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STAR: The Secure Transportable Autonomous Reactor system
Encapsulated Fission Heat Source
NERI Proposal 99-0154

Progress Report
for period terminating March 31, 2000

Comments and questions should be sent to

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Significant progress has been made since the beginning of this project, in September 1999. Part of the project findings are described in References 1 and 2. So far we have found no "show-stopper". In fact, based on the findings accumulated so far it appears to us that the Encapsulated Nuclear Heat Source (ENHS) is technologically feasible and looks even more attractive than initially conceived.

The international collaboration is building up satisfactorily. Recently we have been informed by CRIEPI (The Japanese equivalent of EPRI) that they decided to start now the active collaboration with us. They intend to support few researchers to work on complementary ENHS issues we have defined. Their team will participate in our next project meeting planned for late June (Tentatively, 6/26 and 6/27 at UCB). Prof. Soon Chang of KAIST, Korea, is collaborating with us since the Fall of 1999. Dr. Dan Cacuci, the head of the Reactor Safety Institute of the Karlsruhe Research Center in Germany expressed interest to collaborate with us, but the actual collaboration has not started as yet.

Following the references we give a brief task-by-task summary of the work done.

References

1. E. Greenspan, H. Shimada, D.C. Wade, M.D. Carelli, L. Conway, N.W. Brown and Q. Hossain, "The Encapsulated Nuclear Heat Source Reactor Concept," Paper ICONE-8750 in the Proc. 8th Int. Conf. On Nuclear Engineering, Baltimore, MD, April 2-6, 2000. 13pp.
2. E. Greenspan, H. Shimada and K. Wang, "Long-Life Cores with Small Burnup Reactivity Swing," to be published in the Proc. of the 2000 Int. Topical Mtg. On Advances in Reactor Physics and Mathematics and Computation into the Next Millennium, PHYSOR2000, Pittsburgh, PA, May 7-11, 2000. 26pp.

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Task 1

Several layouts of the ENHS have been developed. These include a novel layout that uses a cover-gas lift-pump for helping the circulation of the primary coolant. Our analysis indicates that it should be possible to manufacture and fuel the ENHS in the factory, and ship it complete and sealed to the site.

Task 2

Our analysis so far indicates that 4 mm is an adequate thickness for the confinement wall, and that with such a wall thickness it is possible to transfer the power from the primary to the secondary coolant with a temperature drop not exceeding 50°C.

Task 3

Acceptable structural wall thickness for the ENHS was found to be 5 cm.

Task 4

It was found possible to design the ENHS primary coolant to have 100% natural circulation. Several dozen alternative designs having 100% natural circulation along with 50°C primary-to-secondary temperature drop were worked out. A one-dimensional simulation model has been developed and is being applied to transient analysis of the primary coolant. It was found that it is feasible to operate the ENHS with 100% natural circulation of the primary coolant over the entire range from very low power level to full power.

Task 7

Different approaches to the design of decay heat removal systems have been evaluated. A PRISM like RVACS system was adopted for the ENHS. If cover-gas lift-pump will be adopted for the ENHS, the cover-gas system will be used as an additional decay heat removal system. Scenarios to analyze for studying the feasibility of decay heat removal were defined. A transient analysis simulation model for decay heat removal is being developed.

Task 9

Reactivity coefficients have been calculated for the reference core. Transient analysis simulation model has been developed for the primary system. Preliminary analysis indicates that the ENHS has a good load following capability.

Task 13

We have identified a scenario for the insertion of the ENHS module into the pool. The insertion will be done with the fuel embedded in solidified lead or Pb-Bi. After insertion into the pool, the Pb (or Pb-Bi) will be melted.

Task 18

No progress was made so far.

Task 19

A couple of novel approaches were conceived for the IHX design. They appear to be simpler to implement than the corrugated confinement wall. One of the design approaches uses circular tubes for the IHX while the other uses channels that are rectangular in cross section. Thermo-

hydraulic analysis performed indicates that the performance of the ENHS that uses these new IHX designs is comparable to that of the reference ENHS.

Task 23

It was found feasible to design simple, uniform composition lead-cooled and reflected cores that maintain nearly zero burnup reactivity swing for up to 15 years of operation without any refueling. These cores have no internal or external blanket assemblies and no reflector assemblies. They are to be loaded inside the module in the factory and taken back to the supplier country in the unopened module. Consequently, the ENHS is expected to be highly proliferation resistant, extremely safe, and very easy to operate.

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THE ENCAPSULATED NUCLEAR HEAT SOURCE REACTOR CONCEPT

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ABSTRACT

The encapsulated nuclear heat source (ENHS) is a novel reactor concept in which the fission-generated heat is transferred from the primary- to the secondary coolant through a reactor vessel wall of a novel design. This enables the reactor module to have a very simple design, to be free of any mechanical connection to other power plant components, and to be simple to install and to replace. The ENHS is to be factory fueled and to deliver 125 MW_{th} per module for 15 effective full power years (EFPY) without any on-site fuel handling. At the end of its core life the ENHS is to be replaced by a new ENHS module.

A study of the feasibility of the ENHS has been initiated in September 1999 under sponsorship of the DOE NERI program. It was found that using lead for the coolant, the ENHS can be designed to have 100% natural circulation and deliver 125 MW_{th} to the secondary coolant through the reactor vessel wall with no more than 50 °C temperature drop. Use of cover-gas lift-pump can significantly reduce the volume and weight of the ENHS module. Feasible diameter and height of an ENHS module for 125 MW_{th} are, respectively, 2.5 m and 20 m. Its weight

for transportation, when loaded with fuel and solidified Pb, is less than 200 tons.

It was also found feasible to design simple, uniform composition, Pb-cooled cores to deliver 125 to 250 MW_{th} for 15 EFPY with very small burnup reactivity swing; approximately 0.5%. The design domain for such cores has been defined.

INTRODUCTION

The encapsulated nuclear heat source (ENHS) is a novel reactor concept¹ that has been selected by the 1999 DOE NERI program as one of the "Generation - IV" reactors to be studied. The ENHS reactor features a high degree of modularity, relatively small and simple modules that are all factory manufactured and easy to install, high degree of passive safety, simplicity of operation and maintenance and long-life cores¹⁻⁴. The ENHS constitute a totally new refueling concept: it is to be factory fuelled, inserted into an in-place power plant, run for 15 effective full power years (EFPY) without any refueling, and then replaced by another module.

One of the unique features of the ENHS is that the fission-generated heat is transferred from the primary coolant to the secondary coolant through the reactor vessel wall. This enables the reactor module

to have a very simple design and to be free of any mechanical connections to the power plant components. The ENHS coolant need have a low vapor pressure at operating temperatures. Lead or Pb-Bi eutectic appears to be among the most promising.

The purpose of this paper is to briefly describe the ENHS concept and to summarize findings from a couple of studies completed since the inception of this NERI project. These studies are: (a) Exploration of the range of dimensions and weights of ENHS for 125 MW_{th} that are to feature up to 50 °C primary-to-secondary temperature drop and up to 100% natural circulation. (b) Definition of the design domain of cores for the ENHS that enable 15 EFPY with close to zero burnup reactivity swing.

SINGLE ENHS REACTOR CONCEPT

A very schematic description of one of many possible embodiments of a reactor concept having a single ENHS module is depicted in Figs. 1 and 2. The reactor consists of ten modules: one ENHS, three steam generators, three reheaters and three secondary coolant pumps. All these modules are inserted into the reactor pool; they are supported by a massive structural platform that is seismically isolated. There is no mechanical connection between the modules. This makes them relatively simple to install and to replace.

A schematic vertical cut through one concept of an ENHS module is shown in Figure 3. This concept features 100% natural circulation. There are three walls to the ENHS reactor vessel – two structural walls and a confinement wall in between. The confinement wall provides the barrier between the primary and secondary coolants. The primary coolant gets from the riser into the downcomer channels that are formed between the inner structural wall and the corrugated confinement wall and gets back into the coolant plenum underneath the core. The secondary coolant gets from the pool into the space between the corrugated confinement wall and the outer structural wall and flows up in the channels located between the downcomer channels. The heated secondary coolant flows back to the pool near its top. Heat is conducted from the primary to the secondary coolants through the confinement wall. The confinement wall is corrugated in order to increase its surface area for heat transfer so as to keep the primary-to-secondary temperature drop at a reasonable value – taken in this work to be 50 °C.

Figure 4 shows a horizontal cut through a corrugated confinement wall. The corrugated wall channels are stiffened against buckling by vertical bends (not shown in Figure 4). Horizontal rings that are welded to the tips of the corrugations provide additional stiffening. They also act as spacers

between the confinement wall and the inner and outer structural-walls. The axial distance between these spacers is approximately 1m. In addition to their structural function, the spacers are to avoid “rubbing” of the thin confinement wall against the adjacent structural walls and to assure that the space between the base of the corrugations and the structural walls will be large enough to avoid formation of “pockets” of stagnant coolant.

Illustrations of possible corrugation dimensions are given in Tables 1 and 2 that are discussed in the following section. The downcomer flow channel formed by the corrugated wall is rectangular with a cross-section of (Downcomer depth) x (Downcomer width). The corrugations of Figure 4 pertain to Case 9.2.1; they are of an extremely large aspect ratio. Smaller aspect-ratio corrugations can also be used (See Tables 1 and 2).

An alternate ENHS concept being considered is depicted in Figure 5. It uses a cover-gas lift-pump that circulates the cover gas from the plenum above the coolant level in the ENHS and injects it into the coolant in the riser through nozzles located at a certain level above the core^a. The cover gas bubbles reduce the effective density of coolant in the riser, thus increasing the head for coolant circulation. The circulator is envisioned to be located above the reactor pool, outside of the ENHS. Thus, no moving mechanical parts and no electrical parts associated with coolant circulation are located inside the ENHS.

There are very few components inside the ENHS reactor vessel. In particular, it does not include intermediate heat exchangers (IHX), decay heat removal systems (DHRS), and mechanical or EM pumps. Its confinement wall serves as the IHX and part of the DHRS. Neither primary nor secondary coolants are pressurized. They exert nearly balancing hydrostatic pressure on the relatively thin confinement wall.

As presently perceived, the ENHS will be manufactured and fuelled in the factory and shipped to the site as a sealed unit with solidified Pb or Pb-Bi filling the vessel up to the upper level of the fuel rods. Upon insertion to the reactor pool hot Pb (or Pb-Bi) is to be pumped into the ENHS vessel. This hot Pb, along with the hot Pb in the pool, will melt the solid Pb at the lower part of the vessel.

At the end of its life, presently perceived to be 15 EFPY, the ENHS will be removed from the reactor pool and stored on site until the decay heat will drop

^a A cover gas lift pump for circulation of liquid lead coolant was first proposed by ANSALDO⁷ for accelerator driven system. The use of steam lift pumps for circulation of secondary lead coolant was proposed earlier by one of the co-authors et al.⁸⁻¹⁰

to a level that will let the Pb (or Pb-Bi) solidify. Before solidification, the coolant will be extracted from the upper part of the ENHS above the level of the fuel rods. The ENHS with the solidified Pb will then serve as a shipping cask.

Currently, the Control and Shutdown Systems (C&SS) used in the 4S reactor^{5,6} have been adopted for the ENHS. The C&SS consists of a single neutron absorber assembly located centrally in the core and six reflector drive mechanisms that move the six radial reflector segments. The shutdown assembly is securely latched in place during shipping and reactor installation. The shutdown rod includes an electromagnetic latch that does not engage until the startup temperature is achieved, about 350°C. At this point the rod can be withdrawn and held out until permanent shutdown is required. Normal operational shutdowns can be accomplished with the reflectors as needed. The reactor is brought critical by a hydraulic system that moves the reflectors at 1 mm/sec to compensate for the negative temperature coefficient of reactivity. At the full power position, the reflectors are stopped from further upward movement by mechanical stops whose movement is established by high-reliability gear drives. These drives restrict the rate of movement of the reflectors to approximately 1 mm/day. This movement compensates for the burn-up reactivity effect. There is no feedback to this drive mechanism. It is anticipated that the power level can be controlled over a wide range simply by using steam pressure and feed-water flow control. In summary, the reactor control is envisioned to be very simple and fully automated: a combination of axial movement of the radial reflector surrounding the core and passive control by temperature feedback.

Each of the six reflector segments is presently perceived to be a 15cm thick cavity that is as long as the active-fuel region. At the shutdown position the cavity level coincides with the active fuel level. The highest reactivity-state of the core is obtained with the cavity withdrawn from the vicinity of the active fuel region; Pb (or Pb-Bi) surrounds the core. The reactivity worth of each reflector segment is less than half a dollar. Hence, a severe reactivity insertion accident is inconceivable. Rather than one cavity per reflector segment, it might be preferable to design the reflector container to have many small compartments or to be filled with a foam-like material that is compatible with Pb and that is not good at slowing down and at scattering neutrons. This will reduce the probability of Pb filling the cavity and making it an ineffective reactivity control mechanism in case of a breach in the container wall. Tungsten-based reflector is also being considered as possible alternative to the cavity reflector.

The path for decay heat removal is from the primary coolant through the corrugated confinement wall into the secondary coolant and through the pool wall into a passive air-cooled reactor vessel emergency cooling system (RVECS). The ENHS is characterized by a very large surface area per MW_{th} of decay heat removal from both the primary- to the secondary coolant, and from the secondary coolant to the RVECS. In addition, the ENHS is characterized by a large thermal inertia due to the large inventory of the primary and secondary Pb. Both features are expected to make the ENHS highly passively safe.

The number of steam generators, reheaters and pumps to be used in the single ENHS module reactor of Figs. 1 and 2 is yet to be optimized. In fact, it may be decided not to use at all reheaters that are being heated by the secondary coolant. This can simplify the energy conversion system design but will somewhat reduce the energy conversion efficiency. The reason for showing the reheaters in Figure 2 is to indicate that it is straightforward to incorporate them in the ENHS reactor. This can not be done in most other reactor design concepts.

ENHS MODULE DESIGN POSSIBILITIES

A preliminary scoping study was performed to identify the required dimensions of the ENHS module for 125 MW_{th} and its corresponding weight. The design goal of this study was to get a primary-to-secondary coolant temperature drop of approximately 50 °C with and without a lift-pump.

Following is a list of assumptions used for this scoping study:

1. Thermal power is 125 MW.
 2. Average linear heat rate is 120 w/cm. In a later part of the study we reduced the average linear heat rate to 80 w/cm.
 3. Core support plate thickness is 0.3 m.
 4. Cavity height below core support plate is 1 m.
 5. Fission gas plenum length above fuel is 75% of the fuel length.
 6. Cover gas cavity height is 1 m.
 7. Vessel bottom and top base thickness is 0.5 m.
 8. Effective coolant layer around core is 3 cm.
 9. Core barrel thickness is 1 cm.
 10. Coolant layer outside of core barrel is 1 cm.
 11. Reflector thickness is 15 cm.
 12. Reflector drive guide wall thickness is 2 cm.
 13. Shield thickness is 51.5 cm.
- Assumptions "5", "8" through "13" are based on the 4S reactor design by Toshiba^{5,6}.
14. Outer structural wall thickness is 3 cm.
 15. Space between corrugations and structural walls is 1cm.
 16. Core coolant inlet/outlet temperature is 420/540 or 400/560 °C.

17. Maximum clad temperature is represented by the outer clad temperature calculated at the core outlet, assuming the average linear heat rate of "2". This follows ANL Stage 1 modeling¹¹.
18. The effect of lift-pump on generating head for Pb circulation is accounted for parametrically by assuming an effective reduction in the Pb density above the gas injection level.
19. The confinement wall thickness is 4 mm. This thickness appears adequate for 15 years of operation.
20. An estimate for the primary-to-secondary temperature drop is $\Delta T_c + 2 \Delta T_{film}$ where ΔT_{film} is the film temperature drop in the primary side.
21. The downcomer flow channels are rectangular in shape. Their hydraulic-diameter was assumed to be the narrow dimension of the channel. This is a conservative approximation.

The core dimensions used for this scoping study have not yet been fully optimized. For a given core height, fuel rod diameter and average linear heat-rate, there is only one pitch (or a very narrow range of pitches) that will give the desirable flat k_{eff} variation with burnup. The definition of these cores is addressed in the following section.

Shielding of the structural components has not been designed, as yet. The shield needed for the ENHS structural wall is likely to be considerably thinner than the 51.5 cm borrowed from the 4S reactor design.^{5,6} In fact, the ENHS structure may not

need at all a special shielding against radiation damage. This is because the ENHS lifetime is constrained by the peak fast neutron induced damage to the clad, and the peak damage rate to the structural components outside the core is smaller than the peak damage rate to the clad.

Tables 1 and 2 present design and performance characteristics of different ENHS. Table 1 pertains to the ENHS of the type shown in Figure 3 (Type A). Table 2 pertains to an ENHS that differs from that depicted in Figure 3 in a couple of ways: (a) Its corrugated confinement wall starts from a level that is 2 m above the top of the core. (b) The inner structural wall radius 2m above the core coincides with that of the core barrel (Type B).

Each entry line ("Case") in Tables 1 and 2 represents an ENHS module in which the combination of the design variables enables the primary coolant to have 100% natural circulation and to transport 125 MW_{th} to the secondary coolant with close to 50 °C temperature drop. The design variables of the study are the five parameters at the left-hand side of the tables: downcomer channel dimensions, riser length, core height and pitch (i.e., core fuel rod lattice pitch). Additional design variables are the core inlet/outlet temperatures and the lift-pump induced effective Pb density in the riser. The other parameters given in the tables are derived.

Table 1 Selected Characteristics of ENHS Modules of Type A

Case No.	Core height (m)	Pitch (cm)	Riser length (m)	Down-comer depth (cm)	Down-comer width (cm)	Max. clad temp. (oC)	Total dT 1->2 (C)	Upper/lower RV radius (m)	No. Total RV height (m)	down-comer chanls	Total weight SS (t)	Total weight Pb (t)	Core Pb velocity (m/s)	Down-comer Pb vel. (m/s)	Comer Film DT (K)
							Tin/Tout=420/540	No lift-pump							
9.1.1	1	1.6	23	45	5	564.2	42.5	1.59/1.63	28.1	77	73.4	1430	0.68	0.4	14.9
							Tin/Tout=400/560	No lift-pump							
9.1.2	1	1.55	23	30	4	583.4	48.1	1.41/1.60	28.1	87	70.8	1208	0.59	0.48	15.8
9.1.3	2	1.7	23	50	5.25	585.8	48.1	1.42/1.42	29.8	63	54.9	1125	0.82	0.31	7.5
							Tin/Tout=400/560;	Lift-pump @ 90% density							
9.1.4	1	1.45	12	16	1.5	577.0	49.1	1.22/1.55	17.1	181	65.2	544	0.81	0.94	11
9.1.5	2	1.5	15	20	2	575.4	48.6	1.05/1.35	21.8	119	48.3	492	1.37	0.93	11.9
9.1.6	2	1.6	18	15	2	580.4	50.8	1.04/1.38	24.8	121	51.5	588	1.03	1.18	11.8
9.1.7	3	1.6	21	17.5	2.5	576.8	51.0	0.96/1.28	29.6	90	43.4	570	1.54	1.15	12.6
9.1.8	3	1.6	21	20	2	576.8	36.9	0.98/1.28	29.6	110	43.4	589	1.54	1.00	8.9
							Tin/Tout=400/560;	Lift-pump @ 80% density							
9.1.9	3	1.6	17	15	1.8	576.8	49	0.93/1.28	25.6	116	43.2	467	1.54	1.34	10.7
9.1.10	3	1.6	19	12.5	2	576.8	57	0.91/1.28	27.6	105	43.3	492	1.54	1.59	12.1

Table 2 Selected Characteristics of ENHS Modules of Type B

Case No.	Core height (m)	Pitch (cm)	Riser length (m)	Down-comer depth (cm)	Down-comer width (cm)	Max. clad temp. (oC)	Total 1->2 dT (C)	upper/	No.		Total height (m)	Down-comer chanls	Total SS weight (t)	Total Pb weight (t)	Core Pb velocity (m/s)	Down-comer Pb vel. (m/s)	Down-
								lower	Total	down-							comer
								RV	RV	comer							film
								radius (m)									dT (K)
								Tin/Tout=420/540		No lift-pump							
9.2.1	1	1.6	23	45	5	564.2	54.8	1.42/1.63	28.1	67	70.9	1078	0.68	0.46	18.8		
								Tin/Tout=400/560		No lift-pump							
9.2.2	1	1.55	22	35	3.5	583.4	50.3	1.29/1.60	27.1	87	68.4	895	0.59	0.47	16.1		
9.2.3	2	1.7	23	50	4.5	585.8	53.9	1.26/1.42	29.8	62	52.9	806	0.82	0.37	18.7		
								Tin/Tout=400/560;		Lift-pump @ 90% density							
9.2.4	1	1.45	17	16	1.5	577.0	52.1	1.05/1.55	22.1	153	63.2	516	0.81	1.11	11.2		
9.2.5	2	1.48	12	40	2	576.4	52.4	1.07/1.34	18.8	109	45.8	345	1.46	0.53	14.3		
9.2.6	2	1.6	14	24	1.5	580.4	50.3	0.96/1.38	20.8	132	49.4	353	1.03	0.89	11.4		
9.2.7	3	1.6	16	30	2	576.8	51.7	0.91/1.28	24.6	94	41.6	345	1.54	0.8	13.1		
9.2.8	3	1.7	22	20	2	581.3	51.6	0.84/1.31	30.6	92	44.2	417	1.22	1.20	11.9		
								Tin/Tout=400/560;		Lift-pump @ 80% density							
9.2.9	3	1.6	20	18	1.5	576.8	48.8	0.79/1.28	28.6	109	41.8	338	1.54	1.4	9.9		
9.2.10	3	1.55	10	36	1.5	574.7	51.5	0.95/1.26	18.6	121	40.2	258	1.77	0.66	12.4		

It is observed that the flow channels formed by the corrugations of the confinement wall need have a relatively large aspect ratio: typically 2 cm wide in the azimuthal direction and an order of magnitude wider in the radial direction. Despite of the narrowness of these channels and the relatively small flow velocity, the Pb flow will be turbulent.

The primary conclusion from the results of the two scores of cases summarized in Tables 1 and 2 is that it is feasible to design an ENHS for 125 MW_{th} with 100% natural circulation, 4 mm thick confinement wall and primary-to-secondary temperature drop not exceeding 50 °C. Another important conclusion is that using a cover gas lift pump it is possible to significantly reduce the ENHS module height, diameter and, hence, weight. It appears that an ENHS can be designed to have a diameter of ~2.5 m, a height of ~20 m, and a total weight of ~400 tons. With the Pb level limited to the upper level of the fuel rods, the total transportation weight of the ENHS is estimated to be ~200 tons. Such a module can be transported fully assembled and fuelled from the factory to the site using existing transportation and lifting means.

Use of cover gas lift pump also provides more flexibility in the selection of the core dimensions for the ENHS and in the startup and load following capability of the ENHS.

A detailed layout of representative ENHS designs has recently been initiated. It appears that we have practical design solutions to major feasibility issues such as how to fabricate and install the confinement wall, how to seal it, and how to accommodate the differential thermal expansion of the structural and confinement walls. Also recently initiated is a thorough structural analysis of the ENHS vessel design. Preliminary results indicate that 4 mm will be an acceptable thickness for the confinement wall.

CORE DESIGN DOMAIN FOR ENHS

One goal of the ENHS is to have a long life core without refueling or fuel shuffling. Another goal of the ENHS is to have this long-life core with a nearly zero burnup reactivity swing. This section summarizes our findings concerning the feasibility of designing a nearly zero burnup reactivity swing core to operate for 15 EFPY.

The design variables of this study are the core height, fuel rod diameter, clad thickness, lattice pitch-to-diameter ratio (p/d), initial plutonium weight percent, total core thermal power and average linear heat rate. Of these, the fuel rod diameter and clad thickness were fixed at 1 cm and 0.1 cm, respectively. The fuel is a metallic alloy of U-Pu with 10 weight % Zr; its density is assumed to be 75% of the nominal density. The uranium is depleted to 0.2

$\text{w}/\%$ ^{235}U and the Pu initial composition is typical to that from LWR spent fuel. The clad is represented by stainless steel having 64.8 $\text{w}/\%$ Fe, 17 $\text{w}/\%$ Cr, 14 $\text{w}/\%$ Ni, 2.8 $\text{w}/\%$ Mo and 1.5 $\text{w}/\%$ Mn. In reality, lower Ni contents steel will be used with the Pb coolant. Two power levels and two average linear heat rates have been considered: 125 and 250 MW_{th} , and 80 and 120 w/cm .

The fuel assembly is assumed to have 217 rods in an hexagonal array. The central assembly site is reserved for a safety rod assembly. There are no control rods. The fine reactivity tuning is done using the radial reflector. The reflector assembly includes a voided container that is 15 cm thick and as long as the fuel section in the core. The minimum reactivity worth of the reflector is obtained when the cavity level coincides with the fuel level in the core. The maximum reactivity worth is obtained when the cavity is all below the core, so that Pb surrounds the core. Each fuel rod has a fission gas plenum above the fuel section; the plenum length is taken to be 75% of the fuel section length.

The core design is subjected to a couple of constraints: the peak fuel burnup should not exceed 150 GWD/tHM , and the peak fluence of $E > 0.1$ MeV neutrons should not exceed $4 \times 10^{23} \text{ n}/\text{cm}^2$. Other restraints on the core design are: (a) Maximum fuel length is 4 m. (b) Minimum p/d is 1.15.

The calculations are done using MCNP¹² and ORIGEN-2¹³ managed by the MOCUP code¹⁴. MCNP calculates k_{eff} , zone-wise fluxes, relative power and effective one group averaged cross sections for use in ORIGEN-2. ORIGEN-2 calculates the change in the isotopic composition as a function of time. The results reported below were obtained by performing the MCNP calculations once every year of reactor full power operation.

Figure 6 defines the combination of core height and p/d ratio that have nearly zero burnup reactivity swing for 15 years of full power operation. The two curved plots shown in the figure each represents a family of cores for a given total power output – 125 MW_{th} for the left-hand plot and 250 MW_{th} for the right-hand plot. The core diameter is uniquely defined by the core height, p/d value, fuel rod outer diameter (1.2 cm, in the present analysis), average linear heat rate (80 w/cm) and core thermal power. The bottom boundary is the criticality constraint. The dashed line saying “Minimum critical height for $dk_{\text{eff}} = 1.0\%$ temperature reactivity loss” defines the minimum height of cores whose reflector reactivity worth is 1.6%. As the unreflected cores of Figure 6 are designed to have $k_{\text{eff}} = 0.994$ when at room temperature, the room temperature k_{eff} of the reflected cores is 1.01. The 1.0% excess reactivity is to compensate for the burnup reactivity swing. If the

burnup reactivity swing is limited to 0.5%, the dotted line will move downwards, increasing the core design domain.

Of the family of acceptable cores defined by Figure 6, the 2m high core having a p/d of 1.3 appears attractive. The corresponding p is 1.56 cm. Its fuel has 11.4 weight % Pu. This core enables designing relatively compact ENHS modules, as illustrated by Case 9.2.5 of Table 2. By designing the ENHS central absorber to compensate for the reactivity swing from room temperature up to 350 °C, as done in the 4S reactor^{5,6}, the 1.5 m high core having p/d of approximately 1.34 will become acceptable; its reflector reactivity worth will be able to compensate for the burnup reactivity swing.

An illustration of the burnup reactivity swing is given in Figure 7. It pertains to the 2m high core having a p/d of 1.30 of Figure 6. k_{eff} varies by not more than ~0.5% over 30 years of EFPY of operation! The fluctuations in k_{eff} are stochastic; they are due to the use of the MCNP Monte-Carlo code for calculating k_{eff} . The optimal k_{eff} versus burnup profile may be somewhat different from that of Figure 6; either flatter or having a slightly negative slope. In the latter case, the reactivity loss is to be compensated by the constant automatic creeping movement of the reflector.

The cores defined in Figure 6 (by the two curved lines) all reach the peak fast neutron fluence constraint after ~15 EFPY. For a 1cm fuel rod diameter (inner), this constraint limits the linear heat rate to 80 w/cm . The resulting peak burnup is ~105 GWD/tHM .

It is concluded that lead-cooled cores can be designed to operate for 15 EFPY without any refueling operation and to have a very small burnup reactivity swing. Details about the neutronic characteristics of these cores can be found in reference 15.

CONCLUDING REMARKS AND DISCUSSION

Using lead for the coolant, the ENHS can be designed to have 100% natural circulation and deliver 125 MW_{th} to the secondary coolant through a 4mm thick confinement wall with no more than 50 °C temperature drop. In fact, there is a good synergism between the requirements for limiting the primary-to-secondary temperature drop to 50 °C and between the requirement for 100% natural circulation. Preliminary structural analysis indicates that 4 mm is an adequate thickness for the confinement wall. This analysis continues.

Cover-gas lift-pump appears very promising for the ENHS; it can significantly reduce the volume and weight of the ENHS module and increase its operational flexibility and stability without

introducing into the module any rotating components or electrical components. Feasible diameter and height of an ENHS module for 125 MW_{th} are, respectively, 2.5 m and 20 m. Its weight for transportation, when loaded with fuel and solidified Pb is less than 200 tons. Hence, it appears that the ENHS could be transported from the factory fully assembled and fuelled.

The design domain has been defined for Pb-cooled cores that can deliver 125 to 250 MW_{th} for 15 EFPY with very small burnup reactivity swing; approximately 0.5%. What limits the core life is the radiation damage to the clad. The radiation damage constraint limits the average linear heat-rate of a core that is to operate for 15 EFPY to 80 w/cm. Had 10 EFPY been acceptable life for the ENHS, its core could have been designed for an average linear heat rate of 120 w/cm. The peak burnup of these cores is approximately 100 GWD/tHM.

It turns out that there is a good match between the core height and pitch-to-diameter values required for providing the desirable neutronic characteristics and between the requirements for 100% natural circulation. Thus, it is possible to design an ENHS that has long life with close to zero burnup reactivity swing to also have 100% natural circulation.

In summary, so far we found no "show stopper"; satisfactory solutions were found for all the feasibility issues considered. The ENHS continues to be a promising concept for Generation IV reactors.

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Fig. 1 A schematic vertical view of a single ENHS module reactor.

Fig. 2 A schematic top view of the single ENHS module reactor pool. The number of steam generators, reheaters (if used) and pumps is yet to be determined.

Fig. 3 A schematic vertical cut through an ENHS design having 100% natural circulation. Confinement wall having vertical corrugations extend to below the core level. Not to scale.

NOTE:

OUTER BAND IS
WELDED TO OUTER
STRUCTURAL WALL.
INNER BAND IS
NOT WELDED TO
INNER STRUCTURAL
WALL.

Fig. 4 A horizontal cut through the corrugated wall of an ENHS module.

Fig. 5 A schematic vertical cut through an ENHS with a cover-gas "lift-pump". Outer structural wall does not have a special shield. Corrugated confinement wall extend to below the core level. Not to scale.

Fig. 6 Design domain of 80 w/cm cores for ENHS featuring nearly zero burnup reactivity swing. Peak fluence constraint limits life to ~15 EFPY.

Fig. 7 k_{eff} evolution with operating time of the 2 m high core having a p/d of 1.3. Fuel inner/outer diameter is 1.0/1.2 cm.

LONG-LIFE CORES WITH SMALL BURNUP REACTIVITY SWING

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ABSTRACT

The neutronic feasibility of designing small and simple lead cooled and reflected cores to operate up to the metallurgical limit with nearly zero burnup reactivity swing is investigated. These cores are loaded with uniform composition fuel, they have no blanket assemblies, and they operate without any fuel shuffling. The design domain has been defined for such cores that can deliver 125 to 250 MW_{th}. The radiation damage to the clad limits the average linear heat-rate of a core that is to operate for 15 effective full power years to 80 w/cm, when the fuel is 1 cm in diameter. The corresponding peak burnup of these cores is approximately 100 GWD/tHM. The dependence of selected neutronic characteristics of the nearly zero burnup reactivity swing Pb-cooled cores on selected core design variables was established.

1. INTRODUCTION

If a core could be designed to have a uniform fuel composition and to maintain, without any fuel handling, a nearly constant k_{eff} up to the radiation damage limit, it will be very attractive for the following reasons: (a) Good proliferation resistance. (b) Simplified control. (c) Improved safety (d) Reduced O&M costs. (e) Improved availability.

For these reasons one of the design goals set for the recently conceived¹ Encapsulated Nuclear Heat Source (ENHS) reactor core was to have a 15 years life with nearly zero burnup reactivity swing. In a preliminary study we illustrated^{2,3} that this might be feasible in small Pb-cooled and reflected cores that are fueled with either Pu-U-Zr(10^{w/o}) or U-Zr(10^{w/o}).

The primary purpose of the present work is to establish the range of core design variables that enable operating the core for 15 Effective Full Power Years (EFPY) with nearly zero reactivity swing without any fuel handling and without exceeding the radiation damage related constraints. The study focuses on lead-cooled cores that have uniform initial composition and do not have any blanket elements or special reflector elements; the lead that surrounds the core provides also for neutron reflection. Another purpose of this work is to explore the neutronic characteristics of

the nearly zero burnup reactivity swing Pb-cooled cores and the systematics of the dependence of these characteristics on the selected core design variables.

Even though Pb is the only coolant specifically considered in this work, all the reported results also apply to Pb-Bi coolant.

In Section 2 we briefly review prior art in the design of close to zero burnup reactivity swing cores. The ENHS core model considered in this work is described in Section 3. Section 4 describes the computational method along with the set of assumptions used. Section 5 describes the dependence of selected neutronic characteristics on core design variables. The design domain identified for nearly zero burnup reactivity swing cores for 15 EFPY is summarized in Section 6. Section 7 discusses the implications of certain of the approximation used in this work. Conclusions are presented in Section 8.

2. PRIOR ART

Orlov and Adamov^{4,5} appears to be the first to suggest that Pb-cooled reactors can be designed to have a nearly zero burnup reactivity swing. One of their reactor design goals was to restrict the available excess reactivity to less than one dollar. To achieve this they have a relatively often refueling – once every year. The fuel residence time in the core is 5 years. The peak discharge burnup is between 6 to 12 percent. The average linear heat rate varies between 210 and 320 w/cm.

Sekimoto^{6,7} et al. designed small Pb-cooled cores using either metallic alloy or nitride fuel. They considered three-zone cores that are to operate for more than 10 years without refueling or fuel shuffling, while maintaining burnup reactivity swings to less than 0.1% $\Delta k/k$. Other restrictions included a negative whole core void coefficient of reactivity over all of the burnup period.

C. Rubbia⁸ et al. designed a Pb-cooled slightly subcritical core that is to be driven by an intense accelerator neutron source. One of their, so called, Energy Amplifier design features is a nearly burnup independent k_{eff} value.

A couple of features that are common to all these core designs are (a) fast spectrum, and (b) breeding ratio that is slightly above unity. The second feature implies that the fissile fuel content is very slightly increasing with burnup. The first feature implies that the accumulation of fission products (FP) does not significantly impair the neutron balance; a slight buildup of fissile fuel content can compensate for the negative reactivity effect of the FP.

The notion that any combination of fertile fuel, coolant, structural materials and possibly moderator has its equilibrium fissile fuel content was established many years ago; see for example reference 9. That reference considered lattices using U-Pu oxide fuel that are cooled by regular or heavy water. It showed that the smaller is the water-to-fuel volume ratio, the higher becomes the equilibrium fissile fuel contents. For the k_{∞} corresponding to an equilibrium fissile fuel content to exceed unity, the neutron spectrum need be hard. This implies using a very small water-to-fuel volume ratio. The added advantage of a hard spectrum is small neutron absorption

probability by the accumulating FP. By designing the core to slightly breed, it is possible to compensate for the poisoning effect of the FP so as to maintain k_{∞} practically burnup independent.

3. THE ENHS CORE MODEL

The ENHS model used for the neutronic analysis is described in figure 1 and in table I. The layout of the core and adjacent components are similar to those of the 4S reactor^{10,11}. The system considered in this work differs from that of the 4S reactor in the coolant type, the composition of the radial reflector and shield, and in the core dimensions.

The core consists of fuel assemblies having 217 rods in a hexagonal array. The central assembly site is reserved for a safety rod assembly. During reactor operation this central site is filled with Pb. There are no control rods. The fine tuning of reactivity is done by axial movement of the radial reflector. Also, there are no blanket assemblies and no reflector assemblies. The Pb that surrounds the core serves as the reflector. All the fuel rods are of a uniform composition. Each fuel rod has a fission gas plenum above the fuel section; the plenum length is taken to be 75% of the fuel section length. The fuel is a metallic alloy of U-Pu with 10 weight % Zr; its density is assumed to be 75% of the nominal density. The uranium is depleted to 0.2 % ²³⁵U and the Pu initial composition is typical to that from LWR spent fuel: the weight % of the isotopes 239, 240, 241 and 242 is, respectively, 67.2, 21.7, 6.4, and 4.7. The clad is represented by stainless steel having 64.8 % Fe, 17 % Cr, 14 % Ni, 2.8 % Mo and 1.5 % Mn. In reality, lower Ni contents steel such as HT-9 will be used with the Pb coolant. The same SS composition is assumed for all the structural components modeled for the neutronic analysis.

The central absorber (denoted in figure 1 as the "safety rod region") is securely latched inside the core as long as the coolant temperature is below 350C. This is accomplished with an electromagnetic latch that connects the central absorber to its withdrawal mechanism; the latch does not engage until its temperature exceeds 350C. After the central absorber is withdrawn, all the reactivity control is done with the radial reflector assembly. That is, the reflector is to compensate for the reactivity swing due to temperature increase above 350C and due to fuel burnup. The nominal core inlet, outlet and average temperatures are assumed to be 420C, 540C and 480C.

The reflector assembly includes a voided container that is 15 cm thick and as long as the fuel section in the core. The minimum reactivity worth of the reflector is obtained when the cavity level coincides with the fuel level in the core. The maximum reactivity worth is obtained when the cavity is all below (or above) the core, so that Pb surrounds the core. Additional information on the ENHS design can be found in reference 12.

4. COMPUTATIONAL METHOD AND ASSUMPTIONS

The calculations are done using MCNP¹³ and ORIGEN2¹⁴ managed by the MOCUP code¹⁵. MCNP calculates k_{eff} , zone-wise fluxes, relative power and effective one group averaged cross

sections for use in ORIGEN2. ORIGEN2 calculates the change in the isotopic composition as a function of time. Most of the burnup dependent results reported below were obtained by performing the MCNP calculations once every year of reactor full power operation, treating the core as a single zone. That is, the isotopic composition across the core is uniform from beginning-of-life (BOL) to end-of-life (EOL). The effect of finer burnup time steps and of space-dependent depletion is discussed in Section 7.

Tables II and III define the cross sections used for the MCNP calculations. Not all the FP with non-negligible absorption probability are accounted for. In Section 7 we will estimate the effect of the FP not accounted for. MCNP generated effective one-group cross sections of all these isotopes are used for the ORIGEN2 burnup analysis. For isotopes not included in MCNP analysis, ORIGEN2 uses cross sections from its pre-processed libraries. For the present study we used ORIGEN2 libraries number 312 for actinides and 313 for FP; they were both intended for LMFBR analysis and were generated using a hard neutron spectrum. In Section 7 we'll check the sensitivity of the results to the ORIGEN2 library selection.

The design variables of this study are the core height, lattice pitch-to-diameter ratio (p/d), initial plutonium weight percent, total core thermal power and average linear heat rate, q'_{av} . Three power levels and two q'_{av} values are considered: 62.5, 125 and 250 MW_{th}, and 80 and 120 w/cm. The fuel rod diameter and clad thickness were fixed at 1 cm and 0.1 cm, respectively.

The core design is subjected to a couple of constraints: the peak fuel burnup should not exceed 150 GWD/tHM, and the peak fluence of $E > 0.1$ MeV neutrons should not exceed 4×10^{23} n/cm². Other restraints on the core design are: maximum fuel length is 4 m and minimum p/d is 1.15.

The critical Pu loading – the Pu weight % to be loaded in the fuel for a given core, is determined so that the k_{eff} value for a cold and clean core will be 0.994. This provides a margin to criticality of 1.3 dollars, as the average delayed neutron fraction for a typical Pb-cooled core, estimated in table IV, is 0.0046. This approach is conservative; in reality it is not acceptable to get the core critical when its temperature is below the coolant melting temperature. As described in Section 3, the ENHS is designed to be at shutdown conditions as long as its coolant temperature is below 350C. A practical design approach would be designing the Pu loading to provide a k_{eff} of ~ 0.995 when the core is at 350C.

The Initial Conversion Ratio (ICR) considered in this work is the conventional definition: the ratio of neutron capture rate in ²³⁸U and ²⁴⁰Pu to neutron absorption rate in ²³⁵U, ²³⁹Pu and ²⁴¹Pu. It turns out that in the hard spectrum of Pb cooled reactors, the even fuel isotopes contribution to fission is not negligible. For example, table IV shows that the ²³⁸U fission rate is second to that of ²³⁹Pu; it is twice as large as that of ²⁴¹Pu. Also shown in table IV is that the BOL fission rate of ²⁴⁰Pu is $\sim 60\%$ of that of ²⁴¹Pu. Hence, the conventional ICR definition used in this work is not a good characterization of the rate of change of the fuel quality.

The flux amplitude used for burnup analysis in ORIGEN2 is obtained from the MCNP results through use of the MOCUP “flux normalization factor” f_{mt} . It is calculated as follows: $f_{mt} (1/s) = \nu P / (1.602 \times 10^{-19} \times k_{eff} \times R_a)$, where ν is the average number of neutrons per fission, P is the

reactor thermal power in MW, and R_a is the average recoverable energy, in MeV, per fission. A value of 204.5 MeV/fission recommended by one of the MOCUP developers¹⁶ was used for R_a throughout this study. Using MCNP BOL fission probability data and isotope dependent recoverable energy data we are estimating, in table V, the recoverable energy for a typical Pb cooled reactor considered in this work to be 209.2 MeV/fission. This implies that the actual power level can be approximately 2% higher than the one assumed for the present analysis.

5. NEUTRONIC CHARACTERISTICS SYSTEMATICS

A parametric study has been performed with the aim of establishing the dependence of various system characteristics on four core design variables: core height, lattice p/d ratio, core power level and average linear heat rate of the fuel rods. The trends obtained for 250 MW_{th} cores that are designed to have a q'_{av} value of 120 w/cm are shown in figures 2 through 12. Supplementary results are provided in tables VI and VII. Following are several observations concerning the systematics of selected neutronic characteristics:

1. The critical Pu concentration increases (Figure 2) and, correspondingly, the ICR decreases (Figure 3) as p/d increases. This is due to an increase of the absorption probability in Pb (Figure 7), to enhancement of the leakage probability (Figure 4) as well as to spectrum softening as p/d increases. In general, the higher is p/d for a given core height, the larger becomes the core diameter (tending to reduce the leakage probability) but the longer becomes the mean distance the neutron travel until it is absorbed (tending to increase the leakage probability). The latter effect is the dominant, as can be deduced from the one-group diffusion theory expression for the non-leakage probability of bare cores: $P_{NL} = [1 + B^2 L^2]^{-1}$. B^2 is the geometric buckling and L^2 is the diffusion area. The radial buckling $B_r^2 \propto R^{-2}$ whereas $L^2 = D/\Sigma_a$. The diffusion coefficient D is, to first order, independent of p/d. Σ_a on the other hand, is nearly proportional to the HM concentration which is $\propto R^{-2}$. Hence, $L^2 \propto R^2$ and $D_r^2 L^2$ is, to first approximation, independent of R . The axial buckling, B_a^2 is independent of R so that $B_a^2 L^2 \propto R^2 \propto (p/d)^2$.
2. The 2 m high cores require the lowest Pu concentration and offer the highest ICR whereas the 4 m high cores require the highest Pu concentration and have the lowest ICR (Figures 2 and 3). This is well correlated with the leakage probability of these cores (Figure 4). Of the three core heights considered, the 2 m high cores are the least leaky, as they have the smallest ratio of surface area to volume. The 1 m high core is the most sensitive to p/d as far as the critical Pu concentration and ICR are concerned. Figure 4 shows that these trends are correlated with the leakage probability dependence on p/d. The dominant leakage in the pancake shaped cores is in the axial direction. And as shown in "point 1" above, the axial leakage is the leakage component that increases with p/d.
3. The reactivity worth of the reflector is sensitive to the core height but not to p/d (Figure 5). This indicates that the radial leakage probability is closely independent of p/d. Combining the trends provided by figures 4 and 5 it appears that p/d affects, primarily, the axial leakage probability. These trends can be anticipated based on simple one-group diffusion theory analysis carried out in "point 1" above. The taller the core, the higher is the reflector reactivity worth. This expected trend is correlated with the tendency of the radial leakage probability to increase with a reduction in the core diameter, i.e., with an increase in the core height. The findings of figure 5 may have important practical implications: pancake shaped

cores may not have large enough reflector reactivity worth to compensate for the temperature and burnup reactivity swings.

4. The void reactivity worth is positive throughout the range considered for the core design variables. It increases with p/d and is not so sensitive to the core height (Figure 6). Voiding the core affects k_{eff} via three phenomena: spectrum hardening (positive effect), reduced parasitic absorption by Pb (positive), and enhanced leakage probability (negative). The reactivity worth reported in figure 6 corresponds to replacement of Pb by void throughout the core volume, but not outside the core. Creation of such a void is practically impossible; the boiling temperature of Pb is 2022K, significantly higher than the melting temperature of SS and of the metallic fuel. We can think of no scenario that can bring the Pb to even close to its boiling temperature. Even if there is a scenario that could have caused the Pb temperature to keep increasing, the clad and fuel will melt and dissolve in the Pb before the Pb will get to its boiling point. Dissolution of the fuel in the Pb will result in a large reactivity loss that is likely to be followed by reactor shutdown. The only conceivable voiding mechanism we could think of is a leakage of Pb from the reactor vessel. This mode of voiding turns out to have a negative reactivity effect, as illustrated in table VIII. This illustration pertains to 2 m high cores. The negative effect is due to the enhanced leakage probability that more than compensates for the positive effects of spectrum hardening and reduction in the parasitic neutron absorption probability. The couple of positive values in table VIII are probably due to statistical uncertainties in the Monte Carlo calculations.
5. Table IX gives the hot-to-cold reactivity swing for selected cores. This reactivity swing consists of four components: (a) Pb density change. (b) Fuel axial expansion. (c) Radial expansion of core support structure. (d) Doppler effect. All these components are the difference in k_{eff} values between the cold temperature and 700C or 480C for, respectively, the Doppler effect or the rest of the effects. Two “cold” temperatures are considered: (a) Room temperature, taken to be 20C, and (b) 350C – the temperature above which it is possible to withdraw the central absorber rod; at lower temperatures it is impossible to bring the reactor to criticality. The Pb density is assumed to be 11.368 g/cm^3 at 20C and $11.072-1.2[T(K)-273] \text{ g/cm}^3$ for T in between 673K and 973K. The coefficient of linear expansion was assumed to be $1.78 \times 10^{-5} \text{ K}^{-1}$ for the core support structure and $1.9 \times 10^{-5} \text{ K}^{-1}$ for the fuel. Upon expansion, the core is assumed to maintain a cylindrical shape; that is, we do not account for enhanced radial expansion of the upper part of the core that is the hottest.
6. The effect of the core power level on several of the characteristics considered above is illustrated in figures 8 through 12. The results pertain to 4 m high cores operating at a q'_{av} of 120 w/cm. The number of fuel rods of these cores is proportional to the power level. For a given p/d , the core diameter is closely proportional to the square root of the power level. That is, the lowest power cores have the largest radial leakage probability, making their reflector reactivity worth the highest (Figure 12) and their void reactivity effect the smallest (Figure 11).
7. The effect of the average fuel linear heat rate on the neutronic characteristics is inversely proportional to the effect of the core power level; the larger the linear heat rate, the smaller is the total length of fuel rods needed for generating a given power level. For a given core height, the core diameter is closely inversely proportional to q'_{av} . Thus, the dimensions of a core for 250 MW_{th} having q'_{av} of 120 w/cm will be identical to the dimensions of a core for

- 125 MW_{th} with q'_{av} of 60 w/cm when both have the same p/d ratio. Cores designed to deliver 125 MW_{th} with q'_{av} of 80 w/cm will be only slightly smaller.
8. It is also found that the power density shape is not sensitive to the core dimensions. Typically, the peak-to-average power density is between 1.8 to 2.0.
 9. Figures 13 and 14 illustrate the sensitivity of the burnup reactivity swing to the p/d ratio. The fluctuations are stochastic in nature; they are due to the use of MCNP for calculating k_{eff} and, even more so, for generating the effective one-group cross sections for the burnup analysis. It is observed that for a core of a given power, height and q'_{av} , there exists a p/d ratio for which k_{eff} can stay practically constant for many years of operation. This happens, for example, for 13 years of effective full power operation of the core for 250 MW_{th}, 4 m, 120 w/cm core of figure 13 having a p/d ratio of 1.30. Designing a similar core to have a slightly larger p/d ratio will lead to k_{eff} that is monotonously declining with operation time. This is illustrated by the p/d = 1.375 core of figure 13 and by the p/d = 1.60 core of figure 14. The burnup reactivity swing of the last two cores over 15 effective full power years (EFPY) is, respectively, approximately 1.5% and 0.8%. This reactivity swing can be compensated by an automated constant movement of the radial reflector at an infinitesimally small speed – on the order of the 1 mm/day proposed for the 4S reactor^{7,8}. When p/d is smaller than a certain value, k_{eff} will tend to initially increase with burnup, as illustrated for the p/d = 1.525 case of figure 14.
 10. The evolution of the concentration of fuel isotopes with EFPY of operation is illustrated in figures 15 and 16. The data in these figures is normalized to the BOL HM inventory. It is observed that the inventory of the two most abundant Pu isotopes – 239 and 240 slightly increases for the first 15 EFPY of operation, while that of the higher Pu isotopes and of ²³⁵U is dropping. Comparing figures 15 and 16 it is observed that, in the p/d range considered, the relative change in the fuel isotope concentration is not sensitive to the core p/d ratio.

6. DESIGN DOMAIN FOR ZERO BURNUP REACTIVITY SWING

Guided by the above information we proceeded to define the combination of core design variables that will yield nearly zero burnup reactivity swing for, at least, 15 EFPY. Figures 17 and 18 define the combination of core height and p/d ratio that have nearly zero burnup reactivity swing for 15 years of full power operation. The two curved plots shown in each of the figures each represents a family of cores for a given total power output – 125 MW_{th} for the left hand plot and 250 MW_{th} for the right-hand plot. The core diameter is uniquely defined by the core height, p/d value, fuel rod outer diameter (1.2 cm, in the present study), q'_{av} (80 or 120 w/cm) and core thermal power. The bottom boundary is the criticality constraint. The dashed line saying “Minimum critical height for $dk_{eff} = 1.0\%$ temperature reactivity loss” defines the minimum height of cores whose reflector reactivity worth is 1.6%. As the unreflected cores of figures 17 and 18 are designed to have $k_{eff} = 0.994$ when at room temperature, the room temperature k_{eff} of the reflected cores is 1.01. The 1% excess reactivity is to compensate for the burnup reactivity swing. The dotted line for 0.28% of dk_{eff} due to hot-to-cold temperature swing corresponds to 350C as the cold temperature for the core design. This line provides the most realistic bound for the core design domain.

Of the family of acceptable cores defined by figure 17, the 2 m high core having a p/d of 1.3 appears attractive. This core enables designing relatively compact ENHS modules, as illustrated

by Case 9.2.5 of table 2 of reference 12. Figure 19 shows the burnup reactivity swing of this core. k_{eff} varies by not more than $\sim 0.5\%$ over 30 years of EFPY of operation! The fluctuations in k_{eff} are stochastic; they are due to the use of the MCNP Monte-Carlo code for calculating k_{eff} . The optimal k_{eff} versus burnup profile may be somewhat different from that of figure 19; either flatter or having a slightly negative slope. The reactivity loss will be compensated, in the latter case, by a constant automatic creeping movement of the reflector.

The cores defined in figure 17 and 18 (by the two curved lines) all reach the peak fast neutron fluence constraint after ~ 15 EFPY. For a 1 cm fuel diameter, this constraint limits the linear heat rate to 80 w/cm. The resulting peak burnup is ~ 105 GWD/tHM. Had 10 EFPY been acceptable life for the ENHS, its core could have been designed for an average linear heat rate of 120 w/cm.

7. DISCUSSION

A number of approximations were used in the calculations the results from which were reported above. In the following we shall estimate the significance of these approximations.

7.1 UNIFORM HOMOGENEOUS CYLINDRICAL CORE MODEL

The core consists of hexagonal heterogeneous unit cells assembled into fuel assemblies. A core made of hexagonal fuel assemblies can not be exactly cylindrical. The question addressed is what error is introduced by volume averaging the heterogeneous unit cell and by cylinderizing the core. Both effects were found negligible.

7.2 SINGLE ZONE BURNUP ANALYSIS

All the burnup dependent results reported above were obtained from a single zone analysis; that is, ignoring space dependent composition. Figure 20 compares k_{eff} evolution as calculated for the same core using two approaches: (a) A 10 zone analysis: the core is divided into ten equal volume radial zones. The burnup of each of the 10 zones is analyzed separately. (b) A single zone analysis. This comparison is done for a 4 m high 250 MW_{th} core having $p/d = 1.15$ and $q'_{\text{av}} = 120$ w/cm. It is observed that the single zone analysis tends to overestimate the value of k_{eff} for long operating times. Figure 21 shows that the buildup of ^{239}Pu in the outermost zone of the 10 zone core is significantly more pronounced than its buildup in the inner zones. The enhanced buildup in the outer zone is due to the effect of the radial reflector. As the importance of fission neutrons born near the core boundary is smaller than the importance of fission neutrons born closer to the core center, the non-uniform buildup of Pu is responsible for a lower k_{eff} . The implication of this finding is that the p/d value for the real core will have to be slightly smaller than the p/d value found optimal using a single zone analysis. The k_{eff} evolution of the real core can be as flat as calculated using the single-zone approximation.

7.3 BURNUP TIME STEPS

All the burnup analysis reported so far made one MCNP calculation every one EFPY. If the neutron spectrum in the core varies significantly during one year, MCNP should recalculate the

effective one-group transmutation cross sections more frequently. Figure 22 compares the k_{eff} evolution with operation time as calculated using two frequencies for MCNP calculations: 0.1 and 1 year per calculation. No difference is observed in the general trend. It is concluded that once a year is a sufficient frequency for MCNP calculations of our Pb cooled systems.

7.4 UNACCOUNTED FOR FISSION PRODUCTS

Forty seven FP are accounted for in the MCNP calculations; they are listed in table III. MCNP cross sections for other FP were not available to us. Using the ORIGEN2 FP data we have estimated that the FP used in the MCNP calculations account for 73% of the absorption by all the FP. We also estimated that the reactivity effect of the FP not accounted for in MCNP is, approximately, 1% after 15 EFPY. This small, but non-negligible effect can be compensated for by a slight reduction in the p/d value.

7.5 SENSITIVITY TO FISSION PRODUCTS LIBRARY

A number of FP libraries for LMFBR are found in the ORIGEN2 code package. Libraries number 312 and 313 were used for the calculations reported upon above. One run was done using libraries number 372 and 373. Figure 23 compares the evolution of k_{eff} as calculated using the two sets of libraries. It is observed that the results are insensitive to the libraries used.

CONCLUSIONS

It is concluded that small and simple lead cooled and reflected cores that are loaded with uniform composition fuel and without blanket assemblies can be designed to operate without any fuel handling up to the metallurgical limit with nearly zero burnup reactivity swing. The range of design variables of such cores has been identified. The dependence of selected neutronic characteristics of the nearly zero burnup reactivity swing cores on core design variables was established.

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Table I. Zone Dimensions and Composition in ENHS Model of Fig. 1

No.	Item	Material (Volume Fraction)			Diameter (cm)	Height (cm)	Cold Temp.(K)	Hot Temp.(K)
		Lead	SS	Fuel				
1	Safety rod region	0.99	0.01	-	10.67332	930	293	753
2	Lower shield	1.00	-	-	53.36822	100	293	693
3	Lower core support	-	1.00	-	53.36822	30	293	693
4	Core	0.31	0.21	0.48	53.36822	400	293	753
5	Fission gas plenum	0.31	0.21	-	53.36822	300	293	813
6	Upper shield	1.00	-	-	53.36822	100	293	813
7	Inner Pb gap	1.00	-	-	56.36822	930	293	753
8	Core barrel	-	1.00	-	57.36822	930	293	753
9	Outer Pb gap	1.00	-	-	58.36822	930	293	693
10	Reflector	1.00	-	-	73.36822	400	293	693
11	Cavity	-	0.10	-	73.36822	400	293	693
12	Upper cavity	0.90	0.10	-	73.36822	400	293	693
13	Reflector guide	0.70	0.30	-	75.36822	930	293	693
14	Down comer	1.00	-	-	82.36822	930	293	693
15	Shield	1.00	-	-	126.8682	930	293	693
16	Vessel	-	1.00	-	127.8682	930	293	693

Table II. Actinides and Cross-Sections used for MCNP Calculations(Temp. in Kelvin)

Nuclides	Cross Section Data
For cold condtion	
^{234,235,236,238} U(293K)	92234.60c, 92235.60c, 92236.60c, 92238.60c
^{238,239,240,241,242} Pu(293K)	94238.60c, 94239.60c, 94240.60c, 94241.60c, 94242.60c
²³⁷ Np(293K)	93237.60c
^{241,243} Am(293K)	95241.60c, 95243.60c
^{242,243,244,245,246} Cm(293K)	96242.60c, 96243.60c, 96244.60c, 96245.60c, 96246.60c
For hot condtion	
^{234,235,236,238} U(755K)	92234.60c, 92235.60c, 92236.60c, 92238.60c
^{238,239,240,241,242} Pu(755K)	94238.60c, 94239.60c, 94240.60c, 94241.60c, 94242.60c

Table III. Fission Products and Cross-Sections used for MCNP Calculations (Temp. in K)

Nuclides	Cross Section Data
^{83,84,86} Kr(294K)	36083.59c, 36084.59c, 36086.59c
^{85,87} Rb(294K)	37085.55c, 37087.55c
^{90,91,92,93,94,96} Zr(600K)	40090.78c, 40091.78c, 40092.78c, 40093.78c, 40094.78c, 40096.78c
⁹⁵ Mo(294K)	42095.50c
⁹⁹ Tc(294K)	43099.60c
¹⁰¹ Ru(294K)	44101.50c
¹⁰³ Rh(600K)	45103.78c
^{105,108} Pd(294K)	46105.50c 46108.50c
¹⁰⁹ Ag(755K)	47109.84c
^{112,113} Cd(600K)	48112.78c, 48113.78c
^{127,129} I(600K)	53127.78c, 53129.78c
^{131,132,134,135,136} Xe(600K)	54131.78c, 54132.78c, 54134.78c, 54135.78c, 54137.78c
^{133,135,137} Cs(600K)	55133.78c, 55135.78c, 55137.78c
^{134,137,138} Ba(600K)	56134.78c, 56137.78c, 56138.78c
¹⁴¹ Pr(294K)	59141.50c
^{143,145} Nd(294K)	60143.50c, 60145.50c
¹⁴⁷ Pm(294K)	61147.50c
^{147,149,150,151,152} Sm(294K)	62147.50c, 62149.50c, 62150.50c, 62151.50c, 62152.50c
^{153,154,155} Eu(294K)	63153.60c, 63154.50c, 63155.50c
^{155,156} Gd(755K)	64155.84c, 64156.84c

Table IV. Average Delayed Neutron Fraction for a 250MW, 4 m High Core of p/d=1.15 and q'av=120W/cm

	Fission Rate	Average v	β value
²³⁵ U	0.005544	2.47	0.0068
²³⁸ U	0.058367	2.53	0.0158
²³⁹ Pu	0.229850	2.95	0.00215
²⁴⁰ Pu	0.017510	2.87	0.0031
²⁴¹ Pu	0.027835	2.98	0.00515
²⁴² Pu	0.002732	2.88	0.0072
Average			0.00461

Table V. Average Recoverable Energy in a 250MW, 4 m High Core of p/d=1.15 and q'av=120W/cm

	Recoverable energy (MeV/fission)	Fission reaction rate from MCNP neutron balance
²³⁵ U	201.7	0.005281
²³⁸ U	202.8	0.058805
²³⁹ Pu	210.6	0.22802
²⁴⁰ Pu	211.0	0.017679
²⁴¹ Pu	211.3	0.027358
²⁴² Pu	211.7	0.002733
Average	209.2	

^aRecoverable energy (MeV/fission) = $0.00129927 Z^2 A^{0.5} + 33.12$ [14], where Z and A are the atomic number and the atomic mass,

Table VI. Data for 250MWth Cores

Case	Core height(m) p/d	4 1.15	4 1.30	4 1.45	4 1.60	2 1.15	2 1.30	2 1.45	2 1.60	1 1.15	1 1.30	1 1.45	1 1.60
Core thermal power (MWth)		250	250	250	250	250	250	250	250	250	250	250	250
Average linear heat rate (W/cm)		120	120	120	120	120	120	120	120	120	120	120	120
Net station efficiency (%)		40	40	40	40	40	40	40	40	40	40	40	40
Core height (cm)		400	400	400	400	200	200	200	200	100	100	100	100
Fission gas plenum length (cm)		300	300	300	300	150	150	150	150	75	75	75	75
Core diameter (cm)		106.8	120.6	134.6	148.6	149.4	169	188.4	208	210.2	237.6	265	292.6
Vessel diameter (cm)		255.8	269.6	283.6	297.6	298.4	318	337.4	357	359.2	386.6	414	441.6
Core inlet temp. (K)		753	753	753	753	753	753	753	753	753	753	753	753
Core outlet temp. (K)		813	813	813	813	813	813	813	813	813	813	813	813
No. fuel assemblies		24	24	24	24	48	48	48	48	96	96	96	96
No. shutdown assemblies		1	1	1	1	1	1	1	1	1	1	1	1
Pu enrichment* (wt%)		10.994	11.797	12.709	13.557	10.000	10.820	11.664	12.553	10.173	11.188	12.243	13.375
No. fuel rods/assembly		217	217	217	217	217	217	217	217	217	217	217	217
Fuel rod outer diameter (cm)		1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
Clad thickness (cm)		0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
Fuel rod pitch/diameter		1.15	1.3	1.45	1.6	1.15	1.3	1.45	1.6	1.15	1.3	1.45	1.6
Fuel material		Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr
Clad material		SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS
Result													
Initial Conversion Ratio(ICR)@ Cold		1.162944	1.103999	1.039082	0.990175	1.294979	1.218041	1.147972	1.083373	1.265015	1.164482	1.081527	1.002335
Reflector reactivity worth (dk)		0.02319	0.02375	0.02347	0.021	0.01363	0.01635	0.01258	0.01011	0.00341	0.00747	0.0038	0.00595
Void reactivity (dk)		0.01938	0.02802	0.04012	0.04476	0.02208	0.03118	0.0393	0.04343	0.01867	0.02552	0.03035	0.03231
Hot to cold reactivity swing** (dk)		-0.00466	-0.00175	-0.00227	-0.00015	-0.00464	-0.00342	-0.00016	0.00061	-0.00148	-0.00299	0.00143	-0.00199
All coolant loss reactivity swing (dk)		-0.12939	-0.15316	-0.17967	-0.21013	-0.05811	-0.07628	-0.10187	-0.12741	-0.03734	-0.05813	-0.08949	-0.11702
BOC fuel material													
²³⁸ U (t)		0.031162	0.030881	0.030562	0.030265	0.03151	0.031223	0.030927	0.030616	0.03145	0.031094	0.030725	0.030328
²³⁵ U (t)		15.54996	15.40965	15.25027	15.10217	15.72355	15.58031	15.43285	15.2776	15.69339	15.51598	15.33176	15.13387
²³⁹ Pu (t)		1.293271	1.387765	1.495103	1.594845	1.176375	1.272838	1.372156	1.476684	1.196718	1.31616	1.440227	1.573458
²⁴⁰ Pu (t)		0.417621	0.448132	0.482794	0.515002	0.379871	0.411021	0.443091	0.476846	0.38644	0.425009	0.465073	0.508098
²⁴¹ Pu (t)		0.123169	0.132168	0.142391	0.15189	0.112036	0.121223	0.130682	0.140637	0.113973	0.125348	0.137165	0.148853
²⁴² Pu (t)		0.090452	0.097051	0.104568	0.111544	0.082276	0.089023	0.095969	0.10328	0.083699	0.092053	0.10073	0.110048
Total (t)		17.50564	17.50565	17.50568	17.50572	17.50562	17.50564	17.50567	17.50566	17.50567	17.50564	17.50567	17.50565
With reflector													
Peak-to-average radial flux factor		1.28	N/A	N/A	1.32	1.37	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Peak-to-average axial flux factor		1.43	N/A	N/A	1.41	1.35	N/A	N/A	N/A	N/A	N/A	N/A	N/A

* Pu(U+Pu)

** Material expansion is neglected except for that of coolant

Table VII. Data for 125MWth cores

Case	Core height(m) p/d	4 1.15	4 1.3	4 1.45	4 1.6	2 1.15	2 1.3	2 1.45	2 1.6	1 1.15	1 1.3	1 1.45	1 1.6
Core thermal power (MWth)		125	125	125	125	125	125	125	125	125	125	125	125
Average linear heat rate (W/cm)		120	120	120	120	120	120	120	120	120	120	120	120
Net station efficiency (%)		40	40	40	40	40	40	40	40	40	40	40	40
Core height (cm)		400	400	400	400	200	200	200	200	100	100	100	100
Fission gas plenum length (cm)		300	300	300	300	150	150	150	150	75	75	75	75
Core diameter (cm)		77	87	97	107	106.8	120.6	134.6	148.6	149.4	169	188.4	208
Vessel diameter (cm)		226	236	246	256	255.8	269.6	283.6	297.6	298.4	318	337.4	357
Core inlet temp. (K)		753	753	753	753	753	753	753	753	753	753	753	753
Core outlet temp. (K)		813	813	813	813	813	813	813	813	813	813	813	813
No. fuel assemblies		12	12	12	12	24	24	24	24	48	48	48	48
No. shutdown assemblies		1	1	1	1	1	1	1	1	1	1	1	1
Pu enrichment* (wt%)		13.59522	14.65678	15.77822	16.73567	11.53811	12.46244	13.45733	14.44044	11.14356	12.25389	13.4000	14.62367
No. fuel rods/assembly		217	217	217	217	217	217	217	217	217	217	217	217
Fuel rod outer diameter (cm)		1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2	1.2
Clad thickness (cm)		0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
Fuel rod pitch/diameter		1.15	1.3	1.45	1.6	1.15	1.3	1.45	1.6	1.15	1.3	1.45	1.6
Fuel material		Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr	Pu-U-Zr
Clad material		SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS	SUS
Result													
Initial Conversion Ratio(ICR)@ Cold		0.903243	0.84931	0.801681	0.769016	1.092047	1.026838	0.967191	0.91334	1.13546	1.043783	0.969303	0.896342
Reflector reactivity worth (dk)		0.04402	0.04395	0.04479	0.04341	0.02426	0.02315	0.02505	0.02379	0.01126	0.01484	0.01114	0.01327
Void reactivity (dk)		0.00903	0.01359	0.01748	0.02172	0.0133	0.01885	0.0235	0.02473	0.01174	0.01382	0.01391	0.01398
Hot to cold reactivity swing** (dk)		-0.00873	-0.00681	-0.00475	-0.0088	-0.0033	-0.00345	-0.0066	-0.00533	-0.00352	-0.00753	-0.0039	-0.00432
BOC fuel material													
²³⁸ U (t)		0.015126	0.01494	0.014744	0.014576	0.015486	0.015324	0.01515	0.014978	0.015555	0.015361	0.01516	0.014946
²³⁵ U (t)		7.547755	7.455009	7.357024	7.273411	7.727449	7.646691	7.559781	7.473909	7.76186	7.66491	7.564796	7.457906
²³⁹ Pu (t)		0.79966	0.862098	0.928056	0.984374	0.678661	0.733029	0.791548	0.849373	0.655452	0.720763	0.78818	0.86015
²⁴⁰ Pu (t)		0.258222	0.278388	0.299685	0.317872	0.219151	0.236707	0.255604	0.274277	0.211656	0.232746	0.254516	0.277757
²⁴¹ Pu (t)		0.076158	0.082104	0.088386	0.09375	0.064634	0.069812	0.075385	0.080893	0.062424	0.068644	0.075065	0.081919
²⁴² Pu (t)		0.055929	0.060295	0.064909	0.068848	0.047466	0.051268	0.055361	0.059406	0.045843	0.050411	0.055125	0.060159
Total (t)		8.752849	8.752832	8.752803	8.752831	8.752847	8.752832	8.752829	8.752836	8.75279	8.752834	8.752842	8.752837
With reflector													
Peak-to-average radial flux factor		N/A	N/A	N/A	N/A	1.29	N/A	N/A	N/A	1.4	N/A	N/A	N/A
Peak-to-average axial flux factor		N/A	N/A	N/A	N/A	1.36	N/A	N/A	N/A	1.23	N/A	N/A	N/A

* Pu(U+Pu)

** Material expansion is neglected except for that of coolant

Table VIII. Change in keff due to Pb Level Drop from in the Reactor Vessel.
2m High Core of 250MWth and 120 W/cm Average Linear Heat Rate.
Reference Pb Level is 4.5 m above Core Bottom

Pb Level from core bottom(m)	p/d			
	1.15	1.30	1.45	1.60
2.75	0.00006	0.00255	-0.00088	-0.00405
2.0 (core top)	-0.00265	-0.00279	-0.00945	-0.01298
1.0	-0.03341	-0.03741	-0.05346	-0.06376
0.0	-0.05625	-0.07101	-0.09436	-0.11884

Table IX. Hot-to-Cold Reactivity Swing for Selected Cores

Total thermal power(MWth) Case Core height(m) + p/d ALHGR(W/cm)	250 4+1.30 120	250 2+1.525 120	125 1+1.15 120	250 4+1.55 80	125 2+1.30 80
Hot to cold reactivity swing(dT=460K) (dk)	-0.01307	-0.00781	-0.01215	-0.00957	-0.00768
Hot to cold reactivity swing(dT=130K) (dk)	-0.00469	-0.00374	-0.00459	-0.00379	-0.00314

Table X. Evolution of Inventory of Fuel Isotopes
in the 125MW_{th}, 2m High Core Designed
with p/d=1.30 and q'_{av}=80W/cm

	YEAR						
	0	5	10	15	20	25	30
²³⁵ U	0.023261	0.019242	0.015963	0.01329	0.011104	0.009316	0.007856
²³⁸ U	11.60731	11.32986	11.05919	10.79629	10.53989	10.28944	10.04469
²³⁹ Pu	1.007076	1.049243	1.07846	1.097299	1.108009	1.112101	1.110856
²⁴⁰ Pu	0.325201	0.332238	0.339489	0.34669	0.353678	0.360402	0.366546
²⁴¹ Pu	0.095912	0.068625	0.051958	0.041839	0.03574	0.032145	0.030096
²⁴² Pu	0.070435	0.069167	0.067579	0.065848	0.064061	0.062287	0.060548
²³⁴ U	1.28E-21	1.15E-05	7.84E-05	0.000228	0.000462	0.000774	0.00115
²³⁶ U	1.29E-21	0.000985	0.00179	0.002449	0.002993	0.003445	0.003821
²³⁸ Pu	1.3E-21	0.000807	0.002855	0.005385	0.007984	0.010445	0.012663
²⁴¹ Am	1.32E-21	0.018059	0.02877	0.035067	0.038725	0.040815	0.041998
²⁴³ Am	1.33E-21	0.002098	0.003882	0.005391	0.006677	0.007757	0.008659
²³⁷ Np	1.3E-21	0.001027	0.002022	0.002952	0.003784	0.00452	0.005166
²⁴² Cm	1.32E-21	0.000208	0.00035	0.000436	0.000484	0.000509	0.000526
²⁴³ Cm	1.33E-21	1.69E-06	5.7E-06	1.02E-05	1.42E-05	1.76E-05	2.03E-05
²⁴⁴ Cm	1.33E-21	0.000103	0.000352	0.000682	0.001048	0.001422	0.001784
²⁴⁵ Cm	1.34E-21	2.14E-06	1.44E-05	4.09E-05	8.18E-05	0.000136	0.0002
²⁴⁶ Cm	1.35E-21	1.29E-08	1.77E-07	7.71E-07	2.11E-06	4.46E-06	8.05E-06

Figure 1. ENHS Core Layout; Vertical Cut

Figure 2. Dependence of Critical Pu Concentration on Lattice p/d and Core Height

Figure 3. Dependence of ICR on Lattice p/d and Core Height

Figure 4. Dependence of Leakage Probability on Lattice p/d and Core Height

Figure 5. Dependence of Reflector Reactivity Worth on Lattice p/d and Core Height

Figure 6. Dependence of Void Reactivity on Lattice p/d and Core Height

Figure 7. Dependence of Absorption Probability in Pb on Lattice p/d and Core Height

Figure 8. Dependence of Critical Pu Concentration on Lattice p/d and Core Power for 4m High Cores

Figure 9. Dependence of ICR on Lattice p/d and Core Power for 4m High Cores

Figure 10. Dependence of Leakage probability on Lattice p/d and Core Power for 4m High Cores

Figure 11. Dependence of Reflector Reactivity Worth on Lattice p/d and Core Power for 4m High Cores

Figure 12. Dependence of Void Reactivity on Lattice p/d and Core Power for 4m High Cores

Figure 13. Variation of k_{eff} with Effective Full Power Years of 4m High Cores for 250MWth having q'_{av} of 120 W/cm

Figure 14. Variation of k_{eff} with Effective Full Power Years of 2m High Cores for 250MWth having q'_{av} of 120 W/cm

Figure 15. Evolution of Fuel Isotope Concentration with Effective Full Power Years of the $p/d = 1.525$ Core of Figure 14

Figure 16. Evolution of Fuel Isotope Concentration with Effective Full Power Years of the $p/d = 1.60$ Core of Figure 14

Figure 17. Design Domain of 80 W/cm Cores for ENHS Featuring Nearly Zero Burnup Reactivity Swing. Peak Fluence Constraint Limits Life to ~ 15EFPY

Figure 18. Design Domain of 120 W/cm Cores for ENHS Featuring Nearly Zero Burnup Reactivity Swing. Peak Fluence Constraint Limits Life to ~ 15EFPY

Figure 19. Variation of k_{eff} with Effective Full Power Years of a 2m High Core for 125MW_{th} having q'_{av} of 80W/cm and p/d of 1.30

Figure 20. Effect of Space Dependent Burnup on k_{eff} Evolution. Core is 4m High with $p/d=1.15$ and $q'_{av}=120\text{W/cm}$ Operating at 250MW_{th}

Figure 21. Effect of Space Dependent Burnup on ^{239}Pu Evolution. Core is 4m High with $p/d=1.15$ and $q'_{av}=120$ W/cm Operating at 250MW_{th} .

Figure 22. Effect of MCNP Calculation Frequency on k_{eff} Evolution. Core is 4m High with $p/d=1.15$ and $q'_{av}=120$ W/cm Operating at 250MW_{th} .

Figure 23. Effect of ORIGIN2 FP Library on k_{eff} Evolution. Core is 4m High with $p/d=1.15$ and $q'_{\text{av}}=120$ W/cm Operating at 250MW_{th} .

Figure 13. Variation of k_{eff} with Effective Full Power Years of 4m High Cores for 250MWth having q'_{av} of 120 W/cm

Figure 14. Variation of k_{eff} with Effective Full Power Years of 2m High Cores for 250MWth having q'_{av} of 120 W/cm

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Figure 23. Effect of ORIGIN2 FP Library on k_{eff} Evolution. Core is 4m High with $p/d=1.15$ and $q'_{\text{av}}=120$ W/cm Operating at 250MW_{th} .

hydraulic analysis performed indicates that the performance of the ENHS that uses these new IHX designs is comparable to that of the reference ENHS.

Task 23

It was found feasible to design simple, uniform composition lead-cooled and reflected cores that maintain nearly zero burnup reactivity swing for up to 15 years of operation without any refueling. These cores have no internal or external blanket assemblies and no reflector assemblies. They are to be loaded inside the module in the factory and taken back to the supplier country in the unopened module. Consequently, the ENHS is expected to be highly proliferation resistant, extremely safe, and very easy to operate.

to have a very simple design and to be free of any mechanical connections to the power plant components. The ENHS coolant need have a low vapor pressure at operating temperatures. Lead or Pb-Bi eutectic appears to be among the most promising.

The purpose of this paper is to briefly describe the ENHS concept and to summarize findings from a couple of studies completed since the inception of this NERI project. These studies are: (a) Exploration of the range of dimensions and weights of ENHS for 125 MW_{th} that are to feature up to 50 °C primary-to-secondary temperature drop and up to 100% natural circulation. (b) Definition of the design domain of cores for the ENHS that enable 15 EFPY with close to zero burnup reactivity swing.

SINGLE ENHS REACTOR CONCEPT

A very schematic description of one of many possible embodiments of a reactor concept having a single ENHS module is depicted in Figs. 1 and 2. The reactor consists of ten modules: one ENHS, three steam generators, three reheaters and three secondary coolant pumps. All these modules are inserted into the reactor pool; they are supported by a massive structural platform that is seismically isolated. There is no mechanical connection between the modules. This makes them relatively simple to install and to replace.

A schematic vertical cut through one concept of an ENHS module is shown in Figure 3. This concept features 100% natural circulation. There are three walls to the ENHS reactor vessel – two structural walls and a confinement wall in between. The confinement wall provides the barrier between the primary and secondary coolants. The primary coolant gets from the riser into the downcomer channels that are formed between the inner structural wall and the corrugated confinement wall and gets back into the coolant plenum underneath the core. The secondary coolant gets from the pool into the space between the corrugated confinement wall and the outer structural wall and flows up in the channels located between the downcomer channels. The heated secondary coolant flows back to the pool near its top. Heat is conducted from the primary to the secondary coolants through the confinement wall. The confinement wall is corrugated in order to increase its surface area for heat transfer so as to keep the primary-to-secondary temperature drop at a reasonable value – taken in this work to be 50 °C.

Figure 4 shows a horizontal cut through a corrugated confinement wall. The corrugated wall channels are stiffened against buckling by vertical bends (not shown in Figure 4). Horizontal rings that are welded to the tips of the corrugations provide additional stiffening. They also act as spacers

between the confinement wall and the inner and outer structural-walls. The axial distance between these spacers is approximately 1m. In addition to their structural function, the spacers are to avoid “rubbing” of the thin confinement wall against the adjacent structural walls and to assure that the space between the base of the corrugations and the structural walls will be large enough to avoid formation of “pockets” of stagnant coolant.

Illustrations of possible corrugation dimensions are given in Tables 1 and 2 that are discussed in the following section. The downcomer flow channel formed by the corrugated wall is rectangular with a cross-section of (Downcomer depth) x (Downcomer width). The corrugations of Figure 4 pertain to Case 9.2.1; they are of an extremely large aspect ratio. Smaller aspect-ratio corrugations can also be used (See Tables 1 and 2).

An alternate ENHS concept being considered is depicted in Figure 5. It uses a cover-gas lift-pump that circulates the cover gas from the plenum above the coolant level in the ENHS and injects it into the coolant in the riser through nozzles located at a certain level above the core^a. The cover gas bubbles reduce the effective density of coolant in the riser, thus increasing the head for coolant circulation. The circulator is envisioned to be located above the reactor pool, outside of the ENHS. Thus, no moving mechanical parts and no electrical parts associated with coolant circulation are located inside the ENHS.

There are very few components inside the ENHS reactor vessel. In particular, it does not include intermediate heat exchangers (IHX), decay heat removal systems (DHRS), and mechanical or EM pumps. Its confinement wall serves as the IHX and part of the DHRS. Neither primary nor secondary coolants are pressurized. They exert nearly balancing hydrostatic pressure on the relatively thin confinement wall.

As presently perceived, the ENHS will be manufactured and fuelled in the factory and shipped to the site as a sealed unit with solidified Pb or Pb-Bi filling the vessel up to the upper level of the fuel rods. Upon insertion to the reactor pool hot Pb (or Pb-Bi) is to be pumped into the ENHS vessel. This hot Pb, along with the hot Pb in the pool, will melt the solid Pb at the lower part of the vessel.

At the end of its life, presently perceived to be 15 EFPY, the ENHS will be removed from the reactor pool and stored on site until the decay heat will drop

^a A cover gas lift pump for circulation of liquid lead coolant was first proposed by ANSALDO⁷ for accelerator driven system. The use of steam lift pumps for circulation of secondary lead coolant was proposed earlier by one of the co-authors et al.⁸⁻¹⁰