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Review of Safety-Related SFR Experimental and Operational Experience in the United States

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Safety Aspects of Sodium Cooled Fast Reactors
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U.S. Fast Reactors Built and Operated

Facility	Location	Mission	Dates	Power (MWt)
EBR-I	Idaho	R&D	1951-1963	1.4
EBR-II	Idaho	R&D	1963-1994	62.5
Fermi-1	Michigan	Power	1963-1972	200
SEFOR	Arkansas	Safety Test	1969-1972	20
FFTF	Washington	Fuel & Material Test	1980-1992	400

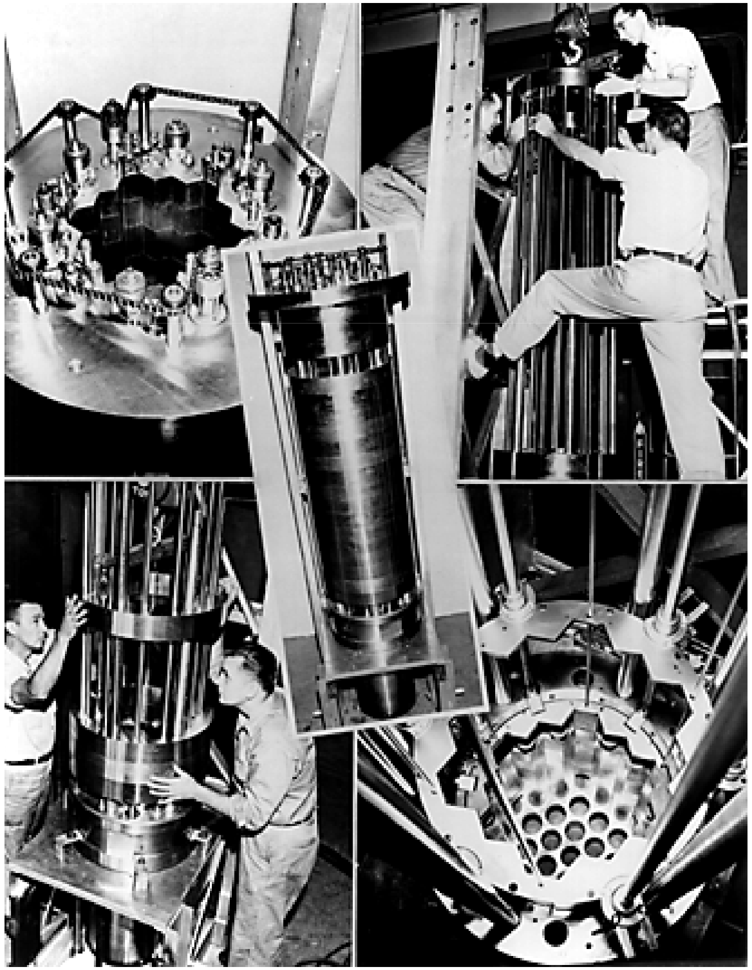
- Experimental Breeder Reactor-I (EBR-I) was built to demonstrate fuel breeding
- Experimental Breeder Reactor-II (EBR-II) was built to demonstrate closure of the metallic fuel cycle and recycling of reactor fuel
- Fermi-1 was built as a metallic uranium-fueled reactor on a utility grid
- Southwest Experimental Fast Oxide Reactor (SEFOR) was built to demonstrate the safety properties of the Doppler feedback for oxide fuel
- Fast Flux Test Facility (FFTF) was built to test fuel and cladding materials for the Liquid Metal Fast Breeder Reactor (LMFBR) program

Experimental Breeder Reactor-1 (EBR-I)

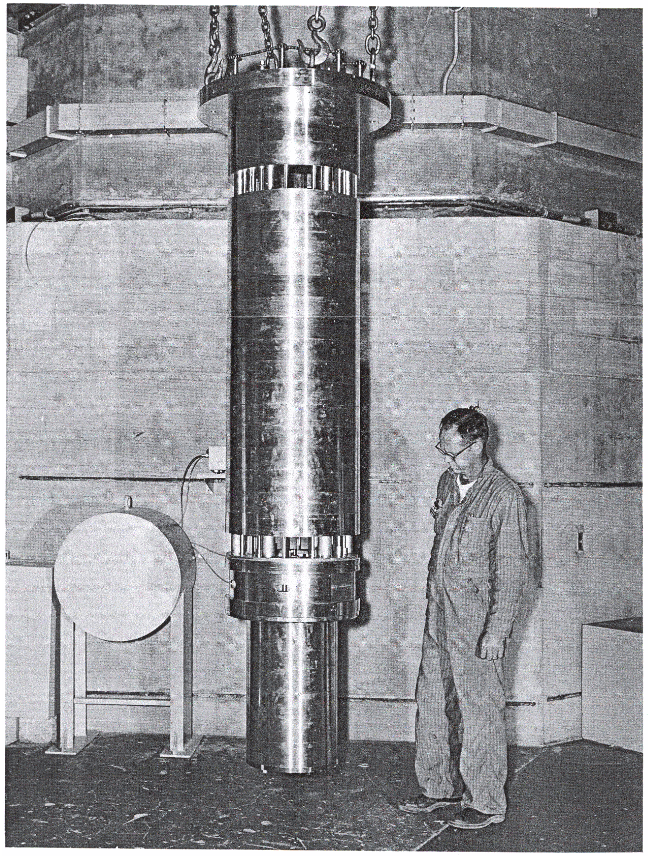
- **EBR-I was operated with four core designs**
  - Pu breeding demonstrated by February, 1952
  - Testing for reactor physics, fluid dynamics, and power generation
  - Generated 200 kW of electricity for NRTS
- **NaK cooled, U-235 (94% enriched) metallic fuel in stainless steel cladding**
  - 217 pin locations, 0.384 in. OD, 4.25 in. core height
  - 227^{o}C inlet, 316^{o}C outlet, 20 psig
  - Mark III core used hexagonal tubes and wire wraps for fuel pin positioning
  - Mark IV core used Pu fuel (1962)
- **In November, 1955, during a test to investigate a positive component of the power coefficient, a power excursion resulted in substantial fuel melting and fuel pin failures**
  - Importance of controlling fuel geometry was identified
  - Core was removed, replaced, and operation continued through 1963

EBR-1 Reactor

Assembly of EBR-I Core



Inner Tank Assembly



(Reactor Shield in Background)

Experimental Breeder Reactor-II (EBR-II)

- **From 1964 through 1994, EBR-II operated as a prototype breeder power station demonstrating fuel cycle closure**
  - Sodium cooled, 371°C inlet, 473°C outlet, 47 psig
  - Fuel pins 0.17 in. OD, 13.5 in. core height; metal fuel in SS cladding
- **First fuel processed in Fuel Cycle Facility in September 1964; recycled fuel irradiation in April 1965**
- **Mission changed to irradiation testing in 1969 to support FFTF and CRBRP oxide fuel development**
- **Integral Fast Reactor (IFR) program began in mid 1980's**
  - Testing and demonstration of high burnup metallic fuels
  - Shutdown Heat Removal Test series 1984-86; natural circulation decay heat removal and passive shutdown in ATWS events (unprotected loss-of-flow and loss-of-heat-sink)
- **Operated through 1994**

EBR-II Site

EBR-II Design

- **Provide demonstration of the fast breeder closed fuel cycle, then an experimental environment for fuel and transient testing**
- **62.5 MWt, metallic fuel (63% U-235, U/Pu), two primary coolant loops, primary system with a large sodium pool**
- **Containment building**
- **Two independent reactor shutdown systems**
- **Forced and natural convection decay heat removal through three independent loops**
  - Pony motors on primary and secondary pumps
  - One loop ‘hardened’ for seismic and tornado effects
- **Core physics and structural design for negative power and temperature reactivity feedbacks**

Fermi-I

- **200 MWt power station located on the western shore of Lake Erie south of Detroit**
  - Critical August 1963, first power August 1966
  - Sodium cooled, 288^{o}C inlet, 427^{o}C outlet, 120 psia
  - Metallic fuel, Zr cladding 0.158 in. OD, 31 in. height, square pin pitch
- **Subassembly flow blockage and fuel melting accident during power ascension on October 5, 1966**
  - Plates were added late in the design and construction phase to function as a core-spreader in case of a core melt accident
    - One plate broke loose and blocked some subassembly inlets
    - One assembly almost completely blocked, another mostly blocked
    - Substantial fuel melting and relocation in two subassemblies
    - Reactor was safety shut down
  - Metallic fuel core removed and replaced with an oxide core; full power 1969
- **Operation ceased in 1972**

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Fermi-I Site

Fast Flux Test Facility (FFTF)

- **FFTF was a fuels and materials test facility for the U.S. breeder reactor program to support technology development for the Clinch River Breeder Reactor (CRBRP)**
  - 400 MWt, sodium cooled, 360^{o}C inlet, 527^{o}C outlet
  - MOX fueled, SS cladding, 0.23 in. OD pin: 217 pin fuel assembly
- **First critical February 1980, full power December 1980, shutdown December 1993**
  - Provided verification of the CRBRP fuel design
  - Investigation of advanced, low-swelling fuel cladding materials
  - Verification of large-scale component designs (pumps, heat exchangers)
- **Mission extension: safety testing**
  - Natural circulation shutdown heat removal
  - Passive power reduction in unprotected loss-of-flow sequence

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FFTF Site – Hanford, Washington

FFTF Design

- Provided a prototypic LMFBR operating environment for testing and development of fuel, cladding, materials, and components
- 400 MWt, loop-type plant, MOX fuel (22% and 27% Pu), three loop primary system, three intermediate sodium loops to air dump heat exchangers
- 10 psi steel containment
- Two independent reactor shutdown systems
- Forced and natural convection decay heat removal through three independent loops
  - Pony motors on primary and secondary pumps
  - One loop 'hardened' for seismic and tornado effects
- Core physics and structural design for negative power and temperature reactivity feedbacks
  - Substantial program on core restraint to provide favorable feedback

U.S. Fast Reactor Safety Development

- **However, even with defense-in-depth measures and natural circulation decay heat cooling capabilities, accidents with severe consequences dominated licensing discussions for later projects such as CRBR**
  - Even though probability of occurrence was very low, about 10^{-5} per reactor year or less, usually associated with failure of the reactor scram systems (and considered to be beyond the design basis in licensing), the consequences were potentially very large, posing a risk to the public
  - Many of the safety-related issues were unresolved at the time the project was ended
- **During the 1980's and 1990's, efforts were undertaken to further improve approaches to safety, including the development of the concept of "inherent safety" to lower the probability of severe accident consequences**

U.S. SFR Safety Experience

- **The US reactor development program has demonstrated that liquid sodium metal cooling in an SFR contributes to excellent safety performance**
  - Excellent heat removal and heat transport characteristics
  - Natural circulation decay heat removal
  - Passive reactor power reduction in beyond-design-basis accidents
- **Core melting accidents have shown that safe shutdown of an SFR is possible without severe consequences**
  - Metallic fuel is compatible with liquid sodium
  - Accident progression can be safely terminated
  - Reactors were refueled and operated after the accidents
- **ANS has started development of ANS Standard 54.1, "Nuclear Safety Criteria and Design Process for Sodium-Cooled Reactor Nuclear Power Plants"**
  - An update and revision of the ANS 54.1 Standard