SODIUM COOLED FAST REACTOR TECHNOLOGIES

ROBERT N HILL
Program Leader, Advanced Nuclear Energy R&D
National Technical Director, ART Fast Reactor R&D

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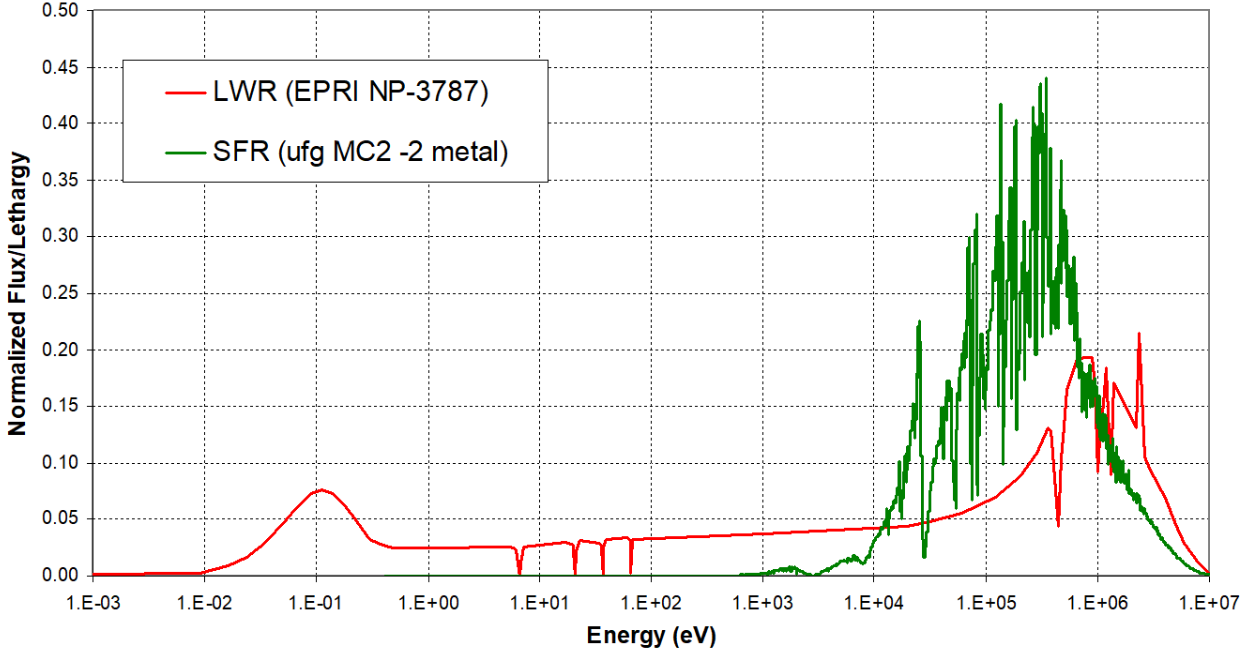
OUTLINE

- 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

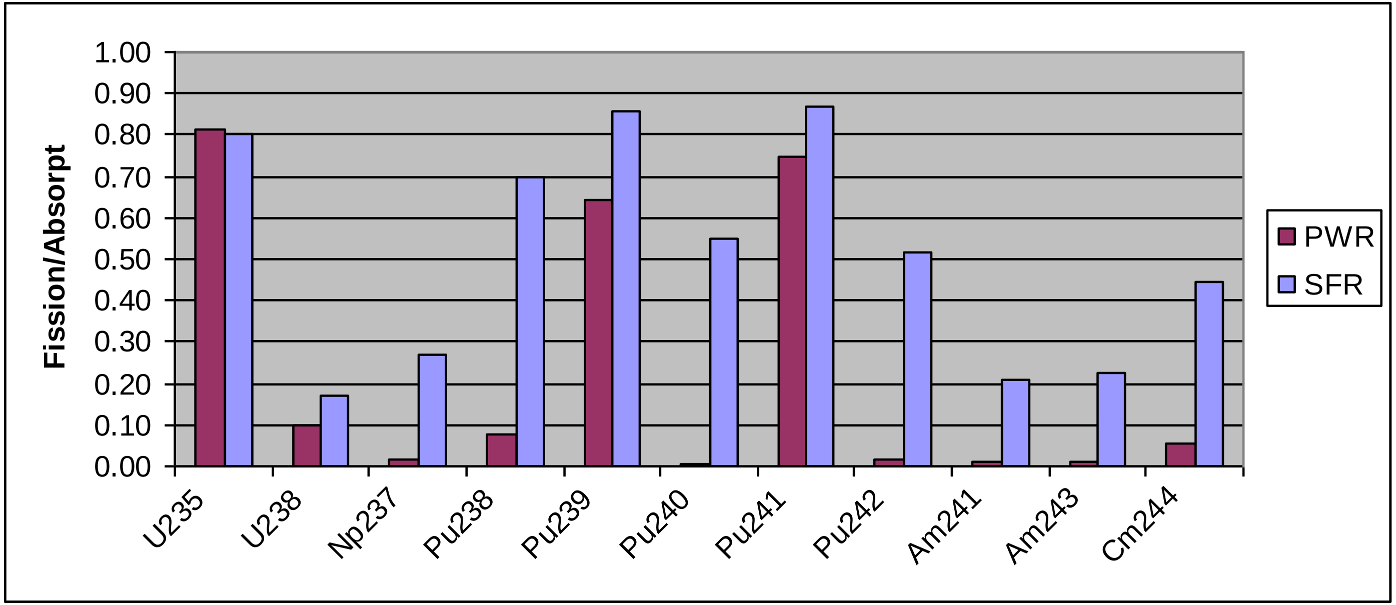
- SFR “Economics”
  - Reactor Demonstration Stages
  - Cost Reduction Design Features
  - Modern SFR Design Approach

COMPARISON OF LWR AND SFR SPECTRA



- 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

2 paths

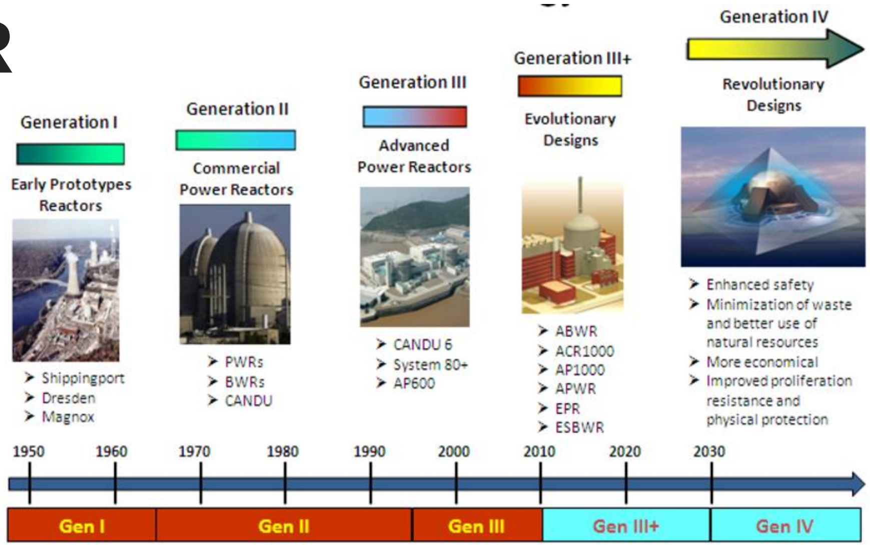
- USA (& others): once through cycle & repository
- France, Japan (& others): closed cycles to mitigate and delay waste disposal

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



System	Neutron Spectrum	Coolant	Outlet Coolant Temperature ºC	Size (MWe)
VHTR (Very high temperature reactor)	thermal	helium	900-1,000	250-300
SFR (Sodium-cooled fast reactor)	fast	sodium	550	30-2,000
SCWR (Supercritical water-cooled reactor)	thermal/fast	water	510-625	300-1,500
GFR (Gas-cooled fast reactor)	fast	helium	850	1200
LFR (Lead-cooled fast reactor)	fast	lead or lead alloy	480-800	20-1,000
MSR (Molten salt reactor)	epithermal/fast	fluoride salts	700-800	1,000

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:

Sodium-Cooled	Lead-Cooled	Gas-Cooled	Molten Salt-Cooled
Oklo	Westinghouse	General Atomics	Elysium
General Electric	Columbia Basin Consulting Group		Southern/TerraPower
TerraPower	Hydromine		Flibe Energy
Advanced Reactor Concepts			

- 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

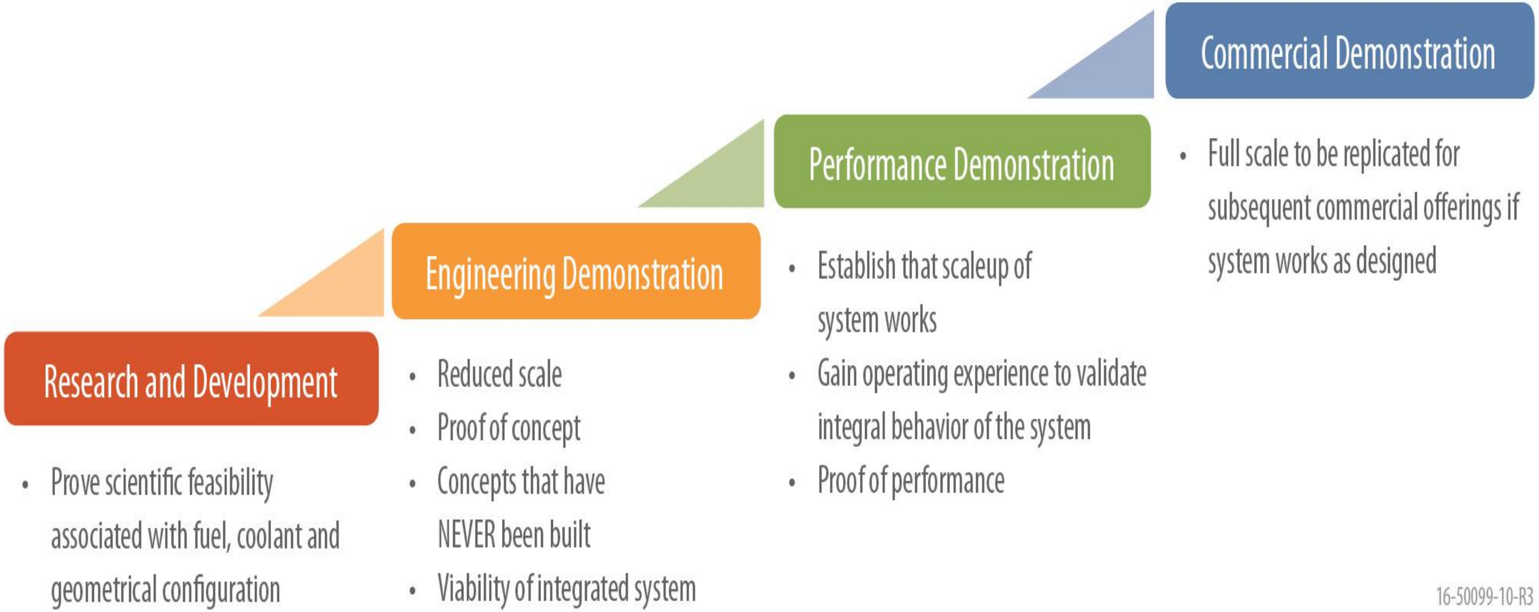
- First usable nuclear electricity was generated by a fast reactor – the EBR-I in 1951
- EBR-II (20 MWe) was operated at Idaho site from 1963 to 1994
- FERMI-1 commercial power reactor (61 MWe) in 1965
- Fast Flux Test Facility (400 MWt) operated from 1980 to 1992

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



INTERNATIONAL FAST REACTORS

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

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 <u>~75% capacity factor for 40 years</u>
  - 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)*

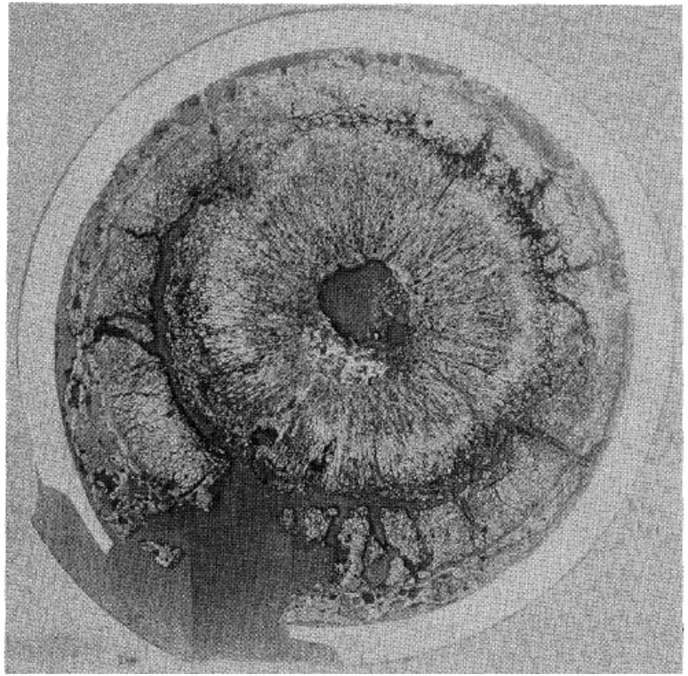
FAST REACTOR FUEL OPTIONS

Fast Reactor Fuel Type Fresh Fuel Properties	**Metal U-20Pu-10Zr**	**Oxide UO_2-20PuO_2**	**Nitride UN-20PuN**	**Carbide UC-20PuC**
Heavy Metal Density, g/cm^3	**14.1**	**9.3**	13.1	12.4
Melting Temperature, ºK	**1350**	3000	3035*	2575
Thermal Conductivity, W/cm-ºK	0.16	**0.023**	0.26	0.20
Operating Centerline Temperature at 40 kW/m, ºK, and (T/T_{melt})	1060 (0.8)	2360 (0.8)	1000 **(0.3)**	1030 **(0.4)**
Fuel-Cladding Solidus, ºK	**1000**	1675	1400	1390
Thermal Expansion, 1/ºK	17E-6	12E-6	10E-6	12E-6
Heat Capacity, J/gºK	0.17	0.34	0.26	0.26
Reactor Experience, Country	US, UK	RUS, FR, JAP US, UK		IND
Research & Testing, Country	US, JAP, ROK, CHI	RUS, FR, JAP, US, CHI	US, RUS, JAP	IND, FR

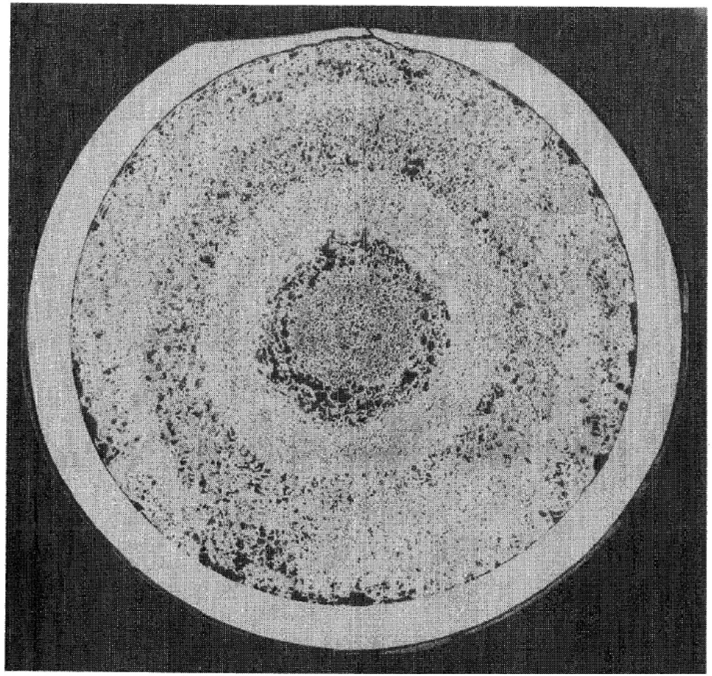
DESIGN ISSUES VARY FOR DIFFERENT FUEL OPTIONS

- 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., 510ºC for metal as compared to ~550ºC 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 <u>ceramic</u> 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

			PWR	SFR
General	Specific power (kWt/kg-fissile)		786	556
	Power density (MWt/m³)		102	300
Fuel	Rod outer diameter (mm)		9.5	7.9
	Clad thickness (mm)		0.57	0.36
	Rod pitch-to-diameter ratio		1.33	1.15
	Enrichment (% fissile)		~4.0	~20
	Average burnup (MWd/kg)		40	100
Thermal Hydraulic	Coolant	pressure (MPa)	15.5	0.1
		inlet temp. (℃)	293	332
		outlet temp. (℃)	329	499
		reactor Δp (MPa)	0.345	0.827
	Rod surface heat flux	average (MW/m²)	0.584	1.1
		maximum MW/m²)	1.46	1.8
	Average linear heat rate (kW/m)		17.5	27.1
	Steam	pressure (MPa)	7.58	15.2
		temperature (℃)	296	455

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

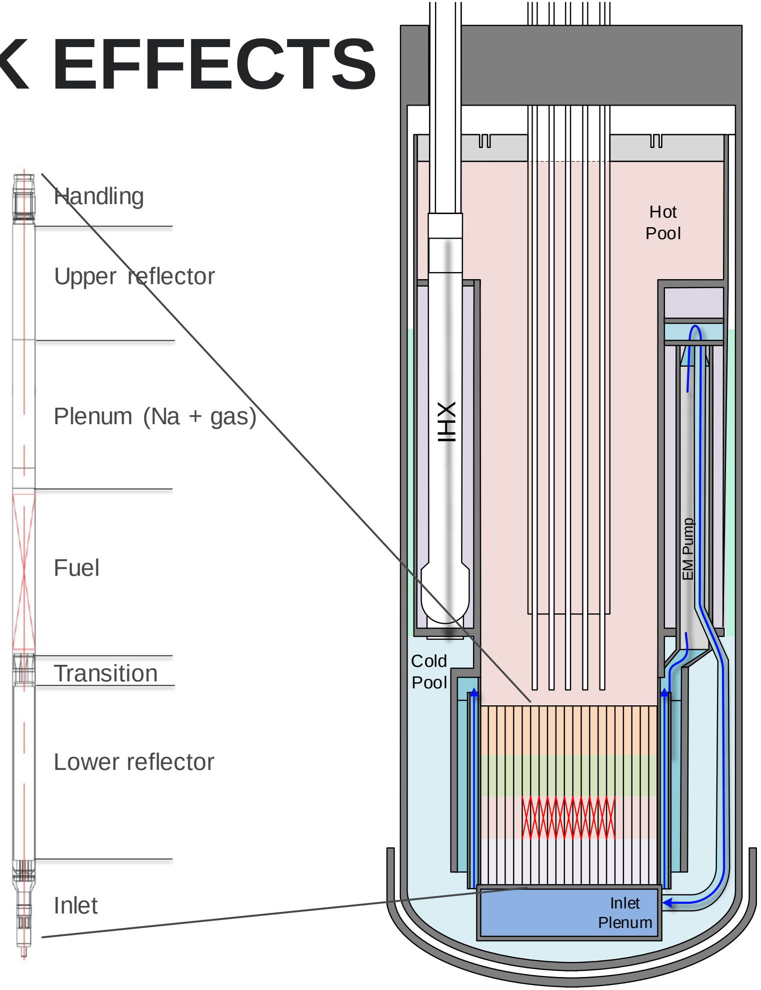
- Fuel pins are tightly packed, typically arranged in a nearly touching triangular pitch, and positioned by a spiral wire spacer within a hexagonal assembly duct
- Average fuel pin linear power ratings are typically 23 to 28 kW/m for pins with cladding diameters of 6 to 8 mm

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

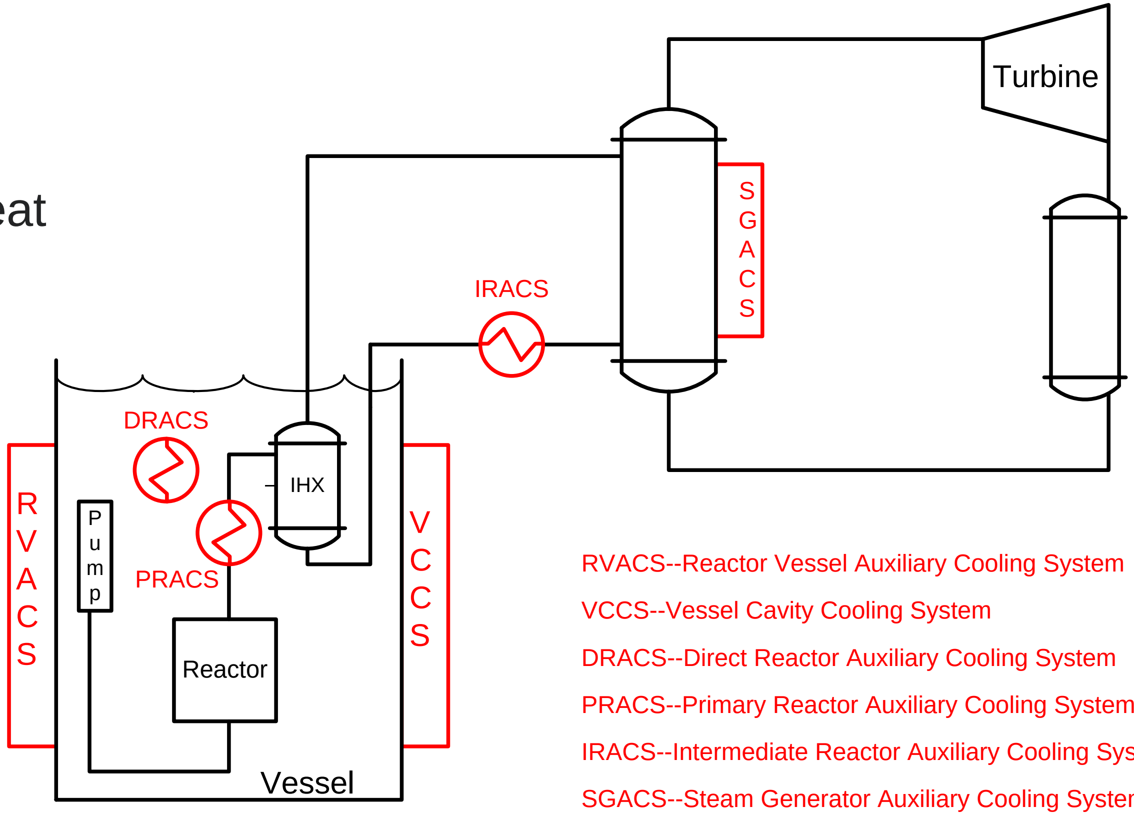


DECAY HEAT REMOVAL SYSTEM OPTIONS

- 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

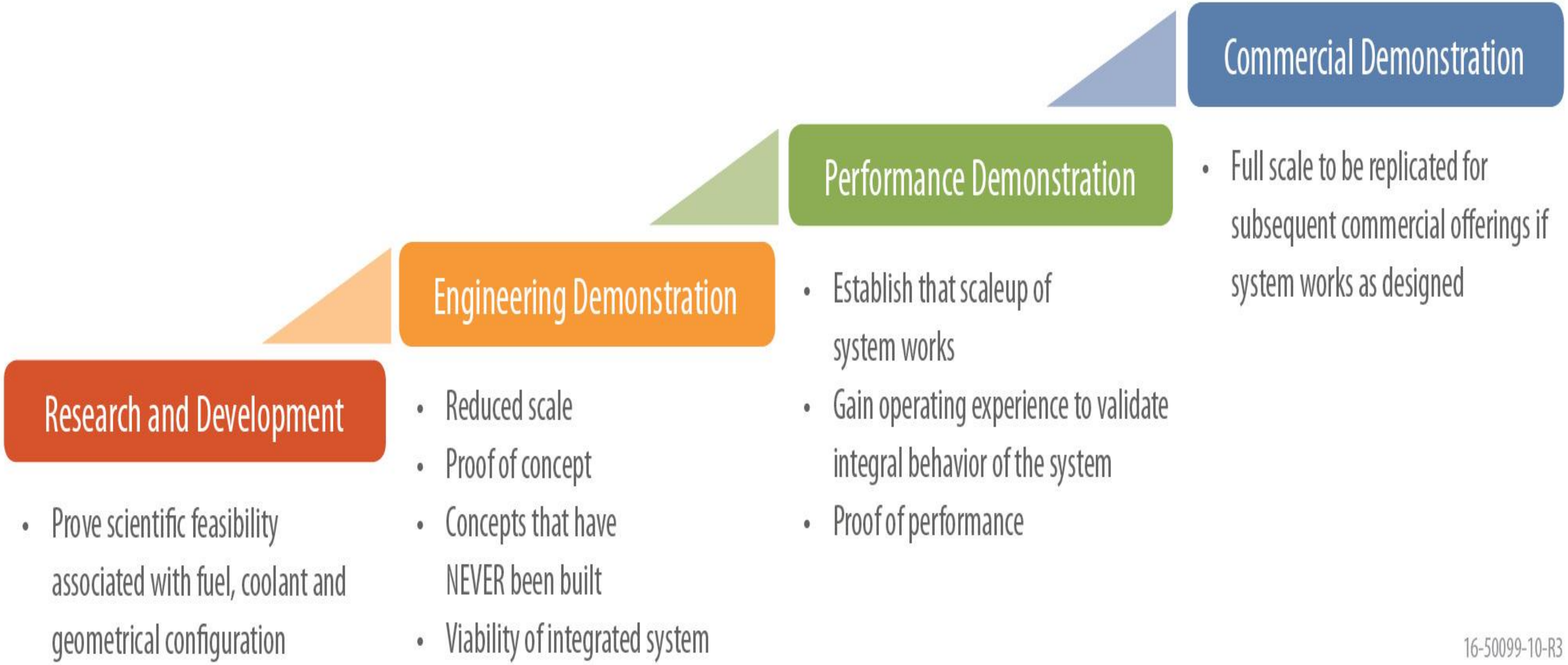
PRACS--Primary Reactor Auxiliary Cooling System

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



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

DEPLOYMENT EXAMPLES

Step in Deployment Path	Light Water Reactor (example)	Sodium Fast Reactor		High Temperature Gas-cooled Reactor		Lead Fast Reactor		Molten Salt Reactor	
	US	US	International	US	International	US	International	US	International
R&D for scientific feasibility	SPERT, BORAX, PBF	SEFOR (20 MWth), TREAT	CABRI						
Engineering Demonstration	S1W EBWR	EBR-I (1.4 MWth) EBR-II (20 MWe)	Dounreay (14 MWe), Rhapsodie (40 MWth),	Peach Bottom (40 MWe)	DRAGON (20 MWth), HTR-10 (10 MWth), HTTR (30 MWth), AVR (15 MWe)		Soviet submarines[a]	Aircraft Reactor Experiment (2.5 MWth); MSRE (7.4 MWth)[b]	
Performance Demonstration	USS Nautilus, Shippingport	Fermi-1 (69 MWe); FFTF (400 MWth)	Phenix (233 MWe), Monju (300 MWe), BN-300 and BN-600 (300 and 600 MWe), PFR (250 MWe)	FSV (842 MWth)[c]	THTR (750 MWth)[c]				
Commercial Demonstration	Yankee Rowe (485-600 MWth)		Superphenix (3000 MWth), BN-800 (800 MWe)		HTR-PM (200 MWe)				

[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

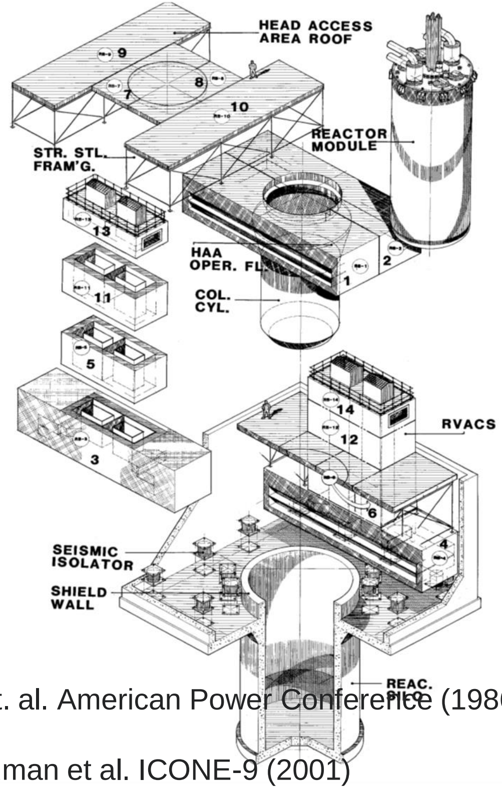
- Early power reactors were focused on performance demonstration of reliability, not cost reduction
  - BN-600 showed reliable operations, but not a commercial demonstration
- Many design refinements introduced in 1980s-2010
  - Design simplification, system integration and compaction, modular fabrication, 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



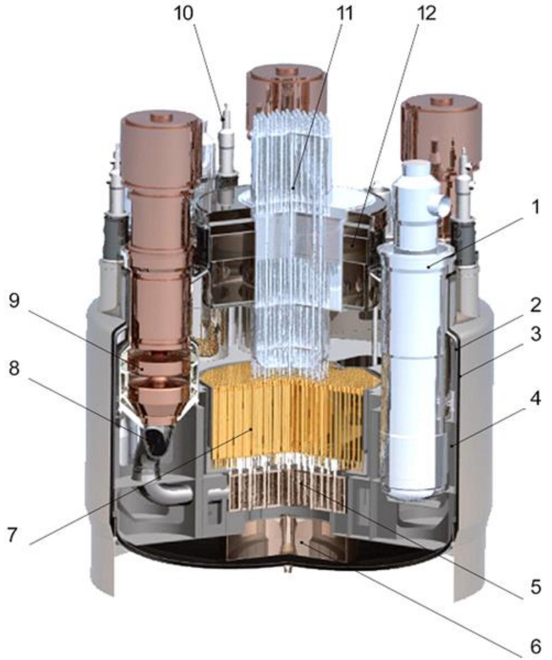
PRISM: A Passively Safe, Economic, and Testable Advanced Power Reactor, Tippets, et. al. American Power Conference (1986)
Optimizing the Size of the SUPER-PRISM Reactor, Boardman et al., ICONE-8 (2000)
Economic Assessment of S-PRISM Including Development and Generation Costs, Boardman et al. ICONE-9 (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



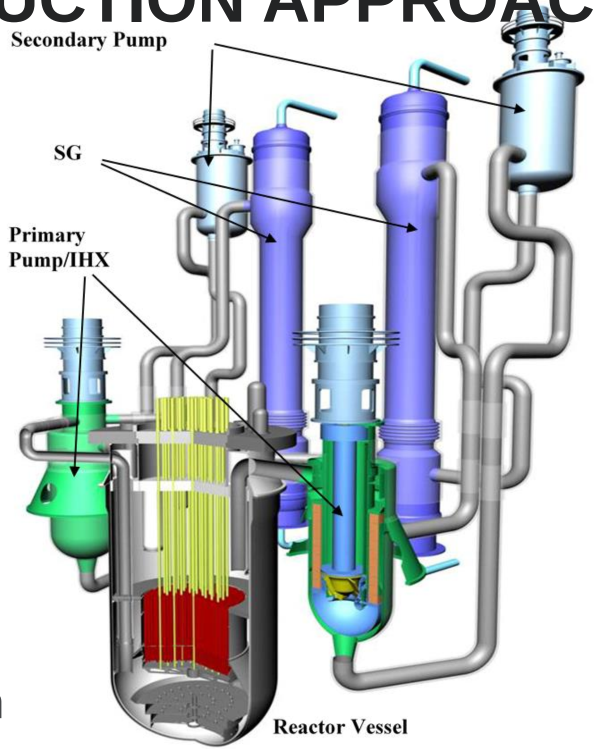
3D Layout of the BN-1200 Primary System

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

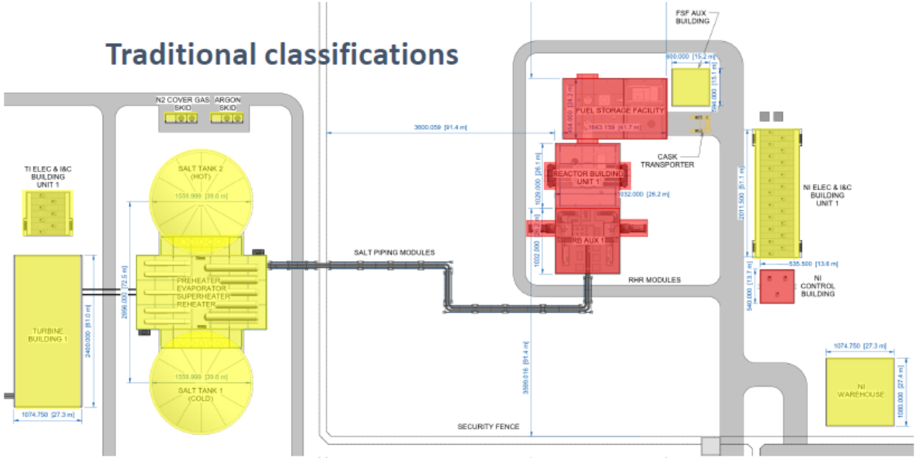
Design Features and Cost Reduction Potential of JSFR, Katoh et. al., NED Journal (2014)

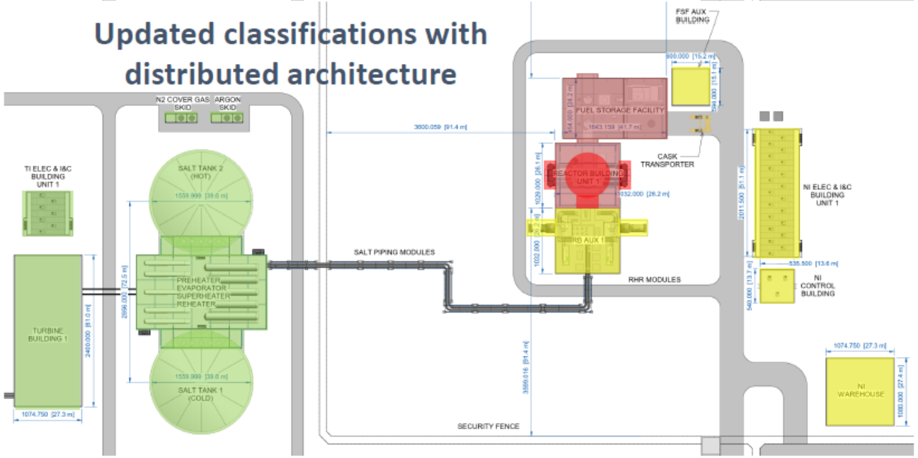
Resulting benefits

- 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

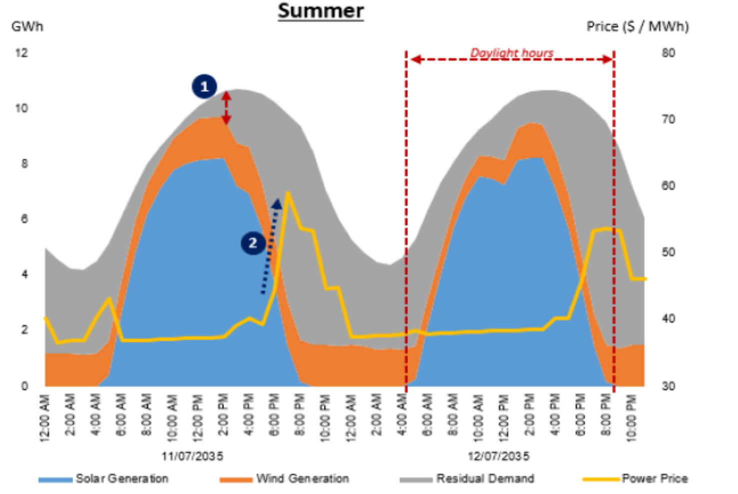




Natrium Reactor Storage and Ramping Balance a Renewables-Based Grid

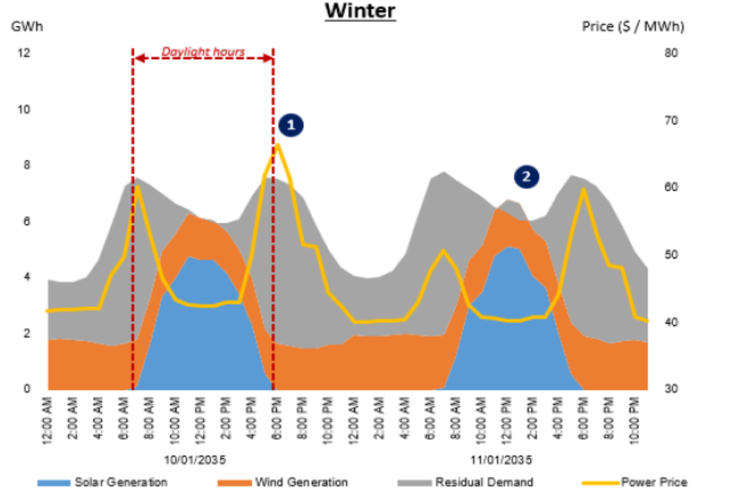
Significant price volatility from solar daily / seasonal variability - WECC Region

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



1. Limited generation needs as wind + solar can almost satisfy all demand during the day
2. 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



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

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

BACKUP SLIDES

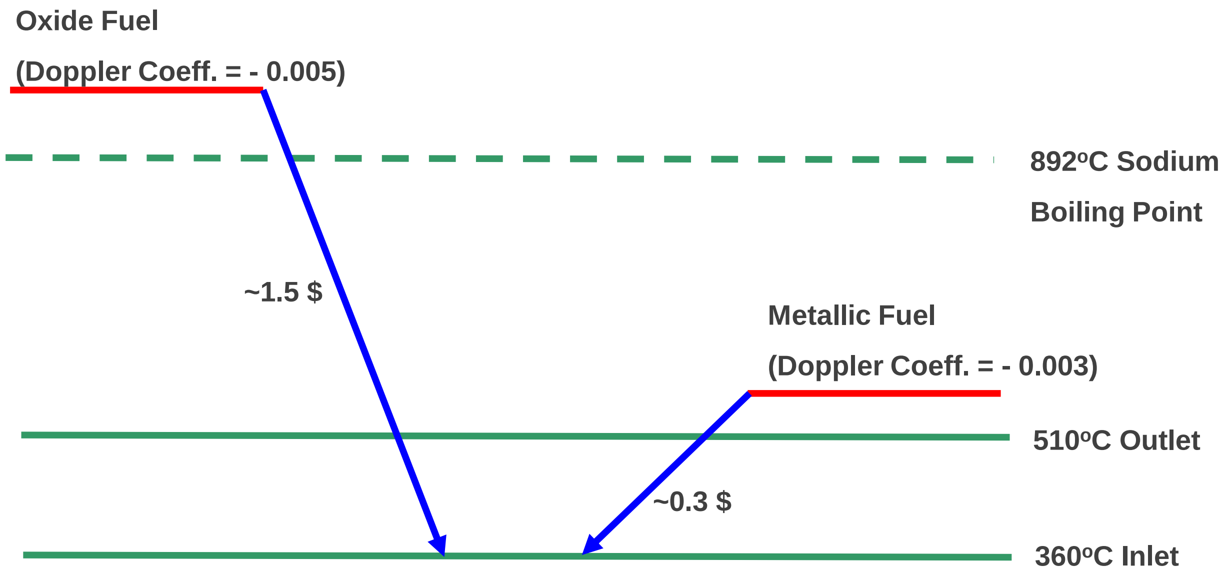
SODIUM VOID WORTH

For context,

- SFRs operate at low pressure with significant margin to boiling (~350^{o}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



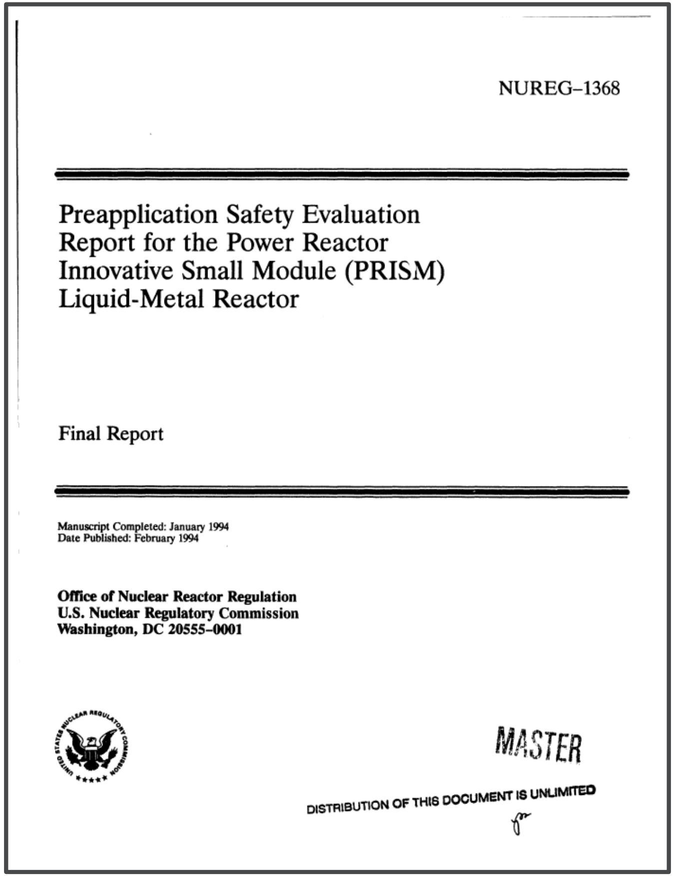
COOLANT VOID WORTH

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



NUREG–1368

Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor

Final Report

Manuscript Completed: January 1994
Date Published: February 1994

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

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