Oak Ridge National Laboratory

Overview of Past U.S. SFR Operations Experience

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U.S. DEPARTMENT OF ENERGY

Outline

- Summary of U.S. Experience
  - Fermi-1
  - EBR-II
  - FFTF
- International experience
  - PFR
  - Phenix
  - Super-Phenix
  - BN-600
  - BN-350
  - Monju

Summary of U.S. Experience (1/3)

- First experimental fast reactor, EBR-I, used NaK coolant
  - Generated first electricity in 1951, confirmed breeding in 1953
  - Experienced partial meltdown in 1955 during a flow test in order to examine a reactivity anomaly
- Sodium Reactor Experiment (SRE)
  - 20MWt graphite-moderated experimental reactor by Atomics International in Moorpark, Ca (1957- 1964)
  - Produced power on grid
  - Due to leak in Tetralin cooled pump, sealant gland organic residue deposited on the cladding, impeding heat transfer
  - Partial meltdown of 13/45 fuel elements led to release of radioactivity to atmosphere in 1959
  - Returned to operation in 1960

Summary of U.S. Experience (2/3)

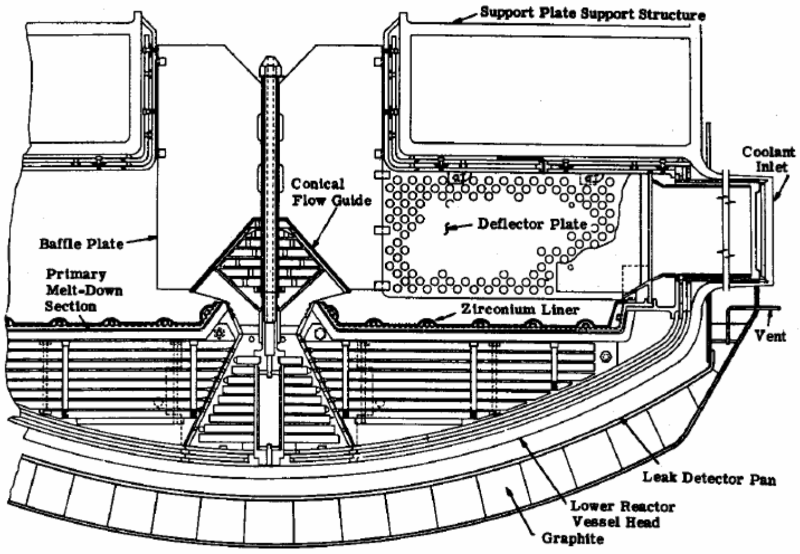
- Hallam Reactor
  - 75 MWe graphite-moderated plant by Nebraska Public Power (1963- 1964)
  - Suffered significant cladding failures due to stress corrosion cracking
  - Prohibitive cost to continue operations led to early shutdown by the utility
- Southwest Experimental Fast Oxide Reactor (SEFOR)
  - 20MWt reactor built to test inherent safety features of oxide fuel, operated by GE for the AEC in Strickler, AK (1969-1972)
  - Built to test inherent safety features of oxide-fueled sodium-cooled reactors
  - Confirmed that core thermal expansion provide negative reactivity feedback during a transient

Summary of U.S. Experience (3/3)

- EBR-II by ANL-West (at current INL site) (1961-1994)
  - Had some minor incidents during operation including a dropped fuel element during refueling
  - Otherwise, excellent operational history, including the safety tests that will be discussed later
- FFTF 400MWt test reactor at Hanford, WA (1980 – 1992)
  - Licensing review by NRC and ACRS, operational certification by DOE
  - Excellent operational history, including the safety tests that will be discussed later
- Fermi-1 as the only commercial SFR (1963-1972)
  - 63MWe metal fueled SFR owned by Power Reactor Development Corporation (Detroit Edison)
  - Flow blockage led to a melt down of 2 subassemblies

Fermi-1

- Concerns were raised during the safety review:
  - In 1956, ACRS issued a letter to the AEC:
    - "*The Committee as a whole was not satisfied with the evidence presented that no credible supercriticality accident resulting from meltdown could breach the container*."
    - ACRS recommended measures "*to insure subcritical distribution of melted fuel and to assure that free fall of core parts cannot reassemble a critical mass suddenly*."
  - In1956, AEC issued construction permit
  - In 1959, to address recriticality concern, six zirconium plates were welded to the core inlet plenum beneath the reactor core to "split" molten mass and assure subcriticality of resulting fragments.



Fermi core catcher with Zr plates

Fermi-1 Fuel Melting Accident

- Prior to accident, some high fuel assembly outlet thermocouple readings were observed during low-power operations
  - Assemblies with abnormal readings were moved to other positions under different thermocouples
  - Locations of the high temperature readings changed on each start-up but not in correlation with the assembly movements
  - Reactor operated at 100 MW(t) without problems
- On October 5, during a power ascension at 34 MWt, building radiation alarms sounded, indicating fuel damage and subsequent investigations revealed fuel melting in two adjacent assemblies
  - Another adjacent assembly was bent, with no internal damage
- A "foreign object" was found in the inlet plenum, which later proved to be a crumpled Zr plate from the melt-down section liner
  - The loose Zr plate had apparently been swept by flowing coolant to cover (partially or completely) the inlet nozzle of various assemblies during the multiple start-ups

Fermi-1 Fuel Melting Accident: Lessons Learned

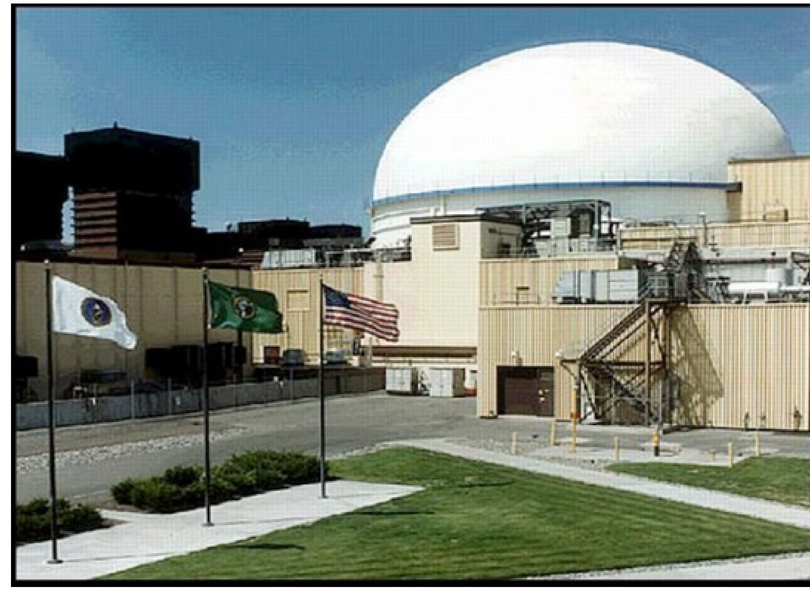
- Assembly inlet nozzle designs since Fermi-1 have included multiple coolant inlet passages so that complete external blockages are "impossible" by design
- Considerable research and testing of both external and internal blockages have been performed to understand and quantify the damage mechanisms and limits
- In the United States, the assembly blockage scenario (external and internal) has been addressed in the
  - assembly design (inlet flow diversity),
  - inlet plenum design (coolant flow distribution and assurance of assembly supply),
  - instrumentation design (detection by multiple thermocouples, delayed neutron detectors, gas tags), and
  - fuel handling equipment design (casks)
- Internationally, in some countries, the fuel assembly blockage scenario has become a design basis accident

More recent and relevant SFR operational experience is with FFTF and EBR II

EBR-II

- A metal-fueled pool-type SFR operating at 62.5 MW-thermal (20 MW-electric)
- 30 years of operation that significantly expanded the technology base for metallic fuel

FFTF

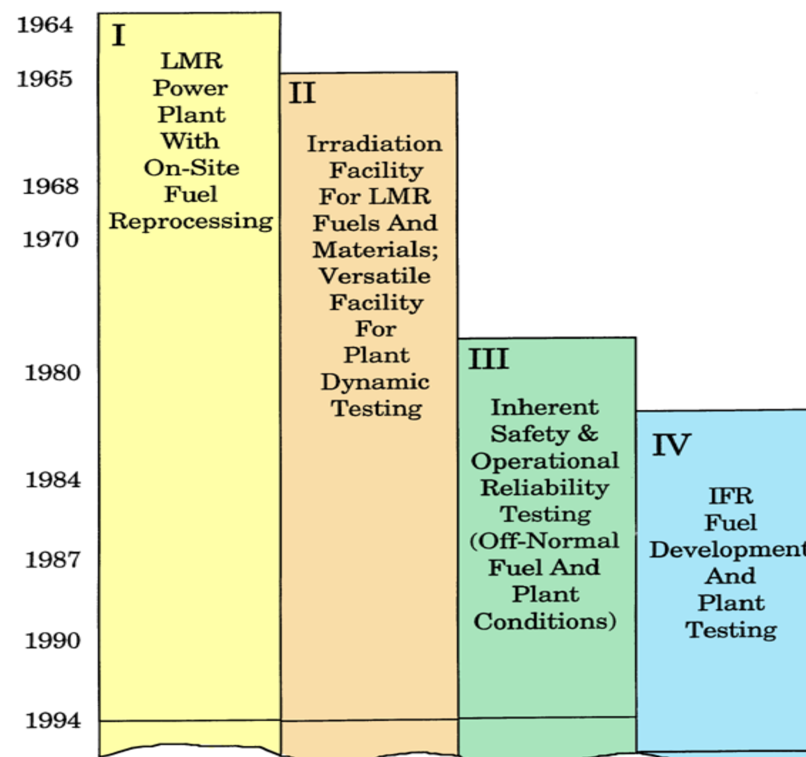


- A mixed-oxide-fueled loop-type SFR operating at 400 MW-thermal
- 10 years of operation that expanded the fuel irradiation experience from oxide to metal to nitride and carbide fuel forms

EBR-II Operation

Key design features

- Pool-type primary system with all PHTS system components in reactor vessel
- Massive heat sink with significant margins to temperature limits of SSCs during accidents
- Unique configuration allowing most of the sodium inventory to be at reactor inlet temperature and minimizing thermal stresses on major primary system components
- ~80% capacity factors achieved even with an aggressive testing program
- Very low exposure to personnel, excellent safety and sodium management record



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FFTF Operation

- Excellent operating record
  - Few incidents (cladding breaches)
    - Release of Cs to cover gas
      - Reactor
      - Spent fuel storage
    - One small leak (75 gal.) from aux, EM pump
- Low radiation exposure to operators
    - 1/100 of commercial power reactors
- Conducted passive safety tests
  - Described later in presentation
  - First use of Gas Expansion Modules (GEM) as a passive reactivity reduction device during ULOF

International Experience

- SFRs with international operation experience are
  - UK's Prototype Fast Reactor (PFR) was a 600 MWt pool-type SFR with 3 secondary loops and steam cycle (1974-94)
  - France's 255 MWe Phenix reactor (1973, 2013) was a pool-type SFR with peak thermal efficiency ~45% and 1.16 breeding ratio
  - France's 1200 MWe Super-Phenix was a pool-type commercial SFR based on Phenix experience (four loops)
  - Russia's BN-600 was a 600 MWe pool-type SFR with bottom supported reactor vessel
  - Kazakhstan's 350 MWe BN-350 reactor was a loop-type SFR as the first commercial LMR in Soviet Union
  - Japan's 400 MWt Monju reactor was also loop-type, but a prototype SFR

PFR Operation (1/2)

- Achieved closed fuel cycle in 1982 when loaded with Pu (highest burnup 17.6 atom-%)
  - Reprocessing plant treated 23 tonnes of oxide fuel and recovered over 3.5 tonnes of Pu through PUREX cycle
- Steam generator leaks (single wall U tubes with all welds above sodium level)
  - 37 gas-space leaks during 1974 to 1984 (33 in evaporators, 3 in superheaters, 1 in a reheater)
  - All the gas-space leaks originated at the welds between the tubes and the tubeplates
  - All austenitic tube bundles in evaporators were replaced
  - Also complete replacement of superheater/reheater tubes by 9 Cr–1 Mo ferritic steel
  - After these fixes, the plant achieved 80% load factor in 1994

PFR Operation (2/2)

- Large under-sodium leak caused failure of rupture disc and plant shutdown
  - 150 kg of water penetrated the sodium system
- Sodium aerosol deposits observed on control rods drives
- Fuel assembly bowing beyond 14 mm allowed limit was an issue
  - Needed to rotate assemblies to stay below 50 dpa limit
- Major oil leak into primary circuit experienced
- Cracks in air heat exchangers of decay heat rejection loops (10 tonnes of NaK in these loops)
- Sodium mixing and vibration problems

Phenix Operation

- Intermediate sodium leaks in the intermediate heat exchanger
- Sodium-water reaction in the steam generators
- Negative reactivity trips
- Cracking of welded joints on certain parts on the main intermediate loop pipes and some components
- Pump vibration due to faulty construction
- Sodium aerosol deposits jammed shutdown rod drives
- Extensive end-of-life testing program and post-shutdown examination
  - Some cracks in 304 SS steam generators dur to thermal striping
- During a renovation, significant upgrades on SG building for protection against large sodium fires
  - Separation by steel insulated walls and doors of firebreak zones
  - Reconstructed SG cells to resist a sodium fire with 1000°C for 30 min
  - Partitioning or housing of cable trays and building steel structures
  - Ventilation and multisampling detection circuits

Super-Phenix Operation

- 11 years of service but 4.5 years of operation (shutdown in 1996)
- 100 events logged, 16 sodium related
  - Very little problems with single wall steam generators
- Sodium leak from used fuel storage drum
  - Concern because safety vessel was made from the same material
  - Destructive examination of samples showed that cracking was due to pre-existenting of microcracks in zones of high hardness, embrittlement by hydrogen, and influence of residual welding stresses
- Major factor for early shutdown was the economics and the impact of Chernobyl accident

BN-600 Operation (1/2)

- Until 2004, 164,000 h of operation and ~88,000 GWh baseload electricity was generated
  - Could operate at 67% power with one loop shut down
  - In each loop, 8 separate steam generator modules consisting of separate evaporator, superheater, and reheater sections (total of 72 separate heat exchangers)
  - Seven superheater and four reheater leaks were experienced (mostly due to manufacturing faults)
    - Monitoring and confinement systems functioned enabling timely detection of leaks and controlled dumping of argon–hydrogen mixture without the maximum design pressure being exceeded
- Operating experience demonstrated capability for
  - Continuing operation of the loop with the leaked section disconnected without reducing the loop power
  - Disconnection of the affected section without shutting down the reactor and even without disconnecting the loop
  - Connecting up a repaired loop without shutting down the reactor

BN-600 Operation (2/2)

- Other major sodium leaks:
  - 800 kg leak from the cold trap pipe due to thermal stresses
  - 600L secondary sodium leak from a drain pipe (~30 kg burned, remaining retained and smothered with extinguishing powder)
  - In both leaks, the protective systems were effective, the damage was not extensive, and repairs were made quickly
  - Of the total 27 cases of leaks, only one case resulted in shut down
- Some fuel failures due to stress-induced corrosion of SS cladding
  - Because of fuel shuffling and rotation particularly in peripheral sections of core with 54 kW/m and 710°C
- Resulting changes in design:
  - Active core height increased from 75 to 100 cm to decrease max. linear heat rating to 47.2 kW/m
  - Reshuffling and rotation of the fuel assemblies were eliminated
  - Ferritic steel was used for duct design and boron-modified, cold-worked austenitic steel for cladding
  - Burnup reached 10 atom-% with 160 effective power days owing to advanced radiation-resistant steels

BN-350 Operation

- Originally designed for 1000 MWt bu latwer reduced to 750 MWt due to unsatisfactory SG operation
- Extended startup due to loss of integrity events in four evaporators (detected by the appearance of hydrogen in the gas plenum) where the SGs were filled with water
- Numerous cladding failures
  - Activity in primary sodium gamma dose on surface of sodium equipment 8.9 microsieverts, 80% of which was from cesium
- Load factor based on restricted operation was 85%
- Reactor vessel highly radioactive during shutdown, so reactor vessel pit is inaccessible

Monju Operation

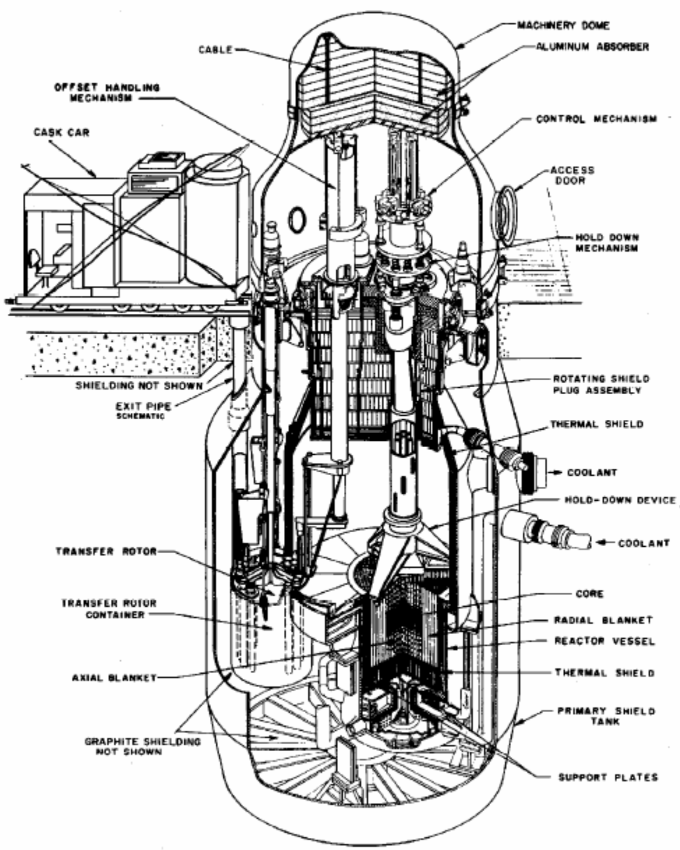
- In 1995, flow-induced-vibration in thermowell caused intermediate sodium leak spilling several hundred kilograms of sodium
  - Inspectors found 3 tons of solidified sodium
  - Sodium mixed with moisture and air creating aerosols and heat reaching several 100°C
  - Reactor was tripped manually and sodium was drained from the affected loop in ~two hours
- In 2010, a fuel assembly fell into the reactor vessel during a scheduled fuel replacement operation
  - Initial attempts to retrieve it was not succesful
  - Fallen device was successfully retrieved from the reactor vessel a year later
  - No damage was reported to the reactor vessel
- Tests for planned restart continued until 2013, but the facility was shutdown by Japanese government 2017

Backup Pages

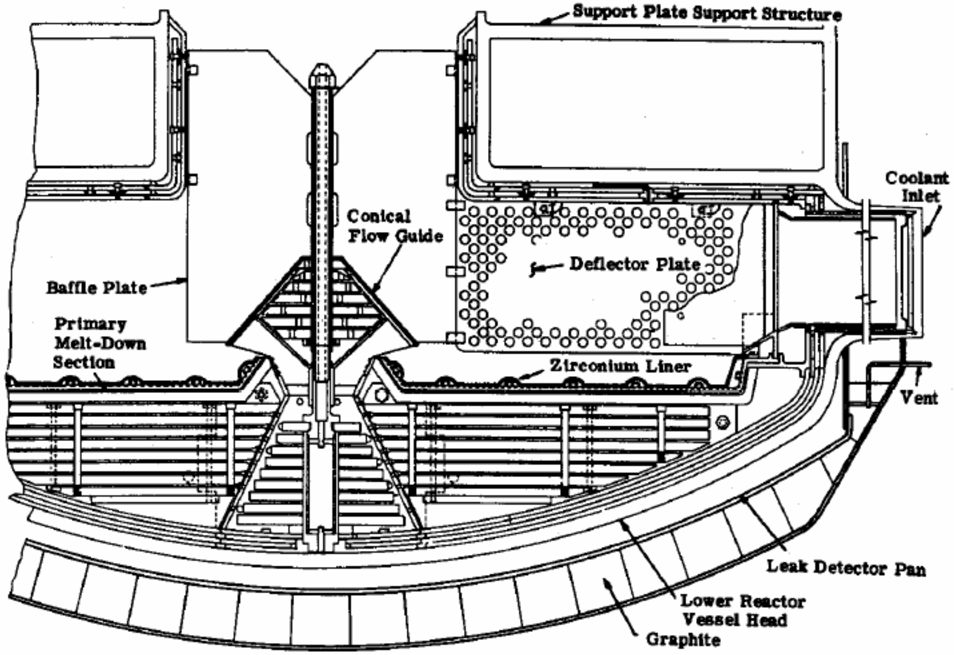
Fermi-1

- 63MWe (200MWt) metal fuel SFR owned by Power Reactor Development Corporation (Detroit Edison) 1963-1972
- Built before EBR II went critical (little experience with SFRs
- Much concern was raised about re-criticality during the safety review

Fermi-1 Reactor Vessel and Core Catcher

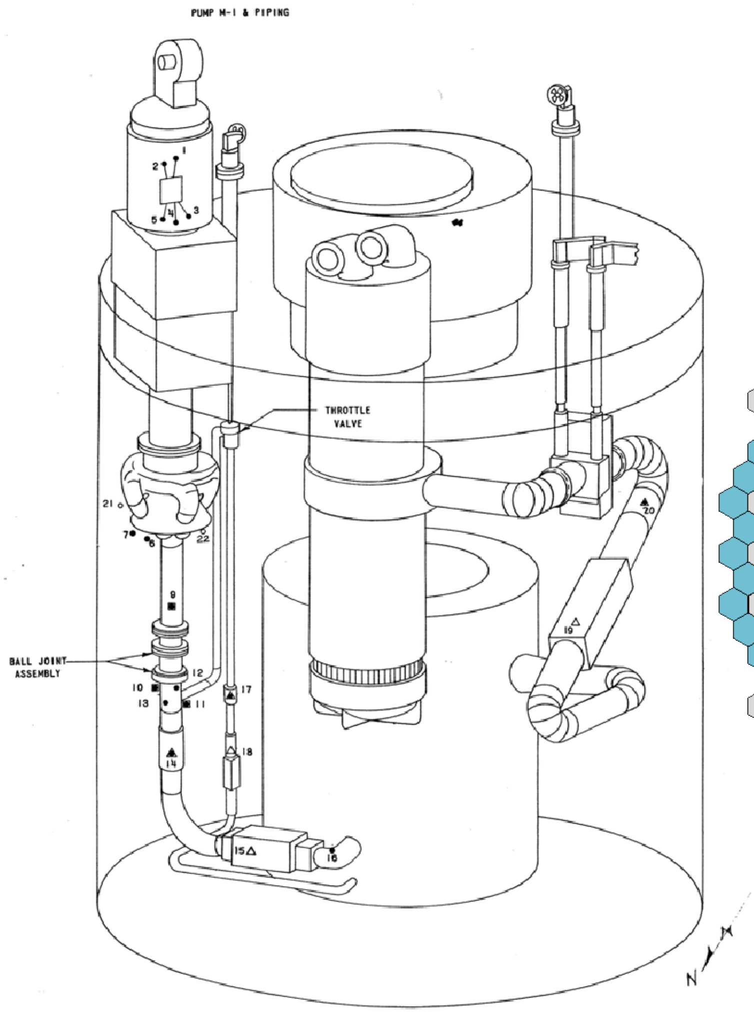


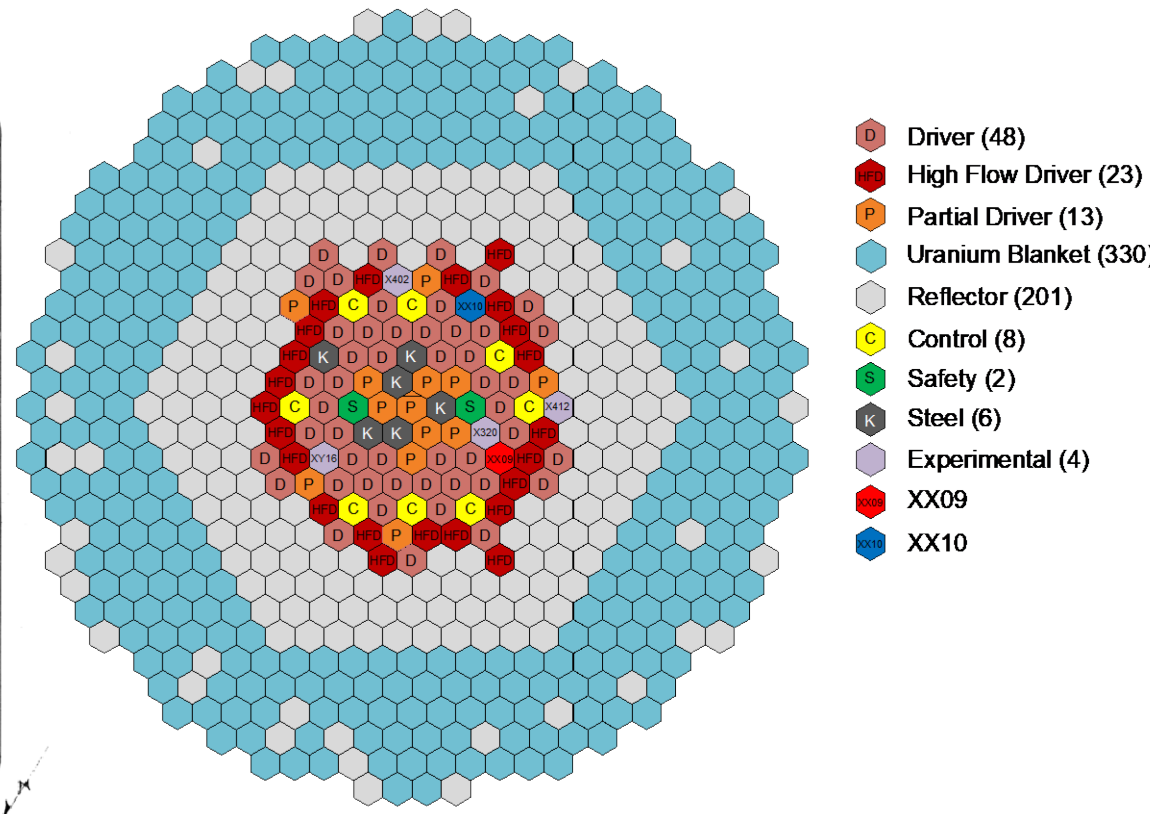
PERSPECTIVE VIEW OF REACTOR



EBR-II (cont.)

PHTS layout and reactor core



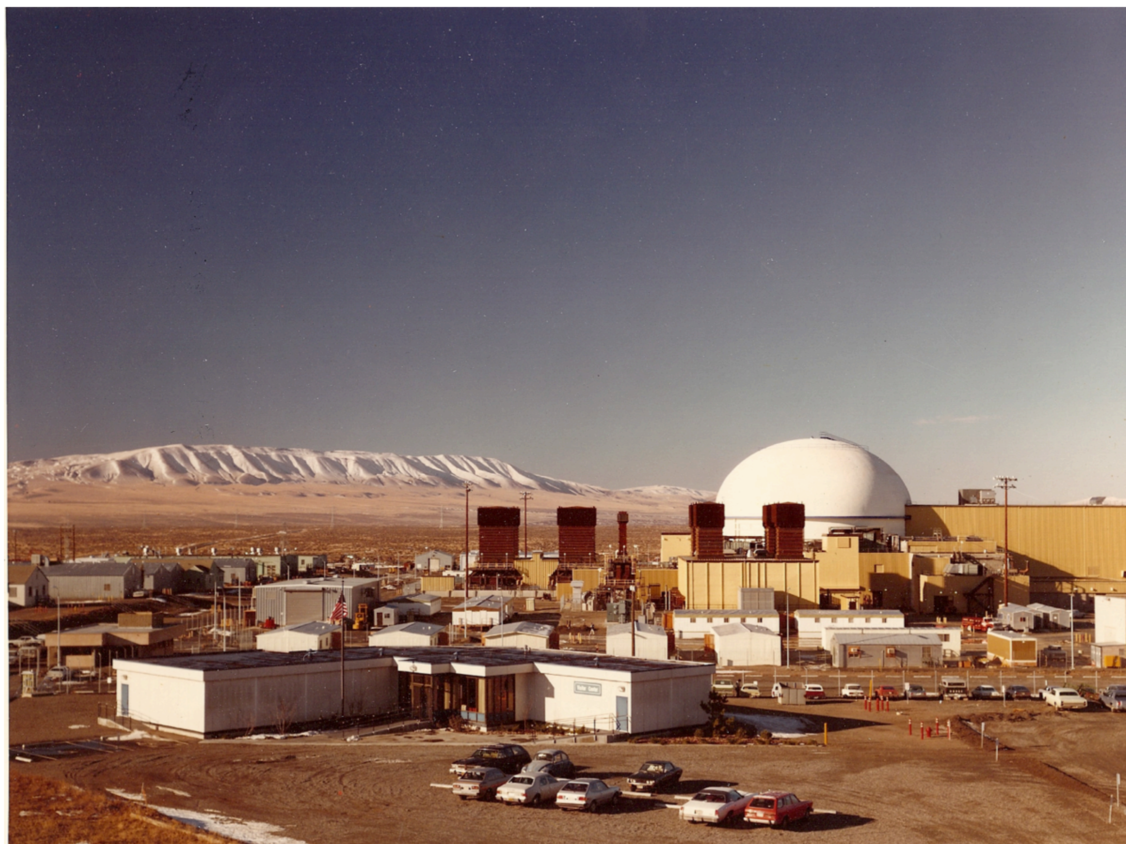


– Primary Sodium: 485 kg/s
– Inter. Sodium: 315 kg/s
– Sec. Steam: 32 kg/s

FFTF

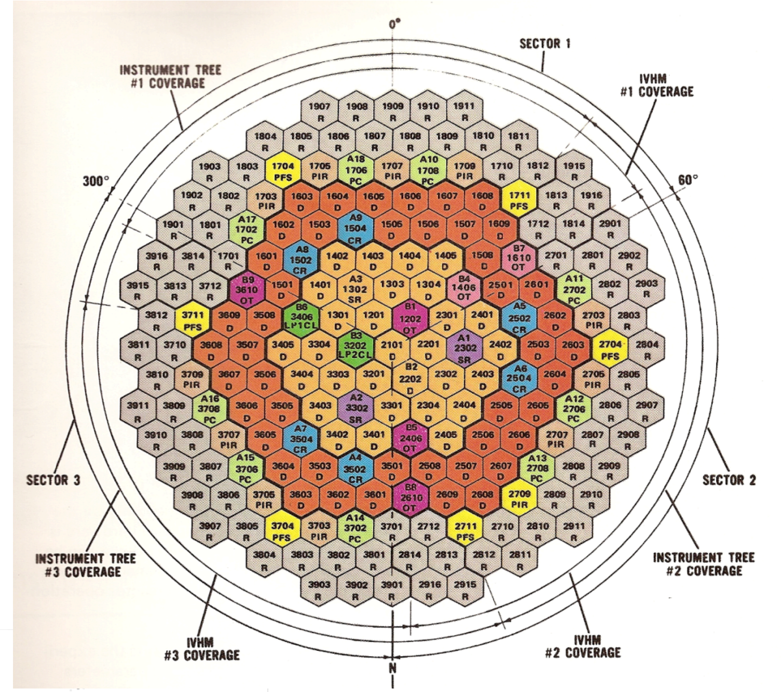
Operated at DOE's Hanford site as a test facility

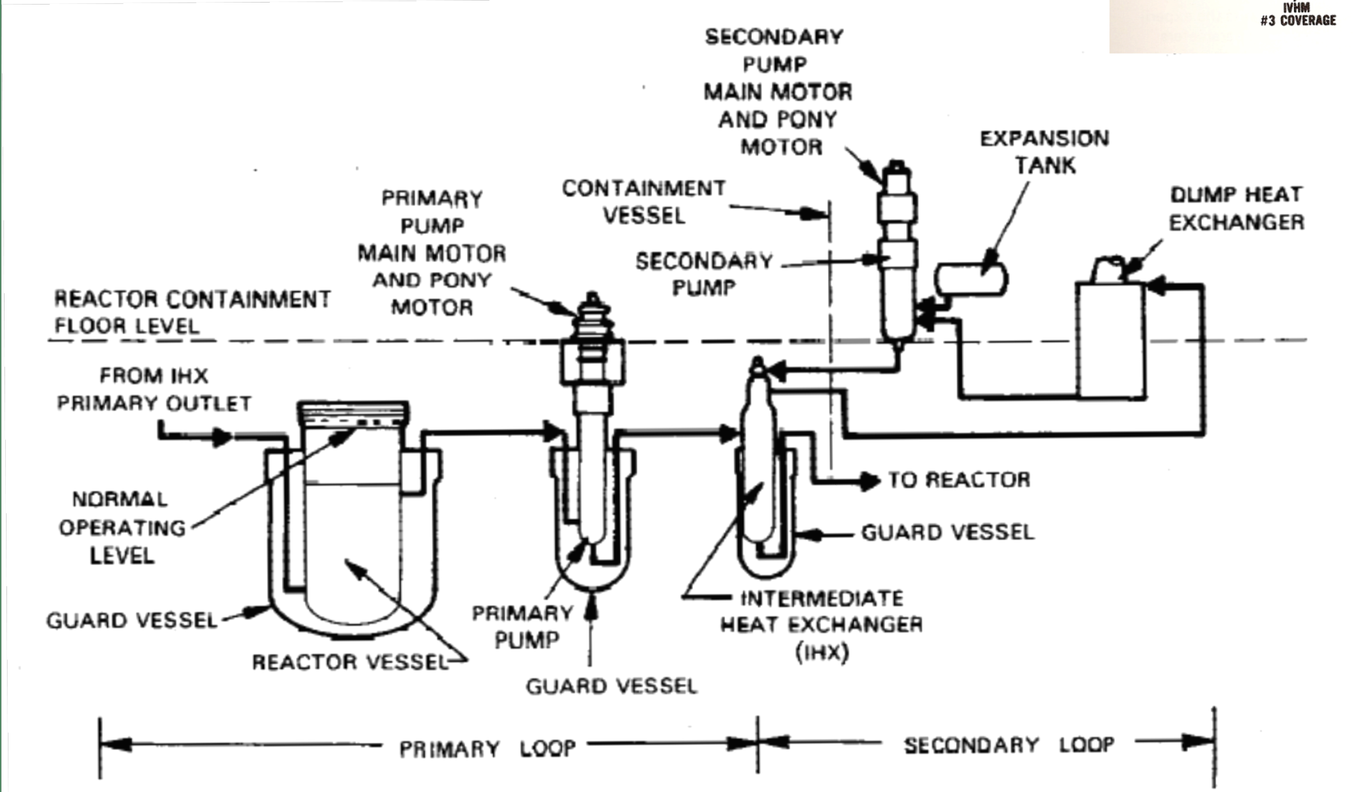
- 400 MW loop-type reactor with oxide fuel in two enrichment zones
- Three loops and 12 DHX modules
- 43,500 gpm primary sodium flow rate with T_{in}=360°C and T_{out}=527°C
- Primary mission: Fuel, materials and component tests



FFTF (cont.)

Heat transport systems and core layout

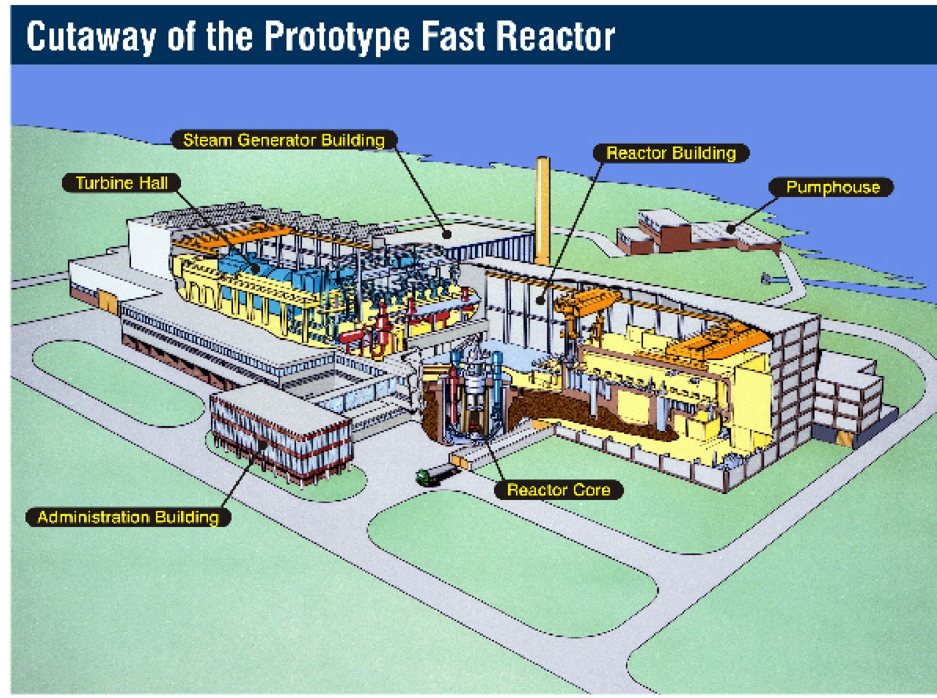




- 217 pins/assembly
- ~150 fuel pellets per pin in 316 SS clad
- Two enrichment zones (20- and 25%)
- Average discharge burnup: 45 MWd/kg
- Limiting peak burnup: 80 MWd/kg
- Power density: 0.39 MW/lt

PFR Prototype Fast Reactor (U.K.)

- Pool-type reactor
- 900 tonnes of sodium
- Criticality 1974
- Sodium inlet temperature – 400 to 430°C, T rise = 160°C
- 600 MW(t) power generation
- 3 secondary loops
- Superheated steam with reheat
- Single wall steam generators
- Fuel handling through one rotating plug
- 80% load factor in 1994



Phenix (France)

- Nominal power May 18, 1973, 255 MW(e)
- 400°C in, 550°C out of core
- Pool type with average burnup 13.5% at periphery of core (breeding ratio 1.16)
- Peak thermal efficiency ~45%
- Closed fuel cycle PUREX
- Phenix Milestones
  - 1973–1990 — Demonstration of fast reactor MOX fuel cycle
  - 1990–1993 — Investigation after negative reactivity shutdowns
  - 1993–2010— Renovation, test, and operation with limited reactor power
  - Currently being decommissioned
  - 350 MW(t), 145 MW(e) on two secondary loops
- Phenix Characteristics
  - All penetrations through top—rotating plug
  - Vessel 11.82-m ID, wall thickness 15 mm below top
  - Pumps and heat exchanger on movable supports, below seals
  - Three primary sodium pumps—variable speed rotating centrifugal

Super-Phenix (France)

- Pool type
- Changes from Phenix
  - Primary sodium purification units within vessel
  - Four helical coil steam generators, 750 MWt each
  - Larger fuel subassemblies to allow more burnup
  - Simplified design of main vessel and roof
  - Dome over upper part of vessel to provide containment
- Characteristics:
  - Power (thermal/electric)—3000MW(t)/1200 MW(e)
  - Thermal efficiency—40%
  - Inner diameter/height of main vessel—21 m/19.5 m
  - Number of loops (primary/secondary)—4/4
  - Number of IHXs—8
  - Sodium inventory (primary/secondary)—3500 tons/500 tons
  - Sodium flow rate (primary/secondary), tonnes/second—4x4.24/4x3.27
  - Primary sodium temperature (hot leg/cold leg)—545°C/395°C
  - Secondary sodium temperature (hot leg/cold leg)—525°C/345°C
  - Steam temperature at turbine inlet—487°C
  - Steam pressure at turbine inlet—177 bar

BN-600 (Russia)

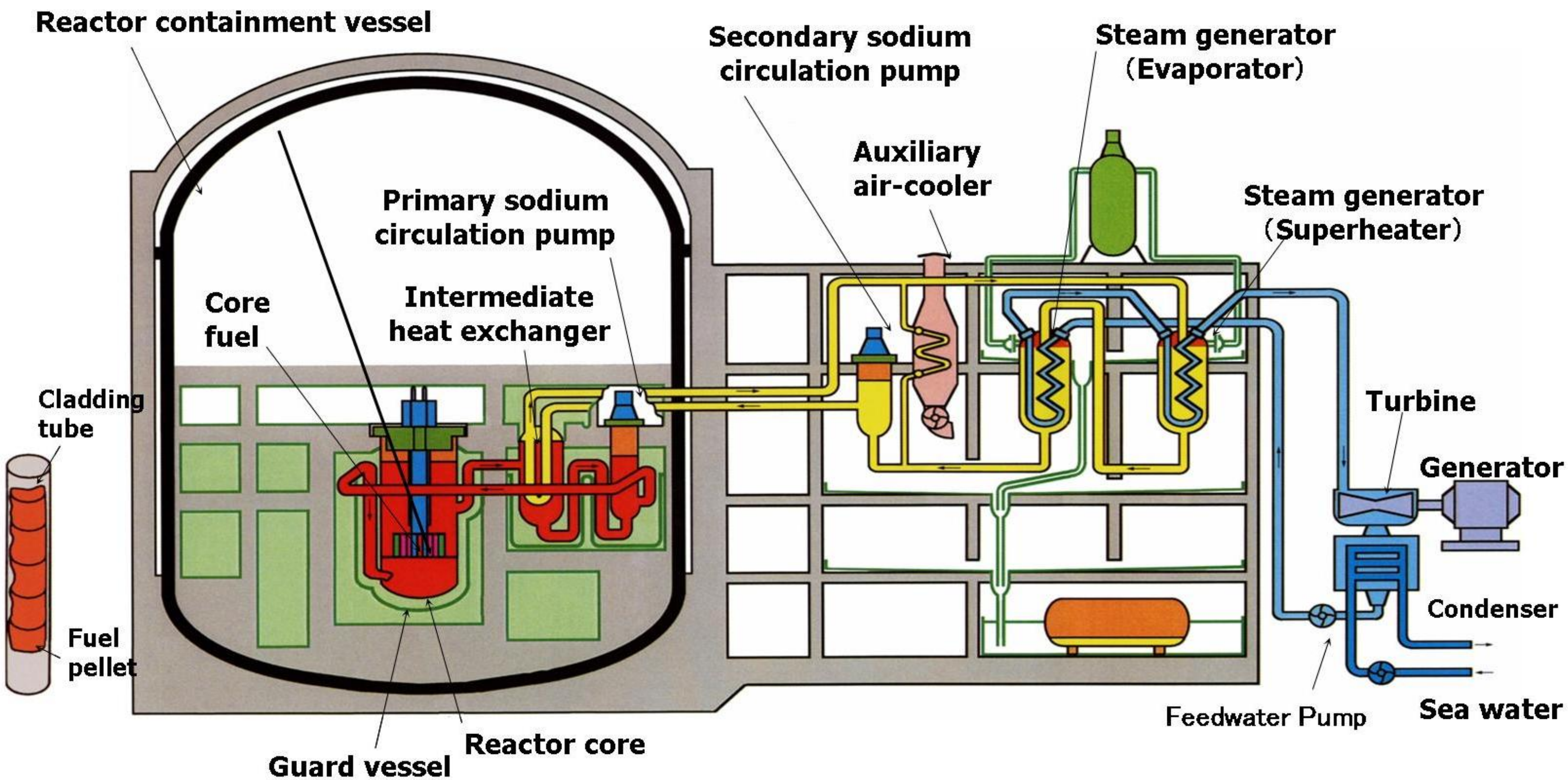
- 600 MWe pool-type SFR
  - 12.8 m in diameter, 12.6-m high cylindrical *reactor* vessel with with no penetrations below the sodium level
  - Low cover gas pressure in the reactor (~0.4 kg/cm^2) to enables small vessel wall thickness (30–40 mm)
  - Bottom supported reactor vessel through a support ring, seated on foundation supports
  - Rotating plug top, shielded IHX, centrifugal pumps
  - Core run 450 days—150 days at full power
  - Fuel Handling system with an eccentric arrangement rotating plugs with two in-pile refueling mechanisms (close and distant relative to the reactor core axis) installed on the small plug, which carries out replacing of assemblies inside the reactor
  - Two drums for new and spent fuel assemblies
  - A spent fuel-to-washing cell transfer mechanism
  - Fuel transfer and washing cells

BN-350 Characteristics (Kazakhstan)

- First commercial plant; on shore of Caspian sea (now Kazakhstan)
- 350 Mwe loop-type plant with six primary and intermediate loops
  - Steam generators
  - Refueling complex (integrated mechanical system)
  - Automated process control and diagnostic systems for monitoring the operating state of the safety-related components and systems
  - EM pumps used in part of secondary system
- Basic Operating Parameters
  - Reactor thermal output, 750 MW
  - Primary sodium temperature at reactor inlet/outlet, 288°C/437°C
  - Sodium flow through reactor, 141,000 tonnes/hour
  - Secondary coolant temperature at SG inlet/outlet, 420°C/260°C
  - Sodium flow in secondary loop, 340 tonnes/hour
  - Number of operating loops (plus one reserve), 5
  - Main steam temperature/pressure, 405°C/4.5 MPa
  - Steam flow, 1070 tonnes/hour

Monju (Japan)

- 714 MWt (290 Mwe) MOX-fueled loop-type SFR with three loops



M. Lind, H. Yoshikawa, S. Jorgensen, M. Yang, K. Tamayama, K. Okusa, "Modeling Operating Modes for the Monju Nuclear Power Plant," January 2012.