OUTLINE

- Background on Fast Reactor Technology
  - Physics and Motivations
  - International Demonstration Reactors

- SFR Fuels and Safety
  - Fuel Options and Key Characteristics
  - Typical System Configuration and Operating Conditions
  - Safety Implications and Inherent Safety Approach

- SFR “Economics”
  - Reactor Demonstration Stages
  - Cost Reduction Design Features
  - Modern SFR Design Approach
In LWRs, most fissions occur in the 0.1 eV thermal “peak”

In SFRs, moderation is avoided – no thermal neutrons
IMPACT OF NEUTRON ENERGY SPECTRUM

- Fissile isotopes are likely to fission in both thermal/fast spectrum
  - Fission fraction is higher in fast spectrum
- Significant (up to 50%) fission of fertile isotopes in fast spectrum

**Net result is more excess neutrons and less higher actinide generation in SFR**
EVOLVING VISION FOR FAST REACTORS

From the initial conception of nuclear energy, it was recognized that full realization of uranium energy content would require fast reactors.

**Fermi:** The vision to close the fuel cycle

**50’s:** First electricity generating reactor: EBR-I with a vision to close the fuel cycle for resource extension

**60-70’s:** Expected Uranium scarcity – international fast reactor programs

**80’s:** Decline of nuclear – Uranium plentiful

**USA (& others):** once through cycle & repository

**France, Japan (& others):** closed cycles to mitigate and delay waste disposal

**Late 90’s in the U.S.:** Rebirth of fast reactor research and development for improved waste management

**Now:** flexible actinide management for fuel cycle benefits
### GENERATION-IV NUCLEAR SYSTEMS

- Six Generation IV Systems considered internationally
- Often target missions beyond electricity
  - High temperature energy products
  - Fuel cycle benefits

<table>
<thead>
<tr>
<th>System</th>
<th>Neutron Spectrum</th>
<th>Coolant</th>
<th>Outlet Coolant Temperature °C</th>
<th>Size (MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>VHTR (Very high temperature reactor)</td>
<td>thermal</td>
<td>helium</td>
<td>900-1,000</td>
<td>250-300</td>
</tr>
<tr>
<td>SFR (Sodium-cooled fast reactor)</td>
<td>fast</td>
<td>sodium</td>
<td>550</td>
<td>30-2,000</td>
</tr>
<tr>
<td>SCWR (Supercritical water-cooled reactor)</td>
<td>thermal/fast</td>
<td>water</td>
<td>510-625</td>
<td>300-1,500</td>
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<tr>
<td>GFR (Gas-cooled fast reactor)</td>
<td>fast</td>
<td>helium</td>
<td>850</td>
<td>1200</td>
</tr>
<tr>
<td>LFR (Lead-cooled fast reactor)</td>
<td>fast</td>
<td>lead or lead alloy</td>
<td>480-800</td>
<td>20-1,000</td>
</tr>
<tr>
<td>MSR (Molten salt reactor)</td>
<td>epithermal/fast</td>
<td>fluoride salts</td>
<td>700-800</td>
<td>1,000</td>
</tr>
</tbody>
</table>
U.S. FAST REACTOR INDUSTRY TODAY

Primary interface with DOE is industry Fast Reactor Working Group (FRWG)

- Started in 2017 to provide developers with access to technical and regulatory resources, continues under NEI leadership

The FRWG Members represent a diverse set of advanced fast reactor technologies:

<table>
<thead>
<tr>
<th>Sodium-Cooled</th>
<th>Lead-Cooled</th>
<th>Gas-Cooled</th>
<th>Molten Salt-Cooled</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oklo</td>
<td>Westinghouse</td>
<td>General Atomics</td>
<td>Elysium</td>
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<tr>
<td>General Electric</td>
<td>Columbia Basin Consulting Group</td>
<td></td>
<td>Southern/TerraPower</td>
</tr>
<tr>
<td>TerraPower</td>
<td>Hydromine</td>
<td></td>
<td>Flibe Energy</td>
</tr>
<tr>
<td>Advanced Reactor Concepts</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

- Electric Utilities - Duke, Exelon, Southern, StudsvikScandpower, EPRI, NEI

The FRWG provides input to DOE on technology development priorities

- Work directly with Lab experts on international engagement and other projects
- Provide consensus feedback to Workshops, Forums, and other advanced reactor groups
FAST REACTOR EXPERIENCE

U.S. Experience

- First usable nuclear electricity was generated by a fast reactor – the EBR-I in 1951
- EBR-II (20 MWe) was operated at Idaho site from 1963 to 1994
- FERMI-1 commercial power reactor (61 MWe) in 1965
- Fast Flux Test Facility (400 MWt) operated from 1980 to 1992

Worldwide Experience

- About 20 fast reactors with >400 operating-years
- Test and/or demonstration reactors built and operated in US, France, UK, Russia, Japan, India, and China
- New power reactors: BN-800 (880 MWe) – 2014, PFBR (500 MWe) - TBD
- Active demonstration projects: CFR600 (China), Natrium (USA)

Viability of sodium-cooled fast reactor technology is demonstrated
REACTOR DEVELOPMENT STEPS: US AND INTERNATIONAL EXPERIENCE FOR LWRS AND ADVANCED REACTOR SYSTEMS

- **Research and Development**
  - Prove scientific feasibility associated with fuel, coolant and geometrical configuration

- **Engineering Demonstration**
  - Reduced scale
  - Proof of concept
  - Concepts that have NEVER been built
  - Viability of integrated system

- **Performance Demonstration**
  - Establish that scaleup of system works
  - Gain operating experience to validate integral behavior of the system
  - Proof of performance

- **Commercial Demonstration**
  - Full scale to be replicated for subsequent commercial offerings if system works as designed
# INTERNATIONAL FAST REACTORS

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Country</th>
<th>MWth</th>
<th>Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>EBR 1</td>
<td>USA</td>
<td>1.4</td>
<td>1951-63</td>
</tr>
<tr>
<td>DFR</td>
<td>UK</td>
<td>60</td>
<td>1959-77</td>
</tr>
<tr>
<td>BR-10</td>
<td>Russia</td>
<td>8</td>
<td>1959-71, 1973-2002</td>
</tr>
<tr>
<td>EBR II</td>
<td>USA</td>
<td>62.5</td>
<td>1963-94</td>
</tr>
<tr>
<td>Fermi 1</td>
<td>USA</td>
<td>200</td>
<td>1963-72</td>
</tr>
<tr>
<td>Rapsodie</td>
<td>France</td>
<td>40</td>
<td>1966-82</td>
</tr>
<tr>
<td>BOR-60</td>
<td>Russia</td>
<td>50</td>
<td>1968-</td>
</tr>
<tr>
<td>BN 350*</td>
<td>Kazakhstan</td>
<td>750</td>
<td>1972-99</td>
</tr>
<tr>
<td>Phenix</td>
<td>France</td>
<td>563</td>
<td>1973-2009</td>
</tr>
<tr>
<td>PFR</td>
<td>UK</td>
<td>650</td>
<td>1974-94</td>
</tr>
<tr>
<td>KNK 2</td>
<td>Germany</td>
<td>58</td>
<td>1977-91</td>
</tr>
<tr>
<td>Joyo</td>
<td>Japan</td>
<td>140</td>
<td>1978-</td>
</tr>
<tr>
<td>FFTF</td>
<td>USA</td>
<td>400</td>
<td>1980-93</td>
</tr>
<tr>
<td>BN 600</td>
<td>Russia</td>
<td>1470</td>
<td>1980-</td>
</tr>
<tr>
<td>Superphenix</td>
<td>France</td>
<td>3000</td>
<td>1985-98</td>
</tr>
<tr>
<td>FBTR</td>
<td>India</td>
<td>40</td>
<td>1985-</td>
</tr>
<tr>
<td>Monju</td>
<td>Japan</td>
<td>714</td>
<td>1994-96, 2010-15</td>
</tr>
<tr>
<td>CEFR</td>
<td>China</td>
<td>65</td>
<td>2010-</td>
</tr>
<tr>
<td>PFBR</td>
<td>India</td>
<td>1250</td>
<td>2021?</td>
</tr>
<tr>
<td>BN-800</td>
<td>Russia</td>
<td>2000</td>
<td>2014-</td>
</tr>
<tr>
<td>CFR600</td>
<td>China</td>
<td>1500</td>
<td>2023</td>
</tr>
<tr>
<td>MBIR</td>
<td>Russia</td>
<td>150</td>
<td>2028</td>
</tr>
<tr>
<td>Natrium</td>
<td>USA</td>
<td>840</td>
<td>2027</td>
</tr>
</tbody>
</table>
FAST POWER REACTORS – MIXED EXPERIENCE

- FERMI-1 was built in early 1960s, only 61 Mwe
  - Flow blockage with local fuel melting in 1966, restarted in 1970
  - Stopped in 1972 due to fuel supply

- Russian BN-600 (600 MWe) started operation in 1980
  - Excellent operating record ~75% capacity factor for 40 years
  - Life extension to 2025 (45 years) with 2040 (60 years) application

- French SUPERPHENIX (1242 MWe) started operation in 1986
  - Limited power in startup phase, secondary loop problems
  - Shutdown in 1998, for political reasons

- Japanese MONJU (280 MWe) started operation in 1995
  - Secondary sodium leak in December 1995
  - Restarted in 2010; fuel handling incident in August 2010
  - Official shutdown in 2016, avoiding post-Fukushima upgrade costs
NEW SFR DEMONSTRATION REACTORS

- New power reactors recently built - BN-800 (880 MWe) in Russia
  - First criticality in June 2014
  - Connection to electrical grid in December 2015
  - Commissioned as power unit in October 2016
  - 82% capacity factor for operations in 2020

- PFBR (500 MWe) in India
  - All construction activities completed
  - Should start operations in late 2021/2022

- Other active fast reactor demonstration projects
  - CFR600 in China under construction
  - Natrium in US siting and licensing
SFR FUELS AND SAFETY
EARLY FAST REACTORS AND FUEL FORMS

Original choice was high density metal fuel (for breeding)

- First usable nuclear electricity—EBR-I in 1951
- EBR-II (1963), Fermi (1963), DFR (UK, 1959) all used metal fuel
- Early designs experienced severely limited fuel burnup because of fuel swelling (U-10Mo burnup of 3 GWD/MT for Fermi)

U.S. and international programs switched to oxide fuel in the late 1960s

- Low swelling and successful Navy oxide fuel experience → high burnup
- Fast Flux Test Facility (400 MWt) operated with oxide from 1980 to 1992

EBR-II (20 MWe) continued metal alloy fuel development from 1963 to 1994

- Solved burnup limitation by allowing adequate space for fuel swelling
- Demonstrated peak burnup comparable to oxide fuel (200 GWD/MT)
# Fast Reactor Fuel Options

## Fast Reactor Fuel Type

### Fresh Fuel Properties

<table>
<thead>
<tr>
<th>Fuel Type</th>
<th>Metal U-20Pu-10Zr</th>
<th>Oxide UO₂-20PuO₂</th>
<th>Nitride UN-20PuN</th>
<th>Carbide UC-20PuC</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heavy Metal Density, g/cm³</td>
<td>14.1</td>
<td>9.3</td>
<td>13.1</td>
<td>12.4</td>
</tr>
<tr>
<td>Melting Temperature, °K</td>
<td>1350</td>
<td>3000</td>
<td>3035*</td>
<td>2575</td>
</tr>
<tr>
<td>Thermal Conductivity, W/cm°K</td>
<td>0.16</td>
<td>0.023</td>
<td>0.26</td>
<td>0.20</td>
</tr>
<tr>
<td>Operating Centerline Temperature at 40 kW/m, °K, and (T/T_melt)</td>
<td>1060 (0.8)</td>
<td>2360 (0.8)</td>
<td>1000 (0.3)</td>
<td>1030 (0.4)</td>
</tr>
<tr>
<td>Fuel-Cladding Solidus, °K</td>
<td>1000</td>
<td>1675</td>
<td>1400</td>
<td>1390</td>
</tr>
<tr>
<td>Thermal Expansion, 1/°K</td>
<td>17E-6</td>
<td>12E-6</td>
<td>10E-6</td>
<td>12E-6</td>
</tr>
<tr>
<td>Heat Capacity, J/g°K</td>
<td>0.17</td>
<td>0.34</td>
<td>0.26</td>
<td>0.26</td>
</tr>
<tr>
<td>Reactor Experience, Country</td>
<td>US, UK</td>
<td>RUS, FR, JAP US, UK</td>
<td>IND</td>
<td></td>
</tr>
</tbody>
</table>
DESIGN ISSUES VARY FOR DIFFERENT FUEL OPTIONS

- **Fuel Swelling**
  - Fission product retention in carbide and nitride fuels can lead to greater swelling than observed for oxide fuels and exacerbate FCMI
  - Current metal and oxide fuel pin designs accommodate fuel swelling

- **Fuel / Cladding Chemical Interaction**
  - Metal uranium and plutonium forms low-melting point eutectic with iron
  - May limit coolant outlet temperature of metal fuel core, e.g., 510°C for metal as compared to ~550°C for oxide (structural materials limiting)

- **Fuel / Cladding Mechanical Interaction (FCMI)**
  - Hard, strong fuel forms push on cladding, particularly at high burnup
  - Worst for nitride and carbide, limits maximum burnup for ceramic fuels

- **Fuel / Coolant Compatibility**
  - Oxide fuel chemically reacts with the sodium coolant
    - Stricter limits on fuel pin failures to prevent potential flow blockages
  - See picture on failed cladding tests
RUN BEYOND CLADDING BREACH TESTS

9% burnup Oxide RBCB Test

12% Burnup Metal RBCB Test
(Operated 169 days after breach)
SAFETY IN DESIGN

- Like LWRs, SFR safety is first based on utilization of multiple (redundant and diverse) engineered protection systems:
  - independent scram systems with provision for stuck rods,
  - multiple coolant pumps, heat transport loops,
  - dedicated decay heat removal systems,
  - multiple barriers to release of radioactive materials.

- Inherent reactivity feedback mechanisms provide additional measures to protect the reactor during double-fault events:
  - Doppler effect,
  - reactivity feedback due to fuel axial and core radial expansion,
  - feedback due to changes in coolant density and void worth,
  - control-rod driveline expansion.

- For some designs, passive reactivity insertion devices (i.e., GEMs), and/or self-actuated shutdown systems are also considered.

- Intermediate loop and pool configuration utilized to assure primary coolant inventory; sodium leak or secondary water reactions monitored closely.
# TYPICAL SPECIFICATIONS OF LWR AND SFR

<table>
<thead>
<tr>
<th></th>
<th>PWR</th>
<th>SFR</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>General</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Specific power (kWt/kg-fissile)</td>
<td>786</td>
<td>556</td>
</tr>
<tr>
<td>Power density (MWt/m³)</td>
<td>102</td>
<td>300</td>
</tr>
<tr>
<td><strong>Fuel</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rod outer diameter (mm)</td>
<td>9.5</td>
<td>7.9</td>
</tr>
<tr>
<td>Clad thickness (mm)</td>
<td>0.57</td>
<td>0.36</td>
</tr>
<tr>
<td>Rod pitch-to-diameter ratio</td>
<td>1.33</td>
<td>1.15</td>
</tr>
<tr>
<td>Enrichment (% fissile)</td>
<td>~4.0</td>
<td>~20</td>
</tr>
<tr>
<td>Average burnup (MWd/kg)</td>
<td>40</td>
<td>100</td>
</tr>
<tr>
<td><strong>Thermal</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td><strong>Hydraulic</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Coolant</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure (MPa)</td>
<td>15.5</td>
<td>0.1</td>
</tr>
<tr>
<td>Inlet temp. (°C)</td>
<td>293</td>
<td>332</td>
</tr>
<tr>
<td>Outlet temp. (°C)</td>
<td>329</td>
<td>499</td>
</tr>
<tr>
<td>Reactor Δp (MPa)</td>
<td>0.345</td>
<td>0.827</td>
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<tr>
<td>Rod surface heat flux</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Average (MW/m²)</td>
<td>0.584</td>
<td>1.1</td>
</tr>
<tr>
<td>Maximum MW/m²</td>
<td>1.46</td>
<td>1.8</td>
</tr>
<tr>
<td>Average linear heat rate (kW/m)</td>
<td>17.5</td>
<td>27.1</td>
</tr>
<tr>
<td>Steam</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure (MPa)</td>
<td>7.58</td>
<td>15.2</td>
</tr>
<tr>
<td>Temperature (°C)</td>
<td>296</td>
<td>455</td>
</tr>
</tbody>
</table>
TYPICAL THERMAL OPERATING CONDITIONS

- Sodium cooled fast reactors operate at near atmospheric pressure; peak pressures are set by core pressure drop and gravity head characteristics (up to about 1.0 MPa max at reactor inlet)
- Reactor coolant outlet temperatures are 510°C to 550°C, depending on cladding material (margin to boiling 330°C to 370°C)
- At reactor temperatures, sodium wets stainless steel, which is typically used as the cladding and structural material
- Average power densities in the reactor core are 300 to 500 kW/liter
- Typical coolant velocities in the fuel pin bundle are 5 to 7 m/s
- Fuel pins are tightly packed, typically arranged in a nearly touching triangular pitch, and positioned by a spiral wire spacer within a hexagonal assembly duct
- Average fuel pin linear power ratings are typically 23 to 28 kW/m for pins with cladding diameters of 6 to 8 mm
SAFETY IMPLICATIONS OF SFR DESIGN APPROACH

- Heat transfer properties of liquid metals allow:
  - Operation at high power density and high fuel volume fraction
  - Low pressure operation with significant margin to boiling
  - Enhanced natural circulation for heat removal

- High leakage fraction implies that the fast reactor reactivity is sensitive to minor geometric changes
  - As temperature increases and materials expand, a net negative reactivity feedback is inherently introduced

- Inherent safety design principles:
  - Tailored reactivity feedbacks to prevent core damage (page 22)
  - Multiple paths for passive decay heat removal envisioned (page 23)

- Favorable inherent feedback to prevent fuel damage has been demonstrated in United States sodium-cooled fast reactors (SFR)
  - EBR-II and FFTF tests for double fault severe transients
FAST REACTOR FEEDBACK EFFECTS

- Sensitivity to geometric changes introduces many feedback effects
  - Doppler
  - Coolant density
  - Core radial expansion
  - Core axial expansion
  - Grid-plate expansion
  - Control-rod driveline expansion
  - Vessel expansion

- System is designed to assure net negative temperature coefficient
DECAY HEAT REMOVAL SYSTEM OPTIONS

- Standard path is through primary/secondary loops
- Passive backup decay heat removal systems
  - natural circulation
  - either continuous operations or passive activation mechanism
- Most designs include multiple DHR systems
  - redundant
  - diverse

RVACS--Reactor Vessel Auxiliary Cooling System
VCCS--Vessel Cavity Cooling System
DRACS--Direct Reactor Auxiliary Cooling System
PRACS--Primary Reactor Auxiliary Cooling System
IRACS--Intermediate Reactor Auxiliary Cooling System
SGACS--Steam Generator Auxiliary Cooling System
SFR ECONOMICS
REACTOR DEVELOPMENT STEPS: US AND INTERNATIONAL EXPERIENCE FOR LWRS AND ADVANCED REACTOR SYSTEMS

Research and Development
- Prove scientific feasibility associated with fuel, coolant and geometrical configuration
- Reduced scale
- Proof of concept
- Concepts that have NEVER been built
- Viability of integrated system

Engineering Demonstration
- Establish that scaleup of system works
- Gain operating experience to validate integral behavior of the system
- Proof of performance

Performance Demonstration
- Full scale to be replicated for subsequent commercial offerings if system works as designed

Commercial Demonstration

From DOE Advanced Demonstration and Test Reactors Study, INL/EXT-16-37867 (January 2017)
## DEPLOYMENT EXAMPLES

<table>
<thead>
<tr>
<th>Step in Deployment Path</th>
<th>Light Water Reactor (example)</th>
<th>Sodium Fast Reactor</th>
<th>High Temperature Gas-cooled Reactor</th>
<th>Lead Fast Reactor</th>
<th>Molten Salt Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>R&amp;D for scientific feasibility</td>
<td>SPERT, BORAX, PBF</td>
<td>SEFOR (20 MWe), TREAT</td>
<td>CABRI</td>
<td>US International</td>
<td>US International</td>
</tr>
<tr>
<td>Engineering Demonstration</td>
<td>S1W EBWR</td>
<td>EBR-I (1.4 MWe), EBR-II (20 MWe)</td>
<td>Dounreay (14 MWe), Rhapsodie (40 MWe), Peach Bottom (40 MWe)</td>
<td>DRAGON (20 MWe), HTR-10 (10 MWe), HTTR (30 MWe), AVR (15 MWe)</td>
<td>Soviet submarines (^a)</td>
</tr>
<tr>
<td>Performance Demonstration</td>
<td>USS Nautilus, Shippingport</td>
<td>Fermi-1 (69 MWe); FFTF (400 MWe)</td>
<td>Phenix (233 MWe), Monju (300 MWe), BN-300 and BN-600 (300 and 600 MWe), PFR (250 MWe)</td>
<td>FSV (842 MWt)</td>
<td>THTR (750 MWt) (^b)</td>
</tr>
<tr>
<td>Commercial Demonstration</td>
<td>Yankee Rowe (485-600 MWe)</td>
<td>Superphenix (3000 MWh), BN-800 (600 MWe)</td>
<td></td>
<td></td>
<td>HTR-PM (200 MWe)</td>
</tr>
</tbody>
</table>

\(^a\) The Soviet experience with lead-bismuth eutectic cooled submarine reactors is relevant but not directly applicable to the LFR point design, therefore they are considered engineering demonstration reactors for the LFR.

\(^b\) The Aircraft Reactor Experiment and MSRE were liquid fueled reactors, with different coolant chemistry than the salt-cooled FHR demonstration reactor point design.

\(^c\) FSV and THTR were commercial demonstrations of large HTGRs, however, for modular HTGRs under consideration today, they serve the role of a performance demonstration.
EVOLUTION OF SFR DESIGN

- Early power reactors were focused on performance demonstration of reliability, not cost reduction
  - BN-600 showed reliable operations, but not a commercial demonstration

- Many design refinements introduced in 1980s-2010
  - Design simplification, system integration and compaction, modular fabrication, power upgrades, etc.
  - Some examples follow: PRISM, BN800 to 1200, JSFR

- Modern designs further refined to current market applications
  - Distributed architecture to segregate non-nuclear components, informed project execution approach, adapt for grid variability, etc.
  - An example follows: Natrium
PRISM DESIGN – COST REDUCTION APPROACH

PRISM SFR was pioneer on modularization
- Factory fabrication
- Construction and transport benefits
- Learning curve for cost reduction

Design was refined in DOE ALMR Program and subsequent work by GE Hitachi
- Compact modular pool configuration
- Electromagnetic pumps – no moving parts
- Optimized plant for minimal footprints
- Multi-module shared infrastructure

Inherent safety allows design simplifications

Optimizing the Size of the SUPER-PRISM Reactor, Boardman et al., ICONE-8 (2000)
Economic Assessment of S-PRISM Including Development and Generation Costs, Boardman et al. ICONE-9 (2001)
RUSSIAN BN DESIGNS – COST REDUCTION

Russian SFR design has evolved in the BN reactors
- BN-600 design reflect BN-350 challenges
- BN-800 is a further performance demonstration
  • Maintain BN-600 reliability, but improve cost
  • Roughly same size as BN-600
  • Power output increased to 880 MWe

BN-1200 design large cost reductions for commercial
- Simplified, compact configuration
- Large modular steam generators
- Simplified refueling system
- Elimination of ex-vessel storage
- High density, high burnup nitride fuel

Claim new design will require 50% less steel

Development of the New Generation Power Unit with the BN-1200 Reactor, Vasilev et. al. FR19 Conference
JAPAN JSFR DESIGN – COST REDUCTION APPROACH

JSFR incorporated innovative cost reduction features
- Advanced materials
- Large 1500 MWe plant with 2 loop configuration
- Integrated pump-IHX component
- Compact, modular fab vessel
- High burnup fuel

Quantitative cost comparisons in paper
- ~25% reduction from FOAK
- 10% of Monju construction cost
- Higher commodity cost than APWR, but, greatly reduced volume/emergency systems cost
- With innovative technologies, net lower cost than APWR is achieved


https://aris.iaea.org/PDF/JSFR.pdf
Resulting benefits

- A distributed architecture allows major parts of the Natrium plant to be built to less demanding standards, reducing cost and construction time.
- Energy island systems can be constructed as a fully commercial (non-nuclear) project.
- These benefits do not require additional technology development.

Approx. cost multipliers based on experience of nuclear constructors:

1.0x  1.2x  3.0x  5.0x
Natrium Reactor Storage and Ramping Balance a Renewables-Based Grid

Significant price volatility from solar daily / seasonal variability - WECC Region

**Demand profile, renewable generation and power prices on typical 48 hour periods in summer and winter**

**Summer**
- Limited generation needs as wind + solar can almost satisfy all demand during the day
- Low day-time prices with short and sharp price spike in early evening as solar tails off
- *Residual demand = net demand after wind and solar*

**Winter**
- More volatile prices driven by shorter daylight hours – two price peaks per day
- Solar variability can produce oversupply and power curtailment in peak supply times

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* TerraPower Proprietary & Confidential - Exempt from Disclosure Under FOIA"
SUMMARY

20 Demonstration SFRs have been built and operated
  – Technical viability of SFR technology confirmed
  – Reliable operations demonstrated in BN-600 over 40 years
  – Power reactors, but not commercial demonstration to date

Modern SFR designs incorporate demonstrated inherent safety features
  – High conductivity metal alloy fuel form
  – Negative temperature coefficient through multiple feedbacks
  – Passive decay heat removal systems

Economics
  – Many, significant cost reduction features employed in modern designs
  – Most recent performance demo – BN-800 shows improvement
  – Thus, modern, innovative designs have the potential to be competitive
  – But efficacy of cost reduction features needs to be demonstrated (to verify improvement performance, cost, and reliability)
QUESTIONS
SODIUM VOID WORTH

For context,

- SFRs operate at low pressure with significant margin to boiling (~350°C)
- The inherent reactivity feedbacks are effective at maintaining this margin, even in severe accident conditions
  - As temperature increases and materials expand, a net negative reactivity feedback is inherently introduced
  - Power is reduced
  - New equilibrium is established once temperatures adjust to heat removal rate

- The traditional challenge to coolant boiling margin in CRBR and other large SFR licensing cases was for an unprotected loss of flow (ULOF) double safety system fault event
  - With oxide fuel, significant reactivity is introduced as fuel cools (page 37)
  - This slows the power reduction rate, with voiding before power reduces sufficiently
  - This behavior does not occur with metal fuel alloy fuel, where the low operating temperature does not result in significant positive Doppler feedback

- This favorable inherent behavior to prevent fuel damage was demonstrated in United States sodium-cooled fast reactors (SFR)
  - EBR-II for double fault transients (1986) for both ULOF and loss-of-heat-sink
METALLIC FUEL SAFETY PERFORMANCE: LOW OPERATING TEMP AND STORED DOPPLER REACTIVITY

- High temperature of oxide fuel implies more stored Doppler
- As power inherently decreases in undercooling event, peak and asymptotic temperatures are determined by reactivity balance

Oxide Fuel
(Doppler Coeff. = -0.005)

~1.5 $

Metallic Fuel
(Doppler Coeff. = -0.003)

~0.3 $
The issue of positive coolant void worth was addressed in the PRISM Preapplication Safety Evaluation Report (NUREG-1368)

- Positive Void Reactivity Coefficient is identified as a concern in Section 3.1.2.7
  - "for sodium voiding to occur, redundant and diverse safety-grade systems would have to experience multiple failures"
  - "Staff conclude that positive sodium void coefficient should not necessarily disqualify a particular reactor design"
  - However, further analyses were identified
  - "Staff will take into account the overall risk perspective"

- The GE paper below captures some of the following discussion with NRC on PRISM licensing
  - "all means identified to lower the void worth have resulted in core designs with other safety issues and with increased costs"

- Subsequent analyses show benign severe accident behavior for metal alloy fuel
  - Melting point of fuel is lower than coolant boiling point
  - This promotes dispersive fuel behavior leading to large negative reactivity effects

U.S. ALMR Licensing Status, Magee, Advanced Reactor Safety Topical Meeting (April 1994)