R &D OF ENERGY TECHNOLOGIES

ANNEX A
II-NUCLEAR FISSION ENERGY

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AII-1 NEW NUCLEAR ENERGY PLANT DESIGNS

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Nuclear energy plants in the United States supply approximately 20% of this nation electricity. In the world, nuclear energy plants supply over 16% of the demand in mostly developed nations. Most of the plants are light water reactors using the steam cycle to generate electricity in either pressurized or boiling water reactors. The expansion of nuclear energy has been dramatically slowed in the United States and the world due to several factors, perhaps the most significant are two major accidents (Three Mile Island in 1979 and Russia's Chernobyl in 1986). While the Chernobyl design was not a conventional reactor used throughout the rest of the world, it did create a strong negative reaction to nuclear energy in Europe. Other significant factors include high capital cost, difficult regulatory climate and lack of electricity demand due to the international slow down in economic development. Southeast Asia is the only area of the world where nuclear energy is expanding however much more slowly than in the past.

Evolutionary nuclear plants (Generation III) have been developed around the world to improve the overall safety performance of the plants by an order of magnitude and to simplify the plants to make their construction less costly. In the US, these evolutionary plants have been "certified" by the Nuclear Regulatory Commission allowing for their construction. Few evolutionary plants have been ordered due to the cost still being much higher than alternative forms of electric generation such as natural gas and coal. Only in Southeast Asia (Japan, for example) have evolutionary plants been built.

In an effort to renew interest in nuclear energy plants in the United States, the United States Department of Energy (DOE) initiated several programs to stimulate construction in the short term and to develop new technologies for the 20 to 30 year horizon. The "Near Term Deployment" initiative[1] is aimed at using more conventional evolutionary reactors that are capable of being deployed by 2010. Examples of Near Term Deployment plants - Generation III - are shown on Table 1.

In 2000, the international community launched the Generation IV program[2] with 10 nations[1] participating. This program was initiated to establish the direction of future nuclear energy plant designs. The experts of these nations worked together to develop a "Roadmap" to identify technologies that they would develop in a collaborative research and development program leading to the deployment of these technologies in 20 to 30 years. This evaluation and plan concluded in 2002 with a set of six nuclear reactor types that would be developed for the future by these nations.

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[1] Argentina, Brazil, Canada, France, Japan, Korea, South Africa, Switzerland, United Kingdom, and the United States.

In the past, nuclear safety was provided by "engineered safety systems" which are systems called upon should the reactor experience some form of upset including a break in one of the major coolant pipes. These active systems rely on pumps and valves that need to be operated to provide the necessary cooling water to prevent the fuel from being damaged, or worse, melting due to the lack of water or coolant. Evolutionary designs include more "passive" systems that respond without the need for much active mechanical and hence electrical actuation. The Generation IV future designs are aimed at inherent safety built into the reactor such that the fundamental design, even with a loss of coolant, does not challenge the integrity of the core.

Table 1 – Near Term Deployment Plants [1]

Design	Supplier	Features
ABWR	GE	1,350 MWe BWR, design certified by NRC and built and operating in Japan
SWR 100	Framatome ANP	1,013 MWe BWR, being designed to meet European requirements
ESBWR	GE	1,380 MWe passively safe BWR, under development
AP600	Westinghouse	610 MWe passively safe PWR, design certified by NRC
AP1000	Westinghouse	1,090 MWe PWR with passive safety features higher capacity version of AP-600 not yet certified
IRIS	Westinghouse	100-300 MWe integral primary system PWR under development
PBMR	ESKOM	110 MWe modular direct cycle helium-cooled pebble bed reactor, currently planned for construction in South Africa
GT-MHR	General Atomics	288 MWe modular direct cycle helium-cooled reactor being licensed for construction in Russia

ABWR – Advanced Boiling Water Reactor
ESBWR – European Simplified Boiling Water Reactor
IRIS – International Reactor Innovative and Secure
PBMR – Pebble Bed Modular Reactor
GT-MHR – Gas Turbine-Modular Helium Reactor

Design features that are being considered include low power density cores, reactors placed in larger vessels that can provide coolant and cooling without the need for external systems, new fuel designs that can withstand high temperatures, and natural heat transfer to the environment, as examples. In general, the objectives of these improvements are designed to reduce the core damage frequency by a factor of 10 lower than current designs to 10^{-6} per year and to provide more efficient and lower costs for construction and operation.

The Generation IV Roadmap[2] established three specific high levels goals for new advanced designs-Sustainability, Safety and Reliability and Economics. Sustainability includes several subgoals relative to resource utilization and waste minimization.Economics focus on competitiveness with other energy sources and minimization of financial risk. Safety and Reliability addresses operational reliability, low likelihood and degree of core damage and the elimination of the need for offsite emergency response[2]. Additionally, consideration was given to proliferation resistance and physical protection.

As a result of this two-year evaluation in which over 100 design ideas were submitted and reviewed by technical experts all over the world, six concepts were selected by the international partners to cooperatively develop that best met the goals established. The six concepts are shown on Table 2.

Table 2 – Generation IV Concepts for Development[2]

Concept	Size MWe	Features
Very High Temper. Reactor (VHTR)	120 – 600	Helium Cooled Brayton cycle 1000 °C outlet temperature in Pebble or Prismatic core form Strengths: safety, 50% efficiency, hydrogen production
Supercritical Water Cooled Reactor	1700	High temperature, high pressure above critical point of water, 550 °C Strengths: economy of scale
Lead Cooled Fast Reactor	50-1200	Fast reactor – closed fuel cycle – long life core (up to 30 years) Strengths: actinide burning, waste minimization, proliferation resistant
Sodium Liquid Metal Cooled Reactor	150-1500	Fast reactor with breeding potential closed fuel cycle Strengths: actinide burning with co-located reprocessing facilities
Gas Cooled Fast Reactor	300	Helium or CO_2 cooled fast reactor, closed fuel cycle Strengths: actinide burning with possible hydrogen mission.
Molten Salt Reactor	1000	Fuel circulating in fluoride mixture of sodium, zirconium, and uranium in an epithermal to thermal spectrum at 700 °C outlet temperature, closed fuel cycle, low pressure Strengths: actinide management

Each of these designs has strengths and weaknesses relative to the criteria established. Some are strong on sustainability – Liquid metal cooled and gas fast reactors; others are projected to be strong on economics due to high efficiencies – supercritical water and the very high temperature reactor; some have good non-proliferation features – molten salt reactor and the small lead cartridge reactor. From the safety perspective, the high temperature gas reactors have important inherent safety features. Each has significant research and development challenges largely focused on materials and proof of concept within economic practicality.

[2] Safety and Reliability Goal # 3

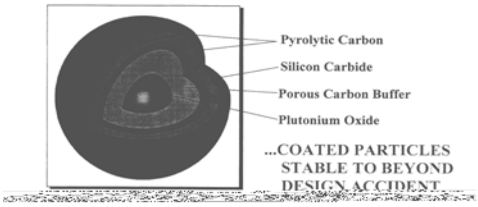
The United States has chosen the Very High Temperature Gas Reactor as their primary choice for future development[3]. This reactor, whose predecessor is the high temperature gas reactor, was chosen because of its high thermal efficiency and high coolant exit temperatures that can be used for thermo-chemical production of hydrogen to support a future hydrogen economy. While the material challenges to deal with 1000 C exhaust temperatures are great, less aggressive future designs which can operate in the range of 900 C can provide the capability to produce hydrogen in the near term but with slightly lower theoretical efficiencies. Some argue that the best use of high temperature reactors is to take advantage of higher thermal efficiencies (approaching 50 %) and hence lower production costs to make hydrogen economically by thermally enhanced electrolysis of water. The advantage of the electrolysis of water approach is that the infrastructure issues relating to the transportation of hydrogen to point of use can be avoided since electric conversion can be performed at point of use.

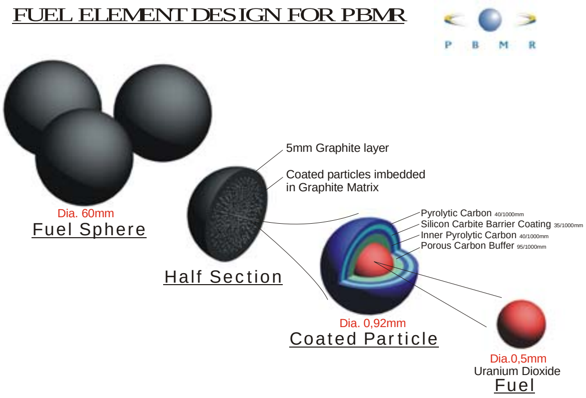
The highest priority for the US Department of Energy's Office of Nuclear Energy, Science and Technology is on the development of "a more economically competitive nuclear energy system" which they hope to accomplish using a VHTR that can also be used to generate hydrogen. The initiative to develop and build such a system is known as the Next Generation Nuclear Plant (NGNP). This project and plant, if approved by Congress, is expected to be built at the Idaho National Engineering and Environmental Laboratory by 2017.

Since high temperature gas reactors appear to be the near term technology of choice for future reactors, there are two types that are under consideration. In the US, the prismatic gas reactor developed by General Atomics has been modified by going to higher temperatures and the use of a direct helium Brayton cycle in which the helium coolant is used to drive gas turbine generators. This technology is being developed by the Russians in a joint US, French and Japanese program. Japan has a 30 MWth prismatic high temperature research reactor (HTTR) currently in operation at the Japan Atomic Energy Research Institute. Past experience with gas reactors in the US using this technology has not been very good with the only large commercial reactor, Fort St. Vrain, prematurely shutdown due to difficulties in operations of this helium cooled core with steam generator Rankine cycle generating electricity.

The other high temperature gas reactor was developed in Germany using the pebble bed reactor concept. Germany built the AVR test and demonstration reactor in 1967 which operated for 22 years. This reactor was also an indirect cycle in which the helium was passed through a steam generator that made electricity in addition to its research mission. Germany also built a 300 MWe pebble bed reactor that was shutdown following public concern about nuclear energy following the Chernobyl accident. At present, the only operating pebble bed reactor is the HTR-10 research reactor in China that operates on the indirect helium/steam cycle. This reactor began operation in 2000 to provide the technology foundation for future commercial plants. The commercial version of the pebble bed reactor is being developed in South Africa by ESKOM, South Africa's national electric utility. This reactor is a 165 MWe direct cycle helium cooled Brayton cycle plant that is in the engineering design phase. The government of South Africa is now in the technical and environmental review of the project. It is expected that the plant can be operational by 2008.

The fundamentals of high temperature reactors are the same in both types. Both yield high thermal efficiencies due to high temperature operations. Both use the same microsphere fuel which are tiny fuel particles coated by pyrocarbons and contained by a silicon carbide shell (Figure 1).





Prismatic Block (~800 cm high) Pebble Sphere (6 cm Diameter)

Figure 1 – Comparison of Fuel Types for Prismatic and Pebble Reators

Both prismatic and pebble bed reactors can operate on direct and indirect cycles in which helium to helium heat exchangers are used for either electricity production or hydrogen production. The prismatic reactor requires annual or 18 month shutdowns for refueling while the pebble bed reactor is an online refueling reactor in which the pebbles are continuously recirculated with fully depleted pebbles withdrawn and fresh pebbles are added when needed. This difference is the fundamental difference between the two technologies providing certain advantages and disadvantages relative to operability, economics and reliability.

The inherent safety features of the high temperature gas reactors is that they operate at relatively low power densities (about a factor of 5 to 10 lower) than light water reactors. They use a chemically inert coolant – helium – which reduces material corrosion and activation problems in the primary circuit. In addition, there is a great deal of graphite in both reactors which is used as a moderator that provides a large amount of heat capacity to absorb heat during normal and accident situations. Because of the low power density and high thermal heat capacity, gas reactors can not "meltdown" and do not require any active emergency cooling systems. The thermal time response of these reactors is very slow in comparison to light water reactors providing another inherent safety feature. One of the issues that requires resolution is that of air ingress in the event of a major pipe break. While additional analysis to prove these claims will be required in the regulatory arena, the tests and experiments performed by the Germans and analyses performed in Japan, South Africa, China and the US support these conclusions. Which of the two technologies will be chosen for the NGNP will depend on the economics and adaptability for the electricity and hydrogen mission in a design competition.

As can be seen by this short summary, there is a great deal of excitement and activity in the nuclear energy research and development arena. The new interest is, in part, stimulated by the challenge of global warming and, in part, by the excellent performance of the 104 nuclear plants in the US in the past 10 years. Most of the current operating plants will have their 40 year Nuclear Regulatory Commission licenses renewed for an additional 20 years. This will allow for the development of near term and long term advanced nuclear energy plants to supply electricity and possibly be used to produce greenhouse gas free hydrogen.

References:

[1] "A Roadmap to Deploy New Nuclear Plants in the United States by 2010," Volumes 1 & 2, US Department of Energy Nuclear Research Advisory Committtee Subcommittee on Generation IV Technology Planning, October 31, 2001.

[2] "A Technology Roadmap For Generation IV Nuclear Energy Systems", Generation IV International Forum, GIF-002-00, December 2002.

[3] "The US Generation IV Implementation Strategy," US Department of Energy, Office of Nuclear Science and Technology, September 2003.

AII-2 R&D NEEDS FOR FISSION ENERGY GENERATION

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R&D Needs are proposed for the development of nuclear energy systems that will generate competitive energy production globally through enhanced operation of already deployed nuclear reactors and their associated fuel cycle facilities as well as the future deployment of advanced reactors and their fuel cycle facilities. This R&D is envisioned to be funded by government, industry, and regulatory authorities in programs launched worldwide. These nuclear energy systems must be highly safe and economical generators which are concurrently responsive to environmental, waste management, and weapons material proliferation concerns. R&D technologies will be presented for three categories of nuclear energy system:

- the existing fleet of water cooled reactors (Generation II reactors except for the few advanced BWRs in operation and construction)
- the advanced water cooled plants of both evolutionary and passive concept already designed but not yet deployed (Generation III reactors) and
- future plants employing various coolants with associated open or fully closed fuel cycles (Generation IV reactors).

AII-2.1 The R&D needs for operating reactors

R&D needs for operating reactors are focused on enhancing concurrent safe and economic operation while achieving effective waste management and proliferation control within the fuel cycle operations already being employed – the open once-thru fuel cycle employed principally in the Americas and Scandinavia and the single-recycle mixed-oxide (MOX) fuel cycle employed principally in France. These needs are:

- Operations in general: management models and methodologies which guide the application of coordinated, effective direction to the execution of the myriad of tasks which must be performed to run a complex modern nuclear power reactor; human training models to enhance delivery and trainee retention of the most effective technologies for plant operation and maintenance.
- Safe operation in particular: human performance models aimed at error mitigation in all aspects of plant conduct and engineering support activities.
- Economic operation in particular:
  - . digital instrumentation and controls, communications and man-machine interface technology including decision-making methodologies to conduct condition monitoring intervention and on-line maintenance and repair of operationally limiting conditions.
  - . cheaper enrichment technologies to make fuel cycles more economical. Further, if U-236 could be stripped from spent fuel (another isotope separation task), then the uranium separated in the single recycle fuel cycle could itself be recycled reducing ore costs and existing storage costs.
- Achievement of projected operational lifetimes: achievement of assumed plant lifetime is challenged by degradation of the material condition of key systems and components. In all light water cooled reactors both boiling and pressurized designs, although the

degradation mechanisms and the most threatened components differ, aqueous corrosion has dictated unexpected and consequent premature replacement of primary system piping, reactor vessel internals, and large components. For each material in its operating environment it is necessary to determine the operative degradation mechanism and the time rate of evolution of the associated failure mechanism so that timely intervention can be executed to preclude unexpected loss of generation capacity. Further in-site diagnostics of corrosion progression and margin-to-failure are useful instrumentation to develop. In particular, current and future degradation if Nickel-Base alloys and their associated weldments used in primary systems will be a dominant issue for life extension of current generation LWR systems. Pressure vessel steel degradation in boric acid environments will be a significant, but not critical, concern.

- Waste management and proliferation control: while the once-through fuel cycle primarily needs the demonstration of an acceptable geological repository through the licensing of the Yucca Mt. Repository in as timely a way as practicable. It could also benefit from R&D to enhance the geological factors and the design of engineered barriers associated with spent fuel repository. The development of the deep borehole approach is a desirable element of this R&D. The single recycle-MOX cycle requires R+D to further enhance safe and economic PUREX reprocessing technology including minimization of secondary generated waste streams and effective encapsulation through vitrification of minor actinides in the spent fuel. Both cycles would benefit from volume reduction achievable by higher fuel burnup. Associated advances in automated, remote safeguards technologies involving information collection and process monitoring and interpretation are desirable.
- Because power uprates of currently operating nuclear power plants are economically very attractive for the utilities some R&D should be also directed to the identification of the best approaches to increase power rating while maintaining safety margins. This may require changes in fuel geometry and qualification of new fuels for higher power levels and for higher burnups to achieve the same 18-month cycle length.

AII-2.2 R&D Needs for near term

- The R&D Needs for near term (by about 2015) deployable reactors are focused upon reduction in construction and operational costs for the group of water cooled reactors already or nearly designed and fuels, materials, and power conversion system needs for high temperature gas cooled reactors. The fuel cycles for these systems are once-through cycles as for Generation II reactors of (1) above.
- Reduced construction costs: The capital cost and construction time requirements for these advanced light water reactors must be reduced to assure their competitiveness with alternate fossil generating systems. Virtual construction capabilities need development. Modularity in design is a technique well exploited in shipbuilding which is being applied in reactor technology. R&D is needed on differentiating the relative aspects of modularity which can be exploited as a function of reactor size. Such investigation will be instrumental in resolving the current debate on whether traditional economy scale arguments combined with adoption of relevant modular approaches dictate that larger size plants (say greater than 600-900 MWe) will be cheaper than smaller plants which can more fully adopt modular approaches. This tradeoff further involves the size of the existing grid for which the candidate plant is contemplated; a

further variable which the overall R&D activity on economics of plant construction should address.

- Safer light water reactors: the apparent safety advantages of integral vessel PWRs need to be confirmed by risk analysis and it must be demonstrated by R&D activities that these reactors can be effectively maintained over their lifetime.
- High temperature gas reactors: to achieve their potential of enhanced performance these reactors require high integrity fuel, radiation resistant structural graphite material and power generating system components capable of sustained operation at high (of order 900°C) gas temperatures. R&D activity in all three areas has been underway, for dozens of years. While progress has been made in the fuel and graphite material area, much more effort is needed to develop and characterize the required material production processes to ensure that identical to those that pass irradiation and performance testing can be consistently produced. Achievement of required high thermal efficiency, reliable power conversion equipment requires development of compressors, recuperators (heat exchangers) and turbines which are compact and operate at high temperatures above levels of current sustained operating experience. Verification and validation of the reliable control scheme for closed Brayton cycle will also be required.

AII-2.3 The R&D Needs for future Generation IV nuclear energy syste*ms*

The R&D Needs for future Generation IV nuclear energy systems encompass both fuel cycles and power reactors. For these future systems which are to be deployable by 2025, time exists to develop R&D results to match design goals. These goals are analogous to but more ambitious than those for the previous categories (1) and (2) of reactor systems.

- The primary goal is to assure the uranium utilization of the reactor systems deployed to meet global energy demand is consistent with uranium resources. The sufficiency of such resource is debatable because it is not cost effective for the commercial sector to establish known reserves beyond certain depletion rate levels. Nevertheless, it seems appropriate based on current resource assessments and the history in the 20th century of mineral price decrease even as consumption has increased significantly to establish the policy that creation of fissile material from fertile U 238 (breeding) is not necessary as a reactor design objective until well into this century and likely throughout the century. This policy can be periodically reviewed and changed should nuclear demand increase or insufficient new reserves be forthcoming. A global exploration program to better establish the earth's uranium reserves is also warranted.

- Two classes of reactors are candidates for Generation IV systems - thermal spectrum reactors, supercritical water-cooled, high temperature gas and molten salt - fast spectrum reactors employing liquid metals (sodium and lead) as well as gas and supercritical water. The overriding common needs for these classes of reactors are
  - the development and implementation of a risk-informed performance based methodology for regulation of the design certification, construction, and operation of these reactor systems.
  - a technically achievable, economic power cycle. A power cycle with such promise is the supercritical CO_2 direct Brayton cycle which offers this potential because of its ability to operate at temperatures in the 600°C range while achieving efficiencies

comparable to helium cycles which need operation at several hundred degrees higher temperatures. Such cycle developments are synergistic with future fusion power cycle needs.

- Both thermal and fast reactor designs can operate with a closed fuel cycle and thereby perform an actinide destruction mission. The primary closed fuel cycle objectives are to develop fuel forms and reprocessing technologies which do not produce weapons usable materials, are economic compared to the once-thru alternative fuel cycle, utilize facilities with safe operation characteristics which also sufficiently minimize the generation of waste streams such that the short term (hundreds of years) risks compare favorably to the avoided long term (hundreds of thousands of years) risks from geologic disposal of unreprocessed spent fuel. Further R+D is necessary to compare and optimize strategies for synergistic combinations of electricity generating and actinide destroying reactors employing various types of thermal and fast reactors to simultaneously achieve future electricity production and waste management needs.

AII-2. 1 Generation IV Thermal Reactors

The primary use of thermal Generation IV reactors is to yield enhanced safe and economic production of electricity while operating on the once-through fuel cycle. The gas cooled reactor is also being designed for very high temperature operation (greater than 1000˚C) to demonstrate the capability to produce hydrogen. The needed reactor coolant outlet temperature depends on the competitive economics of thermo chemical water splitting versus high temperature electrolysis which must be established.

The R&D needs to establish the technology base for each of these thermal reactors follows (Taken from Reference 1). It must be recognized however that sufficient resources are not available even worldwide to develop all these reactor systems in parallel. The 10 nation Generation IV Forum will rapidly need to focus resources on a single thermal system and likely one or, at most, two fast systems. It is suggested here that the thermal system be the gas reactor but at an outlet temperature around 900°C to enhance chances of success and the R&D focus for fast reactors be on the fast gas and lead alloy cooled systems to exploit approaches which have promise and which have not been explored intensively in the past. Ths will allow later effective comparison of these fast reactor concepts with the front runner, the sodium cooled fast reactor.

2.4.1 Supercritical-Water-Cooled Reactor (SCWR)

- SCWR materials and structures, including corrosion and stress corrosion cracking (SCC), radiolysis and water chemistry, dimensional and microstructural stability, and strength, embrittlement, and creep resistance
- SCWR safety, including power-flow stability during operation
- SCWR plant design.

2.4.2 Very-High-Temperature Reactor (VHTR)

- Novel fuels and materials must be developed that permit increasing the core-outlet temperatures from 900°C to over 1000°C, permit the maximum fuel temperature reached following accidents to reach 1800°C, maximum fuel burnup to 150–200 GWD/ MTHM, and avoid power peaking and temperature gradients in the core, as well as hot streaks in the coolant gas.

- Process-specific R&D gaps exist to adapt the chemical process and the nuclear heat source to each other with regard to temperatures, power levels and operational pressures
- Qualification of high-temperature alloys and coatings for resistance to corrosive gases like hydrogen, carbon monoxide, and methane.
- Development of a high-performance helium turbine for efficient generation of electricity. The increase in gas outlet temperature from 900 to over 1000°C will require a step change in turbine hot section technology which may require ceramic materials development for this application. Development of heat exchangers, coolant gas ducts, and valves will be necessary for isolation of the nuclear island from hydrogen production facilities. The development of the intermediate heat exchangers transferring the heat between the primary helium and the fluid carrying high temperature heat to hydrogen plant is the key challenge that needs to be overcome.

Molten Salt Reactors (MSR)

Here the first target is molten salt cooled versus molten salt-fueled reactors.

- Solubility of minor actinides and lanthanides in molten fluoride salt fuel for actinide management with high actinide concentrations
- Lifetime behavior of the molten salt fuel chemistry, and fuel processing during operation and eventual disposal in a final waste form
- Materials compatibility with both fresh and irradiated molten salt fuel for higher temperature applications.
- Metal clustering (noble metals plate-out on of the heat exchanger primary wall).
- Salt processing, separation, and reprocessing technology development, including a simplification of the flowsheet.
- Fuel development, new cross section data, and qualifications to enable selection of the molten salt composition
- Development of tritium control technology
- Graphite sealing technology and graphite stability improvement and testing

AII-2.5 Generation IV Fast Reactors

The R&D needs to establish the technology base for each of the fast reactors follows: (Taken from Reference 1) All fast reactors for sustainable scenario with TRU recycling will have large amount of TRUs, for which the cross section libraries carry high uncertainty at high neutron energies. Thus, improvement of cross section data for minor actinides with quality assurance will be required.

Gas-Cooled Fast Reactor (GFR)

Fuel forms for the fast-neutron spectrum and, if helium cooled, capable of high temperature (>800°C) operation. . In addition, fuels for GFRs will have to be able to achieve high heavy metal loadings in the core.

- If supercritical CO_2 cycle is selected for direct cycle, research of radiolysis of high-pressure CO_2 in hard spectrum will be required as well as the development of materials resistant to CO_2 corrosion up to 750°C under radiolytic products.
- Core design, achieving a fast-neutron spectrum for effective conversion with no fertile blankets.

- Safety, including decay heat removal systems that address the significantly higher power density (in the range of 100 MWth/m^3) and the reduction of the thermal inertia provided by graphite in the modular thermal reactor designs
- Fuel cycle technology, including simple and compact spent-fuel treatment and refabrication for recycling.
- Economics, focusing on modularization and factory fabrication
- Materials with superior resistance to fast-neutron fluence under very-high-temperature conditions
- A high-performance helium turbine for efficient generation of electricity
- Efficient coupling technologies for process heat applications and high temperature nuclear heat.

Lead-Alloy-Cooled Fast Reactor (LFR)

- Fuels and materials (cladding) including nitride fuels development, (Cheap N-15 enrichment is needed for such fuel development.), and environmental issues with lead.
- The development of structural materials that will be compatible with liquid metal coolants at temperatures that may exceed 800°C.
- System design, including open lattice heat removal, both forced, and natural convective; neutronic data and analysis tools; coolant chemistry control, especially oxygen and ^{210}Po control; innovative heat transport methods (such as design for natural circulation, lift pumps, in-vessel steam generators); core internals support and refueling machinery; and seismic isolation.
- Balance of plant, adapting supercritical steam Rankine or developing supercritical CO_2 electricity production technology, and crosscutting R&D on
- hydrogen production technology and heat exchangers for process heat applications
- Economics, focusing on modularization and factory fabrication
- Fuel cycle technology, including remote fabrication of metal alloy and TRU-N fuels.

Sodium-Cooled Fast Reactor

- .Ensuring of passive safe response to all design basis initiators, including anticipated transients without scram (a major advantage for these systems).
- Capital cost reduction.
- Proof by test of the ability of the reactor to accommodate bounding events
- Development of oxide fuel fabrication technology with remote operation and maintenance.

References

[1] "A Technology Roadmap for Generation IV Nuclear Energy Systems", GIF-002-00, U.S.D.O.E. Nuclear Energy Research Advisory Committee and the Generation IV International Forum, December 2002.

[2] "Long-Term Nuclear Technology Research and Development Plan", U.S.D.O.E. Nuclear Energy Research Advisory Committee (NERAC) Subcommittee on Long Term Planning for Nuclear Energy Research, June 2000.

AII-3 PHYSICS-BASED R&D FOR INNOVATIVE FISSION REACTORS

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Nuclear data such as neutron reaction cross sections are indispensable for the R&D of fission reactors. Therefore, nuclear data libraries were constructed and have been revised successively. *At present, the accuracy of nuclear data seems to be enough to design ordinary fission reactors. In the R&D of innovative fission reactors, however, more accurate data are needed for special nuclides, which have not yet played important roles in ordinary reactors.*

For example, more accurate fission and/or capture cross sections are needed for minor actinides (MA: ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, etc.) and long-lived fission products (LLFP: ^{79}Se, ^{93}Zr, ^{99}Tc, ^{107}Pd, ^{126}Sn, ^{129}I, ^{135}Cs) in the R&D of reactors aiming at the nuclear burning and/or transmutation of MA and LLFP. The accuracy of these cross section data is very poor at present compared to major actinides (^{235}U, ^{238}U, ^{239}Pu, etc.) and stable fission products, because the measurement and calculation for these cross sections are difficult. Moreover, cross section data for some material nuclides have become important and the improvement of these data is necessary. For example, the ^{210}Po and ^{210m}Bi production cross sections of ^{209}Bi is important in the R&D of innovative fast reactors with the lead-bismuth eutectic alloy coolant (LBC).

The following is a brief review on the physics-based R&D for innovative fission reactors, i.e. the R&D of nuclear database mentioned above, to which physicists are expected to contribute.

AII-3.1 Nuclear Data on MA and LLFP

Fission reactors produce MA and LLFP that are treated as nuclear wastes and are scheduled to be disposed of under ground at present. However, these MA and LLFP should be burned and/or transmuted into stable or short-lived nuclides to prevent our far-descendants from receiving the negative property. For this reason, the basic R&D of the nuclear burning/transmutation has been made in the world.

The nuclear burning/transmutation will be made by innovative nuclear systems, such as special fission reactors and accelerator-driven systems (ADS). In the ADS, there exist high-energy neutrons up to about 1 GeV, but most of neutrons used for the burning/transmutation have energy less than several MeV because they are generated by the evaporation and/or fission processes of nuclear reaction. Therefore, the important energy range of neutrons is below several MeV.

Fission and/or capture cross sections are most important. Other neutron cross sections are also important because a large amount of MA and LLFP influence the neutron spectrum in the system. The numbers, ν_p and ν_d, of prompt and delayed neutrons generated from the fission of MA are also important for the study on the reactivity and safety of system.

The measurement of these quantities is very important because the accuracy of theoretical calculation for these quantities is poor compared to that of measurement. Neutron energy regions, however, will be practically limited in the measurement, and the evaluation of these quantities will be performed by theoretical calculation in the energy regions where the measurement is practically impossible. Therefore, the improvement of theoretical calculation methods is also important.

The accuracy of about 5 % should be the first goal for the fission and/or capture cross sections of MA with half-lives longer than 100 y and LLFP. As for other quantities, the accuracy of about 10 % should be the first goal. In the case of MA with half-lives shorter than 100 y, the measurement is quite difficult, and the influence of these MA is expected to be small compared to MA with long half-lives. Therefore, the accuracy of about 10 % should be the first goal for the fission and capture cross sections. As for other quantities, the accuracy of about 30 % should be the first goal.
The nuclear data mentioned above should be provided timely, roughly speaking, within 10 y, because the second phase of the nuclear burning/transmutation R&D is expected to start within 10 y and will need the accurate nuclear data on MA and LLFP.

AII-3.2 Nuclear Data on Material Nuclides

Recently, LBC has attracted a great deal of attention in the R&D of innovative fast reactors, because it has various advantages such as chemical inertness, high boiling point, low neutron moderation and large scattering cross section in comparison with the sodium coolant. On the other hand, there are its disadvantages such as corrosiveness, high viscosity and ^{210}Po production. The nuclide of ^{210}Po is an α-ray emitter and its radio-toxicity is strong. It is worth noting that the lead-bismuth eutectic alloy is a promising candidate for both target and coolant materials in ADC.

The bismuth has only one stable isotope, i.e. ^{209}Bi, and ^{210}Po is produced in LBC by the neutron capture of ^{209}Bi and the following β decay: ^{209}Bi(n,γ)^{210g}Bi(β$^-$)^{210}Po. The half-lives of ^{210g}Bi and ^{210}Po are 5.01 d and 138 d, respectively, and the radioactivity of ^{210}Po in LBC causes a problem when a system with LBC is maintained. On the other hand, another α-ray emitter, ^{210m}Bi, is also produced in LBC by the neutron capture: ^{209}Bi(n,γ)^{210m}Bi. The half-life of ^{210m}Bi is very long, 3.04×10^6 y, so its radioactivity increases proportionally with neutron irradiation time and is related to the problem of nuclear wastes when a system with LBC is decommissioned. Therefore, the estimation of the radioactivity of these nuclides is very important for the application of LBC to nuclear systems. The estimation needs accurate data on the ^{210}Po and ^{210m}Bi production cross sections of ^{209}Bi. The present status of these experimental data is, however, poor both in quality and quantity, so these data in nuclear data libraries seem to have a problem in their reliability. Therefore, the accurate measurement of these cross sections is important to improve the accuracy of the nuclear data.

The accuracy less than 5 % should be the first goal for the ^{210}Po and ^{210m}Bi production cross sections. The improvement of nuclear data on material nuclides also should be performed timely as was described for MA and LLFP. In the case of material nuclides, generally speaking, the improvement within 5 years should be necessary.

AII-4 Present status of Worldwide R&D on the Utilisation of Thorium for Power Production

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Even though thorium was considered, since the beginning of the nuclear power development, to be the nuclear fuel to follow uranium, the thorium-based fuel cycles have been studied on a much smaller scale as compared to uranium or uranium/plutonium cycles. Worldwide, thorium resources are larger than those of uranium, and neutron yields of ^{233}U in the thermal and epithermal regions are higher than for ^{239}Pu in the uranium/plutonium fuel cycle. Thus the thorium fuel cycle increases the fissile resource through breeding of ^{233}U. Other reasons identified in past studies are the potential for fuel cycle cost reduction, the reduction in ^{235}U enrichment requirements, safer reactor operation because of lower core excess reactivity requirements, and safer and more reliable operation of ThO_2 fuel as compared to UO_2 fuel at high burnup due to the former's higher irradiation and corrosion resistance. The incidents that took place at TMI and Chernobyl and the growing waste management issues and potential for incineration of plutonium and other long-lived radiotoxic isotopes have revived interest in thorium fuel cycle. However, there are some disadvantages associated with thorium fuel cycle viz. the need for an external fissile material to start thorium cycle, the radiological problems associated with the fabrication of ^{233}U based fuel, higher residual decay heat, and difficulties in spent fuel reprocessing.

The current commercial reactors and all evolutionary and innovative reactors including molten salt reactors and hybrid systems are being studied for utilisation of thorium as part of a study initiated by IAEA. Few Co-ordinated Research Projects have been carried out with participation of different countries on the thorium based fuel cycle for incineration of plutonium and reduction of long-term waste toxicities.

AII-4.1 Present status of thorium fuel development

India

Globally, the availability of thorium is three times that of uranium. India has vast reserves of thorium (five times that of uranium) in contrast to modest quantity of uranium. Accordingly, while formulating the national programme, thorium has been envisaged as the fuel for the third and the largest phase of nuclear power programme. India has been using thorium in small quantities and has demonstrated various stages of thorium fuel cycle – extraction of metal, fabrication, irradiation, and reprocessing. In preparation for the third stage, developmental activities on thorium cycle have commenced right from inception of Indian nuclear programme with the irradiation of thoria `J' rods in the reflector zone of CIRUS reactor. Few thoria pins have been irradiated at the Pressurised Water Loop of CIRUS and in Dhruva. The thoria `J' rods of CIRUS have been reprocessed to make ^{233}U fuel for PURNIMA (liquid fuel) and plate type fuel for KAMINI reactor, the only reactor in the world to use ^{233}U as fuel. From Kakrapar Atomic Power Station onwards, thoria bundles are being used for initial flux flattening in Pressurised Heavy Water Reactors. Laboratory scale studies have been successfully carried out regarding the dissolution and reprocessing of thoria fuel. A Facility for Uranium Separation (FUS) is coming up at Trombay. Alternate fabrication processes to take care of the radiological problem of ^{233}U are being pursued. One of the thoria bundle irradiated in the power reactor has undergone PIE. Thus, work on thorium has been demonstrated on front end, reactor and back end side of thorium fuel cycle, albeit on a small scale.

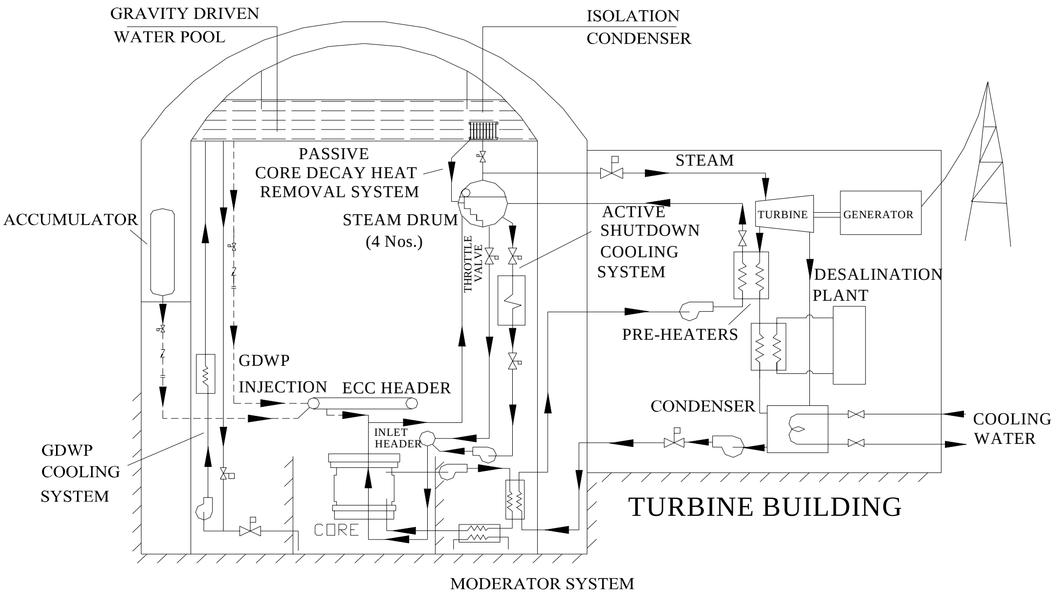

Fig. 1- Schematic of Advanced Heavy Water Reactor

AII-4.2 Advanced nuclear power systems for utilisation of thorium

The third stage of the Indian nuclear power programme envisages setting up of reactors based on Thorium-^{233}U cycle.

- Advanced Heavy Water Reactor
- Compact High Temperature Reactor
- Accelerator Driven System
- Advanced fuel cycle (front end and back end) facilities

India is currently engaged in the design of a thorium fuelled Advanced Heavy Water Reactor (AHWR) for generation of power. AHWR is a 300 MWe, vertical, pressure tube type reactor cooled by boiling light water and moderated by heavy water. The reactor is fuelled with (^{233}U-Th) MOX together with (Pu-Th) MOX. AHWR is nearly self-sustaining in ^{233}U. The design of AHWR is fine tuned towards deriving most of its power from thorium-based fuel, while achieving negative void coefficient of reactivity.

Heat removal from core is achieved by natural circulation of coolant. As shown in Fig. 1., water-steam mixture from core rises through the tail pipes to enter the steam drum. In steam drum, separated water at saturation temperature mixes with feed water and flows down the downcomers to the inlet header. From inlet header, water enters core through inlet feeder pipes.

AHWR incorporates several advanced features to increase its safety, reliability, and economics. These are enumerated below:

- Natural circulation heat removal under normal operation and shutdown conditions
- Low core power density
- Slightly negative void coefficient of reactivity
- Direct spray of Emergency Core Cooling System (ECCS) water into fuel pins during Loss Of Coolant Accident (LOCA)
- Accumulator with fluidic device for ECCS
- Gravity driven cooling system ensuring core cooling for three days following LOCA, without operator intervention
- Passive containment cooling and isolation
- Utilization of moderator heat
- Utilization of low grade heat for desalination

The AHWR fuel comprises a 54 pin composite fuel cluster containing 24 (Th-Pu)O_2 pins and 30 (Th-^{233}U)O_2 pins. It is planned to produce the requisite amount of ^{233}U by in-situ generation in the reactor. Hence it is planned to start with an initial core containing fuel with all 54 pins containing (Th-Pu)O_2. The ^{233}U in the spent fuel will be recycled back to the reactor after reprocessing.

In order to enhance the safety of nuclear systems, a three-pronged approach is followed. In the first step, the systems are so designed that probability of occurrence of an accident is reduced significantly below the level existing in current reactors. Next, even for very low probability events, the consequences are mitigated or reduced significantly so that no accident management is foreseen. Finally, the leakage through multiple barriers to the environment is also significantly reduced. The AHWR relies heavily on passive processes and components for its operation and accident mitigation.

A number of facilities have been either already built or are under construction or planning to validate the evolutionary concepts used in AHWR. The target core damage frequency is 10^{-7}/year or less. For achieving this, a series of potential high consequence events were visualized even when these events are Beyond Design Basis Accidents (BDBA) according to current standards. The safety features are so engineered that the consequences are mitigated. One such class of events is Anticipated Transients Without Scram (ATWS). Such events are analyzed in detail, even though, according to current standards, with two independent and diverse safe shut down systems, these events are BDBA. Results of safety analyses show that for a host of such events the fuel temperatures remain below their failure limits.

For ensuring very limited leakage to environment during accidents, a double containment is used. All potential leakage points are strengthened. These added safety features aim to qualify the design of AHWR for its siting close to populated regions without any need for evacuation planning.

AII-4.3 Compact High Temperature Reactor

Nuclear reactors will need to be increasingly utilised in the future for non-electrical high temperature process heat applications including production of hydrogen or secondary hydrocarbons as a substitute for primary fossil fuel, and for serving as components of compact power packs in remote areas not connected to grid system. To meet such needs a Compact High Temperature Reactor (CHTR) is being developed in Bhabha Atomic Research Centre (BARC).

CHTR is being designed with the following design guidelines
- Use of thorium based fuels
- Passive core heat removal by natural circulation of liquid heavy metal coolant
- Passive power regulation and shutdown mechanism.
- Passive rejection of entire heat to the atmosphere under accidental condition
- Compact design to minimise weight of the reactor

AII-4.4 Accelerator Driven Systems (ADS)

Accelerator driven systems (ADS) throw open several attractive possibilities for extending the Indian nuclear power programme. Such systems have a potential to convert fertile materials to fissionable materials, and to transmute the highly radioactive waste from conventional nuclear power plants to shorter-lived radionuclides, which do not require a very long-term storage under surveillance. In India, a beginning has been made in acquiring the necessary expertise to design and build linear accelerators as well as cyclotrons in India. A set of milestones has been identified, along the way of the development the technologies relevant for an accelerator driven system.

AII-4.5 Research in other countries

Canada

AECL has carried out studies on feasibility of various thorium fuel cycles in PHWRs. AECL has investigated many techniques for thorium fuel fabrication and has fabricated hundreds of thorium-based fuel elements. A number of full-sized, thorium CANDU fuelbundles (uranium-thorium, plutonium-thorium and pure thorium) have been irradiated at full power for years in the NRU research reactor. High quality thorium fuel bundles have been produced and the program of fuel production is continuing. Measurements of fundamental physical quantities important to the physics of thorium fuel cycles have been undertaken in the ZED-2 critical assembly. The properties of spent thorium fuel as a waste product have been studied and evaluated with respect to the Canadian nuclear fuel waste management program.

France

CEA in the 60's carried out few irradiation of thorium subassemblies in a power reactor, followed by reprocessing tests. More recently, CEA's involvement was mainly in the field of neutronics calculations of cores containing thorium, in view of either plutonium burning or long term thorium cycle sustainability:
- PWR with thorium-plutonium fuel or Th-^{233}U closed cycle
- Fast reactors with thorium-plutonium fuel or Th-^{233}U closed cycle
- ADS (like the Energy Amplifier, or molten salt reactors).

Germany

Germany in the past has used thorium in High Temperature Reactors in AVR (a pebble-bed research reactor) and THTR (300 MWe pebble-bed prototype reactor). The AVR operated for more than 2 decades using HEU-Th-fuel and the fuel achieved burnups of more than140 000 MW·d/tHM. The THTR was also operated with HEU-Thorium mixed oxide fuel, until it was shutdown mainly for political-economic reasons. Thus, a broad industrial experience existed in Germany in the field of the fabrication of thorium-based fuel. In the wake of plutonium accumulation, R&D institutions are now having a fresh look at the thorium fuel cycle the rationale being mainly to avoid the production and to burn plutonium.

Republic of Korea

As a part of the long-term national nuclear R&D programme, the advanced PWR project has carried out design simulation studies of various ThO_2-UO_2 mixed cores. The main objectives of these studies are on uranium resource savings and less radiotoxicity. Another national project performed at the Korea Atomic Energy Research Institute, which deals with the possibility of thorium fuel cycle, is the HYPER project, an accelerator-driven subcritical system with molten salt or metallic fuel with thorium as candidate materials.

Netherlands

The main interest of the Netherlands is to attain data for application of the thorium cycle in existing LWRs, and also, in the long term, in ADS and fast reactors. It participates in the coordination of the European Union project on "Use of Thorium Cycle as a Waste Management Option" and "Thorium Cycle: Development steps for PWR and ADS Applications". It will carry out calculations on Th-fuelled cores, Th-fuel pellet fabrication, generation of essential nuclear data, irradiation in the High Flux Material Testing Reactor and post irradiation experiments.

Russian Federation

Minatom's institutes, the Kurchatov Institute, and the other institutes of the Russian Federation, are working on reactor concepts, the physics of thorium systems, and technology. Conceptual investigations on the thorium fuel cycle are performed in the Kurchatov Institute (VVERT reactor, MSR, HTGR), IPPE (WWER type reactors, FR, MSR), VINIEF, ITEF (HWR, ADS). The technological problems of the thorium fuel cycle are studied and developed in the Belarus Institute, Radium Institute, IPPE, Kurchatov Institute, NIIAR.

United States of America

DOE is conducting four projects involving use of the thorium fuel cycle. All four projects are based on a once-through, proliferation resistant, high burnup, long refuelling cycle use of thorium in a light water reactor. Also of interest is the DOE Accelerator Transmutation of Waste(ATW) program on the ATW received final approval I mid November 1999.

AII-5 R&D TOWARDS RECOVERY OF URANIUM FROM SEAWATER

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Uranium is an inevitable natural resource for an atomic power plant to generate electricity. Uranium has been mined as a uranium ore. Seawater, however, contained 3 ppb of uranium as uranium oxide tricarbonate. The total amount of uranium in the seawater reaches 4.5 billion tons, estimated from the product of the uranium concentration in the seawater and the seawater volume, which occupy more than 99 %, of the Pacific, the Atlantic, and the Indian Ocean. This amount is equivalent to a thousand times the discovered uranium in mines.

The recovery of uranium has been carried out by adsorption methods. In this method, the adsorbent needs the high selectivity and capacity for uranium adsorption in the seawater. It was found that hydrous titanium oxide is a god material for the recovery of uranium from seawater. Then, screening researches were carried out to evaluate the many kinds of uranium adsorbents and concluded that the amidoxime was a promising functional group for recovery of uranium from seawater. We have developed a high performance adsorbent (with amidoxime) with radiation-induced graft polymerization and the collection of 1 kg of uranium from seawater was demonstrated. In the recovery system, the stacks of fabric adsorbent was changed to a braid type adsorbent to improve the contact between the adsorbent and the seawater.First the temperature and the depth of the sea of the area, considered for the recovery of uranium, was investigated.

AII-5.1 Recovery of Uranium from Seawater with Adsorbent

AII-5.1.1 Hydrous titanium oxide adsorbent

The first experimental plant for recovery of uranium from seawater was operated by the Agency for Natural Resource and Energy, the Ministry of International Trade and the Industry and Metal Mining Agency of Japan from 1981 to 1988. The adsorbent in this plant was hydrous titanium oxide and its adsorption ability was 0.1g-U/kg-adsorbent (hereafter termed kg-ad).

AII-5.1.2 Amidoxime adsorbent synthesized by radiation induced graft polymerization

The Takasaki Radiation Chemistry Research Establishment and the Japan Atomic Energy Research Institute have been studying the synthesis of amidoxime adsorbent with radiation-induced graft polymerization and subsequent chemical modification since 1981.

Radiation-induced graft polymerization is a radiation processing of polymer. In this processing, the trunk polymer, like polyethylene, is irradiated with electron beams and then contacted with the reactive monomer. The graft chains propagate from the active species in the irradiated trunk polymer.

The fibrous amidoxime adsorbent has been synthesized as follows:

- Nonwoven fabric made of fibrous polyethylene as a trunk polymer was irradiated with electron beams in nitrogen gas.
- Irradiated nonwoven fabric was immersed into the monomer solution which was composed of 50 % dimethyl sulfoxide, 35 % acrylonitrile, and 15 % methacrylic acid after oxygen gas in the monomer solution was substituted with nitrogen gas. The irradiated nonwoven fabric and the monomer solution were warmed up to 40 °C. This temperature was maintained for 4 h for the graft polymerization. The degree of grafting, which was calculated by the increasing weight, reached 150 %.
- Grafted nonwoven fabric was reacted with 3 % hydroxylamine solution at 80 °C for 1 h. In this reaction, the cyano groups in the polyacrylonitrile moiety of the grafted nonwoven fabric were converted into amidoxime in the yield of 95 %.

AII-5.1.3 Preliminary marine experiment in Mutsu

The synthesized fibrous adsorbent in the size of square with 2 cm one side revealed the uranium adsorption of 1.0 g-U/kg-ad after 30 d soaking in the offing of Mutsu (Aomori Prefecture) in 1996. The stacks of these adsorbent, 16 cm x 30 cm x 30 cm, collected 10 g of uranium from seawater in 1997. The average adsorption of uranium in this experiment was 0.3 g-U/kg-ad.

The decrease of fiber diameter from 30 μm to 13 μm in the non-woven fabric was effective in the improvement of adsorption ability. After 30 d soaking in the seawater, newly synthesized adsorbent from the non-woven fabric made of 13 μm polyethylene-coated polypropylene in the size of square with 2 cm side adsorbed 2.0 g-U/kg-ad after 30 d soaking in the offing of Mutsu in 1998.

AII-5.1.4 Marine experiment for 1 kg-U collection in Mutsu

The stacks, 16 cm x 30 cm x 30 cm, made of new non-woven fabric adsorbent were dipped in Mutsu offing. The stacks of 144 pieces were put into a square adsorbent cage, 4 m x 4 m x 45 cm. Three adsorbent cages were hang down from the square floating frame, 8 m side on the sea surface. During the three years, twelve soaking experiments, in which 1800 stacks were immersed, were carried out and 1 kg of uranium was adsorbed on the adsorbent. The average ability of the adsorbent was 0.5 g-U/kg-ad for 30 d soaking. The uranium adsorption was correlated with the temperature of seawater and the wave height. This is because the high temperature of seawater accelerates the chemical adsorption of uranium on the adsorbent. The motion of waves is transferred to the adsorption cage through the hanging ropes and the motion of up and down of adsorption cage realizes the effective contact between seawater and adsorbents.

AII-5.1.5 Braid type adsorbent

Since 2001, a braid type adsorbent has been developed. This adsorbent was manufactured by braiding the adsorbent fiber around the porous polypropylene float, 2cm in diameter. The length of fiber adsorbents surrounding the float was 10 cm. The braid type adsorbent can stand on the bottom of the sea and does not need the adsorbent cage when it is soaked in the sea.

In 2002 and 2003, the uranium adsorption of braid type adsorbent was evaluated in the sea of Okinawa area. The average ability of the adsorbent became 1.5 g-U/kg-ad for 30 d soaking. The nonwoven fabric adsorbent in the size of square with 2 cm one side attached on the braid type adsorbent showed 3.0 g-U/kg-ad. Temperature of the seawater in Okinawa was 30 °C (10 °C higher than that of Mutsu area). The rise of 10 degrees in the seawater temperature increased by a factor 1.5 the uranium adsorption for the nonwoven fabric adsorbent(square size with 2 cm side). Therefore, the braid type adsorbent had 2 times higher adsorption ability of uranium in seawater than the stacks of nonwoven fabric adsorbent ,due to the better contact between seawater and adsorbent.

The Environmental Science Research Laboratory, the Central Research Institute of Electric Power Industry reported that 1,200 t-U/y could be collected from seawater if the 2.1 millions braid type adsorbents, 60 m length, were set in 134 km^2 sea area with an interval of 8 m. Since the higher temperature of seawater accerelate the uranum adsorption, the sea area should have the Japan Current, the depth from 100 m to 200 m, and no fixed net. The sea site (6000 m^2) proposed for the recovery of uranium from seawater is located between Nansei islands and Tosa bay.

AII-5.2 Cost Estimation

The recovery cost of uranium, by using stacks hanging system, was estimated at 400-500 $/kg-U in the following conditions: The scale of uranium recovery is 1,200 t/y. The adsorbent can collect 6 g-U/kg-ad for 60 d soaking. The adsorbent stacks can be repeatedly used 20 times.

Recovery cost of uranium from seawater by using braid type adsorbent has been calculated and the result will be obtained at end of fiscal year 2004.