GEN IV Reactors: Where we are, where we should go

Giorgio Locatelli
University of Lincoln
Lincoln School of Engineering
Brayford Pool - Lincoln LN6 7TS
United Kingdom
Phone;+441522837946
glocatelli@lincoln.ac.uk

Mauro Mancini
Politecnico di Milano
Department of Management,
Economics & Industrial Engineering
Via Lambruschini 4/B Milano – Italy
Phone;+390223994057
mauro.mancini@polimi.it

Nicola Todeschini
Politecnico di Milano
Department of Management,
Economics & Industrial Engineering
Via Lambruschini 4/B Milano – Italy
Phone;+390223994096
nicola.@mail.polimi.it

Abstract – *GEN IV power plants represent the mid-long term option of the nuclear sector. International literature proposes many papers and reports dealing with these reactors, but there is an evident difference of type and shape of information making impossible each kind of detailed comparison. Moreover, authors are often strongly involved in some particular design; this creates many difficulties in their super-partes position. Therefore it is necessary to put order in the most relevant information to understand strengths and weaknesses of each design and derive an overview useful for technicians and policy makers. This paper presents the state-of the art for GEN IV nuclear reactors providing a comprehensive literature review of the different designs with a relate taxonomy. It presents the more relevant references, data, advantages, disadvantages and barriers to the adoptions. In order to promote an efficient and wide adoption of GEN IV reactors the paper provides the pre-conditions that must be accomplished, enabling factors promoting the implementation and barriers limiting the extent and intensity of its implementation. It concludes outlying the state of the art of the most important R&D areas and the future achievements that must be accomplished for a wide adoption of these technologies.*

I. INTRODUCTION

This paper illustrates the most relevant information to understand strengths and weaknesses of each design amenable to a specific GEN IV technology. The aim is to classify and to compare the design according to the most important technical and economical drivers, underlining the most important R&D areas, the enabling factors and the barriers, influencing their implementation.

II. LITERATURE REVIEW: GEN IV TECHNOLOGIES

The development of GEN IV technologies is coordinated by GIF. This organization publishes the referential documents for GEN IV reactors (1). This section introduces the six technologies selected by GIF (VHTR, SFR, SCWR, GFR, LFR and MSR). For each design it presents an introduction, the principal strengths and weaknesses, the main areas of R&D.

II.A. VHTR

VHTR is a thermal reactor cooled by helium (in gaseous phase) and moderated by graphite (in solid phase). The major characteristic is the high OCT (750°-850° with the target of overcoming 1.000°).

The main advantages of this technology are:

- high maximum temperature of thermodynamic cycle to increase energy efficiency and adopt a direct (without heat exchanger) helium Brayton cycle;
- the option of high temperature cogeneration for the production of hydrogen through thermochemical process or medium temperature cogeneration for industrial use.

Helium is a radiologically and chemically inert gas stable in each interesting thermodynamic conditions. Graphite has high thermal conductivity and elevated specific heat capacity, which is useful in accidental situations because they slow down transitory. The disadvantage is the presence of this material in spent fuel requiring specific and innovative decommissioning.

Pebble bed core and prismatic block core are the two options in consideration for VHTR. The main differences are:

- the pebble bed has the least power density, increasing construction cost, but the neutronic stability of the core is better, increasing life of the power plant;
- the operating temperature of the pebble bed is the lowest allowing the adoption of conventional steel for the vessel;
- pebble bed has an higher capacity factor than prismatic block because of the online refueling;
- prismatic block produces less dust (that could damage pipelines and heat exchanger) than pebble bed (2).

VHTR is a modular SMR. The modularization could partially compensate the loss of economies of scale. This selection reduces the total overnight cost (3).

TABLE I Summary of VHTR projects

	VHTR target	HTR-PM	NGNP	GT-MHR	GTHTR300C
Thermal/electrical power (MW)	600/300	2x250/210	600/240	600/286	600/274
Core layout	Prismatic block/Pebble bed	Pebble bed	Prismatic block/Pebble bed	Prismatic block	Prismatic block
Fuel	TRISO cladded with ZrC	TRISO	TRISO	TRISO	TRISO
OCT (°)	850	750	750	850	850
Outlet core pressure (MPa)	9	7	-	7	7
Thermodynamic cycle	Brayton	Rankine	Rankine	Brayton	Brayton
Thermal efficiency	50%	42%	40%	48%	46%
Byproduct	Hydrogen	-	Industrial cogenerative application	Hydrogen Desalinized water	Hydrogen
Economics	High	Similar to LWR	Similar to LWR	Higher than LWR	Higher than LWR

II.A.1. HTR-PM and NGNP

HTR-PM and NGNP are two VHTR operating in the lowest range of temperature for this technology (the OCT is 750°) and adopt an indirect (with IHX) subcritical Rankine cycle. This solution increases construction cost and decreases energy efficiency but promotes the technical feasibility because the proven technology.

HTR-PM is a short term chinese project of a pebble bed core. Two modular cores of 250 MWth generate the thermal power. The reference power plant produces 210 MWe by a single steam turbine (4). The construction costs estimation is similar to current LWR, despite the cost of the vessel (bigger because the low core power density) is height times higher than a conventional RPV (5).

NGNP is the US program for the development of a VHTR. The design of the core has not been defined yet. The two options are a pebble bed core and a prismatic block core. The target thermal power of the NGNP is 600 MWth. The main aim of the design is the cogeneration (2) (6). The reactor can produce heat by steam (at 17 MPa and 540°) or by helium (at 9,1 MPa and 900°), depending on the cogenerative application. The most promising studies are: the production of hydrogen through steam methane reforming (7), of gasoline by methanol (8) and of ammonia (9). The program foresees the construction of a prototype in Idaho within 2021.

II.A.2. GTHTR300C and GT-MHR

GT-MHR and GTHTR300 adopt a prismatic block core and high OCT (850°). This condition allows the usage of a direct helium Brayton cycle permitting higher energy efficiency and less construction cost than a Rankine one.

GT-MHR is an American-Russian project. It has a core thermal power of 600 MWth. The layout of the PCU is vertical and integrated in a single vessel; this solution decreases the energetic losses. GT-MHR has the highest energy efficiency among the VHTR allowing the adoption of a dry cooling for the thermodynamic cycle (10) (11). A desalination plant is connectable with the Brayton cycle without a reduction in its energy efficiency. The estimated cost of desalinated water is the least between fossil fired power plant and other VHTR (12). No GT-MHR prototype has been planned or is in construction.

GTHTR300 is a Japanese project with a core thermal output of 600 MWth. The PCU layout is horizontal. Gas turbine and heat exchangers are in separated modular vessel. These solutions reduce construction cost and facilitate the maintenance (13). No prototype of this design has been planned but an experimental high temperature reactor (HTTR) is operational in Japan (14).

A cogenerative plant for hydrogen production through I-S (Iodine-Sulfur) thermochemical process has been projected for these designs. The OCT increases of 100° to 950°, raising the efficiency of the hydrogen production process, and an heat exchanger is placed at the outlet of the core, transferring heat to the cogenerative plant. In GTHTR300C (GTHTR300 for hydrogen production) the IHX is parallel to the gas turbine with subdivision of the flow rate (15). In GT-MHR the heat exchanger is in series with the PCU decreasing the inlet temperature of the gas turbine (16).

II.A.3. PBMR

PBMR was a South African program for the development of a pebble bed core VHTR. This reactor has a thermal power of 400 MWth, a OCT of 900° and adopts a direct helium Brayton cycle for electricity production. It was planned the construction in Koeberg of a prototype with startup in 2014. The program has been terminated in 2010. The main critical issues were the helium gas turbine (which is in an experimental deployment phase) and the rise in core thermal power (the initial design was 267 MWth) without salient engineering modification (17) (18).

II.B. SFR

SFR is a fast reactor cooled by sodium (in liquid phase). It is the most investigated fast reactor.

The main advantage of this technology is fast spectrum that can convert fertile material in fissile increasing about fifty times the efficient usage of nuclear fuel. This reactor could be burner, transmuting actinides to reduce the production of HLW, converter, with a breeder ratio (ratio between fissile material produced and consumed) near one, or breeder with a net production of fissile material. It requires a closed fuel cycle. Two options are under examination: advanced aqueous process and pyro-metallurgical process.

Sodium is a good coolant with high specific heat without pressurization of the vessel, low melting point (98°) and low corrosiveness. The boiling point (883°) restricts the maximum temperature of thermodynamic cycle. Sodium reacts with water and air in the interesting range of temperature. SFR adopts an airtight primary circuit, high availability steam generators and an intermediate circuit with not radioactive sodium between RPV and PCU for reducing this risk.

The main options for the reactor layout in SFR are loop and pool. The main differences are:

- in a pool layout RPV contains the radioactive sodium and its leak is unlikely while in a loop layout pipelines and heat exchanger are out of the vessel and require special coverings;
- loop layout is more compact and less expensive than pool one and with an easier maintenance;
- pool layout has more thermal inertia allowing slower transitory in accidental situations (19);
- pool layout is the most experimented configuration.

The economics of SFR is low. The presence of intermediate circuit causes high construction cost. The R&D aims at simplify the primary and the intermediate circuits through adoption of high performance steels and the development of economies of scale.

II.B.1. KALIMER and JSFR

JSFR and KALIMER are two SFR of medium-large dimension for the production of electricity though subcritical Rankine cycle and for the management of actinides.

JSFR is a large power plant (the core thermal power is 3.570 MWth), adopting a loop layout reactor. It uses MOX (oxide fuel) with TRU and the closed fuel cycle is based on advanced aqueous process (20). JSFR can be a breeder reactor (maximum ratio of 1,2). Main solutions for reducing construction cost at the levels of current LWR are: integration between primary pump and primary heat exchanger in a single module (21), the reduction of RPV dimensions through high performance materials for walls and reflector, the adoption of innovative high reliable steam generator (called double wall SG) and the decrease of length of pipelines though the adoption of high conductivity and elevated fatigue strength materials (22) (23). Two experimental SFR (Monju and Joyo) are in operation in Japan. The development plan of JSFR foresees a prototype in 2025 and a wide commercial adoption in 2050 (22).

KALIMER is a converter medium power plant (core thermal power is 1.523 MWth), allowing more flexibility than a large one. It adopts a pool layout of the reactor and U-TRU-10%Zr as nuclear fuel. This metal alloy fuel requires pyro-metallurgical process for closing fuel cycle (24). The solutions for decreasing construction cost are reduction of pipelines similarly to JSFR and of RPV dimension through innovative core internals (25). The program predicts the construction of a prototype in 2028 and a wide adoption in 2040 (26).

II.B.2. SMFR

SMFR is an American project for the realization of a SMR with a core thermal power of 125 MWth. The application is the supply of electricity in remote area or developing countries without a connection with electrical grid. This niche market requires reactors without conventional refueling scheme, which need fast spectrum and quite corrosive coolant allowing long autonomies of the core. This is an unexplored application for precedent nuclear generations.

SMFR employs metal alloy fuel and closed fuel cycle based on pyro metallurgical process, adopting an S-CO_2 (Supercritical Carbon Dioxide) Brayton cycle for power production. It is more compact, more efficient and less expensive than a conventional Rankine cycle but less experimented. The reaction between carbon dioxide and sodium requires an intermediate cycle employing not radioactive sodium to reduce risks of LOCA (27). A desalination plant can be connected to the power plant without a reduction in its energetic efficiency (28). No prototype of this technology has been planned.

TABLE 2 Summary of SFR projects

	SFR Target	JSFR	KALIMER	SMFR
Thermal/Electrical Power (MW)	1.000-5.000/ 400-2.000	3.570/1.500	1.523/600	125/50
Fuel	Oxide- Metal alloy	TRU-MOX	U-TRU-10%Zr	U-TRU-10%Zr
Fuel cycle	Aqueous - pyrometallurgical	Aqueous	Pyrometallurgical	Pyrometallurgical
Breeder ratio	0,5-1,3	1,03-1,2	1,0	1,005
RPV Layout	Loop - Pool	Loop	Pool	Pool
OCT (°)	530-550	550	545	510
Thermodynamic cycle	Brayton	Rankine	Rankine	Brayton
Working fluid	S-CO_2	Steam	Steam	S-CO_2
Energy efficiency	> 40%	42%	39,4%	38%
Economics	Medium	Less than LWR	Less than LWR	Less than LWR

II.C. SCWR

SCWR is a thermal/fast reactor cooled by supercritical water. It is considered an evolution of actual BWR because of similar plant layout and size, same coolant and identical main application, which is electricity production.

The main differences with a BWR are:

- higher energetic efficiency (10% respect to the average performance of current LWR);
- reduction of flow rate of water for cooling the reactor, thanks to the superior enthalpy of this coolant allowing the adoption of smaller pipelines and pumps;
- simplification of the plant layout because single phase coolant eliminate steam dryers and recirculation systems.

For these reasons, specific construction and O&M costs seem below the average of current LWR.

The possible core configurations employ classical RPV or pressure tubes. RPV input core water acts as coolant and moderator in RPV layout. This reduces reactor components but requires a complex flow of water in the core due to relevant density and thermodynamic change of supercritical water. In pressure tubes layout coolant and moderator are different fluids, eliminating the above mentioned criticality. This layout requires additional devices, like calandria tank and pressure pipelines.

The main criticality is the high corrosiveness of supercritical water. No material has yet been identified for fuel cladding or core internals, which are subjected to irradiation, high pressure and oxidation (29) (30).

II.C.1. HPLWR and Super LWR

Super LWR and HPLWR are thermal RPV SCWR optimized for electricity production. They have same electrical output (1.000 MWe), identical OCT (500°) and similar layout of PCU. These reactors adopt water rods for moderating nuclear reaction and uranium oxide (UO_2) as fuel. The main difference concerns the layout of the core: HPLWR adopts a three-pass core (31) while Super LWR uses a simpler two-pass core (32). Both of the layouts allow the usage of conventional steel for the vessel. Super LWR is a Japanese project. The program foresees the construction of an experimental reactor in about 2020 to demonstrate feasibility (32). HPLWR is a European program. The objective is to develop the first commercial SCWR before 2035 through a continuous progress in LWR field (33) (34).

II.C.2. Super Fast LWR

Super Fast LWR is based on the design of Super LWR sharing its PCU. It adopts, however, a fast spectrum core fuelled with MOX. Supercritical water is worse moderator than subcritical water and allows this solution. The fast spectrum core has a tight lattice and requires zirconium hydride, reducing the obtainable breeder ratio but insures a negative void coefficient (35). The main benefits are a higher core power density than thermal solution, which reduces construction cost, and ability of actinides management, allowing the usage of this power plant as TRU elements burner (36). Critical issues are elevated irradiation damage for core internals and fuel cladding, which complicates further the selection of a suitable material, and a problematical behavior in accident situation (37). The second phase of deployment for this design is in progress (36).

II.C.3. CANDU-SCWR

CANDU-SCWR is a thermal pressure tubes reactor fuelled with uranium dioxide (or thorium as secondary option) cooled by supercritical water and moderated by heavy water. It has an higher OCT (625°) than former SCWR, which increases energy efficiency of supercritical Rankine cycle and allows medium temperature cogenerative application for industrial use or for hydrogen production through low temperature thermochemical processes. Despite the referential plant thermal power is 2.540 MWth, the pressure tubes reactor allows power modification through reduction of fuel channel in the core. It facilitates, besides, the insertion of reheaters in Rankine cycle (38). Thermo-economic evaluation of PCU isn't fully completed (39). CANDU-SCWR is CANDU proposal reactor for 2025-2080 time range but requires a prototype for experimental test of the design (40).

TABLE 3 Summary of SCWR projects

	SCWR Target	HPLWR	Super LWR	Super Fast LWR	CANDU-SCWR
Thermal/Electrical Power (MW)	3.860/1.700	2.300/1.000	2.300/1.000	2.358/1.000	2.540/1.220
Spectrum	Thermal/Fast	Thermal	Thermal	Fast (BR ≈ 1)	Thermal
Fuel	UO_2/MOX	UO_2	UO_2	MOX	UO_2/Th
Moderator	Light water	Light water	Light water	Light water	Heavy water
Reactor layout	RPV	RPV	RPV	RPV	Pressure tubes
OCT (°)	550	500	500	500	625
Core pressure (MPa)	25	25	25	25	25
Energy efficiency (%)	44	43,5	43,8	43,8	48
Economics	High	Higher than LWR	Higher than LWR	Higher than LWR	Higher than LWR

II.D. GFR

GFR is a fast reactor cooled by helium (in gaseous phase). The aim of the technology is to put together a high temperature reactor and a fast spectrum core.

The main advantages of this reactor are:

- an high OCT (850°) adopting an elevated efficiency helium Brayton cycle for electricity generation and the use of produced heat for cogenerative applications like hydrogen fabrication;
- the ability of actinides management for an efficient exploitation of nuclear fuel (with a converter operating mode).

GFR is the fast reactor with highest OCT. Some R&D areas (like BOP) are in common with VHTR and SFR. This reactor requires a very challenging nuclear fuel with a specific reprocessing process to close the fuel cycle (41) (42). The core internals are exposed at high temperature and elevated irradiation requiring high performance ceramic materials. Helium doesn't react with air or water and it is transparent, allowing the adoption of direct thermodynamic cycle and simpler inspection devices, but has low specific heat and requires pressurization. GFR has more core power density than VHTR and it can't use graphite for increasing thermal inertia of core like the mentioned thermal reactor. The technology under examination requires complex, innovative and expensive security system for insuring pressurization of the vessel and cooling of core in accidental situations.

The referential design is 2.400 MWth reactor, exploiting economies of scale to reduce specific construction costs, adopts GT-MHR RPV and an indirect helium (or helium/nitrogen for simplifying gas-turbine design) Brayton cycle with a bottoming steam Rankine cycle. This solution simplifies primary circuit and security systems but reduces energy efficiency and disadvantages cogenerative applications (43) (44). The construction of a demonstrative reactor, called Allegro, is planned in Europe to evaluate the feasibility of the GFR for commercial use. The hypothesized startup date is 2026 (45) (46).

II.E. LFR

LFR is a fast reactor cooled by pure lead or LBE. It is a liquid metal reactor, similar to SFR, for electricity production and actinides management.

Usage of lead involves several advantages:

- it has excellent neutron and thermo-fluid-dynamic property facilitating the establishment of a natural convection in vessel simplifying heat transfer system and increases intrinsic safety;
- it doesn’t react with air and water, allowing elimination of the intermediate system;
- it has an high boiling point that doesn’t limit the maximum temperature of thermodynamic cycle.

Lead and LBE could be used as coolant. Pure lead is less expensive, more abundant and quite less corrosive (especially at high temperature). The main advantage of LBE is low melting point (125°) respect to pure lead one (327°), reducing the risk of core freezing and related damages during transitory or shut down of reactor. This coolant, however, produces radioactive isotope of polonium (^{210}Po), which must be eliminated through complex treatment of coolant itself and of primary circuit (47). Both coolants corrode structural materials (especially core internals and fuel cladding) through precipitation a low temperature, dissolution at high temperature and erosion caused by movement of the fluid. Main countermeasure for low-medium working temperature is rigorous introduction of oxygen to form a protective oxide layer on exposed steel (48) (49). LFR requires advanced alloys for higher thermic conditions (50). Devices working in flow lead (like pumps or fuel handling system) are critical because they need innovative technologies.

II.E.1. ELSY

ELSY is a European program for deployment of a fast reactor competitive in UE energy market. It adopts a medium size (electricity output of 600 MWe) pool layout reactor, increasing thermal inertia of the core and facilitates the establishment of natural convection in accident situations, and using pure lead as coolant. The high melting point of fluid and its corrosiveness (related at working temperature) limit the difference between inlet and OCT, increasing size of primary circuit and construction cost. To cope with this problem, it employs innovative component, like spiral wound steam generator in RPV and integrated to mechanical pump for forced convection of the coolant. The adopted thermodynamic cycle is a proven subcritical steam Rankine cycle without intermediate circuit (51). The selected fuel is MOX with an advanced aqueous reprocessing process and a fuel handling machine, working in gaseous environments in the superior part of RPV (52). ELSY can work as converter for an efficient use of nuclear fuel or as burner for transmuting TRU (53). The aim of the successive process, called LEADER (Lead-cooled European Advanced Demonstration Reactor), is to realize a prototype reactor within 2020 with a wide adoption of ELSY in about 2040 (51).

II.E.2. SSTAR

SSTAR is a US project for the design of a very small modular reactor (core thermal power of 45 MWth) to supply electricity in remote or developing areas. The power plant is fully modularized and each module (RPV included) is transportable by railway or ship. The construction method is innovative because the site receives completed module, ready to be assembled. The operator has not access to nuclear fuel and the EPC attends to refueling and control of the reactor. This solution increases potentially market for this power plant (54).

SSTAR is a pool layout reactor cooled by natural convection of pure lead and fuelled with innovative nitride fuel, requiring a pyroprocess for closing fuel cycle, without conventional refueling scheme. It is planned a only refueling in the service life of the reactor, requiring a fuel cladding resistant to irradiation, corrosiveness and high temperature for long periods. The refueling is simplified by a single removable fuel assembly. The natural convection influences the layout of the RPV, having a stretched shape, and of the heat exchangers that are inserted in vessel. The higher outlet temperature core (560°) than ELSY allows the adoption of S-CO$_2$ Brayton cycle, being more compact than a conventional solution. It permits the coupling of PCU with a desalination plant, key feature in several remote areas, without reduction in energy efficiency (55) (56).

A prototype design, called SUPERSTAR (SUstainable Proliferation-resistance Enhanced Refined Secure Transportable Autonomous Reactor), has been proposed for surpassing some technical problem preventing a near term deployment of SSTAR. This experimental reactor uses metal alloy fuel and subcritical steam Rankine cycle for electricity production with a lead intermediate circuit (57) (58).

TABLE 5 Summary of LFR projects

	LFR Target	ELSY	SSTAR
Thermal electrical power (MW)	125-3.600/ 60-1.620	1.400/600	45/19,8
Coolant	LBE - Pure lead	Pure lead	Pure lead
Convection	Forced - Natural	Forced	Natural
Fuel	Nitride fuel	MOX	Nitride fuel
Fuel cladding	Ceramic	T91	Coated HT9
Breeder ratio	1.0	1.0	1.0
OCT (°)	800°	480°	560°
Thermodynamic cycle	S-CO$_2$ Brayton	Rankine	S-CO$_2$ Brayton
Energy efficiency (%)	45%	42%	43,8%
Byproduct	Hydrogen Desalinized water	/	Desalinized water
Economics	Medium	Similar to LWR	Similar to LWR

II.F. MSR

MSR is a fast or thermal (with graphite as moderator) reactor cooled by molten salts (in liquid phase). In this technology the nuclear fuel is dispersed in the coolant and therefore it is in liquid phase.

Liquid fuel main advantages are:

- doesn't need fuel fabrication, reducing fuel cycle cost and critical issues;
- has an homogenous composition, allowing addition of any fissile materials without formation of hot spot and a great flexibility in fuel cycle (it is possible to make a breeder cycle in thermal spectrum);
- hasn't problems related to resistance of fuel cladding permitting high working temperature;
- allows online refueling;
- increases intrinsic safety of power plant because of reactivity condition, a lower mass of fissile materials and the option to entirely remove the nuclear fuel from the core in accidental situations.

Molten salts have thermal stability at high temperature (superior to 800°), high specific heat without the need of vessel pressurization and they don't react with air or water. They have, however, a high melting point (about 500°), requiring an intermediate cycle for the coupling to the PCU. The composition of the coolant has to be optimized from an economical and neutronic point of view (the most promising salts are fluorides). The chemistry and the thermo-fluid-dynamic behavior of irradiated molten salts are partially unknown. This coolant, besides, requires a treatment for removing lanthanides, noble gas and noble metal, which are created by the nuclear reaction. The design of a simple and economic process is fundamental for demonstrating the feasibility of MSR (59) (60).

Molten salts are corrosive. Presence of impurities and oxidative fission products increase their corrosiveness. The oxide protective layer, which is formed by the steel with the addition of chrome, silicon and aluminum, is useless in this environment. The elements that resist to molten salt chemical attack are refractory metals and nickel. Most promising materials for the primary circuit of MSR are advanced nickel alloys. Other critical issues concern primary pumps, operating in very corrosive conditions, and heat exchangers, which could be blocked by noble metals (61) (59).

II.F.1. MSFR

MSFR is a fast breeder reactor, using liquid fuel. Such fuel is a mixture of fluorides of thorium and uranium (UF_4, ThF_4) dispersed in a lithium fluoride (^7LiF) molten salt. A fast spectrum core eliminates the critical issues linked to graphite, increases its intrinsic safety. This has negative temperature and reactivity coefficients, and reduces the potentiality of salt treatment plant, which can be offline while the thermal breeder reactor requires an online process (62). Thermal reactor could reach superior OCT and requires an inferior amount of fissile material.

The core is a cylindrical element with length equal to radius. The most promising structural material for this component is a ternary alloy of nickel, tungsten and chrome. The applications of this MSR are electricity production (1.300 MWe) and actinides management. The OCT is about 700°. The intermediate circuit, employing a less expensive molten salt with lower melting point, connects primary circuit with PCU. The design of this system isn't defined. The production of fissile material is increased by the introduction of a fertile blanket in the core filled with a mixture of ThF_4 and LiF (63).

The MSFR is a French project and an evolution of TMSR (Thorium Molten Salt Reactor). The program predicts the construction of a prototype in about 2020 and a wide adoption in 2040 but it seems unrealistic since the design has relevant technical challenges (64).

II.F.2. PB-AHTR

PB-AHTR adopts a solid fuel. This solution eliminates the aforementioned advantages linked to the use of a liquid fuel but reduces the corrosiveness of the coolant, because of the absence of fission products in the fluid, and eliminates the salt treatment plant. The design is similar to VHTR, like the high OCT and cogenerative industrial applications.

PB-AHTR is a thermal reactor moderated by graphite, cooled by a ^7LiF-BeF_2 molten salt and fueled whit a pebble-type element, similar to PBMR. The high specific heat of molten salt and an innovative layout of core internals increase in power density of the core respect to VHTR that involves a rise of reactor thermal power to 900 MWth reducing vessel size; since the vessel is not pressurized is possible to reduce its thickness (65). This design requires an innovative pebble fuel, requiring experimentation (66). The PCU, based on a multiple reheat helium Brayton cycle, is connected with the reactor by an intermediate circuit (67). High OCT (704°) and small size allow a medium temperature cogenerative application. The adoption of a Brayton cycle permits the coupling with a desalination plant without a reduction in energy efficiency (68). The construction of an AHTR prototype is for 2025 (69).

TABLE 6 Summary of MSR projects

	MSR Target	MSFR	PB-AHTR
Thermal/electrical (MW)	2.000/1.000	3.000/1.300	900/410
Spectrum	Thermal (possible breeder)	Fast	Thermal
Moderator	Graphite	-	Graphite
Fuel	Liquid UF_4-ThF_4	Liquid UF_4-ThF_4	Solid TRISO Pebble
Primary molten salt	NaF-ZrF_4	^7LiF	^7LiF-BeF_2
Salt treatment plant	On-line	Offline	-
OCT (°)	700° (850° for hydrogen production)	700°	704°
Thermodynamic cycle	Multiple reheat helium Brayton cycle	Multiple reheat helium Brayton cycle	Multiple reheat helium Brayton cycle
Energy efficiency (%)	44-50	45-55	46
Byproduct	Hydrogen Desalinized water	Low temperature cogeneration	Low/medium temperature cogeneration
Economics	Low	-	Superior to LWR

III. TAXONOMIES OF GEN IV PROJECTS

The overview in the previous chapters introduces several projects amenable to GEN IV with very different features and possible applications. A rigorous assessment requires a classification of the technologies according to main technical and economic drivers. The scientific tool utilized is the taxonomy.

The first taxonomy (see Fig. 1) is an improvement of the classification of GEN IV technologies elaborated by GIF. It adopts three levels tree architecture for a univocal identification of each designs. The first level classifies the designs according to neutron spectrum; this is a fundamental driver, influencing the applications of the reactor, its layout and its fuel cycle. The second level is based on typology of core coolant, influencing reactor working temperature and selection of materials. The single projects are indexed in the last level.

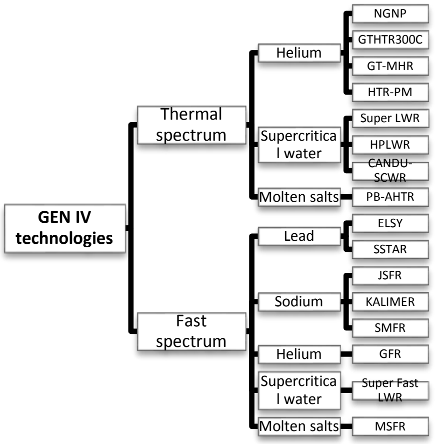

Fig. 1. Levels: neutron spectrum, coolant, designs

The successive taxonomies adopted in this work are based on matrix architecture, categorizing project according to two drivers. The first classification is according to the economics and energy efficiency (Table 7).

TABLE 7 Drivers: economics, energy efficiency

		Energy efficiency		
		High (η>45%)	Medium (45%<η<42%)	Low (η<42%)
Economics	Higher than LWR	GT-MHR GTHTR300C CANDU-SCWR PB-AHTR	Super LWR Super Fast LWR HPLWR	
	Similar to LWR		HTR-PM ELSY SSTAR	NGNP
	Lower than LWR		JSFR	KALIMER SMFR

There is a strong correlation between these factors because, given a certain power output, the sizing of several components (especially vessel and PCU) is directly influenced by electrical efficiency of the power plant. The size of these modules affects construction cost and so the economics of the project. Projects with high energy efficiency and elevated economic competitiveness are GT-MHR, GTHTR300C, CANDU-SCWR and PB-AHTR: thermal high temperature reactor. GT-MHR and GTHTR300C adopts a high efficiency helium Brayton cycle. CANDU-SCWR is the thermal SCWR, which has low construction and O&M costs, with highest OCT, which directly influences efficiency of thermodynamic cycle. Both of them adopt a direct (without heat exchanger) PCU. PB-AHTR adopts an intermediate circuit but these additional costs are compensated for the high thermo-fluid-dynamic properties of molten salts, having high specific heat and don’t require vessel pressurization. The projects with a lower attractiveness are two SFR: KALIMER and SMFR. Addition of an intermediate circuit and other supplementary system allowing the reduction of risks linked to sodium leaks. It increases construction cost. The low boiling point of coolant limits the maximum thermodynamic temperature. These solutions disadvantage this technology. However these reactors have other interesting features (see Table 9 and Table 10). NGNP has a low energy efficiency because is designed mainly for industrial cogeneration.

TABLE 8 Drivers: cogeneration, energy efficiency

		Energy efficiency		
		High (η>45%)	Medium (45%<η<42%)	Low (η<42%)
Cogeneration	High temperature (>800°)	GT-MHR GTHTR300C MSFR	GFR	
	Medium temperature (600°-750°)	CANDU-SCWR PB-AHTR	HTR-PM	NGNP
	Low temperature		HPLWR Super LWR Super Fast LWR ELSY SSTAR JSFR	KALIMER SMFR

TABLE 9 Drivers: actinides management, energy efficiency

		Energy efficiency		
		High (η>45%)	Medium (45%<η<42%)	Low (η<42%)
Actinides management	Breeder ratio > 1	MSFR	JSFR	
	Breeder ratio ≈ 1		Super Fast LWR ELSY SSTAR GFR	KALIMER SMFR
	Breeder ratio < 1	GT-MHR GTHTR300C CANDU-SCWR PB-AHTR	HTR-PM HPLWR Super LWR	NGNP

The taxonomies in Table 8 and Table 9 classify the projects according to their main application (cogeneration, actinide management, electricity production). Depending the cogeneration, the most important driver is OCT, influencing the maximum temperature of thermodynamic cycle, therefore the energy efficiency, and the temperature of delivered heat, influencing the possible cogenerative use (each industrial sector has a specific range of interesting temperatures). The high temperature coolant are helium and molten salts then technologies adopting them (VHTR, GFR, MSR) are preferable in this scenario. Elevated working temperature increases the critical issues linked to corrosion and requires high performance materials, which are expensive and innovative. Some projects (NGNP, HTR-PM) reduce OCT for the first reason, while others (SCWR and LFR) for the second one. MSFR and GFR are large power plants, disadvantaging cogenerative applications. Projects adopting Brayton cycle are suitable for low temperature cogeneration without reducing energy efficiency.

The second taxonomy crosses production of electricity and actinides management. Breeder reactors (JSFR, MSFR) have on average a lower attractiveness due to more demanding design and technical challenge. The target of GEN IV fast reactors is, therefore, more the efficient use of nuclear fuel rather than the net production of fissile material. All the technologies but VHTR have a fast spectrum therefore wide adoption of this reactors is one of the main aims of GEN IV despite demanding design. GFR, a fast high efficiency reactor, has a medium energy efficiency evaluation penalized by the adoption of an indirect cycle. MSFR has high evaluation in both the taxonomies therefore is attractive for deployment.

TABLE 10 Drivers: economics, size of the plant

		Economics		
		Superior than LWR	Similar to LWR	Lower than LWR
Size of the plant	Super-small (<100 MWe)		SSTAR	SMFR
	Small (100-300 MWe)	GT-MHR GTHTR300C	HTR-PM NGNP	
	Medium (300-700 MWe)	PB-AHTR	ELSY	KALIMER
	Large (>700 MWe)	CANDU-SCWR Super LWR Super Fast LWR HPLWR		JSFR

Table 10 deals with economics and plant size. The size subdivision is adopted from IAEA (70) adding super small reactors with power minor than 100 MWe. SSTAR and SMFR are in this class. SSTAR has a better economic evaluation because of the simpler plant layout. PB-AHTR and VHTR have similar properties but the adoption of molten salts in the first one increases its thermal power to a medium size. In this class there are, also, ELSY and KALIMER. The main large size technology is SCWR adopting the same dimension of current LWR. Super Fast LWR is the only fast reactor with higher economic competitiveness than LWR since it shares technical solutions with current nuclear and supercritical fossil fired power plants. Other thermal SCWR have also a high evaluation in economics.

The last taxonomy crosses technical feasibility and coolant. The first is a synthetic parameter, evaluating the deployment time of each project and the risk related to the required technological innovation. The technical feasibility classification relays on the historical availability of commercial power plants (like the SuperPhenix for the SFR or the HTGR for VHTR) or experimental reactors (for example HTTR and HTR-10 for VHTR or Monju and Joyo for SFR). For the reactors with the highest technical challenges (MSFR and GFR), the economics evaluation isn't illustrated because the critical issues, concerning, respectively, the salt treatment plant and the design of fuel element, are very relevant.

Many critical issues are linked to coolant: corrosiveness of lead and supercritical water, the reactivity of sodium with air and water, the lack of knowledge about the behavior of molten salts and other. Another relevant factor is neutronic spectrum since thermal reactors (with same coolant) have higher feasibility evaluation than fast ones. Size and economics have less influence.

TABLE 11 Drivers: coolant, technical feasibility

		Technical feasibility				
		High	Medium	Low	Uncertain	Critical
Coolant	Sodium		JSFR KALIMER	SMFR		
	Lead			ELSY SSTAR		
	Supercritical water			CANDU-SCWR Super LWR HPLWR	Super Fast LWR	
	Helium	HTR-PM NGNP	GTHTR300C GT-MHR			GFR
	Molten salts				PB-AHTR	MSFR

IV. MAIN AREAS OF R&D

Each GEN IV technology has critical issues requiring R&D. This chapter presents the main areas of R&D and reports the progress of the most relevant programs of each sector. The terminology adopted is (in order of advancement): concept, experimental phase, industrialization phase, optimization of productive process, licensing for nuclear sector, ready for deployment. The main areas of R&D are: materials, heat exchangers, PCU and fuel reprocessing.

IV.A. Materials

The development of specific materials is necessary for each GEN IV technologies. Most concepts had been formed in early '60 and '70 but the unavailability of suitable materials prevented their construction and their commercial competitiveness. GEN IV reactors have on average an higher working temperature, irradiation, corrosiveness of coolant and often higher pressure than current reactor. They require materials that had never been adopted in this sector. Some materials are employed in other industrial areas, especially in advanced fossil fired power plants, with a possible technology transfer. The materials for GEN IV are subdivisible in two categories: (1) for high temperature employs, (2) for medium temperature and high corrosiveness employs; therefore, even if each reactor has specific working condition, the classes of materials under exam are limited. The areas of R&D are in common with other research sectors, like fusion power plant and aerospace industry. The Gen IV Materials Handbook (71), which is a digitalized database to share information about the R&D under examination, facilitates a crosscutting approach.

TABLE 12 Summary of R&D areas about materials

Class	Material	Concerned projects	Main reference	R&D Stage
Advanced nickel alloys	Hastelloy XR	GTHTR300C	(72)	Ready
	800H	PB-AHTR VHTR	(73)	Ready
	Inconel 617 Haynes 230	VHTR GFR	(73)	Licensing for nuclear sector
	Super alloy (IN740)	VHTR GFR	(74)	Experimental phase
	Ni-Cr-W	MSR VHTR	(75)	Experimental phase
Ceramic materials	Ceramic cladding	VHTR GFR	(76)	Licensing for nuclear sector
	SiC/SiC composite C/C composite	VHTR GFR	(77)	Industrialization phase
	ZrC	VHTR GFR	(78)	Experimental phase

Refractory metals alloys	Molybdenum and tungsten alloys	LFR for hydrogen production	(74)	Experimental phase
F/M steels	T91	LFR SFR	(79)	Licensing for nuclear sector
	9Cr1MoV	VHTR GFR	(73)	Optimization of productive process
	A533B* A508*	NGNP SCWR	(76)	Optimization of productive process
ODS steels	9Cr 12Cr	SFR	(80)	Optimization of productive process
	Aluminum addition	SCWR LFR	(81)	Experimental phase
	Austenitic		(82)	
	High-chromium ferritic	SFR SCWR LFR	(83)	Experimental phase
Coated steels	Coated steels	LFR MSR SCWR	(84)	Experimental phase

* These steels had been licensed for nuclear applications

IV.B. Heat Exchangers

A crosscutting R&D in this field is difficult because each project has a specific heat exchanger optimized according to its size, thermodynamic cycle and reactor layout. The major criticality is the lack of knowledge about thermo-fluid-dynamic properties of molten salts and lead causing uncertainty about the behavior of this component.

TABLE 13 Summary of R&D areas about heat exchangers

Heat exchangers	Concerned projects	Main reference	R&D stage
Shell and tube	NGNP HTR-PM	(85)	Ready
Printed circuits	VHTR GFR	(86)	Optimization of productive process
High availability	SFR	(19)	Concept
Compact	LFR	(52)	Concept
High resistance to corrosion	MSR	(87)	Concept

IV.C. Power Conversion Unit

The main R&D area for PCU is related to close gas Brayton cycles. In particular is necessary to re-engineer the components and manage unconventional fluids.

TABLE 14 Summary of R&D areas about power conversion unit

PCU	Concerned projects	Main reference	R&D stage
S-CO_2 Brayton cycle	SFR LFR	(88)	Experimental phase
Helium Brayton cycle with IC and recuperator	VHTR GFR	(89)	Experimental phase
Multiple reheat helium Brayton cycle	MSR	(90)	Concept
Subcritical steam Rankine cycle	NGNP HTR-PM JSFR KALIMER ELSY	-	Ready
Supercritical water Rankine cycle	SCWR	(91)	Licensing for nuclear sector

IV.D. Nuclear fuel reprocessing

Fast reactors allow closed fuel cycle able to recycle of spent fuel and an efficient use of fissile materials. The main R&D area concerns fuel reprocessing. Two processes are considered: advanced aqueous process and pyroprocess. The first is optimized for oxide fuel and the second for metal alloy fuel. Each fast reactor necessitates of a specific reprocessing process for its fuel element. Both processes should be available in a mid-term time frame. Unfortunately nowadays closed fuel cycle has a worse economics than the open one.

TABLE 15 Summary of R&D areas about fuel reprocessing

Fuel reprocessing	Concerned projects	Main reference	R&D stage
Advanced aqueous process	Fast reactors	(92)	Optimization of process
Pyroprocess	Fast reactors	(93)	Experimental phase

V. CONCLUSIONS

Overall possible applications of GEN IV technologies (mid/high temperature co-generation, production of electricity, actinides management, energy supply for isolated grids and others) are wider than actual GEN III plants. Economics of GEN IV reactors is averagely similar to actual LWR but the mandatory adoption of CCS for fossil-fired power plants would increase substantially their attractiveness. GEN IV reactors require substantial R&D efforts preventing a short-term or mid-term commercial adoption.

NOMENCLATURE

BOP Balance Of Plant
CANDU CANada Deuterium-Uranium
ELSY European Lead-cooled System
GFR Gas-cooled Fast Reactor
GIF Generation IV International Forum
GTHTR300 Gas Turbine High Temperature Reactor 300
GT-MHR Gas Turbine – Modular Helium Reactor
HLW High Level Waste
HPLWR High Performance Light Water Reactor
HTGR High Temperature Gas Reactor
HTR-PM High Temperature Gas-Cooled Reactor-Pebble bed Module
IAEA International Atomic Energy Agency
IHX Intermediate Heat Exchanger
JSFR Japan Atomic Energy Agency (JAEA) Sodium Fast Reactor
KALIMER Korea Advanced LiquId MEtal Reactor
LBE Lead Bismuth Eutectic
LFR Lead-cooled Fast Reactor
LWR Light Water Reactor
MOX Mixed Oxide Fuel
MSFR Molten Salt Fast Reactor
MSR Molten Salt Reactor
NGNP Next Generation Nuclear Plant
OCT Outlet Core Temperature
ODS Oxide Dispersion Strengthened
PB-AHTR Pebble Bed – Advanced High Temperature Reactor
PBMR Pebble Bed Modular Reactor
PCU Power Conversion Unit
RPV Reactor Pressure Vessel
SCWR Supercritical-Water-Cooled Reactor
SFR Sodium Fast Reactor
SMFR Small Modular Fast Reactor
SMR Small Medium Reactor
SSTAR Small Secure Transportable Autonomous Reactor
TRU Transuranic
VHTR Very High Temperature Reactor

REFERENCES

1 GIF. GIF R&D Outlook for Generation IV Nuclear Energy Systems. Generation IV International Forum. [S.l.], p. 3-24 29-31. 2009.
2 MILLS, P.; SOTO, R.; GIBBS, G. Next Generation Nuclear Plant Pre-Conceptual Design Report. INL. [S.l.], p. iii-XXV 12-68. 2007.
3 INGERSOLL, D. T. Deliberately small reactors and the second nuclear era. Progress in Nuclear Energy, v. 51, n. 4-5, p. 589-603, Maggio-Giugno 2009.
4 ZHANG, Z. et al. Current status and technical description of Chinese 2×250MWth HTR-PM demonstration plant. Nuclear Engineering and Design, v. 239, n. 7, p. 1212-1219, 2009.
5 ZHANG, Z.; YULIANG, S. Economic potential of modular reactor nuclear power plants based on the Chinese HTR-PM project. Nuclear Engineering and Design, v. 237, n. 23, p. 2265-2274, Dicembre 2007.
6 MCKELLAR, M. G.; PATTERSON, M. W.; MILLS, P. M. An Analysis of the Effect of Reactor Outlet Temperature of a High Temperature Reactor on Electric Power Generation, Hydrogen Production, and Process Heat. INL. [S.l.], p. 5-18 25-31. 2010.
7 GANDRIK, A. M. et al. HTGR-Integrated Hydrogen Production via Steam Methane Reforming (SMR) Economic Analysis. INL. [S.l.], p. 3-47. 2010.
8 WOOD, R. A.; PATTERSON, M. W.; BOARDMAN, R. D. Nuclear-Integrated Methanol-to Gasoline Production Analysis. INL. [S.l.], p. 3-26 47-55 63-65. 2010.
9 BOARDMAN, R. D.; WOOD, R. A.; PATTERSON, M. W. Nuclear-Integrated Ammonia Production Analysis. INL. [S.l.], p. 3-28 55-67. 2010. fig. pag. 26.
10 LABAR, M. P. et al. Status of the GT-MHR for Electricity Production. World Nuclear Association Annual Symposium. Londra: [s.n.]. 2003. fig. pag. 12.
11 BAXI, C. B. et al. Evaluation of alternate power conversion unit designs for the GT-MHR. Nuclear Engineering and Design, v. 238, n. 11, p. 2995-3001, Novembre 2008.
12 NISAN, S.; DARDOUR, S. Economic evaluation of nuclear desalination systems. Desalination, v. 205, n. 1-3, p. 231-242, Febbraio 2007.
13 KUNITOMI, K. et al. Japan's future HTR—the GTHTR300. Nuclear Engineering and Design, v. 233, n. 1-3, p. 309-327, Ottobre 2004.

14 TAZIZUKA, T. Reactor technology development under the HTTR Project. Progress in Nuclear Energy, v. 47, n. 1-4, p. 283-291, 2005.

15 KUNITOMI, K. et al. JAEA’S VHTR for hydrogen and electricity cogeneration: GTHTR300C. NUCLEAR ENGINEERING AND TECHNOLOGY, v. 37, n. 1, p. 9-20, 2007. fig. pag. 10-13.

16 RICHARDS, M.; ARKAL, S. H2-MHR pre-conceptual design summary for hydrogen production. Nuclear Engineering and Technology, v. 39, n. 1, p. 1-8, Dicembre 2007.

17 VENTER, P.; MITCHELL, ; FORTIER, F. PBMR reactor design and development. SMiRT 18. Pechino: [s.n.]. 2005.

18 THOMAS, S. The Pebble Bed Modular Reactor: An obituary. Energy Policy, v. 39, n. 5, p. 2431-2440, Maggio 2011.

19 CACUCI, D. G. Sodium Fast Reactor Design: Fuels, Neutronics, Thermal-Hydraulics, Structural Mechanics and Safety. In: ROUAULT, J., et al. Handbook of Nuclear Engineering. [S.l.]: Springer, 2010. p. 2352-2417.

20 ICHIMIYA, M.; MIZUNO, T.; KOTAKE, S. A Next Generation Sodium-Cooled Fast Reactor Concept and its R&D Program. Nuclear Engineering and Technology, v. 39, n. 3, p. 171-186, Giugno 2007. fig. pag. 174.

21 HANDA, T. et al. Status of Integrated IHX/Pump Development for JSFR. Journal of NUCLEAR SCIENCE and TECHNOLOGY, v. 48, n. 4, p. 669-676, 2011.

22 AOTO , K. et al. Design Study and R&D Progress on Japan Sodium-Cooled Fast Reactor. Journal of NUCLEAR SCIENCE and TECHNOLOGY, v. 48, n. 4, p. 463-471, 2011.

23 KONOMURA, M.; ICHIMIYA, M. Design challenges for sodium cooled fast reactors. Journal of Nuclear Materials, v. 371, n. 1-3, p. 250-269, Settembre 2007.

24 KAERI. KALIMER-600 Conceptual Design Report. KAERI. [S.l.], p. 2.1-2.12 5.1-5.82. 2006.

25 HAHN, D. et al. Conceptual Design of the Sodium-Cooled Fast Reactor KALIMER-600. Nuclear engineering and technology, v. 39, n. 3, p. 193-206, Giugno 2007. fig. pag. 194.

26 HAHN, D. et al. Advanced SFR design concepts and R&D. Nuclear Engineering and Technology, v. 41, n. 4, p. 427-446, Maggio 2009.

27 CHANG, Y. I.; LO PINTO, P.; KONOMURA, M. A Case for Small Modular Fast Reactor. Journal of NUCLEAR SCIENCE and TECHNOLOGY, v. 44, n. 3, p. 264-269, 2007. fig. pag. 266.

28 GRANDY, C. et al. Small modular fast reactor design description. ANL CEA JNC. [S.l.], p. 1-8 28-30 48-78. 2005.

29 SUN, C. et al. Progress in corrosion resistant materials for supercritical water reactors. Corrosion Science, v. 51, n. 11, p. 2508-2523, Novembre 2009.

30 WAS, G. S. et al. Corrosion and stress corrosion cracking in supercritical water. Journal of Nuclear Materials, v. 371, n. 1-3, p. 176-201, Settembre 2007.

31 FISCHER, K.; SCHULENBERG, T.; LAURIEN, E. Design of a supercritical water-cooled reactor with a three-pass core arrangement. Nuclear Engineering and Design, v. 239, n. 4, p. 800-812, Aprile 2009. fig. pag. 801.

32 OKA, Y. et al. Super Light Water Reactors and Super Fast Reactors. [S.l.]: Springer, 2010. 6-60 79-100 200-218 221-238 441-522 550-567 571-588 599-619 p. fig. pag. 57-231.

33 STARFLINGER, J. Public Final Report Assessment of the HPLWR Concept. Karlsruhe Institute of Technology. [S.l.], p. 17-39 41-95 227-249 267-274. 2010. fig. pag. 22.

34 SCHULENBERG, T. et al. European supercritical water cooled reactor. Nuclear Engineering and Design, v. 241, n. 9, p. 3505-3513, Settembre 2011.

35 CAO, L. et al. Fuel, Core Design and Subchannel Analysis of a Superfast Reactor. Journal of NUCLEAR SCIENCE and TECHNOLOGY, v. 45, n. 2, p. 138–148, 2008.

36 OKA, Y. et al. Super fast reactor R&D projects in Japan. ICONE19. Chiba: [s.n.]. 2011.

37 IKEJIRIA, S.; ISHIWATARI, Y.; OKA, Y. Safety analysis of a supercritical-pressure water-cooled fast reactor under supercritical pressure. Nuclear Engineering and Design, v. 240, n. 5, p. 1218-1228, Maggio 2010.

38 CHOW, C. K.; KHARTABIL, H. F. Conceptual Fuel Channel Designs For CANDU-SCWR. Nuclear Engineering and Technology, v. 40, n. 2, p. 139-147, 2008.

39 NAIDIN, M. et al. Thermal-Design Options for Pressure-Channel SCWRS With Cogeneration of Hydrogen. Journal of Engineering for Gas Turbines and Power, v. 131, n. 1 012901, Gennaio 2009.

40 TORGERSON, D. F.; SHALABY, B. A.; PANG, S. CANDU technology for Generation III+ and IV reactors. Nuclear Engineering and Design, v. 236, n. 14-16, p. 1565-1572, Agosto 2006.

41 MEYER, M. K.; FIELDINGA, R.; GAN, J. Fuel development for gas-cooled fast reactors. Journal of Nuclear Materials, v. 371, n. 1-3, p. 281-287, Settembre 2007.

42 PETKEVICH, P. et al. Development and benchmarking of a mechanical model for GFR plate-type fuel. Annals of Nuclear Energy, v. 36, n. 8, p. 1262-1269, Agosto 2009.

43 DUMAZ, P. et al. Gas-cooled fast reactors—Status of CEA preliminary design studies. Nuclear Engineering and Design, v. 237, n. 15-17, p. 1618-1627, Settembre 2007.

44 CARRE, F. et al. Update of the French R&D strategy on gas-cooled reactors. Nuclear Engineering and Design, v. 240, n. 10, p. 2401-2408, Ottobre 2010. fig. pag. 2405.

45 BERTHOUX, M.; CADIOU, T. The thermal hydraulics in a rod bundle representative of the start-up core of the ALLEGRO Gas cooled Fast Reactor—Experimental and numerical approaches. Nuclear Engineering and Design, v. 240, n. 10, p. 3372-3386, Ottobre 2010. fig. pag. 3373.

46 STAINSBY, R. et al. The Generation IV Gas-cooled Fast Reactor. ICAPP 2011. Nizza: [s.n.]. 2011. p. Paper 11321.

47 OBARA, T. et al. Polonium evaporation and adhesion experiments for the development of polonium filter in lead–bismuth cooled reactors. Progress in Nuclear Energy, v. 50, n. 2-6, p. 556-559, Marzo-Agosto 2008.

48 MIKITYUK, K. Analytical model of the oxide layer build-up in complex lead-cooled systems. Nuclear Engineering and Design, v. 240, n. 10, p. 3632-3637, Ottobre 2010.

49 KONDO, M. et al. Corrosion of Steels in Lead–Bismuth Flow. Journal of NUCLEAR SCIENCE and TECHNOLOGY, v. 43, n. 2, p. 107–116, 2006.

50 RIVAI, A.; TAKAHASHI, M. Compatibility of surface-coated steels, refractory metals and ceramics to high temperature lead–bismuth eutectic. Progress in Nuclear Energy, v. 50, n. 2-6, p. 560-566, Marzo-Agosto 2008.

51 ALEMBERTI, A. et al. European lead fast reactor— ELSY. Nuclear Engineering and Design, v. 241, n. 9, p. 3470-3480, Settembre 2011. fig. pag. 4-5.

52 CINOTTI, L. et al. Lead-cooled system design and challenges in the frame of Generation IV International Forum. Journal of Nuclear Materials, v. 415, n. 3, p. 245-253, Agosto 2011. fig. pag. 250.

53 BANDINI, G.; MELONI, P.; POLIDORI, M. Thermal-hydraulics analyses of ELSY lead fast reactor with open square core option. Nuclear Engineering and Design, v. 241, n. 4, p. 1165-1171, Aprile 2011.

54 SMITH, C. F. et al. SSTAR: The US lead-cooled fast reactor (LFR). Journal of Nuclear Materials, v. 376, n. 3, p. 255-259, Agosto 2008.

55 SIENICKI, J. J. et al. Status Report on the Small Secure Transportable Autonomous Reactor (SSTAR)/Lead-Cooled Fast Reactor (LFR) and Supporting Research and Developmen. ANL. [S.l.], p. 10-73. 2006. fig. pag. 28.

56 IAEA. Status of Small Reactor Designs Without On-Site Refuelling. IAEA. [S.l.], p. 9-48 511-550 591-713. 2007.

57 BORTOT, S. et al. Core design investigation for a SUPERSTAR small modular lead-cooled fast reactor demonstrator. Nuclear Engineering and Design, v. 241, n. 8, p. 3021-3031, Agosto 2011.

58 SIENICKI, J. J. et al. SUPERSTAR: An Improved Natural Circulation, Lead-Cooled, Small Modular Fast Reactor for International Deployment. ICAPP 2011. Nizza: [s.n.]. 2011. p. Paper 11380.

59 BENES, O. et al. ALISIA - Review report on liquid salts for various applications. Euratom Research and Training Programme. [S.l.], p. 3-9 19-56. 2009.

60 BENES, O.; KONINGS, R. J. M. Thermodynamic properties and phase diagrams of fluoride salts for nuclear applications. Journal of Fluorine Chemistry, v. 130, n. 1, p. 22-29, Gennaio 2009.

61 DELPECH, S. et al. Molten fluorides for nuclear applications. Materialtoday, v. 13, n. 12, p. 34-42, Dicembre 2010.

62 MERLE-LUCOTTE, E. et al. Influence of the Processing and Salt Composition on the Thorium Molten Salt Reactor. Nuclear Technology , v. 163, n. 3, p. 358-365, 2008.

63 MERLE-LUCOTTE, E. et al. Minimizing the fissile inventory of the molten sal fast reactor. Advances in Nuclear Fuel Management IV. Hilton Head Island: [s.n.]. 2009. fig. pag. 3.

64 MERLE-LUCOTTE, E. Le cycle Thorium en réacteurs à sels fondus peut-il être une solution au problème énergétique du XXIème siècle ? Le concept de TMSR-NM. Institut National Polytechnique de Grenoble. [S.l.], p. 24-28 73-100. 2008.

65 YAN, X. L.; HINO, R. Fluoride Salt Advanced High-Temperature Reactor. In: PETERSON, P. F.; BLANDFORD, E. D. Nuclear Hydrogen Production Handbook. CRC. ed. [S.l.]: [s.n.], 2011. p. 329-345.

66 FORSBERG, C. W. Design Options for the Advanced High-Temperature Reactor. ICAPP ‘08. Anaheim: [s.n.]. 2008. fig. n. 6.

67 NE-170 SENIOR DESIGN PROJECT. A modular pebble-bed advanced high temperature reactor. University of California, Berkeley. [S.l.], p. 12-51. 2008. fig. pag. 14.

68 PETERSON, P. F.; ZHAO, H. An Optimized System for Advanced Multi-Effect Distillation (AMED) Using Waste Heat from Closed Gas Brayton Cycles. U.C. Berkeley. [S.l.]. 2006.

69 HOLCOMB, D. E. et al. An Analysis of Testing Requirements for Fluoride Salt-Cooled High Temperature Reactor Components. OAK RIDGE NATIONAL LABORATORY. [S.l.], p. xi-10. 2009.

70 IAEA. Status of Innovative Small and Medium Sized Reactor Designs 2005 - Reactors with conventional refuelling schemes. IAEA. [S.l.]. 2005.

71 REN, W. Gen IV Materials Handbook Functionalities and Operation (1B). Oak Ridge National Laboratory. [S.l.], p. III. 2010.

72 ASAYAMA, T.; TACHIBANA, Y. Collect Available Creep-Fatigue Data and Study Existing Creep-Fatigue Evaluation Procedures for Grade 91 and Hastelloy XR. JAEA. [S.l.], p. 3-45 61. 2007.

73 CORWIN, W. R. Status of Ongoing Research. Within the GIF VHTR Materials Project. GIF Symposium. Parigi: [s.n.]. 2009.

74 MURTY, K. L.; CHARIT, I. Structural materials for Gen-IV nuclear reactors: Challenges and opportunities. Journal of Nuclear Materials, v. 383, n. 1-2, p. 189-195, Dicembre 2008.

75 BAI, G. et al. Effect of temperature on tensile behavior of Ni–Cr–W based superalloy. Materials Science and Engineering: A, v. 528, n. 4-5, p. 1974-1978, Febbraio 2011.

76 CORWIN, W. R. et al. Generation IV Reactors Integrated Materials Technology Program Plan: Focus on Very High Temperature Reactor Materials. Oak Ridge National Laboratory. [S.l.], p. 1-52. 2008.

77 KOHYAMA, A.; PARK, J.; JUNG, H. Advanced SiC fibers and SiC/SiC composites toward industrialization. Journal of Nuclear Materials, v. 417, n. 1-3, p. 340-343, Ottobre 2011.

78 AZEVEDO, C. R. F. Selection of fuel cladding material for nuclear fission reactors. Engineering Failure Analysis, v. 18, n. 8, p. 1943-1962, Dicembre 2011.

79 FAZIO, C. et al. Innovative materials for Gen IV systems and transmutation facilities: The cross-cutting research project GETMAT. Nuclear Engineering and Design, v. 241, n. 9, p. 3514-3520, Settembre 2011.

80 UKAI, S. et al. Tube manufacturing trials by different routes in 9CrW-ODS martensitic steels. Journal of Nuclear Materials, v. 329-333, n. A, p. 356-361, 2004.

81 KIMURA, A. et al. Development of Al added high-Cr ODS steels for fuel cladding of next generation nuclear systems. Journal of Nuclear Materials, v. 417, n. 1-3, p. 176-179, Ottobre 2011.

82 XU, Y. et al. Fabrication and characterization of ODS austenitic steels. Journal of Nuclear Materials, v. 417, n. 1-3, p. 283-285, Ottobre 2011.

83 DE CARLAN, Y. et al. CEA developments of new ferritic ODS alloys for nuclear applications. Journal of Nuclear Materials, v. 386-388, n. 30, p. 430-432, Aprile 2009.

84 KURATA, Y.; YOKOTA, H.; SUZUKI, T. Development of aluminum alloy coating for advanced nuclear. Proceedings of the ASME 2011 Small Modular Reactors Symposium. Washington: [s.n.]. 2011.

85 WRIGHT, J. K.; WRIGHT, R. N.; PETTI, D. A. Next Generation Nuclear Plant Steam Generator and Intermediate Heat Exchanger Materials Research and Development Plan. INL. [S.l.], p. 1-17 93-101 244-248. 2010. fig. pag. 5.

86 MYLAVARAPU, S. K. et al. Fabrication and design aspects of high-temperature compact diffusion bonded heat exchangers. Nuclear Engineering and Design, v. in stampa, 2011.

87 MOIR, R. W. Recommendations for a restart of molten salt reactor development. Energy Conversion and Management, v. 49, n. 7, p. 1849–1858, 2008.

88 WRIGHT, S. A. et al. Operation and Analysis of a Supercritical CO2 Brayton Cycle. Sandia National Laboratories. [S.l.], p. 10-22 43-91. 2010.

89 ZHU, R. et al. Off-design performance research of an axial helium compressor for HTGR-10 power conversion unit. Nuclear Engineering and Design, v. 240, n. 10, p. 2914-2919, 2010.

90 ZHAO, H.; PETERSON, P. F. Optimization of Advanced High-Temperature Brayton Cycles with Multiple-Reheat Stages. Nuclear Technology , v. 158, n. 2, p. 145-157, 2007.

91 ABRAM, T.; ION, S. Generation-IV nuclear power: A review of the state of the science. Energy Policy, v. 39, n. 12, p. 4323-4330, Dicembre 2008.

92 BORGES-SILVERIO, L.; DE QUEIROZ LAMAS, W. An analysis of development and research on spent nuclear fuel reprocessing. Energy Policy, v. 39, n. 1, p. 281-289, Gennaio 2011.

93 SONG, K. et al. Status of Pyroprocessing Technology Development in Korea. Nuclear Engineering and Technology, v. 42, n. 2, p. 131-145, 2010.