

Fuel Material	Heat Transfer Coeff. ($kW.m^{-2}.k^{-1}$) h_f	h_o	h	Time Constant (s)
U/Al	∞	1	1	2.5
	∞	5	5	0.5
UO_2	2	1	0.7	4.2
	2	5	1.4	2.2
	35	5	4.6	0.9

The ELESIM code has been used to predict steady state operating parameters for a range of reactor powers using sheath to coolant heat transfer coefficients calculated by the SPORTS code. Table 5 lists some of this data together with estimated reactivity effects associated with the fuel and coolant temperature changes relative to a reference 20°C value. Also shown is the xenon reactivity load after 12 hours of operation at the stated power.

TABLE 5: STEADY STATE OPERATING CONDITIONS

Reactor Power (kw)	Peak Centre Fuel Temp. (°C)	Coolant Out.Temp (°C)	Reactivity Change (mk)	Xe Worth 12 hr. (mk)
0.1	20	21	-0.1	-0.01
1.0	37	23	-0.2	-0.05
5.0	71	29	-0.6	-0.21
10.0	95	34	-0.8	-0.42
20.0	150	41	-1.4	-0.81
50.0	268	57	-2.8	-1.99

Transients and the Self-Limiting Power Excursion Behavior

The SLOWPOKE safety philosophy is based on a limited maximum excess reactivity during all normal and conceivable abnormal conditions together with a safe self-limiting power excursion behaviour for reactivity insertions greatly in excess of this maximum excess reactivity. The maximum credible reactivity insertion is 3.9 mk which may occur as follows.

If during a reactor start-up (with an installed system excess reactivity of 3.4 mk) with a typical sample (worth 0.1 mk) in one of the small (inner) irradiation sites, an inner irradiation tube floods (worth 0.38 mk) and the control system fails when the control rod is in its fully withdrawn position then the reactor could experience a power excursion with an initial excess reactivity of 3.9 mk.

To demonstrate the safe behaviour of the self-limiting power excursion, transient tests will be performed during reactor commissioning experiments for progressively increasing reactivity insertions. In advance of these experiments, transient simulations have been performed using the SPORTS code both for the LEU and HEU cores.

The SPORTS code was initially developed at CRNL for stability studies on the SLOWPOKE-3 heating reactor. Under single-phase conditions the code is sufficiently accurate. For two phase conditions, an empirical correlation is used to describe the void, and the void is assumed to be homogeneously mixed. Point source neutron kinetics are incorporated, using a reactor period versus reactivity variation based on the INHOUR equation.

The reactivity feedback effects from changes in fuel temperature, coolant temperature and coolant voiding, calculated as described earlier, are used in SPORTS in the form of fitted polynomials. The kinetic parameters, mean neutron lifetime (7.65×10^{-5}s) and the effective delayed neutron fraction (.00803), were determined from WC calculations using flux and adjoint flux weightings, with six delayed neutron groups.

The reactor was modelled in one dimension from the inlet orifice at the bottom of the core to the outlet orifice at the top of the core, and although coolant flow over this region is not strictly one dimensional, because of the cross flow which occurs in certain locations, the one dimensional treatment is a good approximation.

The core was divided evenly into 10 axial nodes and each orifice into 2 nodes. As the circulation in the container surrounding the core is at very low velocity, it imposes a constant hydrostatic pressure drop across the core and fixed boundary conditions of pressure and inlet temperature were assumed. The assumption of a constant inlet temperature is valid for the first few minutes of the transients. After that, the inlet temperature will start to rise and the calculated transients have, therefore, been limited to the first 200 seconds. If the outer thermalhydraulic loop and resulting increasing inlet temperature were modelled, the predicted final power and fuel temperatures would be lower. These results are, therefore, conservative.

Figure 4 shows the calculated power variation with time for a step reactivity insertion of 6 mk. After about 15 seconds into the transient the power exhibits a "prompt" peak of 140 kW. This occurs due to the rapid negative reactivity feedback arising from heating the fuel and moderator at times before a significant fraction of the energy generated in the core has been removed from the core by natural convective flow. As convective flow develops, heat is removed from the core but slowly compared to the rate at which energy is being generated within the core. Core temperatures and power continue to increase while reactivity decreases until a steady power level of 180 kW is reached at which point the initial reactivity insertion is balanced by the negative temperature effects. As already noted these results are expected to be conservative since the increase in inlet temperature which begins after a minute or two is ignored. A further degree of conservatism arises because the calculated reactivity coefficients are less than the measured values in the HEU core (see Table 3) and are, therefore, also expected to underestimate the true values in the LEU core.

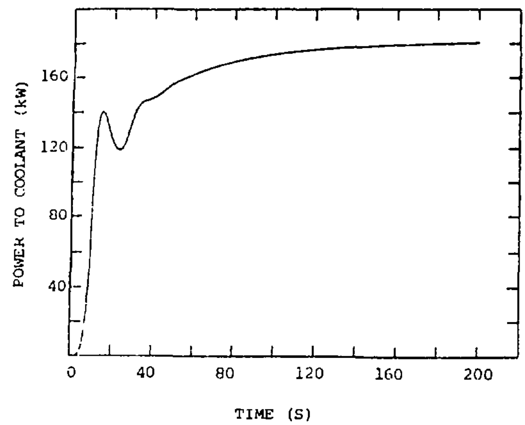


FIGURE 4: 6 mk STEP TRANSIENT

Critical Heat Flux

A conservative equation, developed by J.T. Rogers (7) for predicting the critical heat flux (CHF) in the HEU SLOWPOKE-2 core is:

$$CHF = 580 + 11\ \Delta Ts\ \text{kW/m}^2$$

where ΔTs is the sub-cooling in °C

This expression, also applicable to the LEU core, predicts that even with 0°C sub-cooling, the CHF will not be attained locally in the LEU core until the reactor power exceeds 270 kW. This is far in excess of conceivable abnormal levels which are predicted to be less than 110 kW for the 4 mk largest credible reactivity insertion and only 180 kW for the 6 mk insertion as shown in Figure 4.

SUMMARY

The very low proliferation risk for the current HEU fuelled SLOWPOKE-2 reactor has been removed entirely by the development of an LEU fuelled core which can be used as a replacement core in existing reactors or as a first core in new installations. Calculations indicate that this new LEU core exhibits all the inherent safety characteristics of the original HEU design. The predicted margin to dryout is more than adequate for all conceivable normal and abnormal operating conditions.

ACKNOWLEDGEMENTS

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