ELECTRIC POWER RESEARCH INSTITUTE

Program on Technology Innovation: Material Property Assessment and Data Analysis for Sodium-Cooled Fast Reactors

2020 TECHNICAL REPORT

Program on Technology Innovation: Material Property Assessment and Data Gap Analysis for Sodium-Cooled Fast Reactors

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ABSTRACT

Sodium-cooled fast reactors (SFRs) are poised for worldwide growth as one of the most desirable and mature advanced reactor technologies. This report provides an overview of historical SFR construction and operation and summarizes gaps in SFR technology and materials. Additionally, it includes input from key U.S. stakeholders in SFR design. Based on this review, several areas for research and development are identified, both to support the near-term deployment of conventional SFR designs and to enable the next generation of advanced SFRs.

Keywords

Austenitic stainless steel
Ferritic/martensitic steel
Generation IV reactor
Liquid metal cooled reactor
Nuclear materials
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PRIMARY AUDIENCE: Scientists, engineers, and developers of advanced reactors who are interested in the development, deployment, and operation of sodium-cooled fast reactors (SFRs)

SECONDARY AUDIENCE: Utility staff who are interested in the next generation of reactors

KEY RESEARCH QUESTION

The number of operating SFRs has waned since the 1980s, but SFRs are poised for a resurgence in international construction. This report summarizes key materials issues that need to be addressed to enable both conventional and next-generation SFRs.

RESEARCH OVERVIEW

This report presents the results of a literature review and a U.S. industry survey on materials science-related knowledge gaps for both conventional and advanced SFRs. A brief overview of historic, current, and planned SFRs is given in both tabular and graphical forms. Recent overviews of SFR materials needs are presented, and input from U.S. SFR stakeholders is summarized. Lastly, a brief review of the current scientific literature focused on SFR materials gaps is presented. These findings are summarized and used to identify issues that need to be addressed for both near-term and advanced SFR designs.

KEY FINDINGS

- International SFR construction is projected to increase, with the number of operating reactors growing from 4 to 12 by 2030.
- U.S. stakeholders highlight that industrial supply of nuclear quality ferritic/martensitic steels is required to support development and construction.
- Advanced structural alloys and fuel cladding are required to extend SFR technology, increase its economic competitiveness, and enable a closed fuel cycle.
- Other technological areas in which development is needed include improving weld filler metals, understanding fuel/clad interactions, and developing in-sodium sensors and inspection equipment.

WHY THIS MATTERS

This report provides a current snapshot of planned growth and materials gaps in SFR technology. It enables nuclear designers, builders, suppliers, and researchers to quickly identify issues, gaps, and opportunities in SFR materials.

EXECUTIVE SUMMARY

HOW TO APPLY RESULTS

The results of this report can be applied to provide guidance on materials selection for a lead-cooled fast reactor and direct future research and development on SFR-related materials.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- Members of the EPRI Advanced Nuclear Technology Program will be interested in the broad overview of this report as well as the review of potential alloys considered for SFR applications.

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INTRODUCTION

According to the World Nuclear Association, fast neutron reactors (FNRs) are poised for mainstream implementation due to their more efficient use of uranium as compared to water-cooled reactors and their ability to burn long-lived components of nuclear waste [1]. While FNRs are generally considered more technologically challenging than water-cooled reactors, several types of liquid metal-cooled (lead, lead-bismuth, and sodium) FNRs have been designed and operated [2]. Of these advanced designs, sodium-cooled fast reactors (SFRs) have the most extensive experience, with over 400 reactor-years of operation to date 1[, 2]. SFRs include two basic designs: ‘pool-type’ reactors (which are the most common design) and ‘loop-type’ (e.g. the Monju reactor in Japan) [3]. An example of a pool-type reactor systems is shown below in Figure 1-1.

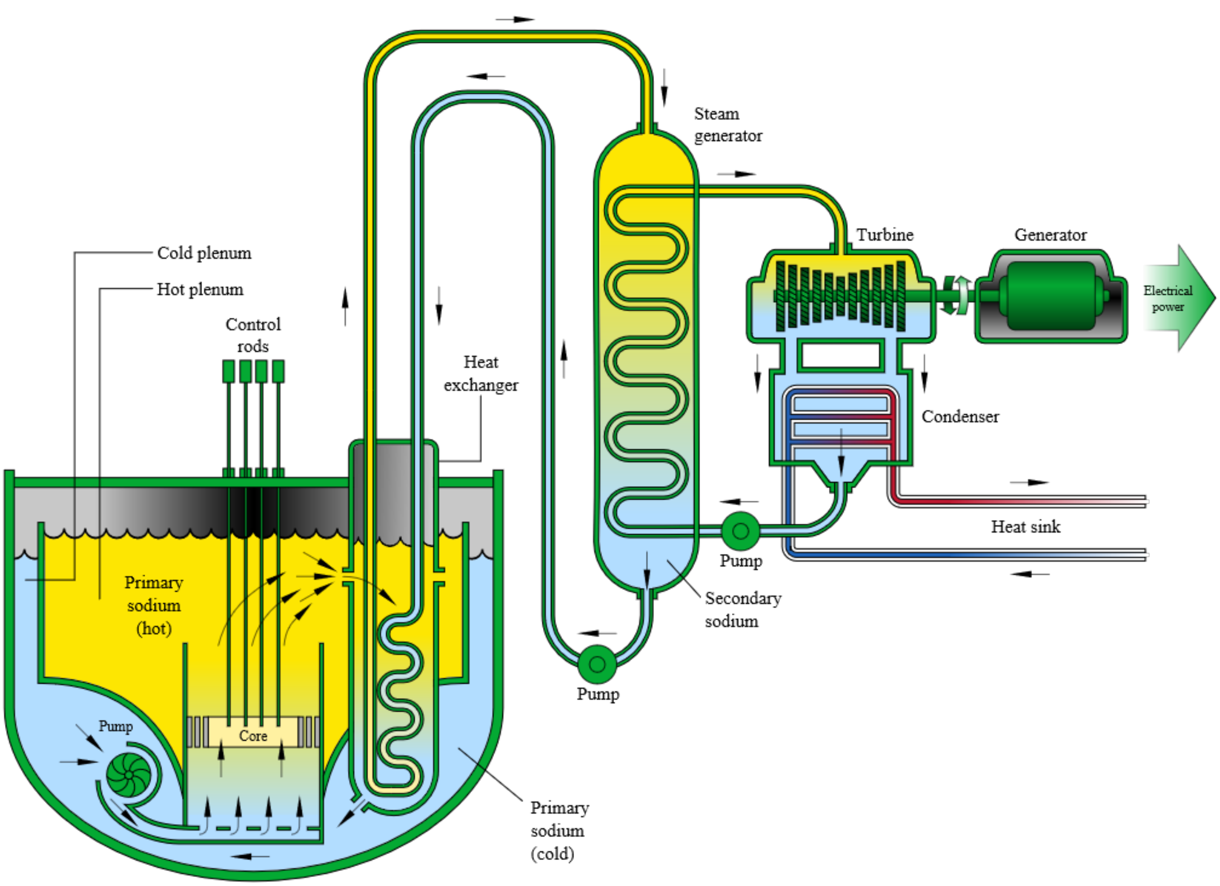

Figure 1-1
Illustration of a pool-type sodium fast reactor [4]

As shown in Figure 1-1, the main features of common SFR's are as follows:

1. The coolant is a low-pressure pool of liquid sodium with a metallic containment vessel and an inert cover gas at the top of the containment.
2. The nuclear core consists of a metallic core support structures and an array of metallic clad fuel pins with either ceramic or metallic fuel.
3. Control rods are typically inserted through the head from the top of the containment vessel.
4. Internal to the vessel are pumps to provide cooling to the core (activated sodium) and additional pumps are located outside of the pressure vessel to circulate the secondary sodium (non-activated) from the heat exchanger to the steam generator.
5. The SFR is mated with a conventional and suitably sized steam generator turbine generator set to produce electric power.

From these basic design features, some key operating characteristics and attributes of sodium fast reactors are listed below. The focus on SFR development stems from the several desirable attributes of both SFR systems and the physical properties of sodium, [2]:

- SFR operating temperatures are on the order of 450-550 which are well above the melting point (~98°C (208°F)) but far below the boiling point (~883°C (1621°F)) of sodium,
- Sodium has relatively high thermal conductivity, which promotes efficient heat transfer (e.g., cooling of core components and removal of decay heat)
- Sodium has relatively low viscosity which facilitates pumping,
- Sodium is compatible with several industrial alloys including both ferritic and austenitic stainless steels and displays good stability under irradiation.

Despite these advantages, SFRs have not seen widespread commercialization due to several factors [5]. These include:

- Global uranium resources are sufficiently high, such that fuel breeding designs have not been economically compelling for power generation reactors.
- Breeder reactors can be costly to build and operate and, in general, sodium-cooled fast reactors have not been as reliable as water-cooled reactors. Several SFR designs have exhibited leaks that caused extensive periods of shutdown. Figure 1-1 compares the cumulative load factors for select SFRs relative to typical conventional water-cooled reactors, showing the much higher reliability of conventional, water cooled reactors.
- Fast neutron reactors have special safety considerations compared to conventional water-cooled reactors. Safety advantages include low-pressure operation and a higher boiling point of the coolant, but significant disadvantages also exist, including, the high chemical reactivity of sodium, the need to control γ-emitting 24Na, and considerations to ensure shutdown of the nuclear chain reaction if there is a loss-of-coolant accident.

- The fast neutron reactor fuel cycle provides access to plutonium that could be used for weapons.
- The fast reactor flux results in high fluence and significant irradiation damage to fuel cladding and components. For example, reactor fuel cladding may see in excess of 150 dpa and the irradiation damage and associated swelling may be limiting [6, 7].

However, significant progress has been made in the technology of handling liquid sodium [2, 8], and SFRs present compelling advantages over other advanced reactor designs. In an international review of fast reactor concepts conducted in 2017, SFRs were shown to exhibit more merits and fewer and more tractable demerits relative to gas fast reactors (GFRs), lead-cooled fast reactors (LFRs), molten-salt reactors (MSRs), and high-temperature fluoride salt reactors (FHRs) [8]. The key merits and demerits of SFRs are summarized in Table 1-1.

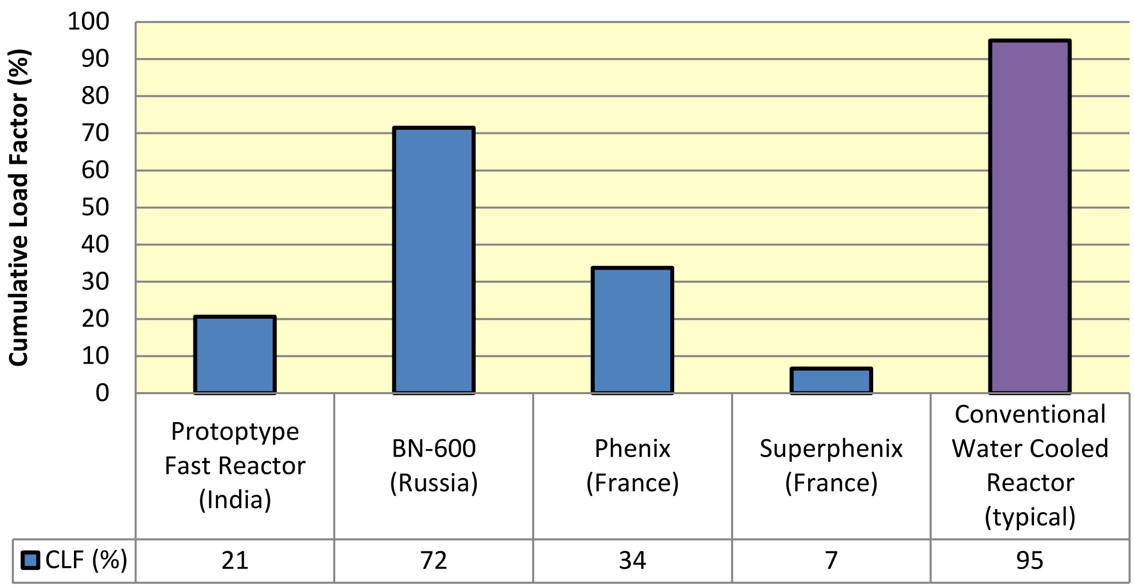

Figure 1-2
Comparison of the cumulative load factors for select sodium-cooled fast reactors, demonstrating concerns with reliability [5, 9]

Table 1-1
Comparison of sodium-cooled fast reactor merits and demerits, summarized from Reference [8]

Merit	Demerit
• Background with and technical maturity of the fuel and fuel cycle • Clear understanding of the remaining technical challenges and the potential for progress before industrial deployment • Potential economic case • Inherent safety due to high boiling point of sodium and low pressure operation • Better fuel utilization relative to light water reactors	• Perception of higher capital costs relative to light water reactor technology • Potential for long unavailability (SuperPhénix experience) • Need for inspection and repair technology development

A summary of historical, operating, and planned sodium-cooled fast reactors by country of operation is listed in Table 1-2 and is displayed graphically in Figure 1-3. Sodium fast reactor operation began in ~1951 with the U.S.-designed breeder reactor, Experimental Breeder Reactor (EBR-I), and reached a peak during 1986-1991 with 13 operating reactors in 7 countries. Starting in 1992, the number of SFRs began to decline, reaching the lowest level of operating reactors (4-5 between 1999 and 2018) since the 1960s. However, consistent with the assessment of the World Nuclear Association [1], planned construction in China, France, India, Russia, South Korea, and the United States is expected to bring the number of operating SFRs up to ~12 by 2030. Given this planned increase in international SFR capability, there is a need to summarize the current state of knowledge on the materials used in these reactors and assess the information gaps that need to be filled for successful SFR implementation. In this regard, the purposes of this report are to:

1. Summarize historical materials performance in sodium-cooled reactors,
2. Conduct an industry survey to elicit the experience and needs of current designers and researchers, and
3. Compile the historical and survey information into a gap analysis that defines needs and fruitful paths for materials development which would advance the safety, performance, and/or economic viability of sodium-cooled fast reactors.

Table 1-2
Summary of sodium-cooled fast reactors by country [1, 2,10, 11]

Country	Sodium-Cooled Reactors	Approximate Dates of Operation
China	China Experimental Fast Reactor CFR-600 CDFR-1000 CDFR-1200	2010–present Planned for 2023 Planned for 2023 Planned for 2028
France	Rapsodie Phénix Superphénix Astrid	1966–1982 1973–2009 1986–1997 2030
Germany	KNK 2 SNR-300	1977–1991 1985–1991
India	Fast Breeder Test Reactor Prototype Fast Breeder Reactor	1985-present Planned for 2019
Japan	Joyo Monju Nuclear Power Plant	1978–2007, restart ~2021 1994–1996, 2010

Table 1-2 (continued)
Summary of sodium-cooled fast reactors by country [1, 2,10, 11]

Country	Sodium-Cooled Reactors	Approximate Dates of Operation
Russia	BR-5 BR-10 BOR-60 BN-350 BN-600 BN-800 BN-1200	1959–1971 1973–1998 1969–2020 1973–1994 1980–2025) 2014–present (construction started)
South Korea	PGSFR	Planned for 2028
United Kingdom	Dounreay Fast Reactor Prototype Fast Reactor	1959–1977 1974–1994
United States	Experimental Breeder Reactor I (EBR-I) Sodium Reactor Experiment (SRE) Experimental Breeder Reactor II (EBR-II) Fermi 1 Southwest Experimental Fast Oxide Reactor (SEFOR) Fast Flux Test Facility (FFTF) S1G (Naval Prototype) S2G (Naval Propulsion) Versatile Test Reactor (VTR)	1951–1963 1957–1964 1963–1994 1963–1966, 1970–1972 1969–1972 1980–1993, 2018–present 1955–1957 1957–1958 Planned for 2026

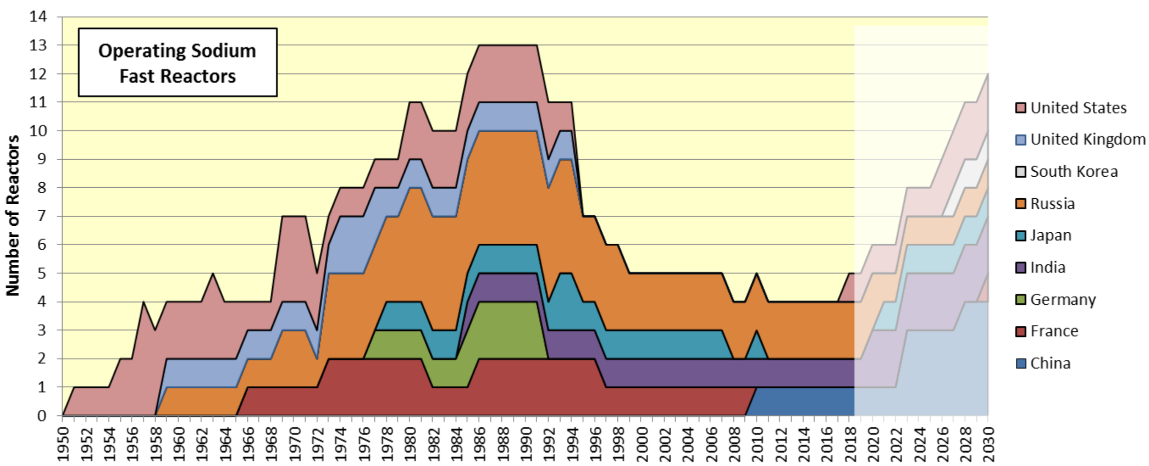

Figure 1-3
Summary of operating sodium-cooled fast reactors by country. Values beyond 2018 are estimates.

2 PREVIOUS REVIEWS OF SFR EXPERIENCE

Several reviews of SFR experience have been performed over the last several years, both on specific designs as well as on the industry as a whole. This section summarizes these reviews chronologically.

2.1 2010 Review of BN-600 Experience

The Russian experience operating the BN-600 sodium-cooled fast reactor was summarized in 2010 [12]. The BN-600 reactor is a pool-type reactor design that has been in operation since 1980 and is licensed for operation until 2025. The fuel cladding was originally made from EI-847 (18Cr-9Ni-Ti austenitic stainless steel) and was later changed to ChS-68kh.d (chromium-nickel-molybdenum steel), while the fuel assembly jacket material was originally 16Cr-11Ni-3Mo austenitic stainless steel, which was ultimately replaced by EP-450 (13Cr-1.5Mo-0.5Nb-0.22V-0.20Ni) ferritic steel. During early operation, several sodium leaks occurred, i.e. 27 leaks to the outside (five of which were of radioactive sodium) and 12 leaks in steam generators. Corrective actions included operator training, design and operating changes, as well as the identification and minimization of manufacturing defects. Based on the effectiveness of these actions, the Authors of Reference [12] highlight:

1. The last sodium leak to the outside occurred in May 1994.
2. Only one small leak from a steam generator occurred in the prior 24 years (Jan 1991).
3. The failures that have occurred in recent years are not associated with sodium systems.

Based on the high reliability of sodium-containing systems in the later operation of the BN-600, the Authors conclude that existing reactor materials and technology are sufficient for commercial power generation and that the lessons learned can further improve new (i.e., fourth generation) sodium-cooled fast reactors [12]. However, it is important to note that an independent review of the BN-600 experience highlights that the relatively high capacity factor of this unit (~72% per Figure 1-1) is, in large part, due to the willingness of the operators to work through multiple sodium leaks and fires [5]. The difficulty in resolving leaks and returning to operation in a timely manner has been challenging for other SFR designs.

2.2 2011 U.S. Department of Energy Materials Research Needs

The U.S. Department of Energy issued a report on research needs for sodium fast reactor fuels and materials in 2011 [7]. The review was conducted by a panel of technical experts and utilized a methodology to rank the relative importance of phenomena and properties both as to importance to a regulatory body and the maturity of the technology. Table 2-1 and Table 2-2 summarize the technical items identified in that review that ‘need more work’ or have ‘almost no data’ available, respectively. Additionally, these tables identify the current materials of construction of the component of interest.

Table 2-1
Summary of items identified as ‘need more work’ in Reference [7]

Structure and Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database
Internal Piping	316	Primary Na	Corrosion Carburization Radioactive mass transport	Temperature ΔT O and C in Na Na flow velocity Na purification capability	Good	Fairly good, needs system assessment for ABR regarding dynamic carbon level
Mechanical Pump (Impeller, Diffuser)	316	Primary or Secondary Na	Corrosion Fatigue Resistance	Flow velocity Vibration Applied load Na purity Temperature	Good, need to identify vendors	Fairly good
Intermediate Heat Exchanger Shell	304 or 316	Primary Na	Na Corrosion Swelling Thermal creep Irradiation creep Fatigue and Creep-Fatigue Interstitial element transfer	O and C in Na Service life Temperature Mechanical load	Good	Adequate
Intermediate Heat Exchanger Tubes	304, Fe-9Cr-Mo Steel	Secondary Na Inside/Primary Na Outside	Na Corrosion Swelling Thermal Creep Fatigue and Creep-Fatigue Interstitial Element Transfer	O and C in Na Service life Temperature Mechanical load	Good	Adequate

Table 2-1 (continued)
Summary of items identified as 'need more work' in Reference [7]

Structure and Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database
Secondary System Pump	TBD	Secondary Na	Corrosion Fatigue Resistance	Flow velocity Vibration Applied load Na purity Temperature	Good, need to identify vendors	Fairly good
Sodium Piping	316	Secondary Na	Corrosion Carburization Radiative Mass Transport	Temperature ΔT O and C in Na Na flow velocity Na purification capability	Good	Adequate
Recuperator	TBD	CO2, Moisture	Oxidation Carburization Creep Fatigue Creep-Fatigue	Temperature Gas purity Applied load	Adequate?	Probably good

Table 2-2
Summary of items identified as 'almost no data' in Reference [7]

Structure and Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database
Electromagnetic Pump	TBD	Primary or Secondary Na	Corrosion Fatigue resistance Electrical compatibility	Flow velocity Vibration Applied load Sodium purity Temperature Electrical interference	Unknown	Poor
Compressor	TBD	CO_2, Moisture, Impurities?, High Pressure	Oxidation Carburization Creep Fatigue Creep-Fatigue	Temperature Gas Purity Applied Load	Probably Adequate	Limited
Turbine Generator	TBD	CO_2, Moisture, Impurities?, High Pressure	Oxidation Carburization Creep Fatigue Creep-Fatigue	Temperature Gas Purity Applied Load Gas Velocity	Unknown	Seriously Lacking
Na to CO_2 Heat Exchanger	TBD	CO_2, Moisture, High Pressure	Oxidation Carburization Creep Fatigue Creep-Fatigue Sodium-CO_2 Reaction	Temperature ΔT Gas Purity Applied Load Tube/Channel Failure Plugging Thin Section Material	Unknown	Seriously Lacking

Consistent with the degree technical maturity of SFRs in the United States, items listed as ‘need more work’ in Table 2-1 are reasonably limited. Areas to highlight include carbon monitoring in primary piping and vendor development for both primary and secondary system mechanical pumps. The more critical items identified as ‘almost no data’ in Table 2-2 include knowledge development and fabrication capability for industrial electromagnetic pumps, knowledge development for compressors, and both knowledge development and fabrication for turbine generators and sodium-to-CO_2 heat exchangers.

This report concluded that an SFR could be designed and licensed based on the technology base and operational experience of EBR-II and the FFTF. However, the design would be constrained within the limitations of that technology base. The limitations to operation from a fuels and materials perspective include:

- Oxide or metal fuel with a maximum burnup of 10 at %,
- Peak cladding temperature of 600°C (1112°F),
- Insufficient database on the performance of D9 stainless steel for the cladding and duct material,
- Peak irradiation exposure of 100 dpa on the cladding and duct,
- Use of ‘fresh’ fuel (fuel with neither addition of minor actinides or fission product carry-over from reprocessing), and
- Limited load-following operation for oxide fuel.

Additionally, the report discussed high level concerns regarding the state of and ability to retrieve information from the U.S. technology base. These concerns were highlighted in a subsequent publication [7, 13] that is described below in Section 2.3.

2.3 2012 U.S. Department of Energy Review

The expert panel that conducted the review discussed in Section 2.2 [7] subsequently published a high-level summary of their recommendations in 2012 [13]. This summary highlighted the state of knowledge of SFR fuels and structural materials according to: (1) the importance of the phenomena with respect to regulatory and reliability concerns, (2) the state of experimental databases, and (3) the state of quantitative understanding of the phenomena. Only non-proprietary and publicly available data were assessed in this review. Based on this assessment, the panel identified the following gaps:

- For fuel with zero to moderate burnup:
  - “An effort should be made to inventory the existing data base, collect the hard copy information and store it in approved storage locations, and transfer this information to an electronic data base that can be readily queried”
  - “Exactly the same effort should be carried out for the fabrication information”, i.e., the fuel fabrication information should be summarized and transferred to an electronic database that can be readily queried.

- For fuel cladding:
  - For D9 stainless steel fuel cladding: More information is needed relative to fuel-cladding chemical interactions for reprocessed fuel with fission product carry-over, particularly the issue of lanthanide migration to the fuel/cladding interface of metallic fuel
  - For HT-9 and T91 ferritic/martensitic alloys:
    - High dose/high temperature swelling data are needed.
    - Vendors are needed that can produce small heats of reactor grade material.
- Fuel performance codes:
  - Most codes are empirically based and are only useful for interpolation when adequately validated with existing data.
  - Few people are adept at using existing codes and documentation is not adequate to train new users.
- Overarching gaps:
  - A test SFR is needed to enhance the existing knowledge base.
  - There is uncertainty in the preservation state of the existing knowledge base. Information and data exist in several locations in a variety of media. It is extremely important to preserve the existing data base.

2.4 2012 Materials Performance Review

Materials performance in sodium was the subject of an extensive review in 2012 by Furukawa and Yoshida [14]. This review covered materials selection, corrosion mechanisms, the effects of sodium exposure on the strength of steels, damage to steels due to sodium-water interactions and tribology in sodium. The review does not explicitly identify knowledge gaps, but it outlines known issues and some methods to mitigate these concerns. The key issues from each section are highlighted below.

- Materials selection:
  - Since oxides are readily reduced in sodium, dissolution of alloying elements occurs due to differences in chemical potential.
  - For austenitic stainless steel, nickel has high solubility in sodium relative to iron and chromium, thus nickel loss and destabilization of the austenite can occur with long time exposure to sodium.
- Corrosion mechanisms:
  - The corrosion rate of alloys is often governed by the solubility of their alloying elements in sodium and solid-state diffusion. While solubility is often low isothermally, thermal gradients can drive corrosion by precipitating out corrosion products in the cold leg of flow loop, the accumulation of which can lead to plugging of the cold leg.
  - Preferential loss of some alloying elements (e.g., nickel, carbon) can lead to phase instability as well as significant changes in mechanical properties.

- Dissolved oxygen in sodium greatly accelerates corrosion rates and therefore must be minimized. Technologically this can be accomplished via cold trapping the oxygen.
- Other impurities such as hydrogen, carbon, and nitrogen likely have to be considered in relation to the alloys they are in contact with and controlled accordingly.
- Corrosion rates typically increase with increased sodium velocity.

- Effect of sodium on the strength of steels:
  - Loss of major alloying elements via corrosion
  - Carburization/decarburization
  - Effects of the reducing nature of sodium (e.g., reducing oxides)
- Damage to steels with sodium compounds:
  - **Sodium-water reaction:** The exothermic reaction between sodium and water results in local heating and produces NaOH and hydrogen gas. Both corrosion and erosion-corrosion (wastage) may occur and promote unstable fracture.
  - **Sodium leak to air:** Sodium reacts with air to typically produce Na_2O (s,l). Excess moisture in air can further drive corrosion via molten-salt type corrosion.
- Tribology
  - Since sodium reduces most oxides, components in intimate contact are prone to self-welding and frictional wear.

2.5 2017 Argonne National Laboratory Research Roadmap

In 2017, Argonne National Laboratory issued technical development roadmaps for two fast reactor concepts: a sodium-cooled fast reactor (SFR) and a lead-cooled fast reactor (LFR) [8]. The report noted the superior technical maturity of the SFR relative to the LFR and identified four SFR designs that could be deployable in the U.S. by the 2030s: the GE PRISM, the Advanced Reactor Concepts ARC-100, the TerraPower TWR-P, and the Department of Energy ABR. Additionally, the technical readiness level (TRL) of key components and systems were assessed from high readiness (10) to low readiness (1). This TRL assessment is reproduced in Table 2-3, which shows that the areas of highest concern (TRL 3-4) are for licensing experience and safety regulations, while technical components have fairly high TRL levels (6-8), with the exception of fuel handling and interim storage systems, which range from 5 to 7, depending on the design.

The Argonne report further details technologies that could be developed for commercial demonstration reactors, targeted in the 2030 timeframe and in commercial reactors by 2050, the list of which is reproduced in Table 2-4. Note that the assessment includes materials selections for the key components of the reactor system. This assessment is predicated on the goal of completing a demonstration reactor around 2030, which requires the use of high TRL technologies. The technologies targeted for 2050 were selected with the aim that implementing them would significantly enhancing commercial reactor performance.

Table 2-3
Technical readiness levels for commercial demonstration of a sodium-cooled fast reactor by the early 2030s in the United States [8]

Key Component	System	Technical Readiness Level
Nuclear Heat Supply	Fuel Element	7–8
	Reactor Core Internals	7
	Reactivity Control Mechanism	7
	Reactor Enclosures (vessels, overhead)	7
	Operations/Inspection/Maintenance	6
	Core Instrumentation	6
Heat Transport	Coolant Chemistry Control/Purification	6
	Primary Heat Transport	6
	Intermediate Heat Exchangers	7
	Pumps/Valves/Piping	6
	Residual Heat Removal	6
Power Conversion	Turbine	8
	Steam Generator	7
	Pumps/Valves/Piping	7
Balance of Plant	Fuel Handling and Interim Storage	5–7
	Instrumentation and Control	6
	Radioactive Waste Management	6
Safety	Inherent (Passive) Safety Features	6
	Active Safety System	6
Licensing	Safety Design Criteria and Regulations	4
	Licensing Experience	3
	Safety and Analysis Tools	7

Table 2-4
Sodium-cooled fast reactor technologies targeted for 2030 demonstration and 2050 commercialization [8]

	2030 Demonstration	2050 Commercialization
Objective	**Commercial Demonstration Reactor**	**High-Performing Reactor with Closed Fuel Cycle**
Fuel	U-Zr or U-Pu-Zr	U-TRU-Zr High burnup based on fission products-vented fuel
Fuel Cladding	HT-9	Advanced alloy
Reactor Structural Materials	Existing ASME code-qualified austenitic stainless steels for 60-year lifetime components Low-chromium ferritic steel for replaceable steam generator design	Advanced ferritic-martensitic stainless steel (modified 9Cr-1Mo) and advanced austenitic stainless steel (Alloy 709)
Primary Pump	Mechanical centrifugal pump (submersible electromagnetic (EM) pump if further developed)	Mechanical centrifugal pump or submersible EM pump
In-Vessel Refueling System	Dual plug system with straight pull and/or fixed arm Single rotatable plug system with pantograph in-vessel transfer machine	Dual plug system with straight pull and/or fixed arm Single rotatable plug system with pantograph in-vessel transfer machine
Reactivity Control System	Primary: segmented-arm control rod drive mechanism with gripper Secondary: drive motors with gravity insertion and fast drive in	2030 technology with possibility of EM latch
Core Restraint System	Engineered limited free bow core restraint design	Engineered limited free bow core restraint design
Power Conversion Cycle	Rankine/steam	Rankine/steam or SCO_2 Brayton
Steam Generator	Separate evaporator and super-heater Once-through design	Once-through design of sodium to CO_2 heat exchanger
Instrumentation and Control	Analog-digital or all-digital hybrid plant control system and protection system, depending on NRC approval Single sensor alarms	Cyber security Supervisory control system System-level automated data reconciliation

Table 2-4 (continued)
Sodium-cooled fast reactor technologies targeted for 2030 demonstration and 2050 commercialization [8]

Objective	2030 Demonstration Commercial Demonstration Reactor	2050 Commercialization High-Performing Reactor with Closed Fuel Cycle
In-Service Inspection	Under-sodium viewing system for in-vessel refueling at refueling temperature Inspection robot for reactor and guard vessels	Under-sodium viewing system for on-line monitoring at core outlet temperature Automated inspection technology for reactor and guard vessels Under-sodium repair technology for fast reactor applications
Safety	Credit for inherent safety based on previous reactor operational experience and calculations	Probabilistic risk assessment
Licensing	Two-step licensing based on 10CFR Part 50	Two-step licensing based on 10CFR Part 50 or one-step licensing based on 10CFR Part 52
Fuel Cycle	Once-through	Fully closed

3
U.S. INDUSTRY SURVEY

As part of this report, stakeholders in U.S. sodium fast reactor development field were surveyed to identify issues and materials gaps for which they would like to see additional research and development. Responses were received from ARC Nuclear, GE-Hitachi, Idaho National Laboratory and TerraPower LLC. Their inputs are summarized in Table 3-1 and detailed below.

The survey responses highlighted that their designs operate based on technology proven in EBR-II, FFTF or other proven operation, thus they can be constructed with commercially available materials. However, it is noted that a commercial supply of the ferritic/martensitic (F/M) steel HT-9 may be limited at the present time and could warrant further development.

Responses also noted that next generation SFRs would benefit from improved fuel cladding and assembly materials. For example, while the excellent swelling resistance of HT-9 helps resolve a primary concern in these applications, material formulations that would also present improved creep-rupture strength and irradiation creep resistance would be even more advantageous. Additionally, an interdiffusion barrier between the fuel and cladding would be desirable as it would enable increased fuel burnup and higher fuel temperature limits.

Demonstrating the properties and performance of F/M steels in liquid sodium and the development of hardfacings for use in sodium environments was also seen as being highly beneficial and an area of future R&D support. Specific areas suggested for future research incuded:

- Industrial supply of nuclear grade F/M steels
- Material Properties in Sodium
  - Long time creep data for F/M steels
  - High temperature creep rupture data for F/M steels (i.e., 700-800°C (1290-1470°F))
  - Creep-fatigue behavior of F/M steels
  - Coefficients of friction for F/M steels in sodium
  - Any F/M material property data acquired under NQA-1 (or equivalent) quality assurance
- Materials Performance in Sodium
  - Long time corrosion data for F/M steels
  - Quantification of the effects of oxygen, temperature, and flow velocity on corrosion in sodium
  - Fretting and self-welding behavior of F/M steels in sodium
  - Environmentally assisted cracking data of F/M steels in sodium

Table 3-1
Summary of input from ARC Nuclear, GE-Hitachi, INL, and TerraPower on desired materials development work to advance sodium fast reactor development

Category	Gap of Area of Potential Benefits with Future Research
Industrial base	Industrial supply or nuclear quality ferritic/martensitic steel (e.g., HT-9)
Material properties in sodium	Creep Creep-fatigue Tribology of F/M steel
Material performance in sodium	Improved cladding* Corrosion data, including effects of O_2, T, and flow Effects of self welding and rates of wear Environmentally assisted cracking on F/M steel
Other materials development	Interdiffusion barrier between clad and fuel* Hardfacing suitable for use in Na and under irradiation

* Desired for future improvement

4
RECENT MATERIALS RESEARCH

In addition to the surveys of SFR technology discussed in Sections 2 and 3, the recent scientific literature focused on sodium fast reactors is an important resource to help identify knowledge gaps. Toward this end, a brief survey of the technical literature was conducted. As detailed below, there appears to be significant efforts to bridge technical gaps in areas such as matching weld joint performance to that of base materials, understanding key fuel/cladding interactions, and developing sensors for use in sodium fast reactors.

Mathew et al., [15] noted that delivering long life of structural components (~60 years) is key to successful economics of employing SFRs for power generation. Toward that goal, they report on studies to increase the creep strength and low cycle fatigue resistance of 316L(N) stainless steel base materials and welds, as well as the creep performance of a 9Cr-1Mo steel and its welds for use in steam generators. Mathew et al., showed that for the 316L(N) material, increased nitrogen content increases creep life but that low cycle fatigue life saturates at a nitrogen content of ~0.14 wt.%. In this material the base metal has superior creep and fatigue resistance relative to its welds. Similarly, the creep rupture strength of 9Cr-1Mo steel welds was significantly lower than that of the base metal. In this material the reduced properties in the weld are due to what is termed 'Type IV' failure in the heat-affected zone. This 'Type IV' cracking describes failure in the fine-grained, inter-critical region of the heat-affected zone and is associated with accelerated creep rupture processes in that microstructure. This research points to potential deficiencies in weld mechanical properties that may limit the performance of assembled components. There is therefore need for more research and development of welding methods for these advanced materials. More recent research by Bhaduri and Laha indicates that additions of boron and nitrogen to 9Cr-1Mo steel results in significantly improved resistance to Type IV cracking [16].

As noted in the 2012 U.S. DOE survey [7, 13] and in the current U.S. industry survey (Section 3), fuel/clad interactions are a key area for future research. Recently, Taeil et al., reported on the interaction of molten uranium with a liquid sodium-filled fuel pin [16]. As expected, this work showed the products of the eutectic reaction with the ferritic/martensitic steel HT-9M as [16]:

- Liquid (U- and Fe-rich) → UFe_2 (solid) + U_6Fe (solid) ~ 725°C (1337ºF)

The observations of increased UFe_2 near the cladding and U_6Fe near the fuel are pertinent to properly identifying potential fuel failure mechanisms. Future work will seek to conduct experiments under higher pressure to better understand fuel relocation behavior in SFRs [16].

A key aspect of the safe and reliable operation of sodium-cooled fast reactors is to ensure that the oxygen content in the liquid sodium remains low. While discrete sampling and analysis of primary or secondary liquid sodium is feasible, the time required for these processes makes then of limited value for reactor control. Instead, an on-line oxygen sensor would be highly desirable. Toward that goal, Nollet et al., developed an electrochemical oxygen sensor for use in liquid sodium [17]. While that work demonstrated that the sensors produce a signal that varied with the oxygen concentration and was similar to electrochemical theory (i.e. the electrochemical potential decreased as the oxygen concentration in the sodium increased), several shortcomings were noted and areas for future improvement and research were suggested. Their research showed that yttria-stabilized thoria (YST)-based sensors are more stable in liquid sodium and are superior to yttria-stabilized zirconia (YSZ) sensors. Understanding the non-equilibrium processes that are influencing the operation of these sensors is a key area for future work.

5
GAP ANALYSIS

Based upon the foregoing discussion of the experiences from operating SFR's and the designs identified for advanced SFR's and the requirements for the materials identified for structural components and fuel cladding to meet performance and life targets for these reactors can be compiled as a systematic schema of knowledge status, knowledge gaps and the required research actions needed to close these gaps. Table 5-1 has been developed to provide an organized gap analysis which the author believes exist for meeting design requirements and for demonstrating that the materials identified for them can deliver performance and durability. This table systematically considers the key design requirements and then the materials considerations (including the specific materials information needed to demonstrate the materials can meet the design requirements) for the major segments(subsystems) of the SFR designs. The subsystems considered in Table 5-1 are the overall vessel shell and piping, fuel cladding, secondary piping and pumping, the highly irradiated vessel internals structures, heat exchanger and secondary/tertiary systems that mate with the turbine-generator systems required to deliver electrical power from grid-installed SFRs.

Table 5-1
Summary of the status, technical gap, and suggested actions from this review of sodium fast reactor technology

Status	Gap	Action
For near term systems designs can be licensed within existing limits (see text) employing materials, constructions and operating procedures based on previous SFRs (EBR-II, FFTF, PHENIX etc.)	Ready availability of industrial components such as electromagnetic pumps, compressors, turbine generators, and heat exchangers	A strategic approach should be defined to determine how to develop a more extensive supply base
For near term systems designs can be licensed within existing limits (see text) employing materials, constructions and operating procedures based on previous SFRs (EBR-II, FFTF, PHENIX etc.)	Accessibility of previously developed technical information (Rationale for design features, Material specification methodologies, properties, optimized processing routes etc.) is not high and information storage is distributed	Develop modernized central storage for all previously developed in formation
316 SS is considered as a material for internal piping for near term system exposed to liquid Na	Long time resistance of 316 to corrosion and carburization, loss of Ni from spinel oxides etc. and consequent effects on e.g., mechanical properties under operating conditions are not accurately/fully known	Develop experimental measurements and models to describe potential carburization from Na environment and effects on mechanical properties. Employ such data to identify the required controls on Oxygen content of molten sodium for long term operations
316CW is considered as a shell material for near term system	Creep and Creep-Fatigue data in the environment are needed to support design	Creep and Creep Fatigue data need to be generated and incorporated into Sec III Div 5 of ASME code
316CW or D9 are considered as materials for fuel cladding and ducting in near term systems	Response of material to high, end of life fluence (>100dpa) is not known. In particular swelling and degree of embrittlement at high fluence need to be quantitatively identified for >100 dpa fluence exposure at operating temperature	Develop data/models that can reliably predict swelling and mechanical properties after high dpa/high temperature (600°C (1110°F)) exposures of 316CW and D9 austenitic stainless steels. Need high temperature data (to sufficient temperature) incorporated into Sec III Div 5 of ASME code (If properties of 316CW and D9 austenitic stainless steels are inadequate to meet design requirements see potential use of HT-9 in advanced system sections)

Table 5-1 (continued)
Summary of the status, technical gap, and suggested actions from this review of sodium fast reactor technology

Status	Gap	Action
Materials for secondary systems (pumping, piping) are "TBD" for near term systems. Materials with experience in Na environment are available.	Cost vs longevity tradeoffs are not explicitly available	Cost effectiveness/Life viability assessments need to be defined to support specific designs
Near term systems expected to use mechanical pumping systems but EM pumping could be a possibility if systems can be matured.	EM pumping systems are not mature, and materials selection is not defined Effect of primary and secondary Na on expected fatigue life of moving and structural parts is not explicitly known	Develop increased database of fatigue performance of candidate materials (316 Stainless steel) in molten Na
Near term systems will use conventional Rankine/Steam cycle with conventional back end flow and TG systems	Long time service of appropriately sized components is not demonstrated (Are key components of the right size commercially available?)	Assess advanced required to enable higher temperature operation and alternate cycles
Near term systems call for moving metal parts to slide against each other.	Relative motion of metallic parts in Na can be affected by dissolving of usually lubricating oxide surfaces and producing the potential for fretting	Resolution of effect of corrosive effect of Na chemistry (O content etc.) with regard to affecting surface conditions and promoting fretting. Identification of limits on Na coolant to maintain metal surface sliding capability Suitable hardfacing options could be research and developed
Ferritic-Martensitic Steels such as HT-9 (9Cr-1Mo) steel offer the potential for reduced swelling performance in irradiated components of SFR (as improved performance substitute for austenitic stainless steels) in near term and advanced SFR systems	Ferritic-Martensitic Steels such as HT-9 (9Cr-1Mo) steel are not available in large sections and sufficient volume to support fabrication of plant components	Development of large-scale production capability of Ferritic Martensitic HT-9 steel. Utilize production lots for materials testing to establish data for incorporation of HT-9 in Sec III Div. 5 of ASME code

Table 5-1 (continued)
Summary of the status, technical gap, and suggested actions from this review of sodium fast reactor technology

Status	Gap	Action
Ferritic-Martensitic Steels such as HT-9 (9Cr-1Mo) steel offer the potential for reduced swelling performance in irradiated components of SFR advanced SFR systems	Low swelling of HT-9 at high fluences for end of life is postulated from lower fluence data. For implementation this behavior needs to be confirmed	Experimentally demonstrate low swelling behavior of Ferritic- Martensitic HT-9 steels
Ferritic-Martensitic Steels such as HT-9 (9Cr-1Mo) steel offer the potential for reduced swelling performance in irradiated components of SFR advanced SFR systems	Fabrication issues including welding of Ferritic-Martensitic Steels have not been reliably resolved	Demonstration of reliable and reproducible welding of HT-9 with acceptable long-term properties (Creep, Creep-fatigue)
Fuel performance codes are used to predict the performance of new fuel systems identified for advanced SFR systems	Most codes are empirically based and most useful for interpolation from previously validated data. Also, there is a low experience base that has experience usage of such codes. Valid extrapolation of codes to advanced systems has not been demonstrated.	Development and validation of improved fuel codes and establishing of experience base for use of these codes.
Future reactors will utilize advanced fuels are needed in order to close the fuel cycle and to enable economically competitive power generation	Advanced Fuels Development is needed	Key issues on advanced fuel performance and potential fuel / clad interactions need to be assessed
Future reactors will utilize advanced fuels are needed in order to close the fuel cycle and to enable economically competitive power generation	Advanced fuel claddings are needed to support advanced fuels. Fuel Clad interactions are unknown	Development of matched fuel-clad systems with possible barrier layers
Thin walled tubing is needed for Na to CO_2 heat transfer in heat exchanger in advanced systems	Lack of knowledge of simultaneous thermal gradient, Na corrosion on one side and possible carburization on the other (with potential diffusion of C through wall ?)	Development and qualification of higher strength alloys such as Alloy 709 may address this gap

Table 5-1 (continued)
Summary of the status, technical gap, and suggested actions from this review of sodium fast reactor technology

Status	Gap	Action
Turbine/Generator components in advanced systems will be required to operate in moist CO_2 environments (vs Air or H_2 in conventional TG sets)	Long life of T-G components operating in moist CO_2 environments is not proven	Requires a systematic review of expected performance of conventional materials of construction in moist CO_2 atmosphere and validation testing
Significant time gap between previous and currently planned build and operations of SFRs	Due to new materials and lost prior experience there is incomplete experience/knowledge base regarding SFR construction and operations	Develop a test SFR to confirm existing knowledge base and provide a test bed for new/improved materials and validation of new methods of component construction

6 SUMMARY

This report has summarized the historical materials performance in sodium-cooled fast reactors and has identified gaps in the current state of knowledge that should be closed to improve the safety, performance, and economic viability of these advanced reactor designs. The main conclusions from this work are as follows:

- Despite their technological complexities relative to water-cooled reactors, sodium-cooled fast reactors are poised for resurgence in construction due to their demonstrated commercial feasibility as well as inherent advantages in safety and fuel utilization. From a current low of four operating reactors internationally, approximately twelve sodium-cooled reactors are planned to be in operation by 2030.
- An evaluation of review articles on advanced reactor development confirms that sodium-cooled fast reactors are the most mature technology for future deployment.
  - In the U.S., critical issues to enable deployment of sodium fast reactors are primarily focused on the availability of industrial components such as electromagnetic pumps, compressors, turbine generators, and heat exchangers. For these components, demonstration of appropriate materials of construction is a common issue.
  - In addition to the technical issues above, development of appropriate safety design criteria, licensing regulations, and licensing experience were identified as critical issues for U.S. sodium fast reactor development.
  - For future designs, improved materials such as advanced fuel and advanced fuel cladding are areas for development in order to close the fuel cycle and to enable economically competitive power generation.
- An industry survey of U.S. stakeholders highlights needs for development of industrial vendors of nuclear-grade ferritic/martensitic steels. Additionally, advanced SFR designs such as the TerraPower Travelling Wave Reactor would benefit from materials testing programs to better define physical and mechanical properties in liquid sodium.
- A review of recent research reveals that several areas for improvement identified by expert reviews are being actively studied. These include improving weld joint properties to better match base metal performance, researching fuel/clad interactions to improve safety, and developing improved sensors to increase SFR reliability.

Specific gaps relative materials that have already been identified for components in the SFR have been compiled. These gaps and suggested research actions have been compiled (in Table 5-1).

7
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