**STAR: The Secure Transportable Autonomous Reactor System
ENCAPSULATED FISSION HEAT SOURCE
(The ENHS Reactor)**

FINAL REPORT

NERI Project No. 99-154
Grant No. DE-FG03-99SF21889

Edited by
Ehud Greenspan
Department of Nuclear Engineering
University of California
Berkeley, CA 94720

October 31, 2003

US Participating Organizations and Researchers

**E. Greenspan (PI), A. Barak, D. Barnes, M. Milosovic', D. Saphier, Z. Shayer, H. Shimada
and S. Wang**

Department of Nuclear Engineering
University of California
Berkeley, CA 94720

D. C. Wade (co-PI), E. Feldman, K. Grimm, R. Hill, J. J. Sienicki and T. Sofu

Reactor Analysis and Engineering Division
Argonne National Laboratory
9700 S. Cass Av.
Argonne, IL 60439

N. W. Brown (co-PI), L. Fischer and Q. Hussain

Lawrence Livermore National Laboratory
7000 East Avenue
Livermore, CA 94550

M. D. Carelli (co-PI), L. Conway and M. Dzodzo

Westinghouse Electric Company
Science & Technology Department
1344 Beulah Rd.
Pittsburgh, PA 15235-5083

Republic of Korea Participating Organizations and Researchers

Yeong Il Kim (co-PI) and Ser Gi Hong

LMR Core Design Technology
Korea Atomic Energy Research Institute (KAERI)
150 Dukjin-dong, Yusong-gu, Taejon, 305-353, Korea

Soon Heung Chang (co-PI) and Kwang Gu Lee

Department of Nuclear Engineering
Korea Advanced Institute for Science and Technology (KAIST)
Kusong-dong, Yusong-gu, Taejon, 305-701, Korea

Il Soon Hwang (co-PI), Byung Gi Park and Seung Ho Jeong

Department of Nuclear Engineering
Seoul National University
56-1 Shinlim-dong, Gwanak-ku, Seoul, 151-742, Korea

Japanese Participating Organizations and Researchers

I. Kinoshita (co-PI), **A. Minato**, **Y. Nishi** and **N. Ueda**
Nuclear Energy Systems Department
Central Research Institute of Electric Power Industry (CRIEPI)
2-11-1 Iwado Kita, Komae-shi, Tokyo 201-8511, Japan

ACKNOWLEDGMENT

This report is based upon work supported by the U S. Department of Energy under Award No. DE-FG03-99SF21889 and by the Nuclear Energy Research Initiative Project of Korean Ministry of Science and Technology (MOST). Valuable contributions from the Central Research Institute of the Electric Power Industry (CRIEPI) of Japan are also acknowledged. Jacopo Buongiorno of the INEEL contributed to this project by evaluating the ENHS fuel rod mechanical integrity. Takaaki Sakai of JNC, Japan, contributed to this project by performing an independent evaluation of the feasibility of natural circulation in the primary and intermediate coolant loops and by calculating the two-dimensional coolant flow distribution in the ENHS core. A number of UC Berkeley graduate students participated in the preliminary assessment of the ENHS feasibility: A.S. Bolori, M. Frank, E. Hill, D.M. Hutchinson, S. Jones, X. Mahini, M. Nichol, B.H. Park, H. Shimada and N. Stone. Luca Gratton helped with the final editing of this report.

DISCLAIMER

Any opinions, findings, and conclusions or recommendations expressed in this material are those of the author(s) and do not necessarily reflect the views of the Department of Energy.

TABLE OF CONTENTS

Executive Summary	1
1. Introduction	1
2. Brief Description of the Reference ENHS Reactor Concept	1
3. Summary of Feasibility Assessment	8
4. Novel Design Variants	14
5. Meeting GEN-IV Goals	15
6. List of ENHS Related Publications	16
1. Introduction	22
1.1 Background	22
1.2 Study Goals	23
1.3 Participating organizations	25
1.4 Report Layout	25
1.5 References	26
2. General Description of the ENHS Reactor Concept	28
2.1 Originally Conceived Single Module Reactor	28
2.2 Design Variations	28
2.3 ENHS Reactor Concept	28
2.4 Multiple ENHS Module Reactor	38
2.5 ENHS Module with Cover-Gas Lift-Pump	39
2.6 References	41
3. Reference ENHS Reactor	42
3.1 Introduction	42
3.2 Dimensions	42
3.3 Core Design	44
3.3.1 Geometry and Composition for Calculating the ENHS Core	44
3.3.2 Core Design Approach	46
3.3.3 Reference Core Neutronic Characteristics	47
3.3.4 Reference Core Power Distribution	52
3.3.5 Reference Core Composition Evolution	57
3.3.6 Reference Core Shielding	59
3.4 Thermal-Hydraulic Characteristics	64
3.5 ENHS Safety	67
3.6 Load Following Capability	71
3.7 Steam Generators Design	72
3.8 Energy Conversion System	75
3.9 ENHS Fuel Cycle Cost	76
3.10 Feasibility of Shipping and Installing a New ENHS Module	79
3.11 Feasibility of Melting the Solidified Pb-Bi in the as Shipped ENHS Module	82
3.12 Feasibility of Solidifying Pb in the ENHS Core at EOL	83
3.13 Feasibility of Removing the Spent ENHS Module from the Reactor Pool	83
3.14 Feasibility of Transporting the Spent ENHS Module	86
3.15 References	88
4. Alternative Design Options	91
4.1 Alternative Fuels	91

4.2	Alternative Coolants	95
4.3	Compact ENHS Module Using Lift-Pumps	101
4.4	Nested Tube IHX	105
4.5	Integrated Steam generator and IHX	106
4.6	Supercritical CO ₂ Energy Conversion System	106
4.7	References	106
5.	Meeting the Goals for Generation-IV Reactors	108
5.1	Sustainable Energy	108
5.1.1	High Fuel Utilization	108
5.1.2	Low Waste	108
5.2	High Safety and Reliability	108
5.2.1	High Safety	108
5.2.2	High Reliability	109
5.2.3	Good Neighbor	109
5.3	High Proliferation Resistance	109
5.3.1	Technological Barrier 1: No Access to Fuel	109
5.3.2	Technological Barrier 2: No Access to Neutrons	110
5.3.3	Technological Barrier 3: No Facilities Suitable for Military Applications	110
5.3.4	Material Barrier 1: Fuel Isotopics	110
5.3.5	Material Barrier 2: Radiation Barrier	111
5.3.6	Desirable Institutional Barriers	111
5.3.7	Feasibility of Proliferation Resistant Multi-Recycling	111
5.4	Economic Viability	114
5.4.1	Competitive life cycle cost	114
5.4.2	Low Financial Risk	115
5.5	References	115
6.	Summary	116
Appendix A:	Reference Thermo-Physical Properties of Pb and Pb-Bi	120
A.1	Introduction	120
A.2	Correlations for Pb-Bi	120
A.3	Correlations for Pb	124
A.4	Conclusions	128
A.5	References	128
Appendix B:	Fuel Cycle Cost Analysis	130
B.1	Introduction	130
B.2	Scenarios Considered	130
B.3	Methodology	131
B.4	Cost Data	132
B.5	Other Assumptions	135
B.6	Results	136
B.7	Conclusions	137
B.8	References	139

EXECUTIVE SUMMARY

1. Introduction

The Encapsulated Nuclear Heat Source (ENHS) is a novel 125 MW_{th} fast spectrum reactor concept that was selected by the 1999 DOE NERI program as a candidate “Generation-IV” reactor. It uses Pb-Bi or other liquid-metal coolant and is intended to be factory manufactured in large numbers to be economically competitive. It is anticipated to be most useful to developing countries. The US team studying the feasibility of the ENHS reactor concept consisted of the University of California, Berkeley, Argonne National Laboratory (ANL), Lawrence Livermore National Laboratory (LLNL) and Westinghouse. Collaborating with the US team were three Korean organizations: Korean Atomic Energy Research Institute (KAERI), Korean Advanced Institute for Science and Technology (KAIST) and the University of Seoul, as well as the Central Research Institute of the Electrical Power Industry (CRIEPI) of Japan. The number of researchers that had some participation in the ENHS study is approximately 40, of which 15 were students.

Unique features of the ENHS include at least 20 years of operation without refueling; no fuel handling in the host country; no pumps and valves; excess reactivity does not exceed 1\$; fully passive removal of the decay heat; very small probability of core damaging accidents; autonomous operation and capability of load-following over a wide range; very long plant life. In addition it offers a close match between demand and supply, large tolerance to human errors, is likely to get public acceptance via demonstration of superb safety, lack of need for offsite response, and very good proliferation resistance. The ENHS reactor is designed to meet the requirements of Generation IV reactors including sustainable energy supply, low waste, high level of proliferation resistance, high level of safety and reliability, acceptable risk to capital and, hopefully, also competitive busbar cost of electricity.

A detailed description of the concept and a summary of the research done so far can be found in close to 40 publications and 2 annual reports (See “ENHS Related Publications” below).

2. Brief Description of the Reference ENHS Reactor Concept

A schematic description of the reference design concept of the ENHS reactor is depicted in Figures 1 and 2. The nuclear steam supply system (NSSS) consists of one ENHS module and eight small steam generators. There is no mechanical connection between the Module and the steam generators. Both primary and secondary coolants flow by natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the Intermediate Heat Exchanger (IHX) and flows back into the coolant plenum underneath the core. The secondary coolant flows from the pool outside of the vessel into the bottom of the IHX and exits back to the pool near the top of the IHX. The IHX, depicted in Figures 2 and 3, consists of rectangular channels that are connected at their top and bottom to a tube sheet (not shown in Figure 2). The 4 mm thick rectangular channel walls provide the barrier between the primary and the secondary coolants whereas the inner and outer walls provide the structural support. More conventional IHX made of circular tubes could be used as well. Relative to circular tube IHX the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller

friction losses due to elimination of grid spacers. The mean temperature difference between the primary and secondary coolants is $\leq 50^{\circ}\text{C}$.

Figure 1 A schematic vertical view of an ENHS reactor. Not to scale

Figure 2 A schematic vertical view of an ENHS module. Not to scale.

Note: Present reference design has secondary coolant enter the module above the top fuel level and exit the module below the level at which the primary coolant enters the IHX; as in Fig 3.

Figure 3 shows a vertical cut through the reference ENHS module while Figure 4 shows an alternative ENHS module design that uses a lift-pump integrated within the riser. Cover gas from the top of the module is injected into the coolant just above the core. The cover gas bubbles reduce the effective density of coolant in the riser, thus increasing the head for coolant circulation. The circulator would be located above the reactor pool, outside of the Module vessel. A lift-pump for the secondary coolant is integrated within the Module above the IHX. Table 1 summarizes selected design and performance parameters of the two reference designs.

Figure 3 Reference ENHS module featuring 100% natural circulation, ENHS1

Figure 4 Reference ENHS module featuring cover-gas lift pump, ENHS2. There are no pumps or valves inside the module, but there are gas circulators outside (on top of) the module.

Table 1 Selected Characteristics of ENHS Reference Designs

Design parameter	ENHS1	ENHS2
Primary Pb coolant circulation	100% natural	With lift-pump
Fuel material	U-12 wt% Pu-10 wt% Zr	
Structural material		HT-9
Fuel length (m)		1.25
Fission gas plenum length (m)		1.25
Core diameter (m)		2.24
Fuel rod outer diameter (cm)		1.56
Clad thickness (cm)		0.13
Lattice (hexagonal) pitch (cm)		2.12
Overall module height (m)	19.6	10.1
Outer module diameter (m)	3.52	3.72
Number of rectangular channels in IHX	135	435
Inner dimensions of channel (cm x cm)	39.2 x 2.5	49.2 x 0.5
IHX channel length (m)	11	3
Approximate weight of fueled module for shipment (t)	360	300
Coolant in/peak coolant out/peak clad/peak fuel temp. (°C)	400/547/579/622 ^a	400/502/531/576 ^a
Primary/intermediate coolant outlet temperature (°C)	503/471 ^a	471/443 ^a
Attainable thermal power (MW _{th})	125	190
Net energy conversion efficiency (%)	38	38

^a At 125 MW_{th}

The eight steam generators (SG) shown in Figures 5 and 6 are anchored to the support structure that covers the pool and are not mechanically connected to the Module. They are designed to meet several unique requirements that are dictated by the ENHS reactor layout: (1) Effective utilization of the pool volume surrounding the Module. (2) Minimum friction losses so as to enable 100% natural circulation of the intermediate coolant. (3) Having no mechanical connection with the Module. (4) Minimum flow rate of water into the intermediate coolant pool in case of a breach in steam generator tube or failure of other water-containing component. (5) Accommodation of a large thermal expansion. (6) Ease of inspection and maintenance. (7) Modular design that is easy to install and replace.

The steam generator is of a once-through tube-in-tube design; feed-water flows in via the inner tube and the steam is generated in the shell between the inner and outer tubes. The liquid metal (either Pb-Bi or Pb) coolant flows outside the tubes. Also, the steam and feed-water piping and nozzles are located outside the ENHS pool, and the feed-water to each steam tube is inherently orificed by a small diameter feed tube. These features all act to minimize the quantity and/or mass flow rate of water or steam that can be introduced into the pool due to a postulated steam line break, or feed-line break, or tube rupture.

A flow partition extending from below the exit from the steam generators down to close to the bottom of the intermediate coolant pool (not shown in Figure 1). Its function is to force the coolant to flow through the bottom of the pool so as to eliminate stagnation of Pb-Bi.

The ENHS core is made of uniform composition fuel rods with no blanket assemblies and designed to maintain a nearly constant k_{eff} over 20 years without refueling or fuel shuffling. The reference fuel is taken to be metallic alloy with 10 weight % Zr. Both enriched uranium and Pu-U are considered for the HM. The structural material is taken to be HT-9; the ferritic-martensitic

SS used for EBR-II. The Module will be manufactured and fuelled in the factory and shipped to the site as a weld-sealed unit with solidified Pb-Bi filling the vessel up to above the fuel rods. A unique feature of Pb-Bi that makes it possible to embed the fuel rods and core structure in solid Pb-Bi without damage is its nearly zero coefficient of volumetric expansion upon phase change⁴². At the end of its core life the Module will be removed from the reactor pool after pumping-out Pb-Bi from above the fuel level and the remainder will be replaced by Pb. The module will be stored on site until the decay heat drops to a level that will permit to solidify the coolant. The Module with the solidified coolant will then serve as a shipping cask.

Currently, the Control and Shutdown Systems (C&SS) used in the 4S reactor of TOSHIBA-CRIEPI⁴³⁻⁴⁵ have been adopted for the ENHS. The C&SS consist of a single neutron absorber assembly located centrally in the core and six segments of an annulus that surrounds the core and referred to as the peripheral absorber (or “reflector”). The active element for both central and peripheral absorbers is B₄C and tungsten; being heavier than Pb-Bi tungsten can scram by gravity. All of these absorbing elements are securely latched in place during shipping and reactor installation. The central absorber has an electromagnetic latch that does not engage until the startup temperature of 350°C is achieved. At this temperature the assembly can be withdrawn. Normal operational shutdowns can be accomplished with the peripheral absorbers. The reactor is brought critical by a hydraulic system that moves the peripheral absorbers up at 1 mm/sec to compensate for the negative temperature coefficient of reactivity. At the full power position, the

Figure 5 ENHS steam generator overall elevation view and ENHS module layout plan

Figure 6 ENHS steam generator cross-section elevation view.

Peripheral absorber segments are stopped from further upward movement by mechanical stoppers whose movement is established by high-reliability gear drives. These drives restrict the rate of movement of the peripheral absorber segments to approximately 1 mm/day. It is anticipated that the height of the peripheral absorbers will be adjusted once a year or two to compensate for a slight drift in reactivity due to burnup.

3. Summary of Feasibility Assessment

The primary objectives of the ENHS study were to assess the feasibility of the following: (1) Designing lead-bismuth or lead cooled reactor cores for 20 years of full power operation without refueling and with nearly zero burnup reactivity swing, using either Pu-U or enriched uranium fuel. (2) Designing the ENHS to have 100 percent natural circulation in both primary and intermediate coolant systems with reasonable module dimensions and weight. (3) Designing the ENHS module to be free of mechanical connections to the energy conversion system. (4) Designing and manufacturing an intermediate heat exchanger (IHx) that is an integral part of the ENHS module vessel. (5) Fueling the ENHS module in the factory, transporting it fuelled and weld sealed to the nuclear power plant site and installing it in the reactor pool. (6) At end of life removing the module from the reactor pool and transporting it unopened to a recycle facility. (7) Designing the ENHS to have autonomous load-following capability. (8) Designing the ENHS reactor to have passive safety so that postulated accidents will not damage reactor components. (9) Attaining a high energy-conversion efficiency. (10) Using alternative coolants while retaining the unique characteristics of the ENHS. (11) Assessing the ENHS reactor ability to meet the goals set for Generation-IV reactors. In the following we summarize the major findings, not adhering exactly to the above numbered objectives.

3.1 Long life core with nearly zero burnup reactivity swing. Figure 7 shows the k_{eff} evolution with burnup for the reference core and its sensitivity to variation of the p/d ratio. The reference core pitch-to-diameter ratio is p/d=1.36 and it uses IFR like metallic fuel with 12.20 w/o Pu from LWR. The maximum value of its burnup reactivity swing over 20 years is < 0.3%; this is less than \$1. Table 2 summarizes selected core physics parameters at BOL and EOL of the reference core. The temperature coefficients of reactivity are all negative except for that of the coolant expansion. However, the negative reactivity coefficient associated with the radial expansion of the core structure can compensate for the coolant thermal expansion effect. The void coefficients for 100% voided inner and middle cores are positive. However, in the ENHS core, creation of voiding is practically impossible since the boiling temperature of Pb-Bi is 1943K that is 1000K higher than the operating temperature and even higher than the melting

Figure 7 Effect of lattice p/d on k_{eff} evolution with operating time. 12.20 w/o Pu

Figure 8 Axial power distribution in the hot channel at BOL and EOL

temperature of the metallic fuel and clad. The fast fluence at 20 EFPY is below 4×10^{23} n/cm² that is assumed to be the radiation damage limit of HT-9. In fact, it is possible to operate this core for ~21 EFPY. Fuel rod analysis concluded^{29,30} that the clad can maintain its mechanical integrity up to the peak burnup of 104 GWD/tHM. Table 2 and Figure 8 show that the core characteristics are exceptionally insensitive to burnup.

Table 2 New ENHS Reference Core Physics Data

Performance Parameter	BOL	EOL
Peak-to-average power density	1.829	1.863
Peak linear heat rate (W/cm)	184.1	187.6
Peak-to-average channel power	1.500	1.529
Peak-to-average power density in hot channel	1.220	1.224
Peak burnup after 20 (20.9 [*]) EFPY (GWD/tHM)		99.89 (104.4 [*])
Average burnup after 20 (20.9 [*]) EFPY (GWD/tHM)		50.8 (53.1 [*])
Peak fast (E>0.1MeV) neutron flux (n/cm ² -s)	8.86E+14	8.94E+14
Peak fast (E>0.1 MeV) fluence at 20 EFPY (n/cm ²)		3.829E+23
Conversion ratio	1.048	1.045
Effective delayed neutron fraction	0.0035	0.0035
Maximum Δk_{eff} with burnup (%)	0.287	
Doppler effect (dk/kk' -°C)	-5.2442E-06	-4.2202E-06
Axial fuel expansion (dk/kk' -°C)	-4.6379E-06	-4.6057E-06
Coolant expansion (dk/kk' -°C)	+1.0917E-07	+5.6866E-07
Grid-plate radial expansion (dk/kk' -°C)	-8.0679E-06	-6.4097E-06
Cold (350°C) to hot (480°C; fuel: 700°C) dk/kk'		
Doppler effect	-1.9600E-03	-1.5399E-03
Axial fuel expansion	-6.1421E-04	-6.0921E-04
Coolant expansion	+2.6400E-05	+8.6558E-05
Grid-plate expansion	-1.0516E-03	-8.3430E-04
Total	-3.5994E-03	-2.8968E-03
Void reactivity effect (dk,%)		
Voiding inner 1/3 core (+upper gas plenum)	+2.718 (+1.516)	+2.809 (+1.618)
Voiding middle 1/3 core (+upper gas plenum)	+0.689 (-0.068)	+0.659 (-0.043)
Voiding outer 1/3 core (+upper gas plenum)	-0.694 (-1.045)	-0.668 (-1.009)
Voiding whole core (+upper gas plenum)	+2.555 (+0.424)	+2.644 (+0.572)
Coolant loss reactivity effect (dk/kk', %)		
Coolant level drops to bottom of fission gas plenum	-1.869	
Coolant level drops to 1/3 below core top	-6.261	
Coolant level drops to 2/3 below core top	-12.219	
Coolant level drops to core bottom	-14.825	
Peripheral absorber reactivity worth (dk,%)	1.990	1.817
Central absorber reactivity worth (dk,%)	4.138	4.104
Peripheral + central absorber reactivity worth (dk,%)	6.811	6.562
Total heavy metal inventory (kg)	1.7505E+04	1.6564E+04
Fissile-to-total plutonium ratio	0.6437	0.6649

* Limited by radiation damage to clad of 4×10^{23} fast (E>0.1 MeV) neutrons per cm².

In conclusion, it was found possible to design cores using Pu-U fuel to operate up to the radiation damage constraint with nearly zero burnup reactivity swing. The resulting cores are very simple:

they have uniform composition, no blanket or reflector assemblies and a single safety rod assembly. The design domain has been defined⁶ for such cores. It was also found possible to design enriched uranium fueled cores to have similar characteristics⁶.

3.2 Feasibility of natural circulation. Table 3 summarizes the results obtained in the thermal – hydraulic analysis of the ENHS primary and secondary coolant systems. The methodology used for this analysis is described in References 11 and 20. The same core was assumed for both ENHS1 and ENHS2. The coolant inlet temperature to the core was assumed to be 350°C instead of 400°C used before^{15,18}. Although this inlet temperature is somewhat too low for lead coolant, the analysis was done for Pb coolant in addition to Pb-Bi, our reference coolant. The results of Table 3 can be easily adjusted to higher coolant inlet temperature. The hot channel temperatures in Table 3 are likely to be an overestimate of the peak temperatures as the hot-channel analysis assumes that there is no cooling mixing between channels. As the ENHS core is made of individual fuel rods, a significant cross flow is expected.

Table 3 Comparison of Design and Thermal Hydraulic Conditions for Lead-Bismuth Eutectic (LBE) and Lead for ENHS Design Variants for 125 MW_{th}

Case Designation	ENHS1	ENHS2 ^(a)
Coolant	LBE/Lead	LBE/Lead
Roughness Height, μm	10	10
Gas Injection Above Core/Lift Pump	No	Yes ^(b)
Heated Core Height, m	1.25	1.25
Fission Gas Plenum Height, m	1.25	1.25
Total Core Height, m	2.5	2.5
Fuel Rod Diameter, cm	1.56	1.56
Fuel Rod Pitch, cm	2.106 ^(d)	2.106 ^(d)
Pitch-to-Diameter Ratio	1.35 ^(d)	1.35 ^(d)
Cladding Thickness, cm	0.13	0.13
Fuel Material	U-Pu-Zr	U-Pu-Zr
Fuel Diameter, cm	1.126	1.126
Cladding-Fuel Bond Material	Sodium	Sodium
Cladding-Fuel Bond Gap Thickness, cm	0.0871	0.0871
Fuel Smeared Density, %	75	75
Number of Fuel Rod Spacer Grids	3	3
Core Radius, m	1.10 ^(d)	1.10 ^(d)
Core Peak-to-Average Power	1.767 ^(d)	1.767 ^(d)
Peak-to-Average Power Factor at Core Mid-Height	1.767 ^(d)	1.767 ^(d)
Peak-to-Average Power Factor at Top of Core	0.942	0.942
Riser Height, m	13	3.0
HX Channel Height, m	11	1.75
Primary Coolant Thermal Centers Separation Height, m	8.125	3.50
HX Channel Width, cm	2.5	0.5
HX Channel Thickness, cm	39.2	49.2
Number of HX Channels	135	434
Number of SG Modules	8	8
SG Height, m	4.8	4.8
SG Tube Outer Diameter, cm	2.0	2.0
Number of SG Tubes per SG Module	613	613
Height Difference Between SG Inlet and HX Outlet, m	1.15	1.15
Intermediate Coolant Thermal Centers Separation, m	4.25	-0.375
Core Flow Area, m ²	1.83	1.83
Primary Coolant HX Flow Area, m ²	1.86	1.87

Case Designation	ENHS1	ENHS2 ^(a)
Intermediate Coolant HX Flow Area, m ²	1.32	1.07
Intermediate Coolant SG Flow Area, m ²	2.04	2.04
Core-to-HX Flow Area Ratio	0.980	0.975
Primary Coolant Flowrate, Kg/s	8320/8220	12100/122
Intermediate Coolant Flowrate, Kg/s	5970/5870	7840/7910
Coolant Velocity in Core, m/s	0.447/0.426	0.646/0.63
Primary Coolant Velocity in HX, m/s	0.438/0.418	0.630/0.61
Intermediate Coolant Velocity in HX, m/s	0.440/0.418	0.715/0.69
Intermediate Coolant Velocity in SG, m/s	0.285/0.271	0.374/0.36
Reynolds Number in Core	47400/30800	66900/438
Primary Coolant Reynolds Number in HX	159000/1030	65400/428
Intermediate Coolant Reynolds Number in HX	128000/8070	42900/275
Intermediate Coolant Reynolds Number in SG	44000/27800	56500/363
Fraction of Primary Coolant Pressure Drop in HX	0.370/0.374	0.303/0.30
Fraction of Intermediate Coolant Pressure Drop Associated with HX	0.701/0.699	0.790/0.78
Primary Coolant Temperature Rise, C	103/104	70.8/69.8
Intermediate Coolant Temperature Rise, C	143/144	109/107
Primary Coolant Outlet Temperature, C	453/454	421/420
Primary Coolant Inlet Temperature, C	350	350
Intermediate Coolant Outlet Temperature, C	421/427	393/395
Hot Channel Primary Coolant Temperature Rise, C	147/149	102/101
Hot Channel Outlet Temperature, C	497/499	452/451
Hot Channel Peak Cladding Outer Surface Temperature, C	518/517	470/466
Hot Channel Peak Cladding Inner Surface Temperature, C	529/529	481/477
Hot Channel Peak Sodium Bond Temperature, C	533/532	484/481
Hot Channel Peak Fuel Temperature, C	572/571	525/522
Intermediate Coolant Inlet Temperature, C	278/283 ^(c)	284/288 ^(c)
Primary-to-Intermediate Coolant Temperature Difference at HX Top, C	31.4/26.60	27.7/24.5
Primary-to-Intermediate Coolant Temperature Difference at HX Bottom, C	71.6/67.4	65.7/61.6

(a) Using a different core and module diameter than in Table 1.

(b) Primary two-phase region height = 1.75 m; Intermediate two-phase region height = 2.0 m; Void fraction = 0.1.

(c) Inconsistent with use of molten lead for which $T_{mp} = 327$ C.

(d) These values have been slightly increased since the T-H analysis has been completed—see Tables 1 and 2. However, the conclusions of the T-H analysis will not be affected.

In conclusion, it was found feasible to design ENHS modules to deliver 125 MW_{th} from the primary coolant to the secondary coolant through a 4 mm thick confinement wall (that separates the two coolants) with maximum primary coolant to maximum secondary coolant temperature drop of 50°C and with 100% natural circulation. The required reactor vessel height is close to 20 meters. In fact, good synergism was found between the requirement for 100% natural circulation and the requirement for primary-to-secondary temperature drop of 50°C.

A significantly more compact and lower weight ENHS module has been conceived and conceptually designed. It uses cover-gas lift-pump for coolant circulation. The gas circulators are located outside of the reactor vessel.

3.3 Designing the Module to be free of mechanical connections. It was found feasible to design the Module to be free of mechanical connections to the energy conversion system. An intermediate heat exchanger (IHx) that is integrated within the Module vessel wall and has practically identical heat transfer characteristics to those of the originally proposed confinement wall has been conceived and analyzed (See Figures 3 and 4). It consists of rectangular channels

circumscribing the primary coolant riser in the space between the inner structural wall and the outer structural wall. This integrated IHX was found able to withstand the loads expected during transportation, installation and operation.

3.4 Fueling the Module in the factory, transporting it fuelled to the nuclear power plant site and installing it in the pool unopened. All above operations were found feasible. The approach envisioned for fuelling the Module in the factory is described in Reference 10. After fuelling the Module will be filled with molten Pb-Bi up to the top core level and the Pb-Bi will be solidified before shipment. The package to be shipped is approximately 19.6m long and 3.2 m in diameter and weighs ~350 tons. This is a relatively modest size package compared to the steam generators and pressure vessels for large LWRs, some of which are nearly 1000 tons. The size and the mass of the package do not present any feasibility issues. The reactor module is however a unique shipping package, in that it contains the reactor fuel and control elements frozen in Pb-Bi or Pb. There is no commercial precedence for licensing such a shipping package. Special gear will have to be constructed for the installation of the Module in the reactor pool.

3.5 Designing the ENHS to have autonomous load-following capability. The ENHS reactor can autonomously follow load variations over a wide range from 110% of nominal power to decay heat level; it is achieved via temperature feedback¹³. Figure 9 illustrates the response of selected core temperatures to a 30% load reduction. The average coolant temperature slightly increases but the peak temperatures go down.

3.6 Removing decay heat passively without damaging core components. The path for decay-heat removal is from the primary coolant through the IHX walls to the secondary coolant and from there either to the steam generators or, through the pool vessel walls, to a passive reactor vessel air cooling system (RVACS). The ENHS is characterized by a very large surface area per MW_{th} of decay heat removal both from the primary to the secondary coolant, and from the secondary coolant to the RVACS. In addition, the ENHS is characterized by a large thermal inertia due to the large inventory of the primary and secondary coolants and the very large

Figure 9 Temperature evolution in selected core locations following a 30% load reduction

Figure 10 Temperature evolution in selected core locations following a LOHS accident

margin to coolant boiling. Both features, combined with a low power density, make the ENHS highly passively safe. This is illustrated in Figure 10 that shows the response of selected core temperatures to a loss of heat sink (LOHS) accident without scram – the worst accident identified so far for the ENHS reactor. It is seen that the maximum temperatures stay far below their constraints. In fact, the peak fuel temperature practically does not increase. The negative reactivity feedback is provided by an increase in the average core temperatures. Accident analysis done so far^{12,13,23,30} revealed that the deviation of the system from nominal operating conditions is very slow and that all plant components do not reach their design limit temperatures. All the postulated events were mitigated by passive phenomena, without actuation or intervention – negative reactivity feedback and natural circulation.

3.7 Designing the pool to have natural circulation. Table 3 results show that this design feature is feasible.

3.8 Melting the Pb-Bi the fuel is shipped imbedded in by filling the Module with hot Pb-Bi and by inserting it into a hot Pb-Bi pool. It is feasible²⁵ to melt the primary coolant and bring it to the reactor startup temperature within approximately two days of its insertion to the pool without having to use special external heaters in the module. In fact, most of the melting may occur in the process of Module installation. The dominant pathway for heat transfer to the solid coolant in and below the core is heat conduction from the secondary coolant in the pool.

3.9 Solidifying the Pb in the core in reasonable time after shutdown and removal the Module from the pool. This was found to be very difficult to achieve. However, we devised a strategy for replacing, after shutdown, the Pb-Bi coolant by Pb. It was found³⁵ feasible to solidify the Pb one week after shutdown by forced cooling the cylindrical outer boundary of the core and the surface of the shutdown assembly channel located at the center of the core. If cooling is provided only at the outer boundary, then the Pb can be solidified after only 22 days. We have conceived of a number of possible approaches for maintaining solid Pb with passive cooling but the practicality of these approaches needs to be further investigated.

3.10 Using the Module as a shipping cask for spent fuel. It appears possible to ship the spent ENHS module intact by placing it inside a specially designed shipping cask. It is likely that the spent ENHS transport cask could be certified by both the US NRC and by the IAEA for domestic and for international transport. This approach is relatively heavy (~1000 MT) and relatively costly (~\$5M for cask) but is do-able. The handling of the lift-pump version of the ENHS will be easier and less expensive. Feasibility of passive removal of decay heat needs still to be established.

3.11 Attaining high-energy conversion efficiency. A no-reheat Rankine cycle having 120 bar – 482°C inlet steam conditions and 34°C (0.056 bar) condenser was chosen for the reference energy-conversion (EC) system²². The feedwater heating of this cycle was optimized with the goal of maximizing the cycle efficiency. The gross EC efficiency of this reference cycle is calculated to be 41.5%. The net efficiency is 37.7%; comparable to that of SUPER-PRISM⁴⁶. By increasing the pressure to 180 bars, the net efficiency will be 40%.

3.12 Using alternative coolants. The neutronic and thermal-hydraulic characteristics of Pb-Bi or Pb cooled ENHS were found to be very similar. Use of sodium, however, dictates major design variations. Attainment of nearly zero burnup reactivity swing calls for a p/d ratio of 1.16

versus 1.36 for Pb-Bi. Natural circulation can be established with sodium in a module of the same height adopted for the reference design, but with significantly smaller temperature margins and significantly larger temperature change across the core than in case of Pb-Bi or Pb cooled reactor. As the volume change associated with solidification of Pb or Na is not as small as that of Pb-Bi, the latter is more suitable for imbedding the fuel for shipment.

In summary, the Encapsulated Nuclear Heat Source (ENHS) concept appears feasible and offers new interesting possibilities for mankind to benefit from nuclear energy. Additional R&D is needed to establish the viability of an ENHS demonstration plant. Important issues that warrant study are the feasibility of handling and shipping the spent module without having to open it on site and the economic viability of the ENHS.

4. Novel Design Variants

In the course of the ENHS feasibility study we came up with a number of novel design solutions that could make the ENHS more economically viable than the originally conceived ENHS. Following is a brief description of the novel design variants we suggest for further examination in a follow-up study.

4.1 Cover gas lift-pump for both the primary and the secondary coolants. The lift-pump enables significant reduction of the module dimensions and fuel inventory or increase of the Module power without introducing a mechanical or an EM pump within the module. Only a very preliminary assessment of this option was carried out in this study; it focused on the effect lift pumps could have on the reactor dimensions but did not get into lift-pump system design, reactor design and safety analysis.

4.2 Nested tube IHX²⁴. This alternative IHX is built out of seamless tubes of various diameters positioned (nested) inside each other. The gap between the tubes is alternately filled with primary then secondary liquid metal coolant, flowing in countercurrent directions. Advantages of this novel IHX design concept include the following: (1) It can be made from off-the-shelf components. (2) High rigidity against buckling. (3) It is easier to fabricate. (4) The nested tube IHX can withstand any pressure differential caused by the different levels of the primary and secondary coolants. (5) It is easy to introduce in between the inner and outer structural walls monitoring devices, Pb-Bi extraction tubes, cooling tubes etc. The main disadvantage is that circular nested IHX needs to be 13% - 30% longer than IHX with rectangular channels, which will result in increased friction losses. On the other hand, the spaces between the tube bundles can be beneficially used for insertion of systems required for servicing the Module.

4.3 Integration of the steam generator within the IHX¹⁴. This design approach will result in a significant reduction in the inventory of the secondary coolant and in the size of the reactor pool. It involves locating steam generator tubes in the primary loop while separating the water/steam tube from the primary coolant by a thin layer of secondary coolant or solid heat conducting material. The drawbacks of this design approach are: (1) Increase in the Module outer diameter. (2) Complicated interface between the Module and the energy conversion system. (3) More complicated inspection and maintenance of the SG.

4.4 Multi-recycling the fuel from the spent ENHS module using a proliferation-resistant dry process such as AIROX^{8,16}. What makes this reactor particularly suitable for such a fuel cycle is the fact that its core is of a uniform composition, does not have any blanket elements, and maintains the fuel k_{∞} nearly constant throughout life. An elaboration on this option can be found in Section 5.3.8.

4.5 Use of supercritical CO₂ instead of water for the working fluid of the energy conversion (EC) system. The advantages of CO₂ - based energy-conversion system is that it can offer a comparable or even higher EC efficiency⁴⁷ while simplifying the interface between the EC system and the liquid metal heat source. The interface simplifications result from the reduced chemical reactivity and reduced pressure of the CO₂ as well as from the smaller volume increase associated with pressure drop (due, say, to a breach in an HX tube). Alternatively, a multi-stage He gas turbine cycle recently conceived at UCB⁴⁸ is worth considering for the ENHS.

5. Meeting GEN-IV Goals

Detailed discussion of the ability of the ENHS concept to meet GEN-IV goals is given in References 15, 19 and 29. It is concluded that the ENHS reactor concept can readily meet all of the goals set for GEN-IV reactors examined in this study. Not examined as yet is the goal of economic viability. Economic viability will be one of the most important issues we recommend to address in Phase 2 of the study. Following is a very brief summary.

5.1 Fuel utilization. The ENHS reactor core is designed so that its fissile fuel content is kept nearly constant throughout the 20 years of core life. As the initial fuel composition is uniform and there are no blanket assemblies, the end-of-life fissile content will be close to uniform across the core. All that is needed to recycle the fuel is to remove FP (not necessarily all), add some depleted uranium makeup and re-fabricate fuel rods. An AIROX-like dry process⁴⁹ may be adequate; only volatile and semi-volatile fission products (FP) can be removed from the fuel. The resulting non-proliferating fuel cycle features an order of magnitude higher fuel utilization than of LWR's.

5.2 Waste management. The above described fuel cycle will result in an order-of-magnitude reduction of the amount of actinides that get to the waste stream, per unit energy generated. Moreover, one of the fuel cycle options considered is to feed the ENHS with spent fuel from LWRs from which part of the uranium and part or all of the FP have been removed. A UREX process being developed for the AFCI program may be adequate. Fluoride volatilization is another process worth examining. The net result will be an alleviation of the LWR nuclear waste problem combined with a low-waste generating nuclear energy system.

5.3 Proliferation resistance. The ENHS nuclear energy system is extremely proliferation resistant due to a unique combination of technological and material barriers: (1) Limited access to fuel – the fuel is loaded into a weld-sealed ENHS module in the factory and is shipped imbedded in solidified Pb-Bi. There is no fuel handling and fuel handling hardware in the power plant. The fuel bearing ENHS module is impossible to divert without notice. (2) Limited access to neutrons – there is no technical way to insert fuel or target materials for irradiation in or near

the core. (3) The fuel always contains FP and thus has a radiation barrier. As the fuel is to be shipped to the site embedded in Pb-Bi, it is possible to seed it with additional radioactive isotopes so as to increase the intensity of the radiation barrier without interfering with shipment and installation.

5.4 Safe operation. There are no pumps, valves or pipes in either the primary or the secondary coolant systems of the ENHS reactor. Moreover, as the ENHS reactor pool is located inside a silo and the coolant level is nearly 15 m above the fuel level, it is unlikely to have a LOCA or a LOFA. In addition, the ENHS reactor cannot have a control rod ejection accident and the maximum reactivity insertion is $< \$1$. Accident analysis done so far indicates that the ENHS reactor has superb safety (See 2.2.6).

5.5 Reliable operation. The ENHS reactor is expected to offer very high reliability due to the following features: no pumps or valves in the primary and secondary systems, nearly zero burnup reactivity swing, very long core life, no on-site refueling, autonomous operation. Being disposable and having relatively few components, the Module is not expected to require much maintenance. There is easy access to inspect and maintain the SG's.

5.6 Good neighbor. It is expected that an ENHS nuclear power plant (NPP) will require no emergency planning zone outside its fence. This is due to the reactor unique safety features discussed above.

5.7 Competitive life cycle cost. No economic analysis of ENHS NPPs has been done so far. Nevertheless, we think that the life-cycle cost of the ENHS NPP has a chance of being competitive with that of large size NPP's due to the following features: (1) Simpler design with fewer components. (2) No refueling and no refueling-hardware on site. (3) Mass production in factory. (4) Very short on-site assembly time (5) Very high availability – due to 20 years core life and high reliability. (6) Very long plant life – as the Module is disposable, the ENHS NPP life could exceed 60 and may reach 100 years⁵⁰. (7) High thermodynamic efficiency of $\sim 40\%$. (8) Small staff. (9) Close match between demand and supply; the incremental capacity increase being $\sim 50 \text{ MW}_e$ per installed Module. (10) Load following capability (autonomous) over a wide range of power levels. (11) No need for long-distance electricity transmission lines. (12) No need for advanced infrastructure. Nevertheless, a thorough cost analysis is required; this is to be a major objective of the work hereby proposed.

A potential economic drawback of the ENHS is the high fuel inventory per installed KW_e that could result in a high fuel cycle cost. This is a generic problem of all long-life core reactors. The ENHS has the advantage of requiring a relatively low enrichment or weight % Pu; about one-half of that of a typical fast reactor such as S-PRISM⁵¹. The ENHS fuel rod diameter is more than double that of S-PRISM, reducing the fuel fabrication cost per kg HM. A preliminary analysis recently performed indicates that the ENHS equilibrium fuel cycle cost could be less than 0.5 cents per kWh (in 2002 \$), which is very reasonable. The fuel cycle cost for the first loading could be somewhat higher, but below 1 cent/kWh. A more thorough study is needed.

5.8 Financial risk. The ENHS NPP is of a highly modular and standard design. All the modules are fabricated and undergo quality assurance in factory. The installation of the modules

is relatively simple so that the probability of cost overruns is small. In addition, the superb safety of the ENHS implies that the integrity of the reactor will be preserved even under the worst credible accidents. The net result is a very low financial risk.

6. List of ENHS Related Publications

1. A.S. Boloori, M. Frank, E. Greenspan, E. Hill, D.M. Hutchinson, S. Jones, X. Mahini, M. Nichol, B.H. Park, H. Shimada, N. Stone and S. Wang, "Once-for-Life Fueled, Highly-Modular, Simple, Super-Safe, Pb-Cooled Reactors," Transactions Am. Nucl. Soc., **80**, 200-201 (1999).
2. H. Shimada, E. Greenspan, N. Stone and S. Wang, "Long-Life Lead-Cooled Core Fueled with Enriched Uranium," Transactions Am. Nucl. Soc., **80**, 199-200 (1999).
3. E. Greenspan and D.C. Wade, "Encapsulated Nuclear Reactor Heat Source Module," Transactions Am. Nucl. Soc., **80**, 197-199 (1999).
4. A.S. Boloori, M. Frank, E. Greenspan, E. Hill, D.M. Hutchinson, S. Jones, X. Mahini, M. Nichol, B.H. Park, H. Shimada, N. Stone and S. Wang, "Once-for-Life Fueled, Highly-Modular, Simple, Super-Safe, Pb-Cooled Reactors," Proc. Global'99: International Conf. On Future Nuclear Systems, Jackson Hole, Wyoming, August 29 - September 3, 1999. Paper No. 183.
5. 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," Proc. 8th Int. Conf. On Nuclear Engineering, Paper ICONE-8750 Baltimore, MD, April 2-6, 2000.
6. E. Greenspan, H. Shimada and K. Wang, "Long-Life Cores with Small Burnup Reactivity Swing," Proc. of the 2000 Int. Topical Mtg. On Advances in Reactor Physics and Mathematics and Computation into the Next Millennium, PHYSOR-2000, Pittsburgh, PA, May 7-11, 2000.
7. 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 for Developing and for Industrial Countries," Proc. of the 5th Nuclear Energy Symposium, Beijing, China, March 27-29, 2000.
8. E. Greenspan and the ENHS Project Team, "The Encapsulated Nuclear Heat Source Reactor for Low-Waste Proliferation-Resistant Nuclear Energy," Invited Paper for the International Seminar on Nuclear Energy with Nearly Zero Release, Susono City, Japan, Nov. 6-9, 2000. To be published in Journal of Nuclear Energy.
9. E. Greenspan and N. Brown, "Small Innovative Reactor Designs – Useful Attributes and Status of Technology," Proc. of a Workshop on New Energy Technologies: A Policy Framework for Micro-Nuclear Technology, The James A. Baker III Institute for Public Policy of Rice University, March 19-20, 2001.
10. L. Conway, Q. Hossain, D. C. Wade, N. W. Brown, M. D. Carelli, M. Dzodzo, E. Greenspan, W.E. Kastenberg, D. Saphier and J. J. Sienicki, "Promising Design Options For The Encapsulated Nuclear Heat Source Reactor" Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
11. J.J. Sienicki and D.C. Wade, "Thermal Hydraulic Analysis of the Encapsulated Nuclear Heat Source," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
12. D. Saphier, E. Greenspan, D. C. Wade, M. Dzodzo, L. Conway and N.W. Brown, "Some

- Safety Aspects of the Encapsulated Nuclear Heat Source LMR Plant' Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
13. D. Saphier, M. Dzodzo and E. Greenspan, "Applying the DSNP Modular Modeling System to Transient and Accident Simulations of Lead Cooled Reactors," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
 14. M. Dzodzo and D. Cuckovic-Dzodzo, "Natural Convection in Enclosures Containing Lead-Bismuth and Lead," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
 15. E. Greenspan, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, Q. Hossain, D. Saphier, J. J. Sienicki, D. C. Wade, "The Encapsulated Nuclear Heat Source Reactor – A Generation IV Reactor," Proc. of GLOBAL-2001, Paris, France, September 2001.
 16. E. Greenspan, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, Q. Hossain, D. Saphier, J. J. Sienicki, D. C. Wade, "The Encapsulated Nuclear Heat Source Reactor for Proliferation-Resistant Nuclear Energy," Proc. of GLOBAL-2001, Paris, France, September 2001.
 17. N. Brown, E. Greenspan, M. Carelli, L. Conway, M. Dzodzo, Q. Hossain, J. Sienicki, D. Saphier, H. Shimada and D. Wade, "The Encapsulated Nuclear Heat Source Reactor for Low-Waste Proliferation-Resistant Nuclear Energy," Proc. of the IAEA Conf. On Small Nuclear Reactors, Cairo, Egypt, 27-31 May 2001. IAEA-SR-218-47.
 18. E. Greenspan, A. Barak, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, E. Feldman, Q. Hossain, D. Saphier, J. J. Sienicki, T. Sofu, D. C. Wade, "The Encapsulated Nuclear Heat Source Reactor Concept," Trans. American Nuclear Society, 85, 71-73, Reno, NV, Nov. 2001.
 19. E. Greenspan, A. Barak, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, E. Feldman, Q. Hossain, D. Saphier, J. J. Sienicki, T. Sofu, D. C. Wade, "The Encapsulated Nuclear Heat Source Potential for Meeting Generation-IV Goals," Trans. American Nuclear Society, 85, 73-74, Reno, NV, Nov. 2001.
 20. J.J. Sienicki, "Feasibility of Natural Circulation Heat Transport in the ENHS", Trans. American Nuclear Society, 85, 74-75, Reno, NV, Nov. 2001.
 21. L. Conway, D. Saphier and M. Dzodzo, "Encapsulated Nuclear Heat Source Compact Modular Steam Generators," Trans. American Nuclear Society, 85, 75-77, Reno, NV, Nov. 2001.
 22. A. Barak and E. Greenspan, "ENHS Reactor Energy-Conversion Efficiency and Steam Generators Size," Trans. American Nuclear Society, 85, 77-78, Reno, NV, Nov. 2001.
 23. D. Saphier, D. C. Wade, N.W. Brown, L. Conway, M. Dzodzo, E. Greenspan, "An Inherently Safe Modular LMR Plant - the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, 85, 78-80, Reno, NV, Nov. 2001.
 24. M. Dzodzo, E. Greenspan and L. Conway, "Nested Channel Intermediate Heat Exchanger Design Concept for the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, 85, 80-81, Reno, NV, Nov. 2001.
 25. E.E. Feldman and T. Sofu, "Feasibility of Melting the Lead or Lead-Bismuth in which the ENHS Core is Embedded," Trans. American Nuclear Society, 85, 81-82, Reno, NV, Nov. 2001.
 26. S.G. Hong and Y.I. Kim, "Comparison of the ENDF/B-VI and JEF2.2 Nuclear Data for the ENHS Benchmark Problem," Trans. American Nuclear Society, 85, 111-113, Reno, NV, Nov. 2001.

27. K.G. Lee and S.H. Chang, "A Feasibility of Sodium Cooled ENHS," Trans. American Nuclear Society, **85**, 38-39, Reno, NV, Nov. 2001.
28. D. C. Wade, A. Barak, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, E. Feldman, E. Greenspan, Q. Hossain, D. Saphier, J. J. Sienicki, T. Sofu, "ENHS: The Encapsulated Nuclear Heat Source – A Nuclear Energy Concept for Emerging Worldwide Energy Markets," Proc. International Conference on Nuclear Engineering, ICONE-10, Arlington, VA, April 14-18, 2002.
29. E. Greenspan and the ENHS project team, "The Long-Life Core Encapsulated Nuclear Heat Source Generation IV Reactor," Proc. of the International Congress on Advanced Nuclear Power Plant, ICAPP, Hollywood, FL, June 9-13, 2002.
30. J. Buongiorno, A. Barak and E. Greenspan, "Irradiation Performance of the Metal Fuel for the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, **86**, 348-349, Hollywood, FL, June 2002.
31. S.G. Hong, E. Greenspan and Y.I. Kim, "Once-for-Life Core for the Encapsulated Nuclear Heat Source (ENHS) Reactor," Trans. American Nuclear Society, **86**, 143-145, Hollywood, FL, June 2002.
32. E. Greenspan, A. Barak, N. Brown, M. Carelli, L. Conway, M. Dzodzo, E. Feldman, D. Saphier, J. Sienicki, T. Sofu and D. Wade, "The Encapsulated Nuclear Heat Source Reactor For Countries With Small And Medium Electricity Grid," Proc. of the 4th Int. Conf. On Nuclear Option in Countries with Small and Medium Electricity Grids, Dubrovnik, Croatia, June 16-20, 2002.
33. S.G. Hong, E. Greenspan and Y. I. Kim, "Once-for-Life Core for the Encapsulated Nuclear Heat Source (ENHS) Reactor," Proc. International Conference on the New Frontiers of Nuclear Technology : Reactor Physics, Safety and High-Performance Computing, PHYSOR-2002, Seoul, Korea, October 7-10, 2002.
34. D. Barnes, M. Milosovic', H. Sagara, K. Wang, E. Greenspan, J. Vujic, Z. Shayer, K. Grimm, R. Hill, S.G. Hong and Y.I. Kim, "The ENHS Core Benchmark," Proc. International Conference on the New Frontiers of Nuclear Technology : Reactor Physics, Safety and High-Performance Computing, PHYSOR-2002, Seoul, Korea, October 7-10, 2002.
35. S. G. Hong et al., "A Comparative Neutronics Analysis by Using ENDF/B-VI and JEF2.2 Libraries for the ENHS Benchmark Problem," Proc. International Conference on the New Frontiers of Nuclear Technology : Reactor Physics, Safety and High-Performance Computing, PHYSOR-2002, Seoul, Korea, October 7-10, 2002.
36. S. G. Hong, E. Greenspan, Y.G. Kim and Y.I. Kim, "Sodium versus Lead-Bismuth Coolants for the ENHS (Encapsulated Nuclear Heat Source) Reactor," Proc. of the Korean Nuclear Society Autumn Meeting, Yongpyong, Korea, October 2002.
37. J. J. Sienicki, "Updated Thermal Hydraulic Analyses for Recent ENHS Design Improvements," Proc. of the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
38. S. G. Hong, E. Greenspan and Y.I. Kim, "Alternative Design Options for the ENHS (Encapsulated Nuclear Heat Source) Reactor," Proc. of the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
39. Z. Shayer, M. Milosevic', D. Barnes and E. Greenspan, "Shielding Optimization for the ENHS Reactor," Proc. of the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.

40. D. Saphier, "Safety Assessment of Advanced Nuclear Power Plants with the DSNP Modular Modeling Systems," Proc. of the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
41. S.G. Hong, E. Greenspan, Y.I Kim and H.R. Kim, "ENHS (Encapsulated Nuclear Heat Source) Core Designs Using Nitride Fuel," to be published in Proc. of GLOBAL'03, New Orleans LA, Nov. 16-20, 2003.
42. Y. Nishi, N. Ueda, I. Kinoshita and E. Greenspan, "Simulation of the Thermal Hydraulic Characteristics of the ENHS," Submitted to Nuclear Technology.
43. T. Sakai and T. Iwasaki, "Thermal-Hydraulic Analysis on the Encapsulated Nuclear Heat Source (ENHS)", Submitted to the 2004 International Congress on Advances in Nuclear Power Plants, Pittsburgh, PA, June 13-17, 2004.

Project Summary Reports:

44. E. Greenspan, N. W. Brown, D. Carelli, L. Conway, M. Dzodzo, K. Grimm, R. Hill, Q. Hossain, D. Saphier, H. Shimada, J. J. Sienicki, D C. Wade and S. Wang, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232, February 2001.
45. E. Greenspan, A. Barak, D. Barnes, J. Buongiorno, N. W. Brown, D. Carelli, S.H. Chang, L. Conway, M. Dzodzo, E. Feldman, S.G. Hong, I.S. Hwang, S.H. Jeong, Y.I. Kim, K.G. Lee, M. Milosevic, B.G. Park, D. Saphier, T. Sofu, J. J. Sienicki, and D C. Wade, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 2nd year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4234.

Other References

46. B.F. Gromov, Yu.I. Orlov, G.I. Toshinsky et al., "Use of Lead-Bismuth Coolant in Nuclear Reactors and Accelerator-Driven Systems," Nuclear Engineering and Design, 173, 207-217, 1997.
47. S. Hattori and N. Handa, "Use of Super-Safe, Small and Simple LMRs to Create Green Belts in Desertification Area," Trans. American Nuclear Society, 60, 437 (1989).
48. K. Aoki, S. Kasai and S. Hattori, "Design of Small and Simple LMR Cores for Power Generation in Remote Communities," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
49. S. Hattori and A. Minato, "A Large Modular LMR Power Station Which Meets Current Requirements," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
50. C.E. Boardman, "Economic Assessment of S-PRISM Including Development and Generating Costs," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
51. Y. Kato, T. Nitawaki and Y. Yoshizawa, "A Carbon Dioxide Cooled Direct Cycle Fast Reactors," Proc. of GLOBAL-2001, Paris, France, September 2001. Paper 028.
52. P. Peterson, Personal communication with E. Greenspan, January 2002.
53. W.H. Hannum, D. Wade and G. Stanford, "Self-Protection in Dry Recycle Technologies,"

- Proc. of GLOBAL 1995, Vol. 1, p. 441, Versailles, France, Sept. 11-14, 1995.
54. D.E. Skorikov, "Fast Lead-Bismuth Cooled Reactor Installation SVBR-75/100," Book of Presentations in the Japan-Russia LBE Coolant Workshop, ISSN 0387-6144, Feb. 22-23, 2001. P. 9
 55. A.E. Dubberley, C.E. Boardman, K. Yoshida and T. Wu, "SUPERPRISM Oxide and Metal Fuel Core Designs," Proc. 8th Int. Conf. On Nuclear Engineering, ICONE-8, Baltimore, MD, April 2-6, 2000. Paper 8002.
 56. H. Branover, S. Lesin and M. Tsirlin, "Some Aspects Of Direct Interaction of Volatile Liquids With Hot Liquid Metals," Proc. of GLOBAL 1999, Jackson Hole, WY, Aug. 29 – Sept. 3, 1999.
 57. L. Cinotti, "Concept and development of a Pb-Bi cooled experimental ADS", Embedded Topical Meeting Acc/App/ADTTA '01 at the ANS Winter Meeting, Reno, NV, Nov. 11-15, 2001.
 58. Y. Karni, E. Greenspan, J. Bendahan and T. Gozani, "Effective Materials and Optimal Design for Radiation Shields," Proc. of the American Nuclear Society Topical Mtg. on New Horizons in Radiation Protection and Shielding, Pasco, WA, April 26 - May 1, 1992, pp. 154-161.
 59. Y. Karni, D. Regev, E. Greenspan and J. Vujic, "Two Dimensional Nuclear Design Optimization Capability," Proc. 8th Int. Conf. on Nuclear Engineering, Paper ICONE-8501, Baltimore, MD, April 2-6, 2000.
 60. O.G. Grigoriev, "Concept of the Modular NPP with SVBR-75/100 Lead-Bismuth Cooled Fast Reactors," Proc. of the Japan-Russia LBE Coolant Workshop, TIT Research Lab. For Nuclear Reactors, Tokyo, Japan, Feb. 22-23, 2001.
 61. J.S. Lee et al., KAERI and H. Keil, AECL, "Research and Development Program of KAERI for DUPIC," DUPIC program in Korea," Proc. Int. Conf. On Future of Nuclear Systems, GLOBAL'93, 2, 733-739, Seattle, WA, September 1993.

1. INTRODUCTION

1.1 Background

In 1996 Lawrence Livermore National Laboratory (LLNL), supported by the University of California (UC) Nuclear Engineering Department, initiated a Laboratory Directed Research and Development (LDRD) program to investigate the possibility of developing a nuclear fuel cycle that would be more proliferation resistant than the current LWR once-through fuel cycle^{1,1}. The objective was to identify a nuclear power system concept that would be highly proliferation resistant, more user-friendly and available to developing countries. The LLNL effort, headed by Neil Brown, contributed significantly to developing the scope of the NERI program and the attributes now specified for the Generation IV reactors.

During the two-year study period of FY97-FY98, the UC/LLNL team identified and studied the potential for a small liquid metal cooled reactor (LMR) to satisfy the requirements developed by LLNL to improve proliferation resistance^{1,2, 1.3}. The work was initially based on the Central Research Institute of Electric Power Industry of Japan (CRIEPI) Super-Safe, Small and Simple (4S) reactor design^{1,4, 1.5, 1.6} that included a long life core that was needed to achieve some of the LLNL objectives. The UC/LLNL study identified new core designs for long life, including the fuel-self-sustaining core that was later adopted for the ENHS.

In the summer of 1998 LLNL organized a consortium of national laboratories, industries and universities to jointly prepare proposals for a Secure, Transportable, Autonomous Reactor (STAR) system concept^{1,7} for the forthcoming NERI program. The Encapsulated Nuclear Heat Source (ENHS) reactor concept was invented in the summer of 1998 by Dave Wade of ANL and by Ehud Greenspan of UCB while brainstorming on the liquid-metal cooled reactor concept to propose for the NERI program. The idea of encapsulating a core inside a reactor vessel and conducting the fission energy through the vessel wall to a secondary coolant pool that contains steam generators was first proposed by Dave Wade^{1,8}. Preliminary analysis led Dr. Wade to conclude that the required vessel height is too large to make the concept practical^{1,8}. This finding led Ehud Greenspan to conceive of the corrugated confinement wall that weaves the space in between two structural walls^{1,9}. The NERI proposal of the ENHS combined this innovative idea of corrugated confinement wall with the long-life core conceived at UCB and with the control elements of the 4S reactor^{1,10}. While preparing the NERI proposal Dave Wade suggested^{1,11} also to consider an ENHS design concept in which PRISM-type IHXs are welded onto the outside of the primary canister with primary coolant passages cut through the canister wall into the shell side of the IHX. In the course of the three years ENHS NERI project there have been several modifications to the original ENHS concept. These modifications will be described in Section 2.

The Reactor Design and Analysis course of the Fall of 1998 taught at UC Berkeley by Ehud Greenspan adopted the ENHS reactor concept for the course design project. In a team work the students carried out a preliminary feasibility study of the ENHS concept. This study concluded that it is feasible to design an ENHS to deliver 125 MW_{th} with acceptable primary-to-secondary

coolant temperature drop and with 100%^a natural circulation^{1,12, 1.13}. These findings provided important input for the ENHS NERI proposal.

1.2 Study Goals

The purpose of this study was to assess the feasibility of the ENHS reactor concept. Twenty-five issues that deserve examination have been identified and defined as the tasks of this feasibility study:

1. Feasibility of the reactor vessel for the ENHS
2. Minimum thickness of the ENHS confinement wall
3. Minimum thickness of the ENHS structural wall
4. Feasibility of natural circulation of the primary coolant
5. How to incorporate pumps in the ENHS
6. Feasibility of natural circulation of the secondary coolant
7. Feasibility of decay heat removal from the ENHS
8. Feasibility of interfacing the ENHS with steam generators (SG) without enthalpy loss and without mechanical connections.
9. Feasibility of controlling the ENHS core reactivity and power level.
10. Feasibility of start-up with 100% natural circulation.
11. Feasibility of designing the reactor pool without a guard vessel.
12. Feasibility of 100% passive safety
13. Feasibility of inserting (and removing) the ENHS into (from) the reactor pool.
14. Feasibility of in-power replacement of ENHS or SG in a multi-ENHS module power plant.
15. Feasibility of transporting the ENHS with fresh fuel and with spent fuel
16. Feasibility of using the reactor vessel as a spent fuel storage cask.
17. Feasibility of alternative coolant for the ENHS
18. Feasibility of designing an ENHS using a stubby heat-pipe layer in the radial reactor vessel wall.
19. Feasibility of designing an ENHS with an external connection-free IHX integrated with the reactor vessel wall.
20. Overall Feasibility Assessment
21. Promising specialized applications of the ENHS
22. Feasibility assessment of multiple-ENHS modules for central power plants
23. Domain of design variables for long-life cores
24. Feasibility of safe operation with positive void coefficient at the core center
25. Feasibility of high-energy conversion efficiency

The order of the study did not follow the task order in the list. Also, in the course of the study the title of some of the tasks was modified but all of the feasibility issues were addressed. Close to half of these were examined during the first year. These feasibility issues are listed below and their assessment is summarized in Reference 1.14:

1. Feasibility of designing cores for at least 15 years of full power operation without refueling and with nearly zero burnup reactivity swing.

^a By 100% natural circulation we mean that the ENHS module can operate at 100% of nominal power with natural circulation of the primary coolant.

2. Feasibility of designing the ENHS module to be free of mechanical connections and to be of reasonable dimensions and weight.
3. Feasibility of designing and fabricating the confinement wall to maintain integrity and to provide the desired performance.
4. Feasibility of designing the ENHS to have 100% natural circulation with reasonable module dimensions.
5. Feasibility of fueling the ENHS module in the factory, transporting it fuelled to site, inserted & removed from pool unopened.
6. Feasibility of operating the ENHS autonomously.
7. Feasibility of decay heat removal.
8. Feasibility of using a heat exchanger instead of confinement wall.
9. Feasibility of using “stubby” heat-pipes instead of confinement wall.

The following studies have been pursued during the second year of the ENHS project and are summarized in Reference 1.15:

1. Completion of the benchmarking of the computational tools and data libraries used for the neutronic design of the ENHS core.
2. Independent evaluation of the neutronic design of the ENHS core.
3. Establishment of a data base for thermo-physical properties of Pb and of Pb-Bi to be used in the ENHS project.
4. Refinement of the analysis of the feasibility of natural circulation in the ENHS.
5. Independent evaluation of the feasibility of natural circulation in the ENHS and optimization of the ENHS module dimensions.
6. Performance of a three-dimensional CFD analysis of the secondary coolant flow.
7. Refinement of the design of the steam generators for the ENHS reactor.
8. Estimation of the thermodynamic efficiency for converting fission energy to electricity in the ENHS nuclear power plant.
9. Analysis of the sensitivity of the ENHS reactor safety to assumptions made and to design variations.
10. Conceptual design of an alternative Intermediate Heat Exchanger (IHx) in the form of nested tubes.
11. Assessment of the feasibility of melting the Pb-Bi in reasonable time after insertion of the ENHS module to the reactor pool without having to use special heater in the module.
12. Assessment of the feasibility of passively freezing the liquid metal around the fuel in reasonable time after removal of the ENHS module from the reactor pool.
13. Development of scenarios for installing the ENHS module in the reactor pool and for removing it from the pool after 20 years of operation.
14. Development of scenarios for transporting the fueled ENHS module from the factory to the reactor site and for transporting the fueled module from the site to a regional or international fuel center.
15. Development of a method for automated control of the oxygen level in the Pb-Bi coolant.

In addition, we devoted time and effort for assessing the ability of the ENHS reactor concept to meet the goals set for Generation IV reactors.

During the third year of the project we refined the analysis of some of the tasks already addressed and completed the remaining tasks.

In addition to the 25 tasks originally defined for this study, we undertook a number of new studies deemed important in the course of the project. Among the additional studies undertaken are the following:

1. Feasibility assessment of a cover gas lift-pump for both the primary and the secondary coolants^{1.16, 1.17}.
2. Feasibility assessment of a nested tube IHX^{1.18}.
3. Assessing the feasibility integrating of the steam generator within the IHX^{1.19}.
4. Feasibility of multi-recycling the fuel from the spent ENHS module using a proliferation-resistant dry process such as AIROX^{1.20, 1.21}.
5. Use of supercritical CO₂ instead of water for the working fluid of the energy conversion (EC) system.
6. A preliminary estimation of the market potential for ENHS reactors^{1.22}.
7. A preliminary estimation of the fuel cycle cost for ENHS reactors.

1.3 Participating Organizations

The U.S. project team working on this DOE NERI sponsored project consisted of researchers from four institutions: The University of California at Berkeley (the lead organization), Lawrence Livermore National Laboratory, Argonne National Laboratory and Westinghouse. Three Korean organizations joined the project at the beginning of the second year and kept collaborating with the US team until the completion of the project two years thereafter. These organizations are Korea Atomic Energy Research Institute (KAERI), Korea Advanced Institute for Science and Technology (KAIST) and Seoul National University. The Korean organizations were supported by the Nuclear Energy Research Initiative program of the Korean Ministry of Science and Technology (MOST).

Throughout the project we had useful interaction with CRIEPI, and to a lesser extent, TOSHIBA researchers that were involved in the design of the 4S reactor. In the second part of our project CRIEPI undertook to perform an independent evaluation of the transient behavior and safety characteristics of the ENHS reactor.

1.4 Report Layout

Most of the research done for this project and the obtained results are described in more than 40 publications that are openly available. Hence in the following we'll provide a general description of the ENHS reactor concept (Section 2) and will focus on a description of the reference ENHS reactor design we came up with and summarize its characteristics (Section 3). A number of alternative designs worked out are described in Section 4. This will be followed by a discussion of how the ENHS reactor can meet the goals of Generation IV reactors (Section 5) and by a brief summary of findings of this study (Section 6). Information that may be useful for future use is included in the appendices: recommended physical properties of Pb and of Pb-Bi eutectic are compiled in Appendix A and data, assumptions and methodology used for the fuel cycle analysis are summarized in Appendix B.

1.5 References

- 1.1 E. Greenspan, E. Elias, "Proliferation Resistant Fission Energy Systems", Int. Conf. on Future Nuclear Systems, Global 97, 2, p. 879-884, Yokohama, Japan, Oct. 5-7, 1997.
- 1.2 E. Greenspan, E. Elias, W.E. Kastenberg, N. Stone, K. Aoki and N.W. Brown, "Compact Once-for-Life Fueled Reactors for Developing Countries," Trans. American Nuclear Society, 78, 239-241 (1998).
- 1.3 E. Greenspan, E. Elias, W.E. Kastenberg, N. Stone, "Non-Proliferation Liquid Metal Cooled Reactors for Developing Countries", Center for Nuclear and Toxic Waste Management, University of California, CNTWM 98-2, Feb. 15, 1998.
- 1.4 S. Hattori and N. Handa "Use of Super-Safe, Small and Simple LMRs to Create Green Belts in Desertification Area," Transaction Amer. Nucl. Soc., 60, 437, 1989.
- 1.5 S. Hattori and A. Minato, "Current Status of 4S Plant Design", 2nd ASME-JSME International Conference On Nuclear Engineering, (ICONE-2), California, Mar. 21-24, 1993.
- 1.6 S. Hattori and A. Minato, "A Large Modular LMR Power Station Which Meets Current Requirements," Proc. 3rd ASME-JSME Int. Conf. on Nucl. Eng., ICONE-3, Tokyo, Oct. 1993.
- 1.7 N. W. Brown, J. A. Hassberger, C. Smith, E. Greenspan, D.C. Wade, R.N. Hill, M.D. Carelli, K.L. Peddicord, K.R. Stroh, "The Secure, Transportable, Autonomous Reactor System", International Conference on Future Nuclear Systems, Global 99, Jackson Hole, Wy., Aug. 29-Sept.3, 1999.
- 1.8 D.C. Wade, "Notions on Use of Pb Coolant to Simplify & Eliminate Systems," ANL Intra-Lab. Memo, July 2, 1998. Followed by a Fax to E. Greenspan dated July 31, 1998.
- 1.9 E. Greenspan, E-mail to D.C. Wade dated August 6, 1998. E-mail Subject: "Response to your letter dated July 1" (Reference 1.8). Also, fax to D.C. Wade and N.W. Brown of E. Greenspan's hand-written Memo: "Convolutd Reactor Vessel Wall Thickness for $\Delta T = 35^{\circ}\text{C}$," dated August 29, 1998.
- 1.10 D.C. Wade, E-mail to E. Greenspan and N.W. Brown dated November 5, 1998.
- 1.11 E. Greenspan and D.C. Wade, "Encapsulated Nuclear Reactor Heat Source Module," Transactions Am. Nucl. Soc., 80, 197-199 (1999).
- 1.12 H. Shimada, E. Greenspan, N. Stone and S. Wang, "Long-Life Lead-Cooled Core Fueled with Enriched Uranium," Transactions Am. Nucl. Soc., 80, 199-200 (1999).
- 1.13 A.S. Boloori et al., "Once-for-Life Fueled, Highly-Modular, Simple, Super-Safe, Pb-Cooled Reactors", Proc. Int. Conf. On Advances in Fuel Cycle, GLOBAL'99, Jackson Hole, Wyoming, Aug. 29-Sept. 3, 1999. See also in Transactions Am. Nucl. Soc., 80, 200-201 (1999).
- 1.14 E. Greenspan, N. W. Brown, D. Carelli, L. Conway, M. Dzodzo, K. Grimm, R. Hill, Q. Hossain, D. Saphier, H. Shimada, J. J. Sienicki, D C. Wade and S. Wang, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232, February 2001.
- 1.15 E. Greenspan, A. Barak, D. Barnes, J. Buongiorno, N. W. Brown, D. Carelli, S.H. Chang, L. Conway, M. Dzodzo, E. Feldman, S.G. Hong, I.S. Hwang, S.H. Jeong, Y.I. Kim, K.G. Lee, M. Milosevic', B.G. Park, D. Saphier, T. Sofu, J. J. Sienicki, and D C. Wade, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 2nd year of

- NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4234.
- 1.16 J.J. Sienicki and D.C. Wade, "Thermal Hydraulic Analysis of the Encapsulated Nuclear Heat Source," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 1.17 J. J. Sienicki, "Updated Thermal Hydraulic Analyses for Recent ENHS Design Improvements," Submitted to the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
- 1.18 M. Dzodzo, E. Greenspan and L. Conway, "Nested Channel Intermediate Heat Exchanger Design Concept for the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, 85, 80-81, Reno, NV, Nov. 2001.
- 1.19 M. Dzodzo and D. Cuckovic-Dzodzo, "Natural Convection in Enclosures Containing Lead-Bismuth and Lead," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 1.20 E. Greenspan and the ENHS Project Team, "The Encapsulated Nuclear Heat Source Reactor for Low-Waste Proliferation-Resistant Nuclear Energy," Invited Paper for the International Seminar on Nuclear Energy with Nearly Zero Release, Susono City, Japan, Nov. 6-9, 2000. To be published in Journal of Nuclear Energy.
- 1.21 E. Greenspan, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, Q. Hossain, D. Saphier, J. J. Sienicki, D. C. Wade, "The Encapsulated Nuclear Heat Source Reactor for Proliferation-Resistant Nuclear Energy," Proc. of GLOBAL-2001, Paris, France, September 2001.
- 1.22 D. C. Wade, A. Barak, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, E. Feldman, E. Greenspan, Q. Hossain, D. Saphier, J. J. Sienicki, T. Sofu, "ENHS: The Encapsulated Nuclear Heat Source – A Nuclear Energy Concept for Emerging Worldwide Energy Markets," Proc. International Conference on Nuclear Engineering, ICONE-10, Arlington, VA, April 14-18, 2002.

2. GENERAL DESCRIPTION OF THE ENHS REACTOR CONCEPT

2.1 Originally Conceived Single Module Reactor

The originally conceived ENHS reactor is described in reference ER5^(a) and will not be elaborated upon here. The primary conceptual change relative to the original concept is the abandonment of the “corrugated confinement wall” in favor of an intermediate heat exchanger (IHX) that features closed ducts. This did not change the basic features of the original reactor concept but made the ENHS module^(b) (the “Module”) more robust and more practical to fabricate.

2.2 Design Variations

Relative to the initially conceived and analyzed ENHS module, the following conceptual design variation were made in the course of the project:

1. Replacement of the corrugated wall by an IHX made of rectangular channels.
2. Designing the ENHS module vessel walls to be of a uniform diameter throughout; in some of the early thinking the vessel diameter was enlarged at its lower part.
3. Increase in the fission gas plenum volume from 50% to 100% of the fuel volume.
4. Increase in the fuel diameter from 1.0 cm to 1.3 cm.
5. Addition of a radiation shield in between the inner and outer structural walls of the Module, in the section below the IHX.

In Section 2.3 we’ll describe the present design concept of the ENHS that is designed to have natural circulation of both the primary and secondary coolants. In Section 2.4 we’ll describe a variant design that uses cover-gas lift-pumps for both the primary and the secondary coolants.

2.3 ENHS Reactor Concept

A schematic description of the design concept of the ENHS reactor is depicted in Figures 2.1 and 2.2. The nuclear steam supply system (NSSS) consists of one ENHS module and eight small steam generators. There is no mechanical connection between the Module and the steam generators. Both primary and secondary coolants flow by natural circulation. The primary coolant that is heated in the core flows up the riser, turns over into the Intermediate Heat Exchanger (IHX) and flows back into the coolant plenum underneath the core. In a vertical counterflow arrangement the secondary coolant flows from the pool outside of the vessel into the bottom of the IHX and exits back to the pool near the top of the IHX. The IHX, depicted in Figures 2.2 and 2.3, consists of rectangular channels that are connected at their top and bottom to a tube sheet (not shown in Figure 2.2). The 4 mm thick rectangular channel walls provide the barrier between the primary and the secondary coolants whereas the inner and outer walls provide the structural support. More conventional IHX made of circular tubes could be used as well. Relative to circular tube IHX the rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid

^(a) The notation ER stands for “ENHS Related publication list” given above at the end of the Executive Summary. The number 5 stands for the publication number in that reference list.

^(b) By ENHS module, or the Module, we refer to everything enclosed by the outer vessel of the ENHS.

spacers. The mean temperature difference between the primary and secondary coolants is $\leq 50^{\circ}\text{C}$. Figure 2.3 shows a vertical cut through a representative ENHS module.

The eight steam generators (SG), shown in Figures 2.4 and 2.5, are anchored to the support structure that covers the pool and are not mechanically connected to the Module. They are designed to meet several unique requirements that are dictated by the ENHS reactor layout: (1) Effective utilization of the pool volume surrounding the Module. (2) Minimum friction losses so as to enable 100% natural circulation of the intermediate coolant. (3) Having no mechanical connection with the Module. (4) Minimum flow rate of water into the intermediate coolant pool in case of a breach in steam generator tube or failure of other water-containing component. (5) Accommodation of a large thermal expansion. (6) Ease of inspection and maintenance. (7) Modular design that is easy to install and replace.

Figure 2.1 A schematic vertical view of an ENHS reactor. Not to scale

Figure 2.2 A schematic vertical view of an ENHS module. Not to scale. Note: Secondary coolant outlet level should be below primary coolant inlet level to IHX, as in Figure 2.3

The steam generator is of a once-through tube-in-tube design; feed-water flows in via the inner tube and the steam is generated in the shell between the inner and outer tubes. The liquid metal (either Pb-Bi or Pb) coolant flows outside the tubes. Also, the steam and feed water piping and nozzles are located outside the ENHS pool, and the feed water to each steam tube is inherently orificed by a small diameter feed tube. These features all act to minimize the quantity and/or mass flow rate of water or steam that can be introduced into the pool due to a postulated steam line break, or feed line break, or tube rupture.

The ENHS core is made of uniform composition fuel rods with no blanket assemblies and designed to maintain a nearly constant k_{eff} over 20 years without refueling or fuel shuffling. The reference fuel is taken to be metallic alloy with 10 weight % Zr. Both enriched uranium and Pu-U are considered for the HM. The structural material is taken to be HT-9; the ferritic-martensitic SS used for EBR-II. The Module will be manufactured and fuelled in the factory and shipped to the site as a weld-sealed unit with solidified Pb-Bi filling the vessel up to above the fuel rods. A

unique feature of Pb-Bi that makes it possible to embed the fuel rods and core structure in solid Pb-Bi without damage is its nearly zero coefficient of volumetric expansion upon change of

Figure 2.3 Representative ENHS module featuring 100% natural circulation

phase^{2.1}. At the end of its core life the Module will be removed from the reactor pool after pumping-out Pb-Bi from above the fuel level and the remainder will be replaced by Pb. The module will be stored on site until the decay heat drops to a level that will permit to solidify the coolant. The Module with the solidified coolant will then serve as a shipping cask.

The fuel and the shutdown control assembly will be loaded into the module at the factory. Factory fabrication and assembly and subsequent disassembly of the entire ENHS module provide the opportunity to consider non-conventional core support and fuel loading options. Several core mechanical design and options have been considered and discussed in Reference 2.2. All involve assembly-less core with individual fuel rods tied to grid plate at the factory. The reference design concept adopted is illustrated in Figure 2.6. It is a whole-core macro pin bundle having no ducts, positioned at the bottom on grid plate pedestals, and with a pin spacer grid near the middle elevation of the pins is adopted for the reference design. It allows for free flowering core radial expansion, This “flowering” of the fuel rod will introduce negative temperature coefficient of reactivity.

The core would be supported vertically within the module from the bottom by the lower grid spacer plate. The pins would be held down against upward hydraulic flow and buoyancy forces by tying each pin to its pedestal on the lower grid plate in a way that allows for 2π steradian degree of freedom (see the insert in Figure 2.6). The pins float in the coolant like an array of buoys, but are pinned in position at the bottom, and are positioned radially and azimuthally near the mid elevation by the upper grid spacer. The bottom pinning is done in a diverse/redundant fashion so that a single failure will not release a pin.

Figure 2.4 ENHS steam generator overall elevation view and ENHS module layout plan

Figure 2.5 ENHS steam generator cross-section elevation view.

Figure 2.6 Schematics of one option for ENHS core support.

The use of pin pedestals allows to eliminate a bottom grid spacer and to thereby avoid its pressure drop – an important consideration for the natural circulation cooling. The coolant enters the lower plenum from the downcomer via large holes in the sides of the core barrel (see Figures 2.2 and 2.3). Then it distributes itself across the core radius, flowing radially between the pedestals, and turns and flows upward through the macro pin lattice. The large coolant gaps of the lattice and the pedestal placement facilitate the radial distribution both in the plenum and throughout the pin lattice.

The grid spacer shown in Figure 2.6 is illustrative only. Its design will be informed by the experience base in the Light Water Reactor industry and the approaches used for fast reactor designs in Germany, Russia, and England where fast reactor grid spacers have been used in place of wire wrap. Its two functions are to position the fuel pins to retain coolability and to dilate the pins radially as the power/flow ratio increases – see Figure 2.6. Pin fretting, coolant pressure loss, and grid spacer integrity in the face of flow induced vibrations and radiation damage are the design issues to be addressed.

The core loading could be assembled (remotely within a hot cell) on a bench and would remain highly subcritical owing to the absence of the Pb-Bi reflector; the central shutdown rod would, of course, be in place and the assembly of the pins into a full core loading would be conducted using a standard “approach to critical” based on 1/M control procedures.

Once assembled on the core support /lower grid plate/pedestal component and with the mid elevation grid spacer already in place, the whole core and support system would be moved to a position underneath the module – which would at that time not yet have the bottom head welded on. The module would be lowered from above onto a top port in the hot cell in a position preparatory for core insertion. The fuel and support system would be raised on a hydraulic lift into position inside the module and would be bolted into place from the bottom side – see Figure 2.7. Subsequently, the bottom head could be welded in place to close the bottom of the ENHS

Figure 2.7 A schematic illustration of core loading into the ENHS vessel

module and the lead-bismuth would be introduced from the top and allowed to solidify. Again, with the central rod in place, the addition of Pb-Bi reflector will not take the assembly to critical. Appropriate bagging procedures would be employed to maintain the outside surfaces of the ENHS module free of tramp radioactivity from inside the hot cell.

Once the module bottom head was welded in place, the fuel pins would be inaccessible until when (~20 years later) the expended module was returned to the factory and the bottom head was cut off for core dismantlement. Disassembly and fuel treatment would be done remotely in the same or a similar hot cell as had been used for assembly.

Currently, the Control and Shutdown Systems (C&SS) used in the 4S reactor of TOSHIBA-CRIEPI^{2,3-2.5} have been adopted for the ENHS. The C&SS consist of a single neutron absorber assembly located centrally in the core also referred to as the “Central Safety Assembly” and six segments of an annulus that surrounds the core sometimes referred to as the “absorbing curtain” or “peripheral absorbers”. Figure 2.8 shown the radial location of these control and safety elements. The active elements for both central absorber and peripheral absorbers are B_4C and tungsten; being heavier than Pb-Bi tungsten can scram by gravity. All of these absorbing elements are securely latched in place during shipping and reactor installation. The central absorber has an electromagnetic latch that does not engage until the startup temperature of 350°C is achieved. At this temperature the assembly can be withdrawn. Normal operational shutdowns can be accomplished with the peripheral absorbers. The reactor is brought critical by a hydraulic system that moves the peripheral absorbers up at 1 mm/sec to compensate for the negative temperature coefficient of reactivity. At the full power position, the peripheral absorber segments

Figure 2.8 ENHS control systems. Top view. Not to scale.

are stopped from further upward movement by mechanical stoppers whose movement is established by high-reliability gear drives. These drives restrict the rate of movement of the reflector segments to approximately 1 mm/day. It is anticipated that the height of the peripheral absorbers will be adjusted once a year or two to compensate for a slight drift in reactivity due to burnup.

2.4 Multiple ENHS Module Reactor

It is possible to construct a power plant made of multiple ENHS modules. There are a couple of general approaches to the design of multiple ENHS module plant: (A) Install several ENHS modules in a single pool of secondary coolant. (B) Use as many independent single module ENHS reactors as desirable in a single power plant.

Approach A is illustrated in Figure 2.9. The secondary coolant pool can have a rectangular cross section and made of thermally insulated concrete. Such a pool design concept is being proposed for a single core, high power, BREST reactor^{2,6}. Steam generators, superheaters, possibly

Figure 2.9 A schematic horizontal view of a 10-ENHS module reactor using insulated

reheaters and possibly secondary coolant pumps are also inserted into the secondary coolant pool. Each of these components is a relatively small module, supported from a seismically isolated platform that covers the pool. There is no mechanical connection between the modules, so it is relatively simple to install and replace these components. In case of some failure in one of the modules, it will be probably most economical to replace it with a new module. If economically justified, the failed module could be fixed “off line” and used as a spare. The secondary reactor pool of Figure 2.9 has a BREST-like RVACS system for decay heat removal.

Approach B is illustrated in Figure 2.10. The power plant consists of 12 ENHS reactors, each including its own reactor pool, steam generators and turbine-generator. The total capacity of this power plant is 600 MWe. This arrangement provides the utmost level of uniformity and modularity. Additionally, this arrangement is most suitable for a gradual increase in the installed capacity of the power plant so as to best fit the increase in demand for electricity.

Figure 2.10 A schematic horizontal view of a 12-ENHS reactor power plant for 600 MWe. Each reactor can be operated as a stand-alone unit. There are common control, services etc. Not to scale.

2.5 ENHS Module with Cover-Gas Lift-Pump

An alternative ENHS concept developed in the course of this project is depicted in Figure 2.11. 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. 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 outside of the Module vessel and above its top cover. Thus, no moving mechanical parts and no electrical parts associated with coolant circulation are located inside the ENHS module. Use of a lift-pump enables reduction of the riser length and, correspondingly, of the length of the Module by nearly 10 meters! However, in order to accommodate the corresponding reduction in the length of the IHX it is necessary to increase the number of IHX channels and the channels width. This will lead to at least a 20 cm increase in the Module diameter.

The ENHS reactor concept shown in Figure 2.11 also features a cover gas lift pump for the secondary coolant. Our present conception incorporates the lift pump in the annular space between the inner and outer structural walls, just above the IHX. In the drawing shown in Figure 2.11 the IHX extends all the way down to close to the Module bottom. This arrangement was conceived before we realized that it is necessary to incorporate a shield in the radial direction around the core. This may lead to a ~40 cm increase in the Module diameter.

Figure 2.11 Representative ENHS module featuring cover-gas lift pump both in the primary and in the secondary coolant systems.

The lift-pump version of the ENHS is subjected to partial loss of flow accidents that are not applicable to the reference (100% natural circulation) ENHS reactor. These new accidents include loss of power to drive the gas circulators, circulator breakdown or cover gas leakage. Analysis of such accidents was beyond the scope of this work. Nevertheless it appears that it would be possible to design the lift-pump ENHS module in such a way that decay heat could efficiently be removed by passive natural circulation in case loss of flow accidents.

2.6 References

- 2.1 B.F. Gromov, Yu.I. Orlov, G.I. Toshinsky et al., "Use of Lead-Bismuth Coolant in Nuclear Reactors and Accelerator-Driven Systems," Nuclear Engineering and Design, 173, 207-217, 1997.
- 2.2 E. Greenspan, N. W. Brown, D. Carelli, L. Conway, M. Dzodzo, K. Grimm, R. Hill, Q. Hossain, D. Saphier, H. Shimada, J. J. Sienicki, D C. Wade and S. Wang, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232, February 2001.
- 2.3 S. Hattori and N. Handa, "Use of Super-Safe, Small and Simple LMRs to Create Green Belts in Desertification Area," Trans. American Nuclear Society, 60, 437 (1989).
- 2.4 K. Aoki, S. Kasai and S. Hattori, "Design of Small and Simple LMR Cores for Power Generation in Remote Communities," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
- 2.5 S. Hattori and A. Minato, "A Large Modular LMR Power Station Which Meets Current Requirements," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
- 2.6 E. Adamov et al., "The Next Generation of Fast Reactors," Nuclear Engineering and Design, 173, pp. 143-150 (1997).

3 REFERENCE ENHS MODULE

3.1 Introduction

This section describes the design of the reference ENHS module and summarizes its calculated characteristics. The reference ENHS reactor considered uses natural circulation, Pu-UZr(10) metallic fuel with the Pu taken from typical PWR spent fuel and Pb-Bi coolants. A number of design variants are described in Section 4. These include core designs that use different fuel composition and designs that use different primary and secondary coolants. The alternative fuel compositions include Pu recycled from fast reactors, enriched uranium, nitride fuel, enriched uranium fuel and thorium-based fuel. The alternative coolants considered include sodium for either the primary loop or for both the primary and secondary coolant loops.

3.2 Dimensions

Table 3.1 summarizes dimensions of the reference ENHS reactor designed for a nominal power output of 125 MW_{th} with primary coolant core inlet and outlet temperatures of, respectively, 400 and 550°C. These dimensions pertain to the ENHS reactor layout shown in Figures 2.1 thru 2.3.

Table 3.1 Dimensions

Parameter	Dimension (cm)
Fuel	
Fuel rod diameter	1.3
Clad thickness	0.13
Pitch to diameter ratio, P/D	1.36
Lattice pitch (triangular) [=1.36 x (1.3+0.26)]	2.12
Fuel length	125
Fission gas plenum length	125
Module radial dimensions	
Outer radius of central absorber	15.41
Thickness of central absorber	5.0
Inner radius of central absorber guide tube (above core)	15.91
Outer radius of central absorber guide tube	16.91
Core inner radius	16.41
Core outer radius	111.83
Thickness of Pb-Bi gap around core	2.0
Thickness of core barrel	2.0
Thickness of Pb-Bi gap between core barrel and peripheral absorber	1.0
Thickness of peripheral absorber assembly	5.0
Thickness of Pb-Bi gap beyond peripheral absorber assembly	1.0
Thickness of inner structural wall	4.5
Thickness of radial shield (adjacent to outer structural wall) ^(a)	20
Thickness of annulus between inner & outer structural walls	45

^(a) Located in lower part of vessel, starting ~20 cm below IHX; it occupies 20 out of 45 cm annulus

Thickness of outer structural wall	4.5
Central absorber drive guide tube outer diameter	32.5
Riser inner diameter	244
IHX inner diameter	253
IHX outer diameter	343
Module outer diameter	352
Pool radial dimensions – lower part	
Distance from Module outer wall to flow partition	6
Flow partition thickness	0.5
Distance from flow partition to inner pool wall	9
Pool wall thickness	5
Pool inner diameter	383
Pool outer diameter	393
Pool radial dimensions – upper part	
Distance from Module outer wall to SG outer wall	14.5
SG outer diameter ^(b)	76
SG wall thickness	0.25
Distance from SG outer wall to pool inner wall	14.5
Pool inner diameter	562
Pool outer diameter	572
Axial dimensions – Module	
Bottom base	50
Shield (not included in previous design)	20
Coolant cavity	65
Core support plate	20
Fuel region	125
Fission gas plenum	125
IHX active bottom from bottom of Module ^(c)	405
IHX length	1300
From top of IHX active length to top of coolant level	115
Cover gas plenum	100
Top base	50
Total module length	1970
Axial dimensions – Pool	
Length of narrow section of pool	1240
Vertical extent of pool cone	50
IHX	
Primary channel inner radial width	40
Primary channel inner azimuthal width	2.5
Channel wall thickness	0.4
Number of channels	138
Number of locations for tube inserts between channels (5.8x45cm)	23

^(b) Details about steam generators are given below.

^(c) A 15cm thick tube sheet and support structure extend below and above the active length of IHX

3.3 Core Design

3.3.1 Geometry and Composition for Calculating the ENHS Core

Figure 3.1 describes the model used for the ENHS core and shield neutronic calculations. The scheme is not to scale. The table in the next page gives the composition and temperatures used for the core neutronic calculations. Table 3.2 summarizes the composition of the different zones defined in Figure 3.1.

Figure 3.1 Schematic vertical cut through the ENHS reactor core and adjacent components. Not to scale.

The core is a single bundle of a uniform composition fuel rods arranged in a triangular lattice and positioned at the bottom on grid plate pedestals. The core has no blanket rods and no solid reflector rods. The fuel and fission gas plenum height is 125 cm each. The clad inner radius and its thickness are 0.65 cm and 0.13 cm, respectively. The core average heat generation rate is 101.4 W/cm. Therefore the number of fuel rods in the core is taken to be $125 \times 10^6 \text{ W} / (125 \text{ cm} \times 101.4 \text{ W/cm}) = 9862$.

Table 3.2 Material Compositions and Temperatures Used for Calculating the ENHS Core

Region No.	Region name	Material volume fractions	Temp (K)
1	Bottom of absorber	99%Pb-Bi + 1%SS	698
2	Core region	50.97%Pb-Bi + 34% fuel+ 14.98%SS	698
4	Inner Pb-Bi ^a gap	100%Pb-Bi	698
5	Core barrel	100%SS	698
6	Outer Pb-Bi gap	100%Pb-Bi	623
7	Space for peripheral absorber	100%Pb-Bi	623
8	Inner structural wall	100%SS	623
9	Outer structural wall	100%SS	623
10	Lower grid plate	50%Pb-Bi + 50%SS	623
11	Below grid plate	80%Pb-Bi + 20%SS	623
12	Bottom base	100%SS	623
13	Fission gas plenum	50.97%Pb-Bi + 14.98%SS	773
14	Upper grid plate	50%Pb-Bi + 50%SS	773
15	Upper reflector	100%Pb-Bi	773
16	Down comer	100%Pb-Bi	624
17	Radial shield	10%B ₄ C+90%ZrH _{1.6}	623
18	Axial shield	10%B ₄ C+90%ZrH _{1.6}	623
19	Pb-Bi region of absorber	100%Pb-Bi	698
20	Space for central absorber	100%Pb-Bi	698
21	Gap of absorber	100%Pb-Bi	698
22	Top of absorber	20%SS	698

The Control and Shutdown Systems (C&SS) designed for the 4S reactor^{3.1-3.3} by TOSHIBA and CRIEPI have been adopted for the ENHS with slight modification. The active elements of the C&SS are a single neutron absorber assembly located centrally in the core and six segments of an annulus containing neutron absorbers that surrounds the core. The cross sectional area of the central absorber region is equal to that of 217 unit cells. During normal operation this central site is filled with coolant and there are no control rods in the core. The operational reactivity control is done by finely moving six annular segments of a peripheral absorber (Region 7 in Fig. 3.1; denoted as “reflector”) that surrounds the core. The absorbing materials for both central and peripheral absorbers are B₄C and tungsten; being heavier than Pb-Bi tungsten can scram by gravity. Table 3.3 summarizes the fuel, absorber and structural material composition assumed for the reference design.

All the absorbing elements are securely latched in place during shipping and reactor installation. The central absorber has an electromagnetic latch that does not engage until the startup temperature of 350°C is achieved. At this temperature the assembly can be withdrawn. Normal operational shutdowns can be accomplished with the peripheral absorbers. The reactor is made critical by a hydraulic system that moves the peripheral absorbers up at 1 mm/sec to compensate for the negative temperature coefficient of reactivity. At the full power position, the peripheral absorbing segments are stopped from further upward movement by mechanical stoppers whose movement is established by high-reliability gear drives. These drives restrict the rate of movement of the reflector segments to approximately 1 mm/day. It is anticipated that the height

of the peripheral absorbers will be adjusted once a year or two to compensate for a slight drift in reactivity due to burnup.

Table 3.3 Fuel, Absorbers and Structure Composition

Component	Material	Composition
Fuel	Pu-U alloy with 10 ^{w/o} Zr	75% of nominal density of 15.85 g/cm ³
Pu	PWR 50 GWD/tHM spent fuel after 10 years of cooling	3.18 ^{w/o} ²³⁸ Pu, 56.35 ^{w/o} ²³⁹ Pu, 26.6 ^{w/o} ²⁴⁰ Pu, 8.02 ^{w/o} ²⁴¹ Pu, 5.83 ^{w/o} ²⁴² Pu
Structures	HT-9	85 ^{w/o} Fe, 11.5 ^{w/o} Cr, 0.5 ^{w/o} Ni, 1.5 ^{w/o} Mo, 0.9 ^{w/o} Mn, 0.6 ^{w/o} Si
Absorbers	Tungsten: 40 ^{v/o} B ₄ C: 40 ^{v/o} HT-9: 20 ^{v/o}	75% nominal density 75% nominal density; boron is enriched to 92% ¹⁰ B

3.3.2 Core Design Approach

The fuel rod diameter, clad thickness and fission gas plenum length were determined from fuel integrity combined with coolant friction loss considerations^{3,4, 3.5}. Figure 3.2 gives the fuel integrity analysis results^{3,4} – with the specified reference fuel rod design it is possible to reach the target peak discharge burnup provided the peak clad temperature does not exceed 590°C. The peak discharge burnup, nearly 110 GWD/tHM, is imposed by fast neutron induced radiation damage to the HT-9 clad. The maximum permissible fluence of neutrons of 0.1 MeV or higher energy is assumed to be 4x10²³ n/(cm²-s).

Coolant friction loss considerations, imposed by the requirement of achieving natural circulation with total module height that does not exceed ~20 m, forced the reference design to have relatively large diameter fuel rods. For a given pitch-to-diameter ratio the larger is the fuel rod diameter the larger the coolant flow channel cross section area and the hydraulic diameter become, and the smaller the pressure drop through the core becomes.

The core design is obtained by searching for the optimal combination of lattice pitch-to-diameter ratio and initial plutonium weight percent giving a BOL k_{eff} of $1+\beta = 1.0042$ and a minimum burnup reactivity swing over the cycle length. The BOL k_{eff} was taken to be β above unity to accommodate uncertainties in the data, computational methods and design.

The core neutronic design was done with the DIF3D^{3.6} and the REBUS-3^{3.7} codes using R-Z geometry and volume homogenized core composition. The DIF3D code is used to obtain the multi-group neutron flux by solving the neutron diffusion equation using the finite difference

Figure 3.2 Peak burnup dependence on peak clad temperature for the reference ENHS fuel rod

discretization method. The REBUS-3 code solves the transmutation equations using the neutron flux obtained from DIF3D. For the burnup analysis, the core is divided into nine equal volume zones: three radial zones and three axial zones. In each zone, the nuclide number densities are assumed constant. The calculations are done using 80 group cross section library derived from ENDF/B-VI. Lumped cross sections, generated using data for 172 fission products isotopes, are used to account for the effect of fission products. There is one lumped fission product for each one of 17 fissionable isotopes. The decay chain considered for the depletion analysis spans the range from ^{234}U to ^{246}Cm . Details about this computational methodology can be found in References 3.8 and 3.9 (Chapter 15).

The DIF3D/REBUS computational methodology was thoroughly benchmarked against the MOCUP code system^{3.10} that uses the Monte Carlo code MCNP^{3.11} for neutron transport calculations and ORIGEN2.2^{3.12} for fuel burnup analysis. Figure 3.3 compares the prediction of the two code systems of k_{eff} evolution with burnup for the benchmark problem. A very good agreement is found between the MOCUP results and the DIF3D/REBUS results of the Koreans. However, the DIF3D/REBUS k_{eff} calculated at ANL is lower throughout the core life. ANL later discovered a mistake in the multi-group cross sections they processed from the ENDF/B-VI files but did not have a chance to repeat the ENHS benchmark analysis with a revised set of cross sections.

3.3.3 Reference Core Neutronic Characteristics

Table 3.4 summarizes beginning-of-life (BOL) and end-of-life (EOL) neutronic characteristics of the reference ENHS core. The results are calculated for Pb but are practically the same for LBE. Figure 3.4 shows the evolution of k_{eff} of the reference core over the core life, when the peripheral absorbers are all out, for three pitch-to-diameter values. The BOL k_{eff} is set to be the same for all

P/D by adjusting the Pu weight % in the fuel. The minimal burnup reactivity swing was obtained for P/D of 1.36 and LWR Pu weight % of 12.20; these value was selected for the reference core.

Figure 3.3 k_{eff} evolution with burnup as predicted for the ENHS benchmark problem using the DIF3D/REBUS code system and the MCNP/ORIGEN2.2 code system

Figure 3.4 Evolution of k_{eff} of the reference core with operating time.

Table 3.4 Reference ENHS Core Physics Characteristics

Performance Parameter	BOL	EOL
Peak-to-average power density	1.829	1.863
Peak linear heat rate (W/cm)	184.1	187.6
Peak-to-average channel power	1.500	1.529
Peak-to-average power density in hot channel	1.220	1.224
Peak burnup after 20 (20.9 [*]) EFPY (GWD/tHM)		99.89 (104.4 [*])
Average burnup after 20 (20.9 [*]) EFPY (GWD/tHM)		50.8 (53.1 [*])
Peak fast (E>0.1MeV) neutron flux (n/cm ² -s)	8.86E+14	8.94E+14
Peak fast (E>0.1 MeV) fluence at 20 EFPY (n/cm ²)		3.829E+23
Conversion ratio	1.048	1.045
Effective delayed neutron fraction	0.0035	0.0035
Maximum Δk_{eff} with burnup (%)	0.287	
Doppler effect (dk/kk' -°C)	-5.2442E-06	-4.2202E-06
Axial fuel expansion (dk/kk' -°C)	-4.6379E-06	-4.6057E-06
Coolant expansion (dk/kk' -°C)	+1.0917E-07	+5.6866E-07
Grid-plate radial expansion (dk/kk' -°C)	-8.0679E-06	-6.4097E-06
Cold (350°C) to hot (480°C; fuel: 700°C) dk/kk'		
Doppler effect	-1.9600E-03	-1.5399E-03
Axial fuel expansion	-6.1421E-04	-6.0921E-04
Coolant expansion	+2.6400E-05	+8.6558E-05
Grid-plate expansion	-1.0516E-03	-8.3430E-04
Total	-3.5994E-03	-2.8968E-03
Void reactivity effect (dk,%)		
Voiding inner 1/3 core (+upper gas plenum)	+2.718 (+1.516)	+2.809 (+1.618)
Voiding middle 1/3 core (+upper gas plenum)	+0.689 (-0.068)	+0.659(-0.043)
Voiding outer 1/3 core (+upper gas plenum)	-0.694 (-1.045)	-0.668 (-1.009)
Voiding whole core (+upper gas plenum)	+2.555 (+0.424)	+2.644 (+0.572)
Coolant loss reactivity effect (dk/kk', %)		
Coolant level drops to bottom of fission gas plenum	-1.869	
Coolant level drops to 1/3 below core top	-6.261	
Coolant level drops to 2/3 below core top	-12.219	
Coolant level drops to core bottom	-14.825	
Peripheral absorber reactivity worth (dk,%)	1.990	1.817
Central absorber reactivity worth (dk,%)	4.138	4.104
Peripheral + central absorber reactivity worth (dk,%)	6.811	6.562
Total heavy metal inventory (kg)	1.7505E+04	1.6564E+04
Fissile-to-total plutonium ratio	0.6437	0.6649

^{*} Corresponding to radiation damage to clad of 4×10^{23} fast (E>0.1 MeV) neutrons per cm².

Table 3.4 shows that the temperature coefficients of reactivity are all negative except for that of the coolant expansion. However, the negative reactivity coefficient associated with the radial expansion of the core structure can compensate for the coolant thermal expansion effect. This has been proven in safety analysis carried out for the ENHS reactor^{3.13-3.16}. The void coefficients

for 100% voided inner and middle cores are positive. However, creation of voiding in the ENHS core is practically impossible since the boiling temperature of Pb-Bi is 1943K; that is 1000K higher than the operating temperature and even higher than the melting temperature of the metallic fuel and clad. Loss-of-coolant accidents, although most unlikely to happen due to the insertion of the ENHS module in a pool of Pb-Bi and the enclosure of the pool in an underground silo, will have a negative reactivity effect. It is worth noting that in a recent study^{3.17} we found that using nitride rather than metallic fuel, the temperature coefficient of the coolant and the void coefficient of reactivity become significantly less positive.

Figure 3.5 shows the average neutron spectrum in the reference core while Table 3.5 give the same data in tabular form. Table 3.6 gives the neutron balance in the reference reactor core at BOL.

Figure 3.5 Average core neutron spectrum of the reference reactor

Additional information on the ENHS core physics characteristics can be found in References 3.9, and 3.17 through 3.21.

Table 3.5 Average Core Neutron Spectrum of the Reference Reactor

Energy group	Upper energy (eV)	Flux (n/cm ² sec)	Energy group	Upper energy (eV)	Flux (n/cm ² sec)
1	2.00000E+07	1.74558E+08	41	1.50340E+04	3.75935E+12
2	1.69050E+07	6.87903E+08	42	1.32680E+04	3.77119E+12
3	1.49180E+07	1.78031E+09	43	1.17090E+04	3.05015E+12
4	1.34990E+07	6.62789E+09	44	1.03330E+04	2.22162E+12
5	1.19130E+07	3.09727E+10	45	9.11880E+03	1.28379E+12
6	1.00000E+07	1.67490E+11	46	8.04730E+03	1.14227E+12
7	7.78800E+06	5.11180E+11	47	7.10170E+03	1.49820E+12
8	6.06530E+06	1.09876E+12	48	6.26730E+03	1.38845E+12
9	4.72370E+06	2.00016E+12	49	5.53080E+03	1.29896E+12
10	3.67880E+06	3.45489E+12	50	4.88100E+03	1.10208E+12
11	2.86500E+06	5.47594E+12	51	4.30740E+03	7.79988E+11
12	2.23130E+06	7.12575E+12	52	3.80130E+03	8.98521E+11
13	1.73770E+06	9.16632E+12	53	3.35460E+03	7.95488E+11
14	1.35330E+06	5.74429E+12	54	2.96040E+03	5.45996E+11
15	1.19430E+06	6.67028E+12	55	2.61260E+03	4.76602E+11
16	1.05400E+06	8.15347E+12	56	2.30560E+03	4.38459E+11
17	9.30140E+05	8.89292E+12	57	2.03470E+03	4.40758E+11
18	8.20850E+05	9.80068E+12	58	1.79560E+03	3.67267E+11
19	7.24400E+05	1.35602E+13	59	1.58460E+03	3.26046E+11
20	6.39280E+05	1.57846E+13	60	1.39840E+03	2.82436E+11
21	5.64160E+05	1.72758E+13	61	1.23410E+03	1.90160E+11
22	4.97870E+05	1.71397E+13	62	1.08910E+03	1.63804E+11
23	4.39370E+05	1.53730E+13	63	9.61120E+02	2.62710E+11
24	3.87740E+05	4.11416E+13	64	7.48520E+02	2.03027E+11
25	3.01970E+05	3.64795E+13	65	5.82950E+02	1.38497E+11
26	2.35180E+05	3.01886E+13	66	4.54000E+02	9.68437E+10
27	1.83150E+05	2.79032E+13	67	3.53580E+02	5.61237E+10
28	1.42640E+05	3.38187E+13	68	2.75360E+02	1.04803E+11
29	1.11090E+05	2.25841E+13	69	1.67020E+02	6.75311E+10
30	8.65170E+04	2.84540E+13	70	1.01300E+02	3.92745E+10
31	6.73790E+04	2.03585E+13	71	6.14420E+01	2.52193E+10
32	5.24750E+04	1.61951E+13	72	3.72670E+01	2.35120E+10
33	4.08680E+04	1.39095E+13	73	2.26030E+01	8.82181E+09
34	3.18280E+04	2.80389E+12	74	1.37100E+01	9.13563E+09
35	2.80880E+04	2.80068E+12	75	8.31530E+00	3.65906E+09
36	2.60580E+04	4.62162E+12	76	5.04350E+00	7.84292E+09
37	2.47880E+04	9.83810E+12	77	3.05910E+00	6.77504E+09
38	2.18750E+04	7.32911E+12	78	1.12540E+00	2.94373E+08
39	1.93050E+04	5.28268E+12	79	4.14000E-01	9.76341E+06
40	1.70360E+04	5.45294E+12	80	1.52300E-01	2.24601E+06
				1.38880E-04	

Table 3.6 Neutron Balance in the Reference Core

Characteristic	Value
k_{eff}	1.00475
Leakage probability	0.271
Absorption probability	
in HM	0.661
in Pb	0.020
in Zr	0.016
in HT-9	0.032
Total	0.729
Fission probability	0.344
ν	2.921
k_{∞}	1.378
ICR	1.0445

3.3.4 Reference Core Power Distribution

The power distribution across the reference core is shown in Figs. 3.6 thru 3.14 and in Table 3.7.

Figure 3.6 Normalized BOL and EOL axial power distribution in the hot channel of the reference core. Numerical data is given in Table 3.7

Figure 3.7 Axial power distribution at selected radial locations in the reference core at BOL

Figure 3.8 Relative axial power distribution at selected radial locations in the reference core at BOL

Figure 3.9 Axially averaged radial power distribution in the reference core at BOL

Figure 3.10 Normalized axially averaged radial power distribution in the reference core at BOL

Figure 3.11 Axial power distribution at selected radial locations in the reference core at EOL

Figure 3.12 Relative axial power distribution at selected radial locations in the reference core at EOL

Figure 3.13 Axially averaged radial power distribution in the reference core at EOL

Figure 3.14 Normalized axially averaged radial power distribution in the reference core at EOL

Table 3.7 Normalized Axial Power Density in Hot Channel of the Reference Core; Plotted in Figure 3.6

Distance from core top (cm)	Normalized power	
	BOL	EOL
9.92063E-01	6.644E-01	6.360E-01
2.97619E+00	6.705E-01	6.476E-01
4.96032E+00	6.877E-01	6.696E-01
6.94444E+00	7.107E-01	6.966E-01
8.92857E+00	7.368E-01	7.261E-01
1.09127E+01	7.645E-01	7.567E-01
1.28968E+01	7.929E-01	7.877E-01
1.48810E+01	8.216E-01	8.186E-01
1.68651E+01	8.501E-01	8.491E-01
1.88492E+01	8.783E-01	8.791E-01
2.08333E+01	9.058E-01	9.082E-01
2.28175E+01	9.327E-01	9.365E-01
2.48016E+01	9.588E-01	9.639E-01
2.67857E+01	9.839E-01	9.902E-01
2.87698E+01	1.008E+00	1.015E+00
3.07540E+01	1.031E+00	1.039E+00
3.27381E+01	1.053E+00	1.062E+00
3.47222E+01	1.074E+00	1.083E+00
3.67063E+01	1.093E+00	1.104E+00
3.86905E+01	1.112E+00	1.122E+00
4.06746E+01	1.129E+00	1.139E+00
4.26587E+01	1.144E+00	1.147E+00
4.46429E+01	1.158E+00	1.161E+00
4.66270E+01	1.171E+00	1.174E+00
4.86111E+01	1.182E+00	1.186E+00
5.05952E+01	1.192E+00	1.196E+00
5.25794E+01	1.201E+00	1.204E+00
5.45635E+01	1.208E+00	1.211E+00
5.65476E+01	1.213E+00	1.217E+00
5.85317E+01	1.217E+00	1.220E+00
6.05159E+01	1.219E+00	1.223E+00
6.25000E+01	1.220E+00	1.224E+00

Distance from	Normalized power	
	BOL	EOL
6.44841E+01	1.220E+00	1.223E+00
6.64683E+01	1.217E+00	1.221E+00
6.84524E+01	1.214E+00	1.217E+00
7.04365E+01	1.208E+00	1.212E+00
7.24206E+01	1.202E+00	1.205E+00
7.44048E+01	1.194E+00	1.197E+00
7.63889E+01	1.184E+00	1.187E+00
7.83730E+01	1.173E+00	1.176E+00
8.03571E+01	1.160E+00	1.163E+00
8.23413E+01	1.146E+00	1.149E+00
8.43254E+01	1.131E+00	1.143E+00
8.63095E+01	1.114E+00	1.126E+00
8.82936E+01	1.096E+00	1.108E+00
9.02778E+01	1.077E+00	1.088E+00
9.22619E+01	1.057E+00	1.067E+00
9.42460E+01	1.035E+00	1.045E+00
9.62302E+01	1.012E+00	1.021E+00
9.82143E+01	9.888E-01	9.960E-01
1.00198E+02	9.642E-01	9.700E-01
1.02183E+02	9.387E-01	9.431E-01
1.04167E+02	9.125E-01	9.152E-01
1.06151E+02	8.858E-01	8.866E-01
1.08135E+02	8.587E-01	8.574E-01
1.10119E+02	8.314E-01	8.277E-01
1.12103E+02	8.043E-01	7.979E-01
1.14087E+02	7.779E-01	7.684E-01
1.16071E+02	7.529E-01	7.398E-01
1.18056E+02	7.305E-01	7.132E-01
1.20040E+02	7.130E-01	6.905E-01
1.22024E+02	7.041E-01	6.755E-01
1.24008E+02	7.115E-01	6.758E-01

It is observed that the power shape stays nearly constant throughout the 20+ years of operation. In fact small, variations are expected due to the adjustment of the axial location of the peripheral absorber that are required to compensate for the small burnup reactivity swing.

3.3.5 Reference Core Composition Evolution

Figures 3.15 shows the evolution of the plutonium isotopes in the reference core while Table 3.8 gives the inventory of the fuel isotopes at BOL and at EOL. Additional information can be found in References 3.9, 3.17, 3.18, 3.19 and 3.21.

Figure 3.15 Evolution of plutonium isotopes in the reference core

Table 3.8 BOL Inventory of Fuel Isotopes and Their Change Over Core Life

Isotope	Inventory (kg)	
	BOL	EOL mass increase
^{234}U	0.0	7.42
^{235}U	30.74	-12.98
^{236}U	0.0	2.53
^{238}U	15339	-1066.3
^{238}Pu	67.99	-22.6
^{237}Np	0.0	3.08
^{239}Pu	1203	177.9
^{240}Pu	568.4	-2.58
^{241}Pu	171.3	-109.0
^{242}Pu	124.5	-8.28
^{241}Am	0.0	75.8
$^{242\text{m}}\text{Am}$	0.0	2.4
^{243}Am	0.0	9.33
^{242}Cm	0.0	0.557
^{243}Cm	0.0	0.013
^{244}Cm	0.0	1.23
^{245}Cm	0.0	0.0845
^{246}Cm	0.0	0.002

3.3.6 Reference Core Shielding

The original design of the ENHS module had no special shield. As the module life is dictated by the core life, radiation damage to the Module structural walls that are subjected to lower fluxes is of no design concern. In particular, it does not impose any shielding requirements. However, it has later become apparent that without incorporation of a special shield the radiation levels outside of the module may be too high to handle. An additional concern is an excessive activation of the secondary coolant that is in contact with the module outer structural wall.

The design objective for the dose rate set for the ENHS shield is less than 100 mrem/hr 1-meter from the ENHS surface. It is expected that this dose-rate will enable the handling of the spent module to comply with the 10CFR20.1201 criteria for occupational dose as well as with 10CFR72.104 criteria for radiation from a storage facility.

The location of the radial and axial shields we came up with^{3.19} is defined in Figure 3.1 (zones 17 and 18). The optimal arrangement found^{3.19} for the radial shield is shown, schematically, in Figure 3.16. The shield consists of HT-9 tubes filled with $\text{ZrH}_{1.6}$ and $^{10}\text{B}_4\text{C}$ at 75% of their nominal density. The tubes are arranged in a very tight array with Pb-Bi in between. The shield rods are made of 24 volume % $^{10}\text{B}_4\text{C}$, 56% $\text{ZrH}_{1.6}$ and 20% HT-9. The boron is enriched to 92% ^{10}B .

Figure 3.16 Schematic description of the radial shield layout.

Figure 3.17 gives the total primary neutron source strength in the ENHS core following shutdown at EOL. These neutrons originate primarily from spontaneous fissions of the fuel constituents. Figure 3.18 gives the total primary photon source intensity in the ENHS core following shutdown at EOL. The primary photon source accounted for is from the decay of the fission products accumulated in the core throughout its life and from the prompt gamma rays emitted from the primary neutrons induced reactions.

Figure 3.17 Primary neutron source strength following EOL shutdown of the ENHS

Figure 3.18 Primary photon source strength following EOL shutdown of the ENHS

Tables 3.9 and 3.10 give, respectively, the primary neutrons and primary photons spectrum for selected decay time. Table 3.11 gives the average neutron flux spectrum, in a coarse group structure, in selected components of the ENHS module during reactor operation. The neutron flux is used for calculating the activation of different ENHS constituents based on which the secondary photon source is derived. Tables 3.12 and 3.13 give the resulting secondary photon source from activation of the HT-9 structural material whereas Tables 3.14 and 3.15 give the secondary photon source from activation of the Pb-Bi primary coolant in the downcomer and of the Pb-Bi secondary coolant in a 10 cm thick annulus surrounding the outer structural wall of the ENHS module.

Table 3.9 Primary Neutron Source Spectrum Following Reactor Shutdown at EOL

Energy group boundaries (MeV)	Spontaneous fission and (α , n) neutron fraction as a function of decay time (days)			
	1 day	30	120	365
6.43 - 20.0	0.176	0.176	0.178	0.181
3.00 - 6.43	0.226	0.225	0.223	0.217
1.85 - 3.0	0.247	0.246	0.243	0.239
1.40 - 1.85	0.125	0.125	0.126	0.128
0.90 - 1.40	0.166	0.167	0.169	0.172
0.40 - 0.90	0.182	0.183	0.185	0.188
0.10 - 0.40	0.036	0.036	0.036	0.037

Table 3.10 Primary Photon Source Spectrum Following Reactor Shutdown at EOL

Lower group energy (MeV)	ENHS cooling time			
	1 day	30 days	120 days	365 days
1.00E-02	0.1970	0.1051	0.1271	0.1394
2.00E-02	0.0435	0.0621	0.0697	0.0764
3.00E-02	0.0577	0.0774	0.0808	0.0922
4.50E-02	0.0277	0.0461	0.0600	0.0702
7.00E-02	0.0513	0.0380	0.0449	0.0515
1.00E-01	0.1856	0.0782	0.0631	0.0586
1.50E-01	0.1198	0.0324	0.0396	0.0443
3.00E-01	0.0417	0.0320	0.0216	0.0242
4.50E-01	0.1267	0.2375	0.2554	0.3452
7.00E-01	0.1090	0.2506	0.2234	0.0813
1.00E+00	0.0080	0.0082	0.0109	0.0136
1.50E+00	0.0299	0.0299	0.0021	0.0017
2.00E+00	0.0009	0.0014	0.0014	0.0014
2.50E+00	0.0011	0.0011	5.02E-05	4.05E-05
3.00E+00	9.74E-06	1.16E-05	4.43E-06	5.03E-06
4.00E+00	2.20E-10	8.57E-10	1.39E-09	1.93E-09
6.00E+00	2.52E-11	9.86E-11	1.60E-10	2.22E-10
8.00E+00	2.90E-12	1.13E-11	1.84E-11	2.56E-11

Table 3.11 Three Group Neutron Flux During Full Power Operation In Selected Regions

Region	Neutron flux (n/cm ² -s)			
	E < 0.5 eV	0.5 eV < E < 1 MeV	E > 1 MeV	Total
Inner wall	4.24E+10	1.79E+14	3.88E+12	1.83E+14
Downcomer	6.30E+10	4.07E+13	2.92E+11	4.11E+13
Outer wall	5.01E+10	4.77E+13	1.22E+11	4.78E+13
Basis	1.65E+09	1.15E+12		1.15E+12
Secondary Coolant	1.63E+10	9.24E+12	1.48E+10	9.28E+12
Core	6.03E+07	4.22E+14	4.28E+13	4.65E+14

Table 3.12 Secondary Photon Source from Activation of HT-9 Components One Day After Shutdown at EOL

Lower energy (MeV)	Disintegration rate (photons/s)		
	Inner structural wall	Outer structural wall	Bottom base plate
1.00E-02	1.21E+10	1.84E+10	2.01E+09
2.00E-02	5.28E+09	8.03E+09	8.76E+08
3.00E-02	4.68E+09	7.12E+09	7.77E+08
4.50E-02	3.93E+09	5.98E+09	6.52E+08
7.00E-02	2.59E+09	3.94E+09	4.30E+08
1.00E-01	6.16E+09	9.36E+09	1.02E+09
1.50E-01	5.62E+09	8.55E+09	9.33E+08
3.00E-01	1.44E+10	2.19E+10	2.39E+09
4.50E-01	2.79E+10	4.24E+10	4.63E+09
7.00E-01	9.30E+11	1.41E+12	1.54E+11
1.00E+00	5.89E+08	8.95E+08	9.77E+07
1.50E+00	7.39E+08	1.12E+09	1.23E+08
2.00E+00	1.08E+08	1.64E+08	1.79E+07
2.50E+00	1.51E+07	2.30E+07	2.51E+06
3.00E+00	1.30E+06	1.97E+06	2.15E+05
4.00E+00	7.64E-09	1.30E-08	4.88E-11
6.00E+00	0.00E+00	0.00E+00	0.00E+00
8.00E+00	0.00E+00	0.00E+00	0.00E+00
1.10E+01			
Total	1.01E+12	1.54E+12	1.68E+11

Table 3.13 Total Secondary Photon Source From Activation of HT-9 Components at Selected Times After Shutdown at EOL

Decay time (days)	Disintegration rate (Photons/sec)		
	Inner wall	Outer wall	Bottom base
1	1.01E+12	1.54E+12	1.68E+11
10	9.43E+11	1.43E+12	1.56E+11
30	8.77E+11	1.33E+12	1.46E+11
60	7.98E+11	1.21E+12	1.32E+11
120	6.73E+11	1.02E+12	1.12E+11
365	3.70E+11	5.62E+11	6.13E+10

Table 3.14 Secondary Photon Source from Activation of Pb-Bi One Day After Shutdown at EOL

Lower energy (MeV)	Disintegration rate (Photons/s)	
	Downcomer	Pool
1.00E-02	2.57E+12	4.11E+11
2.00E-02	1.38E+12	2.20E+11
3.00E-02	1.06E+12	1.69E+11
4.50E-02	1.11E+12	1.77E+11
7.00E-02	7.27E+11	1.16E+11
1.00E-01	5.57E+11	8.91E+10
1.50E-01	5.27E+11	8.43E+10
3.00E-01	1.60E+11	2.56E+10
4.50E-01	3.77E+10	6.02E+09
7.00E-01	5.03E+09	8.04E+08
1.00E+00	4.95E+07	7.90E+06
1.50E+00	3.49E+04	5.58E+03
2.00E+00	1.75E+04	2.80E+03
2.50E+00	8.78E+03	1.40E+03
3.00E+00	6.46E+03	1.03E+03
4.00E+00	1.92E+03	3.07E+02
6.00E+00	1.25E+02	1.99E+01
8.00E+00	7.87E+00	1.26E+00
1.10E+01		
Total	8.14E+12	1.30E+12

Table 3.15 Total Secondary Photon Source from Activation of Pb-Bi at Selected Times after Shutdown at EOL

Decay time (days)	Disintegration rate (γ/sec)	
	Primary	Secondary
1	8.14E+12	1.30E+12
10	2.35E+12	3.76E+11
30	1.58E+11	2.53E+10
60	7.35E+09	1.17E+09
120	1.03E+09	1.64E+08
365	8.78E+08	1.40E+08

Table 3.9 gives the dose-rate at two locations outside the ENHS module one day after shutdown after reaching the clad radiation damage limit; i.e., after more than 20 EFPY. The dose locations are located one meter from the outer structural wall of the ENHS module at the mid-core elevation, and one meter from the base of the ENHS module, at the module central line. The results pertain to a shield that is 22.5 cm in thickness. It is seen that the dominant contribution to dose rate comes from the neutrons. The neutrons that contribute to the dose rate are the primary neutron source defined above, augmented by subcritical chain reaction multiplication. This multiplication was assumed to be 20, corresponding to k_{eff} value at shutdown of 0.95. It should be possible to design the control system to provide a lower shutdown k_{eff} value and thus reduce the shutdown dose rate even further, if necessary. Additional information on the ENHS shield design can be found in Reference 3.22.

Table 3.9 Dose-Rate Outside the Reference Shielded ENHS Module One Day after Shutdown

Dose Rate Source	Dose-rate (mrem/hr)	
	1 m from outer wall at mid-core elevation	1 m below the bottom base at Module axis
Neutrons	82.7	0.7
Fission products γ 's	< 0.001	< 0.001
Other primary γ 's	0.035	0.008
HT-9 activation	0.01	0.05
^{59}Co activation	< 0.001	< 0.001
Total	82.74	0.75

3.4 Thermal-Hydraulic Characteristics

A description of the assumptions and models used for the thermal-hydraulic analysis of the ENHS reactor can be found in references 3.9, 3.19, 3.23, 3.24 and 3.25. Table 3.10 summarizes the thermal hydraulic characteristics of the reference ENHS reactor. Also summarized in Table 3.10 are the dimensions and relative location of the reactor components that were considered in the analysis.

Table 3.10 Thermal Hydraulic Characteristics of the Reference ENHS Reactor Using Either Pb or Pb-Bi (LBE) Coolants

Case Designation		Reference
Primary Coolant		LBE/Lead
Secondary Coolant		LBE/Lead
Roughness Height, μm		10
Power, MW_{th}		125
Heated Core Height, m		1.25
Fission Gas Plenum Height, m		1.25
Total Core Height, m		2.5
Fuel Rod Diameter, cm		1.56
Fuel Rod Pitch, cm		2.106 ^(b)
Pitch-to-Diameter Ratio		1.35 ^(b)
Cladding Thickness, cm		0.13
Fuel Material	U-11 wt% Pu ^(b) -10 wt% Zr	
Fuel Diameter, cm		1.126
Cladding-Fuel Bond Material		Sodium
Cladding-Fuel Bond Gap Thickness, cm		0.0871
Fuel Smeared Density, %		75
Number of Fuel Rod Spacer Grids		3
Core Radius, m		1.10 ^(b)
Core Peak-to-Average Power		1.77 ^(b)
Hot Channel Factor		1.44
Power Factor at Top of Hot Channel		1.06
Riser Height, m		13
HX Channel Height, m		11
Primary Coolant Thermal Centers Separation, m		8.125
HX Channel Width, cm		2.5
HX Channel Thickness, cm		39.2
Number of HX Channels		135
Number of SG Modules		8
SG Height, m		4.8
SG Tube Outer Diameter, cm		2.0
Number of SG Tubes per SG Module		613
SG Coolant Inlet to HX Outlet Height Difference, m		1.15
Intermediate Coolant Thermal Centers Separation, m		4.25
Core Hydraulic Diameter, cm		1.57
Primary Coolant HX Hydraulic Diameter, cm		5.39
Intermediate Coolant HX Hydraulic Diameter, cm		4.70

Case Designation	Reference
Intermediate Coolant SG Hydraulic Diameter, cm	2.50
Core Flow Area, m ²	1.83
Primary Coolant HX Flow Area, m ²	1.86
Intermediate Coolant HX Flow Area, m ²	1.32
Intermediate Coolant SG Flow Area, m ²	2.04
Primary Coolant Flowrate, Kg/s	8320/8220
Intermediate Coolant Flowrate, Kg/s	5970/5870
Coolant Velocity in Core, m/s	0.447/0.426
Primary Coolant Velocity in HX, m/s	0.438/0.418
Intermediate Coolant Velocity in HX, m/s	0.440/0.418
Intermediate Coolant Velocity in SG, m/s	0.285/0.271
Reynolds Number in Core	47400/30800
Primary Coolant Reynolds Number in HX	159000/103000
Intermediate Coolant Reynolds Number in HX	128000/80700
Intermediate Coolant Reynolds Number in SG	44000/27800
Primary Coolant Temperature Rise, C	103/104
Intermediate Coolant Temperature Rise, C	143/144
Primary Coolant Outlet Temperature, C	453/454
Primary Coolant Inlet Temperature, C	350
Intermediate Coolant Outlet Temperature, C	421/427
Intermediate Coolant Inlet Temperature, C	278/283 ^(a)
Hot Channel Primary Coolant Temperature Rise, C	147/149
Hot Channel Outlet Temperature, C	497/499
Hot Channel Peak Cladding Outer Surface Temp, C	518/517
Hot Channel Peak Cladding Inner Surface Temp, C	529/529
Hot Channel Peak Sodium Bond Temperature, C	533/532
Hot Channel Peak Fuel Temperature, C	572/571

^(a) Incompatible with Pb melting temperature of 327°C. ^(b) These values are little bit smaller than in the reference design described in Section 3.3; they correspond to LMR Pu, no shield, and SS structural material.

The physical data used for the analysis is that recommended in Sec. 3 of Ref. 3.19 that provides a comprehensive compilation and evaluation of experimental data published over the years. This data compilation and evaluation is reproduced in Appendix A as it could be useful for any future work on lead and on lead-bismuth cooled reactors. The physical properties used for the analysis of the two reactors addressed in Table 3.10 are summarized in Table 3.11. The values reported in Table 3.11 correspond to the average temperature of the coolant, clad and fuel.

It is found that the operating temperatures depend only very slightly on whether lead or lead-bismuth is used for the coolants. The temperature rises through the hot channel is less than the upper limit criterion of 150 °C. The peak cladding temperatures in the hot channel remain well below a constraint of 650 °C. This constraint is based upon the existing database for ferritic-

martensitic cladding/structural material in contact with LBE as well as eutectic interaction of the cladding with the U-Pu-Zr metallic ternary fuel. Because of the margin between the 650 °C constraint and the calculated peak cladding temperature, the primary coolant inlet temperature could be raised to 400 °C. All the temperature values reported in Table 3.10 will correspondingly be raised by 50 °C without violating the design constraints. The peak fuel temperatures remain well below a constraint of 1040 °C based upon the solidus temperature of the metallic fuel alloy.

Table 3.11 Physical Properties Used for the Thermal-Hydraulic Analysis of Tables 3.10 and 4.11

Case Designation	Reference	With Lift Pump
Primary Coolant Properties		
Density, Kg/m ³	10200/10600	10200/10600
Compressibility, K ⁻¹	1.35E-4/1.28E-4	1.35E-4/1.28E-4
Thermal Conductivity, W/(m•K)	12.8/16.1	12.6/16.2
Viscosity, Kg/(m•s)	1.51E-3/2.30E-3	1.55E-3/2.40E-3
Specific Heat, KJ/(Kg•K)	1.465/1.467	1.465/1.470
Intermediate Coolant Properties		
Density, Kg/m ³	10300/10600	10300/10600
Compressibility, K ⁻¹	1.35E-4/1.27E-4	1.34E-4/1.27E-4
Thermal Conductivity, W/(m•K)	12.0/16.4	11.8/16.5
Viscosity, Kg/(m•s)	1.66E-3/2.58E-3	1.70E-3/2.67E-3
Specific Heat, KJ/(Kg•K)	1.465/1.474	1.465/1.476
Cladding Thermal Conductivity, W/(m•K)	29/29	29/29
Bond Thermal Conductivity at Core Top, W/(m•K)	69.3/69.3	70.8/71.0
Fuel Thermal Conductivity at Top of Core, W/(m•K)	22.2/22.1	21.3/21.2

Stability analysis of the natural circulation flow was done for the reference ENHS module at nominal power level and at 2% of the nominal power; the latter represents decay heat removal power levels. At both power levels the flow was found to be stable.

Three-dimensional computational fluid dynamic (CFD) analysis was done of the natural circulation coolant flow in the ENHS. Two commercially available CFD codes were used for this analysis: Star-CD and FIDAP. This 3-D analysis confirmed that natural circulation could be established in the ENHS, as predicted by the one-dimensional model used for the thermal hydraulic analysis reported above. Details of the 3-D studies can be found in Section 5 of Reference 3.9 and subsequent internal memos.

3.5 ENHS Safety

Accident analysis^{3.9, 3.13-3.15, 3.19} revealed that the deviation of the system from nominal operating conditions is very slow and that all plant components do not reach their design limit temperatures. All the postulated events were mitigated by naturally occurring phenomena – negative reactivity feedback and natural circulation. Four accidents were studied using the DSNP transient simulation model^{3.13,3.14}: (1) A startup accident, (2) A reactivity accident, (3) A loss of

heat sink (LOHS) without scram accident and (4) a stem line break without scram accident. The LOHS was found to be the most severe accident. The results reported in this section pertain to an earlier variant of the ENHS reference design but the conclusions apply to the reference design.

Figure 3.19 illustrates the consequence of a complete LOHS accident on the fuel, clad and coolant temperatures. The top 3 plots pertain to the peak fuel and clad temperatures in the hot channel whereas the 3 plots at the bottom pertain to the fuel, clad and coolant temperatures at mid height location of a typical average core channel. Initially the average core temperature gradually increases – a process that is very slow in the ENHS, due to its very large heat capacity. Due to the overall negative temperature feedback of the ENHS core the reactivity drops, as seen in Figure 3.20, and as a consequence the power starts to drop, as seen in Figure 3.21. As a result of the power drop the core average temperatures start to drop as well (Figure 3.19). At about 1500 seconds into the accident there is a re-criticality (Figure 3.20) as a result of which there is a small increase in the power level (Figure 3.21). It takes few additional oscillations of lesser and lesser amplitude for the system to stabilize.

The important finding of this analysis is that throughout this accident the maximum fuel, clad and coolant temperatures stay far below their constraints. In fact, the peak fuel temperature practically does not increase. The negative reactivity feedback is provided by an increase in the average core temperatures.

Figure 3.19 Temperature evolution in selected core locations following a LOHS accident

The robustness of the ENHS to accidents is due to the following unique features: (1) Full natural circulation in the primary and intermediate coolant loops. (2) Very large overall heat capacity. (3) No active mechanical elements in the primary and intermediate loops need to operate or to be actuated in order to maintain plant safety. (4) Negative power feedback of reactivity. (5) More than 1100°C difference between the highest coolant operating temperature and the boiling temperature. This enables to accommodate the positive void coefficient (See Table II). No scenario for voiding the core has been identified. The coolant boiling temperature is even higher

than the melting temperature of the metallic fuel and clad. Even if LOCA could occur in the ENHS, which is most unlikely (See “10” below), its reactivity effect would be negative due to

Figure 3.20 Reactivity evolution in the LOHS without scram accident in the ENHS

Figure 3.21 ENHS power evolution following a LOHS without scram accident

the removal of Pb-Bi from the top and radial reflectors. Nevertheless, if required by the licensing authority, the core can be redesigned to have negative void coefficient. (6) Low power density and consequently very low operating fuel temperatures. (7) Open lattice core that minimizes the probability of local coolant flow blockage. (8) Long and “slim” design that enhances the passive removal of the decay heat from the core through the pool to the Reactor Vessel Air Cooling System (RVACS). The RVACS is the only special safety system needed for the ENHS reactor. The reactor safety is assured by natural physical phenomena rather than by active engineered safety by systems. (9) There can be no control-rod ejection accident. And the maximum excess reactivity is very small throughout the core life. The excess reactivity required to compensate for burnup effects is only $\sim 50\%$. The drive mechanism of the radial reflector segments used for reactivity control limit the possible rate of insertion of this small excess reactivity to hours if not days. In fact, after reaching full power it is envisioned that the height of these reflector segments will be adjusted once every one, two or even more years. (10) As the ENHS reactor pool is located inside a silo and the coolant level is nearly 15 m above the fuel level, it is inconceivable to have a LOCA. LOFA are inconceivable as the flow is by natural circulation (See also “7” above).

Additional information on the safety characteristics of the ENHS reactor can be found in references 3.9, 3.13, 3.14 and 3.19. In addition, Dr. Nishi et al. of CRIEPI carried out an independent evaluation^{3,15} of the safety of the ENHS reactor. They applied the CERES simulation code they have developed to perform an independent evaluation of the response of the ENHS reactor to a protected loss of heat sink (PLOHS) accident and to an unprotected transient over-power (UTOP) accident. The CERES model for the ENHS is shown in Figure 3.22. The only heat removal system in case of the PLOHS event is the RVACS. The UTOP event was initiated by an external reactivity insertion of 3 cent/sec for 10 sec. In the ENHS it is difficult to imagine a reactivity insertion accident such as a control rod withdrawal accident in conventional reactors. This is because the ENHS maintains a nearly constant k_{eff} with burnup so that the maximum excess reactivity that is theoretically available for insertion by complete removal of the safety and control elements of the reactor that operates at full power is less than \$1. Also, the control elements are not used for load following. Moreover, the control elements drive mechanism is designed to have a maximum possible withdrawal speed of 1 mm per second. Hence it is inconceivable to have a rapid reactivity insertion accident in the ENHS reactor. Nevertheless, Dr. Nishi et al. were curious to find the response of the ENHS reactor to a UTOP event in which the rate of reactivity insertion is the same as in a conventional reactor. The external reactivity insertion is assumed to be 3 cents/sec during 10 sec. This is orders of magnitude larger than that possible in the ENHS reactor. The main findings of this study are as follows.

Throughout the PLOHS accident the power/flow ratio is below 1.0 most of the time. The RVACS heat removal capability is adequate. The hot channel cladding mid wall temperature during the transient is lower than of its value in steady state full power operation. The core is sufficiently cooled under the PLOHS condition. The core cooling flow rate is affected by temperature stratification in the primary hot plenum. It is also found that it is desirable to incorporate a vertical wall below the steam generators in the secondary coolant pool. This wall eliminates formation of stagnant cold Pb-Bi layers at the bottom of the secondary coolant pool.

Figure 3.22 CERES model of the ENHS reactor for transient analysis

In the case of the inconceivable UTOP transient considered, the net reactivity reached a peak in 7 seconds and decreased to zero after 60 sec due to reactivity feedback mechanisms that are inherent in the ENHS design. Even though the rise of reactivity is mitigated by negative reactivity feedback, the core temperature settles at a level that could cause an HT-9 type cladding to fail when the pressure in the fission gas plenum is high, if no negative reactivity is inserted soon enough via control element insertion. Nevertheless, a cladding made of high performance ferritic steel can withstand the UTOP event considered with significant margin. It ought be realized that the reactivity insertion rate assumed for the UTOP analyzed is orders of magnitudes larger than that possible for the ENHS reactor.

3.6 Load Following Capability

Due to its negative temperature feedback and to cooling based on natural circulation, the ENHS reactor has a load-following capability over a wide range of power^{3,14}. The reactor power level will adjust itself to the power demand without operator intervention. Figure 3.23 illustrates the response of selected core temperatures to a 30% drop in the load.

Figure 3.23 Temperature evolution in selected core locations following a 30% load reduction

3.7 Steam Generators Design

Figures 2.4 and 2.5 describe the steam generators for the ENHS as originally conceived^{3.19, 3.26}. The design details of the steam generators (SG) were later modified^{3.19, 3.27}, as summarized below.

Figure 3.24 is a schematic description of the steam generator tube-in-tube or “bayonet” type arrangement. Feedwater is introduced at the top to the internal tube and flows down through this tube, change direction at the bottom and flow upward through the annulus where it is heated by the liquid metal flowing down outside the tubes. The inner tube was designed to have an isolating wall so as to minimize the heat transfer from the steam to the feed-water in the bayonet type SG tube arrangement. It is made of a double concentric pipe of 2.5mmID/5mmOD inner tube inside a 7.5mmID/10mmOD sleeve, with inert gas (possibly CO₂) filling the annular gap of 1.25mm between the inner tube and the sleeve. The heat leakage from the steam on the outer side to the feed-water in the inner side is thus reduced to only 1% of the total heat transferred in the SG, (4% of the superheat).

Table 3.12 gives selected design and performance characteristics of the SG for a plant with no reheat (See Sec. 3.13). The optimal operating temperature span of the intermediate Pb-Bi coolant was found to be from 515 down to 331°C. Relative to the nominal operating temperature range assumed so far: from 500 to 350°C, the optimal temperature range offers the following benefits: (a) Almost doubling of the heat transfer rate from the Pb-Bi to the highly superheated steam. (b) Reduction of the Pb-Bi flow-rate by ~18.5% and its friction losses by ~30%, while (c) Increasing the head for intermediate coolant natural circulation by ~18.5%. It was found that there is a large margin to accommodate uncertainties in data and in the modeling without risking the feasibility of 100% natural circulation.

Figure 3.24 A schematic illustration of the steam generator tube-in-tube arrangement

Figure 3.25 shows the temperature difference between the primary coolant, secondary coolant and water/steam along the IHX and the SG. It provides indication on the exergy match and, hence, on the efficiency of the thermal design.

Figure 3.25 Temperature difference between the primary LM, secondary LM and water/steam

Table 3.12 Design and Performance Characteristics of the SG. No Reheat EC System

Design variable	Units	Value
Power per SG	MW _{th}	15.62
Feed-water (122 bars) inlet temperature	C	242
Steam (120 bars) generation rate	kg/s	7.485
Steam outlet temperature	C	482
LM inlet temperature	C	515
Steam exit TTD [temperature difference]	C	33
LM outlet temperature	C	331
LM temperature cooldown	C	184
LMTD in IHX	C	50
Feed-water inlet temperature	C	276
LM flow	kg/s	580
Steam tube OD/ID	mm	20/14
Tube spacing – triangular pitch	mm	28
Number of tubes		613
Tube circle diameter	mm	756
Steam generator external diameter	mm	760
Distance of SG center from pool & module center	mm	~2285
Feed-water tube description		2 concentric tubes
Feed-water tube wall thickness	mm	1.25mm each
Feed-water tube Isolation, to minimize heat transfer		1.25mm layer CO ₂ , in
Feed-water tube OD/ID	mm	10/2.5
Heat transfer area	m ²	177.2
Steam tube heat transfer minimum length for		
- high superheating	m	1.11 [400-482 C]
- medium superheating	m	0.44 [350-400 C]
- low superheating	m	0.23 [325-350 C]
- boiling	m	1.29 [~325 C]
- feed-water heating	m	0.62 [276-325 C]
Total minimum length	m	3.69
Compensation for heat leakage effects, ~4%	m	0.14
Safety factor for various effects, 20%	m	0.77
Total active tube heat transfer length	m	4.60
LM longitudinal mean pressure drop	Pa	1995
LM cross-tube maximum pressure drop	Pa	309
Estimated overall pressure drop	Pa	2100 - 2200
SG material		HT9, SS (or Ti)

Two-dimensional flow simulation showed that at nominal operating conditions only 3% of the intermediate coolant flow bypasses the steam generators. This small bypass has a negligible effect on the ENHS energy balance. The 2-D flow simulation also revealed that it is desirable to install a vertical flow partition in the ENHS pool, extending from below the steam generators

outlet to close to the pool bottom. This partition eliminates stagnation of the coolant at the bottom of the reactor pool.

3.8 Energy Conversion System

A no-reheat Rankine cycle having 120 bar – 482°C inlet steam conditions and 34°C (0.056 bar) condenser was chosen for the reference energy-conversion (EC) system.^{3.19, 3.27} The feedwater heating of this cycle was optimized with the goal of maximizing the cycle efficiency. Six feedwater heaters were found desirable; the central one is the deaerator. Pre-heaters that are heated by superheated steam have the water exit temperature about 2.5 °C above saturation. The highest pressure steam is extracted at 38 bar and the resulting feedwater temperature at the inlet to the SG is 242°C. The required water flow rate is 6.9 kg/s per SG. The gross EC efficiency of this cycle is 41.5%. Assuming turbine and generator mechanical and electrical efficiency of 0.98 each, the electricity generation efficiency is 39.7%. If the plant consumes 5% of the generated electricity the net efficiency is 37.7%. By increasing the pressure to 180 bars, the net efficiency will be 40%.

Figure 3.26 Simplified schematics of the reference single-reheat Rankine steam cycle

Addition of a single reheat will increase the net efficiency by nearly 1% (relative). More importantly, use of a reheater would reduce the maximum steam wetness from 17% to 9% (for

the 120 bar cycle). Figure 3.26 shows schematically the one-reheat cycle for the ENHS plant. The 43.5 bar steam from the high-pressure-turbine exit is reheated from ~340 °C to 482 °C. Table 3.13 compares selected characteristics of the no-reheat versus one-reheat designs.

Table 3.15 Comparison of the Reference Steam Cycle without Reheat to a Cycle with Reheat

Characteristic	units	No-reheat	Single reheat
Reheated steam	kg/s	0	45.5
Main steam	kg/s	59.9	52.3
Medium-pressure-turbine inlet pressure	bar	43.5	41.1
Thermal load of SG including superheat	MWt	125	109.3
Thermal load of superheater [in SG]	MWt	37	32.3
Maximum steam moisture	%	17 - 19	9 - 11
Thermodynamic mean heat-source temperature	°K	611.5	615.5
	°C	338.3	342.3
Corresponding Carnot efficiency	%	49.8	50.1
Plant efficiency	%	38.16	38.42
Net plant generation capacity	MWe	47.70	48.02

3.9 ENHS Fuel Cycle Cost

A generic feature of long-life cores, like the ENHS core, is low specific power and the fact that the full investment in fuel is made initially rather than in steps over the ENHS lifetime. The resulting high inventory of fuel per installed kW_e, along with the long operation time of the fuel results in a high interest payment that has an adverse effect on the fuel cycle cost (FCC). On the other hand, certain unique features of the ENHS design and fuel cycle offer cost reduction. Relative to conventional liquid metal reactor the ENHS has only about half the fissile fuel concentration and approximately one fifth the number of fuel rods per HM loading. Moreover, as the ENHS maintains a nearly constant fissile fuel content (and k_∞) and has no blanket fuel, recycling of the ENHS fuel requires only removal of volatile fission products, addition of ~6% of depleted uranium makeup and re-fabrication. This process is expected to be significantly less expensive than conventional recycling of LMR fuel. The initial fuel loading for the ENHS can be produced from LWR spent fuel that underwent a UREX process that is to extract only part of the uranium and the fission products. This is likely to be less expensive than the conventional PUREX process that is designed to extract pure plutonium.

Nevertheless, being concerned that the ENHS fuel cycle cost may be prohibitively expensive, we embarked on a preliminary estimation of this cost for the reference ENHS reactor design along with a number of possible design variants. Appendix B describes the fuel cycle cost analysis performed; the following is a summary of the results obtained.

Four ENHS fuel cycle scenarios are considered:

1. Using enriched uranium; the required enrichment is 12.7 w/o^{3.18}. The feed is natural uranium.

2. Using standard once through ALWR spent fuel to provide the Pu and minor actinides (MA) for the ENHS core. The transuranium (TRU) composition assumed as feed is that of the fuel discharged from the “UOX open cycle” case of Table 1 of Reference 3.28. This ALWR is fueled with 4.5% enriched uranium (EU) and burns the fuel in 4 years to 55 GWd/tHM. The Pu concentration assumed for the ENHS is 11.4 w/o; this corresponds to an earlier core design^{3.20} that did not account for the effect of the shields; it is somewhat lower than the 12.2 w/o of the reference design. The ALWR spent fuel (SF) is converted to the fuel feed for the ENHS by applying a simplified UREX process to extract only ~90% of the uranium and fission products. Pu or other TRU isotopes are not partitioned.
3. Using TRU from the spent fuel of a MOX fuelled ALWR. The specific spent MOX fuel composition considered is that of the “MOX mono-recycling” case of Table 1 of Ref. 3.28. The TRU contents in the spent fuel is 8.82 w/o. We are assuming that the spent MOX fuel undergoes a UREX process to extract the fission products (FP) and 43% of the uranium. The k_{∞} of the resulting fuel is identical to that of our reference design (Scenario 2); it contains 13.8 w/o Pu (54% of which is fissile) and 1.6 w/o MA.
4. Using the spent fuel from the ENHS to fuel a new ENHS module. Only FP are extract and depleted uranium is added for the makeup fuel. The k_{∞} of the resulting fuel equals that of our reference fuel (Scenario 2). The processing of the ENHS SF is assumed done using a simple process like the AIROX process that is just to extract volatile and semi-volatile FP. Alternatively, a smelt refining process used at EBR-II in the late sixties can be considered. The number of recyclings from one ENHS module to another ENHS module is, in principle, unlimited; only depleted uranium is needed to make up for the fissioned nuclei. Once the first fuel loading for the ENHS has been procured, the owner of the ENHS reactor has fuel supply security and insensitivity to increases in the market price of uranium, enrichment services or plutonium. Scenario 4 should be considered as the representative scenario for judging the ENHS fuel cycle economic viability.

In addition to inter-comparing these four scenarios, the ENHS fuel cycle cost is also compared against a reference once through ALWR addressed in Scenario 2. Its FCC is calculated using the methodology and cost data described in Appendix B. The net efficiency of this ALWR is assumed to be 33%. For curiosity we also considered an older generation LWR that is fed with uranium enriched to 3.1% and burns its fuel in 3 years to 33 GWd/tHM. The efficiency of this reactor is assumed to be 33.8%.

Tables 3.16 and 3.17 compare the FCC when the discount rate is, respectively, 5% and 10%. It is concluded that the fuel cycle cost of the ENHS reactor is not a showstopper. It is concluded that the equilibrium FCC is quite reasonable, even at 10% discount rate. The first fuel loading is more expensive but not prohibitively so. The lowest FCC for the first ENHS is obtained when using as feed the spent fuel from MOX fuelled ALWR (one MOX cycle; Scenario 3). If the ENHS could be design to have a higher power level, as suggested by the analysis in Section 3.9, and if the lower bound cost data are realistic for the ENHS, the FCC of the first ENHS could be competitive with that of the once through ALWR.

The most expensive fuel for the ENHS is that made from spent fuel from UOX fueled ALWR (Scenario 2). Fueling the first ENHS with enriched uranium (12.7% ^{235}U) might be an acceptable option, particularly if the ENHS design upgrades are successful.

Table 3.16 Fuel Cycle Cost of ENHS Reactor Relative to those of Once-Through ALWR.
5% Discount Rate

			ENHS				LWR	ALWR
Scenario			1	2	3	4		Reference
Fuel			EU	TRU-U	TRU-U	TRU-U	EU 3.1%	EU 4.3%
Fuel feed			Natural U (U ₃ O ₈)	Spent UOX	Spent MOX	Spent ENHS	Natural U (U ₃ O ₈)	
Power MWt	Burnup MWd/kg	Efficiency (%)	Fuel cycle cost (mills/kW _e h)					
NOMINAL COST DATA								
125	55.6	38	13.1	19.5	9.3	6.1	4.4 (5.4 ^a)	3.9 (4.9 ^a)
125	55.6	40	12.4	18.6	8.9	5.7		
125	83.4	38	10.9	16.2	7.7	5.0		
125	83.4	40	10.3	15.4	7.3	4.8		
167	55.6	38	11.6	17.4	8.3	5.4		
167	55.6	40	11.0	16.5	7.9	5.1		
167	83.4	38	9.2	13.8	6.6	4.3		
167	83.4	40	8.8	13.1	6.3	4.0		
LOWER BOUND COST DATA								
125	55.6	38	8.4	12.3	5.5	3.4	3.1 (4.1)	2.7 (3.7)
125	55.6	40	8.0	11.7	5.2	3.2		
125	83.4	38	7.0	10.2	4.6	2.8		
125	83.4	40	6.6	9.7	4.3	2.6		
167	55.6	38	7.5	10.9	4.9	3.0		
167	55.6	40	7.1	10.4	4.7	2.8		
167	83.4	38	5.9	8.6	3.9	2.4		
167	83.4	40	5.6	8.2	3.7	2.3		

^a Including a tax of 1 mill/kW_eh to the Federal Government

Increase in the cost of uranium ore will make the ENHS FCC more favorable relative to the LWR FCC cost. This is because the ENHS FCC is practically independent of the cost of uranium. This robustness to fluctuations in the uranium market need be taken into account in evaluating the viability of the ENHS reactor. In fact, once a utility or a country owns an ENHS power plant along with its first fuel loading, it has the fuel supply for centuries if not millennia, as the only makeup fuel they need is depleted uranium.

Table 3.17 Fuel Cycle Cost of ENHS Reactor Relative to those of Once-Through ALWR.
10% Discount Rate

			ENHS				LWR	ALWR
Scenario			1	2	3	4		Reference
Fuel			EU	TRU-U	TRU-U	TRU-U	EU 3.1%	EU 4.3%
Fuel feed			Natural U (U ₃ O ₈)	Spent UOX	Spent MOX	Spent ENHS	Natural U (U ₃ O ₈)	
Power MWt	Burnup MWd/kg	Efficiency (%)	Fuel cycle cost (mills/kW _e h)					
NOMINAL COST DATA								
125	55.6	38	20.4	37.3	14.8	9.6	4.5 (5.5)	4.2 (5.2)
125	55.6	40	19.4	35.4	14.1	9.1		
167	83.4	38	14.8	27.1	10.8	7.0		
167	83.4	40	14.1	25.7	10.3	6.6		
LOWER BOUND COST DATA								
125	55.6	38	13.2	23.5	8.8	5.3	3.2 (4.2)	2.9 (3.9)
125	55.6	40	12.5	22.3	8.3	5.1		
167	83.4	38	9.6	17.1	6.4	3.9		
167	83.4	40	9.1	16.2	6.1	3.7		

3.10 Feasibility of Shipping and Installing a New ENHS Module

The feasibility of shipping the assembled and fuelled ENHS reactor module from the factory and installing it in the pool vessel at the site has been assessed. The reactor module consists of the reactor vessel containing the reactor core frozen in Pb-Bi and all the components that are internal to the reactor vessel. This will include instrumentation and control components and electrical cabling required for operating the reflectors. Our perception for loading the fuel into the ENHS module and for supporting it in the core is described in Section 2.3. The package to be shipped is approximately 19.6m long and 3.2m in diameter and weighs ~300 tons. This is a relatively modest size package compared to the steam generators and pressure vessels for large LWRs, some of which are nearly 1000 tons. The size and the mass of the package do not present any feasibility issues. The reactor module is however a unique shipping package, in that it contains the reactor fuel and control reflectors frozen in Pb-Bi. There is no commercial precedence for licensing such a shipping package.

Shipping of the ENHS reactor module will be similar to many other shipments of large heavy assemblies. There are several unique features that have been considered and determined to not be “show stoppers” for the shipping of the unique package. It is preferable, but not absolutely necessary, to have the manufacturing facility located adjacent to a waterway that has access to the sea. This would improve access to sites worldwide, many of which are near the sea. If this were not the situation there are multi –wheeled land transporters that can support transfer of the reactor module from the factory to the barge. Even in the preferred case such a transporter may be used and shipped along with the reactor module to support local transportation at the site. Rail transport may be possible over limited routes. It will probably require a special rail car and the

length of the car may limit the rail routes to those that do not have curves that exceed the turning radius of the special rail car.

The reactor module will be placed in a horizontal position following filling with Pb-Bi to a level slightly above the core and freezing the Pb-Bi. The filling operation must be done with the reactor module in the vertical position. It will be secured to a shipping fixture that may also be a strong-back to support lowering the reactor module from vertical to horizontal at the factory, and raising the reactor module to the vertical position for installation at the site. This fixture will be designed to accommodate the concentrated mass of Pb-Bi at one end of the vessel. All of the components will need to be designed to withstand the vibrations and impacts associated with shipping. An assessment of the vessel wall and the IHX tubes to withstand the shipping loads was completed in the first year of this study and determined to be feasible^{13,1}. The components frozen in Pb-Bi are not expected to be vulnerable to these loads since the Pb-Bi experiences very little volume change on freezing. Any void space in the Pb-Bi will consist of very thin gaps between the Pb-Bi and components. Issues associated with instrumentation and cabling that are internal to the vessel have not been evaluated since the design details have not been developed.

Since the reactor module contains radioactive material in the form of low enriched uranium or plutonium it must be licensed for shipment under 49CFR Subpart I and 10CFR71. The package is similar to other shipments of new fuel except the issues related to criticality are unique. The fuel is shipped along with the reflectors and the central shutdown assembly inserted into and around the core. All of these components are frozen in Pb-Bi. In this form criticality is not conceivable. The package will remain sub-critical when submerged in water. As long as the Pb-Bi remains solid there is no possibility of a criticality. The only threat to the Pb-Bi becoming liquid is for it to be heated to above 125°C. If this is determined to be a threat it would only be necessary to include fire protective material around the lower portion of the vessel to avoid the heating. Even if some of the Pb-Bi were to become liquid in a fire, several additional failures would need to occur to reach criticality: removal of the central absorbing assembly and of the reflector segments. It appears feasible to relegate these postulated failure scenarios to the incredible category by design and therefore licensing the fresh reactor module for shipment appears feasible.

Depending on the details of the fuel and core mechanical design it may be necessary to conduct a proof test for the fuel and core mechanical features to assure that the design margins are sufficient to preclude shipping damage that could impact core lifetime or core restraint features that are important to safety. This is not a feasibility issue but may require testing to establish complete confidence in the shipping package, licensing of the shipping package and reliable operation following shipment. It appears possible to design simple feature tests that can bound the shipping environment and demonstrate the adequacy of the design.

Installation of the reactor module into the pool vessel will require special heavy lift equipment and other special equipment to melt Pb-Bi and fill the reactor module. This equipment is also required for installation and filling of the pool vessel that would have preceded installation of the reactor module and is therefore available on a new construction site prior to the arrival of the reactor module. If the installation operation was involving replacement of a spent reactor module the equipment would have also been used for the removal operation, so the necessary equipment

is also available prior to arrival of the reactor module at the site. It is envisioned that this special equipment is mobile and the responsibility of the supplier of the ENHS reactor system, since it is used nominally once every 20 years per reactor module. Portable cranes with capacities up to 850 tons have been used at nuclear power construction sites, so adequate lifting capacity can be provided. Table 3.18 outlines the major steps required installing the reactor module.

Table 3.18 Steps for Reactor Module Installation

Step	Operation
1	Prepare pool vessel with transfer valve over reactor module opening. Maintain pool coolant temperature and chemistry with auxiliary systems.
2	Prepare reactor module and installation envelope* for enclosing the reactor module and transfer valve.
3	Lift reactor module from horizontal to vertical and position over transfer valve. Attach Pb-Bi fill line from auxiliary fill system.
4	Enclose the reactor module and fill line with installation envelope and attach to transfer valve.
5	Inert free volume within envelope. Open transfer valve and lower reactor module into secondary coolant.
6	Begin filling reactor module and lowering into secondary pool.
7	Position reactor module sealing surface with pool cover mating surface. Close the transfer valve and evacuate the gas spaces to gas processing systems.
8	Remove installation envelope, transfer valve, fill line and make permanent attachment of the reactor module to secondary pool cover (structural support platform).

* The installation envelope is envisioned to be a flexible clear plastic bag but a steel cylinder may be necessary for removal in which case it would also be used for installation.

Several of the steps and features in this installation outline have challenges that require further evaluation as the design develops. They do not represent feasibility issues but practical improvements that could be implemented during detail design. The assumption that a clear plastic bag can be used as the transfer envelope is based on similar experience transferring large components in and out of sodium filled vessels. The objective is to avoid both oxygen contamination of the pool coolant and the release of radioactive material that may escape from the pool surface and internal surfaces of the pool vessel. The plastic bag and the transfer valve, which is a large gate valve, permit this operation to be done with a minimum level of contamination. For various reasons it may be determined that it will be desirable to employ a steel cylinder as the transfer enclosure. Design details beyond the scope of this project are needed to make this determination.

The most problematic detail of the installation process is associated with the filling of the reactor module with Pb-Bi. The reactor module when lowered into the pool vessel will submerge about 4 meters while displacing the volume of Pb-Bi equivalent to approximately 300 tons. As it is filled, it will sink further into the pool until the flange of the reactor module mates with the pool flange. The pool coolant level will be increasing as the ENHS module is inserted. The secondary coolant system will be adjusted at the end of this operation.

The major challenge is to have the Pb-Bi fill pipe penetrate the transfer envelope and accommodate the vertical movement of the reactor module as it sinks into the pool. Telescoping, articulated or flexible piping, or a combination of these features, is required and piping must be capable of containing the Pb-Bi at approximately 400°C. Again this is probably not a feasibility issue but a challenge that will require inventiveness to achieve a practical design.

As the ENHS module is inserted into the pool and is filled with hot Pb-Bi, the frozen Pb-Bi shipped with the module will heat up by heat conduction from the pool and from the hot Pb-Bi layer above. Within 2 to 3 days the Pb-Bi in the core will melt and reach the temperature of ~350°C required for reactor startup (See following section). In order to reach the startup temperature it may be necessary to add some external heating. This could be achieved by circulating some of the secondary Pb-Bi from the pool to its dump tank in which a heater will be installed.

It is concluded that it appears feasible to ship and install a factory fueled and sealed ENHS module. The current concept for installation and filling the reactor module with Pb-Bi requires innovative piping that can transport the liquid Pb-Bi from grade level to the top of the reactor module and accommodate the elevation change of the top of the reactor module as it is installed in the pool vessel. This feature or other alternatives for filling the reactor module requires further consideration but it does not appear to be a feasibility issue. Licensing of the shipping package may also provide a challenge since it is a rather unique shipping package for which there are no precedents.

3.11 Feasibility of Melting the Solidified Pb-Bi in the as Shipped ENHS Module

The ENHS module is to be delivered from the factory to the site with the primary system filled to the top of the fuel rods with solid lead or lead-bismuth at ambient temperature (assumed to be 20°C). As the module is inserted into the secondary system pool that is filled with hot molten coolant, molten coolant is added to the module up to the primary coolant operation level. Both primary and secondary molten coolants fill the two sides of the intermediate heat exchanger (IHX). Sensible heat from the added molten coolant and the secondary system pool is conducted to the solid coolant the fuel is embedded in. The question is if the module could be heated to its startup temperature by the sensible heat of the hot liquid metal in the pool and in the ENHS module above the fuel without use of an external heater.

It is found feasible to melt the primary coolant and bring it to the reactor startup temperature within approximately two days of its insertion to the pool without having to use special external heaters in the module. In fact, most of the melting may occur in the process of filling the module with hot coolant and inserting the module into the pool. The dominant pathway for heat transfer to the solid coolant in and below the core is heat conduction from the secondary coolant in the pool. The ~2 days time to melt correspond to use of Pb in the module. A shorter time to melt is expected for Pb-Bi. For more details see Section 10 of Reference 3.9

3.12 Feasibility of Solidifying Pb in the ENHS Core at EOL

At the end of the approximate 20 years of full-power operation, the ENHS module is replaced with a new one. After the core has reached the end of its usable life and is permanently shut down, the module will be cooled by special cooling loops to be inserted, after shutdown, adjacent to the inner structural wall. With this external cooling operating, the lead-bismuth coolant is to be pumped out of the module and lead is to be pumped up to above the core level (the core stays always covered with liquid metal). While the coolant level inside the primary vessel drops, the module is raised. After the module is out of the secondary coolant, the special cooling loops cause the lead to solidify. At this point the module is taken out completely from the pool and installed in a nearby silo for a cooling period. The length of this period is determined by the time it takes for the lead to remain solid if the special cooling loops or other means for forced cooling is turned off. Only then can the module be shipped out of the plant. The feasibility question addressed in this section is how long needs the module cool in the silo before it could be shipped out. Two cooling approaches were considered: natural convection cooling and forced cooling. The analysis was done using simplified analytic models as well as using a three-dimensional computational fluid dynamic (CFD) analysis.

The cooling period required for solidifying Pb in the core using natural convective cooling in air was estimated to be on the order of 10 years. However, it was found feasible to remove the decay heat from the ENHS core shortly after one week by cooling the cylindrical outer boundary of the core and the surface of the shutdown assembly channel located at the center of the core, provided that lead is frozen in the core. If cooling is provided only at the outer boundary, then the lead can be kept solidified after 22 days. These estimates are based on a very simplified model and are likely to be an underestimation of the required cooling time. Moreover, practical design of the decay heat removal system from the spent ENHS module need be worked out.

The usage of lead rather than lead-bismuth eutectic is crucial to the feasibility of the surface cooling approach. The higher freezing temperature and greater thermal conductivity of solid lead enable the calculated removal of power levels with lead that are ten times those calculated for LBE.

For details see Sections 11 and 12 of Reference 3.9. A more thorough study is required in order to identify a practical design that will offer an acceptable time for solidifying the lead in the core.

3.13 Feasibility of Removing the Spent ENHS Module from the Reactor Pool

It is desirable to remove the spent reactor module and ship it to the recycling facility as soon as is practical following its ultimate shutdown. Achieving a short period of on-site residence for the spent reactor module following removal is believed to be a favorable non-proliferation measure. The 300 ton radioactive vessel with fuel weld sealed inside presents a formidable deterrent by itself. However, leaving the spent reactor module stored on-site for many years while it cools is not desirable from non-proliferation and economic considerations. Therefore a goal of less than one year for removal and shipping has been selected. The evaluation reported in the previous section indicates that the Pb-Bi must be replaced with lead to achieve this goal.

The task of removing the reactor module essentially reverses the steps required for its installation, described in Section 3.10. However, it has the added challenges of radiation and decay heat removal. Also, removing the heavy Pb-Bi from within the vessel is not as easy as pumping it in.

At shutdown both the Pb-Bi inside the reactor module and the Pb-Bi in the pool are liquid. In order to solidify the coolant in the core before module removal within reasonable time after shutdown, it is necessary to remove all the Pb-Bi and back fill the core with lead. The lead should be filled up to the top core level as it was when originally shipped from the factory, so as to reduce the mass of the reactor module to facilitate removal and shipping. The coolants in the pool and reactor module must be maintained liquid throughout this operation. This can be done using the decay heat and the pool coolant service system. Heat to the pool coolant could be supplied to avoid freezing by electrical heaters in the coolant service system when decay heat transfer to the pool is not available.

The pumping out of Pb-Bi is envisioned done using positive displacement pumps. Such pumps could be inserted in between the inner and outer structural walls in the space between the nested tube modules of the intermediate heat exchanger. A lead fill pipe will be permanently installed in the vessel, possible in the region between the inner and outer structural wall. This pipe will extend nearly to the bottom of the vessel. As the Pb-Bi is pumped out of the vessel lead will replace it by flowing into the vessel near the bottom. This replacement will continue until the correct volume of lead has been introduced into the core. The supply of lead will be shut off and the Pb-Bi will continue to be pumped out. The pumps will be located at the top of the core so that they will not remove the lead that has replaced the Pb-Bi. This approach is possible because the Pb-Bi is less dense than the lead and will float on the lead as it is removed and the lead replaces it. Although the idea of placing a positive displacement pump into the vessel to remove the Pb-Bi appears feasible it requires a design effort to determine exactly how this might be accomplished and the capacity that might be achieved in such a pump. It would be desirable to pump the level down in a few days but obtaining the pumping capacity to do this in the available space in the reactor module may be difficult.

External cooling to the reactor module will be started shortly after the lead has completely replaced the Pb-Bi and will continue until the lead has solidified. This cooling is done with coolant flowing through pipes that were installed during manufacture and have remained unused throughout the reactor module operational life. Presently, water is considered for this coolant. The cooling water will remove the decay heat and the sensible heat required to bring the lead to its solid state. Freezing could be achieved within approximately one week of shutdown (See Section 3.12).

Table 3.19 outlines the major steps for removal and preparation of the reactor module for shipping to the recycling center. The objective of the procedure is to obtain a shipping package that can be licensed by DOT and NRC as a standard package. There do not appear to be any feasibility issues associated with the removal of the spent reactor module, however the evaluation of the shielding requirements have not been completed. If the shielding requirements were to be massive there could be various problems with adding the additional mass. Lifting and heat removal may become a challenge and structural support for the extra shielding may prove

difficult. These items require further design detail and evaluation to determine their impact. A steel shell could be used in place of the plastic enclosure envisioned for the transfer envelope. This piece of equipment might be adequate for shielding the reactor module during removal and transfer but is unlikely to be suitable for shipping. Also, if the dominant radiation source is the reactor vessel activation it may be possible to reduce the level of this source with additional internal shielding. The dose at the surface of the shipping package must be reduced to less than 100mR/hr to meet current 10CFR71 shipping package requirements.

Table 3.19 Steps for Removing The ENHS Module and Preparing it for Shipment

Step	Operation
1	There will be a period of preparation prior to shutdown in which all auxiliary equipment needed to complete the operation is brought into place on the site.
2	The reactor is shutdown and the features or special equipment necessary to maintain the pool coolant liquid is brought into operation. This may be heat supplied electrically to the pool coolant or externally to the pool vessel. Alternately, it could be heater could be associated with the coolant service system used to maintain the pool level.
3	A transfer gate valve is secured to the top of the pool vessel in the open position and the pump and piping for removing the Pb-Bi and the piping for back-filling with lead are attached. The cooling water connections are made so that the lead can be frozen at the appropriate time.
4	The reactor vessel attachments are disconnected, including the safety rod and reflector drive power leads. These mechanisms are locked in their safe positions.
5	The pool vessel cover gas is evacuated and replaced with clean inert gas. The radioactive cover gas is exhausted into a container for retention and disposal
6	The enclosure into which the reactor module is to be withdrawn is put in place. This operation must be integrated with the installation of the piping for Pb-Bi, lead and cooling water. The air within the enclosure is replaced with inert gas and the gate valve is opened.
7	The positive displacement pump and its piping are inserted into the reactor vessel down to near the top of the core. The primary coolant is pumped from the vessel through penetrations in the enclosure into an external tank. The piping features external to the reactor must accommodate the rise of the vessel as the coolant is removed. A pipe installed in the reactor module at the factory that goes to a point near the bottom of the vessel is used to insert Pb and displace Pb-Bi. When the operation is complete there will be Pb only in the vessel to a level slightly above the core and the reactor module is floating partially submerged in the secondary pool.
8	The water cooling flow is started until the lead is frozen. At this point the reactor module can be lifted from the pool and transferred to the on-site storage vault.
9	Following the transfer to storage location the cooling will continue until the lead is solid.

By regulation, the active cooling used to freeze the lead cannot be used during shipping. Therefore the module will be moved from above the reactor pool vault to an on site storage silo. This facility is envisioned to be a nearby, below grade, silo that is capable of safely supporting the module while it cools and the decay heat is reduced. It is necessary for the decay heat level to

reduce to the level that the module can be cooled by radiation and natural convection with the lead maintained frozen solid throughout the core.

The spent-module shipping package is expected to be similar to the original package shipped to the site with some additional mass for steel shielding around the core region of the vessel. Packages of this form in the range of 1000 tons have been shipped.^{14.1, 14.2} It is expected that the shielding will add, at most 100 additional tons which means the spent shipping package should be about 400 tons. Such a package can clearly be transported, however, the design of the package and a conceptual review of its licensability needs to be completed.

In conclusion, the removal and shipping of the spent reactor module from the reactor pool is a major challenge but appears to be feasible. The assessment is incomplete at this time; shielding of the spent module, licensable decay heat removal, and mechanisms for removing the Pb-Bi all need to be addressed in greater detail.

3.14 Feasibility of Transporting the Spent ENHS Module

The spent ENHS module contains more than type A quantities of radioactive material and must be shipped as a type B package. Two possible configurations for packaging the spent module have been identified. The first package configuration uses a cask to contain the ENHS module and is based on conventional Type B spent fuel shipping casks. The second configuration uses the ENHS module as the containment system and is based on the decommissioned Trojan and Shippingport reactor packages. Both reactors had their spent fuel removed and were shipped to the Hanford site in the state of Washington where they were buried.

The spent ENHS module shipment must meet all applicable transportation regulations for radioactive materials. The shipping package for the spent ENHS module must be certified by both 10CFR71 US regulations for domestic shipments and IAEA Safety Series 6 regulations for international shipments. The two regulations are similar. Rather than going into detailed requirements and differences for the two regulations, only the three requirements that are most difficult to meet and are potential show stoppers were considered. The three requirements are: Safety Analysis Report (SAR) preparation and submittal (10CFR71, Subpart D), package evaluation for normal conditions of transport (10CFR71.71) and package evaluation for hypothetical accident conditions 10CFR71.73.

Careful preparation of the SAR is important because the SAR provides the basis for the certification of the transport package. One input to the SAR that is particularly relevant to the spent ENHS Module is the Type B package category designation. The package category determines which code and other criteria are appropriate for components that affect the structural integrity of containment, criticality or shielding systems. The spent ENHS module package category is the highest, category one, because it contains more than 30,000 curies of radioactive material. The Trojan reactor package was category 1 whereas the Shippingport reactor was category 2.

The spent ENHS module package must be evaluated for all normal conditions of transport (NCT). The evaluations must show if a radioactive release occurs under NCT, the release must

be less than a $1\text{E-}06$ A2 quantity per hour. The evaluations can be performed by test and/or by analysis. The two toughest NCT requirements are the one-foot drop onto an unyielding surface and the 85°C maximum temperature limit on any accessible surface. The one-foot drop requirement can be met by attaching impact limiters to the package to reduce the deceleration of the package. The 85°C requirement can be met by using a screen enclosure to limit access to the ENHS package.

The spent ENHS module package must also be evaluated for all hypothetical accident conditions (HAC). The evaluations must show if a radioactive release occurs under HAC, the releases must be less than an A2 quantity in one week. The evaluations can be performed by test and/or by analysis. The three toughest HAC requirements are the 30-foot drop onto an unyielding surface, the 45-inch drop onto a carbon steel pin, and the one-half hour engulfing 800°C fire. The 30-foot drop requirement can be met by attaching impact limiters to the package to reduce the deceleration of the package. The 45-inch drop requirement can be met with a ductile material and a containment wall thicker than 4 inches. The fire requirement can be met because of the large mass of the spent ENHS module and the use of a relief valve if cooling water is contained in the module.

The first spent ENHS module transport package configuration uses conventional proven technologies and design principles to meet type B, category 1 requirements. A cylindrical cask that is 22 meters long is used to contain the spent ENHS module. The maximum outer diameter of the cask is 6.25 meters and the minimum outer diameter is 4.25 meters. The cask wall thickness is 12.7 centimeters. The bottom end is a 2:1 ellipsoid enclosure that is 12.7 centimeters thick and is welded to the main cask cylindrical body. The top end is a bolted 2:1 ellipsoid enclosure that is 12.7 centimeters thick. The top enclosure uses 72 bolts and a bore seal with two o-rings to close the cask and make it leak tight. Support posts at the top and bottom closures hold the spent ENHS module in place during NTC and HAC.

A basket is used to center and support the spent ENHS module in the cask. The basket is a tubular liner with welded spacer disks. The spacer disks are alternate pie sections and arranged to allow water and steam to flow around the ENHS module and transfer heat to the inner wall of the cask. The heat transfer mechanism and analysis for cooling the ENHS module are described in the IF 300 SAR (Chapter VI of NEDO-10084-3 report; September 1984).

Impact limiters are attached at both ends of the cask. The limiters use a 304 stainless steel sheath to contain low-density cellular grout (LDCG). The impact limiters are designed to limit the loads to the cask to less than 100 g's by crushing and deforming the LDCG contained in the sheath.

The cask and basket will meet all category I type B packaging requirements. They are fabricated from 304 stainless steel in accordance with the ASME BPVC, Section III Division I, Subsections ND, NG and NF. Brittle fracture requirements are met by using 304 stainless steel. A pressure relief valve is probably needed to meet HAC requirements.

The second configuration uses the spent ENHS module for the containment system. The entrance and exit ports to the module are welded shut to contain the radioactive material. Impact limiters

similar to those used in configuration 1 are attached directly to the spent ENHS module. This approach was used for both the Trojan and Shippingport reactor vessels but they first had the spent fuel removed. Both vessels were at essentially room temperature because they did not have any significant heat source. It is not apparent how the heat from the spent fuel in the ENHS reactor can be transferred and distributed to the outer vessel surface of the module. Also it is unlikely that the ENHS containment material can meet US NRC brittle fracture requirements after 20 years of irradiation.

The Trojan reactor vessel package was defined as category 1 but did not meet many of the requirements. It did not meet the 30-foot drop, the 45-inch drop on a pin and US NRC brittle fracture requirements. The vessel was shipped from Portland Oregon up the Columbia River to the Hanford site under an exemption to 10CFR71 based on a Probabilistic Risk Assessment.

The Shippingport reactor vessel package was defined as category 2 and met all regulatory requirements. The reactor vessel was surrounded by a thermal shield, which had low radiation damage and met US NRC brittle fracture criteria for category 2. The thermal shield was extended and became part of the containment vessel for the package. The package was certified under both the 10CFR71 and IAEA regulations. The package was shipped down the Ohio and Mississippi Rivers, through the Panama Canal and up the Columbia River to the Hanford site for burial.

It is concluded that it is likely that Configuration 1 of the spent ENHS module transport cask could be certified by both the US NRC and the IAEA for domestic and international transport. This approach is relatively heavy (1000 MT) and relatively costly (\$5M for cask) but is do-able.

It is unlikely that Configuration 2 of the spent ENHS module could be certified unless some significant regulatory and regulatory-guide changes occur. The first issue is the large number of curries that are contained in an irradiated reactor vessel that are not readily dispersible. Changes in regulatory guidance can address both dispersible and non-dispersible radioactive material in establishing package categories. Also brittle fracture requirements need be reviewed again to determine if a linear elastic fracture approach might be acceptable for packages with little or no dispersible radioactive material. Finally it is uncertain that spent fuel shipped in a reactor frozen in place by lead can be considered to be non-dispersible.

3.15 References

- 3.1 S. Hattori and N. Handa, "Use of Super-Safe, Small and Simple LMRs to Create Green Belts in Desertification Area," Trans. American Nuclear Society, 60, 437 (1989).
- 3.2 K. Aoki, S. Kasai and S. Hattori, "Design of Small and Simple LMR Cores for Power Generation in Remote Communities," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
- 3.3 S. Hattori and A. Minato, "A Large Modular LMR Power Station Which Meets Current Requirements," Proc. 3rd ASME-JSME Int. Conf. On Nuclear Engineering, ICONE-3, Tokyo, Japan, Oct. 1995.
- 3.4 J. Buongiorno, A. Barak and E. Greenspan, "Irradiation Performance of the Metal Fuel for the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, 86, 348-349, Hollywood, FL, June 2002.

- 3.5 S.G. Hong, E. Greenspan and Y.I. Kim, "Once-for-Life Core for the Encapsulated Nuclear Heat Source (ENHS) Reactor," Trans. American Nuclear Society, **86**, 143-145, Hollywood, FL, June 2002.
- 3.6 K. L. Derstine, "DIF3D : A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, ANL (April 1984).
- 3.7 B. J. Toppel, "A User's Guide for the REBUS-3 Fuel Cycle Analysis Capability," ANL-83-2, Argonne National Lab. (Mar. 1983)
- 3.8 S. G. Hong et al., "A Comparative Neutronics Analysis by Using ENDF/B-VI and JEF2.2 Libraries for the ENHS Benchmark Problem," Proc. International Conference on the New Frontiers of Nuclear Technology: Reactor Physics, Safety and High-Performance Computing, PHYSOR-2002, Seoul, Korea, October 7-10, 2002.
- 3.9 E. Greenspan, A. Barak, D. Barnes, J. Buongiorno, N. W. Brown, D. Carelli, S.H. Chang, L. Conway, M. Dzodzo, E. Feldman, S.G. Hong, I.S. Hwang, S.H. Jeong, Y.I. Kim, K.G. Lee, M. Milosevic, B.G. Park, D. Saphier, T. Sofu, J. J. Sienicki, and D C. Wade, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 2nd year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4234.
- 3.10 R. L. Moore et al., "MOCUP: MCNP-ORIGEN2 Coupled Utility Program," Idaho National Engineering Laboratory Report INEL-95/0523, 1995.
- 3.11 J. F. Briesmeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code", Los-Alamos National Laboratory Report LA-12625-M, Version 4B (1997).
- 3.12 A. G. Croff, "ORIGEN2 - A Revised and Updated Version of the Oak Ridge Code," Oak Ridge National Laboratory Report ORNL/TM-7175 (1980).
- 3.13 D. Saphier, E. Greenspan, D. C. Wade, M. Dzodzo, L. Conway and N.W. Brown, "Some Safety Aspects of the Encapsulated Nuclear Heat Source LMR Plant" Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 3.14 D. Saphier, M. Dzodzo and E. Greenspan, "Applying the DSNP Modular Modeling System to Transient and Accident Simulations of Lead Cooled Reactors," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 3.15 Y. Nishi, N. Ueda, I. Kinoshita and E. Greenspan, "Simulation of the Thermal Hydraulic Characteristics of the ENHS," Submitted to Nuclear Technology.
- 3.16 T. Sakai and T. Iwasaki, "Thermal-Hydraulic Analysis on the Encapsulated Nuclear Heat Source (ENHS)", Submitted to the 2004 International Congress on Advances in Nuclear Power Plants, Pittsburgh, PA, June 13-17, 2004.
- 3.17 S.G. Hong, E. Greenspan, Y.I Kim and H.R. Kim, "ENHS (Encapsulated Nuclear Heat Source) Core Designs Using Nitride Fuel," Proc. of GLOBAL'03, New Orleans LA, Nov. 16-20, 2003.
- 3.18 E. Greenspan, H. Shimada and K. Wang, "Long-Life Cores with Small Burnup Reactivity Swing," Proc. of the 2000 Int. Topical Mtg. On Advances in Reactor Physics and Mathematics and Computation into the Next Millennium, PHYSOR-2000, Pittsburgh, PA, May 7-11, 2000.
- 3.19 E. Greenspan, N. W. Brown, D. Carelli, L. Conway, M. Dzodzo, K. Grimm, R. Hill, Q. Hossain, D. Saphier, H. Shimada, J. J. Sienicki, D C. Wade and S. Wang, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232, February 2001.

- 3.20 Greenspan and the ENHS project team, "The Long-Life Core Encapsulated Nuclear Heat Source Generation IV Reactor," Proc. of the International Congress on Advanced Nuclear Power Plant, ICAPP, Hollywood, FL, June 9-13, 2002.
- 3.21 S.G. Hong, E. Greenspan and Y. I. Kim, "Once-for-Life Core for the Encapsulated Nuclear Heat Source (ENHS) Reactor," Proc. International Conference on the New Frontiers of Nuclear Technology : Reactor Physics, Safety and High-Performance Computing, PHYSOR-2002, Seoul, Korea, October 7-10, 2002.
- 3.22 Z. Shayer, M. Milosevic, D. Barnes and E. Greenspan, "Shielding Optimization for the ENHS Reactor," Submitted to the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
- 3.23 J.J. Sienicki and D.C. Wade, "Thermal Hydraulic Analysis of the Encapsulated Nuclear Heat Source," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 3.24 J.J. Sienicki, "Feasibility of Natural Circulation Heat Transport in the ENHS", Trans. American Nuclear Society, 85, 74-75, Reno, NV, Nov. 2001.
- 3.25 J. J. Sienicki, "Updated Thermal Hydraulic Analyses for Recent ENHS Design Improvements," Submitted to the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
- 3.26 L. Conway, Q. Hossain, D. C. Wade, N. W. Brown, M. D. Carelli, , M. Dzodzo, E. Greenspan, W.E. Kastenberg, D. Saphier and J. J. Sienicki, "Promising Design Options For The Encapsulated Nuclear Heat Source Reactor" Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 3.27 A. Barak and E. Greenspan, "ENHS Reactor Energy-Conversion Efficiency and Steam Generators Size," Trans. American Nuclear Society, 85, 77-78, Reno, NV, Nov. 2001.
- 3.28 J.P. Grouiller et al., "Different Possible Scenarios for Plutonium Recycling in PWRs," Proc. Global'01, Paris, France, September 9-13, 2001.

4 ALTERNATIVE DESIGN OPTIONS

4.1 Alternative Fuels

A study was performed of the sensitivity of the core design and performance characteristics to the initial fuel composition^{4.1}. Four alternative metallic alloy fuel compositions containing 10^w% Zirconium are considered: (1) U with Pu from a typical FBR; this composition was used in early studies of the ENHS cores. (2) U with Pu from LWR spent fuel; this is the reference ENHS core composition addressed in Section 3. (3) U with all the Transuranics (TRU) from LWR. (4) Th with all the TRU from LWR. Table 4.1 gives the isotopic composition of the TRU of the four alternative compositions. The core design goal, assumptions and methodology are the same used for the design of the reference core (Sec. 3.3) with two exceptions: in the present study SS-316 rather than HT-9 was used for the structural material and the special radiation shields were not accounted for.

Table 4.1 Fuel Compositions Considered in the ENHS Sensitivity Analysis

Isotope	Weight %			
	FBR Pu (Design I)	LWR Pu (Design Ia)	LWR TRU ^a (Design Ib)	LWR TRU ^a (Design Ic)
²³⁷ Np			6.641	6.641
²³⁸ Pu		3.18	2.749	2.749
²³⁹ Pu	67.2	56.352	48.652	48.652
²⁴⁰ Pu	21.7	26.62	22.98	22.98
²⁴¹ Pu	6.4	8.02	6.926	6.926
²⁴² Pu	4.7	5.828	5.033	5.033
²⁴¹ Am			4.654	4.654
^{242m} Am			0.019	0.019
²⁴² Cm			1.472	1.472
²⁴³ Cm			0.005	0.005
²⁴⁴ Cm			0.496	0.496
²⁴⁵ Cm			0.038	0.038
²⁴⁶ Cm			0.006	0.006
Fertile	²³⁸ U	²³⁸ U	²³⁸ U	²³² Th

^a Corresponding to 50 GWd/tHM and 10 years cooling. Design Ib uses U while Ic & Id use Th.

Table 4.2 compares selected design and performance characteristics of the four cores. For the TRU-Th(Zr) core, Design-I(c), we could not find an acceptable design featuring nearly zero burnup reactivity swing for 20 EFPY but only for ~10 EFPY. It calls for a tight lattice of p/d=1.18 and for a high initial TRU concentration of 18.5^w%. This poorer performance of thorium-based fuel is due to its relatively low fast fission probability, small conversion ratio, and to the lower η value of ²³³U vs. fissile Pu above ~0.1 MeV. In the following we exclude from the comparison cores having thorium as the fertile fuel.

Tables 4.3 and 4.4 summarize selected core physics characteristics of Design-I, Design-I(a), and Design-I(b) at BOL and EOL, respectively.

Table 4.2 Lattice Pitch-to-Diameter Ratio, TRU Weight % and BU Reactivity Swing of the Alternative Core Designs

Design	p/d ratio	TRU or Pu wt%	Burnup reactivity swing (% Δk)
Design I	1.35	11.344	0.197
Design I(a)	1.33	12.101	0.234
Design I(b)	1.39	14.515	0.204
Design I(c)	1.18	18.519	0.411 ^a

^a Reactivity swing for 10 years.

Table 4.3 Selected Physics Data at BOL of Alternative Cores

Performance Parameter	Design I	Design I(a)	Design I(b)
Peak-to-average power density	1.767	1.814	1.817
Peak linear heat rate (W/cm)	177.8	181.8	182.1
Peak-to-average channel power	1.425	1.472	1.486
Peak-to-average power density in hot channel	1.241	1.233	1.231
Peak burnup after 20 (21.9) EFPY (GWD/tHM)	95.2	98.8	98.7
Average burnup after 20 (21.9) EFPY	50.7	50.6	50.6
Peak fast ($E > 0.1$ MeV) neutron flux (n/cm ² -s)	5.681E+14	5.921E+14	5.639E+14
Peak fast ($E > 0.1$ MeV) fluence at 20 (21.9)	3.646E+23	3.807E+23	3.635E+23
Conversion ratio	1.153	1.231	1.159
Effective delayed neutron fraction	0.00383	0.00393	-
Doppler effect (dk/kk' -°C)	-5.785E-06	-5.595E-06	-4.453E-06
Axial fuel expansion (dk/kk' -°C)	-4.422E-06	-4.623E-06	-4.611E-06
Coolant expansion (dk/kk' -°C)	+5.171E-07	+8.430E-07	+1.752E-06
Grid-plate radial expansion (dk/kk' -°C)	-8.974E-06	-8.705E-06	-8.907E-06
Cold (350°C) to hot (480°C; fuel: 700°C) dk/kk'			
Doppler effect	-2.034E-03	-1.729E-03	-1.386E-03
Axial fuel expansion	-0.566E-03	-0.641E-03	-0.643E-03
Coolant expansion	+0.045E-03	+0.060E-03	+0.156E-03
Grid-plate expansion	-1.153E-03	-1.086E-03	-1.141E-03
Total	-3.708E-03	-3.396E-03	-3.013E-03
Void reactivity effect (dk)			
Voiding inner 1/3 core	+2.549E-02	+2.711E-02	+3.200E-02
Voiding middle 1/3 core	+0.886E-02	+0.847E-02	+1.032E-02
Voiding outer 1/3 core	-0.316E-02	-0.426E-02	-0.466E-02
Voiding whole core	+3.111E-02	+3.028E-02	+3.631E-02
Tungsten annulus reactivity worth (dk)	0.0208	0.0207	0.0190
Central absorber reactivity worth (dk)	0.035	0.036	0.035
Peripheral absorber+central absorber worth (dk)	0.0609	0.06324	0.05978
Total plutonium mass (kg)	1985.9	2118.2	2193.9
Fissile-to-total plutonium ratio	0.7360	0.6437	0.6437

The characteristics of the three cores are quite similar. Notable differences are found in the Doppler effect and in the coolant expansion reactivity coefficient. Due to the higher weight percent of TRU, the Doppler coefficient of the core fueled with TRU from LWR [Design I(b)] is smaller than that of the core fueled with typical FBR Pu. The reactivity coefficient of coolant expansion for Design-I(c) is most positive because its coolant volume fraction is the largest. A similar trend is observed in the void coefficient.

Table 4.4 Selected Physics Data at EOL of Alternative Cores

Performance Parameter	Design I	Design I(a)	Design I(b)
Peak-to-average power density	1.778	1.8542	1.8467
Peak linear heat rate (W/cm)	178.7	185.9	185.2
Peak-to-average channel power	1.435	1.501	1.516
Peak-to-average power density in hot channel	1.235	1.235	1.227
Peak fast ($E > 0.1\text{MeV}$) neutron flux ($\text{n/cm}^2\text{-s}$)	5.895E+14	6.030E+14	5.766E+14
Conversion ratio	1.065	1.1097	1.0845
Doppler effect ($\text{dk/kk}'\text{-}^\circ\text{C}$)	-4.556E-06	-4.399E-06	-3.792E-06
Axial fuel expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	-4.396E-06	-4.624E-06	-4.654E-06
Coolant expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	+9.681E-07	+1.111E-06	+1.796E-06
Grid-plate radial expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	-6.804E-06	-5.725E-06	-6.263E-06
Cold (350°C) to hot (480°C ; fuel: 700°C) $\text{dk/kk}'$			
Doppler effect	-1.656E-03	-1.560E-03	-1.373E-03
Axial fuel expansion	-0.577E-03	-0.611E-03	-0.615E-03
Coolant expansion	+0.140E-03	+0.153E-03	+0.244E-03
Grid-plate expansion	-0.897E-03	-0.761E-03	-0.818E-03
Total	-2.990E-03	-2.818E-03	-2.563E-03
Void reactivity effect (dk)			
Voiding inner 1/3 core	+2.657E-02	+2.732E-02	+3.168E-02
Voiding middle 1/3 core	+0.945E-02	+0.819E-02	+0.981E-02
Voiding outer 1/3 core	-0.298E-02	-0.427E-02	-0.468E-02
Voiding whole core	+3.281E-02	+2.999E-02	+3.526E-02
Tungsten annulus reactivity worth (dk)	0.01047	0.01882	0.0177
Central absorber reactivity worth (dk)	0.042	0.03579	0.03503
Peripheral absorber+central absorber worth (dk)	0.0609	0.06087	0.05875
Total plutonium mass (kg)	2057.5	2155.5	2244.7
Fissile-to-total plutonium ratio	0.7288	0.6647	0.6417

Table 4.5 compares the actinides inventory in the four cores, including the thorium-containing core, Design I(c). In Design I(c) nearly 530 kg of ^{239}Pu are consumed and a similar amount of ^{233}U is produced.

It is concluded that there is significant flexibility in designing once-for-life cores for the ENHS, as long as ^{238}U is the primary fertile fuel. The use of different TRU compositions will have only a small effect on the feasibility of natural circulation as the p/d ratio and the peak-to-average power density are not very sensitive to the TRU composition loaded into the core.

Table 4.5 Initial Actinides Inventory and its Change Over Core Life

Isotopes	Design I		Design I(a)		Design I(b)		Design I(c)	
	Mass (kg) at BOL	Increased Mass (kg)	Mass (kg) at BOL	Increased mass (kg)	Mass (kg) at BOL	Increased mass (kg)	Mass (kg) at BOL	Increased mass (kg)
²³² Th	-	-	-	-	-	-	10880	-835
²³³ U	-	-	-	-	-	-	0.0	532.3
²³⁵ U	31.04	-13.54	30.77	-13.03	30.44	-12.17	0.48	1.354
²³⁸ U	15490	-1070	15360	-1063	14940	-980	7.84	-0.626
²³⁸ Pu	0.0	6.96	67.44	-22.48	69.85	23.56	66.34	20.9
²³⁷ Np	0.0	3.17	0.0	3.17	168.8	-52.9	160.3	-57.86
²³⁹ Pu	1335	116	1194	177.2	1236	140	1174	-528.1
²⁴⁰ Pu	430.9	33.1	563.8	-1.18	583.9	3.0	554.6	-40.12
²⁴¹ Pu	127.1	-78.92	169.9	-108.0	176.0	-111.7	167.1	-109.5
²⁴² Pu	93.34	-6.26	123.5	-8.28	127.9	-4.0	121.5	-5.24
²⁴¹ Am	0.0	56.95	0.0	75.14	118.3	39.6	112.3	30.70
^{242m} Am	0.0	1.803	0.0	2.390	0.0483	6.11	0.458	5.84
²⁴³ Am	0.0	7.038	0.0	9.278	37.40	-1.6	35.5	-2.21
²⁴² Cm	0.0	0.422	0.0	0.555	0.0	1.12	0.0	1.07
²⁴³ Cm	0.0	0.01	0.0	0.014	0.127	-0.052	0.121	-0.053
²⁴⁴ Cm	0.0	0.916	0.0	1.213	12.6	-1.5	11.97	-1.298
²⁴⁵ Cm	0.0	0.062	0.0	0.082	0.965	1.14	0.917	1.164
²⁴⁶ Cm	0.0	0.0014	0.0	0.002	0.153	0.088	0.145	0.091

In a study performed after the termination of the ENHS NERI project it was found^{4,2} that it is feasible to design uniform composition blanket free cores for the ENHS reactor that feature nearly zero burnup reactivity swing using nitride fuel with either natural N or enriched N. Nitride fuel using enriched N enables achieving up to ~25% higher discharge burnup and longer core life than metallic fuel due to their somewhat softer spectrum, i.e., reduced radiation damage rate to the clad material. Combined with the ~ 10% larger HM loading of the nitride fuel, nitride fuel cores can generate up to ~38% more energy than the reference metallic fuel core.

Nitride fuel core with enriched N has the largest p/d ratio of 1.45 implying a reduction of the coolant friction loss through the core and, therefore, the possibility of increasing the core power that can be removed by natural circulation cooling. On the other hand, nitride fuel core using natural N has the smallest p/d ratio of 1.21 and, hence, the most compact core and smallest ENHS module diameter.

Concerning safety-related reactor physics characteristics, relative to the reference metallic fuel core, nitride fuel cores offer the following advantages: (a) Significantly more negative Doppler reactivity coefficient; due to the softer spectrum. (b) In case of natural N, less positive coolant expansion reactivity coefficient, and (c) less positive core coolant voiding reactivity effect. Disadvantages of the nitride fuel cores are: (a) Not as negative fuel expansion reactivity coefficient. (b) In case of enriched nitride fuel, more positive coolant expansion reactivity coefficient and more positive void reactivity coefficient. (c) Somewhat lower reactivity worth of

the central and peripheral absorbers. The net balance appears to be positive; the enhanced Doppler coefficient more than compensates for the reduced fuel expansion coefficient, and it is possible to increase the control elements reactivity worth by a minor design modification.

4.2 Alternative Coolants

A study was performed of the sensitivity of the core design and performance characteristics to the initial fuel composition^{4.1}. Three combinations of primary and secondary coolants were considered: (1) Pb-Bi for both primary and secondary coolant – the reference design, (2) sodium for both the primary and secondary coolants, and (3) sodium for the primary coolant and Pb-Bi for the secondary coolant. The core design goal, assumptions and methodology are the same used for the design of the reference core (Sec. 3.3) with two exceptions: in the present study SS-316 rather than HT-9 was used for the structural material and the special radiation shields were not accounted for. LMR Pu is assumed.

Two alternative core designs were worked-out: Design II uses sodium for the primary and secondary coolants and Design III uses sodium for the primary coolant and lead-bismuth for the secondary coolant. The design of an optimal core involves a parametric search of a combination of the lattice p/d ratio and the initial plutonium weight percent giving a BOL k_{eff} of 1.0042 and a minimum value of burnup reactivity swing over 20 years of full power operation. The outcome of this search is summarized in Table 4.6. It is found that uniform composition cores having no blanket rods and no solid reflectors can be designed for the ENHS using sodium for the primary coolant with either sodium or lead-bismuth for the secondary coolant. The optimal p/d ratio with all sodium cooling (Design II) is smaller than with all Pb-Bi cooling (Design I) by 14% and the use of Pb-Bi for the secondary coolant increases the p/d ratio by 3.45%.

Table 4.6 P/D Ratio, Pu wt%, and Burnup Reactivity Swing for Different Coolant Combinations

Characteristic	Design I	Design II	Design III
Primary coolant	Pb-Bi	Na	Na
Secondary coolant	Pb-Bi	Na	Pb-Bi
P/D ratio	1.35	1.16	1.20
Initial Pu wt%	11.344	11.012	11.145
Burnup reactivity swing ($\% \Delta k$)	0.197	0.125	0.144

Figure 4.1 compares the neutron spectra of Design I, Design II, and Design III. It is seen that Design I has the hardest neutron spectrum. Of the sodium-cooled cores, Design III has a slightly softer spectrum than Design II. This is probably due to the somewhat larger p/d of Design III. The pitch of Design III is larger than that of Design II because the leakage probability from Design III is smaller due to the superior reflection properties of Pb-Bi. Table 4.7 compares the neutron balance in the different cores. Although sodium cooled cores are more leaky, the fraction of neutrons absorbed in the coolant is smaller than that of Design I. The combined neutron loss via leakage and absorption in coolant varies slightly from design to design. Had the same pitch (that of Design I) was used for all cores, the neutron loss rate in the sodium-cooled cores would have been significantly larger and the spectrum of these cores would have been significantly softer.

Figure 4.1 Average neutron spectrum of ENHS cores of different coolant combinations

Table 4.7 Neutron Balance in the Reference Designs

Fraction of neutron lost in core (%)	Design I	Design II	Design III
Via leakage	17.2	19.5	18.47
Via absorption in coolant	1.53	0.122	0.149
Combined	18.7	19.62	18.61

Tables 4.8 and 4.9 compare selected characteristics of the three core designs at BOL and EOL, respectively. It is found that sodium cooled cores have significantly more positive reactivity coefficients of coolant expansion than lead-bismuth cooled cores. This is probably due to the fact that the thermal expansion coefficient of sodium is more than twice that of Pb-Bi. The void coefficients for 100% voiding of the inner and middle zones of the cores are positive. The void coefficient for the all-sodium reactor is the smallest as sodium does not make as good a reflector as does Pb-Bi. This positive void reactivity in the sodium-cooled core can be problematic, unlike the case for Pb-Bi cooled core. This is due to the relatively low boiling temperature of sodium – 1165K; it is nearly 800K lower than the boiling temperature of Pb-Bi.

Table 4.8 Selected Core Physics Data at BOL for ENHS Reactor With Different Coolants

Performance Parameter	Design I	Design II	Design III
Peak-to-average power density	1.767	1.979	1.838
Peak linear heat rate (W/cm)	177.8	198.4	184.1
Peak-to-average channel power	1.425	1.609	1.518
Peak-to-average power density in hot channel	1.241	1.222	1.205
Peak fast ($E > 0.1 \text{ MeV}$) neutron flux ($\text{n/cm}^2\text{-s}$)	$5.681\text{E}+14$	$6.016\text{E}+14$	$5.381\text{E}+14$
Doppler effect ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-5.785\text{E}-06$	$-7.740\text{E}-06$	$-8.784\text{E}-06$
Axial fuel expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-4.422\text{E}-06$	$-4.673\text{E}-06$	$-4.580\text{E}-06$
Coolant expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$+5.171\text{E}-07$	$+2.552\text{E}-06$	$+4.903\text{E}-06$
Grid-plate radial expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-8.974\text{E}-06$	$-9.746\text{E}-06$	$-9.788\text{E}-06$
Cold (350°C) to hot (480°C ; fuel: 700°C) $\text{dk/kk}'$			
Doppler effect	$-2.034\text{E}-03$	$-2.583\text{E}-03$	$-2.727\text{E}-03$
Axial fuel expansion	$-0.566\text{E}-03$	$-0.645\text{E}-03$	$-0.643\text{E}-03$
Coolant expansion	$+0.045\text{E}-03$	$+0.501\text{E}-03$	$+0.576\text{E}-03$
Grid-plate expansion	$-1.153\text{E}-03$	$-1.297\text{E}-03$	$-1.271\text{E}-03$
Total	$-3.708\text{E}-03$	$-4.024\text{E}-03$	$-4.065\text{E}-03$
Void reactivity effect (dk)			
Voiding inner 1/3 core	$+2.549\text{E}-02$	$+1.940\text{E}-02$	$+2.152\text{E}-02$
Voiding middle 1/3 core	$+0.886\text{E}-02$	$+0.732\text{E}-02$	$+0.972\text{E}-02$
Voiding outer 1/3 core	$-0.316\text{E}-02$	$-0.429\text{E}-03$	$-0.123\text{E}-02$
Voiding whole core	$+3.111\text{E}-02$	$+2.651\text{E}-02$	$+3.316\text{E}-02$
Peripheral absorber reactivity worth (dk)	0.0208	0.01306	0.02343
Central absorber reactivity worth (dk)	0.0350	0.03692	0.03456
Peripheral absorber+central absorber worth(d k)	0.0609	0.05343	0.06404

The peripheral absorbers of the sodium-cooled cores are made of 40% tungsten at 75% nominal density, 40 % B_4C (92% ^{10}B) and 20% SS and are 5.0 cm thick. For the same material and thickness as of Pb-Bi cooled design, the reactivity worth of the peripheral absorber in the sodium-cooled cores is significantly smaller. This is because the Na in the downcomer does not make as good a reflector as Pb-Bi. The reactivity worth of the peripheral absorber of Design III is almost twice that of Design II. This is due primarily to the better neutron reflection properties of Pb-Bi used for the secondary coolant.

The central absorber of Design II and III is made of the same material as that of their peripheral absorbers and its thickness is 5.0 cm. The reactivity worth of the central absorber of Design II is smaller than that of Design III. If needed, the reactivity worth of the central and peripheral absorbers could be increased by increasing their thickness.

Table 4.9 Selected Core Physics Data at EOL for ENHS Reactor With Different Coolants

Performance Parameter	Design I	Design II	Design III
Peak-to-average power density	1.778	2.000	1.859
Peak linear heat rate (W/cm)	178.7	200.5	186.2
Peak-to-average channel power	1.435	1.626	1.526
Peak-to-average power density in hot channel	1.235	1.227	1.213
Peak fast ($E > 0.1\text{MeV}$) neutron flux ($\text{n/cm}^2\text{-s}$)	$5.895\text{E}+14$	$6.104\text{E}+14$	$5.469\text{E}+14$
Peak burnup after 20 EFPY (GWD/tHM)	95.2	107.1	99.22
Average burnup after 20 EFPY (GWD/tHM)	50.7	50.7	50.5
Peak fast fluence ($E > 0.1\text{MeV}$) (n/cm^2)	$3.646\text{E}+23$	$3.847\text{E}+23$	$3.476\text{E}+23$
Doppler effect ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-4.556\text{E}-06$	$-6.003\text{E}-06$	$-6.851\text{E}-06$
Axial fuel expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-4.396\text{E}-06$	$-4.747\text{E}-06$	$-4.679\text{E}-06$
Coolant expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$+9.681\text{E}-07$	$+4.276\text{E}-06$	$+6.730\text{E}-06$
Grid-plate radial expansion ($\text{dk/kk}'\text{-}^\circ\text{C}$)	$-6.804\text{E}-06$	$-7.166\text{E}-06$	$-6.954\text{E}-06$
Cold (350°C) to hot (480°C ; fuel: 700°C) $\text{dk/kk}'$			
Doppler effect	$-1.656\text{E}-03$	$-2.221\text{E}-03$	$-2.538\text{E}-03$
Axial fuel expansion	$-0.577\text{E}-03$	$-0.607\text{E}-03$	$-0.616\text{E}-03$
Coolant expansion	$+0.140\text{E}-03$	$+0.586\text{E}-03$	$+0.872\text{E}-03$
Grid-plate expansion	$-0.897\text{E}-03$	$-0.923\text{E}-03$	$-0.917\text{E}-03$
Total	$-2.990\text{E}-03$	$-3.165\text{E}-03$	$-3.200\text{E}-03$
Void reactivity effect (dk)			
Voiding inner 1/3 core	$+2.657\text{E}-02$	$+2.213\text{E}-02$	$+2.450\text{E}-02$
Voiding middle 1/3 core	$+0.945\text{E}-02$	$+0.836\text{E}-02$	$+1.094\text{E}-02$
Voiding outer 1/3 core	$-0.298\text{E}-02$	$-0.011\text{E}-02$	$-0.163\text{E}-02$
Voiding whole core	$+3.281\text{E}-02$	$+3.060\text{E}-02$	$+3.775\text{E}-02$
Peripheral absorber reactivity worth (dk)	0.01047	0.01163	0.02109
Central absorber reactivity worth (dk)	0.042	0.03599	0.03359
Peripheral absorber+central absorber worth(d k)	0.0609	0.50972	0.60132

Figure 4.2 compares the axial power distributions in the hottest channel at BOL and Figure 4.3 compares the radial power distributions at BOL for the three reference core designs – Design I, II, III. Of the three designs, Design II has the largest power peaking factor of 1.979 whereas Design I has the smallest power peaking factor of 1.767. Figure 4.3 shows that the lead reflector flattens the radial power distribution relative to sodium reflector. With sodium coolant there is a power density spiking at the inner core boundary; it is due to the enhanced slowing down of neutrons at the central cavity when filled with sodium.

Figure 4.2 Coolant effect on axial power distributions in the hottest channel at BOL

Figure 4.3 Coolant effect on radial power distribution at BOL

Table 4.10 compares selected thermal-hydraulic characteristics of the ENHS reactor using the three different coolant combinations.

Table 4.10 Coolant Types Effect on Selected Thermal Hydraulic Characteristics of the ENHS

Characteristic	Coolant, Primary/Intermediate		
	Design I	Design II	Design III
Coolant type, Primary/Intermediate	LBE/LBE	Na/Na	Na/Pb
Fuel Rod Pitch, cm	2.106	1.8096	1.872
Pitch-to-Diameter Ratio	1.35	1.16	1.20
Core Peak-to-Average Power	1.77	1.98	1.84
Hot Channel Factor	1.44	1.62	1.52
Power Factor at Top of Hot Channel	1.06	0.993	1.03
Core Hydraulic Diameter, cm	1.57	0.755	0.917
Core Flow Area, m ²	1.83	1.20	1.36
Primary Coolant Flowrate, Kg/s	8320	666	751
Intermediate Coolant Flowrate, Kg/s	5970	686	5885
Coolant Velocity in Core, m/s	0.447	0.651	0.647
Primary Coolant Velocity in HX, m/s	0.438	0.419	0.472
Intermediate Coolant Velocity in HX, m/s	0.440	0.604	0.420
Intermediate Coolant Velocity in SG, m/s	0.285	0.391	0.272
Reynolds Number in Core	47400	15800	18800
Primary Coolant Reynolds Number in HX	159000	72400	80400
Intermediate Coolant Reynolds Number in HX	128000	87500	86200
Intermediate Coolant Reynolds Number in SG	44000	30100	29700
Primary Coolant Temperature Rise, C	103	147	130
Intermediate Coolant Temperature Rise, C	143	142	144
Primary Coolant Outlet Temperature, C	453	497	480
Primary Coolant Inlet Temperature, C	350	350	350
Intermediate Coolant Outlet Temperature, C	421	469	452
Intermediate Coolant Inlet Temperature, C	278	326	308 ^(a)
Hot Channel Primary Coolant T Rise, C	147	238	198
Hot Channel Outlet Temperature, C	497	588	548
Hot Channel Peak Clad Outer Surface T, C	518	591	552
Hot Channel Peak Clad Inner Surface T, C	529	599	560
Hot Channel Peak Sodium Bond T, C	533	602	563
Hot Channel Peak Fuel Temperature, C	572	628	593

^(a) Inconsistent with use of molten lead for which $T_{mp} = 327$ C. However, Pb-Bi will give similar performance.

Of the three coolant combinations considered, only that using LBE (or Pb) for the primary coolant meets the hot channel coolant temperature rise constraint of 150°C. However, this temperature rise constraint was imposed by mass transport consideration and does not apply to Pb or Pb-Bi coolant.

It is concluded that it is possible to meet all the design goals set for the ENHS core using any one of the LM coolant combinations considered. Relative to Pb-Bi or Pb cooled cores, Na cooled cores feature tighter lattice and therefore more compact cores but spikier power distribution, more positive coolant temperature reactivity coefficient but comparable overall negative reactivity coefficient, and smaller reactivity worth of the absorbers used for reactivity control.

4.3 Compact ENHS Module Using Lift Pumps

Section 2.5 briefly describes the lift-pump variant of the ENHS reactor. A simplified thermal-hydraulic analysis has been done for the ENHS with lift-pumps for both the primary and the secondary coolants. Assumptions and model used for this analysis can be found in references 4.3 through 4.7. The dimensions of the cores and their performance correspond to designs using LMR Pu and SS structural material and that are not shielded.

Table 4.11 compares design and performance characteristics of the preferred lift pump ENHS design with that of the reference ENHS (No lift pump). It can be seen that the riser length needed for the lift-pump design is only about 3 meters as compared with ~13 meters required for the 100% natural circulation (reference) design. Consequently, the lift-pump ENHS vessel height needs to be only approximately 10 meters. Even with half the total vessel height, the lift-pump design of Table 4.11 has significantly larger temperature margins than the reference design: for example, primary coolant hot channel temperature rise of 102°C versus 147°C

Table 4.12 summarizes results of a simplified thermal-hydraulic analysis of lift-pump driven modules for the three different coolant combinations considered in Section 4.2. It is found that only the Pb-Bi (or Pb) cooled design can meet the 150°C limit on the coolant temperature rise across the core, and it can do so with a large margin. Hence, it should be possible to increase the power per module. The hot channel temperature rise for the all-Na cooled reactor is 240°C – more than twice that of all-LBE (or Pb). Since the Na cooled core does not have a technological constraint on the coolant temperature rise, the Na cooled designs of the lift-pump version of the ENHS may also be designed to have somewhat above nominal thermal power level.

Table 4.13 compares selected thermal-hydraulic characteristics of the lift-pump ENHS using Pb-Bi (or Pb) coolants for several power levels: up to 2.5 times the nominal power. It is found that the power per module could be increased by 50% -- from 125 to nearly 190 MW_{th}, without exceeding the design constraints. Such a power increase can have a significant contribution to the economic viability of the ENHS (See Section 3.9). For sodium cooled ENHS with lift pumps it was found possible to increase the nominal thermal power by ~25%. However, for the same average core outlet temperature the sodium cooled design will have a lower average heat-delivery temperature than the LBE cooled design so the thermal-to-electrical energy conversion efficiency of the sodium cooled reactor will be smaller than that of the LBE (or Pb) cooled

reactor. It is concluded that sodium is inferior to LBE (or Pb) as a coolant for the ENHS both from the viewpoint of attainable power and from safety considerations.

More accurate modeling of the lift-pump performance as well as detailed safety analysis are required before sound conclusions could be derived on the promise of the lift-pump design. The safety analysis should include loss-of-lift-pump (LOLP) accidents that are irrelevant for the reference (100% natural circulation ENHS reactor). LOLP accidents could result from loss of power to drive the cover-gas circulators, breakdown of circulators or cover gas leakage. It appears, though, that the lift-pump ENHS reactor could be designed to passively remove decay heat in case of LOLP accidents.

Table 4.11 Design and Performance Characteristics of the Preferred Lift-Pump ENHS versus the Reference ENHS

Case Designation	Reference	With Lift Pump
Primary Coolant	LBE/Lead	LBE/Lead
Secondary Coolant	as primary	as primary
Roughness Height, μm	10	10
Power, MW_{th}	125	125
Heated Core Height, m	1.25	1.25
Fission Gas Plenum Height, m	1.25	1.25
Total Core Height, m	2.5	2.5
Fuel Rod Diameter, cm	1.56	1.56
Fuel Rod Pitch, cm	2.106	2.106
Pitch-to-Diameter Ratio	1.35	1.35
Cladding Thickness, cm	0.13	0.13
Fuel Material	U-11 wt% Pu-10 wt% Zr	
Fuel Diameter, cm	1.126	1.126
Cladding-Fuel Bond Material	Sodium	Sodium
Cladding-Fuel Bond Gap Thickness, cm	0.0871	0.0871
Fuel Smeared Density, %	75	75
Number of Fuel Rod Spacer Grids	3	3
Core Radius, m	1.10	1.10
Core Peak-to-Average Power	1.77	1.77
Hot Channel Factor	1.44	1.44
Power Factor at Top of Hot Channel	1.06	1.06
Riser Height, m	13	3.0
HX Channel Height, m	11	1.75
Primary Coolant Thermal Centers Separation, m	8.125	3.50
HX Channel Width, cm	2.5	0.5
HX Channel Thickness, cm	39.2	49.2
Number of HX Channels	135	434

Case Designation	Reference	With Lift Pump
Number of SG Modules	8	8
SG Height, m	4.8	4.8
SG Tube Outer Diameter, cm	2.0	2.0
Number of SG Tubes per SG Module	613	613
SG Coolant Inlet to HX Outlet Height Difference, m	1.15	1.15
Intermediate Coolant Thermal Centers Separation, m	4.25	-0.375
Core Hydraulic Diameter, cm	1.57	1.57
Primary Coolant HX Hydraulic Diameter, cm	5.39	1.58
Intermediate Coolant HX Hydraulic Diameter, cm	4.70	0.990
Intermediate Coolant SG Hydraulic Diameter, cm	2.50	2.50
Core Flow Area, m ²	1.83	1.83
Primary Coolant HX Flow Area, m ²	1.86	1.87
Intermediate Coolant HX Flow Area, m ²	1.32	1.07
Intermediate Coolant SG Flow Area, m ²	2.04	2.04
Primary Coolant Flowrate, Kg/s	8320/8220	12100/12200
Intermediate Coolant Flowrate, Kg/s	5970/5870	7840/7910
Coolant Velocity in Core, m/s	0.447/0.426	0.646/0.630
Primary Coolant Velocity in HX, m/s	0.438/0.418	0.630/0.615
Intermediate Coolant Velocity in HX, m/s	0.440/0.418	0.715/0.697
Intermediate Coolant Velocity in SG, m/s	0.285/0.271	0.374/0.364
Reynolds Number in Core	47400/30800	66900/43800
Primary Coolant Reynolds Number in HX	159000/103000	65400/42800
Intermediate Coolant Reynolds Number in HX	128000/80700	42900/27500
Intermediate Coolant Reynolds Number in SG	44000/27800	56500/36300
Primary Coolant Temperature Rise, C	103/104	70.8/69.8
Intermediate Coolant Temperature Rise, C	143/144	109/107
Primary Coolant Outlet Temperature, C	453/454	421/420
Primary Coolant Inlet Temperature, C	350	350
Intermediate Coolant Outlet Temperature, C	421/427	393/395
Intermediate Coolant Inlet Temperature, C	278/283 ^(a)	284/288 ^(a)
Hot Channel Primary Coolant Temperature Rise, C	147/149	102/101
Hot Channel Outlet Temperature, C	497/499	452/451
Hot Channel Peak Cladding Outer Surface T, C	518/517	470/466
Hot Channel Peak Cladding Inner Surface T, C	529/529	481/477
Hot Channel Peak Sodium Bond Temperature, C	533/532	484/481
Hot Channel Peak Fuel Temperature, C	572/571	526/522

^(a) Inconsistent with use of molten lead for which $T_{mp} = 327$ C. However, Pb-Bi will give similar performance.

Table 4.12 Coolant Types Effect on Selected Thermal Hydraulic Characteristics of the ENHS
With Lift Pumps

Characteristic	Coolant, Primary/Intermediate		
	Design I	Design II	Design III
Coolant type, Primary/Intermediate	LBE/LBE	Na/Na	Na/Pb
Fuel Rod Pitch, cm	2.106	1.8096	1.872
Pitch-to-Diameter Ratio	1.35	1.16	1.20
Core Peak-to-Average Power	1.77	1.98	1.84
Hot Channel Factor	1.44	1.62	1.52
Power Factor at Top of Hot Channel	1.06	0.993	1.03
Core Hydraulic Diameter, cm	1.57	0.755	0.917
Core Flow Area, m ²	1.83	1.20	1.36
Primary Coolant Flowrate, Kg/s	12100	665	779
Intermediate Coolant Flowrate, Kg/s	7840	601	7920
Coolant Velocity in Core, m/s	0.646	0.650	0.671
Primary Coolant Velocity in HX, m/s	0.630	0.416	0.487
Intermediate Coolant Velocity in HX, m/s	0.715	0.656	0.701
Intermediate Coolant Velocity in SG, m/s	0.374	0.343	0.367
Reynolds Number in Core	66900	15700	19400
Primary Coolant Reynolds Number in HX	65400	21100	24200
Intermediate Coolant Reynolds Number in HX	42900	19900	30200
Intermediate Coolant Reynolds Number in SG	56500	26300	39800
Primary Coolant Temperature Rise, C	70.8	147	125
Intermediate Coolant Temperature Rise, C	109	162	107
Primary Coolant Outlet Temperature, C	421	497	475
Primary Coolant Inlet Temperature, C	350	350	350
Intermediate Coolant Outlet Temperature, C	393	476	432
Intermediate Coolant Inlet Temperature, C	284	314	325 ^(a)
Hot Channel Primary Coolant T Rise, C	102	240	192
Hot Channel Outlet Temperature, C	452	590	542
Hot Channel Peak Clad Outer Surface T, C	470	593	545
Hot Channel Peak Clad Inner Surface T, C	481	601	554
Hot Channel Peak Sodium Bond T, C	484	604	557
Hot Channel Peak Fuel Temperature, C	526	630	587

^(a) Inconsistent with use of molten lead for which $T_{mp} = 327$ C. However, Pb-Bi will give similar performance.

Table 4.13 Selected Thermal-Hydraulic Characteristics for the ENHS Operating at Different Power Levels. With Lift-Pumps

Power, MWt	125	156	188	250	313
Power, Times Nominal	1.0	1.25	1.5	2.0	2.5
Coolant	LBE/Lead	LBE	LBE	LBE	LBE
Primary Coolant Flowrate, Kg/s	12100/12200	12700	12500	12900	13200
Intermediate Coolant Flowrate, Kg/s	7840/7910	7810	7770	7690	7610
Coolant Velocity in Core, m/s	0.646/0.630	0.659	0.671	0.693	0.714
Primary Coolant Velocity in HX, m/s	0.630/0.615	0.642	0.654	0.676	0.696
Intermediate Coolant Velocity in HX, m/s	0.715/0.697	0.711	0.707	0.699	0.691
Intermediate Coolant Velocity in SG, m/s	0.374/0.364	0.372	0.370	0.366	0.362
Reynolds Number in Core	66900/43800	69000	71100	75200	79000
Primary Coolant Reynolds No in HX	65400/42800	67500	69500	73500	77300
Intermediate Coolant Reynolds No in HX	42900/27500	42300	41800	40500	39200
Intermediate Coolant Reynolds No in SG	56500/36300	55800	55100	53500	51700
Primary Coolant Temperature Rise, C	70.8/69.8	86.9	103	133	161
Intermediate Coolant T Rise, C	109/107	137	165	222	280
Primary Coolant Outlet Temperature, C	421/420	437	453	483	511
Primary Coolant Inlet Temperature, C	350	350	350	350	350
Intermediate Coolant Outlet T, C	393/395	403	413	433	452
Intermediate Coolant Inlet Temp, C	284/288 ^(a)	267	248	211	171
Hot Channel Primary Coolant T Rise, C	102/101	125	148	191	232
Hot Channel Outlet Temperature, C	452/451	475	498	541	582
Hot Channel Peak Clad Outer T, C	470/466	497	523	574	623
Hot Channel Peak Clad Inner T, C	481/477	511	540	596	651
Hot Channel Peak Sodium Bond T, C	484/481	516	546	605	661
Hot Channel Peak Fuel Temperature, C	526/522	565	603	676	745

^(a) Inconsistent with use of molten lead for which $T_{mp} = 327$ C. However, Pb-Bi will give similar performance.

4.4 Nested Tube IHX

The presently preferred IHX design concept for the ENHS is made of rectangular channels that are stacked in the annulus between the inner- and outer- structural walls. The primary coolant flows in the channels from top to bottom whereas the secondary coolant flows in the space

around the channels from bottom to top. An alternative IHX design concept identified for the ENHS uses circular tubes instead of rectangular ones. This design approach is described in Section 9 of References 4.6 and 4.7.

Advantages of the proposed nested tube IHX concept include the following: (1) This IHX can be made from off-the-shelf components. (2) It has high rigidity against buckling. (3) It is easier to fabricate than the rectangular channel IHX. (4) It can withstand higher pressure-differential than the rectangular channel IHX. In fact, it can withstand any pressure differential caused by different levels of the primary and secondary coolants. (5) The gap between nested tube IHX modules can be utilized for insertion of cooling loops for decay heat removal and solidification of the primary coolant before removal of the used module.

The main disadvantage is that circular nested IHX needs to be 13% - 30% longer than IHX with rectangular channels. This will result in increased friction losses and decreased head for natural circulation. Alternatively, larger ENHS module diameter will be required.

4.5 Integrated Steam Generator and IHX

This IHX concept eliminates the secondary coolant loop by integrating the steam generation with the intermediate heat exchanger (IHx)^{4,9}. It involves replacing the secondary liquid metal coolant in the IHx by water/steam while separating the water/steam from the primary coolant by a thin layer of liquid metal or by a solid heat conducting material. The drawbacks of this design approach are: (1) Increase in the Module outer diameter. (2) Complicated interface between the Module and the energy conversion system. (3) More complicated inspection and maintenance of the SG. However, if proven feasible this design approach will result in a significant reduction in the size and cost of the ENHS power plant. The feasibility of this design concept has not been studied, as yet.

4.6 Supercritical CO₂ Energy Conversion System

The idea is to use supercritical CO₂ instead of water for the working fluid of the energy conversion (EC) system. The advantages of CO₂ - based energy-conversion system is that it can offer a comparable or even higher EC efficiency⁴¹ while simplifying the interface between the EC system and the liquid metal heat source. The interface simplifications result from the reduced chemical reactivity and reduced pressure of the CO₂ as well as from the smaller volume increase associated with pressure drop (due, say, to a breach in an HX tube). Another advantage of the supercritical CO₂ EC system is that it is much more compact and, hence, less expensive, than the water/steam Rankine EC system. The feasibility of this design concept has not been studied, as yet, in the framework of the ENHS.

4.7 References

- 4.1 S.G. Hong, E. Greenspan and Y.I. Kim, "Alternative Design Options for the ENHS (Encapsulated Nuclear Heat Source) Reactor," Submitted to the 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.

- 4.2 S.G. Hong, E. Greenspan, Y.I Kim and H.R. Kim, "ENHS (Encapsulated Nuclear Heat Source) Core Designs Using Nitride Fuel," to be published in Proc. of GLOBAL'03, New Orleans LA, Nov. 16-20, 2003.
- 4.3 J. J. Sienicki, "Updated Thermal Hydraulic Analyses for Recent ENHS Design Improvements," 2003 International Congress on Advances in Nuclear Power Plants, Cordoba, Spain, May 4-7, 2003.
- 4.4 E. Greenspan, N. W. Brown, D. Carelli, L. Conway, M. Dzodzo, K. Grimm, R. Hill, Q. Hossain, D. Saphier, H. Shimada, J. J. Sienicki, D C. Wade and S. Wang, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 1st year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4232, February 2001.
- 4.5 J.J. Sienicki and D.C. Wade, "Thermal Hydraulic Analysis of the Encapsulated Nuclear Heat Source," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 4.6 E. Greenspan, A. Barak, D. Barnes, J. Buongiorno, N. W. Brown, D. Carelli, S.H. Chang, L. Conway, M. Dzodzo, E. Feldman, S.G. Hong, I.S. Hwang, S.H. Jeong, Y.I. Kim, K.G. Lee, M. Milosevic', B.G. Park, D. Saphier, T. Sofu, J. J. Sienicki, and D C. Wade, "The Encapsulated Nuclear Heat-Source (ENHS) Reactor," Summary Report of 2nd year of NERI Project No. 990154 Feasibility Study, University of California-Berkeley Nuclear Engineering Internal Report UCBNE-4234.
- 4.7 J.J. Sienicki and D.C. Wade, "Thermal Hydraulic Analysis of the Encapsulated Nuclear Heat Source," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 4.8 M. Dzodzo, E. Greenspan and L. Conway, "Nested Channel Intermediate Heat Exchanger Design Concept for the Encapsulated Nuclear Heat Source," Trans. American Nuclear Society, **85**, 80-81, Reno, NV, Nov. 2001.
- 4.9 M. Dzodzo and D. Cuckovic-Dzodzo, "Natural Convection in Enclosures Containing Lead-Bismuth and Lead," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.
- 4.10 Y. Kato, T. Nitawaki and Y. Yoshizawa, "A Carbon Dioxide Cooled Direct Cycle Fast Reactors," Proc. of GLOBAL-2001, Paris, France, September 2001. Paper 028.

5. MEETING THE GOALS FOR GENERATION-VI REACTORS

5.1 Sustainable energy

5.1.1 High Fuel Utilization

The ENHS can be fueled with either Pu($\sim 12^w/o$)-U or enriched ($\sim 13^w/o$) uranium^{5.1, 5.2}. The fuel material considered at present is metallic alloy with Zr ($10^w/o$) although nitride fuel may be preferable. The ENHS reactor core is designed so that the fissile fuel content is kept nearly constant throughout the 20 years of core life. As the initial fuel composition is uniform and there are no blanket assemblies, the end-of-life fissile content will be close to uniform across the core. As a result it is possible to recycle the fuel using a dry process that cannot separate Pu or other actinides from the heavy metal. The particular process we have in mind is the AIROX dry process^{5.4}. Only volatile and semi-volatile fission products and trace amounts of actinides will be removed and some make-up fuel added. The outcome will be an overall utilization of several dozen percent of the energy content of the fuel fed into the ENHS; close to 50 times higher than that of the once-through fuel cycle presently used with LWRs. The peak discharge burnup is approximately 105 GWd/tHM; it is constrained by the radiation damage to the clad.

5.1.2 Low Waste

The above described fuel cycle will result in an order-of-magnitude reduction of the amount of actinides that get to the waste stream, per unit energy generated. Moreover, one of the fuel cycle options considered is to use properly processed spent fuel from LWR as the feed material for the ENHS. The idea is to use a proliferation resistant dry process, such as an electro-refining process, to remove from the LWR spent fuel only part of the uranium and part or all of the fission-products^{5.2}. The net result will be a dramatic alleviation of the nuclear waste problem from LWRs combined with a low-waste generating nuclear energy system (See Section 5.3.8).

5.2 High Safety and Reliability

5.2.1 High Safety

There are no pumps, valves or pipes in both the primary and secondary coolant systems of the ENHS reactor. Moreover, as the ENHS reactor pool is located inside a silo and the coolant level is nearly 15 m above the fuel level, it is inconceivable to have a LOCA or a LOFA. In addition, the ENHS reactor cannot have a control rod ejection accident or any other accident that results in more than \$1 reactivity insertion. Accident analysis done so far revealed that the deviation of the system from nominal operating conditions is very slow as compared to other LMR or LWR reactors. All the postulated events were mitigated by naturally occurring phenomena – negative reactivity feedback and natural circulation. No operator intervention is necessary. The RVACS system is sufficient to remove the decay heat and keep all plant components at below damage level temperatures.

5.2.2 High Reliability

The ENHS reactor is expected to offer very high reliability due to the following features: no pumps or valves in the primary and secondary systems, nearly zero burnup reactivity swing, very long core life, low power density. In fact, after the reactor is brought to nominal power, it can operate autonomously. Once a year or so there may be a need for a slight adjustment of the height of the peripheral absorber segments so as to compensate for small reactivity changes with burnup. The reactor can follow the load autonomously over a wide range of power levels. Also, being disposable and having relatively few components, the ENHS module is not expected to require much maintenance. There is easy access to inspect and maintain the steam generators.

5.2.3 Good Neighbor

It is expected that an ENHS nuclear power plant will require no emergency planning zone outside its fence. This is due to the reactor safety-features discussed above.

5.3 High Proliferation Resistance

The ENHS offers a unique combination of technological barriers and material barriers that, along with adequate institutional barriers can make the nuclear energy system extremely proliferation resistant.

5.3.1 Technological Barrier 1: No Access to Fuel

The combination of long-life core and nearly constant k_{eff} makes it possible to eliminate on-site refueling altogether. The ENHS module is designed to be factory fueled and to be disposed or recycled after ~20 EFPY of operation. It is envisioned that the ENHS factories and recycling facilities would incorporate stringent international safeguards and security controls. The fuel is to be sealed inside the ENHS module from the time the module leaves the factory until the spent module is returned to the waste disposal site or to a regional or international recycling center. It is envisioned that the ENHS power plants will not even have on-site hardware for refueling.

The absence of on-site fuel handling, combined with the small number of components inside the ENHS module, enables designing the module in a way that will make it unnecessary to ever open this module. The components inside the module are robust and will be designed to operate reliably for 20 EFPY without a need to access them. Thus we envision the ENHS module to have the fuel sealed inside a welded enclosure that could serve as a shipping and disposal container or would only be recycled at a secure, internationally controlled, regional recycling center.

Even if individuals in the client country were to break into the top cover of the ENHS module, they would not be able to easily remove the fuel. The fuel is effectively secured in the core support structure since it is not designed for replacement on site. When outside the factory and outside of the secondary coolant pool, the fuel is imbedded in solid Pb-Bi (or Pb).

It is practically impossible to steal the ENHS module with the fuel: The module is ~20 m long and ~3 m in diameter and weighs ~ 300 tons. The fact that the fuel is shipped imbedded in solid Pb-Bi or Pb makes it even more difficult to steal the fuel; it will take long time and special

mechanical and heating equipment to destructively separate the fuel from the module. Any attempt to break into the module could be immediately detected by the IAEA by using automatically operating monitors that are connected to wireless transmission devices. The long time it will take even a trained team of people to break into the fuel will give the international community ample time to take measures to prevent diversion of the ENHS fuel.

5.3.2 Technological Barrier 2: No Access to Neutrons

There is no access to the neutrons in the client country. There are very few components inside the module; none requires maintenance. Hence, no need to open the module in the client country for operation and maintenance. Moreover, the module is sealed in the factory so that efforts to open it in the client country can be detected almost immediately. Even if there were ways to open the module in the client country undetected, it would be physically impossible to insert fertile material for irradiation into or in the vicinity of the core.

This is because the fuel rods fill all the space inside the core barrel and there is no way to remove fuel from the top of the core. There is no blanket fuel in or around the core as is common in designs of sodium-cooled fast reactors. The current of neutrons in the pool in the vicinity of the ENHS module vessel is too low to be useful for any strategic material production application.

5.3.3 Technological Barrier 3: No Facilities Suitable for Military Applications

Installing and operating ENHS reactors will not require the country to obtain sensitive technologies that can be used for clandestine production of strategic nuclear materials. Specifically, no fuel fabrication or handling facilities and no fuel reprocessing capability are needed in the client country.

In summary, the ENHS can provide a sequence of effective technological barriers to diversion of nuclear fuel. To the best of our knowledge, no other reactor concept can match the ENHS in this respect.

5.3.4 Material Barrier 1: Isotopics of the Fuel

It is technically possible to fuel the ENHS with uranium enriched to approximately 13% ^{235}U . This fuel has little value for development of nuclear explosives. There are, however, a couple of concerns related to proliferation with the use of low enriched uranium fuel: (1) The resulting spent fuel will contain significant amounts of plutonium. The plutonium can be of weapons grade if the operation of the ENHS is stopped after a short period of time. (2) There will be a continuous buildup of the global plutonium inventory.

On the other hand, fuelling with LWR spent fuel plutonium has the following proliferation-resistance attributes: (1) The concentration of ^{238}Pu keeps building up fast. (2) The Pu is mixed with many minor-actinides (MA); it is never separated from the uranium and MA. The concentration of the MA keeps building up. (3) The in-core inventory of Pu increases very slightly; there is essentially no accumulation of Pu. (4) The loaded fuel has a significant radiation barrier, due to the MA.

5.3.5 Material Barrier 2: Radiation Barrier

A unique feature of the ENHS is the possibility to seed in the core a very effective radiation barrier other than the fuel. This is because the fuel is loaded in the factory and is shipped to the site imbedded in Pb-Bi or Pb. Thus, after loading the fuel into the ENHS vessel and before pouring in the Pb-Bi or Pb it is possible to insert into the core or its close vicinity strong sources of gamma rays. After filling the module with Pb-Bi or Pb up to the top of the fission gas plenum, the radiation level outside of the ENHS module will be very low; the radiation sources can be designed so that they will not interfere with the shipment and installation of the ENHS module. However, access to the fuel will be deterred by the potential exposure to the high radiation fields of the seeded radiation sources when trying to remove the Pb-Bi or Pb.

5.3.6 Desirable Institutional Barriers

For the unique technological and material barriers offered by the ENHS to be most effective, it is desirable to institute certain international agreements. For example: (1) The establishment of well safeguarded nuclear recycling centers as well as of a limited number of fuel fabrication and ENHS module fabrication facilities. (2) A restriction on the construction of enrichment, fuel reprocessing and fuel handling facilities. ENHS reactors may facilitate instituting such agreements as they provide very long-term (~20 yr) security of energy supply without the country having an independent fuel industry.

Ideally, the module recycling and fuel fabrication facilities are to be co-located with the ENHS module fabrication plant as illustrated, schematically, in Figure 5.1. The fuel feed to such a site could be spent fuel from LWRs and, when needed, natural or depleted uranium. The only fuel sent out from these sites will be embedded in ENHS modules. The radioactive waste from the fuel processing and from the used ENHS modules will be shipped for disposal in regional or international waste centers. Alternatively, fuel reprocessing could be done in the regional or international waste centers from which the fuel will be sent to the ENHS fabrication facility, as illustrated in Figure 5.2.

5.3.7 Feasibility of Proliferation Resistant Multi-Recycling

The ENHS core is designed to be Fuel Self-Sustaining or Fuel Self-Sufficient (FSS); the fuel k_{∞} is nearly independent of the core location and burnup. This feature is obtained by designing the core to maintain the fissile-fuel content nearly constant with burnup; that is, to have a breeding ratio that is few percent above unity. The slight buildup of fissile fuel with burnup is used to compensate for the negative reactivity effect of the fission products that build up during the cycle. The fuel is discharged from the FSS core when reaching its radiation damage limit. What is necessary for reusing this discharged fuel is to remove all or part of the fission products, mix the heavy metal (HM) with makeup fuel and re-fabricate fuel elements. The make-up fuel, approximately 5% of the fuel loading, can be either depleted uranium or natural uranium or spent fuel from LWR. The reactivity worth (or k_{∞}) of the resulting fuel is almost the same as the reactivity worth of the previous fuel loading and of the beginning-of-life fuel loading. This is because the reactivity released by the removal of fission products is used to compensate for the low reactivity worth of the makeup fuel.

Figure 5.1 A schematic illustration of an ENHS-based nuclear energy systems in which the ENHS module fabrication and fuel processing and handling facilities are co-located.

Figure 5.2 A schematic illustration of an ENHS-based nuclear energy systems in which the ENHS module fabrication and recycling facilities are separately located.

The net result is that such a FSS can recycle its actinides many times in a highly proliferation resistant manner – without partitioning of actinides and without using uranium enrichment services. Relatively simple and proliferation resistant processes could be used for the extraction of fission products.

One of the processes proposed^{5.2, 5.3} for consideration for the extraction of fission products is an AIROX like process^{5.4-5.6} that removes only volatile (^3H , I, Xe, Kr, ^{14}C) and semi-volatile (Ru-90%; Cd-75%; Te-75%) fission products (FP) from the discharged fuel such as the one being developed in the DUPIC program^{5.6} in Korea. The AIROX process has also been considered for a single recycling in LWRs^{5.4, 5.5}. The AIROX process, originally developed in the USA, is the most proliferation resistant FP removal process since (a) it leaves significant concentration of FP, thus making the fuel more radioactive, and (b) it can not partition actinides, since the separation of the FP is based on volatility induced by heating rather than by chemical or electrochemical means. For this reason the US State Department gives its blessing to the DUPIC Program in Korea that is aiming at using the AIROX process to recycle LWR spent fuel into CANDU reactors. As these water reactors do not breed, the fuel will be recycled only one time and the added burnup is limited to the order of 50%. In the case of ENHS like reactors, on the other hand, the fuel could be recycled many times leading to a dramatic reduction in the volume and toxicity of the high-level waste along with a dramatic increase in the fuel utilization.

There remains the question of how to come by the first core loading without separation of Pu. One possibility^{5.2, 5.3} is to use LWR spent fuel as the feed material and remove from it only part of the uranium and part or all of the FP. For example, if the LWR SF contains 1% Pu and MA, it is necessary to remove approximately 90% of the uranium to make a fuel with 11 to 12 % Pu and MA. This could hopefully be done using a highly proliferation-resistant process, possibly a combination of an AIROX process and a fluoride volatilization process or a simplified version of the UREX process. Another feed option that could be considered is the spent fuel from MOX fueled LWR. The TRU content in such spent fuel can be approximately half of that needed for ENHS like reactor. Hence, only ~50% of the uranium need be extracted along with FP to make fuel for ENHS like reactor. The latter is likely to offer a more economical fuel cycle.

The resulting non-proliferating multi-recycle fuel cycle features a high fuel utilization – an order of magnitude higher than in LWR, along with a great reduction in the inventory of the high level waste that need be disposed of in an underground repository. An important question is whether it is possible to wait with the recycling of the TRU, and especially of the Pu, until FSS fast reactors are ready for deployment. Let us examine the situation in the USA as a case study. The following analysis is very crude, as it is based on approximate assumptions: (a) The total inventory of LWR spent fuel (SF) to be accumulated by 2010 is 62000 tHM. The average Pu inventory is 1 % of the HM inventory; i.e., 620 tons. (b) From 2010 to 2030 the installed capacity of operating LWR will be 100 GWe. These reactors will operate at an average capacity factor of 0.9 and will discharge fuel having an average burnup of 50 GWD/tHM. (c) The average Pu inventory in the fuel discharged from LWR SF at 50 GWD/tHM is 1.5 %. (d) From 2030 on, LWR will be replaced by FSS fast reactors at a rate of 5 GWe per year. The FSS reactors will be fueled by the LWR SF. (e) The specific power of the FSS fast reactor is 62.5 kWe/kg Pu; this is in between the design value of the BREST^{5.7} HLM reactors (50 kWe/kg Pu) and of the S-PRISM sodium cooled fast reactor^{5.8} (73 kWe/kg when using oxide fuel and 123 kWe/kg when using metallic fuel).

It is found^{5,9} that the total TRU inventory to be accumulated from LWR could provide the fuel loading for ~55 GWe of FSS reactors. This capacity is only 60% of the current LWR generating capacity in the USA. It will take only ~12 years from the initiation of deployment of the FSS fast reactors to load into their core the entire Pu inventory to be accumulated in LWR SF. If the LWR SF was to fuel ENHS reactors, it is estimated that the entire inventory of LWR spent fuel presently destined for the Yucca Mountain repository (YMR) could be converted into the fuel loading of ~ 320 ENHS reactors, a large inventory of uranium that need not be disposed of in the YMR and FP with trace amounts of actinides that will have to be disposed of in the YMR. With makeup from either depleted uranium or uranium separated from the LWR spent fuel, the fuel discharged from the 320 cores could possibly be recycled into new 320 cores for hundreds of years. This simplified analysis also indicates a potential shortage of Pu (or TRU) shortly after installation of commercial FSS reactors will be initiated. Recycling the Pu in the form of MOX fuel in existing LWR will worsen the shortage of TRU.

An energy system based on the proposed combination of FSS reactors and proliferation-resistant multi-recycling using spent fuel from LWR as the feed is environmentally attractive for three reasons: (1) It converts most of the waste from LWR into high quality fuel. (2) It generates nuclear energy without increasing the total inventory of Pu. The total Pu and MA inventory in the FSS-based energy system of a given capacity is nearly constant. Most of the Pu and MA inventory is well secured inside the reactor core. (3) It maximizes utilization of the existing uranium resources – possibly generating ~50 times more energy per unit weight of mined U than generated in an LWR operating with the once-through fuel cycle.

The only waste anticipated from the above proposed energy system consists of the following components: (1) The FP extracted in the dry process. (2) TRU and FP that cannot be recovered from the spent fuel clad material. (3) TRU and FP waste from the fuel fabrication process. (4) Coolant and structural material activation products. This waste has not yet been quantitatively assessed.

5.4 Economic Viability

5.4.1 Competitive life cycle cost

An economic analysis of ENHS NPPs has not yet been performed. Nevertheless, the life-cycle cost of the ENHS NPP is expected to be competitive with that of large size NPPs due to the following features: (1) Simpler design with fewer components. (2) No on-site refueling and refueling-hardware. (3) Standard and modular design lends itself to mass production. (4) Very short on-site assembly time (5) Very high availability – due to 20 years core life and low maintenance of the NSSS. (6) Very long plant life – as the ENHS module is disposable, the ENHS NPP life could be at least 60 years and possibly even close to 100 years. (7) High thermodynamic efficiency – possibly 42%. (8) Small staff requirement for operation and maintenance. (9) Close match between demand and supply – Installed capacity can be increased in increments of 50 MW_e. (10) It is not necessary to have long transmission lines – Generating unit could be located near the demand, as in localized distribution grid. (11) It is not necessary to have an advanced infrastructure.

5.4.2 Low Financial Risk

The ENHS NPP is of a highly modular and standard design. All the modules are fabricated and undergo quality assurance inspection at the factory. The installation of the modules is relatively simple so that the probability of cost overruns in fabrication and installation is very small. In addition, the superb safety of the ENHS (Sec. 5.2.1) implies that the integrity of the reactor will be preserved even under the most severe accidents.

5.5 References

- 5.1 H. Shimada, E. Greenspan, N. Stone and S. Wang, "Long-Life Lead-Cooled Core Fueled with Enriched Uranium," Transactions Am. Nucl. Soc., **80**, 199-200 (1999).
- 5.2 E. Greenspan, N. W. Brown, M. D. Carelli, L. Conway, M. Dzodzo, Q. Hossain, D. Saphier, J. J. Sienicki, D. C. Wade, "The Encapsulated Nuclear Heat Source Reactor for Proliferation-Resistant Nuclear Energy," Proc. of GLOBAL-2001, Paris, France, September 2001.
- 5.3 E. Greenspan et al., "Multi-Recycling of Spent Fuel with Low Proliferation Risk," Proc. Int. Conf. on Emerging Nuclear Energy Systems, ICENES-98, Herzelia, Israel, 28 June 1998.
- 5.4 H. Feinroth et al., "Dry Recycle of Light Water Reactor Spent Fuel," Proc. of 8th International Conference on Nuclear Engineering: ICONE-7, Paper 7485, Tokyo, Japan, 19-23 April 1999.
- 5.5 H. Feinroth, J. Guon, D. Majumdar, "An Overview of the AIROX Process and Its Potential for Nuclear Fuel Recycle," Proc. of the Global International Fuel Cycle Conference, Seattle, 1993.
- 5.6 J.S. Lee, et al., "Research and Development Program of KAERI for DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors)," also M.S. Yang et al., "Conceptual Study on the DUPIC Fuel Manufacturing Technology," Proc. Int. Conf. On Future Energy Systems: Emerging Nuclear Fuel Cycles and Waste Disposal Options, GLOBAL'93, Seattle, Washington (Sept. 1993).
- 5.7 E.O. Adamov, V.V. Orlov, A.I. Filin, V.N. Leonov, A.G. Sila-Novitski, V.S. Smirnov, and V.S. Tsikunov, "The Next Generation of Fast Reactors," Nuclear Engineering and Design, **173**, pp. 143-150 (1997).
- 5.8 C.E. Boardman, A.E. Dubberley, D.G. Carroll, M. Hui, A.W. Fanning and W. Kwant, "A Description of the S-PRISM Plant," Proceedings of ICONE-8, Baltimore, Maryland, April 2-6, 2000.
- 5.9 E. Greenspan, P. Hezlar, H. Sekimoto, G. Toshinsky and D. Wade, "New Fuel Cycle And Fuel Management Options In Heavy Liquid Metal Cooled Reactors," Proc. Advances in Nuclear Fuel Management III (ANFM 2003), Hilton Head Island, South Carolina, October 5-8, 2003.

6. SUMMARY

It was found possible to design uniform composition blanket-free cores to operate at 125 MW_{th} for 20 years with nearly zero burnup reactivity swing. The design domain for such cores has been identified; the design variables are the core height, lattice pitch-to-diameter ratio and fissile fuel contents. It was found possible to design such cores using either Pu-U fuel having ~11-12 weight % Pu or uranium enriched to ~13^{w/o}, both in alloy with Zr (10^{w/o}). A wide variety of Pu compositions as well as a mix of Pu and minor actinides are acceptable. The core life is limited by radiation damage to the clad. A fission gas plenum volume that is equal to the fuel volume is required to accommodate the fission gas pressure buildup and maintain the clad integrity. The core designs are simple: they have uniform composition, no blanket or reflector assemblies and a single central safety assembly. Six axially movable absorber assemblies surrounding the core are used for reactivity control. They are lifted almost completely above the fuel level to bring the core to full power. A combination of tungsten and B₄C is used for the absorbing material. The specific density of the control assembly is larger than that of Pb-Bi so that scrambling can be done by gravity. The maximum change in k_{eff} throughout 20 years of full power operation is only 0.2% or about \$0.5. After the reactor is brought to full power it may be necessary to adjust the peripheral absorber elevation once every 5 years or so. The core power shape and reactivity coefficients virtually remain constant throughout life.

It was found feasible to design ENHS modules to deliver 125 MW_{th} from the primary to the secondary coolant through a 4 mm thick intermediate heat exchanger with maximum primary coolant to maximum secondary coolant temperature drop of 50°C and with 100 % natural circulation. The required reactor vessel height is close to 20 meters. In fact, good engineering compatibility was found between the requirements for 100 percent natural circulation and a primary-to-secondary temperature drop of 50°C. It was also found that the ENHS reactor can maintain 100 percent natural circulation over the entire range from above nominal power level down to the decay heat power level.

A significantly more compact and lower weight ENHS module has been conceived and partially analyzed. It uses cover-gas lift-pumps to enhance coolant circulation. The gas circulators are located outside of the reactor vessel. The use of a lift pump reduces the ENHS module length to approximately 10 m – about half the length and volume required for the reference (100% natural circulation at full power) design. The power level of the ENHS module with a lift pump can significantly exceed 125 MW_{th}. As a result of the reduced volume and increased power level the power per unit module volume of the lift-pump ENHS is estimated to be approximately three times that of the reference, 100% natural circulation ENHS. However, a more thorough evaluation of the lift-pump performance and a safety analysis are required before the feasibility of the lift-pump version of the ENHS could reliably be assessed.

An intermediate heat exchanger (IHx) that is integrated within the ENHS module vessel walls and possessing substantially identical heat transfer characteristics to those of the original corrugated confinement wall concept, has been conceived and analyzed. It consists of rectangular channels circumscribing the primary coolant riser in the space between the inner structural wall and the outer structural wall. This integrated IHx was found able to withstand the loads that are expected during transportation, installation and operation. Relative to circular tube IHx the

rectangular channel IHX features close to an order-of-magnitude smaller number of channels and smaller friction losses due to elimination of grid spacers. Another innovative IHX design has been conceived; it consists of nested tube bundles. Advantages of the proposed nested channels IHX concept include the following: it can be made from off-the-shelf components; it has high rigidity against buckling; and it is easier to fabricate.

A strategy was developed for fueling the ENHS module in factory and for transporting it to the nuclear power plant site as a weld-sealed unit. To avoid fuel damage during transport and module installation, the fuel is embedded in solidified Pb-Bi that fills the vessel up to above the fuel rods. The total weight of the reference ENHS module for transportation is estimated to be about 300 tons. This is approximately half the weight of steam generators for large pressurized water reactors (PWRs) that have been transported from the factory to nuclear power plant sites. Dimensions of the reference ENHS module are somewhat smaller than dimensions of a large PWR steam generator. It appears feasible to factory manufacture and fuel the ENHS module and ship it to the site using available transportation equipment.

At the site the ENHS module will be inserted into the pre-heated secondary Pb-Bi pool vessel. Hot Pb-Bi is then to be pumped into the ENHS vessel through a pipe. This hot Pb-Bi, along with the hot Pb-Bi in the pool, will melt the solid Pb-Bi at the lower part of the vessel. It was found feasible to melt the primary coolant and bring it to the reactor startup temperature of 350°C within approximately two days of its insertion to the pool without having to use special external heaters in the module. After 20 effective full power years the ENHS will be removed from the reactor pool. Before removal, the coolant will be extracted from the upper part of the ENHS module above the level of the fuel rods. The Pb-Bi coolant will then be replaced by Pb. It appears feasible to solidify the Pb one week after shutdown by forced cooling the cylindrical outer boundary of the core and the surface of the shutdown assembly channel located at the center of the core. If cooling is provided only at the outer boundary, then the Pb can be solidified after only 22 days. The module will then be taken out of the pool and moved to storage vault on site until the decay heat drops to a level such that passive cooling will permit the Pb to stay solid. The ENHS with the solidified Pb will then be prepared as a licensed shipping package and transported to a recycling facility. The cooling time required for solidifying the coolant in the spent module may be longer than one year. Innovative passive means for significantly shortening the required cooling period need be explored.

It appears possible to ship the spent ENHS module intact by placing it inside a specially designed shipping cask. It is likely that the spent ENHS transport cask could be certified by both the US NRC and by the IAEA for domestic and for international transport. This approach is relatively heavy (~1000 MT) and relatively costly (~\$5M for cask) but is do-able. The handling of the lift-pump version of the ENHS will be easier and less expensive. More thorough assessment of the feasibility of transporting the ENHS module unopened is required.

Due to its negative temperature feedback and to cooling based on natural circulation, the ENHS reactor has a load-following capability over a wide range of power. The reactor power level will adjust itself to the power demand without operator intervention. This autonomous load following capability has been demonstrated in numerical simulation. The operators need only perform a startup and a periodic surveillance function. Both functions may potentially be accomplished

remotely from a centralized facility that services many units.

A preliminary safety analysis was performed for the reference ENHS design. The accidents considered so far include a startup accident, a loss of heat sink accident, and a steam line break without scram accident. It was found that under all accident conditions considered, fuel and clad temperatures will remain significantly below the safety limits and the integrity of all systems will be maintained. Interestingly, in a loss of heat sink without scram accident – the worst plausible accident identified, the peak fuel temperature will not exceed its steady state operating conditions. The decay heat is passively removed by the reactor vessel air-cooling system (RVACS). Contributing to the exceptional safety features of the ENHS are the lack of pumps and valves in the primary and intermediate cooling systems, use of full natural circulation, high heat capacity, low power density, more than 1000°C margin between the coolant operating and boiling temperatures and availability of very small excess reactivity for postulated positive reactivity insertion accidents.

Steam generators have been designed to meet several unique requirements that are dictated by the ENHS reactor layout: (1) Effective utilization of the pool volume surrounding the ENHS module. (2) Minimization of the friction losses enabling 100% natural circulation of the intermediate coolant. (3) Elimination of mechanical connections to the ENHS module. (4) Minimization of water flow rate into the intermediate coolant pool in case of a steam generator tube breach, or failure of other water-containing component. (5) Accommodation of a large thermal expansion. (6) Ease of inspection and maintenance. (7) Modular design that is easy to install and replace. Two-dimensional flow simulation showed that at nominal operating conditions only 3% of the intermediate coolant flow bypasses the steam generators. This small bypass has a negligible effect on the ENHS energy balance. The 2-D flow simulation also revealed that it is desirable to install a vertical flow partition in the ENHS pool, extending from below the steam generators outlet to close to the pool bottom. This partition eliminates stagnation of the coolant at the bottom of the reactor pool. Another novel plate type steam generator has recently been conceived. It may allow a more compact reactor pool design, or the integration of the steam generator with the IHX.

A Rankine steam cycle was designed to match the heat source characteristics of the ENHS steam generators. The gross thermodynamic efficiency for converting the ENHS heat to electricity was calculated to be 40%; the corresponding net efficiency is close to 38%. These efficiency values correspond to a simple energy conversion system that has no reheat and a moderate steam pressure of 120 bars. The main effect of the reheat is a reduction of the undesirable low-pressure steam moisture. Drying the low-pressure steam has the marginal effect of a 1% power recovery, and should extend the low-pressure turbine lifetime. Increasing the steam pressure to 180 bars can increase the net efficiency to 40%. Even higher efficiency can be attained using supercritical steam. However, the steam moisture level increases with steam pressure increase, unless reheaters are used. An energy conversion system using CO₂ working fluid appears to make a better match with the ENHS than a steam driven energy conversion system. However, such a CO₂ system has not been designed, as yet, for the ENHS.

The neutronic and thermal hydraulic characteristics of Pb-Bi and Pb are very similar. The same ENHS reactor design can, in principle, operate with either one of those coolants. ENHS can also

be designed to provide nearly zero burnup reactivity swing and 100% natural circulation using sodium as the primary and secondary coolants. The sodium-cooled designs feature more compact, smaller diameter cores, but larger core peak to average power ratios, and significantly larger temperature rise across the core. Hence, smaller thermal margins and lower energy conversion efficiency result from the sodium-cooled designs. The safety and performance advantages with Pb-Bi or Pb make these superior to sodium for the ENHS reactor coolants.

The study found that the ENHS reactor concept can meet very well all of the goals set for Generation IV reactors that have been examined including the following: (1) Highly sustainable energy supply; the ENHS maintains its fissile fuel inventory constant; all that is needed for reusing the fuel discharged from one module is to remove fission products, add depleted uranium and re-fabricate fuel rods. (2) Low waste; the TRU can be recycled many times. (3) Extremely high level of proliferation resistance. There is no access to the fuel in the host country and no access to neutrons. The host country gets energy security with no need to invest in fuel cycle technologies. (4) Superb safety and reliability. As a consequence, there is no need for emergency planning zone outside of ENHS nuclear power plant fence. (5) Very low risk to capital. This is due to relatively low cost per module, standard design with factory assembly line fabrication, short construction time, superb safety and large tolerance to human errors. However, the economic viability of the ENHS has not been examined so far; it was out of the scope of work. Nevertheless, general considerations and very preliminary examination of the ENHS fuel cycle cost indicate that there is a good probability that the ENHS will be economically viable despite of its small unit size.

The study found that the ENHS reactor concept can well satisfy all of the goals set for Generation IV reactors that have been examined, including the following: (1) Highly sustainable energy supply – the ENHS maintains a nearly constant fissile fuel inventory; all that is needed for reusing the fuel discharged from one module is the removal of fission products, the addition of depleted uranium and the re-fabrication of the fuel rods. (2) Low waste – the TRU can be recycled many times. (3) Extremely high level of proliferation resistance – there is no access to the fuel in the host country and no access to neutrons. The host country gets energy security without having to invest in resident fuel cycle technologies. (4) Superb safety and reliability. As a consequence, there is no need for emergency planning zone outside of ENHS nuclear power plant fence. (5) Very low risk to capital. This is due to relatively low cost per module, standard design with factory assembly line fabrication, short construction time, superb safety and large tolerance to human errors. However, the economic viability of the ENHS has not been examined so far; it was out of the scope of work. Nevertheless, general considerations and very preliminary examination of the ENHS fuel cycle cost indicate that there is a good probability that the ENHS will be economically viable despite of its small unit size.

In summary, the Encapsulated Nuclear Heat Source (ENHS) concept appears feasible and offers new interesting possibilities for mankind to benefit from nuclear energy. More thorough R&D needs to be undertaken before an ENHS demonstration plant construction can be justified. The most important issues that need more thorough study are the feasibility of handling and shipping the spent module without having to open it on site and the economic viability of the ENHS.

APPENDIX A: REFERENCE THERMO-PHYSICAL PROPERTIES OF Pb-Bi AND Pb

A.1 Introduction

It was noticed during FY01 that different researchers in our team used different published correlations for the thermo-physical properties of Pb and Pb-Bi. This led to some inconsistencies among internal assessments. The purpose of this undertaking was to establish a reference set of correlations for the thermo-physical properties of Pb-Bi and Pb to be used in standardizing the assessments for the ENHS project.

Guiding criteria for the determination of reference correlations were the following: The correlation is to err on the conservative side and to be applicable over wide temperature ranges. Where new experimental data and correlations for Pb-Bi and Pb properties are available, they receive a higher weight.

The correlations chosen for the reference set are not necessarily superior to other correlations in the literature. When significant variation in physical properties exists, sensitivity studies are recommended to quantify the uncertainties in design and performance parameters due to the data uncertainties. To provide a database for the sensitivity analysis, all the experimental data available to us is presented in the figures below. Data scatter around the reference correlation allows the definition of uncertainty bands.

A.2 Correlations for Lead-Bismuth thermo-physical properties

A.2.1 Density

The polynomial expression for density derived by Been et al.^{A.1} is adopted:

$$\rho = 10740 - 1.38 * t \text{ [kg/m}^3\text{]}, \quad (\text{A.1})$$

where t is in °C. The expression (A.1) is valid for the range of temperatures 200°C to 1000°C. The comparison with other data presented in Lyon and Poppendiek^{A.2}, Kutateladze et al.^{A.3} and Miller^{A.4} is given in Figure A.1.

A.2.2 Thermal expansion coefficient

Expression for thermal expansion coefficient of lead-bismuth (A.2) is derived using the density correlation of Eq. A.1:

$$\beta = \frac{1}{7782.609 - t} \text{ [1/K]} \quad (\text{A.2})$$

where t is in °C (see Figure A.2). The expression (A.2) is valid for the range of temperatures 200°C to 1000°C.

Figure A.1 Correlation for Pb-Bi density and comparison with experimental data

Figure A.2 Thermal expansion coefficient of Pb-Bi

A.2.3 Dynamic viscosity

The expression for coefficient of dynamic viscosity of lead-bismuth is that recommended by J. K. Fink^{A.5} (available at www.insc.anl.gov):

$$\mu = 4.9 \cdot 10^{-4} \cdot \exp\left(\frac{760.1}{T}\right) \text{ [kg/m s]} \quad (\text{A.3})$$

where T is in K. It is valid for 126°C to 900°C range. The comparison with other data (Lyon and Poppendiek^{A.2}, Miller^{A.4}, Kutateladze et al.^{A.3} and International Critical Tables of Numerical Data, Physics, Chemistry and Technology^{A.6}) is presented in Figure A.3.

Figure A.3 Correlation for Pb-Bi dynamic viscosity and comparison with experimental data

A.2.4 Thermal conductivity

Data from Lyon and Poppendiek^{A.2}, Miller^{A.4}, Nikol'skii et al.^{A.7}, (Kutateladze et al.^{A.3}), Powell and Childs^{A.8}, and Brown^{A.9} is used for the thermal conductivity:

$$k = 12.4119 - 3.37233 \cdot 10^{-2} \cdot T + 7.75802 \cdot 10^{-5} \cdot T^2 - 3.94661 \cdot 10^{-8} \cdot T^3 \text{ [W/m K]}, \quad (\text{A.4})$$

where T is in Kelvin. The temperature range is from 148°C to 700°C. The comparison with available data is presented in Figure A.4.

Figure A.4 Correlation for Pb-Bi thermal conductivity and comparison with experimental data

A.2.5 Specific heat

The specific heat of lead-bismuth is based on Lyon and Poppendiek^{A.2}, Miller^{A.4}, and Nikol'skii et al.^{A.7}, (Kutateladze et al.^{A.3}). This data is $C_p = 146.51$ [J/kg K] for the temperature range 130°C to 700°C (see Figure A.5).

Figure A.5 Correlation for Pb-Bi specific heat and comparison with experimental data

A.3 Correlations for Lead thermo-physical properties

A.3.1 Density

Polynomial expression for Pb density derived by Saar and Ruppertsberg^{A.10} and recommended by Sienicki^{A.11} is adopted:

$$\rho = 10660 - 1.35 \cdot (T - 600) \text{ [kg/m}^3\text{]}, \quad (\text{A.5})$$

where T is in K. The expression (A.5) is valid for the range of temperatures 328°C to 800°C. The comparison of the correlation with experimental data in Lyon and Poppendiek^{A.2} and Kutateladze et al.^{A.3} is presented in Figure A.6.

Figure A.6 The correlation for Pb density and comparison with experimental data

A.3.2 Thermal expansion coefficient

Expression for thermal expansion coefficient of lead is derived from Eq. A.5:

$$b = \frac{1}{8496.3 - T} \text{ [1/K]} \quad (\text{A.6})$$

where T is in K (see Figure A.7). The expression (A.6) is valid for the range of temperatures 328°C to 800°C. Comparison with data calculated according to the Pb density presented in Lyon and Poppendiek^{A.2} is presented in Figure A.7.

Figure A.7 Thermal expansion coefficient for Pb

A.3.3 Dynamic viscosity

The expression for coefficient of dynamic viscosity of lead is obtained from the data base of Lyon and Poppendiek^{A.2} and Touloukian et al.^{A.12} (see Figure A.8). For the higher range of temperatures (700°C < t < 981°C) data generated by Ofte and Wittenberg^{A.13} is used. The recommended correlation is:

$$m = 6.57424 * 10^{-3} - 1.6840 * 10^{-5} * t + 1.8148 * 10^{-8} * t^2 - 6.7514 * 10^{-12} * t^3 \text{ [kg/m s]} \quad (\text{A.7})$$

It is valid for the temperatures range from 329°C to 981°C and t is in °C. It is worth to emphasize that viscosity is sensitive to liquid and solid impurities. Correction of the expression for viscosity would be necessary if the content and ratio of additives is known.

A.3.4 Thermal conductivity

Data from Lyon and Poppendiek^{A.2} and data sets number 10 and 135 from Touloukian et al.^{A.14} (Figure A.9) are used for generating the thermal conductivity correlation:

$$k = 31.5824 - 4.6967 * 10^{-2} * T + 4.5425 * 10^{-5} * T^2 - 1.4410 * 10^{-8} * T^3 \text{ [W/m K]}, \quad (\text{A.8})$$

where T is in degrees Kelvin. The temperature range is from 327.44°C to 1081.84°C. Eq. A.8 produces values of the thermal conductivity between 15 and 16 W/m K which is on the conservative side for the heat transfer calculations. Data sets in Touloukian et al.^{A.14} indicate that values can approach 24 W/m K.

Figure A.8 Correlation for Pb dynamic viscosity and comparison with experimental data

Figure A.9 Correlation for Pb thermal conductivity and comparison with experimental data

The correlation developed by Mills et al.^{A.15} is recommended by Fink^{A.16} and applied by Cahalan^{A.17} (see Figure A.9). It represents a compromise between the low and high values of experimentally obtained Pb thermal conductivities. However, in the range of higher temperatures the correlation is away from the both sets of data. On the contrary, the correlation recommended in Touloukian et al.^{A.14} is favoring the higher values of Pb thermal conductivities.

A.3.5 Specific heat

The equation for specific heat of lead recommended by J. K. Fink^{A.18} (available at www.insc.anl.govA.19) is adopted:

$$C_p = 175.1 - 4.961 \times 10^{-2} * T - 1.524 \times 10^6 * T^{-2} + 1.985 \times 10^{-5} * T^2 - 2.0994 \times 10^{-9} * T^3, \quad [\text{J/kg K}] \quad (\text{A.9})$$

where T is in Kelvin. This data is for the temperature range 600.7 K to 3600 K. A comparison with data presented by Lyon and Poppendiek^{A.2} and Touloukian et al.^{A.20} is shown in Figure A.10.

Figure A.10 Correlation for Pb specific heat and comparison with experimental data

A.4 Conclusions

Correlations for the reference thermo-physical properties of Pb-Bi and Pb are presented. Some of them are adopted from available publications while others are developed in this project. The derived correlations are on the conservative side and/or cover a wider range of temperatures covered by published correlations. For some properties, such as mass densities and Pb thermal conductivity, there is a large discrepancy among various published data sets. It is recommended the sensitivity analyses be used to quantify the effects of these uncertainties on the system design and performance.

A.6 References

- A.1 S. A. Been, H. S. Edwards, C. E. Teeter, Jr., V. P. Calkins, "The Densities of Liquids at Elevated Temperatures: I. The Densities of Lead, Bismuth, Lead-Bismuth Eutectic, and Lithium in the Range Melting Point to 1000°C (1832°F)," NEPA 1585, NEPA Division, Oak Ridge, Tennessee, September 7, 1950
- A.2 R. N. Lyon and H. F. Poppendiek, "Liquid-Metal Heat Transfer," Chapter 5 in Liquid-Metals Handbook, Editor in Chief, R. N. Lyon, "Liquid Metals Handbook," 2d ed. (revised), Atomic Energy Commission, Department of the Navy, Washington, D.C., pp. 184 – 212, 1954.
- A.3 S. S. Kutateladze, V. M. Borishanskii, I. I. Novikov and O. S. Fedynskii, "Liquid-Metal Heat Transfer Media," Supplement No. 2 of the Soviet Journal of Atomic Energy Atomnaia Energiia, Atomic Press, Moscow, 1958, Translated from Russian, Consultants Bureau, Inc., New York, 1959.
- A.4 R. R. Miller, "Physical Properties of Liquid Metals," Chapter 2 in Liquid-Metals Handbook, Editor in Chief, R. N. Lyon, "Liquid Metals Handbook," 2d ed. (revised), Atomic Energy Commission, Department of the Navy, Washington, D.C., pp. 38 – 49, 1954.
- A.5 J. K. Fink, "Pb-Bi Viscosity, Preliminary Recommendation," available at www.insc.anl.gov
- A.6 International Critical Tables of Numerical Data, Physics, Chemistry and Technology, Volume 5
- A.7 N. A. Nikol'skii, N. A. Kalakutskaya, I. M. Pchelkin, T. V. Klassen and A. Vel'tishcheva, "Thermal and Physical Properties of Molten Metals and Alloys," AEC-tr-4511, Problems of Heat Transfer, M. A. Mikheev, Ed., Academy of Sciences SSSR, Moskva, pp. 1-38, 1959. (translated into English as AEC-tr-4511).
- A.8 R. L. Powell and G. E. Childs, "American Institute of Physics Handbook," p4-142.
- A.9 W. C. Brown, "Physical Review," vol. 22, p. 171, 192A.
- A.10 J. Saar and H. Ruppertsberg, "Calculation of $C_p(T)$ for Liquid Li/Pb Alloys from Experimental $r(T)$ and $(\partial P/\partial T)_s$ Data," Journal of Physics F Metal Physics, Vol. 17, p. 305, 1987.
- A.11 J. J. Sienicki, 2001, "Lead-Bismuth Eutectic and Lead Thermophysical Properties for Thermal Hydraulics Analyses," ANL Intra-Laboratory Memo, May 15, 2001.
- A.12 Y. S. Touloukian, S. C. Saxena and P. Hestermans, "Thermo-physical Properties of Matter. Viscosity", vol. 11, The TPRC Data Series, IFI/Plenum, New York, 1970.

- A.13 D. Ofte and L. J. Wittenberg, "Viscosity of Bismuth, Lead, and Zinc to 1000°C," Trans. Metallurgical Society of AIME 227, June, pp. 706-711, 196A.
- A.14 Y. S. Touloukian, R. W. Powell, C. Y. Ho, P. G. Klemens, "Thermo-physical Properties of Matter, Volume 1: Thermal Conductivity - Metallic Elements and Alloys", The TPRC Data Series, IFI/Plenum, New York, pp 175-191, 1970.
- A.15 K. C. Mills, C. J. Monaghan, and C. J. Keene, "Thermal Conductivities of Molten Metals: Part 1 Pure Metals, " International Metallurgical Reviews," vol. 41, pp. 209-242, 1996.
- A.16 J. K. Fink, "Results of Cursory Literature Review on Lead Thermophysical Properties," ANL Intra-Office Memo, July 7, 1998.
- A.17 J. E. Cahalan, "Molten Lead Thermophysical Properties for SASA4A/SASSYS-1," ANL Intra-Laboratory Memo, July 14, 1998.
- A.18 J. K. Fink, "Enthalpy and Heat Capacity of Lead," ANL Intra-Office Memo, June 15, 1998.
- A.19 J. K. Fink, "Enthalpy and Heat Capacity of Solid and Liquid Lead, Recommendations," available at www.insc.anl.gov.
- A.20 Y. S. Touloukian, E. H. Buyco, "Thermo-physical Properties of Matter. Specific Heat – Metallic Elements and Alloys", vol. 4, The TPRC Data Series, IFI/Plenum, New York, 1970.

APPENDIX B: FUEL CYCLE COST OF THE ENHS^a

A Preliminary Assessment

B.1 Introduction

A generic feature of long-life cores, like the ENHS core, is low specific power and the fact that the full investment in fuel is made initially rather than in steps over the ENHS lifetime. The resulting high inventory of fuel per installed kW_e, along with the long operation time of the fuel results in a high interest payment that has an adverse effect on the fuel cycle cost (FCC).

On the other hand, certain unique features of the ENHS design and fuel cycle offer cost reduction. Relative to conventional liquid metal reactor, for which we take SUPER-PRISM as the reference, the ENHS has only about half the fissile fuel concentration and approximately one fifth the number of fuel rods per HM loading. Moreover, as the ENHS maintains a nearly constant fissile fuel content (and k_{∞}) and has no blanket fuel, recycling of the ENHS fuel requires only removal of volatile fission products, addition of ~6% of depleted uranium makeup and re-fabrication. This process is expected to be significantly less expensive than conventional recycling of LMR fuel. The initial fuel loading for the ENHS can be produced from LWR spent fuel that underwent a UREX process that is to extract only part of the uranium and the fission products. This is likely to be less expensive than the conventional PUREX process that is designed to extract pure plutonium.

Nevertheless, being concerned that the ENHS fuel cycle cost may be prohibitively expensive, we embarked on a preliminary estimation of this cost for the reference ENHS reactor design along with a number of possible design variants. This Note briefly summarizes the approach taken and the preliminary findings. These preliminary findings are encouraging – the ENHS fuel cycle cost is not likely to be prohibitively expensive.

B.2 Scenarios Considered

Four ENHS fuel cycle scenarios are considered:

1. Using enriched uranium core; the required enrichment is 12.7 w/o^{B.1}. The feed material is natural uranium.
2. Using standard once through ALWR spent fuel to provide the Pu and minor actinides (MA) for the ENHS core. The transuranium (TRU) composition assumed as feed is that of the fuel discharged from the “UOX open cycle” case of Table B.1 of Reference B.2. This ALWR is fueled with 4.5% enriched uranium (EU) and burns the fuel in 4 years to 55 GWd/tHM. The Pu concentration required for the ENHS is 11.4 w/o^{B.3}. The ALWR spent fuel (SF) is converted to the fuel feed for the ENHS by applying the UREX process to extract only ~90% of the uranium and fission products. Pu or other TRU isotopes are not partitioned.

^a Based on an Internal Memorandum by A. Barak and E. Greenspan of the Department of Nuclear Engineering of the University of California, Berkeley, dated May 6, 2002.

3. Using TRU from the spent fuel of a MOX fuelled ALWR. The specific spent MOX fuel composition considered is that of the “MOX mono-recycling” case of Table B.1 of Reference B.2. The TRU contents in the spent fuel is 8.82^w%. We are assuming that the spent MOX fuel undergoes a UREX process to extract the fission products (FP) and 43% of the uranium. The k_{∞} of the resulting fuel is identical to that of our reference design (Scenario 2); it contains 13.8^w% Pu (54% of which is fissile) and 1.6^w% MA.
4. Using the spent fuel from the ENHS to fuel a new ENHS module. Only FP are extract and depleted uranium is added for the makeup fuel. The k_{∞} of the resulting fuel equals that of our reference fuel (Scenario 2). The processing of the ENHS SF is assumed done using a simple process like the AIROX process that is just to extract volatile and semi-volatile FP. Alternatively, a smelt refining process used at EBR-II in the late sixties can be considered. The number of recyclings from one ENHS module to another ENHS module is, in principle, unlimited; only depleted uranium is needed to make up for the fissioned nuclei. Once the first fuel loading for the ENHS has been procured, the owner of the ENHS reactor has fuel supply security and insensitivity to increases in the market price of uranium, enrichment services or plutonium. Scenario 4 should be considered as the representative scenario for judging the ENHS fuel cycle economic viability.

In addition to inter-comparing these four scenarios, the ENHS fuel cycle cost is also compared against a reference once through ALWR addressed in Scenario 2. Its FCC is calculated using the methodology and cost data base described below. The net efficiency of this ALWR is assumed to be 33%. For curiosity we also considered an older generation LWR that is fed with uranium enriched to 3.1% and burns its fuel in 3 years to 33 GWd/tHM^{B.4}. The efficiency of this reactor is 33.8%^{B.4}.

B.3 Methodology

The costing of the nuclear FCC is done according to well-established procedures; it accounts for all the front-end and the back-end cost components. The front-end cost components include the following:

1. Feed material: U₃O₈ for Scenario 1 and spent fuel before reprocessing for Scenarios 2-4.
2. Conversion of U to UF₆ for enrichment (Scenario 1).
3. Enrichment of U (Scenario 1) or reprocessing spent fuel (Scenarios 2-4).
4. Fabrication of fuel rods.
5. Transportation to and from the given stage.
6. Interim storage before and after the given stage.

The back-end cost components include the following:

1. Interim storage before either reprocessing or final disposal.
2. Transportation to either reprocessing or final disposal.
3. Final conditioning, packaging and disposal.

These apply to all of the scenarios considered.

In order to account for the time factor, all these expenses are discounted for the time period between the actual occurrence of the expense and some reference date. This reference date is

taken arbitrarily to be the date at which the fuel utilization in the reactor begins. Consequently, the time periods are positive for the front-end cost components and are negative for the back-end cost components. All the costs are given in year 2000 US dollars.

Having a low fissile content, ALWR spent fuel is of low value as a raw material for new fuel production. Consequently it is economically better to dispose of this fuel rather than to reprocess it. The fuel disposal expenses are added as back end expenses to the ALWR FCC. In Scenario 2, where the low-grade ALWR spent fuel is processed to make a fuel load for the ENHS, the back end expenses saved from the ALWR are credited to the ENHS FCC.

In Scenario 4, on the other hand, the spent fuel from the ENHS has high enough fissile content and high enough value so that the cost of its recycling to a new ENHS is charged to the FCC of the new ENHS.

B.4 Cost Data

The fuel cycle costing guidelines adopted for the Generation IV Roadmap Project^{B.5} were used as the basis for our study. We adjusted the cost data to the specific features of the ENHS reactor and its fuel cycle as described below. Table B.1 summarizes the cost data used for the present evaluation. Entries denoted as “input” use the cost data of Reference B.5 as is. Entries denoted as “calc.” use cost data that are adjusted to the ENHS system. Following is a description of the adjustments made to get the “calc.” cost figures of Table B.1; they pertain to the “nominal” cost values:

1. Reprocessing of SF from UOX fueled ALWR by the UREX process is assumed to cost $\frac{3}{4}$ of the conventional PUREX reprocessing. Only FP and ~90% of the U are separated from the spent fuel. There is no extraction and purification of Pu or MA. Thus, if the conventional ALWR SF reprocessing cost is \$800/kg^{B.5}, the UREX reprocessed spent fuel cost is assumed to be \$600/kg.
2. Reprocessing cost of SF from MOX fueled ALWRs corresponding to Scenario 2, if done conventionally by a PUREX process, is assumed to be \$1400/kg; this is the average of the cost of PUREX reprocessing for SF of the reference UOX fuelled ALWR (\$800/kg) and for SF from >20% Pu fuelled LMR MOX (FR-MOX) fuel (\$2000/kg^{B.5}). The spent MOX from ALWRs has less than 10^w% Pu; somewhat less than the average between the ALWR and FR SF referred to above. However, for the ENHS the SF reprocessing from the MOX fuelled ALWR will be done using the UREX rather than the PUREX process. Using the above-mentioned ratio of $\frac{3}{4}$ between UREX and PUREX, the reprocessing cost for Scenario 2 is assumed to be \$1050/kg.

Table B.1 Unit Cost Components Used for ENHS Fuel Cycle Analysis
Based on Gen-IV data^{B.5}.

Sub-component		unit	Low expected cost	Nominal cost	Years (average) before putting fuel into core
Natural U	input	\$/kgU, 2% losses	20	30	2
conversion	input	\$/kgU	3	5	1.5
enrichment	input	\$/kgSWU	50	80	1.2
Depleted U storage	input	\$/kg DU/y	.30	.50	1.1
Fabrication EU	calc.	\$/kgEU [metallic]	300	420	0.6
Fabrication U-TRU	calc.	\$/kgPuU	370	600	0.6
Reprocessing: net	calc.	\$/kg spentUOX	380	600	1.1
Reprocessing: net	calc.	\$/kg spentMOX	570	1050	1.1
Reprocessing: net	calc.	\$/kg spentENHS	540	1000	1.1
- SpentUOX storage	calc.	\$/kg spentUOX	8.5 + 5/yr	8.5 + 5/yr	7 to 2 [5y storage]
- Capitalized spent UOX interim storage	calc.	\$/kg spentUOX	38	38	2
- Transport UOX	input	\$/kg spentUOX	40	50	2
HLW contents	input	kgHLW/kgSF	.05	.05	
- HLWaste Disposal	input	\$/kg spentUOX	80	200	1
Credits for alternative back-end once-thru UOX fuel cycle expenses and for recovered U					
Spent UOX storage	input	\$/kg spentUOX/yr	50 + 5/yr	50 + 5/yr	7 to -23 [30 years storage]
capitalized Spent UOX interim storage	calc.	\$/kg spentUOX	179	179	0
Transport UOX	input	\$/kg spentUOX	40	50	-23
Final burial of spent UOX	input	\$/kg spentUOX	130	300	-23
Capitalized spent UOX credit	calc.	\$/kg spentUOX	233	292	0
Credited kgU/kg spentUOX (assumed)			.91	.91	1.1

3. Reprocessing of spent fuel from the ENHS is assumed to be done using an AIROX (DUPIC) like process - mechanical decladding followed by removal of volatile and semi-volatile fission products. The cost is assumed to be half of the cost of reprocessing FR fuel (\$2000/kg^{B.5}), namely \$1000/kg.
4. The ENHS fuel fabrication cost was inferred from the fabrication costs of the GEN-IV road-mapping costing guidelines^{B.5} taking into account differences in the HM weight per fuel rod and in the TRU concentration. Super-PRISM fuel data was used to represent the reference FR. S-PRISM seed fuel pin is 6.322 mm in diameter (smeared) and 1016 mm long whereas the ENHS fuel pin is 13 mm in diameter (smeared) and 1250 mm long. That is, the HM inventory per ENHS fuel rod is 5.2 times higher than that of S-PRISM. In addition, the Pu or TRU contents in the ENHS fuel is only ~1/2 that in S-PRISM. The ENHS fuel fabrication

cost was approximately estimated by assuming four different mathematical models for extrapolating GEN-4 data as a function of the amount of HM per fabricated fuel rod and for interpolating of the fuel TRU contents. The results of the four models are - 532, 571, 615 and 625 dollars per kg of ENHS fuel. As the more sophisticated models yield the two lower values, a rounded-up specific cost of \$600/kg is assumed for our ENHS FCC analysis.

5. For the case of EU fueled ENHS (Scenario 1), which is much less active than Pu, these models gave 388, 390, 405 and 419 \$/kg. A conservative value of \$420/kg is assumed for our ENHS fuel cycle cost analysis.

Two of the four models used for getting the cost values presented in “4” and “5” above will now be described. The simplest model assumes that the fabrication cost of a fuel rod is reciprocally proportional to a fraction of the mass ratio. The value of this fraction was assumed to be 0.8. Using the value of \$2600/kg^{B.5} as pertinent to S-PRISM metallic fuel fabrication and the ratio of 5.2 between the masses of the fuel rods the ENHS fuel fabrication cost obtained is \$625/kg [=2600/(5.2*0.8)]. For EU fuel (Scenario 1), the fabrication cost of blanket rods is used for the reference. With \$500/kg^{B.5} and 1.546 ratio of ENHS fuel rod mass to S-PRISM blanket rod mass, the ENHS EU fuel fabrication cost is \$404/kg [= 500/(1.546*0.8)]. The S-PRISM blanket rods are 10.89 mm in diameter (smeared).

In the more sophisticated mathematical model used for cost interpolation the fabrication cost of a fuel rod, C_r , is assumed to be a product of two functions:

- F_c = a function of the fissile material content [quantity and quality].
- F_m = a function of the rod HM mass, m .

$$C_r = F_c * F_m \quad (B.1)$$

F_m dependence on m is assumed to be approximately linear:

$$F_m = A + B * m. \quad (B.2)$$

The fabrication cost per kg C is therefore

$$C = C_r / m = F_c * (A/m + B). \quad (B.3)$$

The following relative values are assumed for F_c :

- $F_c = 2$ for S-PRISM FR metal driver fuel rod having 21.4% Pu,
- $F_c = 1.5$ for ENHS TRU-U metal fuel rod having 11.4% Pu,
- $F_c = 1.1$ for ENHS EU metal fuel rod having 12.7% ²³⁵U, and
- $F_c = 1$ for S-PRISM blanket U metal rods.

Further assuming equal fuel density in the above fuel rods, using the nominal (unsmeared) dimensions of the S-PRISM rods (5.477 mm and 10.046 mm for, respectively, driver and blanket rod diameter^{B.6}) and using the pertinent cost data from Gen-IV guidelines^{B.5} (2600 \$/kg and 500 \$/kg, respectively) we obtain the numerical values of the A and B coefficients to be inserted into Equation B.3:

$$C = F_c * (338 * m_{bl} / m + 162) = F_c * (1138 * m_{dr} / m + 162). \quad (B.4)$$

In the above m_{bl} and m_{dr} are, respectively, the mass of the blanket and driver rods. Using this expression we get

$$C_{ENHS} = 1.5 * (338 / 1.546 + 162) = \$571/\text{kg} \text{ or}$$

$C_{\text{ENHS}} = 1.5 \cdot (1138/5.2 + 162) = \$571/\text{kg}$
 for TRU-U fueled ENHS (Scenarios 2, 3 and 4) whereas for EU fueled ENHS (Scenario 1):
 $C_{\text{ENHS}} = 1.1 \cdot (338 \cdot 0.647 + 162) = \$419/\text{kg}$

6. An extra charge of 1 mill/kW_eh need be imposed, so we think, on the fuel cycle cost of the LWR but not on that of the ENHS. This is because we interpret the present tax on the sale of nuclear electricity in the US as intended to cover primarily the cost of providing repository space for the high level waste. The repository space required for HLW from the ENHS fuel cycle is a very small fraction of that required for LWR spent fuel. The tables displaying the results of this FCC evaluation bring two FCC figures for LWR; one without this tax and one (in brackets) with the tax.

B.5 Other Assumptions

Additional assumptions used for calculating the fuel cycle cost include the following:

1. The reference real discount rate is 5% per year. Such a discount rate is suggested in a number of references, such as References B.4 and B.6. It is probably suitable for developing countries that will be getting loans from international organizations such as the World Bank for the purchase of their nuclear power plants. We also use a discount rate of 7% and of 10% for sensitivity analysis.
2. 93% capacity factor, as assumed for S-PRISM^{B.7, B.8}. The ENHS reactor is likely to have a higher availability.
3. The reference net efficiency of the reference ENHS is 38%. We calculated this value for a 120 bars Rankine cycle. This value happens also to have been used for S-PRISM^{B.8}. Using 180 bars steam pressure we find it possible to design the ENHS to have 40% efficiency. Hence 40% is also used for the fuel cycle cost of an improved ENHS design.
4. The reference ENHS design operates at 125 MWt. It is our estimate that it will be possible to upgrade the ENHS module to deliver 1/3 extra power without increasing the fuel inventory or enrichment. Hence, a power level of 167 MWt is considered as well.
5. The average discharge burnup of the reference ENHS design is 55.6 MWd/kgHM. What dictates this discharge burnup is the constraint of 4×10^{23} n/cm² of fast (E>0.1 MeV) neutron fluence on the HT-9 clad. However, it is our estimate that it will be possible to design cores for the ENHS that will feature ~50% higher average burnup. Consequently, a discharge burnup of 83.4 MWd/kgHM is considered as well in the FCC analysis of upgraded ENHS.

B.6 Results

The results of the FCC analysis using 5% discount rate are summarized in Table B.2. They are compared against the fuel cycle cost of the reference once-thru ALWR estimated using the same methodology and data base as used for the ENHS fuel cycle cost evaluation. The upper part of the table displays the calculated costs based on the nominal cost input data from Reference B.5. In the lower part of the table the calculated costs are based on the “lower-bound” cost data from Reference B.5. In addition to the reference ENHS design, Table B.2 brings fuel cycle cost estimates of a number of ENHS design upgrades. Following are a number of observations:

Table B.2 Fuel Cycle Cost of ENHS Reactor Relative to those of Once-Through ALWR.
5% Discount Rate

			ENHS				LWR	ALWR
Scenario			1	2	3	4		Reference
Fuel			EU	TRU-U	TRU-U	TRU-U	EU 3.1%	EU 4.3%
Fuel feed			U _{natural} (U ₃ O ₈)	Spent UOX	Spent MOX	Spent ENHS	Natural U (U ₃ O ₈)	
Power MWt	Burnup MWd/kg	Efficiency (%)	Fuel cycle cost (mills/kW _e h)					
NOMINAL COST DATA ^{B.5}								
125	55.6	38	13.1 / 15.2 ^b	19.5	9.3	6.1	4.4 (5.4 ^a)	3.9 (4.9 ^a) 4.5 ^b (5.5)
125	55.6	40	12.4 / 14.4	18.6	8.9	5.7		
125	83.4	38	10.9 / 12.6	16.2	7.7	5.0		
125	83.4	40	10.3 / 12.0	15.4	7.4	4.8		
167	55.6	38	11.6 / 13.5	17.4	8.3	5.4		
167	55.6	40	11.0 / 12.8	16.5	7.9	5.1		
167	83.4	38	9.2 / 10.7	13.8	6.6	4.3		
167	83.4	40	8.8 / 10.2	13.1	6.3	4.0		
LOWER BOUND COST DATA ^{B.5}								
125	55.6	38	8.4 / 11.4 ^b	12.3	5.5	3.4	3.1 (4.1)	2.7 (3.7) 3.5 ^b (4.5)
125	55.6	40	8.0 / 10.8	11.7	5.2	3.2		
125	83.4	38	7.0 / 9.4	10.2	4.6	2.8		
125	83.4	40	6.6 / 9.0	9.7	4.3	2.6		
167	55.6	38	7.5 / 10.1	10.9	4.9	3.0		
167	55.6	40	7.1 / 9.6	10.4	4.7	2.8		
167	83.4	38	5.9 / 8.0	8.6	3.9	2.4		
167	83.4	40	5.6 / 7.6	8.2	3.7	2.3		

^a Including a tax of 1 mill/kW_eh to the Federal Government

^b for uranium price of \$50/kg

1. The fuel cycle cost of recycled ENHS spent fuel is quite acceptable: 6.1 mills/kW_eh for the reference design. If the ENHS design will successfully be upgraded as we hope it can, the nominal FCC will drop to only 3.0 mills/kW_eh. If commercial and technological improvements in the nuclear fuel market and industry anticipated by the GEN-IV working group^{B.5} will realize, the FCC will be 3.4 and 2.3 mills/kW_eh for, respectively, the reference

design and the upgraded design. If the ENHS design upgrades are successful, the equilibrium ENHS cycle FCC might be competitive with the FCC of the once through ALWR.

2. The lowest FCC for the first ENHS is obtained when using as feed the spent fuel from MOX fuelled ALWR (one MOX cycle; Scenario 3); it is below 10 mills/kW_eh and is likely to be acceptable. If the ENHS design upgrades are successful and the lower bound cost data will turn out to be realistic for the ENHS, the FCC of the first ENHS could be competitive with that of the once through ALWR. This speculative conclusion assumes that the front-end costs for the ALWR will stay close to their nominal value of Table B.1 whereas the MOX fuel reprocessing (using UREX) cost will be its lower bound value.
3. The most expensive fuel for the ENHS is the one made from spent fuel from UOX fueled ALWR (Scenario 2). Its FCC is more than twice that of Scenario 3 and more than 3 times compared to ENHS recycled fuel (Scenario 4).
4. Fueling the first ENHS with enriched uranium (12.7% ²³⁵U) might be an acceptable option, particularly if the ENHS design upgrades are successful.

Table B.3 gives the FCC for bounding combinations of ENHS power, burnup and efficiencies for the four ENHS scenarios and a couple of LWR references corresponding to a discount rate of 7%. It is found that the LWR FCC does not vary to the first significant figure, the ENHS FCC does increase when the discount rate increases from 5% (Table B.2) to 7% (Table B.3). Nevertheless, The equilibrium ENHS FCC is still acceptable and so is the FCC pertaining to Scenario 3. The insensitivity of the LWR FCC to the discount rate is due to the fact that the increase in the front-end charges due to discount rate increase happens to be largely compensated by a reduction in the back-end charges.

Table B.4 gives FCC comparison for selected scenarios for a discount rate of 10%. The ENHS FCC for the second and following ENHS modules is still acceptable. Scenario 3 might also be acceptable for providing the loading for the first ENHS module, especially if the cost of reprocessing (using UREX) and ENHS fuel fabrication will be close to their lower bound values.

B.7 Conclusions

The fuel cycle cost of the ENHS reactor is not a showstopper. The equilibrium FCC is quite reasonable, even at 10% discount rate. The first fuel loading is more expensive but not prohibitively so. Increase in the cost of uranium ore will make the ENHS FCC more favorable relative to the LWR FCC cost. This is because the ENHS FCC is practically independent of the cost of uranium. This robustness to fluctuations in the uranium market need be taken into account in evaluating the viability of the ENHS reactor. In fact, once a utility or a country owns an ENHS power plant along with its first fuel loading, it has the fuel supply for centuries if not millennia as the only makeup fuel they need is depleted uranium.

A more thorough analysis need be carried out in order to assess the overall economic viability of the ENHS reactor and its fuel cycle.

Table B.3 Fuel Cycle Cost of ENHS Reactor Relative to those of Once-Through ALWR.
7% Discount Rate

			ENHS				LWR	ALWR
Scenario			1	2	3	4		Reference
Fuel			EU	TRU-U	TRU-U	TRU-U	EU 3.1%	EU 4.3%
Fuel feed			U _{natural} (U ₃ O ₈)	Spent UOX	Spent MOX	Spent ENHS	Natural U(U ₃ O ₈)	
Power MWt	Burnup MWd/kg	efficiency (%)	Fuel cycle cost (mills/kW _e h)					
NOMINAL COST DATA ^{B.5}								
125	55.6	38	15.7 / 18.4 ^b	26.3	11.4	7.4	4.4 (5.4 ^a)	4.0 (5.0 ^a)
125	55.6	40	15.0 / 17.4	25.0	10.8	7.0		4.6 ^b (5.6)
167	83.4	38	11.3 / 13.1	18.8	8.1	5.3		
167	83.4	40	10.7 / 12.5	17.9	7.7	5.0		
LOWER BOUND COST DATA ^{B.5}								
125	55.6	38	10.1 / 13.8 ^b	16.5	6.7	4.1	3.1 (4.1)	2.7 (3.7)
125	55.6	40	9.6 / 13.2	15.7	6.4	3.9		3.6 ^b (4.6)
167	83.4	38	7.2 / 9.9	11.8	4.8	2.9		
167	83.4	40	6.9 / 9.4	11.2	4.6	2.8		

^a Including a tax of 1 mill/kW_eh to the Federal Government

^b for uranium price of \$50/kg

Table B.4 Fuel Cycle Cost of ENHS Reactor Relative to those of Once-Through ALWR
10% Discount Rate

			ENHS				LWR	ALWR
Scenario			1	2	3	4		Reference
Fuel			EU	TRU-U	TRU-U	TRU-U	EU 3.1%	EU 4.3%
Fuel feed			Natural U (U ₃ O ₈)	Spent UOX	Spent MOX	Spent ENHS	Natural U (U ₃ O ₈)	
Power MW _t	Burnup MWd/kg	efficiency (%)	Fuel cycle cost (mills/kW _e h)					
NOMINAL COST DATA ^{B.5}								
125	55.6	38	20.4 / 23.9 ^b	37.3	14.8	9.6	4.5 (5.5 ^a)	4.2 (5.2 ^a)
125	55.6	40	19.4 / 22.7	35.4	14.1	9.1		4.9 ^b
167	83.4	38	14.8 / 17.4	27.1	10.8	7.0		(5.9)
167	83.4	40	14.1 / 16.5	25.7	10.3	6.6		
LOWER BOUND COST DATA ^{B.5}								
125	55.6	38	13.2 /18.2 ^b	23.5	8.8	5.3	3.2 (4.2)	2.9 (3.9)
125	55.6	40	12.5 / 17.3	22.3	8.3	5.1		3.9 ^b
167	83.4	38	9.6 / 13.2	17.1	6.4	3.9		(4.9)
167	83.4	40	9.1 / 12.5	16.2	6.1	3.7		

^a Including a tax of 1 mill/kW_eh to the Federal Government

^b for uranium price of \$50/kg

B.8 References

- B.1 E. Greenspan, H. Shimada, K. Wang, "Long-Life Cores with Small Burnup Reactivity Swing," Proc. ANS. Int. Topl. Mtg. Advances in Reactor Physics and Mathematics and Computation into the Next Millennium (PHYSOR2000), May 7-12, 2000.
- B.2 J.P. Grouiller et al., "Different Possible Scenarios for Plutonium Recycling in PWRs," Proc. Global'01, Paris, France, September 9-13, 2001.
- B.3 E. Greenspan and the ENHS project team, "The Long-Life Core Encapsulated Nuclear Heat Source Generation IV Reactor," To be presented at the International Congress on Advanced Nuclear Power Plant, ICAPP, Hollywood, FL, June 9-13, 2002.
- B.4 W. Burch, E. Rodwell, I. Taylor and M. Thompson, "A Review of the Economic Potential of Plutonium in Spent Nuclear Fuel," Electric Power Research Institute Report TR-106072, Feb. 1996.
- B.5 Generation IV Roadmap, "Report of the Fuel Cycle Crosscut Group (FCCG)", March 18, 2002.
- B.6 NEA OECD Expert Group, "The Economics of the Nuclear Fuel Cycle," OECD Report, 1985.
- B.7 A.E. Dubberley, C.E. Boardman, K. Yoshida and T. Wu, "SUPERPRISM Oxide and Metal Fuel Core Designs," Paper 8002 in Proc. 8th Int. Conf. on Nuclear Engineering, ICONE-8, Baltimore, MD, April 2-6, 2000.
- B.8 C.E. Boardman, "Economic Assessment of S-PRISM Including Development and Generating Costs," Proc. 9th Int. Conf. On Nuclear Engineering, ICONE-9, Nice, France, April 8-12, 2001.