Fast neutron Generation IV reactors

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1) Introduction

Learning outcomes:

After working through this chapter, you will be able to:

1) Evaluate if a reactor concept should be classified as a Generation IV system
2) Critically review the conflicting goals of Generation IV systems
3) Assess the pros and cons of coolant and fuel technologies that form the basis of Generation IV concepts.

Background

In 1999, the world had 433 large nuclear reactors producing electricity to the grid, but only 37 units were under construction, down from over 200 units in 1980 [IAEA]. Among the problems perceived to hamper the development were difficulties in finding public acceptance for waste management approaches, the competitiveness of the industry under new safety requirements and the potential for proliferation of weapons usable materials. In response to these issues, the US Department of Energy (DoE) invited government representatives from eight other countries to discuss international collaboration in the development of ”Generation IV nuclear energy systems” [GIF website]. The founding charter of the Generation IV International Forum (GIF) states that nuclear power plant technology had evolved as ”three distinct design generations”, namely

I) Prototypes
II) Current operating plants
III) Advanced reactors

and that the next generation, to be labelled ”Generation IV”, must address the aforementioned issues. In order support the development of Generation IV systems, the Members of GIF agreed to identify potential areas for international collaboration on R&D for such systems and foster corresponding collaborative projects [GIF charter].

In the first technology roadmap published by GIF, eight ”goals” that Generation IV systems should meet were defined [GIF 2002]:

1) Resource utilisation:

Generation IV nuclear energy systems will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilisation for worldwide energy production.

2) Waste minimisation and management:

Generation IV nuclear energy systems will minimise and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.

3) Life cycle cost:

Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

4) Financial risk:

Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

5) Operational safety and reliability:

Generation IV nuclear energy systems operations will excel in safety and reliability.

6) Core damage:

Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

7) Off-site emergency response:

Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

8) Proliferation Resistance and Physical Protection

Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.

Several of the aforementioned goals are obviously not defined in terms of measurable quantities, and are therefore open to interpretation. Example given, the meaning of ”effective fuel utilisation” in Goal (1) may be discussed at length. Others appear impossible to satisfy already from the very start, such as Goal (4) on financial risk. Without doubt, small gas plants constitute a financial risk smaller than any perceivable nuclear reactor project. Nevertheless, over 100 experts from ten countries were involved in identifying concepts that could be further developed to meet the eight goals, and selected the ”six most promising systems”, as being (in alphabetic order):

1) Gas-cooled fast reactor systems (GFR)
2) Lead-cooled fast reactors systems (LFR)
3) Molten salt reactor systems (MSR)
4) Sodium-cooled fast reactor systems (SFR)
5) Supercritical-water-cooled reactor systems (SCWR)
6) Very-high-temperature reactor systems (VHTR)

Whereas several of these systems are based on coolant technologies that have been used in research reactors or military reactors, only sodium-cooled reactors have been deployed for grid-connected power production. It is to be noted that the definition of VHTRs employed by GIF includes operation at temperatures above 1000°C, which is far above the temperature regime under which conventional high temperature reactors with coated particle fuel have been operating. Hence, a substantial amount of R&D is necessary to bring all concepts beside the SFR to industrial maturity, and GIF provided a list of recommendations for such work. However, it should be noted that not even recently operational SFRs, such as BN-800, meet all the eight goals set by GIF. Consequently, BN-800 is classified as a reactor ”step towards” Generation-IV design [GIF FAQ].

Underlying technologies

It is somewhat remarkable that all of the systems selected by GIF are based on the use of coolants and/or fuels that have been deployed in prototype reactors already in the 1960’s. Albeit the coolants feature advantages that would permit to design reactors meeting the goals of GIF, the associated concepts historically were not able to compete with light water (or heavy water) reactors on commercial grounds. In order to address this problem, and identify solutions making the six concepts competitive with Generation III systems, we need to review existing operational experience (OPEX) pertaining to the respective coolant.

Experience from operation of sodium-cooled reactors

More than 200 years of operational experience has been accumulated in a total of 25 sodium and sodium-potassium cooled reactors (see Table 1, ref IAEA PRIS + OPEX). The first reactor taken into operation was EBR-I in Idaho Falls (USA), which also was the first reactor of any kind to produce electricity for in-house needs. The biggest reactor was Super-Phenix in Creys-Malville, producing 1200 MW of electricity to the French grid for 300 full power days. The most reliable reactor has been BN-600 in Beloyarsk, which has provided 27 full power years of operational experience since its grid connection in 1980. The reactor most recently connected to the grid is BN-800, also in Beloyarsk.

Reactor	Country	Coolant	Power	Fuel	Criticality	OPEX	Shutdown
EBR-I	USA	NaK	1.2 MW_{th}/0.2 MW_e	U	1951	0.5 y	1963
SIR/S1G	USA	Na	11 MW	UO_2	1955	900 h	1955
S2G	USA	Na	11 MW	UO_2	1955	1 y	1958
SRE	USA	Na	20 MW_{th}/6 MW_e	U	1957		1964
BR-5/BR-10	USSR	Na	5-8 MW_{th}	PuO_2/UC/UN	1958		2004
DFR	UK	NaK	72 MW_{th}/15 MW_e	U-Mo	1959	5 y	1977
LAMPRE	USA	Na	1 MW_{th}	Pu (liquid)	1961		1965
Hallam	USA	Na	200 MW_{th}/75 MW_e	U-Mo/UC	1963		1964
Fermi	USA	Na	200 MW_{th}/69 MW_e	U-Mo	1963	500 h	1972
EBR-II	USA	Na	62 MW_{th}/20 MW_e	U-Fissium	1963	15 y	1994
Rapsodie	France	Na	24-40 MW_{th}	$(U,Pu)O_2$	1967		1983
SEFOR	USA	Na	20 MW_{th}	$(U,Pu)O_2$	1969		1972
BOR-60	USSR	Na	60 MW_{th}/12 MW_e	UO_2	1969		operates
BN-350	USSR	Na	1000 MW_{th}/150 MW_e	UO_2	1973		1999
Phénix	France	Na	568 MW_{th}/250 MW_e	$(U,Pu)O_2$	1973	12 y	2009
PFR	UK	Na	600 MW_{th}/250 MW_e	$(U,Pu)O_2$	1974	4 y	1981
JOYO	Japan	Na	75-100 MW_{th}	$(U,Pu)O_2$	1977		operates
KNK-II	Germany	Na	58 MW_{th}/21 MW_e	$(U,Pu)O_2$	1977	2 y	1991
FFTF	USA	Na	400 MW_{th}	$(U,Pu)O_2$	1980		1992
BN-600	USSR	Na	1470 MW_{th}/250 MW_e	UO_2	1980	27 y	operates
SuperPhénix	France	Na	3000 MW_{th}/1200 MW_e	$(U,Pu)O_2$	1985	0.8 y	1997
FBTR	India	Na	40 MW_{th}/13 MW_e	(U,Pu)C	1985	6 y (?)	operates
Monju	Japan	Na	714 MW_{th}/280 MW_e	$(U,Pu)O_2$	1994	1 h	2017
CEFR	China	Na	65 MW_{th}/20 MW_e	UO_2	2010		operates
BN-800	Russia	Na	2100 MW_{th}/880 MW_e	UO_2/$(U,Pu)O_2$	2014	2 y	operates

Whereas the operation of BN-600 has proven that SFRs can produced electricity to the grid with a reliability that is on par with (or better than) LWRs, the specific capital investment still is higher than for Generation III reactors in the same country. This is partially due to technology specific issues that require more expensive solutions. Among the problems that SFRs have encountered are [IAEA 2013, Sauvage 2004]:

1) Steam generator corrosion (S2G, PFR, Phénix).
2) Sodium water interaction (Fermi, BN-350, Phénix, BN-600)
3) Sodium fires (S1G, BN-350, Phénix, Monju)
4) Fuel reloading machine failures (JOYO)

In 1982-1983, sodium-water interaction due to leaking steam generator tubes occurred four times in Phénix, resulting in six months of outage. During the first event, the combustion flame caused erosion-corrosion failure on an adjacent tube and damaged the steam generator shell [Sauvage 2004].

Sodium fires occur typically results from leaks in intermediate heat exchanger piping, and may be extinguished using sodium carbonate powders [IGCAR 2010]. The main consequences are the evolution of smoke during the event, the down-time required for repairing and cleaning the secondary system, and to obtain permission to restart the reactor. In Phénix, outages of a few months were sufficient to return to power, but in Monju it took 16 years to restart the reactor.

BN-600 has operated with an average availability of 74% since start of commercial operation in 1982. In total twelve sodium-water interaction events has occurred, caused by flaws in weld joints. Six of these happened during the first year of operation. Thanks to improved welding procedures and quality control, BN-600 has not experienced any sodium leaks since 1993 and no sodium-water interactions since 1991 [Potapov 2015]. In the same time, the service life of the primary sodium pump has been possible to extend from three to six years.

Albeit well managed in Russia, the problems related to sodium leaks, sodium-water interaction and sodium fires, has led the CEA to replace the secondary side steam cycle with a nitrogen based solution in the design of their (now discontinued) Generation-IV prototype ASTRID. A consequence thereof is a reduced efficiency in conversion of heat to electricity.

A particular issue related to the use of sodium is its relatively low boiling temperature, which in reactors with positive coolant void worth will accentuate power excursions. It is to be noted that most of the SFRs from which significant OPEX exists, were fuelled with enriched uranium, or with mixed oxide fuel where the uranium was enriched. Such reactors usually have a negative coolant temperature coefficient and void worth. Consequently, they behave better during transients.

Three core melts have occurred in sodium-cooled reactors. In 1954, the first core of EBR-I melted due to fuel bowing. In 1959, an organic compound used for component cooling in the SRE graphite moderated reactor [Starr 1958], decomposed and clogged several cooling channels, damaging 13 of the reactor's 43 fuel elements. In 1966, a coolant channel blockage in Fermi resulted in partial core melt. None of these accidents were specifically caused by the use of sodium coolant.

Currently, four sodium-cooled fast reactors are under construction, the Prototype Fast Breeder Reactor (PFBR) in India, with a design power of 500 MW_e, MBIR in Russia, a 150 MW_{th} research and materials testing reactor, and two CFT-600 units in China, each with a power of 600 MW_e.

Figure 1: Interactive photo exhibition from sodium fast reactors.

Experience from operation of lead-cooled reactors

80 reactor years (likely not at full power) has been accumulated from the operation of eleven lead-bismuth (LBE) cooled reactors in the Soviet Union and Russia (see Table 2). These reactors were either used for propulsion of military sub-marines, or as land-based prototypes. The first reactor taken into operation was VT-27 in Obninsk, which reached criticality in 1959. Two variants of propulsion reactors with a thermal power of 155 MW and a turbine power of 30 MW were produced, OK-550 by OKBM Afrikantov, and BM-40A by OKB Gidropress. The last operating sub-marine reactor was shut-down in 1996.

Reactor	Country	Coolant	Thermal power	Fuel	Criticality	OPEX	Shutdown
VT-27	USSR	LBE	70 MW_{th}	U-Be	1959	3500 h	1976
K-27 I-II	USSR	LBE	2 x 73 MW_{th}	U-Be	1963	?	1968
K-64	USSR	LBE	155 MW_{th}	U-Be	1969	?	1972
K-123	USSR	LBE	155 MW_{th}	U-Be	1977	?	1996
K-316	USSR	LBE	155 MW_{th}	U-Be	1978	?	1990
KM-1	USSR	LBE	155 MW_{th} (?)	U-Be	1978	5000 h	1986
K-432	USSR	LBE	155 MW_{th}	U-Be	1978	?	1990
K-373	USSR	LBE	155 MW_{th}	U-Be	1979	?	1990
K-493	USSR	LBE	155 MW_{th}	U-Be	1981	?	1990
K-463	USSR	LBE	155 MW_{th}	U-Be	1981	?	1990

The operation of the VT-27 prototype suffered from failure of steam generator tubes, and revealed significant oxidation of the same. Large irregularities in core temperatures (200 K) were observed while operating the second core [Suvorov 1998].

During operation of the nuclear sub-marine reactors, three accidents occurred, which made it impossible to further operate the vessels [Gromov 1997]:

- Clogging of the reactor inlet due to accumulation of large amounts of lead oxides leading to a core-melt. This was the first and also the only nuclear accident related to the use of lead-bismuth as reactor coolant.

- One accident due to not meeting the required water chemistry in the heat exchanger loop, leading to penetration of steam into the primary loop and a subsequent spill of lead-bismuth into the reactor compartment through a broken manometer.

- One reactor failure due to cold surface sweating water condensation on cold surfaces, causing corrosion cracking of heat exchanger pipes, leading to removal of the reactor from service.

It may be noted that no reactor has operated on pure lead-coolant. The nominal temperature regime of the sub-marine reactors was between 230 and 450 degrees C. Moreover, the core life of the KM-1 prototype was quoted to be 5000 full power hours [Pankratov]. If this number also applies to operation of sub-marines with the same reactor type, the total experience from operating lead-bismuth cooled reactors at their design power shrinks to about five years.

Construction of the BREST-300 reactor started in Russia in 2020. The 300 MWe lead-cooled reactor with (U,Pu)N fuel is expected to commence operating in 2026, in association with a fuel recycling facility.

Experience from operation of helium-cooled reactors

The Generation-IV Very High Temperature Reactor (VHTR) system is defined to be operating at a helium coolant outlet temperature exceeding 1000°C. No reactor has ever operated under such conditions. However, seven helium cooled reactors with coated particle fuel have operated at lower temperatures [IAEA 2010]. These are listed in Table 3.

Reactor	Country	Coolant	Thermal power	Fuel	Criticality	OPEX	Shutdown
DRAGON	UK	He	20 MW_{th}	UO_2-TRISO	1964		1976
AVR	Germany	He	46 MW_{th}/15 MW_e	UO_2-BISO	1966	14 y	1987
Peach Bottom	USA	He	115 MW_{th}/40 MW_e	UC-BISO	1967	4 y	1974
Fort St Vrain	USA	He	842 MW_{th}/330 MW_e	UC-TRISO	1976	2 y	1989
THTR	Germany	He	750 MW_{th}/300 MW_e	$(Th,U)O_2$-BISO	1983	420 d	1989
HTTR	Japan	He	30 MW_{th}	UO_2-TRISO	1998		operates
HTR-10	China	He	10 MW_{th}	UO_2-TRISO	2000		operates

The temperature regime under which these reactors have operated varies from maximum 750°C (DRAGON, Peach Bottom, THTR) to 950°C (AVR and HTTR).

A particular safety advantage pertaining to the concept is the use of coated particle fuel, embedded in graphite. The triplicate TRISO coating ensures a very high failure temperature (~ 1600°C) of the barrier for fission gas release. Conversely, the coating makes the fuel very difficult and costly to reprocess, which is an advantage in terms of non-proliferation, but a disadvantage in the context of resource utilisation.

Other disadvantages of the concept include [Dawson 2012]:

- Leakage of highly mobile helium
- High cost of helium coolant
- Mechanical properties of structural materials at high temperature
- Large size (and associated costs) of secondary system
- Activation of nitrogen impurities in graphite

Continuous leak rates of 0.2 kg He per day (0.15% of the total inventory) were observed throughout the DRAGON project [Simon].

In the case where the graphite (into which the coated particle fuel is embedded) has the shape of billiard-ball-sized pebbles, there is also an issue related to graphite dust production resulting from friction between the pebbles. On May 4th 1986, a a fuel pebble in the THTR got lodged in a fuel feed pipe. During attempts to dislodge the pebble, it was damaged, leading to release of a small amount (100 MBq) radioactivity into the atmosphere. THTR was built without a containment.

The high temperature operation of AVR resulted in release of Sr-90 from its BISO coated fuel, which was ignored, in spite of concerns for occupational safety [Moorman 2015]. In total 100 TBq of Sr-90 (2 kg) is estimated to have been released during operation of AVR. The high release rate observed in AVR may be possible to mitigate by application of TRISO coated particle fuels.

Recently, a 2x105 MWe helium-cooled twin unit (HTR-PM) was connected to the grid in China, using pebble bed TRISO fuel. Its operating temperature is limited to 750°C [Zhang 2016].

Experience from operation of molten salt reactors.

Four molten salt reactors have been operated. In the US, the MSRE (Molten Salt Reactor Experiment) of Oak Ridge National Laboratory was based on an 8 MWth reactor which achieved criticality in 1965 and the reactor was shutdown in 1969. During this period, it operated at power for a total of less than 18 months, on a U-Li-Be-Zr fluoride salt at a nominal temperature of 632 - 640°C. Once chemistry control was achieved, the corrosion rate of the Hastelloy-N vessel was observed to be about 2 microns per year.

During storage of the spent fuel from the MSRE, oxidation of UF_4 by fluorine (the latter forming due to radiolysis of the salt) had started to convert the fuel into highly volatile UF_6. In 1994 it was discovered that 4.4 kg of (87% fissile) uranium had evaporated in the storage tanks and was transported out from the main reactor building, out of which 2.6 kg had precipitated in charcoal filters for off-gases [Peretz 1996, US NRC 1997].

A zero-power molten salt reactor was built in China 1970-1971 [Zou 2019]. Very little public information is available on the project, which was abandoned due to materials related problems.

Currently, a 2 MW_{th} MSR is under construction in China.

Experience from operation of gas-cooled fast reactors

There is no experience from operating gas-cooled fast reactors. Among the particular challenges facing the realisation of this concept are:

- Difficulty in ensuring passive decay heat removal following a de-pressurisation accident
- The large size of primary and secondary systems, with an associated impact on economy

Experience from operation of super-critical water reactors

There is no experience from operating super-critical water reactors. Among the challenges specific for the concept are:

- Very high corrosion rates of zircalloy and stainless steels in supercritical water
- Instability of the system around the critical point
- The poor conversion ratio, making it difficult to achieve sustainability

Exercise

As we have seen, the technologies that are proposed for further development into Generation IV systems sometimes meet one or several of the GIF goals in a manner that probably excludes it from meeting other goals.

- Discuss, for each of the six technologies, which of the GIF goals that are best met by the concept, and why the reasons for being able to meet these goals may contradict the ability of the concept meeting other goals.

Questions

- Which are the criteria for classifying a reactor concept as a Generation IV system?
- Which Generation IV systems are based on a coolant technology for which there is operational experience?
- How can the economics goals be made compatible with other Gen-IV system goals?

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