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**ADVANCED NON-LIGHT-WATER REACTORS MATERIALS AND
OPERATIONAL EXPERIENCE**

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TABLE OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AGR	advanced gas-cooled reactor
AISI	American Iron and Steel Institute
ANLWR	advanced non-light-water reactor
ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing and Materials
AVR	Arbeitsgemeinschaft Versuchsreaktor
B&PVC	Boiler and Pressure Vessel Code (ASME)
C	Celsius
CRC	central rotating column
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor II
F	Fahrenheit
ft	foot
FBTR	fast breeder test reactor
FFTF	Fast Flux Test Facility
FSV	Fort St. Vrain
gal	gallon
HGC	helium gas compressor
HTGR	high-temperature gas-cooled reactor
HTR	high-temperature reactor
HTTR	high-temperature test reactor
IAEA	International Atomic Energy Agency
IHX	intermediate heat exchanger
kg	kilogram
lb	pound
LBB	leak-before-break

LMFR	liquid metal fuel reactor
LMR	liquid-metal-cooled reactor
LWR	light-water reactor
MIR	Fast Reactor Inspection Module
mm	millimeter
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
MW _e	megawatt electric
MW _t	megawatt thermal
NDE	nondestructive evaluation
NGNP	Next Generation Nuclear Plant
NIV	neutron-induced void
NRC	U.S. Nuclear Regulatory Commission
OD	outside diameter
OpE	operating experience
PCRV	prestressed concrete reactor vessel
PFR	prototype fast reactor
PIRT	phenomenon identification and ranking table
PNNL	Pacific Northwest National Laboratory
PVC	polyvinyl chloride
RPV	reactor pressure vessel
RSS	reserve shutdown system
SFR	sodium-cooled fast reactor
SG	steam generator
SONAR	sound navigation and ranging (system)
SS	stainless steel
THTR	thorium high-temperature reactor
xLPR	Extremely Low Probability of Rupture (code)

EXECUTIVE SUMMARY

This report summarizes the available domestic and international operating experience (OpE) for both power and research advanced non-light-water reactors with regard to materials and component integrity. It focuses on both sodium-cooled fast reactors (SFRs) and high-temperature gas-cooled reactors (HTGRs). The OpE among salt reactors is limited (i.e., 4 years at the Molten Salt Reactor Experiment at Oak Ridge National Laboratory (1965–1969)). The available information on this experiment was captured in “Technical Gap Assessment for Materials and Component Integrity Issues for Molten Salt Reactors” (Agencywide Documents Access and Management System Accession No. ML19077A137).

This report identifies OpE relevant to the following:

- materials used, including a summary of the range of materials used in both SFRs and HTGRs
- observed and anticipated material degradation mechanisms for both SFRs and HTGRs
- component integrity issues
- possible solutions to challenges involving materials and component integrity
- specific issues based on OpE that should be addressed in the development of regulatory infrastructure
- assessment tools and evaluation techniques (e.g., nondestructive evaluation (NDE)) used to identify and address component integrity issues

This effort identifies OpE in SFRs, HTGRs, and related test loops to aid the U.S. Nuclear Regulatory Commission’s preparations for licensing SFRs and HTGRs. The components of interest include, but are not limited to, primary and secondary piping, steam generator (SG) components, pumps, and reactor pressure vessels (RPVs). Future companion reports will identify gaps in consensus codes and standards and computational codes used in the construction and operation of SFRs and HTGRs.

This report makes the following significant findings related to SFRs:

- The Experimental Breeder Reactor II (EBR-II) never experienced a sodium/water reaction because the EBR-II was designed with double-walled (concentric tubes brazed or swaged together) SG tubes. Other SFRs that did not use double-walled SG tubes experienced sodium/water reactions.
- Corrosion of immersed stainless steel (SS) components in sodium is not a concern if sodium purity is maintained. No significant corrosion of materials in the sodium circuits of the BN-10 reactor occurred during 44 years of operation; therefore, a 60 year life of piping circuits and future SFR designs is possible.
- Weld design and quality control are critically important. Residual weld stresses, excess weld metal, and weld constraints should be minimized, and direct tube-to-tube-plate welds should be avoided entirely in SFRs. Lowering the threshold for quality in welds and secondary loops has resulted in operational problems.

- Reheat cracking is a concern in SFR components operating at high temperatures. In particular, weld repairs should be carefully managed because they may give rise to very high tensile residual stresses. Some types of austenitic SS (e.g., American Iron and Steel Institute (AISI) 321 SS) are significantly more prone to reheat cracking than other austenitic SS.
- Stresses induced by thermal expansion, particularly in areas of constraint, must be carefully considered. The stresses have often been the source of structural integrity issues in SFR operation.
- Thermal fatigue (thermal striping) caused by mixing sodium flows at different temperatures is a significant issue in SFRs.
- Management of the startup and cooldown transients in SFRs to control vibration, thermal expansion loads, and possible fatigue issues are important.
- Shrink-fit parts should be avoided because they could loosen during thermal transients.
- Electromagnetic pumps have operated reliably.
- Oil-based lubricants should be avoided in SFRs.
- Possible valve failures (for all system valves, especially those operating at high temperature) are a concern for SFRs. Valve reliability under operating conditions should be determined accurately.
- Austenitic steels are unsuitable for SFR SGs because of the potential for caustic stress-corrosion damage following even small leaks.
- Testing should confirm the chemical compatibility between molten sodium and insulation material.
- Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure and sodium leaks.
- Accurate detection methods of corrosion and leaks are necessary, particularly in regions coated with insulation. The design phase should consider sensor placement and reliability under operating conditions. Inadequate or unreliable leak detection systems have resulted in extensive shutdowns caused by sodium contamination and excessive sodium leaks with consequent fires.
- The licensing process needs to scrutinize seismic and external dynamic loading events of SFRs. During an emergency shutdown (scram), the intermediate heat exchanger may experience thermal shock caused by the influx of cold sodium. This could lead to buckling and structural issues amplified by an external loading.

The report makes the following significant findings related to HTGRs:

- The accurate prediction of core temperatures in HTGRs is problematic. Even for more recent designs (e.g., the high-temperature test reactor), core temperatures have exceeded anticipated design temperatures.

- Moisture ingress and leakage events are a reoccurring problem with HTGRs. HTGRs should be designed to accommodate and mitigate moisture ingress.
- Failures within the SG could introduce water into the primary loop and introduce the potential for unanticipated reactivity.
- Management of thermal stresses is important in HTGRs because thermal expansion stresses can cause large loads and creep.
- The design of HTGRs must consider the accumulation of cycles during testing. Failure to account for these additional cycles led to fatigue failures in both Peach Bottom Atomic Power Station and Germany's Arbeitsgemeinschaft Versuchsreaktor.
- Pumps, seals, and compressors have a history of poor reliability in HTGRs. The design and testing of these components should be well scrutinized.
- HTGRs should be designed to minimize sources of graphite dust (e.g., fretting) and should include filters or other mitigating measures to address graphite dust.
- Coarse-grained alloys are used for improved creep resistance; however, they are more vulnerable to cracking. Control of alloy grain sizes should be considered because alloys with excessive grain size may have insufficient toughness.
- Oil-based lubricants should be avoided in HTGRs.
- HTGRs need to ensure the structural integrity of the RPVs and the connecting vessels, especially under low helium flow and loss-of-forced convection conditions because buckling may occur.
- Backup systems must be properly designed to handle overloads and system upsets such as seismic loads.

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1. INTRODUCTION

Several advanced non-light-water reactor (ANLWR) concepts are being considered at this time, including, among others, the following:

- very-high-temperature reactors (with outlet temperatures of 900–1,000 degrees Celsius (C)) (1,652–1,832 degrees Fahrenheit (F))
- sodium-cooled fast reactors (SFRs) (500–600 degrees C) (932–1,112 degrees F), including the traveling wave reactor
- supercritical water reactors (510–625 degrees C) (950 degrees–1,107 degrees F)
- high-temperature gas-cooled reactors (HTGRs) (700–800 degrees C) (1,292–1,472 degrees F)
- lead-cooled fast reactors (480–570 degrees C) (896–1,058 degrees F)
- molten salt reactors (MSRs) (600–800 degrees C) (1,112–1,472 degrees F)

Recent workshops presented by the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) discussed the regulatory challenges and needs related to these ANLWRs [1], [2], [3].

This effort is concerned with examining domestic and international operating experience (OpE), technical gaps, consensus codes and standards, and computational codes for SFRs and HTGRs. This report did not include OpE on MSRs because of the general lack of OpE. Only one MSR, the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory, operated for 4 years (1965–1969) [4]. The available information on this experiment was captured in “Technical Gap Assessment for Materials and Component Integrity Issues for Molten Salt Reactors” (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19077A137) [5]. This report focuses on OpE with materials and component integrity.

NUREG/CR-6944, “Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs): High-Temperature Materials PIRTs,” Volume 4, issued March 2008 [6], ranks the most significant phenomena associated with HTGRs with regard to thermohydraulics, fission products transport, structural graphite, irradiation effects on materials, material stability at elevated temperatures, weld residual stress relaxation section damage, environmental degradation, and fabrication issues, including the properties of heavy section steel used for reactor pressure vessels (RPVs).

NUREG/KM-0007, “NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors,” issued April 2014 [7] (and references cited therein), summarizes knowledge management efforts to develop and compile information on liquid-metal-cooled reactors (LMRs), including issues with the materials in SFRs. NUREG/KM-0007 documents NRC licensing activities and safety analyses associated with U.S. reactors along with international reactors. Appendix A, “List of Documents in the NRC Knowledge Center,” to NUREG/KM-0007 lists many available references. These include information and documentation on LMR severe accidents, operational issues, and analysis tools relevant for licensing purposes.

This effort identifies OpE in SFRs, HTGRs, and related test loops to aid the NRC's preparations for licensing SFRs and HTGRs. References [6] and [7] summarize the expected materials for different components. The components of interest include, but are not limited to, primary and secondary piping, steam generator (SG) components, pumps, and RPVs. This compendium will also be used to inform future work on consensus codes and standards and computational codes used in the construction and operation of SFRs and HTGRs.

2. OVERVIEW

2.1 Background Summary

Multiple domestic and international corporations have stated their intent to conduct prelicensing or licensing activities for ANLWRs with the NRC in the next 5 years. The NRC is seeking to develop additional technical capabilities and update its regulatory infrastructure to license new, innovative ANLWRs in an efficient and effective manner. This report summarizes OpE for domestic and international power and research SFRs and HTGRs. Material and structural degradation issues are the main interest; however, the report also describes nondestructive evaluation (NDE) experience. This document summarizes OpE from publicly available documents. The NRC will use OpE to identify gaps in technology necessary to expand consensus codes and standards (e.g., American Society of Mechanical Engineers (ASME)) and modify or develop computational codes for assessing damage mechanisms of ANLWRs.

This report identifies and compiles service experience and potential issues, including damage development mechanisms and anticipated issues of concern for the NRC licensing of future ANLWRs. It is possible some prior experience with older reactors may not apply to anticipated ANLWR designs. This report also identifies additional issues, to the maximum extent possible, such as crack and damage detection OpE.

Component integrity issues identified for SFRs and HTGRs include:

- sodium/helium leakage, including leakage at seals and pumps
- thermal shock and thermal mismatch
- seismic responsive structures
- creep damage and crack growth
- low cycle fatigue
- void swelling

2.2 Scope of Advanced Non-Light-Water Reactor Operational Experience

For both SFRs and HTGRs, this report obtained and discussed the following OpE:

- materials used in SFRs and HTGRs, including metals, nonmetals (e.g., graphite, ceramic components, and concrete), and protective coatings
- observed and anticipated material degradation mechanisms for both SFRs and HTGRs

- component integrity issues, including operational damage (e.g., metal creep, cracking, fracture, swelling, concrete degradation, SG leakage and plugging, and weld cracking)
- possible solutions to materials and component integrity challenges, including inservice evaluation and repair
- specific issues based on OpE that should be addressed in developing the regulatory infrastructure
- evaluation techniques (e.g., NDE) used to identify and address component integrity issues unique to these ANLWR technologies

2.3 Sodium-Cooled Fast Reactors

Concerns related to SFRs include a focus on preventing leaks of molten sodium coolant because sodium is reactive in the presence of air and water. The potential for thermal shock, thermal mismatch stresses, and seismic response are of particular importance in SFRs. Some SFR designs must manage low-cycle fatigue and creep damage, depending on operating conditions and materials. Furthermore, SFRs have experienced leaks and pump failures at seals. Finally, the secondary sodium loop in SFRs operates at near atmospheric pressure and the steam system operates at high pressure. These significant pressure differentials combined with the reactivity of sodium make it critical to prevent SG tube failures [8]. Guidez et al. [9] discussed issues with the possible interaction of water and sodium in SFR SGs. In these events, pressurized water reacts with sodium in a highly exothermic way, causing a high-speed water/sodium jet to damage nearby structures and produce a corrosive byproduct (sodium hydroxide). If the water leakage rate is large, the tube may swell and subsequently burst. For example, during startup of Phénix [10], excess weld metal at butt welds caused stress concentrations, which led to tube cracking and sodium/water interactions. Sodium/water interactions produce sodium hydroxide and hydrogen, which can be detected to shut down the reactor, if needed. In general, SGs have not been reliable, and sodium heat transport systems have leaked at welds.

Material performance requirements for SFRs differ from those for light-water reactors (LWRs) as follows:

- Typical operating pressures within the sodium coolant areas are low (near atmospheric pressure), although differential pressures between the secondary sodium and the water/steam side of the SGs are higher in SFRs than those seen between the primary and secondary water loops in LWRs.
- Thermal loads are more important. These loads are induced by the high thermal conductivity of sodium combined with high temperatures and temperature fluctuations (thermal shock) and complex fluid flows in some areas. Thermal properties of the materials must be chosen with this in mind.
- During normal operating conditions, temperatures usually range between 200–560 degrees C (392–1,040 degrees F) and are higher during transients. The lower temperatures occur during startup. This requires materials to retain adequate toughness after experiencing creep damage.
- Irradiation effects on structural materials outside the core are less significant.

- Materials must be compatible with sodium and contaminants (e.g., sodium hydroxides).

Material considerations include, among others, sensitivity to material creep, creep fatigue, thermal aging, and performance in sodium. The system must maintain ductility over the life of the plant in case of accidental loadings and material aging, which has affected LWRs. Grades of 304 and 316 stainless steel (SS) maintained good creep and corrosion resistance with high toughness along with Alloy 800 for SG parts, as discussed by Guidez et al. [9]. In addition, three major incidents (discussed later) led to shutdowns another 19 percent of the time.

2.4 High-Temperature Gas-Cooled Reactors

Despite the larger potential for high-temperature damage mechanisms compared to SFRs or LWRs, the majority of significant OpE at HTGRs appears to be related to the contamination of the primary loop by either moisture, lubricants, or excessive graphite dust or the performance of seals and compressors. In general, design flaws in HTGRs unrelated to fabrication led to conditions resulting in material and component failures. The materials used in HTGRs include SA-508/SA-533 steel for lower temperature RPVs and for LWR vessels and modified 9Cr-1Mo (Grade 91) steel for higher temperature RPVs. Alloy 800 has been chosen for some HTGR SGs. McDowell et al. [11] discusses materials used in HTGRs.

Materials performance requirements for HTGRs are markedly different from those for LWRs as follows:

- Creep resistance is critically important because of the higher operating temperatures (700–950 degrees C) (1,292–1,752 degrees F) of HTGRs. The requirement for creep resistance of the RPV material is design specific because HTGR designs typically cool the RPV with helium from the cold leg.
- HTGRs have a high-temperature strength that allows them to resist buckling during low heating flow or loss-of-forced convection conditions.
- HTGRs are resistant to high-temperature decarburization (primary loop).
- HTGRs are resistant to corrosion from high-temperature water and steam (steam loop).

HTGR graphite has a number of separate requirements, including isotropic or near isotropic properties and a high degree of resistance to irradiation damage. The performance requirements of structural materials, including graphite, are principally dictated by HTGR operating temperatures, which in some cases have been markedly higher than temperatures predicted by design analyses.

This report examined the information obtained (e.g., reports, conversations) to identify the important OpE issues of concern for materials and structures in ANLWRs summarized in the OpE list for HTGRs below.

2.5 Relation to Advanced Non-Light-Water Reactor Licensing

The results in this report will identify the OpE expected in HTGRs and SFRs and corresponding damage development mechanisms. However, some OpE may not apply to new ANLWRs. This report will also identify additional issues for consideration by the NRC to the maximum extent possible. The results may specifically help identify necessary changes to the regulatory

licensing framework supporting ANLWR licensing (e.g., changes to Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” or changes to the NRC computational codes, such as the Extremely Low Probability of Rupture (xLPR) code).

3 SODIUM-COOLED FAST REACTORS

This section summarizes the OpE for SFRs by component and includes important lessons learned in regard to material performance and structural integrity. For reference, the OpE examples are numbered for each component below.

Table 1 lists international SFR OpE compiled by the IAEA [12] which discusses the evolution of these reactors. Although OpE for all of these plants could not be obtained, this report discusses OpE as much as possible below.

Table 1 Sodium Fast Reactors [12]

Reactor	Country	Power (MW _t)	Power (MW _e)	Criticality (yr)	Shut Down (yr)	Primary Hot Leg (°C)	Primary Cold Leg (°C)
EBR-II	United States	62.5	20	1961	1994	473	371
FFTR	United States	400	0	1980	1992	503	360
PFR	United Kingdom	650	250	1974	1994	560	399
BN-350	Russia	1,000	130	1972	1999	430	280
BN-600	Russia	1,470	600	1980	N/A	535	365
BN-800	Russia	2,100	870	2014	N/A	547	354
BOR-60	Russia	55	12	1968	N/A	530	330
BR-10	Russia	8	0	1958	2003	470	350
Joyo	Japan	50–140	0	1977	N/A	465–500	350–370
Monju	Japan	714	280	1995	2010	529	397
FBTR	India	40	12	1985	N/A	530	380
Phénix	France	563	250	1973	2009	560	395
Rapsodie	France	40	0	1967	1983	515	400
Superphénix	France	3,000	1,200	1985	1998	545	395

Sodium supports a fast neutron spectrum because it has low neutron moderation and absorption. The large margin to boiling of sodium (about 700 degrees C) (1,292 degrees F) allows for efficient operation at atmospheric pressure [7]. Sodium is also chemically compatible with structural materials, which minimizes corrosion in plant cooling systems. However, an inert atmosphere covering the sodium is needed because sodium is reactive with air and water. Molten sodium will burn if it is exposed to air or water; therefore, special fire-suppression systems are an important part of the design. The reactivity of sodium led to many of the issues described below.

3.1 Fuel Structure and Cladding, Subassemblies, and Core Components

- (1) (Phénix, 1979): Cladding failure in a fuel pin [10] resulted in the largest fission gas (xenon-135) release at the Phénix plant. Cracking of the Type 316 SS cladding in a faulty fuel subassembly caused the release of the fission gas. Initial fabrication defects during construction must be identified.
- (2) (Phénix, early operation): Sodium- and irradiation-induced swelling in heat-treated Type 316 SS was more pronounced than expected. During the first years of Phénix's operation, the behavior of the pin bundle with spacing wires under irradiation was a major problem that limited the lifetime of the plant's subassemblies as damage developed from fretting caused by thermal expansion. The most significant deformations included cladding swelling, helical twisting of the pins, and localized bulging of the hexagonal tube surface at the wire pitch by mechanical interaction with the bundles. To remedy this situation, the hexagonal guide sheaths were made slightly shorter 5 millimeters (mm) (0.2 in.) to better manage inservice distortion induced by swelling of the fuel pins [10]. The SS used in the hexagonal fuel wrapper was replaced with titanium-stabilized and -hardened SS, which greatly reduced swelling (the exact grade was not specified). Irradiation-induced embrittlement, swelling, and distortion must be accounted for properly.
- (3) (Phénix, early operation): A serious fuel pin failure during dismantling of the hexagonal wrapper occurred because of irradiation-induced ductility reduction in the Type 316 SS hexagonal wrapper. The dismantling process involved sawing and tearing off metal from two sides [10], leading to a rupture. Modifications were made to the cutting and milling process to avoid such ruptures in future dismantling operations. The proper choice of materials for SFRs requires complete material databases, which should account for irradiation embrittlement, swelling, and distortion.
- (4) (Phénix, general): Fifteen out of more than 140,000 fuel pins leaked over the lifetime of the Phénix reactor. In case of cladding failure, a slow reaction proceeds between the oxide fuel and sodium, forming a lower density sodium oxide compound. This induced swelling leads to cracking and failure of the cladding. Therefore, it is very important to detect the failure as soon as possible and to unload the defective fuel assembly. Spectroscopic and chromatographic monitoring of the primary sodium cover gas was the basis for the Phénix clad failure detection and localization system. Cladding failures resulted in shutdowns of 3 days, which is the time required to replace the defective fuel subassembly [10]. In most cases of cladding failure, the reactor was stopped before it reached an automatic trip threshold. Reliable sensors for detecting fission gas release are necessary.
- (5) (BN-350, general): The primary issue for fuel assembly design in Russian reactors was the swelling of steel under high neutron fluences. Radiation exposure of SS OH18N10T (similar to AISI Type 321 SS) used in the fuel rod structure resulted in differential swelling, high local stresses, and cracking. Irradiation testing on structural materials during the BN-350 project development was minimal which led, in part, to problems with the fuel assemblies. An extensive program had to be developed, and multiple types of steels were investigated to determine the effects of irradiation swelling, embrittlement, and creep characteristics to select the optimum steels for fuel element cladding, 4C-68 (the Russian equivalent to cold-worked titanium stabilized Type 316 SS), and ferritic-martensitic steel EP-450. The design of the BN-350 was also modified to account for

this issue [13]. Proper selection of materials for SFRs requires complete material databases, which should account for irradiation embrittlement, swelling, and distortion.

- (6) (BN-600, 1983–1987): Because of multiple instances of fuel pin failures, the Russian BN-600 reactor was shut down six times for unplanned refuelling [14,15]. The investigations of the failed fuel revealed stress-induced corrosion of the annealed (Russian-grade) austenitic steel claddings as one of the main causes of their early failures. The claddings were damaged mainly in the peripheral region of the core because the periphery had the most unfavourable operating conditions (highest fluence) in the core. Because the fuel pins were reshuffled and rotated during operation, their linear heat rating and cladding temperature rose up to 54 kilowatts per minute and 710 degrees C (1,310 degrees F), respectively, at the end of the fuel cycle (favourable operating conditions were not specified). An improved (unspecified) cladding alloy eliminated the problems. The proper choice of materials for SFRs requires complete material databases, which should account for irradiation embrittlement, swelling, and distortion.
- (7) (Fast Flux Test Facility (FFTF), early operation): Although the FFTF fuel performed extremely well (low outage time), one driver fuel pin and several test pins experienced cladding breaches during the more than 10 years of reactor operation [16]. These breaches released relatively small amounts of radioactive cesium into the primary sodium and cover gas systems, which led to operational complications. A cesium trap was installed in the FFTF primary sodium processing system in 1987, following nearly 7 years of operation that included several fuel pin failure events. The cesium trap mitigated contamination of the primary loop. SFR designs should account for fuel failure.
- (8) (FFTF, general): Reflector assemblies were made of Alloy 600 and experienced swelling behaviour caused by the neutron flux in the FFTF [17]. This was a poor choice of material, resulting from a design that used insufficient test data to describe the swelling response of Alloy 600. Proper choice of materials for SFRs requires complete materials databases, which should account for irradiation embrittlement, swelling and distortion.
- (9) (Fast Breeder Test Reactor (FBTR), May 1987): India's FBTR experienced a problem as a fuel subassembly was being transferred from the core to the periphery [18]. The improper design of the fuel transport system caused extensive bending of the subassembly and the heads of the reflector subassemblies during transport. Details were not specified; however, the transport system was improved, and it was noted that this was an expensive solution.
- (10) (Prototype Fast Reactor (PFR), general): Neutron-induced distortion of core components in the United Kingdom's PFR and its effect on plant operation is a similar issue to some of the problems discussed above. Radiation damage resulting from the high neutron fluxes and operating temperatures of a fast reactor can give rise to dimensional changes in core components. The mechanisms involved are swelling caused by neutron-induced voids (NIV) and radiation creep. Large differences in NIV swelling rates occurred in different batches of the same material because of a quality control problem that was not adequately addressed and thus led to handling problems of components made of cold-worked EN58B. The materials chosen later in the lifetime of the PFR, such as Nimonic Alloy PE 16, had considerably lower swelling rates.

Components manufactured from the ferritic steel FV448, which was undergoing testing at the time of the PFR's closure, had extremely low swelling rates. NIV distortion was not expected to be life limiting for this material [19]. The proper choice of materials for SFRs requires complete material databases, which should account for irradiation embrittlement, swelling, and distortion.

- (11) (BN-600, 1995): The central rotating column (CRC) in BN-600 is used for refuelling the core. In 1995, an increase in force required to turn the CRC was observed. The CRC rotates during fuel assembly and fuel reloading operations. During examination of the problem, sodium found in the bearing assembly was impeding the CRC turning motion. This phenomenon was caused by the transfer of sodium vapor from the reactor cover gas (argon) through the gap between the CRC and rotation plug and the subsequent accumulation of sodium and its compositions in the bearing assembly. Designs should consider the potential for sodium vapor penetration and condensation.
- (12) (FFTF, general): The FFTF outer row assemblies consisted of a stack of Inconel-600 blocks penetrated by SS coolant tubes. These assemblies acted as a radial neutron reflector and as a straight, but flexible, core boundary. During design, these assemblies were assumed to exhibit low-swelling behaviour in a neutron flux based on a collection of high nickel-alloy data available at the inception of FFTF. However, during an FFTF refuelling outage, the degree of difficulty withdrawing an outer row driver fuel assembly was a function of the peak fast fluency of neighbouring reflector assemblies. Post-irradiation examinations showed that the reflector assemblies were both bowed and stiff. Differential swelling in a steep radial flux gradient [16, 17] had distorted the Inconel 600 blocks into a trapezoidal cross-section. SFR designers must calculate fluxes accurately to minimize irradiation damage.

3.2 Reactor and Pumps

- (1) (Phénix, 1976): The primary coolant pump in Phénix experienced excessive vibrations [10, page 39]. As a result, operators decreased the pump speed until the pump could be extracted several months later. During inspection of the pump, a hydrostatic bearing that was shrink-fit to the pump shaft during fabrication expanded during a severe thermal transient (automatic shutdown) and slipped down the pump shaft, damaging the shaft and also causing excessive vibrations. The Superphénix design eliminated shrink-fit fabrications to avoid this possible problem. Shrink-fit parts should be avoided in pumps that could loosen during thermal transients.
- (2) (Phénix, 1974–1975): Three separate secondary coolant leaks were traced to joining welds on large-diameter butterfly valves upstream from the SG. These leaks led to small fires in the insulation as a result of sodium reacting with air. Weld repairs were found to be ineffective, but replacement of the valves with diaphragms solved the problem. Possible valve failures remain a concern in all SFR designs [10].
- (3) (Superphénix, early operation): During Superphénix startup trials [9, Chapter 3], excessive vibration of the reactor internal structures was observed as soon as the primary pumps were started. This was a problem of flow-induced self-excitation caused by the cooling rate that was of the same order of excitation as the flow rate pulsations. The vibration could have led to structural damage of the internals. Increasing the vessel cooling flow rate by 50 percent solved the problem. This OpE illustrates the importance

of managing the startup and cooldown transients in SFRs to control vibration and possible fatigue issues in the components.

- (4) (Superphénix, general): The primary pumps in Superphénix were immersed in sodium and designed based on the Phénix experience. The primary pumps operated without incident the entire working life of Superphénix. Each pump was heavily instrumented with three vibration sensors, three shaft rotation speed sensors, multiple flow rate sensors, and 32 thermocouples [9]. These sensors ensured that the pump operated properly and that the bearings operated properly to prevent cavitation. The main parts of the pumps were made from Type 316 SS and the castings (impeller, lantern, and diffuser) were made from Z3 CN20-10 with no molybdenum (American Society for Testing and Materials (ASTM) equivalent CF-3M) because OpE with Phénix had demonstrated that the presence of molybdenum increased the risk of spinodal decomposition. All materials performed as designed. In addition, based on the Phénix operation, the pumps had no shrink-fit parts that could loosen during thermal transients. As discussed in OpE No. 5 below, a coupling failed in the primary pump; however, this was not considered a pump failure. This OpE illustrates the usefulness of building upon lessons learned from a prototype reactor.
- (5) (Superphénix, September 1993): A sodium coolant coupling in a primary pump broke [9]. The poor alignment of this coupling led to a significant wear at the failure site. All couplings were replaced and properly aligned. No pump coupling failures were observed afterward.
- (6) (BN-600, 1981–1985): The unstable operation of pump speed control systems caused unplanned power losses between 1981 and 1985 [15, 24]; this problem also led to additional fatigue cycles. Impacts from the pump electric motor on the pump housing caused failures in the pump electric motor coupling (leading to excessive distortions) from increased vibration and fatigue cracks in pump shafts. Using a better shaft and coupling and operating the pumps in a steady mode after attaining the desired level of reactor power prevented any failures of the reactor coolant pumps since 1985. Thermal fatigue or periodic operation outside of normal design parameters may greatly reduce the lifetime of reactor components.
- (7) (FFTF, early operation): Sodium systems operated for over 20 years, including more than 10 years of reactor power operation. During this time, only one sodium leak (approximately 284 liters (75 gallons (gal)) of primary sodium) occurred during a refueling outage from a small electromagnetic pump. A combination of freeze and thaw cycles caused a leak that led to localized deformation of the electromagnetic pump duct wall and to subsequent operation under cavitation conditions. This process eroded the pump duct wall at one of the deformations and caused it to eventually fail. Plant operating procedures were subsequently revised to tightly control pump freeze and thaw cycles and to prevent pump cavitation [17]. Thermal fatigue or periodic operation outside of normal design parameters may greatly reduce the lifetime of reactor components
- (8) (Superphénix, date not provided): A blade broke on the impeller in a secondary pump. The secondary pumps circulated sodium in the secondary heat exchangers and resided in expansion tanks. The failure did not lead to any significant issue. The cause of the failure is not publicly documented [9].

- (9) (Superphénix, July 1990): Air leaked into the sodium coolant [9, 20] from a compressor. It took the operators approximately 1 month to identify and shut down the plant for repair and subsequent purification of the sodium. An argon branch connection leading to an argon-activity measurement chamber was fitted with a small compressor (with a diameter of a few centimeters (cm)). This branch was designed to send argon to the measurement chamber. The membrane between the air side and argon side tore, which led to the sodium contamination. The lengthy delay in identifying this issue was traced to the designers' choice not to measure the chemical composition of the core cover gas. The designers believed air ingress was not plausible because the core cover argon gas was always relatively overpressurized towards the ambient air and the overpressure was being continuously monitored. As a result, contamination of the sodium went unnoticed for months [9]. The membrane material and the reason for the tear were not described [9, 17]. This example highlights the value of confirmatory surveillance.

3.3 Sodium Storage Tanks and Drums

- (1) (Superphénix, 1987): A leak occurred in the sodium storage drum (about a 24 cubic meter (848 cubic ft) volume of sodium). Examinations showed both through-wall and partial through-wall cracks were pervasive in the wall-to-base welds. The cracking (shown in Figure 1) was primarily in the heat-affected zone; however, some cracks extended into the weld metal. Hydrogen embrittlement, enhanced by weld residual stresses,¹ created the cracks.



Figure 1 Storage drum cracking in Superphénix (from [9, Figure 17.9])

The storage drums were fabricated from 15D3, which is a low alloy ferritic steel consisting of iron (98–99 percent), manganese (0.5–0.8 percent), molybdenum (0.25–0.35 percent), silicon (0.15–0.3 percent), and chromium (0.0–0.3 percent). Similar cracking was found with 15D3 during the construction of the SNR-300² reactor in Kalkar, Germany [9, Chapter 17]. This material was subsequently found unsuitable for SFRs because of hydrogen embrittlement issues. Improper material selection in combination with inadequate welding procedures was the cause of the hydrogen cracking.

¹ This is a particular problem in refinery operations.

² The SNR-300 reactor was a German SFR that was completed in 1985 but never operated.

3.4 Secondary (Intermediate) Heat Exchanger

- (1) (Phénix, 1976): The intermediate heat exchanger (IHX) experienced two separate sodium leak events [10, page 41] followed by fires. The events were caused by cracking in the welds joining the weld on the metal plate that closed off the sodium outlet nozzle in the secondary system above the heat exchanger. The metal plate connected two long shells that operated at different temperatures, which led to thermal expansion stresses that subsequently caused the cracks. The solution was to replace the plate with a more flexible design to accommodate the thermal mismatch. In addition, a flow-mixing device was added to the sodium header to reduce the temperature differences between the shells. Thermal mismatch loads and mixing of different sodium flows must be properly considered.
- (2) (Phénix, June 1985): In June 1985, a small sodium leak occurred in the secondary heat exchanger pipe circuit at a thermocouple thimble weld [10]. The source of the leak was a through-wall fatigue crack driven by vibration excitations of the thermocouple caused by sodium flow. A similar issue led to a serious fire at the Monju Nuclear Power Plant (Monju) in Japan (see OpE No. (4) below). Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure and sodium leaks.
- (3) (Phénix, April 1990): A small sodium leak was detected at an auxiliary circuit tee pipe joint [10, Chapter 14]. An operator failed to completely close a valve. This error led to residual sodium flows, which created thermal fluctuations (thermal striping) that resulted in fatigue cracking. This is another example of a damage mechanism in SFRs driven by temperature gradients caused by sodium flow.
- (4) (Monju, December 1995): After a scheduled shutdown and the following plant startup [21], a sodium leak occurred in the secondary heat transport system of pipe loop C (a hot-leg geometry) and led to a serious fire when it interacted with oxygen and moisture. The leak melted steel structures in the room and resulted in a solidified sodium residue on the room floor almost 3 meter (m) (9 feet (ft)) in diameter and approximately 30 cm (1 ft) deep. The leak was located at a 3 mm (0.12 inch (in.)) diameter thermocouple penetration (Figure 2). The failure of the thermocouple well tube released the well tip and permitted sodium to flow through the thermocouple and into the room. The well tube failure was caused by high-cycle fatigue from flow-induced vibration caused by vortex shedding in the direction of the sodium flow. Figure 2 shows the microscopic investigation of the fracture surface, which indicates fatigue striations. The designers had inappropriately applied the ASME Performance Test Code³ to the Monju tube thermocouple weld because the thermocouple was used in a location where vortex-induced vibrations can occur. Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure and sodium leaks.

³ The ASME Performance Test Code was redone in 2010. The code establishes the practical design considerations for thermocouple installations in power and process piping.

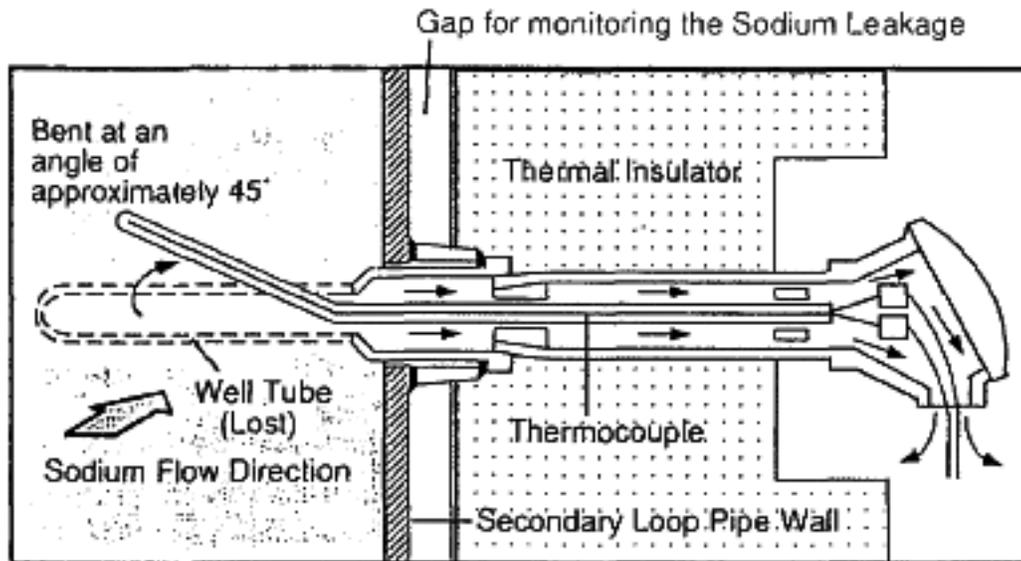


Figure 2 Bent thermocouple and missing well tube in Monju (from [21, Figure 5])

- (5) (Superphénix, December 1994): The reactor shut down because of an argon pressure seal leak [9]. Eddy current measurements identified a 30 mm (1.2 in.) long crack (half the tube circumference) at the weld between the argon supply pipe and the seal nozzle. The manufacturing defect led to unanticipated additional bending stresses at the weld, which caused inservice crack growth, likely from fatigue. Inspections of other pressure tube seals did not reveal a problem that would indicate a generic fabrication defect. A sleeve was inserted for repair. Initial fabrication defects during construction must be identified.
- (6) (BN-600, May 6, 1994): During the 15 year operation of the BN-600 reactor operation, 30 sodium leaks (2 large and 28 small) occurred. The largest leakage of secondary sodium occurred from an IHX drainpipe. The incident took place during the replacement of an isolation valve while the reactor was shut down. A fire accompanied the sodium leak, and the leak caused equipment damage in the adjacent area. Approximately 1.2–1.3 cubic meters (about 1,000 kilograms (kg) 2200 pounds (lb)) of sodium leaked, but only several tens of kilograms of sodium ignited. The remaining sodium was retained in the smothering catch pan system and covered with extinguishing powder. Fabrication defects or poor management of thermal stresses leading to cracking at the welds caused these issues [22].
- (7) (Experimental Breeder Reactor (EBR)-II, 1968): A major sodium leak occurred in the secondary sodium system [23] in EBR-II. About 379 liters (100 gal) of hot sodium spilled onto the floor in the secondary sodium control room where sodium was purified. This was apparently a maintenance repair issue. A section of pipe had been removed as part of repairs on a bellows-seal isolation valve. The sodium in the line was frozen and, therefore, created a plug for the repair. However, the frozen sodium plug did not extend far enough beyond the repaired section, and molten sodium spilled onto the floor. A major fire erupted but was extinguished by a salt mixture (Metalex) that starved the fire of oxygen. The cleanup took 13 days. The piping material was Type 304 SS [23]. Repair procedures in SFRs must be carefully planned.

- (8) (EBR-II, November 14, 1970): A loud banging noise was heard near the IHX [23]. The noise source was within the Type 304 SS IHX inlet pipe. Visual examinations using both a periscope and a remote television system revealed that two support clips were damaged. One of the two support clips holding a 25.4 mm (1 in.) diameter evacuation tube in place was broken, and the top clip was loose. The missing clip supports allowed the evacuation tube to vibrate against the wall of the 324 mm (12 in.) outer diameter (OD) inlet pipe because of sodium flow. Both the 324 mm (12 in.) OD pipe and the 25.4 mm (1 in.) tube showed evidence of wear. Repairs were made to permit operation. Although the incident report did not specifically identify the cause [23], it was likely the result of flow-induced vibration and thermal mismatch loading.
- (9) (Phénix, general): Phénix experienced 11 sodium leak events during its 35 years of operation. These problems were solved using various design modifications. An example of the only significant issue is the sodium leak into the inner space at the secondary sodium outlet header [10, 20, 24]. Different thermal expansion loads between the inner and outer shells that were underestimated in the original design caused the leak. Figure 3 illustrates the problem and solution; the redesign improved both the mixing of the secondary sodium outlet from the tube bundle (i.e., improving accommodation for thermal expansion) and the flexibility of the IHX hot header between the top closure plate and the inner shell (i.e., the upper right portion in Figure 3). Control of structural constraints in SFRs in regions of thermal expansion mismatch is an important design feature and must be considered during the license assessment of the plants.

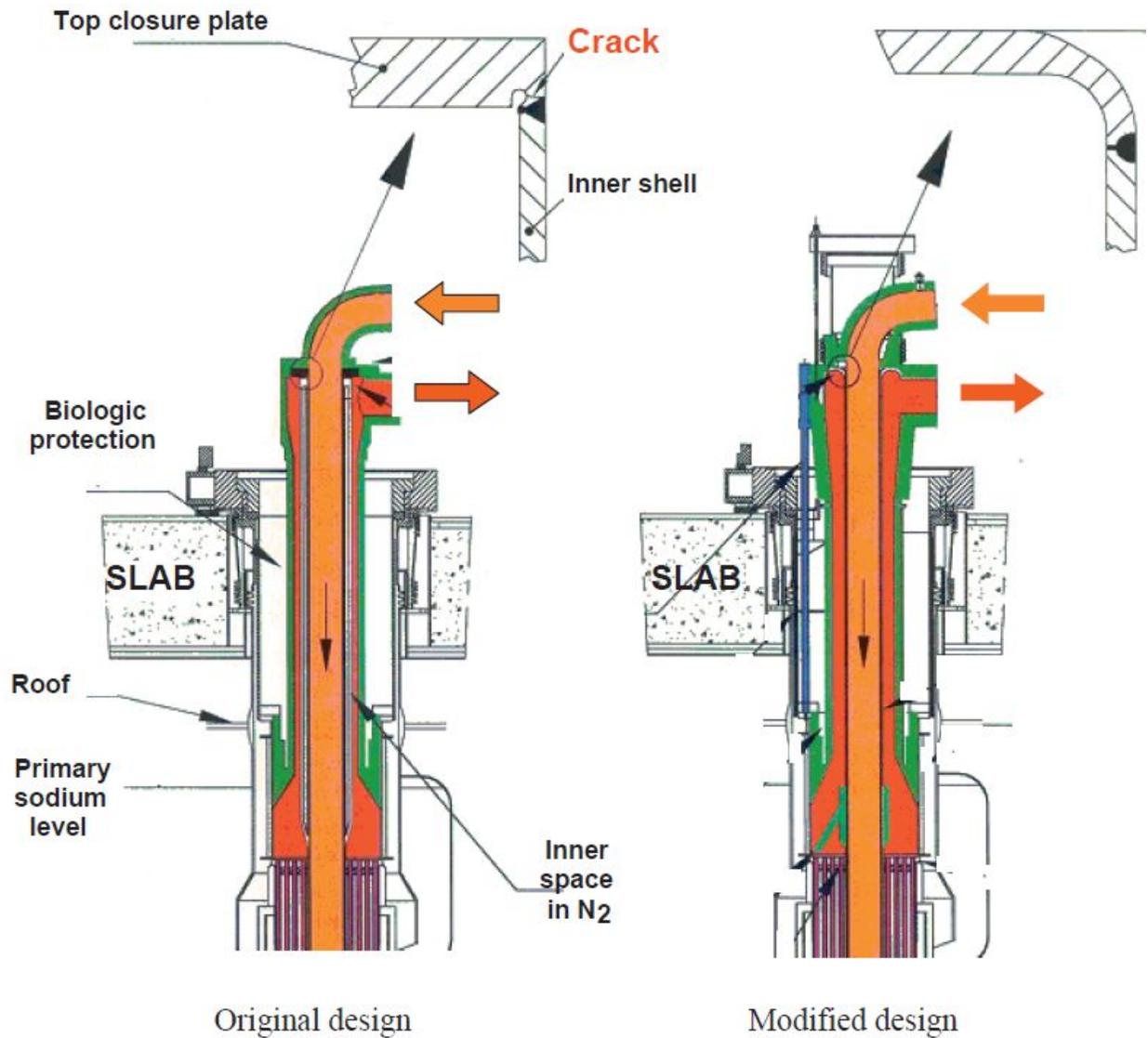


Figure 3 Modified design of secondary sodium outlet header (from [24, Figure 2.26])

- (10) (Phénix, November 1998, December 2000): Sodium leaks were found in the IHX of Phénix. The leaking tubes were near the upper tubesheet (see Figure 4). A metallurgical analysis revealed that caustic stress corrosion from the presence of polluted sodium (sodium hydroxide from air ingress) caused the cracking. The sodium was contaminated during testing of the IHX (i.e., air ingress during work on the secondary loop between 1995 and 1997). Proper procedures during repairs must be carefully maintained.

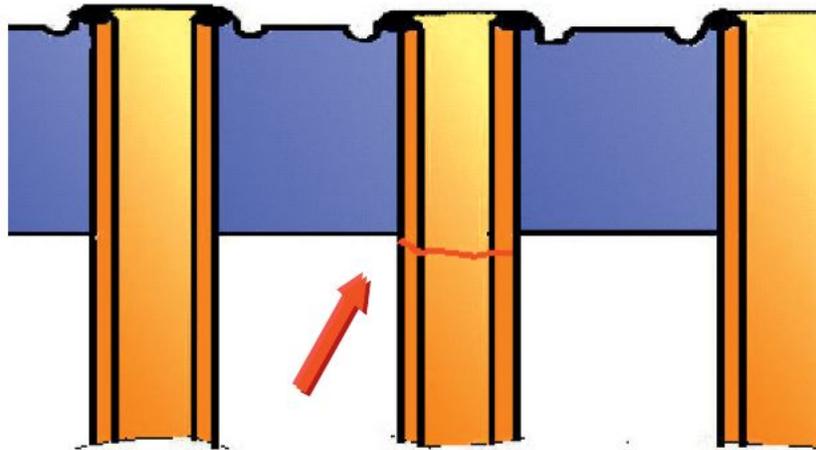


Figure 4 Tube leaks in Phénix (from [24, Figure 2.27])

3.5 Sodium Piping Circuits

- (1) (FFTF, general): Overall, the main FFTF sodium systems performed with no major problems during the nearly 20 years they were in operation [16]. One exception was the occurrence of periodic flow and pressure oscillations in the secondary main heat transport system loops. An extensive investigation showed that the formation of periodic sodium flow vortexes developing at piping tees near the inlets of the heat exchangers caused the oscillations. After evaluating several potential concerns associated with these oscillations and investigating the means of eliminating them, the oscillations were deemed acceptable after changes were made to some plant control systems and operating procedures. The specific operational modifications were not specified. Sodium flows must be carefully assessed to prevent fatigue problems in SFRs.
- (2) (PFR, general): Operators discovered that oil from oil bearings had leaked into the primary loop. Following the oil ingress, the oil bearings were replaced with gas bearings [19]. Oil from pumps, bearings, and other places that leaks into the primary circuits of an SFR can produce methane gas through the core sodium coolant, causing reactivity effects and possible blockage of the subassemblies with solid carbon debris. In the case of the PFR, no reactivity effects were seen possibly because the oil was retained in the pump cone for a prolonged period and slowly broke down without the formation of large bubbles. Oil bearings should be avoided in SFR designs.
- (3) (BOR-5 and BOR-10, general): Over the operational lifetime of Russia's BOR-5 and BOR-10, a combined total of 19 sodium leaks occurred in the sodium equipment and pipelines in the reactors [24]. A combination of inservice cracking and manufacturing fabrication defects, many of which were unspecified, caused the sodium leaks [24]. As with cracking in other operating SFRs, the valve leaks were probably caused by seal problems and improper seating of contacting parts. Pipe leaks not caused by manufacturing defects were likely caused by thermal fatigue and thermal expansion mismatch issues in the SS materials (i.e., X18H10T Russian designation, Type 316L equivalent). Reference [24] states that most of the sodium leaks actually happened in the early stages of reactor operation, which began in 1958, while the plant was mastering sodium technology and developing various equipment design solutions. No

sodium leaks occurred after 1986, and the plant was retired in 2002. No significant corrosion of materials in the sodium circuits occurred during the 40 years of operation. The earlier operation of these reactors demonstrates that fabrication defects during construction must be identified, especially with welds.

- (4) (PFR, 1987–1990): Another problem with the SG in the second decade of PFR operation was the deterioration of welds in the SS outer vessels of the superheaters and reheaters [19]. Leaks in reheater vessels in 1987 and 1988 revealed cracks (in one case, more than 100 mm (3.9 in.) long) in the original interplate welds. Subsequent inspections of the other vessels revealed large, nonpenetrating cracks similarly located in two of the superheater vessels. All cracks were in welds that were reworked during fabrication, or they occurred in areas where fabrication welds had overlapped. It was decided to dump the sodium, cut out the defects, and repair the vessels. However, one of the nonpenetrating cracks in the Superheater 3 vessel was not cut out. Later, an in situ assessment indicated a low likelihood of rapid propagation. Strain gauges were fitted to the crack region as monitors. Two repair methods were used. In the earlier method, the excised region was filled with weld metal against a backing plate. One of these repairs caused a further leak in 1990. In the later repair technique, a circular stub surrounding the defect area was welded on, the defect was removed, and the vessel was resealed by welding a cap on to the stub end. In subsequent years, this method evolved into a “stood off” patch, whereby the cracks were left in situ with holes drilled at both ends to stop them from propagating any further. An investigation of the problem indicated that delayed reheat cracking caused by the relaxation of weld residual stresses initiated the cracks. A high-temperature brittle intergranular mechanism driven by the residual stress field caused the cracks to subsequently grow. Repair procedures in SFRs must be carefully planned because weld residual stresses in repair welds are often highly tensile.

3.6 Steam Generator

SG failures are a particular concern for SFRs because of the potential reaction between molten sodium and water. Materials must also meet the demanding conditions of the steam generator environment. The review of the Phénix reactor [10] also describes general concerns of SFR SGs.

- (1) (Phénix, 1974, 1975): Four water leaks were found in the economizer/evaporator inlet of the SGs [10, page 46]. The leaks were caused by wear of the subheader underframes from vibrations produced by high-pressure water flow in the orifice plates in the evaporators. As a result, the plate geometry was modified to improve flow characteristics of the orifice plates. Parts in contact must be prevented from wearing out in SFR SGs (tribology management).
- (2) (Phénix, 1982): A check valve that failed in the SG nitrogen filling system resulted in sodium slowly entering the water-steam region through a leak. The sodium reached a reheater isolation valve, creating two holes from corrosion (total area 2 square centimeters) in two SG tubes. Repairs were made, and the plant was restarted [10]. An accurate estimate of component reliability minimizes the chance of a sodium leak.
- (3) (Phénix, 1983–1984): Four separate sodium/water reactions took place in the SG reheater tubes. The leaks and subsequent sodium/water reactions were the result of cracking at tube-butt welds. The butt welds had excess metal leading to stress concentrations at the junction of the weld to the tube. Thermal transients caused these

cracks during startup operations as water mixed with steam, which caused thermal shock and fatigue. Consequently, the design was modified to reduce these transients, and excess butt weld material was removed [10]. Excess weld material in SFRs can lead to over constraint, increasing the likelihood of cracking under thermal transients.

- (4) (Phénix, general): Every 10,000 hours of operation, one of the burst discs operating near the hottest SG regions was removed, tested to ensure its burst rating, and replaced. Burst discs on the SGs are carefully monitored to prevent sodium/water reactions and excessive operating pressures in Phénix [10]. This was a precaution to prevent sodium/water fires. An accurate estimate of component reliability minimizes the chance of a sodium leak.
- (5) (BN-350, 1973): A major sodium fire occurred because of poor weld quality control of the SG tubing. The next Russian reactor (BN-600) was designed with the SGs in separate compartments to contain sodium/water fires. Additionally, the BN-600 was designed with an extra SG to allow the repair of fire-damaged SGs while the reactor continued to operate [14]. Initial fabrication defects during construction must be identified, especially with welds. Vendors should anticipate SG operational issues during design.
- (6) (BN-350, general): The BN-350 had six loops and six SGs, each of which had two vertical evaporators with field-type heat exchange tubes [25]. During testing, leakage was detected at welds where the tubes had been welded to the tube sheets. The materials were not identified; however, References [24, 25] describe the tube material as 2.25Cr-1Mo steel. Metallographic examination showed the presence of microcracks in the tube weld joints caused by cold stamping. The SGs were repaired in situ; all the tubes were replaced, and all the welds were thoroughly examined. In 1974 and early 1975, another incident occurred, the first three SGs were repaired, and the reactor operated at 350 megawatts thermal (MW_t). After 9 days of operation, a large interloop leak was detected in SG No. 5. The reactor was shut down, and the valves in the lines of feedwater were closed. The drainage of water from the damaged evaporator was started. However, operators failed to drain the sodium properly because the drainage pipeline got plugged with reaction products. This apparently led to a violent sodium/water reaction. The majority of the evaporator heat-transfer tubes were corroded by the resulting sodium hydroxide solution; the evaporator housing was also partially damaged. About 300 kg (660 lb) of sodium went into the SG room. Proper procedures and quality control (particularly for welds) and repairs are important.
- (7) (BN-600, early operation): Leaks in the sodium interloops [15, 24] occurred during the first years of plant operation in the SG superheaters. Each interloop consisted of eight sections, including an evaporator module, a superheater module, and a reheater module. The evaporator modules were made of ferritic-martensitic steel (1X2M) and superheater modules of Russian SS (OH18N10T, Type 316L equivalent). All cases of interloop leaks occurred at the point of tube-to-SG tube plate welds—a crack problem location found in many SFR designs. Deficiencies in the welding procedure and subsequent weld control also caused cracking. After design improvements, only one interloop leak was observed after 1985. Before these improvements, in the first three cases of interloop leaks, a fairly large amount of water went into the secondary circuit and a considerable quantity of sodium entered the tertiary circuit. Moreover, the first case showed hydrogen leakage through the packing bearings of the shutoff valves. Welds should be designed to minimize stresses during operation. Additionally weld

fabrication and inspection are important. Accurate estimates of valve reliability minimize sodium leaks.

- (8) (BN-600): As of 1997, the BN-600 Russian SFR had 27 sodium leaks, with the largest leak being 1,000 liters (264 gal) [15, 24]. Fourteen of the leaks resulted in fires. The causes of the leaks, their locations, or the materials used were not summarized; however, item 7 above includes some of them.
- (9) (Phénix, 1986): During a scheduled outage, a long-term sodium leak was discovered beneath insulation at an SG's heater inlet T-part junction [10]. The leak occurred through a crack under the insulation that could not be easily detected (i.e., there was no smoke or other visible signs). Flow assisted corrosion removed 7 mm (0.28 in.) of the pipe's thickness (the pipe is nominally 19 mm (0.75 in.) thick in this region). The piping insulation and molten sodium mixed to form a corrosive amalgam. This incident highlighted the need for better detection methods of corrosion and leaks, particularly in regions coated with insulation. It also highlighted the need to test chemical compatibility between molten sodium and insulation materials.
- (10) (PFR, 1974–1984): A total of 37 gas-space leaks occurred in PFR SG units, with 33 of these leaks in evaporators, 3 leaks in superheaters, and 1 leak in a reheater [19]. All the gas-space leaks originated at the welds between the tubes and the tube-plates. These leaks had a considerable effect on PFR availability; the highest annual load factor before 1984 was only 12 percent. Gas-space leaks proved to be readily detectable by means of the hydrogen they generated. Careful washing of the tube-plates with hot sodium limited the number of leaks and avoided further damage, but it did not resolve the problem. The solution involved reinforcing all the tube-to-tube-plate welds in the three evaporators. Future SFRs should avoid the type of direct tube-to-tube-plate welds adopted initially at the PFR, which could not be heat treated after manufacture. In addition, austenitic steels are unsuitable for SFR SGs because of the high risk of caustic stress-corrosion damage following even small leaks. Flow-induced vibrations also contributed to some of the cracking at the welds.
- (11) (PFR, February 1987): A failure in Superheater 2 led to a major leakage of steam into the secondary sodium circuit [19] in the PFR. This caused a 28-week shutdown for repairs. After the leak event, an examination of the austenitic SS superheater revealed a fretting failure of a single tube, which had been subjected to unexpected flow-induced vibration. Thirty-nine neighboring tubes failed in the resulting sodium/water reaction event. This event rendered the unit unserviceable. One of the replacement tube bundles, stored on site as a strategic spare, was installed. The tube location suggested the possibility of a similar event occurring in the other original superheaters, and a decision was made to replace all of them. Observations of damage from vibration in the other units after removal warranted tube replacement. The sodium leaks in Superheater 2 demonstrated the possibility for a large number of tubes to fail from overheating in a period of a few seconds. This event was unlikely to cause significant over pressurization damage in the secondary circuit or the IHX, although it was possible. The incident led to a reassessment of the design-basis accident for the PFR SGs. In the case of the PFR, the design-basis accident was changed from a single double-ended guillotine fracture to 40 double-ended guillotine fractures spread over a period of 10 seconds. Vibrations caused by mixing sodium flows of different temperatures must be carefully assessed to prevent fretting.

- (12) (PFR, general): Cracking in PFR SG containment vessels was observed during maintenance activities. The evidence indicated that a delayed reheat mechanism driven by residual stresses in non-stress-relieved welds initiated cracking in PFR SGs. Weld repairs (Figure 5) made during the manufacturing process gave rise to conditions that favored cracking.⁴ Weld residual stresses must be properly managed in systems where creep may occur and where corrosion may take place.

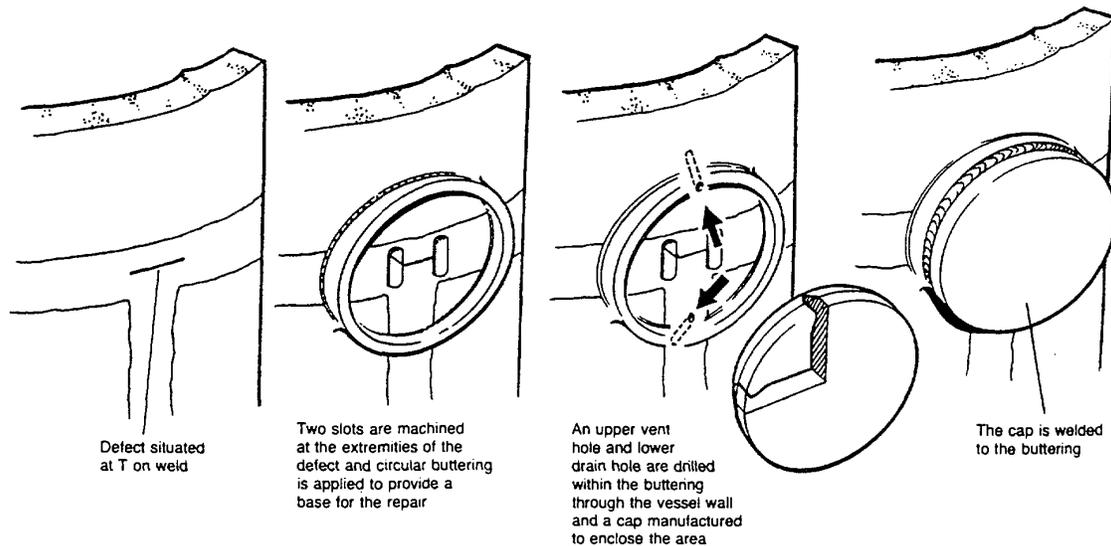


Figure 5 SG vessel repair methods (from [19, Figure 3.2])

- (13) (EBR-II, February 7, 1965): During a shutdown period with the SG system at ambient temperature, the operating crew reported liquid water between the steam and sodium tubesheets at the upper end of an evaporator (Figure 6). The double-tube design was made to minimize the possible interaction of sodium with water and steam. The source of water was traced to a crater crack in one of the tube-to-tubesheet welds (2-1/4Cr-Mo steel). It was a fabrication defect that the original helium leak test had not detected, but it was manually repaired. Access was gained by removing a section of the steam riser from the evaporator. No additional leaks were detected on any of the SGs. After this incident, additional inspections were made seven times from 1969 to 1978. Although there was some discoloration, there was no evidence of corrosion or fatigue cracking at the tube-to-tubesheet welds. Initial fabrication defects during construction must be identified.

⁴ It is well known that repair welds often greatly increase tensile weld residual stresses.

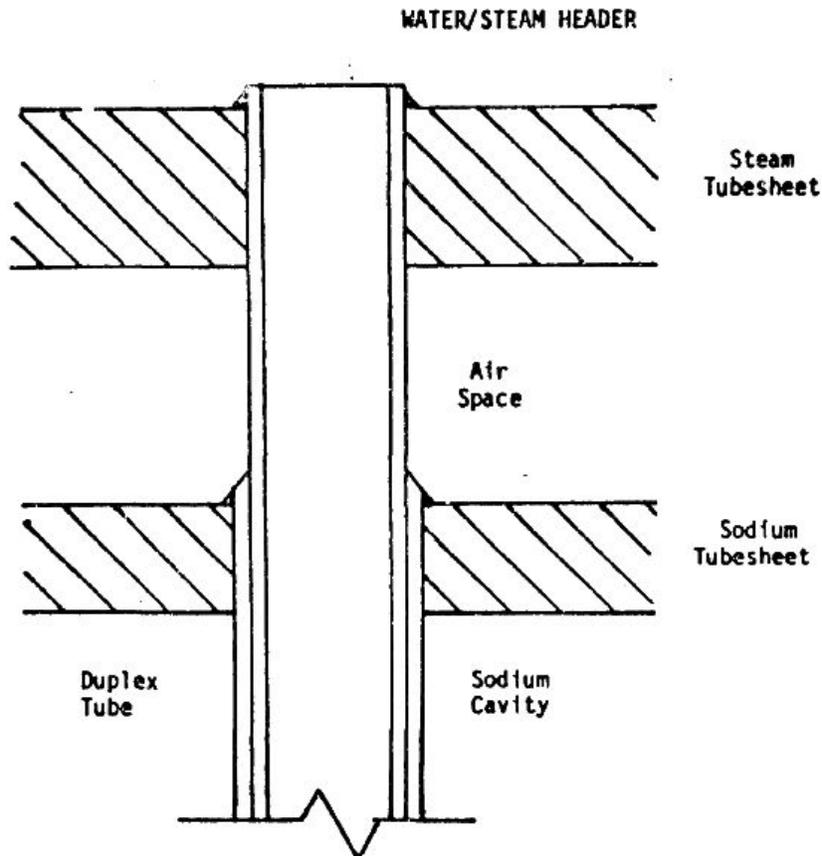


Figure 6 EBR-II steam generator tube cracking (from [23, Figure 7])

3.7 Sodium Purification System Loop

- (1) (FBTR, 2002): After 17 years of operation, a sodium leak in India's FBTR [25, 26] occurred in the purification building where the primary sodium purification piping resides. Publicly available sources do not precisely describe the event; however, it appears to have occurred because of a manufacturing defect in the bellows of sealed sodium service valves. The cleanup was particularly expensive and time consuming because the sodium was radioactive; its removal was a major effort.
- (2) (BN-600, July 1993): A crack on the auxiliary primary sodium purification system pipeline in a 48 mm (1.89 in.) diameter pipe [24] caused a sodium leak. This resulted in an insignificant radioactive discharge to the atmosphere (less than 5 microsieverts (0.5 millirem)) at the plant boundary. The cause of the crack was not identified.
- (3) (PFR, June 1991): Overheating on the top bearing of a primary sodium pump [19] resulted in a manual trip of the PFR. The upper bearing of the pump lost a significant quantity of oil (up to 35 liters (9.25 gal)), which entered the primary sodium circuit. The primary pump filters were also damaged and required replacement. This was another incident of pump bearing failure that allowed oil into the sodium and resulted in a long shutdown. The bearing materials and the precise tribology of the failure were not reported. Oil bearings should be avoided whenever possible in SFRs.

- (4) (FBTR, April 2002): While the reactor was operating at 17.4 MW_t, 75 kg (165 lb) of primary sodium leaked from the purification circuit [26]. The leaked sodium froze on the floor and pipelines. The sodium was manually removed under inert gas purging. The leak came from the body of a bellows-sealed valve through one of the three blind holes used by the manufacturer for machining the valve body (Figure 7). Sodium leaked inside a purification compartment from a hole in the valve body. Because the problem was generic to that specific valve, all valves with that make were inspected and rectified by welding tightly fitting plugs. The leaked sodium converted to sodium hydroxide, which was neutralized by phosphoric acid and disposed of as active liquid effluent. The material thickness of the valve was 0.1 mm (0.004 in.). This thickness was sufficient to hold sodium for 17 years, which indirectly indicates that minimal corrosion occurs between SS and high-purity sodium. In a similar incident, sodium leaked past the failed bellows and went through a crack in the weld joint of the nipple used for mounting a spark plug detector. A faulty valve was later cut and replaced with a new valve (see Figure 7). Proper valve design and accurate estimates of valve reliability will minimize sodium leaks.

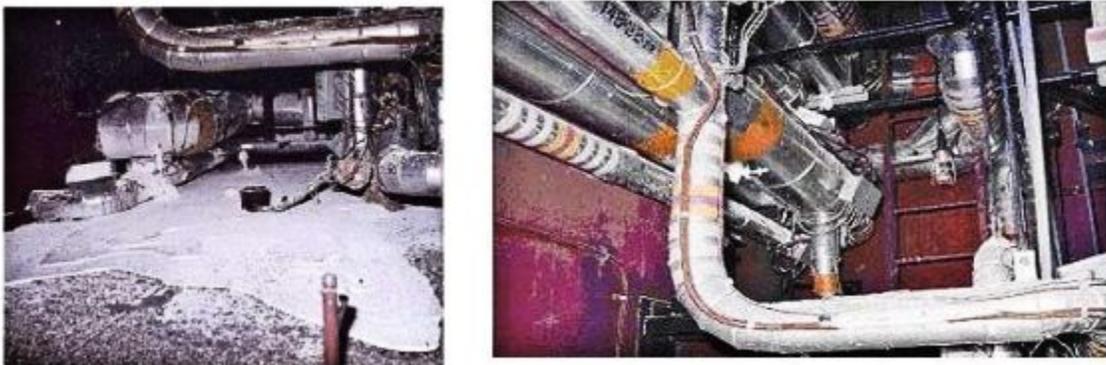


Figure 7 FBTR valve leakage failure (left) and sodium leakage on the floor and piping (right) (from [26, Figure 12])

3.8 Emergency Sodium Cooling Circuits

- (1) (Superphénix, May 1991): A sodium leak occurred at the thimble of the thermocouple that measures the plugging indicator pellet temperature of a loop purification circuit. This design error led to varying temperatures in the area, which caused excessive expansion and distortion followed by cracking in the thimble. Thermal fatigue must be carefully assessed to prevent component failure [9].
- (2) (Superphénix, April 1994): An argon leak with sodium aerosols from a through-wall crack occurred on a secondary circuit gas line connecting the interspace of the rupture disk (discharge membranes for severe sodium/water reactions) to the cold storage tank. Corrosion caused by aqueous sodium hydroxide initiated the crack. The root cause was a poorly chosen gas fitting [9].

3.9 Hydraulics and Thermohydraulics

In nominal operating conditions, SFRs must handle hot sodium (550 degrees C) (1,022 degrees F) (and cold sodium (400 degrees C (752 degrees F) on the primary side and 350 degrees C

(662 degrees F) on the secondary side), which could lead to potential thermohydraulic concerns [9], including the following:

- fluctuation and thermal striping risks in the mixing zones with flows at different temperatures
- complex sodium flow zones and interactions with flow patterns between subassemblies
- thermal stratification of sodium and its consequences on the structures, including the inner vessel
- cold shocks or hot shocks during transient conditions

In general, thermal loads in SFRs play a large role in structural performance and possible cracking, as pressures are low. Chapter 18 of Reference [9] discusses thermohydraulic codes that were developed and specifically verified during the design and operation of Superphénix.

3.10 Severe Accident

No SFR has had a severe accident; therefore, little information was found on this issue. However, studies of BN-350 have been underway since the 1990s to ensure safety under external dynamic loading, such as an earthquake [24]. Investigations were carried out to refine the parameters of the maximum design seismic impact, including building structures, pipelines, heat exchange equipment, and reactor vessels. Analysis of the consequences of seismic impact on the reactor plant building structures, equipment, and pipelines showed existing safety systems, such as the power supply system, the feedwater system, and the service water supply to components important to safety, would be either partially or completely destroyed during a seismic design-basis event. Taking this into account, a design was developed to provide for the arrangement of safety equipment in the seismically robust part of the reactor building to ensure the removal of reactor residual heat under seismic impact conditions. A separate design is currently under development to provide reactor seismic protection by triggering the shutdown system in response to signals from seismic sensors.

In addition, following a seismic safety upgrade study of the Phénix reactor in the late 1990s [28], the plant's buildings had to be reinforced, especially the structural steel work and steel reinforcements in the reinforced concrete. The most extensive reinforcement work was in the SG building in 2000. The licensing process needs to scrutinize the seismic and external dynamic loading events of SFRs.

3.11 Inservice Inspection

Although very little information on inservice inspection was available, below is an example of inservice inspection in SFRs.

The VISUS system was an ultrasound system developed to detect objects in opaque sodium in Phénix [10, page 44]; it monitored the movement of the subassemblies within the reactor to detect possible problems. After various adjustment problems, the VISUS system worked extremely well. This device was a very useful instrument for “seeing through” sodium by compensating for its opacity during potentially risky handling operations.

Phénix used a periscope to monitor the space above the sodium in the storage drum [10]. In June 1976, during the loading of new fuel subassemblies, an unusual object resembling a metallic rod appeared on top of a handling flask in a breeder subassembly placed half an hour before the rod's appearance. After it was visualized and photographed several times, the object disappeared. This unusual object was later determined to be a rod of solid sodium that froze in the upper neutron shielding channel of the new fuel subassembly while it was being inserted into the storage drum and that had been held upright under the effect of the hydrostatic pressure. The temperature of the argon atmosphere in the storage drum was not high enough to melt the sodium rod. It was subsequently melted by raising the temperature.

Phénix [10] set up a sound navigation and ranging (SONAR) system for inspection in 1993 following a negative reactivity trip. Its role was to detect movements in core subassemblies in case a new reactivity trip occurred. The SONAR device also detected acoustic signals in the core and measured the magnetic field in the vessel.

In 1982, the French safety authority requested a comprehensive inspection before fuel loading in Superphénix [9]. A robotic device called the MIR (Fast Reactor Inspection Module) was developed and could move in the containment vessels. The MIR used ultrasonic sensors for inspecting welds. The MIR could also take distance measurements between components within the SFR vessel. In addition, the MIR was equipped with cameras to make videos of the main vessel and safety vessel (Figure 8). The use of the MIR robot and the excellent correlation between the measurements carried out during construction and those taken by the MIR demonstrated the feasibility of reactor vessel inservice inspection by ultrasonic techniques.



Figure 8 MIR inspection device (from [9, Figure 19.2])

The last operational period for Superphénix extended from August 1994 to December 1996, during which the most noticeable event was the occurrence of a small leak at the argon feed tube of the gas sealing bell of one IHX in January 1995. Visual inspection using an endoscope inserted in the tube above the reactor roof found the precise location of the leak. An internal sleeve was expanded through pressurization to plug the leak without the need to remove the IHX.

Superphénix had several methods for detecting sodium leaks [9]. In addition to conventional fire detectors, 62 aerosol detection systems equipped with spectrophotometers were installed throughout Superphénix. Approximately 270 spark plug-type devices were used to detect sodium leaks. These devices operated on the principle that sodium leakage would cause a short circuit between the ground and the spark plug electrode, thus triggering an alarm. The devices proved to be unreliable for two principal reasons: (1) the degassing of products in the thermal insulation deposited on the spark portal electrodes caused them to short to ground and (2) keeping the spark plugs at high temperatures caused a loss of electrical insulation by degrading the spark plug's internal insulation. Roughly 900 beaded wires were also installed beneath thermal insulation where the wires were set to short circuit during a sodium leak and sound an alarm. In general, the spark plug and beaded wire detectors were unreliable, and a "sandwich" detector was developed consisting of a steel sheet electrode placed between two layers of insulating felt. The new detectors operated satisfactorily, and installation of the new detectors began in 1992. However, the OpE of the detectors was curtailed because of the closure of Superphénix.

In EBR-II, the primary containment shield tank consisted of an inner and outer Type 304 SS tank with a 127 mm (5 in.) annulus between the two that was filled with inert gas and surrounded by a 1.83 m (6 ft) concrete biological shield [27]. The annulus region was continually monitored for sodium aerosols and air leakage through either tank wall (monitoring devices not specified).

3.12 General Design Issues

During the design phase, many potential material and structural integrity issues were carefully considered. The designers of Superphénix were much more cognizant of issues related to thermal fluctuations in various components after operation of the Phénix. Some issues that needed consideration are listed below; however, these issues were not necessarily problems in the Superphénix plant. This section discusses lessons learned from OpE related to materials performance, including favorable performance, and potential materials issues:

- Thermal stratification and mixing between hot and cold sodium pools and zones must be managed because these can lead to thermal fatigue of the metal in these zones. In normal operation, sodium temperatures in Superphénix could vary between 545 degrees C (1,013 degrees F) at the core outlet to 395 degrees C (743 degrees F) at the IHX outlets, raising the possibility of thermal stratification in the mixing zones and significant local stresses. Superphénix developed several thermohydraulic codes [9, Chapter 18] (e.g., SUPERCAVNA to model thermal stratification and NAJECO and NAJET for jet mixtures). In addition, the RCC-MR code had many developments and additions to handle high-temperature design and damage assessments to support Phénix and Superphénix.
- For similar reasons, startup transient temperature fluctuations must also be managed because significant stresses can occur during the startup and shutdown transients. A

number of mockup tests in Superphénix validated thermohydraulic codes to ensure that thermal mixing and stratification were properly managed [9].

3.13 Other Topics

3.13.1 Steam Generator Tear Down of Experimental Breeder Reactor II

In April 1981, the superheater was removed from the EBR-II steam system for destructive examination after more than 16 years of operation [23]. The superheater was disassembled in a sequence that would progressively yield examination results before their obliteration by subsequent disassembly. The steam surfaces contained lightly scattered corrosion pitting. The pits were less than 0.25 mm (0.01 in.) in depth and appeared to have been formed early in the life of the tubing, as evidenced by the oxide coating. The inside diameter measurements were within 0.05 mm (0.002 in.) of the nominal fabricated diameter of 27.05 mm (1.06 in.). The straightness measurements of the tubes indicated that some tubes were bowed and the peripheral tubes were bowed more than the central tubes. The bowing was consistently in the same direction (i.e., the tube bundle was twisted in one direction and then returned in the opposite direction). Plasticity, creep from operations, or thermal expansion stresses could have caused this distortion. Investigations of the duplex (double-walled) SG tubing indicated it was “feasible that some creep or stress relaxation of the material has occurred in the tubes” [23, page 13]. As discussed in Reference [57], EBR-II never experienced a sodium/water reaction because of the use of duplex (double-walled) SG tubes (i.e., concentric brazed tubes swaged together).

The SG tube-to-tubesheet welds were a matter of concern because they were located at the highest stress point [23]. Liquid penetrant examination showed discontinuities in some welds. The baffle nest, which was suspended from inadequate welds and had broken away from its support ring, was discovered 22 inches from its as-built location. During operation, the baffle nest had apparently “floated” in the sodium flow stream within 2 inches of its as-built location. In general, however, the superheater was found to be in remarkably “like new” condition with even the original soapstone marks clearly visible on the baffle nest. It may be useful to examine teardown inspections of many other SFR components from this and other SFR plants to infer more information on the long-term OpE of SFRs. There are probably other teardown results that the review could not identify.

Sodium leaks occurred at all SFRs examined in this report at some point during their operational lives. For example, the Phénix plant is typical for the types of leaks that have occurred in SFRs. During its 35 years of operation, Phénix faced 31 sodium leaks. Most of these leaks were located on welds of secondary loops and auxiliary circuits; fewer leaks occurred in the primary system. A leak caused by corrosion occurred in Phénix only once. Each leak in Phénix had consequences on plant availability but never on its safety. Table 2.11 of Reference [24] provides a complete summary of all sodium leaks that occurred in Phénix. This summary table shows the leak location, leak detection method used, amount of leakage, and temperature. The fluid temperatures at the leak location ranged from about 100 degrees C (212 degrees F) to 550 degrees C (1,022 degrees F). In addition, during the 35 years of Phénix operation, five sodium/water reactions occurred, all in the reheater module. Most began soon after the startup of the plant after a shutdown and were located at butt welds on the hottest part of the SG tubes. Startup and shutdown transients may have played a role in these leak reactions.

3.13.2 Potential Emergency Issues with Intermediate Heat Exchangers in Sodium-Cooled Fast Reactors

During an emergency shutdown (scram) operation in an SFR, the IHXs (Figure 9) may experience a thermal shock when the cold sodium arrives, which could lead to tube buckling among other structural issues [9]. This is amplified in the case of an earthquake. Future SFR designs must consider this issue. The IHX could operate up to 542 degrees C (1,008 degrees F) in Superphénix.

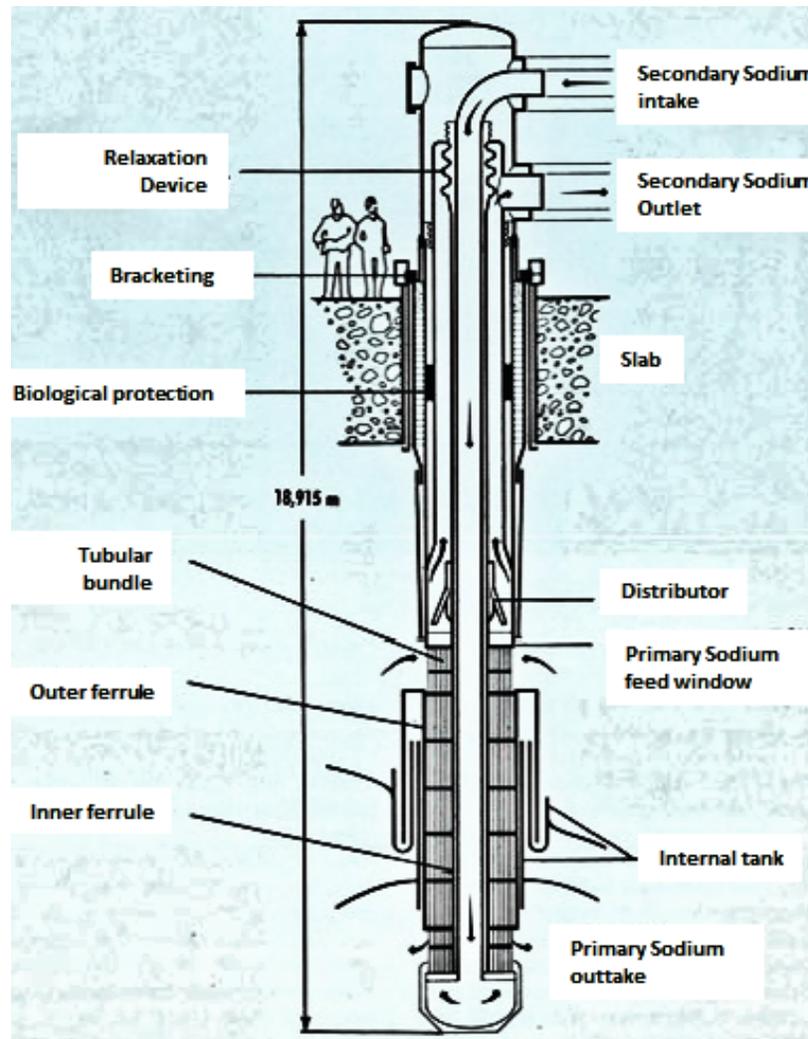


Figure 9 Superphénix heat exchanger (from [9], Figure 11.1])

3.13.3 Weld Reheat Cracking

The problem of delayed reheat cracking is a concern that can affect any welded structure operating at high temperature and did affect welds at several SFR plants, including Phénix and the PFR. At Phénix, the secondary piping and buffer tanks were susceptible to reheat cracking [28, 29]. These components are made of Type 321 SS (stabilized with titanium) and were some of the hottest components of the sodium envelopes of the SG superheater and reheater modules. The secondary piping loop was not considered safety significant, which may also

have reduced the welding quality and contributed to failure at the welds. Replacement with 316LN material in Superphénix, based on Phénix experience, minimized this issue. Reheat cracking was also observed in PFR [19].

3.13.4 Double-Walled Tube Steam Generator Proposal

To enhance the reliability of SGs against a water/sodium reaction caused by tube failure, Japan has developed a double-walled SG tube [30]. Figure 6 gives an example of a double-walled SG tube in EBR-II. In addition, a collaboration between Japan Atomic Power Company and the DOE is progressing on a performance test of an SG with a double-walled tube of 2Cr-1Mo steel.

3.13.5 Fuel Handling

Although fuel handling is not considered a structural or materials issue, mishaps during fuel handling can significantly affect reactor operation and can potentially lead to structural problems; this presents a challenge because sodium is opaque. Reference [10] discusses some fuel incidents that occurred in EBR-II.

3.13.6 Sources of Plant Shutdowns

Superphénix shutdowns were dominated by failures in the steam side of the plant in accordance with Table 2 [9]. The complexity of the nonnuclear portion of the plant, which included two separate 600-megawatt-electric (MW_e) turbo generators, may have been a strong contributor.

Table 2 Origins of Superphénix Shutdowns (from [9, Figure 6.2])

Origin	Contribution
Water-steam plant	59%
Scheduled	18%
Instrumentation and control	10%
Reactor	6%
Steam generator	5%
Secondary	2%
Core	0%
External	0%

3.13.7 Material Performance

Table 3, adapted from [12], summarizes the materials used for major components in many of the world's SFR reactors. Different grades of SS dominate the material designs for the various SFR components. Nickel-based superalloys (e.g., 9Cr-1Mo and modified 9Cr-1Mo (Grade 91)) are being strongly considered for cladding in future SFRs because of their good resistance to void swelling.

Selection parameters for reactor assemblies include tensile strength, creep response, low-cycle fatigue performance, creep-fatigue interaction, and high-cycle fatigue caused by vibrations. Fracture toughness and weldability are also important. Low-alloy steels are not considered suitable in the SFR heat transport system because they usually do not have adequate high-temperature material properties.

SFR criteria [31] for selecting the SG materials in Table 3 address the same concerns as those discussed above for the reactors; however, good ductility and resistance to aging effects are also important. Mechanical properties allowing performance in sodium are necessary, including with regard to the material's susceptibility to decarburization. Corrosion resistance to molten sodium and aqueous sodium hydroxide, wear resistance, and proper weldability (noted in the OpE) are important. Fabrication issues have also caused many failures. Reference [31, 32] discusses the material requirements and selection criteria for SFRs.

Table 3 Materials Selection for SFRs (from [12])

Reactor	Cladding	SG Evaporator Material	SG Superheater Material	SG Reheater Material	IHX Shell Material	IHX Tube Material	Primary Hot Leg (°C)	Primary Cold Leg (°C)	Primary Hot Leg Material	Primary Cold Leg Material	Secondary Hot Leg Material	Secondary Cold Leg Material	Secondary Hot Leg (°C)	Secondary Cold Leg (°C)
EBR-II	316	2.25Cr-1Mo	2.25Cr-1Mo		304	304	473	371	304	304	304	304	467	270
FFTR	316 (20% CW)				304	304	503	360	316	304	316	304	459	316
DFR	Nb				18Cr-18Ni-1Mo	18Cr-18Ni-1Mo	350	230	18Cr-18Ni-1Mo	18Cr-18Ni-1Mo	18Cr-18Ni-1Mo	18Cr-18Ni-1Mo	335	195
PFR	Various (includes Nimonic PE16)	2.25Cr-1Mo Nb stab.	Modified 9Cr-1Mo	Modified 9Cr-1Mo	316 BS (1501)	316 BS (1501)	560	399	321	321	321	321	540	370
BN 350	16Cr-15Ni-2Mo-Ti-Si (CW)	2.25Cr-1Mo	2.25Cr-1Mo		18Cr-9Ni	18Cr-9Ni	430	280	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	415	260
BN 600	15Cr-15Ni-2Mo-Ti-Si (CW)	2.25Cr-1Mo	2.25Cr-1Mo	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	535	365	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	510	315
BN-800	16Cr-15Ni-2Mo-Ti-Si (CW)	10Cr-2Mo-V-Nb	10Cr-2Mo-V-Nb				547	354	18Cr-9Ni	18Cr-9Ni	16Cr-11Ni-3Mo	18Cr-9Ni	505	309
BOR-60	16Cr-15Ni	2.25Cr-1Mo	2.25Cr-1Mo & SS		18Cr-9Ni	18Cr-9Ni	530	330	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	480	210
BR-10	16Cr-15Ni-3Mo-Nb				18Cr-9Ni-Ti	18Cr-9Ni-Ti	470	350	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	18Cr-9Ni	380	270
JOYO	316 (20% CW)				316FR & 304	316FR & 304	465-500	350-370	304	304	2.25Cr-1Mo	2.25Cr-1Mo	445-470	300-355
MONJU	Modified 316	2.25Cr-1Mo	Austenitic SS		304	304	529	397	304	304	304	321	505	325
Phoenix	17Cr-13Ni-2.5Mo-1.5Mn-Ti-Si	2.25Cr-1Mo stab.+unst ab.	321 H	321 H	316	316	560	395	316	316	321	304	550	343
Rapsodie	316				316	316	515	400	316	316	316	316	485	360
Super Phoenix	17Cr-13Ni-2.5Mo-1.5Mn-Ti-Si	33Ni-21Cr-Ti-Al-Mn			316LN	316LN	545	395	18Cr-12Ni-2.5Mo-1.8Mn-Si	18Cr-10Ni	18Cr-12Ni-2.5Mo-1.8Mn-Si	18Cr-12Ni-2.5Mo-1.8Mn-Si	525	345
FBTR	316 (20% CW)	2.25Cr-1Mo stab.	2.25Cr-1Mo stab.		316	316	515	380	316	316	316LN	316LN	510	284

The Japan Atomic Energy Agency developed Type 316FR SS. This structural material is used for the IHX in the Joyo reactor and is the lead material for the future fast breeder reactor in Japan. It is an austenitic SS with improved high-temperature creep properties. The low carbon content and the optimization of the phosphorus and nitrogen content enhance the strength of

Type 316FR SS within the component range of the Japan Industrial Standards for Type 316 SS [24].

3.13.8 Material Choices for Future Sodium-Cooled Fast Reactors

This section discusses advanced SFR plants being built in other countries.

ASTRID (France)

ASTRID is the proposed advanced SFR (600 MW_e) to be built in France. The material candidates for ASTRID are based on prior SFR designs (i.e., Phénix, Superphénix, and Superphénix 2) and the European Fast Reactor design [32]. For the materials in contact with sodium, the choice of materials is based on the following three criteria:

- (1) corrosion resistance to sodium, including resistance to intergranular attack
- (2) easy to weld
- (3) sufficient/superior mechanical properties

Based on these criteria, the designers chose Type 316L austenitic SS with controlled nitrogen for the main vessel and the internals operating at high temperature (550 degrees C) (1,022 degrees F) and possibly Type 304L for lower stress structures operating in the negligible creep regime. These materials satisfy the above three criteria. The low carbon content ensures better corrosion resistance, particularly against impurities that maintenance and repair activities may introduce into the sodium.

Choosing an SG material is complex because of the need to consider various aspects. In addition to the criteria noted above, SGs must have (1) sufficient creep resistance, (2) excellent resistance to water and water vapor corrosion [32], (3) the ability to undergo inspections and in situ maintenance and repair, and (4) in particular, a design that facilitates cleaning operations. Phénix SGs used chromium-molybdenum-grade material, and the vapor collectors of Superphénix were made of 2.25Cr-1Mo steel. Owing to its excellent oxidation resistance and fabrication characteristics, Alloy 800H is a good candidate for ASTRID.

Prototype Fast Breeder Reactor (India)

India's next generation SFR, the Prototype Fast Breeder Reactor, will use of 304LN in place of 316LN for the cold-leg near-core components and the use of chromium-molybdenum in place of 304LN/316LN for the secondary sodium system are the focus areas for reducing material costs. The 2.25Cr-1Mo steel is under study for the secondary sodium storage tank and auxiliary cold-leg piping, and modified 9Cr-1Mo steel is being considered for the surge tank and hot-leg piping [33].

3.14 Lessons Learned (Sodium-Cooled Fast Reactor)

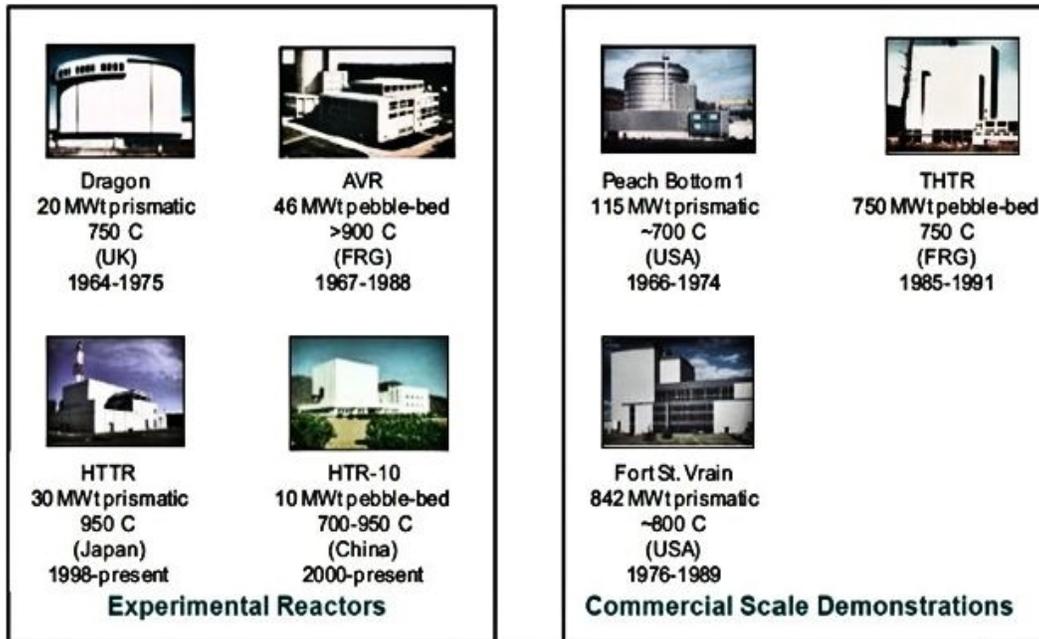
- (1) Sodium heat-transport systems experienced a significant number of leaks caused by poor weld design and poor weld quality control. Initial fabrication defects, particularly at welds, must be identified during construction. Welds should be carefully designed to minimize residual stresses, and direct tube-to-tube-plate welds should be avoided. Lowering the threshold for quality in welds and secondary loops has resulted in operational problems.

- (2) Repair procedures in SFRs must be carefully planned because weld residual stresses in repair welds are often highly tensile.
- (3) Reheat cracking is a concern in SFR components operating at high temperatures. In particular, weld repairs should be carefully managed because they may give rise to very high tensile residual stresses. Some types of austenitic SS (e.g., Type 321) are significantly more prone to reheat cracking than other austenitic SS.
- (4) Stresses induced by thermal expansion, particularly in areas of constraint, must be carefully considered for SFRs. These stresses have often been the source of structural integrity issues in SFR operation. In the case of Phénix, excess weld material led to over-constraint and cracking. The issue was resolved by eliminating unnecessary weld material.
- (5) Thermal fatigue (thermal striping) is a much more significant issue than it is for LWRs. Thermal fatigue in SFRs is caused by mixing sodium flows of different temperatures and must be carefully assessed to prevent fatigue.
- (6) Management of the startup and cooldown transients in SFRs to control vibration, thermal expansion loads, and possible fatigue issues in the components is important.
- (7) Avoid shrink-fit parts in pumps that could loosen during some thermal transients and inspect all pump welds carefully.
- (8) Oil-based lubricants should be avoided in SFRs.
- (9) Possible valve failures (all system valves especially those operating at high temperature) are a concern for SFRs. Valve reliability under operating conditions should be accurately determined.
- (10) Austenitic steels are unsuitable for SFR SGs because of the high risk of caustic stress-corrosion damage following even small leaks.
- (11) Testing should confirm the chemical compatibility between molten sodium and insulation material.
- (12) The proper choice of materials for SFRs requires complete material databases.
- (13) Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure and sodium leaks.
- (14) The failure of in-sodium components without an adequate means for removal and repair has resulted in costly and time-consuming recovery.
- (15) Sodium contamination and the consequent formation of sodium oxide have caused the binding of rotating machinery and control rod drives.
- (16) Better detection methods of corrosion and leaks are necessary, particularly in regions coated with insulation. The design phase should consider sensor placement and reliability under operating conditions. Inadequate or unreliable leak detection systems have caused extensive shutdowns because of sodium contamination and excessive sodium leaks with consequent fires.

- (17) The licensing process needs to scrutinize seismic and external dynamic loading events of SFRs. During an emergency shutdown (scram), the IHX may experience thermal shock caused by the influx of cold sodium. This condition could lead to buckling and structural issues amplified by an external loading.
- (18) Reference [34] assesses applicable standards and codes for SFRs, and future companion reports will provide more information on such codes.

4 HIGH-TEMPERATURE GAS-COOLED REACTORS

As of April 2010, seven HTGRs have been designed and operated throughout the world [37]. Figure 10 [37] depicts these, along with information on the plants themselves. Reference [35] discusses standards and codes applicable to HTGRs. This section describes OpE for HTGRs; OpE is provided by component for many of these plants. This section of the report lists the overarching lessons learned from the OpE with regard to material performance and structural integrity. For reference, the OpEs are numbered for each component below. Chapter 2 of Reference [37] provides a general, high-level overview of the OpE of these plants.



Reactor	Country	Design	Criticality (yr)	Shut Down (yr)	Power (MWth)	Power (MWe)	He Pressure (MPa)	Primary Hot Leg (°C)	Primary Cold Leg (°C)
FSV	USA	Prismatic	1976	1989	842	330	4.8	775	400
Peach Bottom I	USA	Rod	1967	1974	115	40	2.3	725	325
Dragon	UK	Rod	1968	1975	20	N/A	2	750	350
HTRR	Japan	Rod-in-Block	1998	N/A	30	N/A	4	950	395
AVR	Germany	Pebble Bed	1967	1998	46	15	1.1	950	275
THTR-300	Germany	Pebble Bed	1986	1989	700	300	4	750	250
HTR-10	China	Pebble Bed	2000	N/A	10	N/A	3	700	250

Figure 10 Important operating gas-cooled reactors (from [37, Figure ES-1])

4.1 Fuel Structure

- (1) (Fort St. Vrain (FSV), October 1981): The licensee discovered a crack had propagated through two stacked fuel elements caused by high tensile stresses induced by irradiation. High tensile stresses can develop in the fuel elements caused by irradiation swelling at high temperatures, leading to cracking [36, 37]. Reference [38] summarizes attempts to improve graphite and carbon-graphite resistance to this effect. In general,

the operational history of FSV can be characterized by low availability (capacity factor) and inconsistent power production [39]. Graphite swelling under irradiation must be carefully evaluated with supporting data.

- (2) (Arbeitsgemeinschaft Versuchsreaktor (AVR), general): During a test campaign using special pebbles containing melt wires, Germany's AVR pebble bed HTGR [40] showed maximum core temperatures greater than 1,280 degrees C (2,336 degrees F) at an average coolant outlet temperature of 950 degrees C (1,742 degrees F). The maximum core temperature was 200 degrees C (392 degrees F) greater than maximum design temperature of the fuel pebbles [41]. The excess temperatures led to higher releases of fission products and overheating of heat exchanger components and hot-gas ducts. Elevated temperatures may have also contributed to cracking of the graphite reflectors. There is no convenient way to monitor active core temperatures because pebble movement destroys standard detection equipment [41, 49]. Hot-gas temperatures were measured outside the reactor core, and computational codes were used to infer core temperatures. These codes were inaccurate as evidenced by the excessive core temperatures during the test campaign. One reason for the high temperatures is local pebble densification occurred, for which the designers did not account. Denser packed regions experienced a greater pressure drop and higher temperatures, as observed from measurements. Finally, the hot-gas streams observed in AVR may lead to overheating in parts of the SG or other metallic components. Chapter 11 of Reference [49] describes the AVR pebble bed fuel design, circulation, and computational temperature estimation procedures. These were originally based on a two-dimensional diffusion computer code but were improved to three-dimensional calculations in the 1980s as computer speed and memory improved. Reference [49] also describes the continuous processes of fuel pebble loading, pebble movement in the core, and pebble removal after burnup. Calculating HTGR core temperatures is problematic, particularly for pebble bed designs.
- (3) (Thorium High-Temperature Reactor (THTR), circa 1986): Germany's THTR-300 (a pebble bed THTR) experienced higher temperatures than the design had anticipated [37]. Thermal fatigue from excessive thermal gradients across the core outlet, with a possible contribution from thermal neutrons, caused the insulation attachment bolts in the hot-gas duct to fail very early in the operation of the reactor. Uncontrolled core temperatures were thought to be a major cause. In addition, the insertion of control rods directly into the core during "unfavorable conditions" damaged the fuel elements. Such conditions were not mentioned. Calculating HTGR core temperatures is problematic, particularly for pebble bed designs.
- (4) (FSV, August 1984): Moisture entering into the helium cooling gas leached volatile chlorides from various sources, causing chlorine-induced stress-corrosion cracking of an SS control rod cable [37]. The solution was to replace the SS cables (the type was not identified) with corrosion-resistant Inconel cabling (the type was not specified). Because moisture can cause numerous structural corrosion and cracking issues, future HTGRs must manage moisture correctly and establish convenient methods to control it.
- (5) (AVR, 1971–1981): The reducer in the AVR was a slowly rotating slotted double disk at the end of the pebble extraction pipe (Figure 11). The fuel pebbles move through the disk and down the pipe. As the disk rotates, the pebbles are statistically distributed into the core. Numerous problems with this device occurred over the years. In January 1971, increased internal friction caused the drive motor to fail. In October 1974, a drive cam shear failure caused the reducer to malfunction. In July 1976, the bearing shield became loose, preventing the reducer from rotating and dispensing fuel. In

August 1976, a feather key sheared off in the reducer drive, stopping the reducer entirely. In December 1981, the drive no longer turned because of corrosion caused by water condensation. The repeated failure of the reducer, in addition to other components of the AVR, can be attributed to poor design [49]. Although these malfunctions did not necessarily endanger the public, the plant was shut down numerous times. Reference [49] discusses the numerous problems and incidents with the reducer that caused the plant to shut down. Other mechanical issues with the fuel dispensing and storage system that also occurred indicate a poor design of the entire system.

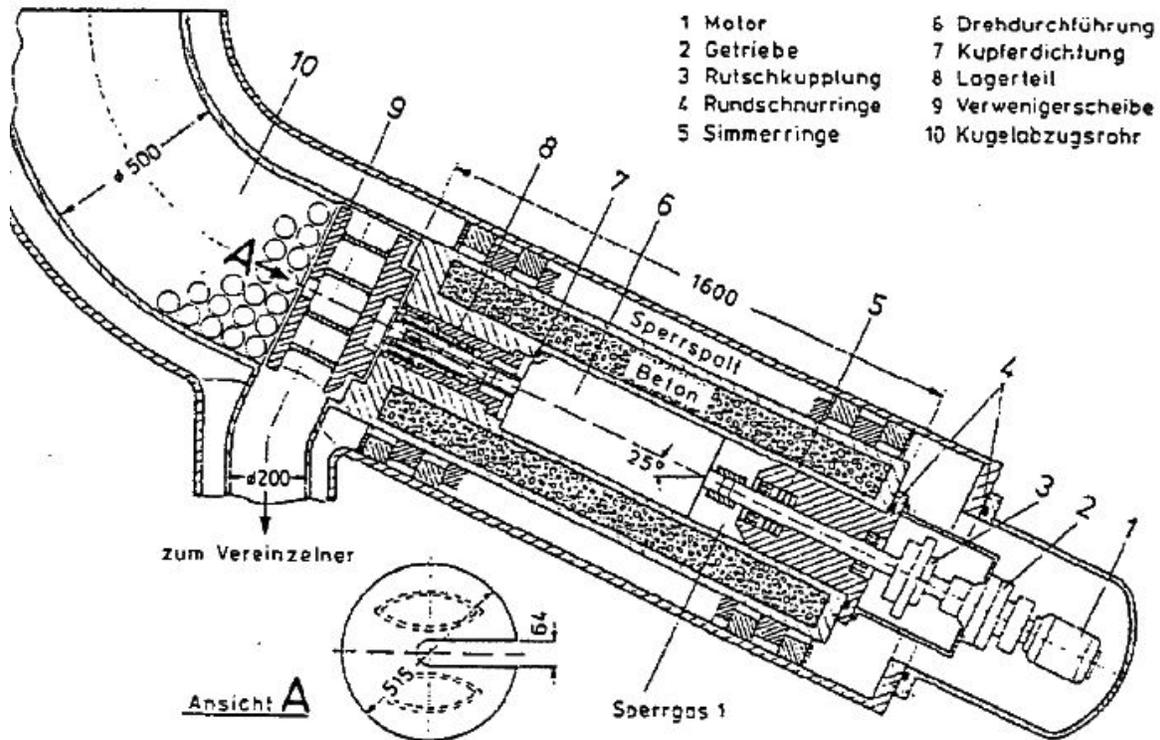


Figure 11 AVR reducer (from [49, Figure 13.3-2])

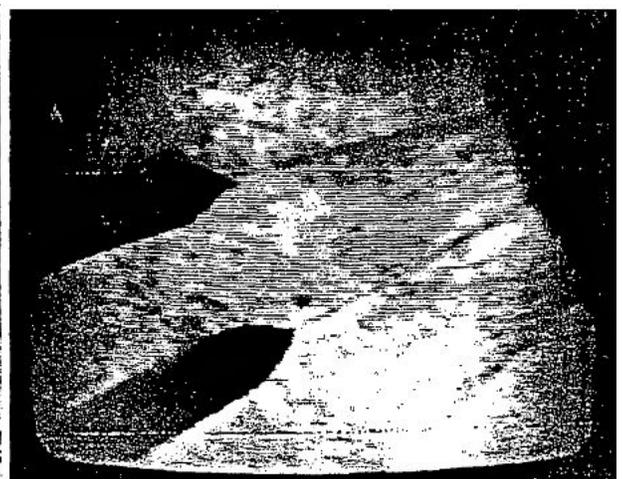
4.2 Core and Core Structures

- (1) (High-Temperature Reactor (HTR)-10, general): In China's HTR-10 (a pebble bed reactor), non-uniform temperature distributions and deformation from neutron irradiation resulted in increased graphite wear caused by fretting of graphite blocks. The resulting graphite dust can clog the heat exchanger and induce erosion. Graphite damage must be minimized [37].
- (2) (AVR, May 1986): A visual inspection of the AVR core after defueling showed cracks in the bottom graphite reflector. A number of pebbles had sunk into various coolant penetration slits of the graphite bottom reflector structure and could neither roll off into the fuel discharge tube nor be removed by a manipulator [40]. The source of the graphite cracking was not reported.

- (3) (High-Temperature Test Reactor (HTTR), 1999): During power-rise tests up to 20 megawatts, the temperature unexpectedly rose at the core support plate caused by gas flow through gaps in the core support structure. Because a repair was not possible, the design margins were reevaluated. The analysis showed the higher temperatures could be tolerated. This illustrates the importance of considering helium flow in gaps for HTGR structures [47, 53].
- (4) (AVR, 1984): In 1984, AVR went through a rigorous inspection (after about 10 years of service) of the core structure to determine whether damage had occurred [49, Chapter 13], with special attention paid to examining the graphite. The inspection revealed little damage (Figure 12) to the graphite from material abrasion, corrosion, or shrinkage [49]. Graphite damage can be minimized through proper reactor design and quality control of graphite fabrication.



Original graphite ceiling reflector



Reflector after 10 years service

Figure 12 Graphite damage in an AVR reflector after service (from [49, Figure 13.1-3])

- (5) (Peach Bottom Atomic Power Station (Peach Bottom), 1967): Several factors contributed to the fatigue failures experienced by the control rods in ball-screw assemblies [37, 42]. The plant accumulated large numbers of cycles during the testing and personnel training period before plant startup. These additional cycles induced additional fatigue damage for which the plant was not designed. It is important to consider the accumulation of cycles in HTGRs during testing phases.

4.3 Reactor and Pumps

- (1) (FSV, lifetime): Moisture was a significant problem for the FSV reactor. The two primary sources of moisture were through the circulator bearings [37] and through small weld cracks in the prestressed concrete reactor vessel (PCR) liner during startup. These small weld cracks sealed during high-temperature operation. FSV had limited ability to remove moisture once it got into the primary system because it lacked a reactor drain. By the end of November 1988, almost 3,800 liters (1,000 gal) of water were removed

from the reactor primary system [36]. Reactors should have a way to remove water that inadvertently enters the core.

- (2) (FSV, related to OpE No. 1 above): Moisture in the helium coolant could attack the graphite fuel element and cause corrosion of steel components, particularly the control rod drive mechanisms that affect reactivity. Because moisture can cause numerous structural corrosion and cracking issues, it must be managed correctly. It is critical for future HTGRs to have convenient methods for preventing moisture ingress.
- (3) (FSV, 1984): The PCRV tendon wires in the liner revealed significant corrosion issues [36]. The PCRV system consisted of 448 tendons, each made of 3.87 meters (12 feet 10-1/4 in.) wires. Load cells were used to detect any loss of prestress. The licensee [36] found broken or corroded tendon wires in at least six tendons. The tendons are stored in sealed boxes to prevent moisture from entering. Prestress was used to produce compression in the PCRV liner (Figure 13). The material was steel containing 0.72- to 0.93-percent carbon, 0.4- to 1.10-percent manganese, 0.1- to 0.35-percent silicon, and 0.04-percent (maximum) phosphorous and sulfur, and it met the requirements of ASTM A421, "Standard Specification for Stress-Relieved Steel Wire for Prestressed Concrete," issued 1970 [43]. Sulfonate grease used on the tendon wires combined with oxygen from air ingress created acetic acid, which corroded the tendons. Several possible sources of corrosion were determined and summarized in Reference [36]. One example of tendon wire corrosion was traced to grease missing from the ends of tendon wires in combination with moisture in the sealed boxes where the tendon wires were stored before construction. Later, the plant found that microbiological attack had also contributed to the corrosion and damage to the wires [44]. PCRVs are subject to degradation mechanisms that traditional RPV steels do not encounter in service.

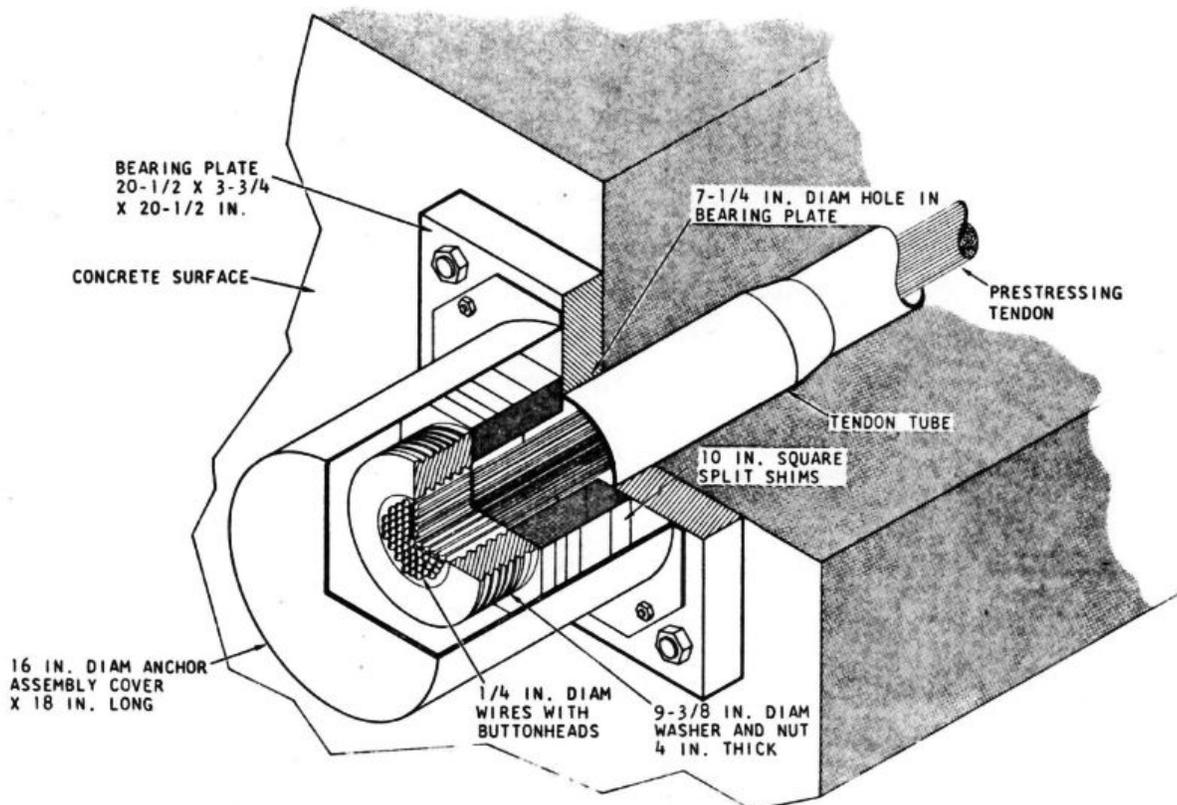


Figure 13 Prestressed tendon wires affected by corrosion damage (from [44])

- (4) (FSV, 1985): Moisture that collected between the 3/4-inch supply line in the 1/8-inch inlet line resulted in corrosion products that plugged the carbon steel helium pressurizing lines. Current HTGR designs may not encounter this issue if carbon steel is not used [36]. Carbon steel is particularly susceptible to corrosion from moisture ingress.
- (5) (AVR, first quarter 1980): In the AVR, four control rod drives were inserted from below into holes bored into the graphite reflector (the rods do not drive into the pebble bed). In general, the control rods performed well during operation. Inspection of the four control rods drives found cracks in the bellows of the couplings, and new bellows were welded in place. The cause of the cracking was not specified, but thermal fatigue may have caused the cracking because the cooling flow is anticipated to be complex near the bellows [49].
- (6) (AVR, fourth quarter 1974 and 1977): In 1974, a crack was found in the membrane compressor head of AVR [49, page 421]. Ten membrane compressors were used to pump helium from the reactor annulus into bottle storage and to transport it through the gas circuits and gas purification plants. After replacement, a similar crack was found in 1977. The original cast iron head was replaced with an austenitic steel (the type was not specified). Although the exact cause of the cracking was not listed, it was suspected that thermal fatigue or creep fatigue⁵ caused it. Operators eventually replaced all of the

⁵ Creep fatigue design and modeling techniques are quite complex, and all code bodies attempt to use very conservative interaction diagrams for assessment.

membrane compressors with dry running piston compressors because the latter were found to perform better. In general, the membrane compressors did not perform as well as the dry running piston compressor because the membranes had to be replaced frequently.

- (7) (HTTR, lifetime): The helium gas compressors (HGCs) in Japan's HTTR had problems similar to Germany's AVR [45]. The HTTR has 14 HGCs for circulating helium coolant in the helium purification system, helium gas sampling system, helium storage and supply systems, and the radioactive waste treatment system. These reciprocating-type compressors are specific to HTGRs. Degradation in compressor performance occurred repeatedly in the HTTR. In one incident in 2004, the volumetric flow rate of the HGCs decreased during operation. Inspections revealed that piston rings were abnormally abraded (Figure 14), which degraded the performance of the piston-ring seals. Further, the resulting graphite dust from the piston rings circulated in the primary cooling system. This dust increased differential pressures in the filters of the primary helium gas circulators [46], which increased pressures in the filters and overloaded them on the primary HGCs. As result, the HGCs had to be replaced often (this issue also occurs in the AVR). Originally, the replacement of these filters was not considered to be necessary throughout the lifetime of the design. To solve the abnormal abrasion of the piston rings, the tolerance for wear was optimized so that the flow rate of the HGCs did not fall significantly until the abrasion of piston rings reached the design limit. This also effectively prevented increases in differential pressures in the filters of primary gas circulators. Graphite dust should be minimized in HTGRs.

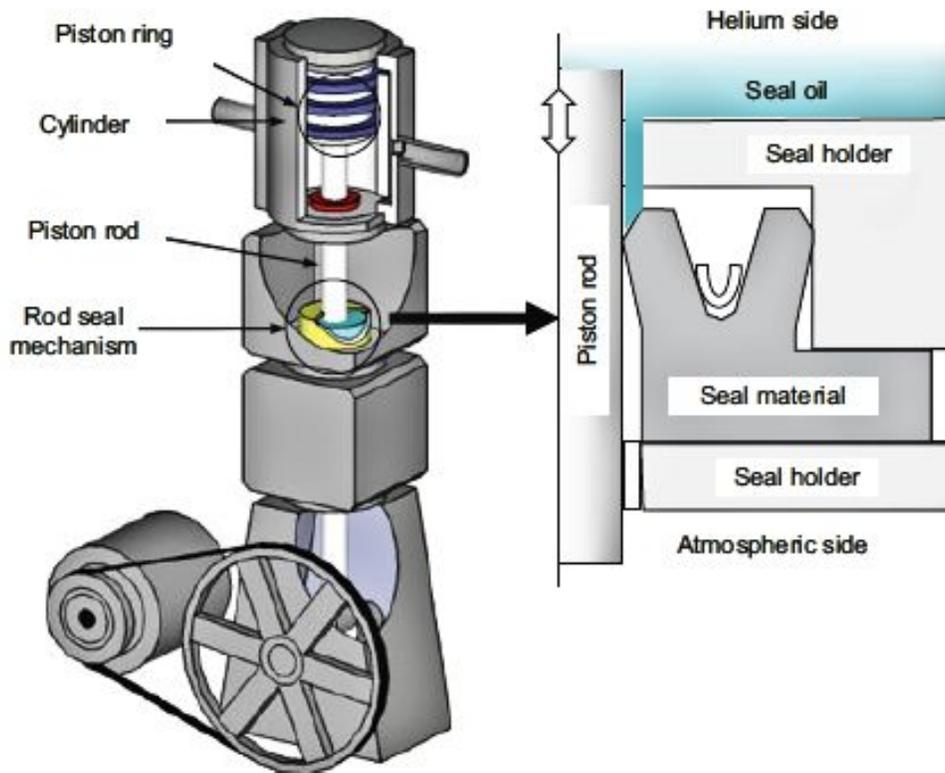


Figure 14 Schematic of the exterior and internal mechanisms of a reciprocating HGC (from [45, Figure 4])

- (8) (HTTR, lifetime up to 2010): Another generic problem with the HTTR compressors involved oil seal performance (a similar issue occurred in the AVR). Seal oil repeatedly leaked from the piston rod mechanism of the HTTR HGCs during long-term operations [47]. The oil seal forms a pressure boundary between the piston mechanics (helium side) and the piston crank mechanism (the atmospheric side; see also Figure 14). Studies showed that operating at high speeds caused leakage. This exceeded the material tribology limits of the seal material, which was originally a polyurethane-based material selected for its heat and radiation resistance. The solution was to replace the polyurethane seal material with a Teflon-based material that has superior heat resistance. This solution appeared to remedy the problem until at least 2010. Whenever possible, oil seals should be avoided in HTGR designs.
- (9) (FSV, 1988): The FSV was shut down for a scheduled 12-week outage to replace bolting material on the helium circulators [37]. The bolts holding the insulation shroud and steam seal in place failed because of caustic stress-corrosion cracking. In addition, the “D” helium circulator experienced corrosion cracking and had to be shut down because of excessive circulator shaft vibration. The materials used for construction were not reported, but they were probably a type of ferritic steel. A small crack in the core support floor section of the liner cooling system [37] allowed entry of the moisture, which led to the corrosion. The cause of the crack was not specified. HTGR designs must account for corrosive effects and caustic embrittlement of fastener and other mechanical components.
- (10) (Dragon, 1968): An inspection found that the bearings in the Dragon single-stage centrifugal blowers were damaged during an inspection [37,48]. The bearing lubrication could not support the necessary weight without a minimum circulator rotational speed. The damage occurred during startup and shutdown, when the rotation speed was low and the bearings were under higher stresses. Reactor components should be able to withstand stresses during shutdown and lower than full-power operation.

4.4 Steam Generator/Turbine

- (1) (Peach Bottom, lifetime): The Peach Bottom reactor was the first nuclear power plant to use Alloy 800 for the SGs. The reactor did not experience any leaks during 7 years of operation [35]. Section 4.8 briefly discusses the use of Alloy 800 in HTGRs (Item (9)).
- (2) (FSV, 1987): Hydraulic oil leaked from a valve causing a fire in the FSV turbine building. The hydraulically controlled valve was the Loop 2 main steam bypass valve [35]. The fire damaged many pieces of equipment and electrical cables, causing a loss of power to the control room radiation monitor. This incident was actually an operational problem [37] because the root cause determination found that a restrictor orifice was left out of a thermal relief for this valve, causing the relief to open and the pressure to surge. This fire is unusual because a steam pipe ignited it. HTGR designers must consider potential sources of fire that LWRs are not subject to, including the ignition of hydraulic oils [35].
- (3) (AVR, general): Water resulting from SG tube ruptures entered the AVR HTGR [40]. The design lesson from the AVR experience is to ensure that the impact of SG tube ruptures in HTGRs will be limited (design-basis accident control). The cause of the tube ruptures was not identified. Placement of the SG relative to the reactor core is important to minimize water ingress. In the AVR, the SG was above the core.

- (4) (AVR, 1971): The fuel oil line broke on a control accelerator for one of the two main steam turbine shutoffs in the AVR [49]. The fuel leak led to a fire, which spread rapidly until the turbine emergency switch was turned off. The fire was extinguished after 15 minutes. The instruments and electrical installation of the engine console in front of the turbine were destroyed. Figure 15 shows the fire damage to the control stand of the turbine. The problem was traced to the fabrication of a clamped screw assembly. The bore of the union nut had sharp edges and had been pushed into the pipe, causing a small starter flaw. Line vibrations and thermal fatigue subsequently grew this crack to a critical size, leading to rupture. To prevent similar failures in the future, several design modifications were made. Among other design changes, welded-on cone couplings replaced the former couplings, and more elastic pipe loops replaced the rigid pipe connections to reduce vibrations.

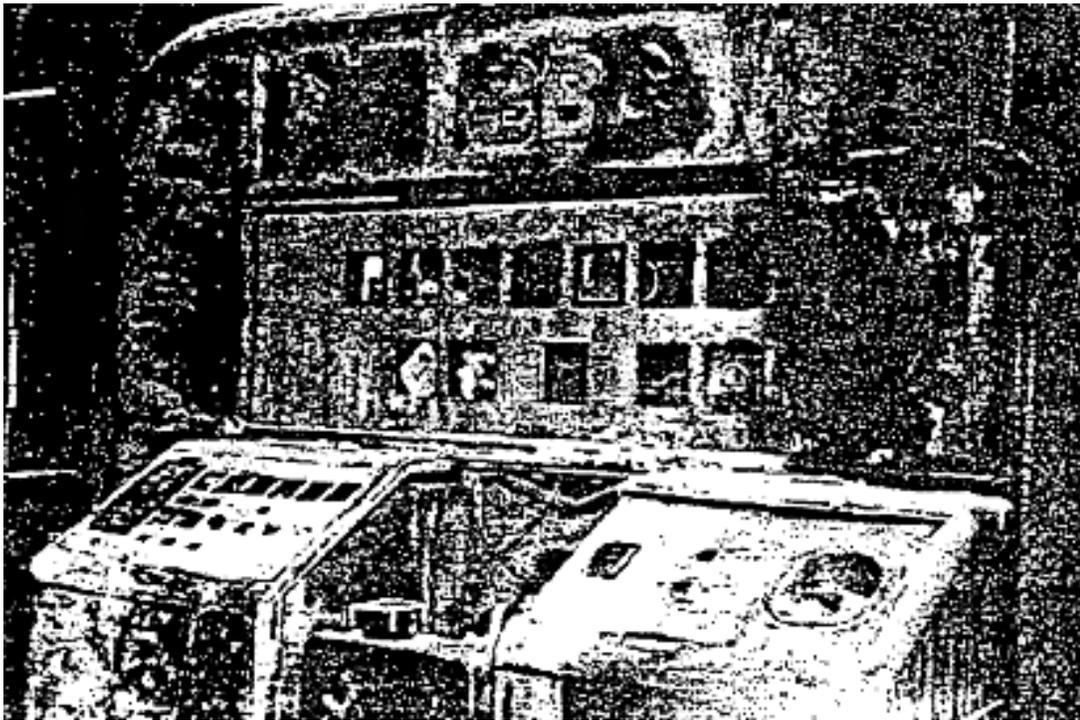


Figure 15 Fire damage in the turbine building (from [49 Figure 10.1-1])

- (5) (AVR, 1978): A serious water leak occurred in AVR [49]. Elevated moisture levels were observed earlier in the helium coolant gas. The source of the leakage in the AVR pebble bed reactor was difficult to find because there were many possible leak sources, including compressor leaks, leaks in the cooling unit of the gas purification system, leaks in the SGs, and leaks in cooling units of the coolant gas blowers. This particular leak was a 3-square-millimeter crack in pipe 8 of Superheater System 1. Reports are ambiguous, but the leak occurred either at a weld repair or at a pipe bend [49]. The cause of the crack was not specified, but creep fatigue ratcheting likely caused the crack. As a consequence, water sensors and drainage devices were installed. Lessons learned include (1) all containers, particularly the reactor vessels, must be equipped with drainage devices, and (2) U-shaped pipe lines are to be avoided. This incident caused an outage of about 8 months. The pipe materials were not specified.

- (6) (AVR, January 1979): After the SG incident in the AVR (Item 5 above), a problem occurred during startup in the coolant gas circulator system [49]. Water had leaked into the circulator blower area from the cooling water circuit for the cooling gas blowers. This was likely a fabrication issue. Other than this incident, the helium cooling gas blowers performed very well during plant operation; however, screws and bolted connections may be problematic in HTGRs. The materials were not specified. This incident demonstrates the need for proper fabrication and quality control.
- (7) (Dragon, early operation): After the first Dragon boiler tube failure in 1967, the reactor suffered serious waterside corrosion troubles. Shutdowns for tube plugging were frequent, and 13 complete boiler changes took place [50]. The planned water treatment was typical for a conventional once-through boiler. In 1963, a reappraisal of the safety criteria showed that a reactivity accident caused by partial flooding of the reactor core could potentially occur and that some degree of neutron poisoning of the cooling water was essential. The only suitable poison under alkaline conditions was lithium sulphate at a required concentration of 1.6 percent, which caused waterside corrosion. This incident demonstrates that careful material selection is necessary and that the steam side can indirectly create significant safety issues.

4.5 Primary Piping

- (1) (EVA-II⁶, general): The former 10-MW_t EVA-II test loop experienced two SG leaks caused by a strong radial jet flow of the entering hot helium. This led to increased thermal stresses on the tube material, causing cracks in the tube walls when water entered into the helium system [51].
- (2) (FSV, life end): The event that finally brought FSV operations to an end was the severe cracking of the Incoloy 800 (Alloy 800) SG superheater headers. Replacement of the headers was deemed too expensive to justify a plant restart, given the long history of operational problems at the plant. The header cracks were caused by vibration and thermal cycling of the header material, which had coarse grain sizes, making the header prone to cracking [37].
- (3) (AVR, December 1982): A leak occurred in a compensator in the piping lines between the cooling tower and the machine housing building in the AVR [49]. The compensators were installed to compensate for thermal expansion mismatch effects to control thermal expansion mismatch stresses. The compensator fretted against a support because of the thermal expansion of the line. The materials and temperatures were not specified. Because of a lack of instrumentation, the incident was noticed only when the basement of the machine housing filled with water almost to the ceiling. Thermal stress management is important in HTGR plants.
- (4) (Dragon, 1974): The helium coolant leak rate detected during operation reached 2 kg (4.4 lb) per day [52]. The leaks were eventually found in the SS pipework leading to the helium purification plant. The leaks caused by chloride corrosion were small pits and crevices in otherwise healthy lengths of pipe. All leaks occurred in narrow sections of the pipes that were wrapped with polyvinyl chloride (PVC) insulating tape during commissioning to mark the various flowpaths and components of the circuit. When the circuit was approximately 80 to 120 degrees C (176 to 248 degrees F), the innermost

⁶ also referred to as the Single Tube Test Facility

layer of PVC tape decomposed and trapped gaseous hydrochloric acid under the outer, but still intact, layers of tape. (The PVC tape on pipe sections with temperatures of less than 80 degrees C (176 degrees F) remained stable; however, the PVC tape on pipe sections with temperatures above 120 degrees C (248 or degrees F) cracked and fell off.) After discovery of the leaks, all accessible SS pipework was inspected, and more than 200 tape markings were removed. All sections of SS pipework with tape markings were cleaned, and sections operating above 80 degrees C (176 degrees F) were replaced. This resulted in a 10-fold reduction of unaccounted helium losses, to 0.2 kg (0.44 lb) per day. This demonstrates the need for proper quality control during initial fabrication.

4.6 Severe Accident

- (1) (HTTR, 2005): The reactivity control system in Japan's HTTR consists of two separate systems: (1) a control rod system and (2) the reserve shutdown system (RSS), which consists of boron. Pellets are released when necessary [45, 46, 53]. If the control rod system fails to engage, the RSS system drops the pellets into the core to shut down the reactor. During a periodic test of the RSS, a problem was observed. The oil seal at the top of the drive motor was distorted, and a spring had failed. A factory fabrication error of the seal caused the problem.
- (2) (HTTR, general): The Great East Japan Earthquake disaster of March 2011 caused accelerations in the HTTR greater than design values [45]. The HTTR was shut down at the time for a periodic inspection that found several problems, including sludge deposition in various components and a reduction in thickness of the combustor liner in an emergency gas turbine generator. These problems were caused by a long-time wave from the strong earthquake and were exacerbated by vibrations of fuel tanks and the long-time operation of the emergency gas turbine generators after the event. Future HTGRs would benefit by using this experience as an important lesson for protecting against the effects of earthquakes. In particular, backup systems must be properly designed to handle overloads as well. The comprehensive report from the Japan Atomic Energy Research Institute [53] contains details of the HTTR design but does not provide OpE.

4.7 Other Topics

The following other OpE topics do not necessarily fall into the component categories discussed above and cover more general OpE:

- (1) (AVR, September 1971): While regenerating the cation exchanger of the condensate desalination system, hydrochloric acid leaked into the gravel bed filter at the AVR [49]. This occurred because two shutoff valves in front of the gravel bed filter leaked, and operator error caused the relief valve between the two shutoff valves to close. To avoid corrosion damage, the power plant was shut down and the affected regions, including the SG, were rinsed numerous times with deionized water.
- (2) (HTTR, 2006): During a periodic inspection of the HTTR emergency gas turbine generators [45], small cracks were found on one of the three turbine blade nozzles. The cracks were attributed to fatigue from the long operating time of the emergency gas turbine generator. Emergency generators used in other industries do not operate as long, and this generator had 1,000 hours of operation. Therefore, generators that are

not specifically designed for nuclear operations should be avoided, and design-specific backup generators for the nuclear industry should be used.

- (3) Leak-before-break (LBB) issues are important for HTGR designs. Zhang et al. [54] summarizes LBB considerations for HTR-10 that should largely apply to next-generation HTGRs. Section 3 of Reference [35] provides a good overview of LBB considerations for HTGRs. As with LWRs, if LBB cannot be satisfied in an HTGR, a piping break must be postulated, and appropriate protection against the dynamic effects of the break must be provided for the safety-related structures, systems, and components. LBB analyses allow for the elimination of pipe whip restraints, jet impingement barriers, and other safety features. Formally, the LBB methodology could not be applied to piping that was degraded by an active degradation mechanism such as stress-corrosion cracking or creep cracking. The development of the xLPR code to place the active degradation mechanism of primary water stress-corrosion cracking in bimetallic welds in LWRs into a probabilistic framework may permit LBB considerations to bypass this requirement. A similar development could also be made for HTGR active degradation mechanisms such as creep, creep fatigue, and corrosion. Enhancing the xLPR code⁷ to account for these degradation mechanisms is a logical step for the assessment of HTGR licenses. (See References [35, 54] for more details on the LBB application in HTGRs.) Finally, application of LBB requires robust leak detection methods.
- (4) The white paper on Next Generation Nuclear Plant (NGNP) high-temperature materials [55] provides an updated materials summary. The white paper discusses the candidate materials for the NGNP and the material applicable to the different HTGR systems. In addition, this summary includes composite applications and ceramic materials. This report also provides an update on the applicability of the ASME Boiler and Pressure Vessel Code (B&PVC) to HTGRs.

4.8 Design Considerations

- (1) The designers of the Dragon [52] frequently expressed concerns about prolonged contact between metal components in a high-temperature and high-purity helium environment causing a self-welding seizure; however, this concern did not materialize. Even under load and operating at a temperature of approximately 650 degrees C (1,202 degrees F), frictional force was the only resistance to disengagement at the Nimonic control rod shield tube interface with the Monel 400 reflector head assemblies.
- (2) Power plant startup was carefully controlled in the AVR plant. To prevent high tensile stresses in the graphite during startup, which heats from the outside to the core, the hot-gas temperature could only be increased by 3 degrees C (37.4 degrees F) per minute. Reference [49] summarizes the procedures used to control temperature during startup (beginning on page 454) and summarizes the plant procedures used to control thermal gradients in the core graphite during reactor shutdown (beginning on page 456).
- (3) The coolant in HTGRs can result in flow-induced vibrations driven by energized gas flow. This flow across heat exchanger tubes, reactor internals, and other flexible locations can lead to damage and fatigue cracks if it is not properly managed. This problem also occurred with the Magnox and advanced gas-cooled reactors (AGRs) with carbon

⁷ A companion code, PROMETHEUS, might be more convenient for this purpose than the xLPR code.

dioxide coolant [37], and it can also affect helium-cooled reactors. Indeed, damage and crack development caused by unanticipated or incorrect loads have led to fatigue problems in both Magnox and AGR these reactors.

- (4) Graphite dust is produced in HTGRs from the contact and movement of graphite blocks. Graphite dust production is exacerbated by thermal gradients, coolant flow, vibration, and irradiation-induced swelling [37]. The generation of graphite dust is a particular problem in pebble reactors. This dust can be transported to other locations (e.g., the heat exchangers), reducing efficiency and possibly causing local “hot” spots that can lead to damage. Additionally, the graphite dust has a high propensity to absorb fission products, creating a radioactive graphite aerosol in the primary circuit. HTGR designers should seek to minimize the production of graphite dust and have methods (e.g., a filtering system) to remove the graphite dust from the helium loop.
- (5) Control rod lubrication in HTGRs is challenging. At HTGR temperatures, conventional lubricants (such as oils) cannot be used because of the high-temperature and the high-radiation environment [37]. Dry lubricants can be used; however, tribology studies of dry lubricants and their life performance is necessary and important to materials and component integrity.
- (6) Oil leaks in HTGRs were a persistent problem. The Peach Bottom HTGR experienced many leaks in the hydraulic components that were attributed to manufacturing defects in the sealing of surfaces [37]. Many of these leaks occurred in the compressor circulators. HTGRs should eliminate the use of hydraulics and oils as much as possible to prevent oil ingress from hydraulic components. This issue should be a source of scrutiny in licensing.
- (7) Research focused on the advancement of heat exchangers for HTGRs recommends the use of Alloy 617 for temperatures above 850 degrees C (1,562 degrees F) and Alloy 800H for temperatures below 850 degrees C (1,562 degrees F) [37, 56]. This research demonstrated the stress rupture behaviors of these alloys and how carburization or decarburization can occur depending on the materials used and on the heat exchanger flow rates. The reactions between the metal and the impurities in the helium coolant can lead to carburization or decarburization, depending on the gas kinetics. Carburization reduces low temperature ductility and decarburization leads to a severe loss in creep strength, but control of the impurity contents can keep these effects within acceptable limits [37]. Specific high performance alloys are needed for the temperatures and environments experienced in HTGR reactor components. Alloy 617 and Alloy 800H are being considered for use in the NGNP IHX.⁸
- (8) The preferred candidate materials for the hot duct are Alloy 617 or Alloy 800H, depending on the temperature [37, 56]. For Alloy 617, one important issue to consider is its high cobalt content. The activation of the cobalt could pose a problem in future HTGRs; however, in Germany’s HTGRs, cobalt was not present in the oxide scale; therefore, radioactive cobalt could not enter the hot-gas circuit even if the oxide spalled off.
- (9) The SG in Peach Bottom [37] was constructed of carbon steel, but the tubes were made from Alloy 800H. The Alloy 800H incurred significant age-hardening damage during operation but retained its ductility. These results were found acceptable and accurately

⁸ ASME is presently adding Alloy 617 to the ASME B&PVC.

predicted. One study showed that Alloy 800H has a lower oxidation rate if it is hot rolled instead of cold rolled [37, Section 4.1.5]. Germany's THTR used cold-rolled Alloy 800, Grade 1, for the SG [37, Section 4.1.3]. The THTR header design required inelastic computational analyses (elastic-plastic creep) of 30 different transients with temperatures of about 560 degrees C (1,040 degrees F) [37]. The ASME B&PVC has recently added a simplified nonlinear computational analysis procedure for high-temperature designs. Although the THTR only operated for 6 years, Alloy 800, Grade 1, performed well.

- (10) The United Kingdom's AGR SGs were predominantly helical coil and used carbon dioxide coolant. The AGR had failures associated with dynamic stress generated by the acoustic vibration (noise) from the gas circulators. SG designs must properly consider acoustic vibrations.

4.9 Lessons Learned (High-Temperature Gas-Cooled Reactors)

- (1) High tensile stresses caused by irradiation swelling at high temperatures can develop in the fuel elements, leading to the cracking of graphite fuel elements.
- (2) Calculating HTGR core temperatures is challenging, particularly for pebble bed designs. Although it is easy to attribute the overheating in the AVR to limitations of computational codes in the 1980s, the core temperature of the prismatic HTTR has also been underestimated. Inaccurate core temperatures have led to thermal fatigue and fuel failure in pebble bed HTGRs.
- (3) HTGRs must account for all sources of possible corrosion, and corresponding materials and components must be designed appropriately. Early designers focused on the impact of moisture ingress and consequent graphite oxidation; however, OpE has demonstrated moisture ingress can also lead to structural component failures. Future HTGRs must be designed to consider all aspects of potential moisture ingress and incorporate methods for removing moisture from the core and primary loop.
- (4) HTGRs must consider the accumulation of cycles during testing. Failure to account for these additional cycles led to fatigue failures in both Peach Bottom and the AVR.
- (5) Management of thermal stresses is important in HTGRs because thermal expansion stresses can cause large loads and creep.
- (6) Corrosion of the prestressed concrete tendons could weaken the integrity of the reactor vessel and must be considered.
- (7) Abnormal abrasion in helium coolant compressors can degrade the performance of piston-ring seals and cause leakage. Compressors in HTGRs must be carefully chosen.
- (8) HTGR should be designed to minimize sources of graphite dust (e.g., fretting) and should include filters or other mitigating measures to address graphite dust.
- (9) Coarse-grained alloys are used for improved creep resistance, but such alloys are more vulnerable to cracking. Control of alloy grain sizes should be considered because alloys with excessive grain sizes may have insufficient toughness.

- (10) Oil-based lubricants should be avoided in HTGRs. Lubricants have repeatedly leaked into the primary loop in HTGRs, and the high temperatures of the HTGRs are sufficient to ignite these oils elsewhere in the plant, even when they are not associated with the primary loop.
- (11) The design lesson from the Dragon experience is to ensure that the impact of SG tube ruptures in HTGRs will be limited (design-basis accident control).
- (12) Lessons learned from AVR include (1) all HTGR vessels, particularly the reactor vessels, must be equipped with drainage devices, and (2) “U-shaped” pipe lines are to be avoided.
- (13) HTGRs need to ensure the structural integrity of the RPV and the connecting vessels, especially under low helium flow and loss-of-forced convection conditions, because buckling may occur.
- (14) Backup systems must be properly designed to handle overloads and system upsets, such as seismic loads.

5 SUMMARY

5.1 Sodium-Cooled Fast Reactors

SFR OpE is extensive and has resulted in the following conclusions with regard to materials and structural performance:

- As discussed in Reference [57], EBR-II never experienced a sodium/water reaction because of the use of double-walled (concentric tubes brazed or swaged together) SG tubes. This has not been the case with many other SFRs.
- Corrosion of immersed SS components is not a concern if sodium purity is maintained. No significant corrosion of materials in the sodium circuits of BN-10 occurred during its 44 years of operation; therefore, a 60 year life of the piping circuits in future SFR designs is possible.
- Weld design and quality control are critically important. Residual weld stresses, excess weld metal, and weld constraints should be minimized, and direct tube-to-tube-plate welds should be avoided entirely in SFR. Lowering the threshold for quality in welds and secondary loops has resulted in operational problems.
- Reheat cracking is a concern in SFRs in components operating at high temperatures. In particular, weld repairs should be carefully managed because they may give rise to very high tensile residual stresses. Some types of austenitic SS (e.g., Type 321) are significantly more prone to reheat cracking than other austenitic SS.
- Stresses induced by thermal expansion, particularly in areas of constraint, must be carefully considered. The stresses have often been the source of structural integrity issues in SFR operation.
- Thermal fatigue (i.e., thermal striping) caused by mixing sodium flows at different temperatures is a significant issue in SFRs.

- The management of startup and cooldown transients in SFRs to control vibration, thermal expansion loads, and possible fatigue issue is important.
- Shrink-fit parts should be avoided because they could loosen during thermal transients.
- Electromagnetic pumps have operated reliably.
- Oil-based lubricants should be avoided in SFRs.
- Possible valve failures (all system valves, especially those operating at high temperature) are a concern for SFRs. Valve reliability under operating conditions should be accurately determined.
- Austenitic steels are unsuitable for SFR SGs because of the potential for caustic stress-corrosion damage following even small leaks.
- Testing should confirm the chemical compatibility between molten sodium and insulation material.
- Secondary measurement devices (e.g., thermocouples) must be properly designed to prevent leaks. Flow-induced vibrations and complex fluid flows in these areas can cause failure and sodium leaks.
- Accurate detection methods of corrosion and leaks are necessary, particularly in regions coated with insulation. The design phase should consider sensor placement and reliability under operating conditions. Inadequate or unreliable leak detection systems have resulted in extensive shutdowns because of sodium contamination and excessive sodium leaks with consequent fires.
- The licensing process needs to scrutinize seismic and external dynamic loading events of SFRs. During an emergency shutdown (scram), the IHX may experience thermal shock caused by the influx of cold sodium. This could lead to buckling and structural issues amplified by an external loading.

5.2 High-Temperature Gas-Cooled Reactors

HTGR reactor OpE is extensive and has resulted in the following conclusions with regard to materials and structural performance:

- An accurate prediction of core temperatures in HTGRs is technically challenging. Even for more recent designs (e.g., the HTTR), core temperatures have exceeded anticipated design temperatures.
- Moisture ingress and leakage events are a reoccurring problem with HTGRs. HTGRs should be designed to accommodate and mitigate moisture ingress.
- Failures within the SG could introduce water into the primary loop and introduce the potential for unanticipated reactivity.
- Management of thermal stresses is important in HTGRs because thermal expansion stresses can cause large loads and creep.

- HTGRs must consider the accumulation of cycles during testing. Failure to account for these additional cycles led to fatigue failures in both Peach Bottom and the AVR.
- Pumps, seals, and compressors have a history of poor reliability in HTGRs. The design and testing of these components should be well scrutinized.
- HTGR should be designed to minimize sources of graphite dust (e.g., fretting) and should include filters or other mitigating measures to address graphite dust.
- Coarse-grained alloys are used for improved creep resistance, but they are more vulnerable to cracking. Control of alloy grain sizes should be considered because alloys with excessive grain size may have insufficient toughness.
- Oil-based lubricants should be avoided in HTGRs.
- HTGRs need to ensure the structural integrity of the RPV and the connecting vessels, especially under low helium flow and loss-of-forced convection conditions because buckling may occur.
- Backup systems must be properly designed to handle overloads and system upsets, such as seismic loads.

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