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THE ENCAPSULATED NUCLEAR HEAT SOURCE FOR PROLIFERATION-RESISTANT LOW-WASTE NUCLEAR ENERGY

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Abstract

Encapsulated Nuclear Heat Source (ENHS) is a small innovative reactor suitable for use in developing countries. The reference design is a 50MWe lead-bismuth eutectic (Pb-Bi) cooled fast reactor. It is designed so that the fuel is installed and sealed into the reactor module at the factory. The nuclear controls, a major portion of the instrumentation and the Pb-Bi covering the core are also installed at the factory. At the site of operations the reactor module is inserted into a pool of Pb-Bi that contains the steam generators. Major components, such as the pool vessel and steam generators, are permanent and remain in place while the reactor module is replaced every 15 years. At the end of life the sealed reactor module is removed and returned to an internationally controlled recycling center. Thus, the ENHS provides a unique capability for ensuring the security of the nuclear fuel throughout its life. The design also can minimize the user country investment in nuclear technology and staff. Following operation and return of the module to the recycling facility, the useable components, including the fuel, are refurbished and available for reuse. A fuel cycle compatible with this approach has been identified that reduces the amount of nuclear waste.

1. INTRODUCTION

The encapsulated nuclear heat source (ENHS) is a small innovative reactor suitable for use in developing countries. The research on this concept is being conducted under the U.S. Department of Energy (DOE), Nuclear Energy Research Initiative (NERI). The research was initiated in the summer of 1999[1]. The most innovative feature in the ENHS is the fact that the nuclear fuel is encapsulated in the primary coolant vessel throughout its life. The heat generated in the reactor core is transferred through uniquely designed section of the primary coolant vessel wall to a pool of secondary coolant. The primary and secondary coolants are a lead bismuth eutectic. The secondary pool also contains the steam generators.

2. PERFORMANCE OBJECTIVES

The ENHS performance objectives are those established by DOE for the Generation IV(GEN IV) reactors [2]. These are summarized in the box below. The objectives address the entire fuel cycle as does the ENHS concept. The key features in the ENHS concept that support achievement of these objectives are the approach to the reactor module design and the use of fuel cycles that are compatible with its fast neutron spectrum.

Generation IV
Performance Objectives

- Competitive Busbar Cost of Electricity
- Acceptable Risk to Capital
- Limited Project and Construction Lead Times
- Low Liklihood of Core Damage
- Demonstration of No Severe Core Damage
- No Need for Offsite Response
- As Low as Resonable Achievable Radiation Exposure
- Tolerant to Human Error
- Solutions for all Waste Streams
- Public Acceptance of Waste Streams
- Minimal Waste
- Minimal Attractivness to Potential Weapons Proliferation
- Intrinsic and Extrinsic Proliferation Resistance

The latter characteristic and recycling of the fuel are fundamental to the minimization of high level waste. The ability of ENHS to achieve the first two economic objectives has not yet been determined, but the approach is expected to result in the absolute minimum construction and lead times.

The safety objectives will be met with significant margins. It should be possible to demonstrate through full scale testing that there will be no core damage under severe accident conditions, and no need for offsite response. Since the nuclear operations onsite are minimum and the response to any balance of plant events only impacts the availability, the plant should be very tolerant to human error. Po210 is generated in the coolants and needs to be given greater attention than it has been. However, there are data that indicate that its presence should not be a major problem[3].

A key driver in the design was the desire to achieve an improvement in the proliferation resistance. We believe this has been done, and with the realization of international recycling facilities the approach will permit the minimization of radioactive waste while retaining a high level of proliferation resistance.

3. SYSTEM DESCRIPTION

3.1 Nuclear Island

The ENHS reference design produces 50MWe from 125MWt using natural circulation in both the primary and secondary Pb-Bi coolants. The reference coolant was selected as a lead bismuth eutectic (Pb-Bi) of 45% lead and 55% bismuth. In principal, the concept would also work with sodium as the primary coolant. The installed nuclear island layout is illustrated schematically in Figure 1.

The heated primary coolant flows up from the core into downcomer passages that form the primary vessel wall. The secondary coolant flows up the primary vessel wall cooling the primary coolant as it flows down to the core inlet. The heated secondary coolant is then cooled by the steam generators in the secondary pool and returns to the bottom of the pool vessel. These flows are achieved by natural circulation. The natural circulation achieves design simplicity and avoids the need for active components but it requires a rather tall 19m primary vessel. An alternative design that uses a so called "lift pump" to introduce voids above the core outlet plenum by pumping cover gas into a sparger has been evaluated. This design is only 10m high and reduces the coolant mass significantly. This improvement is achieved at the expensive of adding an active component to pump the cover gas.

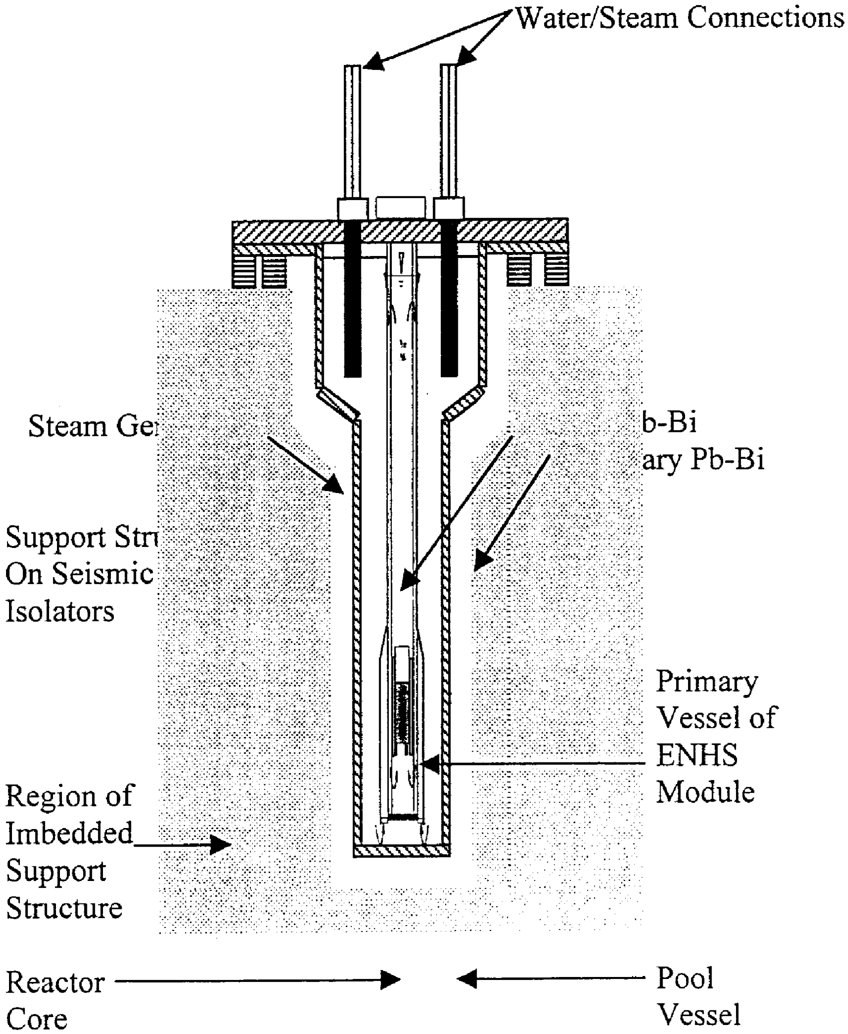

Figure 1 A Schematic Vertical View of a Single ENHS (Not to scale.)

An innovative tube-in-tube steam generator module has been designed. Each module produces 15.625 MWt of steam at 135 $bars_g$ (2000psig) and 482C (900F). In the reference single module design, eight of these fit around the reactor module in a pool vessel that has an upper diameter of 4.6 meters.

3.2 Manufacturing and Shipping

The ENHS module is designed for shop assembly to the maximum extent possible. This means all structure and equipment internal to the primary coolant vessel, including the fuel is installed at the factory. Parts internal to the primary coolant vessel will be designed for the life of the module. This includes reflectors and the control drive mechanisms that are internal. Cabling for instrumentation and preparation of the structures that mate with the pool vessel are all completed in the shop to support rapid installation at the site. In addition, Pb-Bi coolant, sufficient to cover the core, will be poured into the vessel and frozen. This approach is intended to both physically protect the fuel against shipping damage and provide additional discouragement to theft.

It is anticipated that special equipment for shipping and installation would be fabricated assembled along with the reactor module. This equipment would be used in the shop to erect the module to the vertical position so that the Pb-Bi can be poured into the core. It could later be used at the site to move the module from the horizontal shipping position into the vertical position for installation. This reusable equipment would be provided by the nuclear system supplier.

3.3 Installation at the Site

Upon arrival at the port of delivery the reactor module will be unloaded onto either rail transport or a special vehicle for road transport to the site. The transport fixtures and any special vehicles would be provided by the supplier and used repeatedly for delivery of multiple modules to one site, or for deliveries to various sites. The delivery would be made on a schedule consistent with completion of the secondary pool and installation of the steam generators. The module would be hoisted into the vertical position and lowered into the secondary pool vessel. When replacing a module however it would be desirable to make the exchange with the secondary Pb-Bi in place. It is anticipated this can be done while filling of the reactor module with primary Pb-Bi. The empty weight of the module will not be sufficient to cause it to sink into the pool to its operating depth. Therefore, as the reactor vessel is filled it will sink until fully immersed and mated with the secondary pool closure surface. The process of filling the module with hot Pb-Bi and sinking it into a pool of hot Pb-Bi will melt the Pb-Bi in the core. At this point the module will be structurally attached and seal welded to the pool vessel. Instrumentation and control cables will be connected to complete the installation.

The installation will be done with the Pb-Bi at an acceptably low temperature and using fixtures that maintain the inert cover gases over the Pb-Bi surfaces. These installation fixtures would be used repeatedly by the module supplier. In addition, special systems will be provided for Pb-Bi heating, circulating and purification. There is an open question as to whether or not it is possible to passively control the coolant chemistry quality throughout the operating life. This is a very desirable characteristic that might be possible with a sealed coolant system. If this is not possible, it will be necessary to provide a permanent active system to maintain the coolant chemistry. Installation of the reactor module and purifying the coolants is the last major task. Following checkout of the nuclear instrumentation and controls and their integration with the plant supervisory system the plant can start acceptance testing.

3.4 Operation

The plant may have multiple modules, either in individual pools or in a single larger pool. The plant, whether single or multiple modules, would be operated from a single control building. The operations would be limited to monitoring and maintenance. Planned maintenance would be limited to the balance of plant and steam generators. The lack of pumps and valves in the nuclear system and the expectation of maintaining a sealed corrosively benign coolant environment support meeting this objective.

Plant start-up will be initiated with a single switch and will proceed under the control of redundant supervisory computers. The system will be heated to 350C and the reflectors will be raised to bring the reactor critical. A preferred scheme for raising the temperature and initiating the natural circulation has not been selected. It may be feasible to do this using low reactor power. However, there may be other reasons for providing trace heating on the module and this equipment may play a role in the startup process, at least until sufficient decay heat is available to sustain the natural circulation.

Full power operation on natural circulation in both the primary and secondary coolant has been demonstrated analytically[4]. This feature is compatible with passive load following and autonomous control. The operational simplicity and improved safety margins should support a significant reduction in the size of the operating staff and cost of operations. The size of the staff will be dominated by those needed to maintain the balance of plant. The number of staff trained in nuclear technology will be minimal. The fact that there is no servicing of the module, including no refueling, is further reason for a minimal size nuclear staff.

3.5 Recycling and Waste Disposal

At the end of 15 years of operation and a short cooling period the module is removed from the pool to a storage location on site. This will be done by reversing the steps used for installation. The module will remain there, continuing to cool. Following solidification of the core in Pb-Bi the module is expected to be suitable once again for shipping to a recycle facility. Thus, the fuel is locked inside the ENHS from "cradle to grave". This feature restricts access to the fuel and simplifies safeguarding throughout the life of the module. The ENHS module also provides substantial barriers to accessing neutrons for use in production of fissionable material.

The module recycling would be completed in a few international monitored and controlled facilities located strategically through out the world. The total Pu and minor actinide inventory in the proposed ENHS-based energy system of a given capacity is fixed. Most of this transuranics (TRU) inventory is well secured inside ENHS modules. The only high level radioactive waste anticipated from this energy system consists of the following: (1) The fission products (FP) extracted in the dry fuel recycling process, (2) Trace losses of TRU and FP that can not be recovered from the spent fuel clad material during recycle, (3) Trace losses of TRU and FP waste from the fuel fabrication process, (4) Structural material activation products. It may be possible to reuse the vessel following annealing. The waste has not yet been characterized and quantified, but it is anticipated that it will be small and not a proliferation concern.

Recently Greenspan et al. [5] proposed a fuel cycle scheme that would further reduce the amount of high level radioactive waste. In this reference it is noted that the core life limit of ~105 GWd/tHM maximum burnup is due to the radiation damage to the cladding and fuel support structures. By partially reprocessing this spent fuel just to extract the volatile and some of the semi-volatile FP and mixing the product with makeup fuel the fuel can be recycled. The make-up fuel, approximately 7% of the fuel loading, can be spent fuel from light water reactors (LWRs) from which the volatile and semi-volatile FP have been removed. It may be possible to repeat this process many times. So far we have not carried out the burnup analysis beyond an average discharge burnup of 20%.

In addition to the reductions in high level waste, ENHS will create a minimum of low level radioactive waste because there is no need to provide normal servicing of the module. Some low level waste will result from the removal and replacement of the module every 15 years.

4. REACTOR MODULE DESCRIPTION

4.1 Design Parameters

As mentioned, two configurations for the ENHS reactor module are being evaluated. Table 1 gives some of the design parameters of both the natural circulation version of the module, ENHS1 and the lift-pump version, ENHS2.

Table 1: Selected Design Parameters of Representative ENHS Modules for $125MW_{th}$

Design parameter	ENHS1	ENHS2
Primary Pb-Bi coolant circulation	100% natural	With lift-pump
Average linear heat-rate (W/cm)	60	60
Average discharge BU* (MWd/tHM)	52,000	52,000
Core life* (effective full power years)	20	20
BU reactivity swing	<1$	<1$
Maximum excess reactivity	<1$	<1$
Core height (m)	1.25	1.50
Core diameter (m)	1.98	1.87
Fuel rod diameter (cm)	1.0	1.0
Clad thickness (cm)	0.1	0.1
Lattice (hexagonal) pitch (cm)	1.45	1.50
Overall module height (m)	19.6	10.1
Outer module diameter (m)	3.24	3.35
Number of rectangular channels in IHX	135	245
Inner dimensions of channel (cm x cm)	40 x 2.5	50 x 1.0
IHX channel length (m)	13	6
Weight of fueled module for shipment (t)	360	300
Coolant core inlet/outlet temperature (°C)	400/564	400/543
Primary-to-secondary mean ΔT (°C)	49.1	47.3
Number of steam generators per ENHS	8	8
Steam generator module diameter (m)	1.0	1.0
Active length of SG tubes (m)	7.5	7.5

* Limited by radiation damage to clad @ $4x10^{23}$ n/cm^2 >0.1 MeV

It can been seen that a major reduction in height is realized with the lift-pump design along with some reduction in shipping weight. In order to obtain sufficient heat transfer the primary coolant flow channels in the lift-pump design have been reduce to 1 cm and over 100 tubes are added. Appendix A provides a schematic of the ENHS1 module.

4.2 Intermediate Heat Exchanger

As mentioned, one of the key issues of feasibility was a concern about the unique characteristic of transferring heat through the primary vessel wall. This wall was conceived initially as a corrugated thin walled structure. The corrugations provided the necessary increase in heat transfer area between the primary coolant flowing downward along the inside of the wall and the secondary coolant flow upward on the outside of the wall. Both the fabrication and structural support of this configuration presented problems. Several more practical alternatives have been identified. The reference design uses rectangular tubes that are 2cm by 40cm with 2mm wall thickness. Section view A-A in Appendix A depicts the top view of rectangular coolant flow tubes that the primary coolant flows downward through. The secondary coolant flows upward through similar alternate channels formed by the tubes. Both the structural and thermal performance feasibility of such a design have been confirmed [6], but fabricability issues remain and other more innovative as well as conventional tubular designs are being considered.

4.3 Core Design

A key requirement of the ENHS is to have a long life core without access to the fuel. To accomplish this it means that the initially loaded core must have a very long life. From a safety standpoint it is desirable to do this without a large excess of reactivity throughout the life. The design domain for cores that meet these design objectives for at least 15 years of full power (EFPY) is defined in Greenspan et al.[1]. What limits the core life is the radiation damage to fuel structural materials, assumed to be $4x10^{+23}$ n/cm^2. The corresponding peak fuel burnup is approximately 105,000 MWD/tHM. The fuel considered for these cores is metallic Pu-UZr fuel with $10^{w}/_{o}$ Zr. Typical Pu concentration is 11-12 $^{w}/_{o}$ of HM. By adjusting the lattice p/d ratio and the Pu $^{w}/_{o}$, it is possible to change the slope of k_{eff} vs. burnup.

4.4 Safety

A series of transient simulations for the ENHS were performed using the DNSP computer code and have demonstrated large margins to damage of either the fuel or structures [7]. In addition various operational transients such as normal startup, the following postulated accidents were studied:

- A postulated \$1.5 reactivity insertion from low power followed by a scram
- Loss of heat sink without scram
- Steam line break without scram

Although more than one dollar of external reactivity was inserted in the postulated reactivity accident, the strong negative feedback mitigated the consequences. The core response is benign and slow. The fuel integrity is retained even in this event for which there is no credible mechanism.

The system heatup during the LOHS without scram is very slow and will reach a maximum in the range of 600-700°C, in a few days, after which the temperature will decrease according to the decay heat curve, and finally the system is expected to freeze if not restarted. Under these conditions the small amount of remaining decay heat will be removed by conduction through the solid lead to the surface of the containment vessel.

The steam line break produced only a minor effect in the ENHS, and following the SG dry-out the behavior is similar to the LOHS accident.

The temperatures in the fuel and structures are sufficiently in low in these events that it is very likely that a series of severe accident tests can be implemented in a prototype to demonstrate the inherent safety of the ENHS.

5. STATUS OF RESEARCH

The ENHS project was proposed to the NERI as a three-year effort and we are more than half way through the project. A number of the feasibility issues have been resolved but much design refinement and optimization remains to be completed. Progress to date has been encouraging and we hope to develop continued interest both domestic and international in continuing the research. Currently we have collaborators from Japan (Toshiba and CRIEPI) and from South Korea (KAERI), and we would welcome the participation of other interested organizations. On completion of the currently planned three-year project we would hope to identify resources that would permit further design optimization and some key feature testing.

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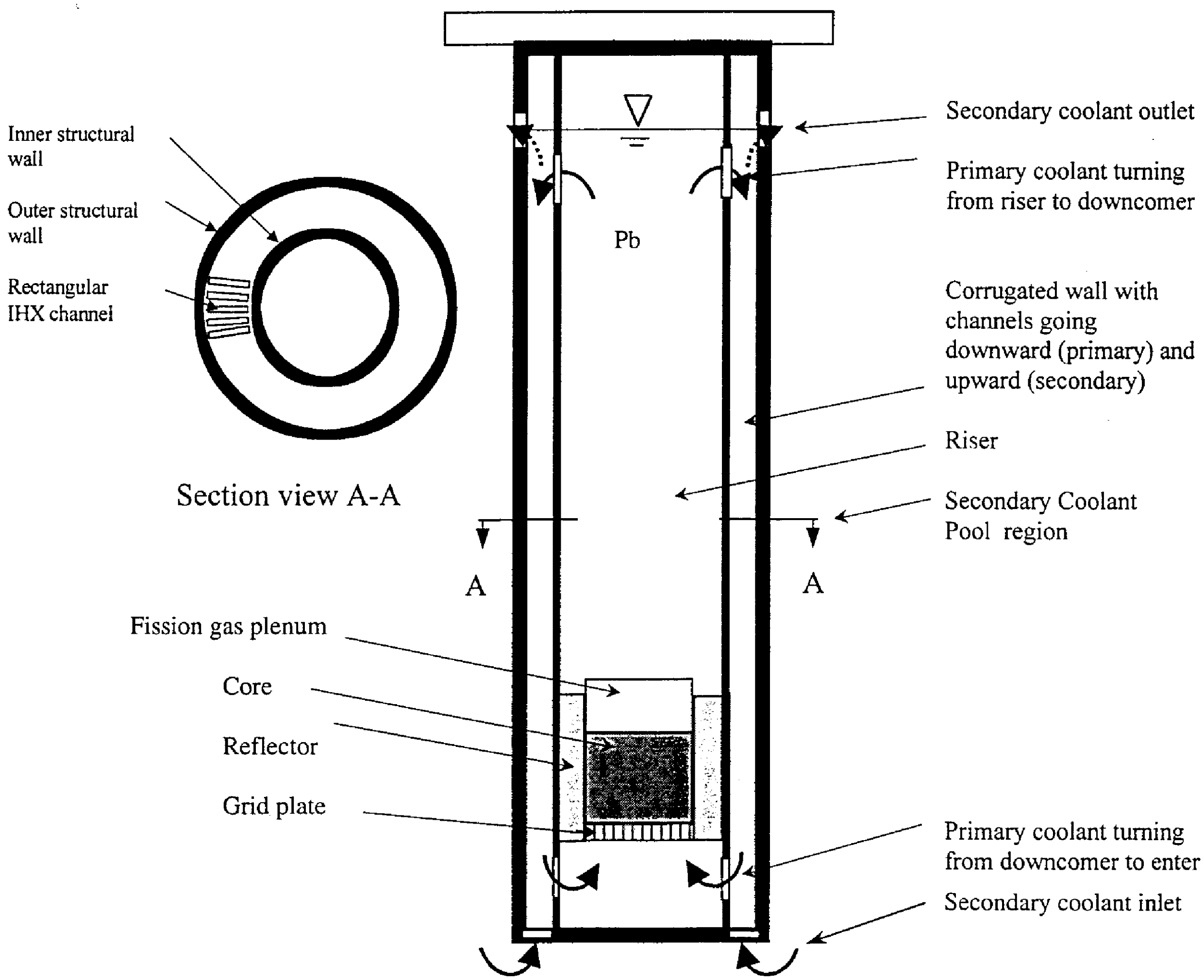
APPENDIX A : ENHS1 100% NATURAL CIRCULATION MODULE
(Schematic, not to scale)



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