OAK RIDGE National Laboratory

Canadian Nuclear Safety Commission Information Seminar on Sodium Fast Reactors

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Module 1 – Introduction

ORNL is managed by UT-Battelle, LLC
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U.S. DEPARTMENT OF ENERGY

Seminar is Arranged by Modules

- Module 1—Introduction
- Module 2—SFR Neutronics
- Module 3—SFR Coolants and Thermal Hydraulics
- Module 4—Fuel Characteristics
- Module 5—SFR Systems and Components
- Module 6—Safety and Accident Analysis
- Module 7—Licensing Issues
- Module 8—Containment Systems
- Module 9—Selected SFR Operating Experience
- Module 10—Summary of PWR, 4S, and PRISM Characteristics

The seminar modules are is arranged by topic. Information in the modules may be repeated when relevant to other modules.

Sodium Fast Reactor CNSC information Seminar Objectives

Provide an understanding of

- Sodium fast reactor (SFR) technology
- Experience related to safety to support CNSC assessment, review, regulation, and licensing of SFR systems

Why A Sodium Fast Reactor?

Positive Aspects

- Faster neutron spectrum can be used for breeding or transmutation of transuranic waste products
- Sodium is an excellent heat transfer agent with high-temperature boiling point
- Sodium allows low-pressure components
- Sodium allows high-thermal efficiency and superheated steam conditions

Positive Aspects of Sodium Fast Reactors (continued)

- Sodium is compatible with structural components (stainless steels) and metallic fuels
- Electromagnetic pumps and flow instrumentation are possible
- Large supply of sodium is available due to extensive use in chemical industry
- Internal breeding of plutonium can result in long times before refueling
- Large database from EBR-II, CRBR and FFTF programs (but many years old)

SFR technology was chosen over other coolants (see subsequent slides) for breeder reactors. Subsequently, the slower growth of nuclear reactors and ample supplies of uranium made the need for breeders less urgent. SFRs also offer a way of burning transuranic fission products and minimizing the capacity needed for permanent repositories.

Challenging Aspects of Sodium Fast Reactors

- Requires compact core to minimize slowing down of neutrons due to sodium moderating effect
- High temperatures can lead to thermal stresses and creep in structural materials
- Sodium burns on contact with air or water—requires inert cover gas
- Sodium containing systems must be in inerted cells and concrete must be steel lined
- Sodium freezes at 100°C and components and piping must be trace heated

Sodium-cooled fast reactors, however, do have some disadvantages that require appropriate design considerations. These will be addressed in the subsequent slides and modules, with a view toward future regulatory actions.

Challenging Aspects of Sodium Fast Reactors (continued)

- Core is not in most reactive configuration—neutronic excursions possible with core compaction (severe accidents)
- Sodium and sodium-oxide and -hydroxide aerosols are not benign
- Sodium is opaque making ISI and refueling difficult
- No established regulatory experience in Canada
- Sodium is incompatible with oils and most foreign materials
- Sodium fast reactors require higher initial fuel enrichment

History of SFRs

- Liquid-metal fast reactors were among the first to be designed, built, and operated
- The first ever electrical power was generated by EBR-I, which fed into the power system for Arco, Idaho
- Most of the initial designs were intended to support the development of breeder reactors
- More detailed information on certain plants is provided in Module 9 – Selected SFR Operating Experience
- Detailed information for SFR power levels, temperatures, configuration, and dates of operation are given in IAEA "Fast Reactor Database," IAEA-TECDOC-1531, 2006

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Module 1 – Introduction

Details of all SFRs can be found in reference 1.

Ref 1. International Atomic Energy Agency, Fast Reactor Database, 2006 Update, IAEA-TECDOC-1531, December, 2006.

Summary of LMR Experience

Plant	Dates of major events				
	Start of construction	First criticality	First electricity generation	First full-power operation	Final shutdown
Rapsodie (France)	1962	January 1967		March 1967	April 1983
KNK-II (Germany)		October 1972	April 1978	1978	October 1991
FBTR (India)	1972	October 1985	1994	1996	
PEC (Italy)	January 1974	Project canceled			
JOYO (Japan)	February 1970	July 2003		October 2003	
DFR (UK)	1954	1959	1962	1963	1977
BOR-60 (Russian Federation)	1964	1968	1969	1970	
EBR-II (USA)	June 1958		August 1964	1965	1998
Fermi (USA)	August 1956	August 1963	August 1966	October 1970	1975
FFTF (USA)	June 1970	February 1980		December 1980	1996
BR-10 (Russian Federation	1956	1958		1959	December 2003
CEFR (China)	May 2000	To be determined			

Ref: International Atomic Energy Agency, *Fast Reactor Database, 2006 Update*, IAEA-TECDOC-1531, December 2006.

Ref: International Atomic Energy Agency, Fast Reactor Database, 2006 Update, IAEA-TECDOC – 1531, December 2006, p14.

Summary of LMR Experience (continued)

Plant (demonstration or prototype fast reactors)	Dates of major events				
	Start of construction	First criticality	First electricity generation	First full-power operation	Final shutdown
Phenix (France	1968	1973	1973	March 1974	
SNR-300 (Germany	1973, finished in 1985; in 1991 the Government announced that SNR-300 should not proceed to commence operation				
PFBR (India)	2003	To be determined			
MONJU (Japan)	1985	1994	1995		
PFR (UK)	1966	1974	1975	1977	March 1994
CBRBP (USA)	Project canceled				
BN-350 (Kazakhstan)	1964	November 1972	1973	Mid-1973	April 1999
BN-600 (RF)	1967	February 1980	April 1980	December 1981	Not determined
ALMR (USA)	Not determined				
Kalimer-150 (Republic of Korea)	Not determined				
SVBR-75/100 (RF)	To be determined				
BREST-OD-300 (RF)	To be determined				

BN-600 is discussed in more depth in the Module 9 – Selected SFR Operating Experience

Summary of LMR Experience (continued)

Plant (commercial size reactors)	Dates of major events				
	Start of construction	First criticality	First electricity generation	First full-power operation	Final shutdown
Super-Phenix 1 (France)	1976	1985	1986	1986	1998
Super-Phenix 2 (France)	Project subsumed into EFR				
SNR 2 (Germany	Project subsumed into EFR				
DFBR (Japan)	Not determined				
CDFR (UK)	Project subsumed into EFR				
BN-1600 (RF)	Project subsumed into BN-1800				
BN-800 (RF)	2002	2012 (planned)	2014		
EFR	Not determined				
ALMR (USA)	Not determined				
SVBR-75/100 (RF)	Not determined				
BN-1800 (RF)	Not determined				
BREST-1200 (RF)	Not determined				
JSFR-1500 (Japan)	Not determined				

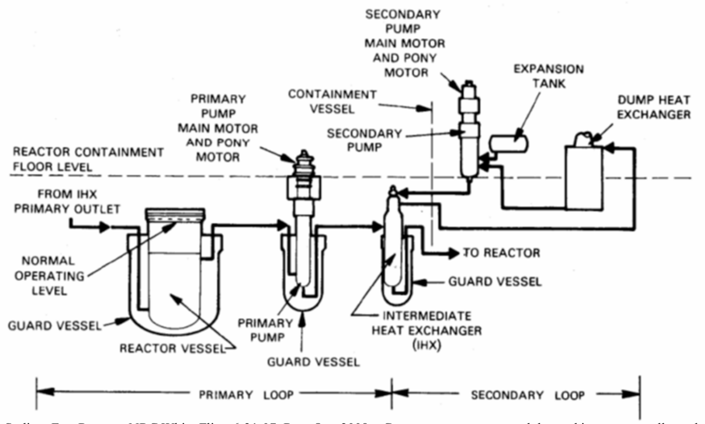
These slides indicate that many advanced nations have pursued SFR breeder technology, but the various European efforts have been subsumed into the EFR program, which has not led to a construction decision.
The CRBR (Clinch River Breeder Reactor) design was significantly completed when it was cancelled.

Loop and Pool Designs

- Two major types of layouts have been considered for SFRs—LOOP and POOL types. These are described in more depth later.
- The LOOP type has the components separated and connected by piping, where the POOL type has the core, pumps, intermediate heat exchangers, etc., internal to a primary sodium pool

Example of Loop Configuration

FFTF Loop-Type Primary and Secondary Systems

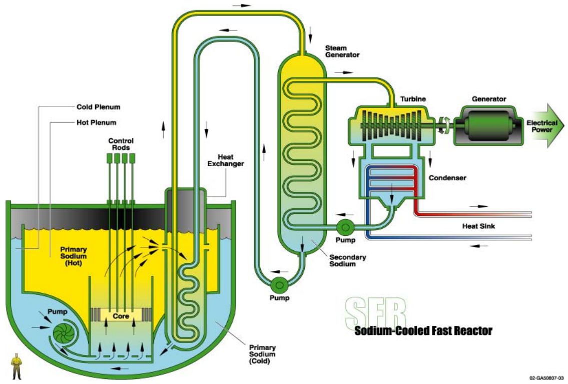


Ref: J. Cahalan, Sodium Fast Reactor, NRC White Flint, 6-21-07, Rev. Oct. 2008—Components are separated, housed in separate cells, and connected by piping.

Ref: Cahalan, J., Sodium Fast Reactor, NRC White Flint, 6-21-07 rev Oct 2008
Components are separated, housed in separate cells, and connected by piping

Example of Pool Configuration

- Core, primary piping, IHX, and primary pumps are in a pool of sodium



Steam generator and balance of plant are external to pool, connected through an intermediate heat transfer system.

Ref: http://ne.doe.gov/geniv/document/gen iv roadman/pdf

Reference: http://ne.doe.gov/gen iv/document/gen iv roadmap/pdf
Core, pumps, primary sodium, IHX, in pool of sodium. Only steam generator is external to pool.

Key SFR Characteristics Compared to LWRs

COOLANT FLUID

Liquid sodium (SFR)

- Highly reactive with air or water
- High boiling point (980°C at 1 atm)
- Low pressure operation ~1 atm (0.1 MPa pump head plus static head)
- Very high thermal conductivity
- Compatible with appropriate structural materials (mostly stainless steels)
- Activates to ^{24}Na and ^{22}Na under neutron irradiation
- Opaque

Water (LWR)

- Compatible with air
- Low boiling point (100°C at 1 atm)
- High-pressure operation (14 MPa for PWRs, 10 MPa for BWRs)
- Low thermal conductivity
- Highly corrosive (requires Zircaloy cladding)
- Transport of mostly corrosion products
- Transparent

Key SFR Characteristics Compared to LWRs (cont.)

PLANT SYSTEMS

SFR

- Low-pressure operation
- Thin vessels
- Need for nonreactive cover gas
- Opaqueness and chemical reactivity complicates fuel handling—generally done with vessel head in place
- High temperatures allow more efficient power conversion system
- Requires intermediate coolant loop
- No concerns for thermal shock
- Can use electromagnetic pumps and flow-meters
- May need design features against core disruptive accident
- Must provide trace heating to preclude solidification of sodium
- Need steel-lined cells to preclude sodium-concrete interactions after a sodium leak
- Can use a pool-type or loop-type layout

LWR

- High-pressure operation
- Thick vessels
- No need for inert cover gas (steam in pressurizers)
- Fuel handling easier because of "direct" access (vessel head off)
- Temperatures limited by water pressurization limits
- Intermediate coolant loop used in pressurized-water reactors (PWRs) only
- Thermal shock is an issue with certain emergency core cooling system (ECCS) operations
- May need to preclude meltdown but no energetic nuclear excursion
- Dryout of zirconium cladding could produce excessing amounts of hydrogen, possibly leading to explosive mixtures
- Heat tracing required for PWR systems with high boron content
- Loop-type layouts almost universally used except for new integrated SMR designs and the iPWR

Key SFR Characteristics Compared to LWRs (cont.)

FUEL DESIGN/NEUTRONICS

SFR

- Core is not in most reactive configuration (compaction can lead to prompt critical state)
- Fuel configuration maximizes fuel/coolant ratios
- Higher initial enrichment and plutonium concentrations
- Hexagonal pitch
- Can treat core as homogeneous
- Can use stainless steel claddings
- Fuel can be oxide or metal
- Uses large fission gas plenum for substantially higher burnup
- Must be arranged to limit voids near center of core (could increase reactivity)
- Plutonium has fewer delayed neutrons and shorter time for control action
- Fuel and cladding does not react with coolant
- Metal fuel can create a low melting point eutectic with cladding (generally uses sodium filler inside cladding)

LWR

- Core is in most reactive configuration
- Fuel configuration optimizes water moderation
- Generates plutonium with burnup but starts with ~3-4% ^{235}U enrichment (some reactors use mixed oxide fuel)
- Square pitch
- Must account for local effects in neutron analysis
- Must use zirconium (low-neutron absorption) cladding
- Uses small fission gas plenum
- Voids in core tend to reduce reactivity
- Zirconium in cladding can react with water at high temperatures (1200°C) and generate hydrogen
- PWRs use boric acid solutions for reactivity control, which can cause corrosion

Key SFR Characteristics Compared to LWRs (cont.)

THERMAL HYDRAULICS

SFR

- Sodium has excellent heat transfer characteristics
- Allows minimum sodium inventory in the core
- Relatively low melting point (~100°C)* and high boiling point (980°C)*
- Allows low-pressure operation
- Fluid is easy to pump—flow properties are like water—requires much less pumping power than lead, mercury, or bismuth
- Tolerant of partial flow blockages
- Can use wire wrap or grid spacers
- Electrically conductive, so can use electromagnetic pumps
- Free surfaces must have inert cover gas
- Dominated by thermal conduction (low Peclet number)
- Should minimize sodium boiling and void formation
- Amenable to natural circulation

LWR

- Water has relatively poorer heat transfer characteristics
- Water is also the neutron moderator; must be in sufficient quantities to moderate neutrons
- Water must be pressurized to remain liquid at reactor conditions (~14 MPa)
- Hot channel factors must be evaluated (local boiling issue in PWRs)
- Can be used as direct cycle in boiling-water reactors (BWRs) (boiling in the core)
- Fluid is generally pumped, but some new designs use natural circulation

Key SFR Characteristics Compared to LWRs (cont.)

TRANSIENTS AND SAFETY CONSIDERATIONS

SFR

- Plutonium has shorter time-delayed neutrons—requires faster shutdown system
- Voids near center of reactor can cause reactivity increase
- Compaction can lead to a more reactive configuration
- May need to design against a core disruptive accident
- Sodium can burn in contact with air or water—forms aerosols
- Doppler limits on reactivity excursions more important if fertile and fissile materials are close to each other
- Doppler enhanced in sodium systems because of spectrum softening compared to other fast reactor spectrums
- Material growth (thermal expansion)—should design for reactivity control during heat up
- Fast neutron fluence can exacerbate material swelling and bowing
- Sodium must be prevented from contacting concrete
- Sodium must be prevented from freezing (~100°C)
- Fission products (except for rare gases may be transported as sodium aerosols
- Radial expansion of core introduces negative
- Fuel movement can introduce strong reactivity effects
- Pool-type designs with large primary sodium inventory have slower response to transients

LWR

- Core already in optimal geometric configuration
- Removal of water stops neutron reaction (but makes decay heat removal a challenge)
- Core can melt down
- Large reactors require post-accident heat removal systems and backup emergency power
- Containment design must accommodate pressure increase due to release of high-pressure water
- Fission products probably would transport largely independently of water and steam
- Dryout of fuel can rapidly generate hydrogen that can challenge containment integrity
- Response to transients can be rapid

Key SFR Characteristics Compared to LWRs (cont.)

CONTAINMENT

SFR

- May need to contain beyond design basis accidents
- May need to include inerted enclosures against sodium leaks
- May need to limit and contain sodium spills
- Cannot use water for post-accident heat removal
- Need to prevent contact of sodium with concrete
- May be able to accommodate post-accident heat removal (PAHR) by natural convection
- May not allow hands-on maintenance
- Can benefit from modern accident identification methods risk-informed approach)

LWR

- Designed for large break loss-of-coolant accident (LOCA)
- Not designed for beyond design basis accidents
- Must depend on ECCS systems and electrical power for post-accident heat removal (possible exception for Gen-III+ type)
- May need to accommodate hydrogen combustion
- Allows hands-on maintenance
- Reduced size containments (ice condensers and suppression pool types) may have inadequate volumes
- New designs can benefit from modern accident identification methods risk-informed approach)

Key SFR Characteristics Compared to LWRs (cont.)

OPERATING EXPERIENCE

SFR

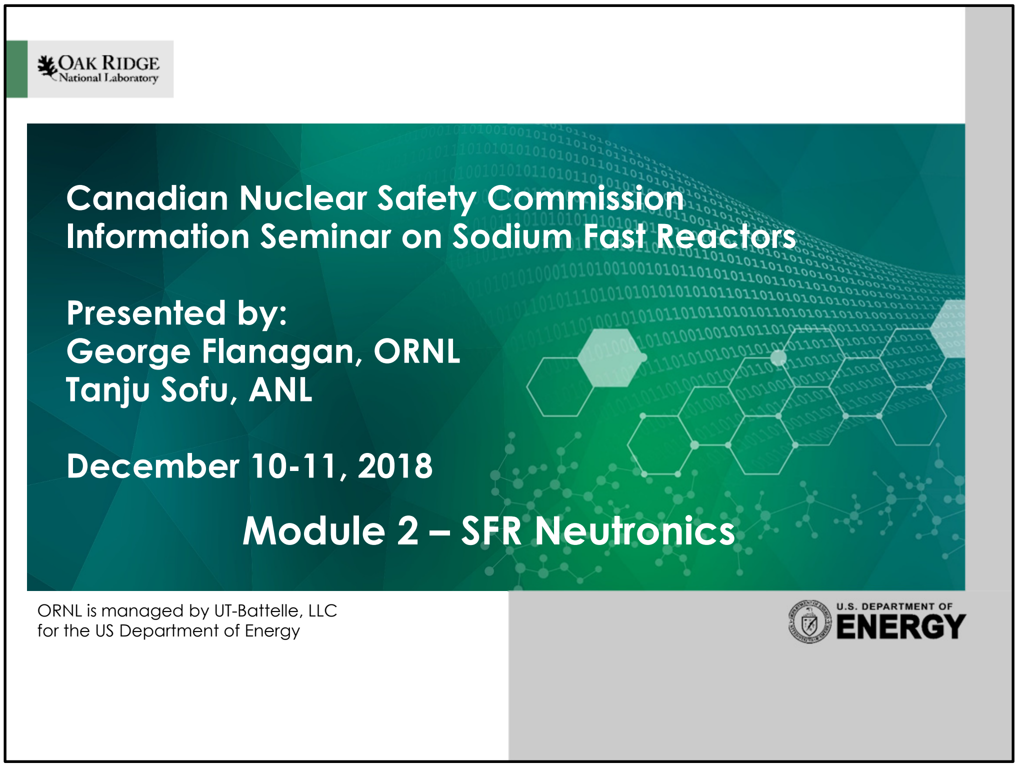
- SFRs have been operated by the United States, United Kingdom, France, Germany, India, Japan, Kazakhstan, Russia, and China
- Some have experienced sodium leaks, problems with fuel handling, steam generator problems, etc.
- EBR-I experienced core melt due to core compaction
- Fermi-1 experienced fuel assembly melting and its propagation due to total inlet blockage
- EBR-II and FFTF had deliberate unprotected core heatup with no damage
- FFTF and EBR-II ran well; others with leaks ran well with accommodation

LWR

- Generally good experience except for TMI-2 and Fukushima
- TMI-2 core melt due to improper training of operating staff, which shut down coolant makeup to the core—little release of radioactivity. This was a small break LOCA.
- Fukushima accident was largely due to external event larger than design basis and poor placement of emergency diesel, fuel tanks, and electrical systems. This resulted in a long-term station blackout (ESBO) (BDBA).
- Current designs can be susceptible to ESBO
- New designs, such as iPWR, have 72-hour loss-of-offsite electrical power capability

Suggested Reading

- Alan E. Waltar and Albert B. Reynolds, *Fast Breeder Reactors*, Pergamon Press, 1981
- International Atomic Energy Agency, *Liquid Metal Cooled Reactors: Experience in Design and Operation*, IAEA-TECDOC-1569, December 2007
- International Atomic Energy Agency, *Fast Reactor Database, 2006 Update*, IAEA-TECDOC-1531, December 2006
- John Graham, *Fast Reactor Safety,* Academic Press, 1971.
- Thomas Cochran et al., "Fast Breeder Reactor Programs, History and Status," International Panel on Fissile Materials, February 2010



A major reason for developing fast reactors was to take advantage of the fast neutron spectrum that could be used to breed fuel (plutonium from U238 in this case). A more recent objective is that the fast spectrum could be used to transmute “burn” long lived transuranic fission products. However, the characteristics required to obtain the fast spectrum result in a reactor significantly different from present generation light water reactors.

Neutronics Basic Principles

- SFRs have a fast-neutron spectrum, which results in significant differences from LWRs
- No deliberate neutron moderators—results in a "fast" or "hard" neutron energy spectrum compared to LWRs
- Takes advantage of high energy fission cross sections and smaller parasitic capture cross sections at high-energies
- Liquid metal, typically sodium, used as coolant, ~100 times more effective heat transfer medium than water
- Nominal coolant (sodium) operating conditions are far below the boiling point along with low vapor pressures to allow low-operating pressure

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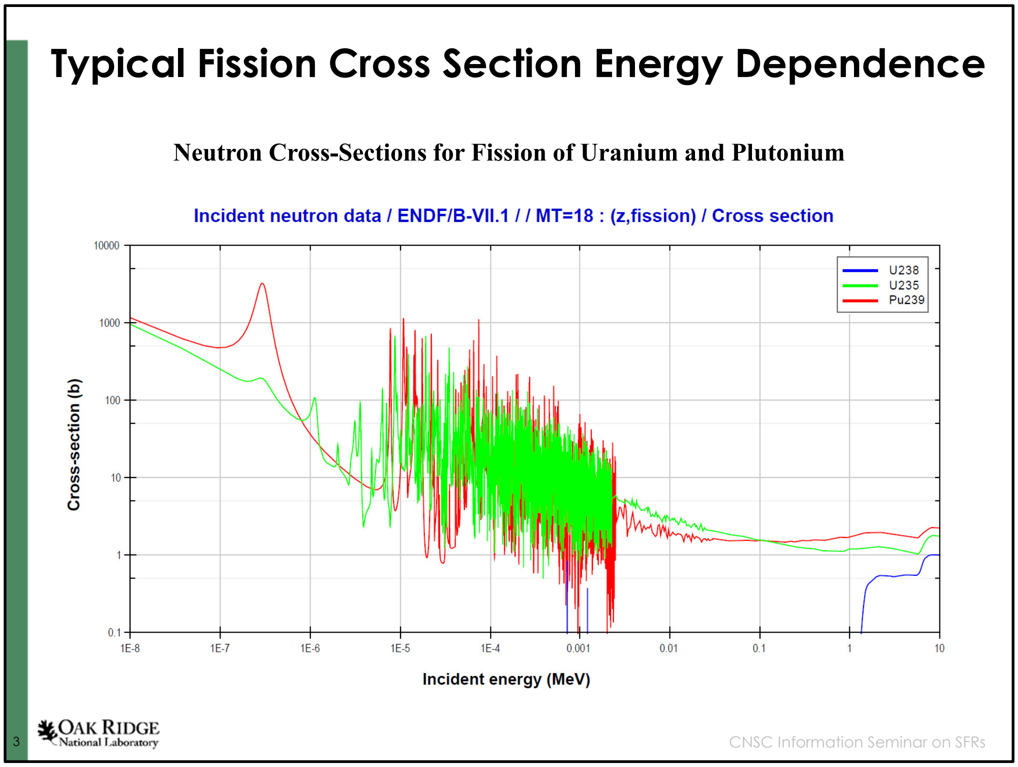
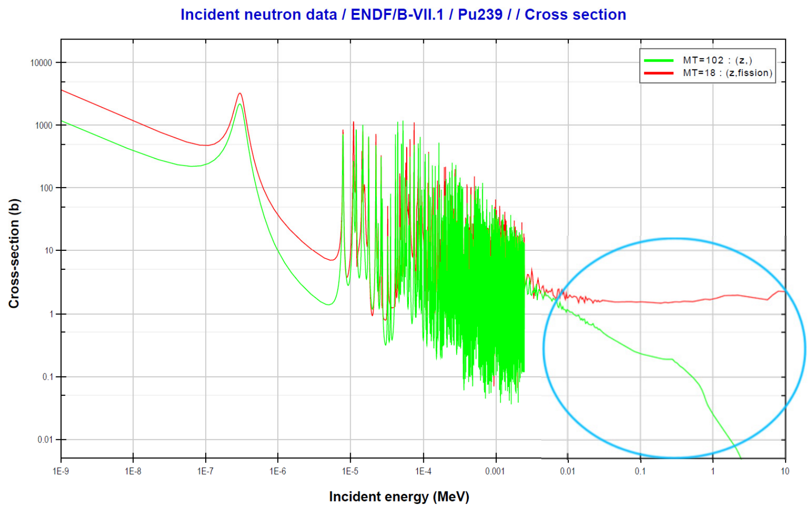
A major reason for using sodium as a coolant is that sodium does not slow neutrons down appreciably (to a much less extent than water). Sodium has the additional advantages beyond its reduced moderation of neutrons in that it has a high boiling point, allowing thinner components and very high heat transfer characteristics.

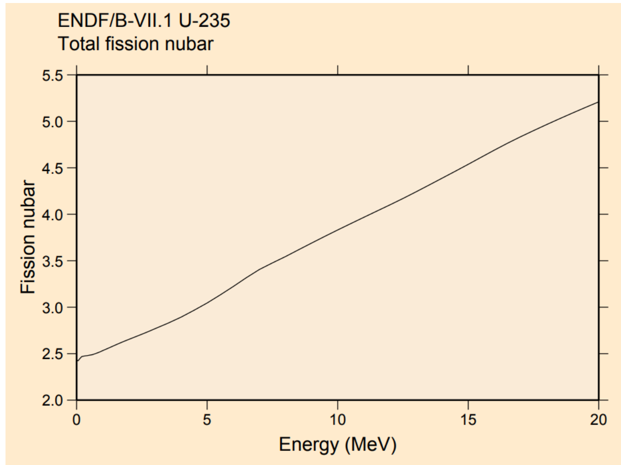


Note: The fission cross section decreases with increasing neutron energy, but U 238 becomes fissionable at high neutron energies. At higher neutron energies, the fission cross section becomes greater than the capture cross sections and the number of neutrons per fission increases. This allows fast reactors to operate at fast neutron spectra.

Fission Cross Section More Dominant at High Energies

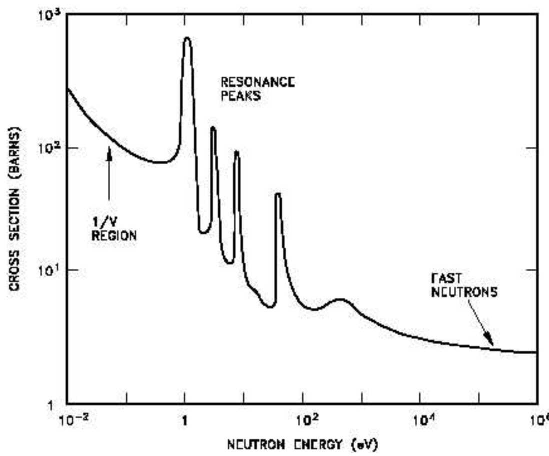


Typical Neutron Yield from Fission Energy Dependence



Neutron yield from fission increases with increasing neutron energy. Neutron spectrum is also more energetic than for water-cooled reactors.

Typical Capture Cross Section Energy Dependence



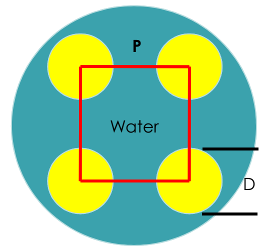
Ref: Sterling Bailey, *Industry Perspectives and Experience in the Design of Liquid Metal Cooled Reactors*, Appendix C.3 of *NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors*, NUREG/KM-0007, ORNL/TM-2013/79, G. F. Flanagan, et. al., April 2014.

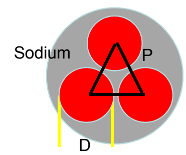
Ref: Sterling Bailey, *Industry Perspectives and Experience in the Design of Liquid Metal Cooled Reactors*, Appendix C.3 of *NRC Program on Knowledge Management for Liquid-Metal-Cooled Reactors*, NUREG/KM-0007, ORNL/TM-2013/79, G. F. Flanagan, et. al., April 2014.

Note that the capture cross section decreases with increasing neutron energy; most resonances are at lower energies

LWR vs LMR Lattice

- In an LWR, water acts as both a coolant and a moderator. An optimal P/D ratio is adjusted so that
  1. Adequate moderation is obtained (i.e., not undermoderated and not overmoderted)
  2. Generated nuclear heat is effectively removed by the coolant
- In an LMR, no moderation is needed. Sodium acts only as a coolant. Because of excellent cooling properties of sodium (liquid metals in general), fuel pins can be placed much closer on a triangular pitch
- Fuel details to be provided in Fuel Characteristics module





Since no moderation is needed, the fuel spacing can be reduced to as low as possible, but still allowing for the heat to be removed from the fuel pins by the coolant. This leads to a hexagonal array for the SFR fuel pins vs a square array for LWRs, because the moderator to fuel ratio in an LWR is critical to its optimum operation.

Why Fast Spectrum

- Fast spectrum reactors were originally intended for breeding
- Fast spectrum also allows burnup of transuranic fission products
- Breeding can be achieved in a thermal reactor if it were fueled with ^{233}U
- For other fissile material, breeding can only be sustained in a reactor with fast spectrum

Fast Spectrum

- Difficulty with fast spectrum is that the fission cross section is too low in comparison to thermal range
- Therefore, fast reactors require more initial U fissile mass than LWRs to sustain a chain reaction
  - Relatively high enrichment ~15%

This is an important point. Whereas LWRs operate with enrichments of less than 4-5%, SFRS must operate with higher fissile fuel loadings, usually using plutonium.

Design Impacts of Fast Spectrum

- Very high neutron speed and lower cross sections at higher energies lead to much longer neutron mean free path
- Therefore, fast reactors can be analyzed as homogeneous (as compared to LWRs for which the impact of local heterogeneities is much more important)
- Higher flux level required for equivalent reaction rate
- Power distribution is flatter and less sensitive to local geometry

Design Impacts of Fast Spectrum (continued)

- Higher leakage can make reactivity more sensitive to dimensional changes
- Mid-energy ^{238}U resonances contribute to significant Doppler reactivity coefficient (Doppler feedback is improved with oxide fuels and with sodium coolant because of slower spectrum)
- Fission cross section also has resonance regions, which can result in reactivity increase due to Doppler effect, but net effect of Doppler is negative in fast-reactor designs
- Lower reactivity reduction with burnup
- Shorter neutron lifetime and reduced delayed neutron fraction may impact dynamic behavior

Important safety issues are affected by the fast neutron spectrum. Dimensional changes of the core affects the reactor response to nuclear transients. Also, the Doppler effect is important in limiting the increase in power during nuclear excursions. Sodium, because it is a moderator of sorts because of its low atomic weight, and oxide fuel tend to increase the Doppler effect and thus have a positive effect on safety issues.

Conversion vs Breeding

Conversion

- Conversion Ratio (CR)

  [fissile mass produced] ÷ [fissile mass consumed]

The ratio is called CR if <1

Typical LWR:

CR $\cong$ 0.6

Breeding

- Breeding Ratio (BR)

If the ratio is >1, then more fissile material is produced than is consumed

For CR >1, it is commonly called BR

LMFBR Goal: BR ~1.4

(Transuranic burnup goal TBD)

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Note: Burnup goals for transuranic transmutation TBD

Breeding and Burning

- For breeding and burning to take place, the reactor is divided into two regions:
  - Active "core" region (high-fissile content)
  - "Blanket" region (high-fertile content or burnup target material)
- Neutrons leaking from the core are absorbed in the blanket material, converting the fertile isotopes into fissile isotopes, thereby creating more fissile inventory or at least maintaining it or burning target fission product

Fast reactors have a "blanket" for breeding or burning that is peripheral to the core, whereas LWRs generate some plutonium internally to the fuel elements. In some designs, such as the 4S, the "Blanket" is basically a reflector that conserves neutrons and is used to control reactivity.

Breeding Definitions

- Fissile nuclides
  - Nuclides that can be induced to fission with neutrons of essentially zero kinetic energy (thermal neutrons)
- Fissionable nuclides
  - Nuclides that can only be fissioned with energetic "fast" neutrons
- Fertile nuclides
  - Nuclides that can be transmuted into fissionable nuclides via neutron capture

Burnup in SFRs

- Breeders have high initial fissile loading
- For economic reasons, fuel must tolerate much higher burnup
  - Typical LWR burnup ~2-3 atom-%
  - SFRs can achieve burnups well in excess of 10 atom-%, which corresponds to thermal energy generation of Q = 150,000 MW_d/T_e
- High burnup requires consideration of possible swelling of cladding (HT9 steel is now the most commonly used) and/or distortion of fuel assembly
  - Swelling of cladding may cause fuel failure and assembly bowing might result in alteration in core configuration
  - Irradiation experiments show that fuel HT9 cladding can withstand 300,000 MWd/T_e

Ref: S. F. Hayes and D. L. Porter, SFR Fuel Performance and Approach to Qualification, DOE/NRC Seminar Series on Sodium Fast Reactor Fuel Performance and Qaulification, Nov 27-28, 2007.

CRBR and similar vintage designs used oxide and mixed oxide (containing Pu) fuels. EBR II irradiation experience showed advantages of HT9 cladding and metallic fuels

The need for higher burnup values has exacerbated the issue of swelling of fuel cladding and duct materials. HT9 appears to be adequate in this respect. (HT9 is ferritic-martensitic iron alloy 12Cr-1Mo-0.3V-0.5 W)

DPA = Displacements per atom – a measure of radiation “damage”.

Possible Burnup Effects

- Possible change in core configuration is of more concern in LMRs than in LWRs due to higher fissile content and greater sensitivity to leakage of neutrons with much longer mean free path
  - Changes in core geometry may result in increase or decrease in local multiplication factor
- Local jump in temperature might also lead to sodium boiling, which effectively reduces the slight moderation and absorption by sodium. This may increase the multiplication factor if it occurs near the center of the reactor and decrease it if it occurs near the periphery due to higher neutron leakage.

Sodium voiding, particularly in the central zones of an SFR can have positive reactivity effect. This can lead to a core disruptive accident, which has been a major concern with early SMR safety analyses. Some configurations, such as a pancake design, have been proposed to reduce this effect but most designs actually built and operated have not used this configuration because of deleterious neutron economy.

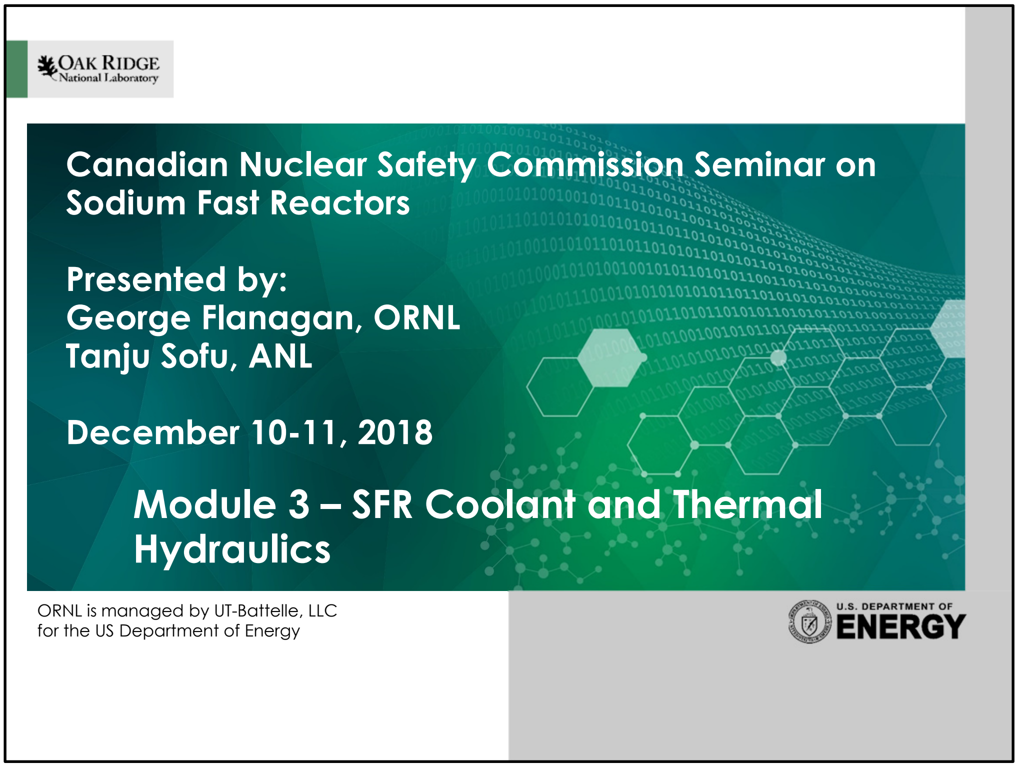
Possible Burnup Effects (continued)

- Wire wraps and grid spacers prevent pin-to-pin contact even at high burnup.
- Experiments have shown that sodium-cooled fuel assemblies are very tolerant of partial blockages because of the very high-thermal conductivity of sodium
- Extensive set of experiments at ORNL using electrical fuel pin simulators indicated that sodium fuel assemblies, particularly of the CRBR or FFTF type with wire wrapped pins, are highly tolerant of partial blockages

For example, see M. H. Fontana, et. al., Effects of Partial Blockages in Simulated LMFBR Fuel Assemblies, paper given at American Nuclear Society Topical Meeting on Fast Reactor Safety, Los Angeles, California, April 2-4, 1974, CONF-740401-P3 (with others).

Possible Burnup Effects (continued)

- The core structure design must prevent significant changes in geometry
- This is achieved by wire wraps and/or grid spacers and core restraint design
- Other intrinsic feedback mechanisms must balance coolant void effects (i.e., other feedback mechanisms must provide sufficient negative reactivity so that the overall feedback coefficient must be reasonably negative at elevated temperatures)
- Examples are core restraints that allow core radial expansion or fuel assembly bending so that the net reactivity is decreased
- Gas expansion modules (GEMS) have been added (e.g., FFTF) as passive devices to assure enhanced neutron leakage as primary flow rate is reduced



The coolant and associated thermal hydraulic issues are a major factor affecting the design and operation of SFRs. Sodium was chosen after evaluation of several other choices. Although it has considerable problems with compatibility with air and water, sodium is an excellent heat transfer agent, has relatively low neutron absorption, is easy to pump, and has low vapor pressure at high temperatures.

Thermal-Fluid Considerations

- Primarily due to neutronic requirements, fast reactor fuel lattice has to be kept compact
- This requirement results in higher power density than the conventional LWRs
- This stipulates that a coolant with better heat transfer capabilities be used for commensurate heat removal

Liquid metals are excellent candidates that effectively satisfy all the design objectives and requirements.

Because the neutron energy spectrum should be "fast", moderating materials should not be used in the SFR core. Sodium is somewhat moderating, so there should be as little as possible in the core. However, heat needs to be removed from the fuel, so some sodium is necessary for heat transfer. The result is that the fuel is packed as closely as practical, usually in a hexagonal pattern.

Coolant Alternatives

- Fast neutron spectrum requires that coolant with low *moderating power* be used
- From the neutronics standpoint, coolants with low mass number, such as those that contain hydrogen (e.g., H_2O) are not suitable
  - Energy loss in a collision event is inversely correlated to the mass number of the isotope

Collision parameter, α $$\alpha = \left(\frac{A-1}{A+1}\right)^2$$

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Collision parameter is defined as

$(E')_{min}$ = [*(A–1)/(A+1)]⊥2* E = *(alpha)* E

Coolant Alternatives (continued)

- Sodium (Na)
- Potassium (K)
- NaK (22% Na, 68% K)
- Li (92.5% ^{7}Li, 7.5% ^{6}Li)
- Lead (Pb)
- Mercury (Hg)

Coolant Alternatives (continued)

Isotope	Mass number	Collision parameter α
H*	1	0.0000
Li	7	0.5625
Na	23	0.8403
K	39	0.9025
Hg	202	0.9804
Pb	208	0.9810

*Given as a reference; not a liquid metal-coolant.

Coolant Alternatives (continued)

- Mercury, bismuth, and lead were eliminated due to their high densities that result in high-mass flow rates and very high pumping power requirements
- Lead–bismuth has problems with formation of ^{210}Po, which could exacerbate maintenance problems
- NaK solution, liquid at room temperature, was used in early designs; later eliminated due to high parasitic neutron absorption cross section of potassium. Also, NaK exposed to air can degenerate to an explosive compound that can be ignited by mechanical shocks
- Sodium was given the most serious consideration in the Fast Breeder Reactor Program

Activity Associated with Irradiation of Liquid Metals

Metal	Induced activity (Ci/g)
Sodium	0.20 (^{24}Na, ^{22}Na)
Potassium	0.11 (^{38}K, ^{42}K)
NaK (22% Na, 78% K)	0.11 (^{38}K, ^{42}K)
Li (92.5% ^{7}Li, 7.5% ^{6}Li)	0.03 (^{8}Li, no gamma)
Lead	0.09 (^{208}Pb, no gamma)
Mercury	0.28 (^{199}Hg)

* Ref: Tang et al., "Thermal Analysis of Liquid Metal Fast Breeder Reactors"

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Ref: Y. A. Tang, R. D. Coffield, Jr., and R. A. Markely, "Thermal Analysis of Liquid Metal Fast Breeder Reactors," American Nuclear Society (1978).
Sodium can be activated by neutron irradiation, forming Na 24 and Na 22. This requires that irradiated sodium needs to be allowed to decay (half life of 15 hours).

Thermal-Physical Properties of Liquid-Metal Coolants

Coolant	Vapor pressure [p (atm)]	Density (kg/m^3)	Viscosity (μPa . s)
Sodium	0.012	823	2.95
Potassium	0.08	714	2.2
NaK (22/78)	0.06	742	2.3
Lithium	Negligible	479	4.4
Lead	Negligible	1041	22
Mercury	12.2	1323	150
Water	~100	726	88.5

* Ref: Tang et al., "Thermal Analysis of Liquid Metal Fast Breeder Reactors"

Ref: Y. A. Tang, R. D. Coffield, Jr., and R. A. Markely, "Thermal Analysis of Liquid Metal Fast Breeder Reactors," American Nuclear Society (1978).
This table gives the properties under the conditions in the reactor during operation. For example, the value for water is for a pressure of 100 atm (~ 1500 psia). (Units of viscosity are micropoise [Pa. s]: 10E-2 Pa.s is one centipoise, which is the viscosity of water at 20 C and atmospheric pressure.)

Summary, Coolants, and Thermal Hydraulics

Key SFR Characteristics especially those different than those for LWRs

SFR

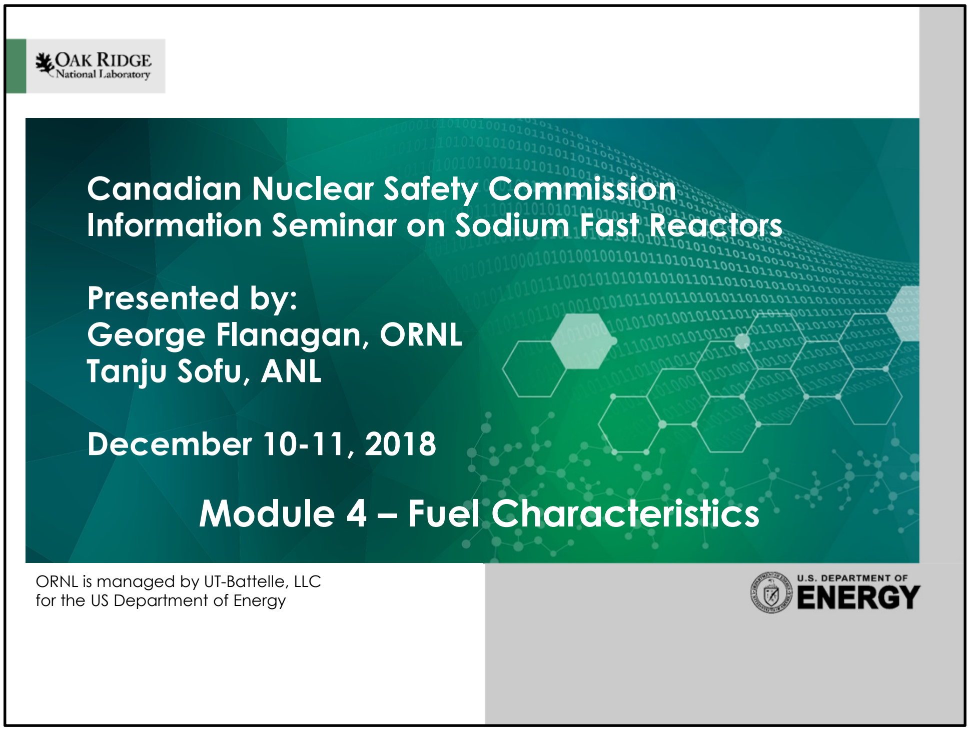
- Sodium has excellent heat transfer characteristics
- Allows minimum sodium inventory in the core
- Relatively low-melting point (~100°C)* and high-boiling point (980°C)*
- Allows low-pressure operation
- Fluid is easy to pump—flow properties are like water—requires much less pumping power than lead, mercury, or bismuth
- Highly tolerant of partial flow blockages
- Can use wire wrap or grid spacers
- Electrically conductive so can use electromagnetic pumps and flowmeters
- Free surfaces must have inert cover gas
- Dominated by thermal conduction (low Peclet number)
- Should minimize sodium boiling and void formation
- Amenable to natural circulation

LWR

- Water has relatively poorer heat transfer characteristics
- Water is also the neutron moderator; must be in sufficient quantities to moderate neutrons
- Water must be pressurized to remain liquid at reactor conditions (~14 MPa)
- Hot channel factors must be evaluated (local boiling issue in PWRs)
- Can be used as direct cycle in BWRs (boiling in the core)

Sodium was the Coolant of Choice

- Lead/bismuth has been used by the Russians but not for breeders
- Potassium was used in space reactor experiments at ORNL
- Dense liquids require very high pumping power
- NaK has been used for EBR-I and Dounreay
- Sodium is the coolant of choice for most breeder reactor programs, although it softens the neutron spectrum somewhat



The requirements for fuel performance in SFR are major factors affecting the success of the concept. Because SFRs need to experience higher burnup than LWRs, and the fuel is exposed to higher radiation damage, the cladding and duct material should be resistant to radiation induced swelling as well as high temperatures. The fuel should also be tolerant of tighter packing and minimum space for coolants. HT9 (ferritic-martensitic steel, 12Cr-1.0%Mo-0.3V-0.5W) appears to be the cladding material of choice in recent designs.

Fast Reactor Fuel Types

- Fast reactor fuels can reach much higher burnup than LWR fuel
  - In LWRs, fuel kept in the core until the reactor loses its criticality
  - In FRs, fuel can theoretically be kept in the core indefinitely, imposing different restrictions on the fuel design and performance
- Decision on fuel type is based on many criteria (fabrication, performance, safety, and choice of fuel cycle)
  - Oxide fuel—UO_2, MOX
  - Metal fuel
    - Molybdenum experience (Dounreay and Fermi reactors)
    - Fissium and zirconium alloys
  - Nitride fuel
  - Carbide fuel—UC
  - Other fuel types
    - Uranium sulphide (US)
    - U_3Si
    - Uranium phosphate (UP)

Most nations have focused on oxide fuels or metal fuels, although other forms were evaluated.

Most Common Fast Reactor Fuels

- Often the choice is reduced to irradiation experience
  - Oxide fuel
    - Sintered pellet UO_2 or MOX fuel similar in design to a PWR oxide fuel pellet
    - Helium-filled gap between fuel and cladding
    - Large fission gas plenum (helium filled at manufacture)
  - Metal fuel
    - Uranium-zirconium or uranium-plutonium-zirconium alloy rods
    - Sodium-filled gap between the fuel and cladding
    - Large fission gas plenum (argon filled at manufacture)
- Substantially different thermophysical properties of oxide and metal fuel forms play a significant role in the safety performance
  - Thermal conductivity, stored energy, melting point, failure mechanism, …

Because selection of fuel form is often the first decision in a design project and qualifying a new fuel form could be a lengthy and very expensive ordeal, most important criteria becomes the earlier experience.
Oxide fuel was used in CRBR and FFTF largely because of the greater experience in LWRs
More recent designs (PRISM, 4S, ARC-100, TWR) use metallic alloys, largely based on EBR-II experience and limited irradiation testing at FFTF

U.S. SFR Fuels Experience

- SFRs have been extensively studied and operated by DOE and its predecessor, AEC
  - Experience with EBR-I, EBR-II, FFTF, and CRBR project
- Early U.S. SFR experience focused on metal-alloy fuel
  - EBR-II tests in late 1960's showed limited success for low burnup
- Oxide fuel form was selected for further development in FFTF and CRBR project
  - Based on experience in commercial LWRs and naval reactors
- After CRBR project was canceled, DOE continued on with Advanced Liquid Metal Reactor (ALMR) and Integral Fast Reactor (IFR) programs
  - Emphasis back on a pool-type SFRs with metal alloy fuel to address regulatory concerns related to severe accidents

1932: Chadwick's discovery of neutrons; 1942: CP-1; 1951: First nuclear electricity in EBR-I; 1964: First criticality in EBR-II (all in quick succession)
Severe accident concerns impeded CRBR licensing. Even though U.S. NRC Atomic Safety and Licensing Board (ASLB) eventually excluded HCDAs from the licensing basis, it stated that "probability of core melt and disruptive accidents must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum". That was one of the main motivations of the ALMR and IFR programs.

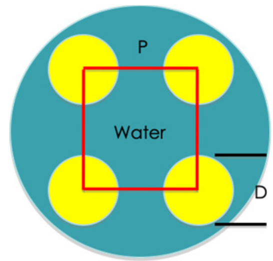
U.S. SFR Fuels Experience (cont.)

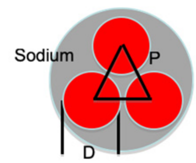
- Subsequent metallic fuel testing in the 1980s (as part of the IFR program) demonstrated that burnup limitation could be overcome by changing the fuel design
  - Lower smeared density with more room to accommodate irradiation-induced swelling
- Under the ALMR program, PRISM (GE) and SAFR (Rockwell/WEC) concepts submitted their Preliminary Safety Information Document to NRC in 1986
  - NRC's Pre-application Safety Evaluation Reports (NUREG-1368 and 1369) identified "incomplete information on the proposed metallic fuel" as one of the key regulatory issues
- IFR program (until its termination in 1994) as well as ongoing work under DOE's Advanced Reactor Technologies (ART) program and Advanced Fuels Campaign continued addressing this issue

Fuel swelling is greater in fast spectrum. Early metallic fuel forms (Mark I/IA in next slide) were limited in terms of burnup due to initial fuel swelling.
This was later overcome (in Mark-II to –IV) with more room inside the cladding that resulted in a fuel with lower smeared density (for up to 20 atom-% burnup).
Metal fuel has 50x higher thermal conductivity than oxide fuel. Also with bond sodium increasing the gap conductance, operating centerline temperature with metallic fuel is less than half of that for oxide fuel.

Typical SFR Fuel Pin Layout

- Sodium moderation is not desired, so fuel is in most dense configuration: triangular pitch, hexagonal fuel assembly
- Fuel assemblies contain tens to hundreds of pins in a duct
  - Duct allows control of flow between fuel assemblies—unlike PWRs, which have an open core (BWRs have ducts for better control of boiling)
- Fuel pins can be spaced by wire-wrap or grid spacers

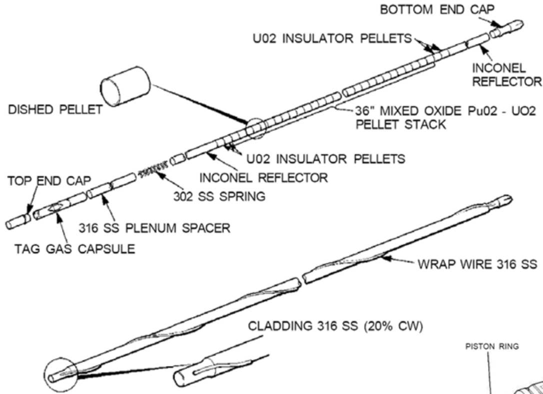


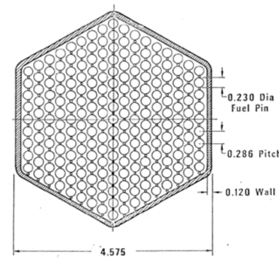


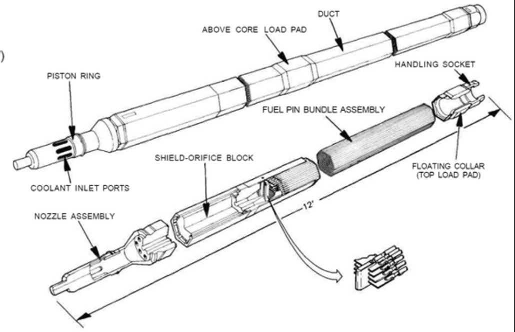
SFR Fuel Forms

- FFTF and CRBR used UO_2-PuO_2 mixed fuel (MOX)
- PRISM, TWR, ARC-100 and 4S use metallic fuel based on EBR-II experience
- Fuel is clad in 316 stainless steel for CRBR or HT9 for newer designs
- Breeders have axial blankets at top and bottom
- Fuel pin has a fission gas plenum above or below the fuel stack(top location has higher fission gas pressure)
- Oxide fuels run with much higher maximum internal temperature than metallic fuels

FFTF Fuel Pin and Assembly Design (Oxide Fuel)







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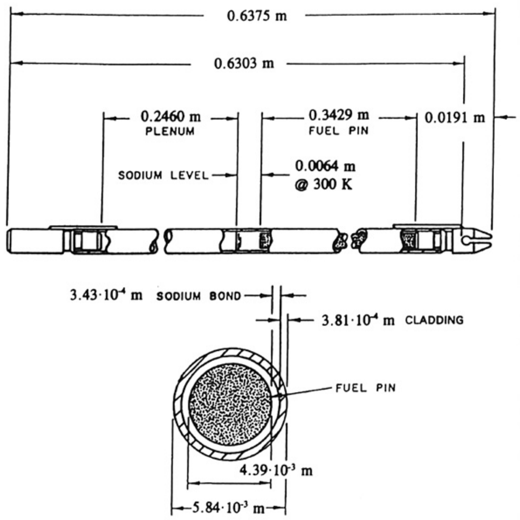
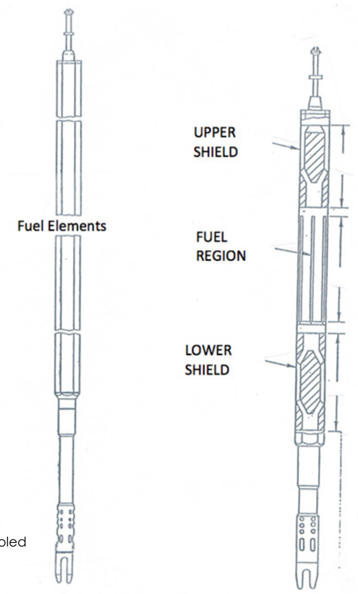
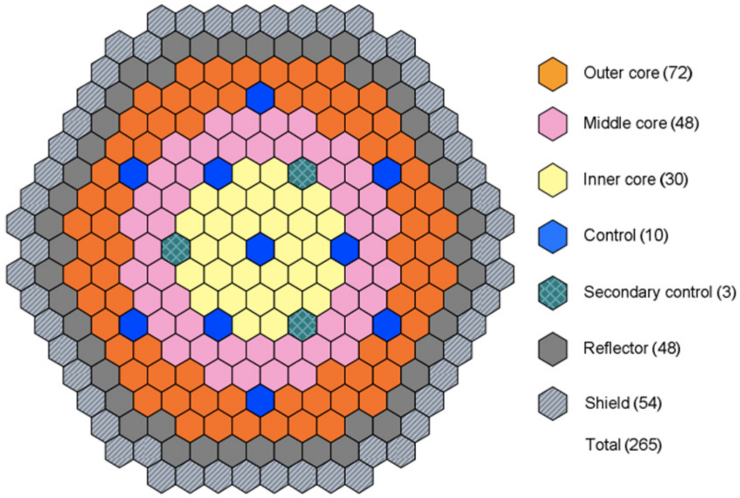
EBR-II Fuel Pin and Assembly Design (Metal Fuel)





T. Sofu, "A review of inherent safety characteristics of metal alloy sodium-cooled fast reactor fuel against postulated accidents," *Nuclear Engineering & Technology*, Volume 47, Issue 3, pages 227-239 (August 2015).

Typical SFR Core Configuration



ANL's AFR-100 Core Design

Fuel Pin Design Considerations

- Phenomena affecting fuel pin performance
  - Creep (cladding)
  - Swelling (fuel)
  - Linear power, temperature
  - Gap conductance
  - Fuel restructuring and constituent migration
  - Fission gas release and transport (open pore formation)
  - Fuel cracking
  - Differential thermal expansion
  - Yield strength
  - Irradiation damage
  - Cladding attack by rare-earth fission products
  - Interdiffusion between fuel alloy and cladding
  - Changes in thermo-physical properties with burnup

Definitions

- Fuel burnup—Defined in terms of energy yield (MWd/kg) or as the fraction of heavy atoms fissioned (atom-% burned)
- Microstructural changes—Actual changes in grain size and orientation can be affected by alloying of certain elements
- Creep—Time-dependent strain occurring under constant stress over long periods of time
- Strength—Material mechanical properties such as hardness, yield, and ultimate strength are less important parameters in fuel selection

Definitions (cont.)

- Swelling—Most fission products lodge within the fuel matrix and contribute to an overall volumetric increase known as *fission product fuel swelling* $\left(\frac{\Delta V}{V}\right)_{\text{solid fp}} = \frac{V - V_0}{V_0}$
  - Net swelling of the fuel is derived from the balance between fission gas retention vs. release, grain structure, porosity distribution, temperature, and temperature gradient
- Fission gas release from the fuel into the fuel cladding gap and plenum
  - Not all the fission product gases remain confined within the fuel, some can diffuse to the grain boundaries and escape to the pin plenum via interconnected porosity and cracks
  - Released fission gas, unless intentionally bled off, pressurizes the overall fuel pin and applies stress to the cladding

Fast Reactor Fuel Design Challenges

- Fast reactor fuels are typically designed to reach much higher burnup to take advantage of higher initial fissile loading as well as the "breed and burn" characteristics
  - Typical LWR fuel burnup is ~ 2–3%
  - SFR fuels typically reach burnup well in excess of 10%
- Greater fuel swelling in fast spectrum
  - Current metallic and oxide fuel pin designs can accommodate it
- Fuel-Cladding Mechanical Interaction
  - Hard, strong ceramic fuel forms can push on cladding imposing limits on maximum burnup
  - Not a major fuel failure mode because fission gas and coolant pressure difference is a bigger factor
- Fuel-Cladding Chemical Interaction
  - Limits coolant outlet temperature of metallic fuel core
- Fuel-Coolant Compatibility
  - Oxide fuel chemically reacts with the sodium coolant imposing stricter limits on fuel pin failures to prevent potential flow blockages

Greater fuel swelling in fast spectrum was the limiting factor for the early metallic fuel forms.
For FCMI, the difference wrt LWR fuel is high burnup (higher FG pressure) and much lower coolant pressure.

Oxide Fuel: UO_2, PuO_2, $(U_{0.8}Pu_{0.2})O_2$

- Fabrication
  - Oxide fuels have the highest operating and manufacturing experience from LWR experience
  - Oxide fuel fabrication is normally accomplished through powder metallurgy; the mixture is then cold-compacted into a pellet
  - The pellets are *sintered* at ~1600°C to achieve the desired level of densification
    - (85–90% theoretical density)

Oxide Fuel: UO_2, PuO_2, $(U_{0.8}Pu_{0.2})O_2$ (cont.)

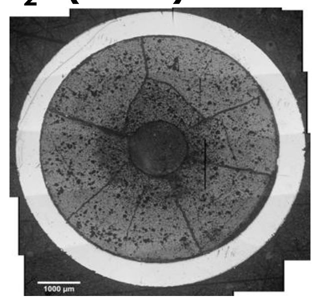
- Physical properties
  - Oxygen ions are arrayed in a simple cubic structure, and the heavy metal ions form a face-centered cubic sublattice
  - Relatively brittle material at temperatures less than half the melting point
- Swelling
  - Some porosity is intentionally incorporated to accommodate fuel swelling
  - 0.15 to 0.45% per atom-% burnup of total swelling is due to solid fission products
  - Substantially greater swelling results from fission gases
- Microstructure
  - Steep radial temperature profiles cause columnar and equiaxed grains to develop after a few hours of irradiation

Oxide Fuel: UO_2, PuO_2, $(U_{0.8}Pu_{0.2})O_2$ (cont.)

- Fission gas release
  - Once released from the fuel matrix, fission gas is vented to collecting zones
    - Usually a fission gas plenum above or below the fuel stack
  - Fuel temperature
    - Below 1300 K, fission gas mobility is low, and there is essentially no gas escape
    - Between 1300 and 1900 K, atomic motion allows some diffusion, and some amount of gas can escape from fuel matrix
    - Above 1900 K, thermal gradients can drive gas bubbles and closed pores over distances comparable to grain sizes

Oxide Fuel: UO_2, PuO_2, $(U_{0.8}Pu_{0.2})O_2$ (cont.)

- Irradiation experience
  - Large irradiation experience in FFTF and international reactors in France, Russia, and Japan
  - Acceptable performance and reliability demonstrated at 10 at.% burnup, with capability established up to 20 at.% burnup

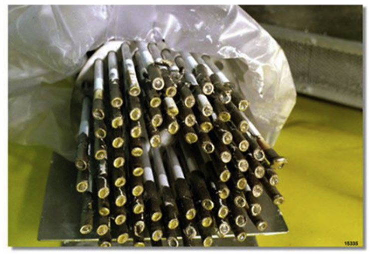


High Burnup Oxide Fuel

  - Robust overpower capability established in TREAT tests: ~ 3 to 4x's nominal power
    - Well above primary and secondary FFTF trips
    - Pre-failure axial molten fuel motion
    - Clad failures near core mid-plane
  - Performance issues typically related to creep rupture of cladding at high burnup, accelerated due to FCMI

Metallic Fuel: U-Fs, U-Zr, U-Pu-Zr Alloys

- Fabrication
  - Developed at Argonne based on experience gained through 20+ years operation of EBR-II
  - Injection cast as cylindrical slugs and placed inside the cladding
  - Liquid-metal sodium is used inside the pin to thermally bond the fuel/cladding and increase gap conductance
    - Along with the high fuel thermal conductivity, maintains significantly lower fuel operating temperatures compared to oxide fuel



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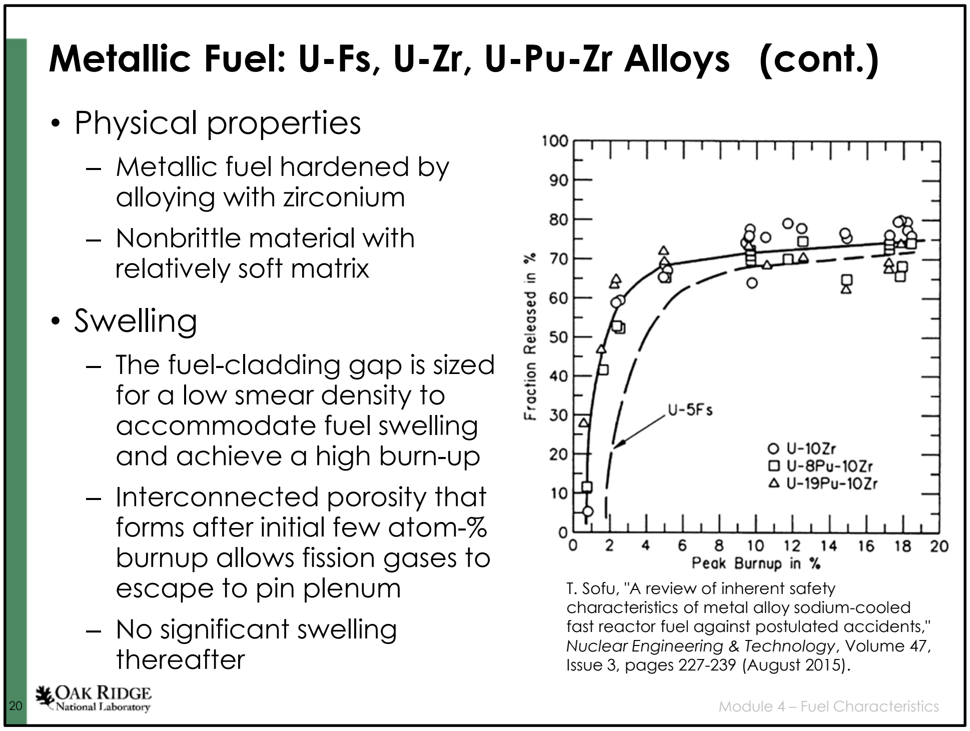
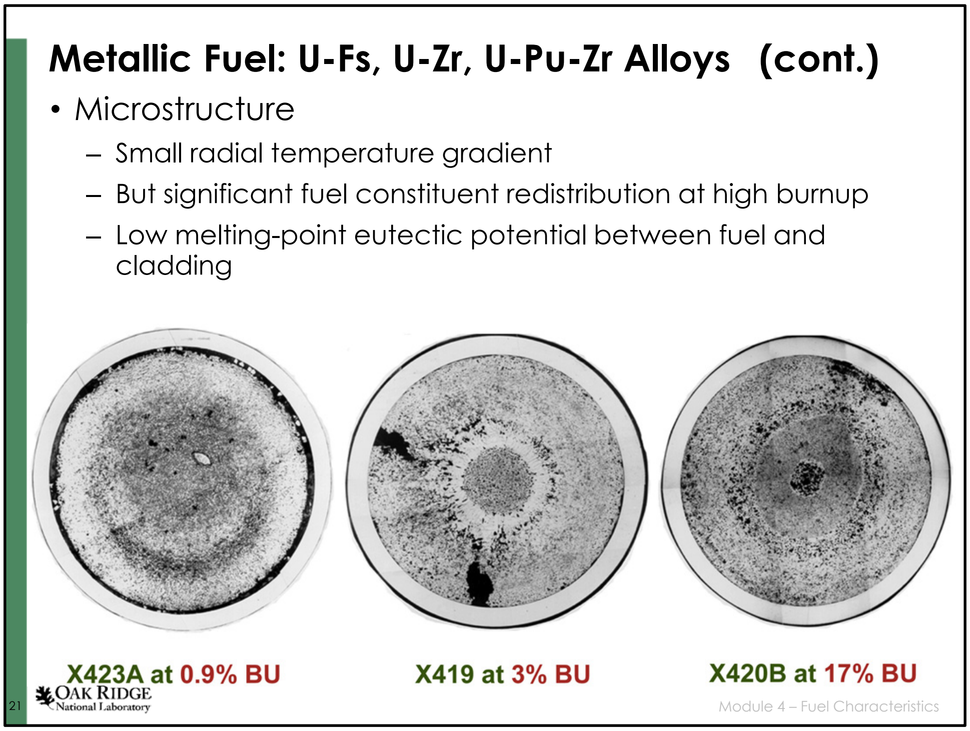


Early experience with trying to restrain swelling of the metal fuel with strong cladding was not successful, and it limited the burnup that can be achieved with it. The key to overcome this limitation was the discovery that, although its soft structure allows metal alloy fuel to swell easily, the total swelling is limited to the swelling at only a few percent burnup as shown in Fig. 3. After the initial few percent burnup, the interconnection of pores in the fuel matrix allows venting of fission gas to the pin plenum avoiding further swelling. Therefore, by allowing sufficient room inside the cladding to accommodate this initial swelling, the FCMI limitation to achieve higher burnup was eliminated for the metal fuel forms.



Post-irradiation examination of metallic U–Pu–Zr fuel pins shows the formation of annular zones with considerably different alloy compositions, fuel porosities, and densities. Uranium migrates from the central and outer zones to the middle zone, whereas Zr and fission products tend to migrate in opposite directions. The resulting zonal densities can vary from 8 g/mL in the central zone to 16 g/mL in the middle zone. The Zr depletion in the middle zone also reduces the melting temperature significantly and impacts the thermophysical properties.

Metallic Fuel: U-Fs, U-Zr, U-Pu-Zr Alloys (cont.)

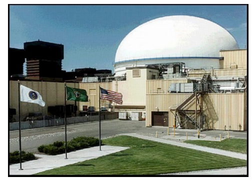
- Irradiation experience
  - Also a large database with metal fuel from EBR-II and FFTF
  - Fuel of choice for U.S. fast reactor R&D program and commercial vendors
  - Acceptable performance and reliability demonstrated at 10 at.% burnup, with capability established up to 20 at.% burnup
  - Robust overpower capability established in TREAT tests: ~ 4 to 5x nominal power
    - Axial fuel expansion prior to melting
    - Pre-failure axial molten fuel motion
    - Failures near top of fuel column
  - Typical performance issue is creep rupture of cladding at high burnup, accelerated due to FCCI
    - Similar performance for U-Fs, U-Zr and U-Pu-Zr fuel forms
    - Burnup, T, and cladding performance are key parameters

Metallic Fuel Experience

EBR-II

- Fuel fabrication and design parameters
- Prototype fuel behavior
- Swelling and restructuring vs. burnup
- Influence of high temperatures
- Fuel failure mode
- Impact of fuel impurities
- Blanket safety
- Run beyond cladding breach tests
  - Six tests with U-Fs/U-Pu-Zr/U-Zr fuel to assess failed fuel performance in the core

FFTF



- Fuel column length effects
- Lead metal fuel tests with HT9 cladding
- Commercial metal fuel prototype
- Metal fuel qualification

Bullets show the topics for performance assessments.

Metallic Fuel Experience (cont.)

- Metal-alloy fuels are manufactured as slugs/rods (full-length in EBR-II) in SS (316) or advanced alloy (D9, HT9) cladding
- Fuel-cladding gap is filled with bond sodium to achieve high gap conductance during early irradiation
- Binary (U-Zr) fuel is the (initial) choice of fuel for all U.S. SFR developers

Reactor	Fuel Type	# of Fuel Pins	Clad	Peak Burnup
EBR-II	Mark-I/IA (U-5Fs)	~90,000	316SS, D9, HT9	~2.5%
	Mark-II (U-5Fs)	~40,000		~8%
	Mark-IIC/IICS/III/IIIA/IV (U-10Zr)	~16,000		~10%
	U-Pu-Zr	>600		~15-20%
FFTF	U-10Zr	>1050	HT9	~14%
	U-Pu-Zr	37		~9%

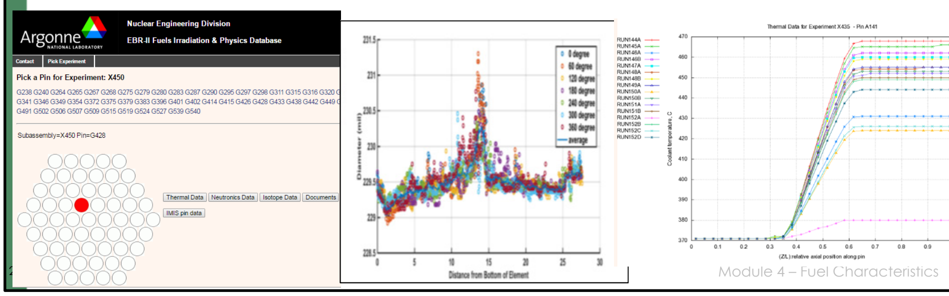
Fs – Simulated Fission Products; Burnup unit is atom-%
First row (~90,000 Mark-I/IA pins) were the first generation metallic fuel pins that could not achieve high burnup (developed and tested in late 60s).
Second generation fuel forms overcame the burnup limitation with lower smeared density.
Metallic fuel is non-reactive with sodium, it allows us to put sodium inside the fuel pin to improve gap conductance by several orders of magnitude.
Along with the high fuel thermal conductivity, this maintains significantly lower fuel operating temperatures compared to oxide fuel.

Metallic Fuel Design Parameters

Key Parameter	EBR-II/FFTF
Peak Burnup, 10^4MWd/t	5.0 – 20
Max. linear power, kW/m	33 – 50
Cladding hotspot temp., ^oC	650
Peak center line temp., ^oC	<700
Peak radial fuel temp. difference, ^oC	100 - 250
Cladding fast fluence, n/cm^2	up to 4 x 10^{23}
Cladding outer diameter, mm	4.4 - 6.9
Cladding thickness, mm	0.38 – 0.56
Fuel slug diameter, mm	3.33 – 4.98
Fuel length, m	0.3 (0.9 in FFTF)
Plenum/fuel volume ratio	0.84 to 1.45
Fuel residence time, years	1 - 3
Smeared density, %	75

Metal Fuel Irradiation and Physics Analysis Databases

- EBR-II Metallic Fuel Irradiation Testing Databases (ANL)
  - PIE reports, digitized micrographs, profilometry measurements, gamma scans, porosity and cladding strain measurements, and scans for other microstructural characteristics to support fuel qualification and code validation
  - Also pin-by-pin fuel fabrication and core load information for each EBR-II operating cycle (operating parameters, temperature, fluence, and burnup predictions as input to fuels performance codes)



Module 4 – Fuel Characteristics

PIE: Post-Irradiation Examination performed in hot-cells

Metal Fuel Irradiation and Physics Analysis Databases (cont.)

- FFTF Metallic Fuel Irradiation Testing Database (PNNL)
  - Data from aggressive irradiation testing of 8 metallic fuel assemblies containing long fuel pins (prototypic of commercial SFR fuels)
    - No cladding breach up to burnups approaching 150 MWd/kgM
  - Test design descriptions (fabrication data and QA documentation) for IFR-1 and MFF series of metal fuel tests
  - Available operational data for irradiation cycles
    - Power, flow rates, temperatures
  - Test reports
    - Fabrication records, irradiation reports, PIE reports
  - Results for impact of metal fuel tests on reactor operating parameters such as reactivity feedbacks and direct measurement data (in-core assembly growth, assembly pull forces, IEM cell exams)

Design Impact of SFR Fuels on Safety

- Difference in thermal conductivity and gap conductance offers significant advantage for the metallic fuel
  - Much lower steady-state and transient temperatures
  - Flatter radial temperature profile
- Despite big difference in melting point, both oxide and metal fuels have similar margin to melting during transients
- Phenomena depending on diffusional rate processes, such as creep and fission gas release, are also similar
- Since metal fuel cladding generally fails below the coolant boiling point, damaged metal fuel pins remain coolable
- Metal fuel is also compatible with sodium coolant
  - Can operate with minor clad failures
  - Oxide fuel chemically reacts with sodium
- These, and the low retained heat, are significant factors to more benign response of metallic fuel during accidents

Comparison with Oxide and Metallic Fuel Forms

	Oxide (UO_2-20PuO_2)	Metal (U-20Pu-10Zr)
Heavy Metal Density, g/cm^3	9.3	14.1
Melting Temperature, K	3000	1400
Thermal Conductivity, W/cm-K	0.023	0.16
Operating Centerline Temp. at 40 kW/m, K	2360	1060
T/T_{melt}	0.79	0.76
Fuel-Cladding Solidus, K	1675	1000 (eutectic)
Thermal Expansion, 1/K	12×10^{-6}	17×10^{-6}
Heat Capacity, J/g-K	0.34	0.17

Impact of Neutron Spectrum

	Oxide fuel	Metal Fuel
Spectrum	Softer due to oxygen moderation in the fuel	Harder due to reduced moderation
Effective heavy metal density	Lower due to oxygen in fuel	Higher
Neutron yield	Lower due to softer spectrum and lower heavy-metal density	Higher due to harder spectrum and higher heavy-metal density
Conversion rate	Lower due to all above	Higher due to all above
Burnup reactivity swing	Higher due to lower conversion rate	Lower due to higher conversion rate
Excess external reactivity needed	Larger due to higher burnup reactivity swing	Smaller due to lower burnup reactivity swing
Mean free path	Shorter but still with sufficient sensitivity to core radial expansion	Longer with greater sensitivity to fuel axial and core radial expansion

Smaller excess external reactivity needed to control the reactor limits the reactivity available for accidental insertion.

Impact of Operating Temperature

	Oxide fuel	Metal Fuel
Operating temperature	Higher due to much lower thermal conductivity and gap conductance	Much lower due to high thermal conductivity and gap conductance
Radial temperature gradient	Higher due to much lower thermal conductivity	Much lower due to high thermal conductivity
Heat capacity	Higher	Lower
Stored heat	Higher due to higher fuel temperatures and heat capacity	Lower due to lower fuel temperatures and heat capacity
Grace period for operator action to correct cooling deficiencies during accidents	Shorter due to larger stored heat	Longer due to smaller stored heat

Impact of Doppler Feedback

	Oxide fuel	Metal Fuel
Doppler feedback	Larger due to softer spectrum and higher operating temperature	Smaller due to harder spectrum and lower operating temperature
Zero- to full-power Doppler reactivity swing	Larger also due to much higher radial temperature gradient across the fuel	Smaller also due to low radial temperature gradient across the fuel
Reactivity control requirement	Larger external reactivity needed due to above	Smaller external reactivity needed due to above
External reactivity available for accidental insertion	Larger due to above	Smaller due to above

Doppler feedback in metallic fuel is about 1/3 of what it is in oxide fuel.
Since strong Doppler feedback resists the system to return to equilibrium temperatures, metallic fuel provides better inherent safety performance as the system approaches an asymptotic state.

Fuel Response During Unprotected Accidents

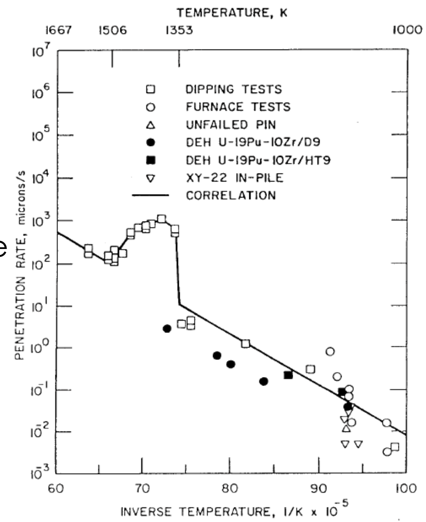
- Some multiple-fault accident initiators can lead to fuel failures (typical cases involve unprotected accidents)
  - When PPS fails to scram the reactor, key early measure is to maintain the coolant temperature below its boiling point
  - Net negative reactivity feedback eventually brings the reactor power into equilibrium with the available heat rejection rate as the system approaches an asymptotic temperature distribution
  - In the long term, goal is to keep the asymptotic cladding, vessel, support structure temperatures below creep limits
- Avoiding core damage therefore depends on:
  - Providing sufficient negative reactivity feedback to overcome the initial power-to-cooling mismatch, and
  - Reducing the reactivity feedback components (mainly Doppler) that resist the return of the system to equilibrium

Metal-Fuel Failure Modes

- FCMI: Since low fuel smeared density allows development of inter-connected porosity in fuel matrix and release of fission gas to pin plenum early in operation, FCMI is not a common failure mode
- FCCI: Major mode of pin failure in metallic fuel due to formation of low melting-point intermetallic eutectic between the uranium and iron at the fuel-cladding interface
  - When zirconium is used as a component in the metal fuel alloy, this eutectic penetration is delayed and reduced
  - If the transient temperatures are sufficiently high for an extended period of time, however, the potential exists for a thinning of the cladding and subsequent breach

Metallic Fuel-Cladding Eutectic Formation

- Temperature limit depends on fuel/cladding compositions and the irradiation history, but measurable cladding thinning starts around 1000 K
- Penetration rate is slow up to the point when fuel melting begins
- As the molten fuel eventually comes into contact with the cladding, the eutectic penetration rate becomes very fast



C. M. Walter and L. R. Kelman,
- "The Interaction of Iron With Molten Uranium," J. Nucl. Mat. 20 (1966).
- "Penetration Rate Studies of Stainless Steel by Molten Uranium and Uranium-Fissium Alloy," J. Nucl. Mat. 6, (1962).

Metal-Fuel Failure Consequences

- Due to the high conductivity of the metal fuel, peak fuel temperature in steady operation and most transients is well above the axial mid-plane
- The peak cladding temperature is also near the top of the fuel column where the cladding is the weakest
- Therefore, metal fuel pin failure locations are predictably near the top of active core height where any in-pin and/or ex-pin molten fuel relocation reduces core reactivity
  - Propagation of molten fuel cavity through the top of the fuel column may lead to expelling of the molten fuel into the upper pin fission gas plenum prior to cladding breach

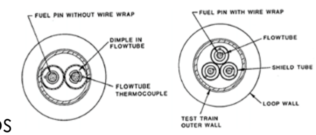
T. H. Bauer, A. E. Wright, W. R. Robinson, J. W. Holland and E. A. Rhodes, "Behavior of Modern Metallic Fuel in TREAT Overpower Tests," Nuclear Technology, Volume 92 (1990)

Metal-Fuel Failure Consequences (cont.)

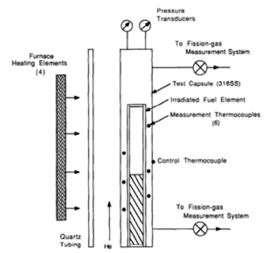
- When cladding fails, metal-alloy fuel's compatibility with sodium coolant offers a significant advantage
  - Significantly different from the chemical reaction that occurs between oxide fuels and sodium coolant
- Metal-fuel and cladding eutectic mix disperses in the sodium coolant and gets entrained out of the core instead of freezing and creating coolant channel blockages that can propagate the damage
- Cladding damage typically occurs at temperatures generally below the boiling point of the sodium coolant
  - Damaged configurations are usually coolable and limited in scope

Metallic Fuel Transient Testing Experience

- EBR-II passive and inherent safety tests
  - ~80 integral experiments from comprehensive shutdown heat removal, BOP, and inherent plant control testing program
    - Including several unprotected (without scram) LOF and LOHS tests
  - No challenge to fuel integrity during entire testing program
- TREAT M-series tests
  - Rapid transient overpower tests to examine margin to cladding failure, fuel melting and relocation
  - Whole irradiated EBR-II pins in flowing Na loops
  - U-5Fs/SS, U-10Zr/HT9, U-19Pu-10Zr/D9 fuel types



- Out-of-pile tests in radiant furnaces
  - Fuel Behavior Test Apparatus (FBTA)
    - Irradiated U-10Zr, U-Pu-Zr pin segments
    - Examined liquid phase formation and FCCI rate
  - Whole Pin Furnace (WPF) Tests
    - Irradiated whole U-Zr, U-Pu-Zr pins
    - Examined margin to cladding failure



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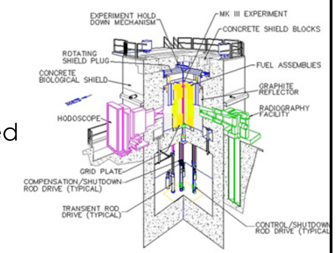
Module 4 – Fuel Characteristics

LOF: Loss of Flow
LOHS: Loss of Heat Sink

TREAT Experiments Relational Database (ANL)

- Searchable collection of transient tests conducted in TREAT (1959-1994)
  - ~900 tests & categories w/ parametric information (e.g. fuel, transient info, results)
  - ~6000 searchable PDFs with links to referenced tests



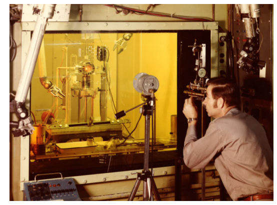
- Metallic Fuel Transient Overpower Tests
  - Test specifications, test plans, digital data…

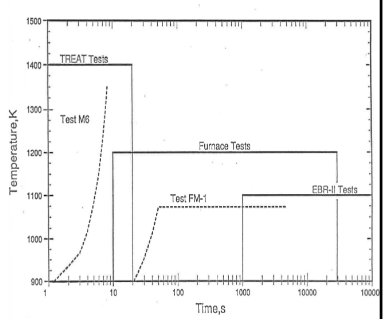
	objectives or outcomes							fuel		B/U			clad	
	Limited fuel damage, no clad breach	Pre-failure in-pin fuel motion	Cladding failure threshold	Mild cladding failure	Fission product release	Fuel-coolant interaction	Post-failure fuel and cladding disruption & dispersal	U-Zr	U-Pu-Zr	Zero (fresh fuel)	Low (up to 5 at.%)	Medium (5 to 10 at.%)	D9	HT9
M5	✓	✓							✓		✓		✓	
M6	✓	✓	✓	✓		✓	✓		✓		✓	✓	✓	
M7	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓

	transient type		transient meas.		post-test analyses						posttest examinations & measurements			
	Overpower	Heat balance	Fast Neutron Hodoscope	Thermal-hüdraulic (Press, Temp, Flow)	Thermal-hydraulics	Cladding failure	Fuel motion	Severe-accident behavior, interaction of multiple Phenomena	Microstructural and mm-scale changes	Fuel-cladding interface or gap behavior	Examinations of irradiated sibling samples	Non-destructive examination of test remains	Destructive examination of test remains	Micro-examination of fuel & cladding remains
M5	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
M6	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓
M7	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓	✓

Database for Out of Pile Experiments (ANL)

- Transient furnace tests in hot cells
  - Chopped irradiated pin segments in Fuel Pin Test Apparatus (FBTA)
  - Full length irradiated pins in Whole Pin Furnace (WPF)
  - Simulated reactor accidents, varying ramp rates and peak temperatures
  - Showed significant safety margin for selected transient conditions
- U-(0-26)Pu-10Zr pins in D9, HT9, 316SS clad
  - Burnup: 2-3 a/o in WPF, 6-12 a/o in FBTA
  - Fuel compatibility tests on clad fuel segments
  - Fission gas retention examinations
  - Cladding penetration depth measurements
- Results being archived in an online database:
  - Metallurgical examination of tested materials
  - Fission product release measurements





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U-(0-26)Pu-10Zr essentially implies both binary (U-Zr) as well as ternary (U-Pu-Zr) fuels with varrying Pu ratio up to 26 %.

Backup slides

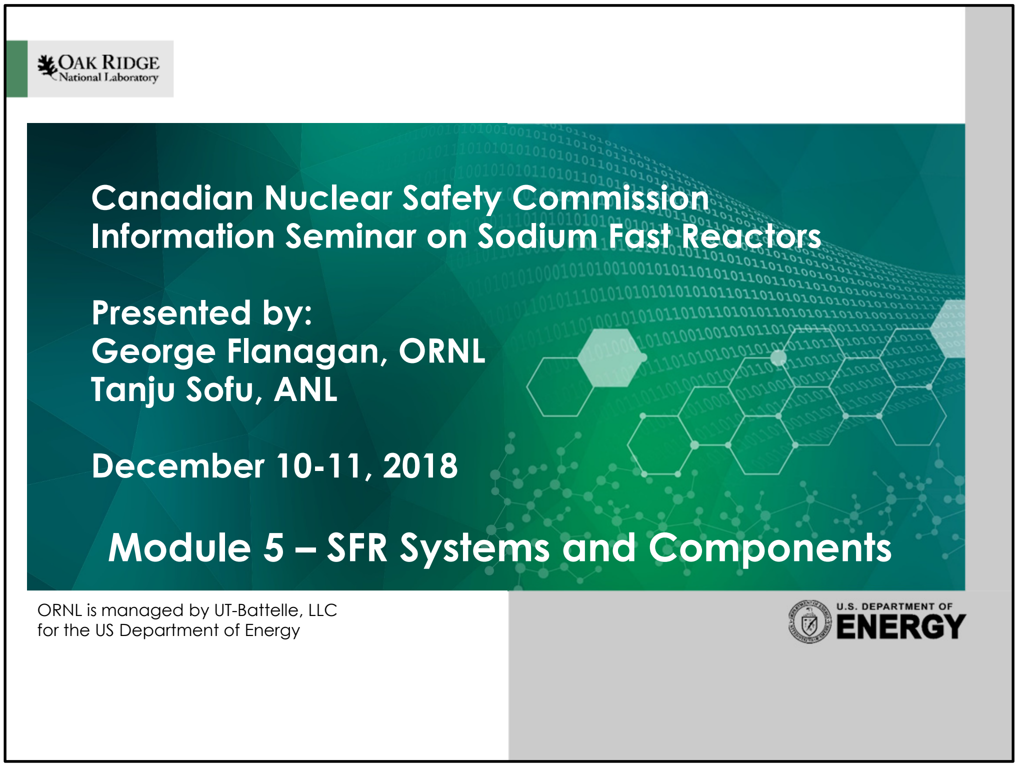
PWR vs SFR Fuel for PRISM Example

Category	PWR	PRISM (metal fuel)	S-PRISM (oxide fuel)
Fuel type	UO_2	U-Pu-10% Zr metal	MOX
Pin OD	~0.382	0.29 OD	0.335 OD pin
Bonding	Helium-bonded	Sodium-bonded	Helium-bonded
Cladding	Zirconium cladding	HT9 cladding	HT9 cladding
Spacers	Grid	Wire wrap	Wire wrap
Pitch	Square	Triangular	Triangular
Lattice	Open lattice	Hexagonal duct	Hexagonal duct
Assembly information	Square, 17 × 17, 8.5-in. pitch	Hexagonal, 271, 6.282-in. pitch	Hexagonal, 217, 6.355-in. pitch
Enrichment	Maximum 5%	Plutonium, typically less than 30%	Plutonium, typically less than 30%

Needs references for PRISM and source of PWR info

PWR vs SFR Fuel for PRISM Example (continued)

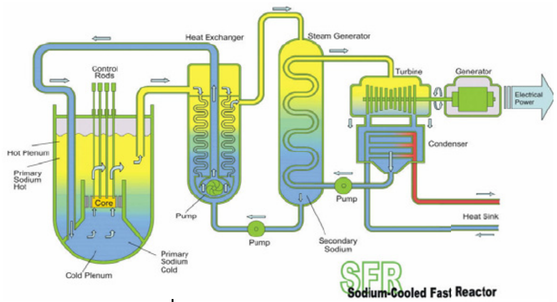
Category	PWR	PRISM (metal fuel)	S-PRISM (oxide fuel)
Active length	12-ft active length	47-in. active length	45-in. active length
Gas plenum	Relatively small	70 in.	67.25 in.
Burnup	50,000 MWd/T	~100,000 MWd/T	~100,000 MWd/T
Blanket			
Pin OD	NA	0.3983 in.	0.4326 in.
Blanket composition	NA	Zirconium-natural or depleted uranium	UO_2 (natural)
Blanket assembly	NA	Hexagonal assembly, 127, 6.282-in. pitch, 40-in. long	Hexagonal assembly, 127, 6.355-in. pitch
Total fuel pin length	144–168 in.	157 in.	



This module presents the major characteristics of SFR systems and components, particularly as they differ from LWRs.

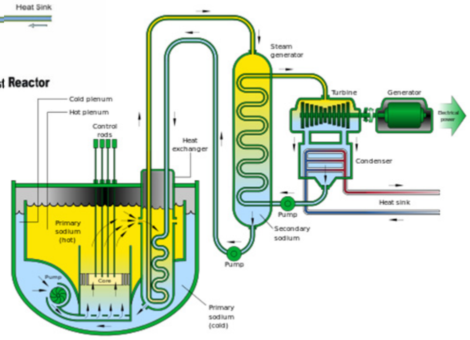
Plant Configurations

- Two major types of configurations are used in Sodium Fast Reactor Systems (SFRs): Loop and pool types



Loop

Pool



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This module presents the two types of arrangements used in SFRs. Both the pool type and the loop types have been used in large, commercial scale reactors, and each has its advantages.

Loop Configuration

- Pumps, primary sodium to secondary sodium intermediate heat exchanger (IHX), piping, etc., are separated from the reactor vessel
  - The primary coolant leaves the reactor vessel
  - Intermediate heat exchanger (IHX) is located in the containment area
  - Has reliability improvements—easier to isolate the loop and perform IHX maintenance
  - Primary vessel surrounded by a guard vessel
  - Usually requires double-walled piping for primary sodium in areas outside the vessel
  - Preferred in Japan
  - FFTF was a loop-type plant

Advantages of Loop Layout

- Major components are separate for easier maintenance and replacement
- Components are in separate cells—often individually housed in steel-lined concrete
- Simpler vessel head design (compared to pool type)
- Allows more flexible relative elevation of components to enhance natural circulation
- Less neutron shielding needed to reduce secondary sodium activation
- Quicker response to changes in steam and secondary system (because of smaller sodium inventory in primary system)

Disadvantages of Loop Layout

- More susceptible to pipe breaks, especially in primary coolant loop with activated sodium
- Requires more space and larger containment
- Requires more cells
- Less sodium in the primary system to act as a heat sink with shorter grace period
- Can require guard pipes around piping to contain leaks
- Requires long piping system

Pool Configuration

- Core, primary piping, IHX, and primary pumps are in a pool of sodium
  - Primary coolant is kept within the reactor vessel which also encompasses the IHX
  - Larger reactor vessel, but reduces the impact of a primary pipe break or leak
  - Preferred in the United States, France, Russia, S. Korea, China, and India
  - Primary vessel surrounded by guard vessel
  - EBR-II was a pool-type plant
  - Choice for current U.S. fast reactor R&D program and commercial vendors (also 4S)

Advantages of Pool Configuration

- Leakage in primary system components and piping does not result in leakage of activated primary coolant
- Mass of primary sodium is ~3× that of a loop system, providing greater heat capacity
- Large thermal inertia of pool dampens transients and provides longer grace period during accidents
- Simpler cover gas system with only one free surface
- Reduced need for guard piping to contain leaks
- All primary piping is located inside the vessel

Disadvantages of Pool Configuration

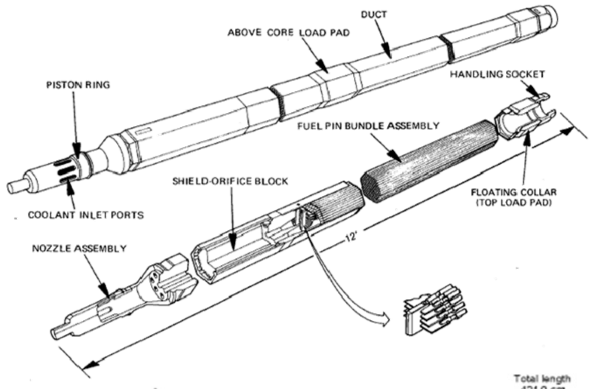
- Large, complex vessel head must support more systems and fuel handling equipment
- Restricted access to components—harder to perform maintenance on components
- Requires a larger pool vessel
- Larger primary system is needed to ensure natural convection
- Requires more neutron shielding to minimize activation of secondary sodium

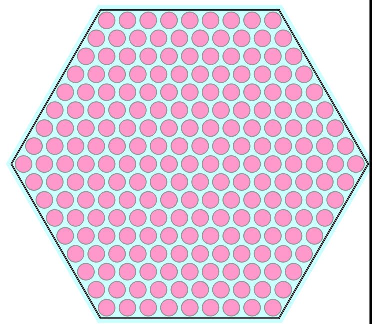
Major Systems and Components

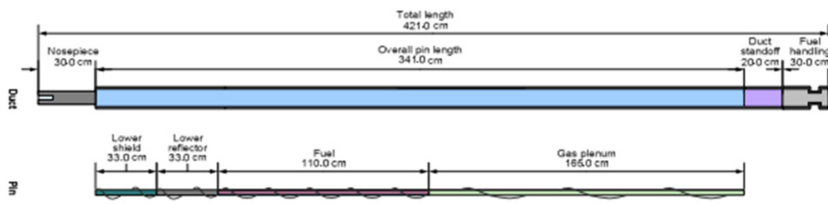
- Reactor core
- Reactivity control and shutdown systems
- Reactor and guard vessels
- Heat transport systems (primary and intermediate)
- Energy conversion system (balance of plant)
- Decay heat removal systems
- Containment
- I&C, coolant and cover-gas cleanup systems, spent fuel storage

Reactor Core

- Fuel pin and fuel assembly (ANL's AFR-100 design)







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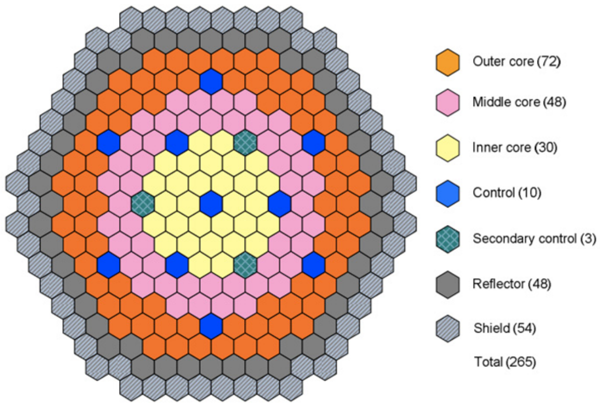
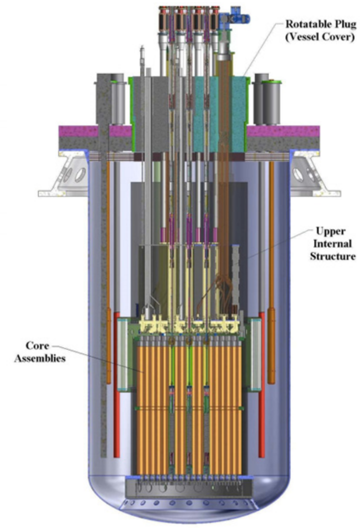
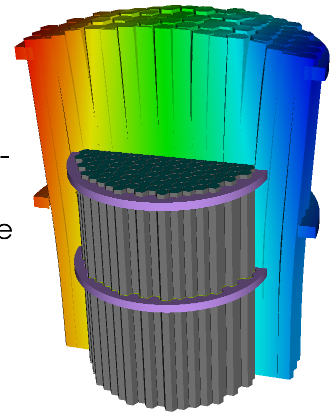
Reactor Core (cont.)

- Typical SFR core configuration (ANL's AFR-100 design)





Core Restraint System



- Controls horizontal movements of core assemblies from thermal expansion, irradiation-induced swelling, irradiation-enhanced creep
  - Reactivity effects should be acceptable
  - Control-rod driveline alignments should be maintained within specified tolerances
- Accommodates horizontal seismic motions within alignment and stress specifications
- Maintains sufficient clearances to facilitate refueling
- Design parameters include
  - Length and stiffness of lower adaptors
  - Number, location, and configuration of assembly load pads
  - Rigidity of peripheral boundary

Most international reactors adopt “free-flowering core” concept

U.S. designs favor “limited free bow” approach

Used and tested in FFTF.

Reactivity Control and Shutdown Systems

- Two independent active systems to control the reactivity and achieve shutdown
  - Reactivity control system: Capable to bring the reactor from any operating condition to hot standby condition with most reactive control assembly inoperative
    - Also serves to compensate for burnup reactivity swing and accommodates uncertainties in criticality and fissile loading
  - Shutdown system: Capable to bring the reactor from any operating condition to subcritical state at refueling temperature (~200°C) with most reactive control assembly inoperative

Reactivity Control and Shutdown Systems (cont.)

- Other typical supplementary reactivity control systems
  - Rod stop system: Prevents substantial power increase during unintended rod withdrawal event
  - Passive reactivity control devices that require no electric power or actuation signal
    - Curie point magnetic alloy that facilitates automatic detachment of control rods when the coolant temperature rises
    - Hydraulically suspended rods
    - Gas expansion modules
  - Ultimate shutdown system: Manually shuts down reactor in the event that all methods of active or passive scram options have failed

Reactor Vessel

- Reactor vessel envelopes the core and most of primary heat transport system (PHTS) components
  - In pool type systems, entire PHTS is placed inside the reactor vessel (reactor primary coolant boundary)
  - Provides support for reactor core, inner barrel, thermal barriers, shielding…
  - Also acts as a barrier against the release of radioactivity
- Typically made of austenitic stainless steel and shaped as a cylindrical shell with a dome or torospherical bottom
  - Either hung from the top by a support ring, or supported at the bottom
- The fuel assemblies rest on a core support structure
  - Core support grid guides the flow from the inlet plenum
  - Upper internal structures guide the flow into the upper plenum
- An inert cover gas separates the sodium from the reactor head that provides access for control rods and rotating plugs as refueling ports
  - No penetrations of the reactor vessel in a pool type system

2 inches of thickness vs 8-12 inches in an LWR

Reactor Vessel

SFR (PRISM Pool Type)	PWR
316 stainless steel	Stainless steel clad manganese moly steel
~5.75 m diam, ~17 m high	4 m ID, 11 m high
5 cm thick	20-30 cm thick
Guard vessel: ~6 m OD, 2.5 cm thick 2.5 Cr–Mo steel alloy	No guard vessel

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Ref: PRISM Preliminary Safety Information, ML082880369 GEFR-00793 – Vol 1, December, 1987; and ML082880397 GEFR-00793 – Vol 4, December, 1987. Ref: PWR: -Pressurized Water Systems, USNRC Technical Training Center, June, 2003

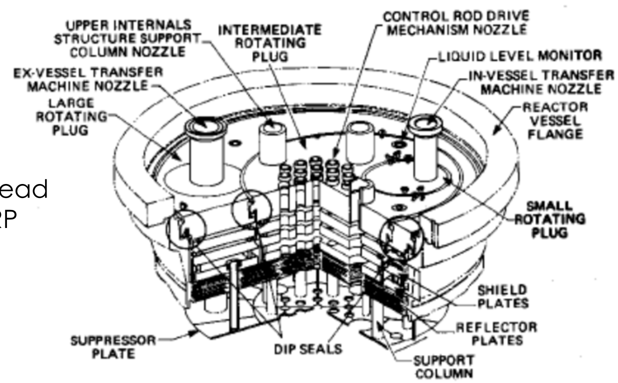
Reactor Vessel Considerations

SFR	LWR
Compatibility with sodium (low-carbon stainless steel)	Compatibility with water using SS cladding on vessel material (boric acid could cause corrosion problems)
No thermal shock concern	Thermal shock is an issue under certain ECCS conditions
Head contains rotating plug (usually); control rod drives and refueling system done with head in place	Head contains control rod drives—refueling done with head removed (bottom head for CRD in BWRs)
Concern with fast neutron fluence	Concern with radiation embrittlement under high pressure
Needs argon cover gas	No cover gas needed (steam in pressurizer)
Needs guard vessel to contain sodium leaks and maintain sodium inventory	High-pressure injection system to control coolant inventory

Reactor Vessel Head

- Seals the primary sodium and must contain an inert cover gas (typically argon)
  - Must be thermally insulated from the hot surface sodium
- Pool-type head is more complex because it must support the core, pumps, IHX, and allow fuel handling access

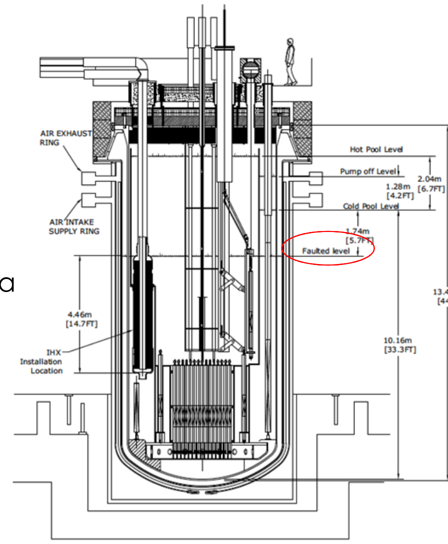
Reactor Vessel Head for CRBRP



Reactor vessel head and components need to be thermally shielded from the surface of the hot sodium. Also, sodium aerosols have been an issue in tending to jam control rod drives and refueling system components.

Guard Vessel

- In case of failure of the reactor vessel (from seismic events or thermal creep induced rupture), the guard vessel wraps the reactor vessel
  - Gap between the reactor and guard vessels does not contain Na under normal conditions
  - It is sufficiently wide to allow inspection but narrow enough to maintain high enough sodium level
    - to keep the core covered and decay heat removal systems functional



- Both cold and hot legs (i.e., sodium inlet and outlet pipes) enter above the guard vessel so that any pipe rupture does not result in coolant loss

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In the diagram, pipes penetrating the vessel head are of IHTS.
In a pool type system, no PHTS pipes to penetrate the rx or guard vessels

Heat Transport Systems

- SFRs generally have three heat transfer systems:
  - **Primary heat transfer system** (PHTS)—cools the core
  - **Intermediate heat transfer system** (IHTS)—transfers heat from the primary system to steam generator (usually with sodium)
    - To avoid the possibility of activated primary sodium burning with steam and pressurization of PHTS as a result of a steam generator tube rupture
  - **Energy conversion system** or balance of plant (BOP)—to generate electricity with a turbine
- Both PHTS and IHTS are kept at low pressure (near ambient) since the boiling point of Na is significantly higher than normal operational temperatures
  - Peak pressure is set by core/IHX pressure drop and gravity head characteristics (up to about 1.0 MPa at reactor inlet)
- Turbine/generator, condenser, feedwater systems are similar to a PWR but they run at a higher temperature
  - Higher energy conversion efficiency

Heat Transport Systems (cont.)

PHTS characteristics:

- Each reactor fuel assembly typically produces about 5 MW of power
- Average core power density is typically 350 to 500 kW/liter (1100 to 1500 kW/liter in the fuel)
  - Average fuel pin linear power ratings are typically 23 to 28 kw/m for pins with cladding diameters of usually < 1 cm
- Typical coolant velocities in the fuel pin bundle are 5 to 7 m/s
- Primary coolant outlet temperatures are ~500-550°C, depending on cladding material (boiling margin ~350°C)

Heat Transport Systems (cont.)

BOP characteristics:

- Based on choice of energy conversion system
  - Conventional steam cycle or supercritical CO_2 Brayton cycle
- SFRs allow more efficient steam conditions than a conventional steam cycle
  - Water reactors are limited to ~325°C outlet temperatures and 15.5 MPa—limited to saturated steam cycles—with efficiencies ~35%
  - SFR can attain temperatures high enough for superheated steam and "modern" steam conditions with sodium outlet temperature ~550°C
    - Allows ~453°C 10.5 MPa steam and higher thermodynamic efficiency ~40+%
    - Conventional designs uses saturated steam cycle

Ref: PWR: PWR: -Pressurized Water Systems, USNRC Technical Training Center, June 2003. SFR: Alan E. Waltar and Albert B. Reynolds, *Fast Breeder Reactors*, Pergamon Press, 1981, Table 12-1.

Conventional water reactors required a derating of the steam system because of the relatively low temperature of the steam as compared to conventional steam plants. Sodium cooled reactors, due to their high temperatures, allow superheated steam with higher temperatures and consequently higher efficiencies. Efficiency, however, for nuclear plants is a less important issue than for conventional plants because of the low cost of fuel as compared to coal or gas fired plants.

Sodium Pumps

- Mechanical pumps in the primary and intermediate loops are generally vertical-shaft, single-stage, double-suction impeller, free-surface centrifugal pumps
  - Always in the cold leg in pool type systems
    - Easier on seals, bearings, etc., because of cooler temperatures
  - Can be in hot or cold leg in loop type systems
    - Hot-leg pump location is usually preferred because of easier control of free surface in pump
  - Mechanical pumps are normally in the cold leg of the intermediate loop
- Electromagnetic pumps (induction or J+B type) can also be used in SFRs since sodium has a very high-electrical conductivity
  - Used on intermediate loop in EBR-II and SEFOR, the primary loop of the Dounreay, in backup decay heat removal systems of SNR-300 and SuperPhenix
  - Supplementary flow coastdown feature is often needed to assure adequate flow inertia during loss of flow accidents
    - Inertia driven electrical supply generator to avoid abrupt stop

One of the factors affecting EM pumps is that flow coastdown is needed to assure adequate transient flow after loss of power accidents. EM pumps must have an inertia driven electrical supply generator to provide this coastdown; otherwise, the flow would stop abruptly.

Intermediate Heat Exchanger (IHX)

- All SFRs have intermediate heat exchangers and intermediate loop to transfer heat from the primary coolant to the steam generator
- IHX isolates primary system from leaks in steam generators, and steam generators from radioactive primary sodium
- Loop designs can locate IHX outside the reactor vessel to enhance natural circulation
- Steam generator pressures are much higher than IHX pressure, which is slightly higher than primary system pressures
  - Leaks propagate from intermediate loop to primary system

Thermal expansions can be significant due to differences in temperature of various components in loop systems.

Intermediate Heat Exchanger (cont.)

- Generally shell-and-tube heat exchangers in counter flow configuration are used
- Design considerations include
  - Straight vs. bent tubes
  - Shell vs. tube-side primary flow
  - Counter-current vs. parallel vs. cross flow
- Usually made of 316 or 304 stainless steel
- Primary sodium on shell side (slightly higher pressure in tubes)
  - Since the water-steam pressure is higher than the sodium pressure, steam or water will flow into the sodium in intermediate loop, preventing contamination of the turbine with sodium oxide
  - Intermediate loop is generally equipped with a pressure relief system to prevent overpressurization in case of a tube rupture event
- Smaller differential thermal expansion between tubes and shell than in steam generators because of smaller temperature differences

Steam Generator

- Steam generators transfer the heat from the intermediate sodium to the water/steam in the power conversion system
  - Steam drives the turbine generators that produce electricity
- Can be separate boilers and superheaters or once-through boiler/superheater systems
- Isolate the high-pressure steam at ~7 MPa from the low-pressure sodium systems
- Often made with 2-1/4 Cr–1 Mo % ferritic steel
  - Super-Phenix used Incoloy 800, PFR used austenitic SS
- Require accommodations of thermal expansion to a greater extent than IHXs
- Steam generators with single tube wall separating steam from intermediate sodium are susceptible to sodium water reactions and are difficult to identify/isolate the leaking tubes
- Double wall steam generator tubes have two walls separated by a mesh that allows leaked material to transport to a sensing device
  - Water leaks (from the outside) and sodium leaks (from the inside) can be easily detected
- Double wall steam generators are costlier and less efficient because of the greater heat transfer resistance in double walls

Steam generator thermal expansion has been accommodated by floating lower tubesheets, bellows, helical tubes, hockey stick tubes, and special designs to accommodate thermal expansion (BN 350)

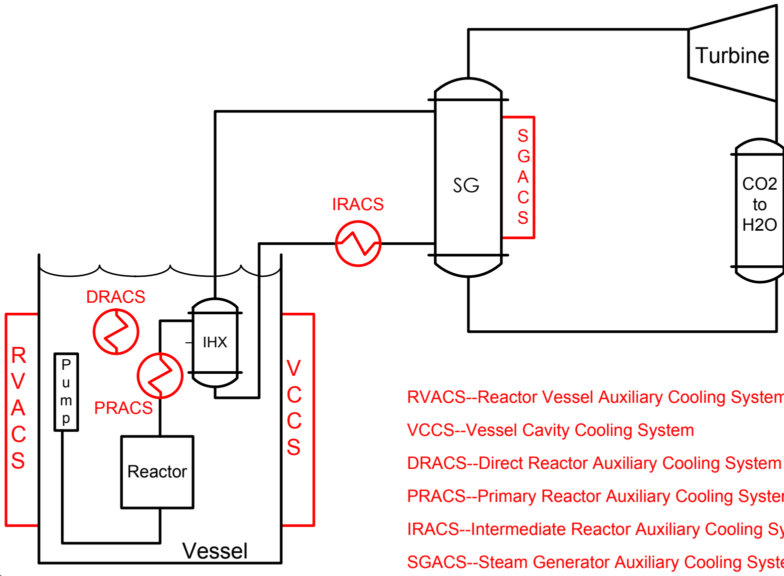
Steam generators have been a cause of trouble for most sodium cooled plants. Some plants, such as the Toshiba 4S, where cost is less of an object because of the anticipated remote locations, use double wall steam generator tubes. The Russian BN – 600 has addressed the issue of steam generator leaks by constructing eight separate steam generator loops. for each of three secondary sodium circuits. These can be isolated from the rest of the system and repaired while the rest of the plant continues to operate.

Decay Heat Removal Systems

- SFRs rely on independent and diverse means for removal of decay heat
- Normal shutdown heat removal is usually via balance-of plant (BOP)
  - Based on diverting steam (or supercritical CO_2 in Brayton cycle) from the turbine to heat sink via a bypass line
  - Usually not a safety-grade system
- In the event BOP path is not available, shutdown heat removal is achieved via redundant safety grade decay heat removal systems
  - To maintain continuous effective core cooling and keep the primary system component temperatures below allowed limits during postulated accidents
  - Can be based on passive heat removal mechanisms (using natural convection with no valves or mechanical devices to control its operation)

Decay Heat Removal Systems (cont.)

- Decay heat removal system options



RVACS--Reactor Vessel Auxiliary Cooling System

VCCS--Vessel Cavity Cooling System

DRACS--Direct Reactor Auxiliary Cooling System

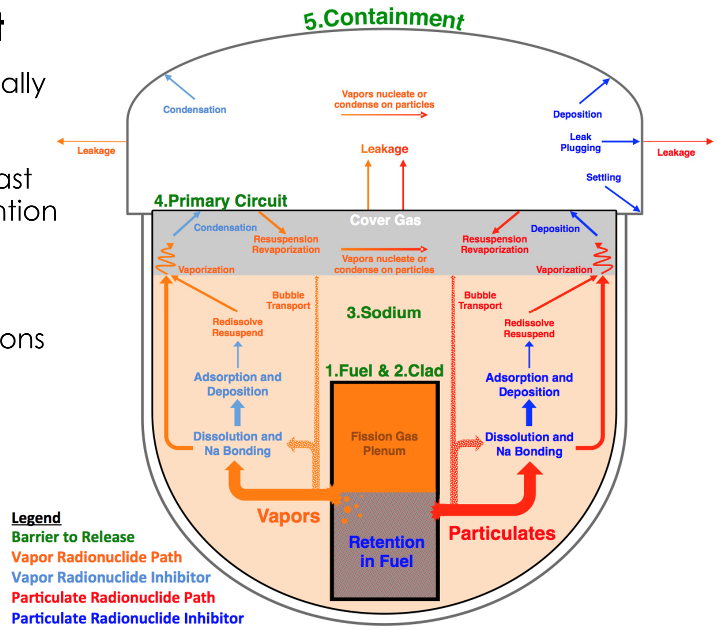
PRACS--Primary Reactor Auxiliary Cooling System

IRACS--Intermediate Reactor Auxiliary Cooling System

SGACS--Steam Generator Auxiliary Cooling System

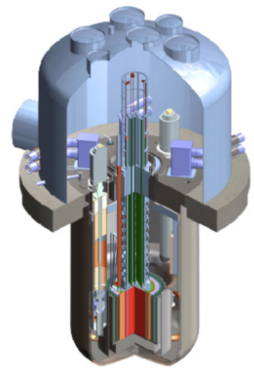
Containment

- SFR designs typically rely on a containment structure as the last barrier for prevention of uncontrolled release of radioactivity in accident conditions



Containment (cont.)

- SFR containment systems have evolved
  - Early systems were over-designed against energetic events from a Hypothetical Core Disruptive Accident (HCDA)
  - Experiments and analyses indicated that such events are exceedingly rare, and the energy releases are far less than early analyses indicated
  - In most modern designs, containment design basis is a large sodium fire
    - Sodium aerosol analyses and experiments indicate agglomeration along with plate-out in the systems inside containment
  - In pool designs, guard vessel may provide containment function
  - In the loop designs, all primary piping is double walled to provide containment function
  - Some designs propose an underground reactor with a dome over the reactor vessel



Refueling System

- Refueling is done with the vessel head in place, unlike LWRs where the head is removed
- Sodium is opaque, so visual guidance is not available (some concepts of ultrasonic imaging are being considered)
- Refueling can be done through rotating plugs in a rotating head, providing access to all areas of the core
- Some concepts use refueling mechanism independent of top head plugs
- Some concepts store spent fuel in the primary vessel—some store externally
- Fuel must be in inert gas throughout the process
- Typically, refueling starts 2 days after shutdown (a fuel assembly will generate ~30–40 kW at that time) and takes 2 weeks
  - Typically replaces one-third of the core every year

Refueling in SFRs is a more complex issue than for LWRs because the sodium is opaque and highly reactive with air and moisture and fuel must be under sodium and protected from contact with air or water. After sufficient cooling, the fuel is washed with water to remove traces of sodium before transport to the reprocessing plant or storage.

Instrumentation

- Liquid metal coolants pose unique instrumentation challenges
- Critical core parameters:
  - **Flux**: In-core, ex-core (in-vessel), and ex-vessel neutron detectors
  - **Temperature**: Resistance Temperature Detectors (RTDs) and thermocouples throughout the primary and intermediate loops to determine thermal power, operating conditions, and monitoring for anomalies
  - **Flow**: Venturi flowmeters (accurate but with slow response time) and magnetic flowmeters (less accurate but faster response time) to complete the thermal power calculations, determine loop operating conditions and monitor flow anomalies
  - **Pressure**: Via NaK-filled capillary tube.

Instrumentation (cont.)

Flux

- Flux monitoring is typically done by a group of neutron detectors located in the reactor cavity external to the reactor vessel
  - Feasible due to longer mean free path of fast neutrons
  - Needed to protect the instruments from irradiation damage (in-core or in-vessel detectors may be used during initial startup)
- Signals from these detectors are used for both the reactor control system and the plant protection system

Instrumentation (cont.)

Temperature

- Sodium temperature is measured throughout the primary and secondary loops
  - Calculate thermal power
  - Determine loop operating conditions
  - Monitor for potential abnormal activities
- Two types of sensors are commonly used
  - Resistance temperature detectors (RTDs)
    - Provide a highly accurate and reliable measurement
    - Do not require a cold junction (as do thermocouples)
  - Thermocouples

Instrumentation (cont.)

Flow

- Flow measurements complete the thermal power calculations and to determine loop operating conditions
- Flow measurement sensor types
  - Venturi flow meter (converging–diverging nozzle with pressure difference measurement)
    - Highly accurate but with slow response time
    - Not appropriate for reactivity control and shutdown systems
  - Magnetic flow meter
    - Less accurate but with rapid response time
- Flow sensor calibration
  - Venturi flow meter is used to provide in-place calibration of the rapid-response magnetic flow meter
  - Another calibration method is sodium activation with a pulsed neutron device and using the time-of-flight technique
    - May not be practical in large-scale engineering systems with high-background radiation level

Flow meters, or other instrumentation, requiring separation of the fluid stream from the sensors, must be protected against freezing of the sodium, (which freezes roughly at the boiling point of water).

Instrumentation (cont.)

Pressure

- Liquid pressure measurements are normally made by routing a small column of high-pressure liquid onto one side of a sensing diaphragm
- This causes complication with sodium because sodium solidifies well above room temperature
- Alternate method is to interface the sodium with NaK via a bellows system. (NaK is liquid at room temperatures and requires no trace heating)

Instrumentation (cont.)

Failed fuel detection and location

- Failed fuel can be detected by sensing fission products in the cover gas; xenon isotopes have sufficiently high gamma energy to allow detection. Cover gas adsorption by carbon filters can concentrate the xenon
- With ~300 fuel assemblies in the reactor, locating the failed fuel is a more difficult task
- A technique that has been used successfully in EBR-II and FFTF is gas tagging
  - Mixtures of xenon and krypton isotopes can provide over 100 unique gas tags (each fuel assembly can have its unique tag identification mixture)

Gas tagging increases the expense of fabrication and tracking of fuel elements, which can be an issue with commercial SFRs.

Instrumentation (cont.)

Leak detection

- Important because:
  - Primary sodium is radioactive
  - Liquid sodium will burn in air
  - Loss of sodium could impair heat transport systems
- Leaks can be detected by conductivity probes (usually in low spots below sodium-containing tanks) or by sensing of sodium aerosols
- Sodium level monitoring—particularly important where sodium inventory is crucial
  - Level monitoring can be done with electrical induction probes
  - Needed for sodium inventory tracking

Auxiliary Systems

Inert cover gas

- Nitrogen used as inert gas in cells with sodium-containing systems
- Nitrogen cannot be used at temperatures >400°C because of nitriding problems with steel
- Argon used as cover gas within vessels and components because it does not react with structural materials and is inexpensive for an inert gas
- Argon subsystems provide pressure control and atmosphere for all sodium-gas interfaces
- Because of possible radioactive contamination, radioactive argon processing system (RAPS) is needed to remove xenon and krypton isotopes

Auxiliary Systems (cont.)

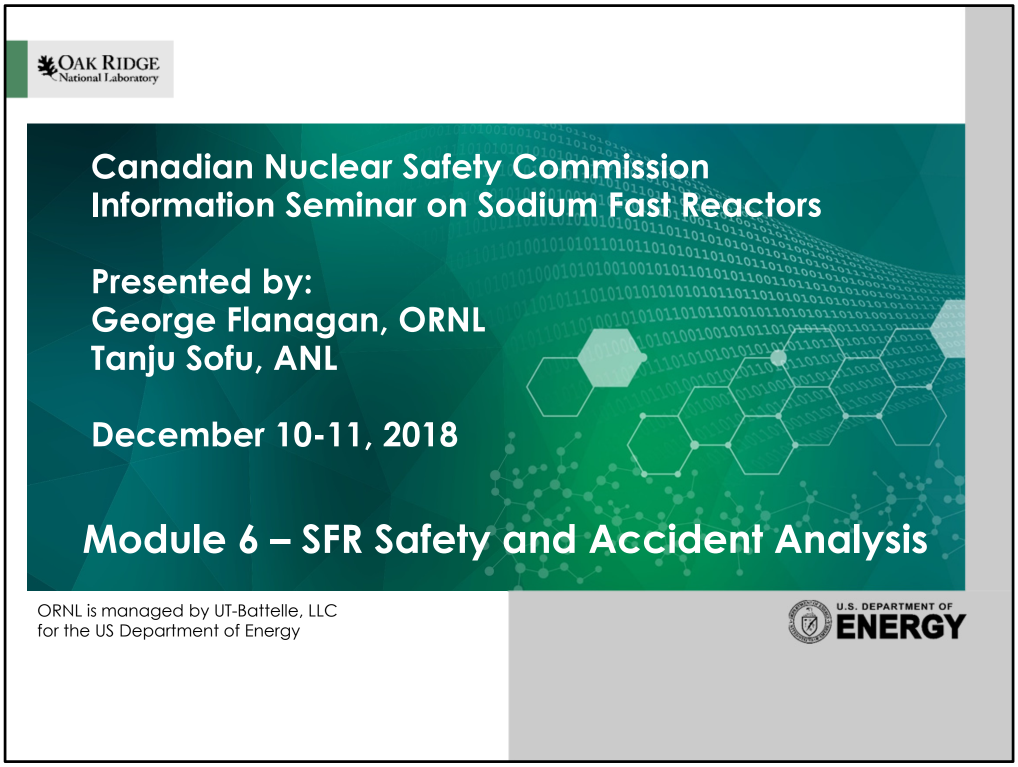
Trace heating

- Sodium solidifies at 98°C, so it must be heated at reactor low power to keep it in liquid state
- Trace heaters provide a heat flux of about 10 to 20 kW/m^2
- Trace heating systems can require 10 MW during startup (cold core) conditions—less for when pumps can be used for heating

Ref: Waltar and Reynolds, op. cit., pages 498-499

Other Systems

- Other systems unique for SFR may include:
  - Sodium purification system
  - Cover-gas cleanup system
  - Na fire protection
  - Cell inerting systems
  - Cell liners
  - Under the head refueling systems
  - Ex-vessel fuel handling and storage
  - Seismic isolation
  - Unique inservice inspection systems for opaque coolant



Safety and accident analysis is a core issue regarding NRC involvement in SFR licensing. The safety analysis culture arising from LWR experience started with deterministic analysis based on prescribed accident initiators. Over time, probabilistic accident analysis has provided a route to risk informed regulation. SFR safety analysis is based on a historical base of deterministic analysis, modified over time with probabilistic analysis.

Outline

- Defense-in-depth and plant states considered in design
- Design characteristics that impact safety
- Safety approach for AOOs, DBAs, BDBAs (DECs), and severe accidents
- Accidents and their classification
- Important transient phenomena and fuel behavior
- Safety analysis codes and methods
- Backup material
  - Accident Types Comparison
  - SFR Event Descriptions
  - Evaluation of Phenomena

Accident events fall into two basic categories: design basis events (DBE) and beyond basis events (BDBE). BDBEs can lead to severe accidents. Analysis of accident sequences depend on sufficient understanding of phenomena and is embodied in accident analysis codes, which are covered in another area.

AOO-Anticipated operational occurrence
DEC-Design extension condition

Defense-in-Depth, Risk and Safety

- Defense-in-depth is the key concept on which fast reactor safety is based
- Fast reactor safety and reliability goals:

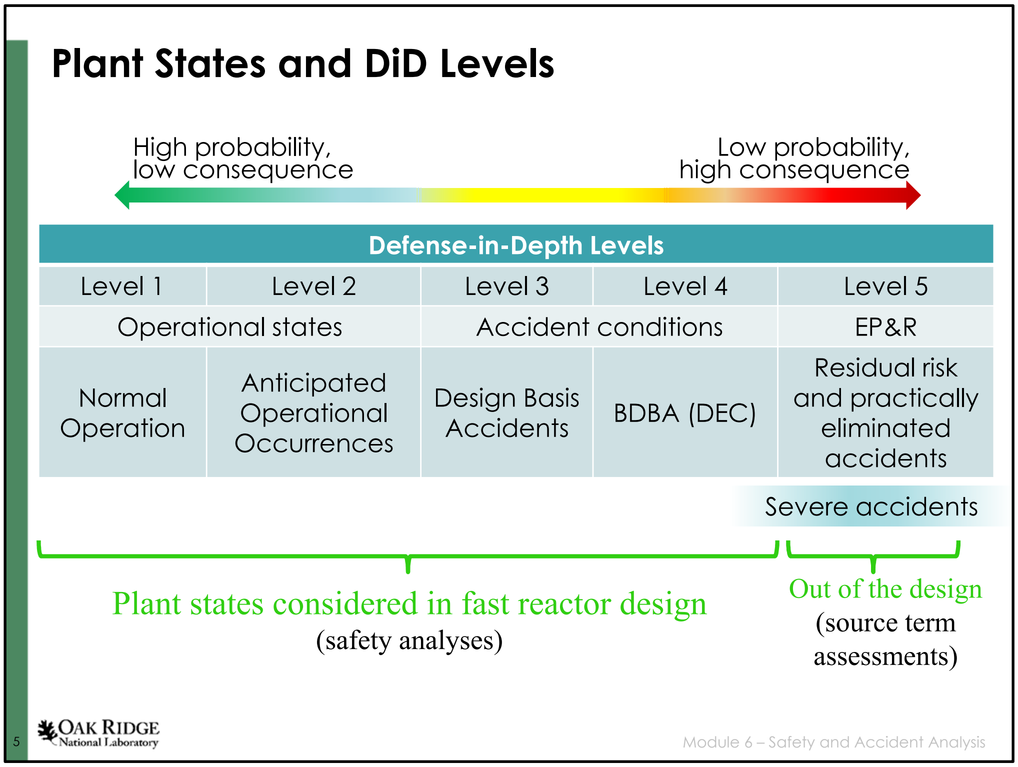
	DiD level
– Excellence in operational safety and reliability	1 and 2
– Low likelihood and degree of core damage	3 and 4
– Smaller emergency planning zone	4 and 5

- The traditional approach to demonstrating adequacy of defense-in-depth is deterministic, but a combination of deterministic and probabilistic approaches is increasingly being adopted
- Risk-informed safety approach considers both probability and consequences of postulated accidents
  - Accidents with large consequences are reduced in risk significance by requiring that their likelihood are acceptably small

Defense-in-Depth Levels

- **Level 1** – Prevention of operational failures
  - Achieved by proper (and compatible) selection of fuel, cladding, coolant, and structural materials, and by following high quality practices in construction and operation
- **Level 2** – Control of abnormal operation and detection of failures
  - Achieved by providing large margins between normal operating conditions and limiting failure conditions, and surveillance features for detection of anomalies
- **Level 3** – Control of accidents within the design basis
  - Achieved by conservative design and engineered safety systems for reactor shutdown, decay heat removal, and emergency power
- **Level 4** – Control of severe plant conditions, including prevention of accident progression and mitigation of consequences
  - Achieved mainly by the containment structure but also via accident mitigation measures including in-vessel retention and maintaining a coolable configuration, as well as accident management guidelines
- **Level 5** – Mitigation of radiological consequences should significant releases of radioactive materials occur
  - Achieved by off-site emergency response (sheltering, evacuation, …)

Definition of DiD levels are based on IAEA standards.
Levels 1 and 2 are for normal operation and SAFDL applies.
Level 3 covers DBAs. The acceptance criteria for these events such that they should not have a release greater than 10% of the TEDE. Analyses of the events in this category needs to be conservative.
Level 4 are for BDBAs, and may involve reliance on inherent safety features of the design (in case engineered protection systems fail). The acceptance criteria for these events such that they should have a release less than the TEDE. Analyses of the events in this category are based on best-estimate methodologies using realistic values.
Level 5 is for emergency response for accident with radioactivity releases to the environment.



Severe accidents can be pushed into the residual risk category, especially with metal-alloy fueled, pool type SFRs.

But they can also be considered within level-4 of DiD since most international concepts are based on oxide fuel with vulnerability to HCDAs.

EP&R: Emergency Planning & Response

Plant States and DiD Levels (cont.)

- AOOs are typically handled via reactivity control system and BOP for heat sink whereas DBA's are handled via safety grade systems (shutdown system and DHRS)
- BDBAs/DECs are typically handled by inherent safety (metallic fuel), or with addition of non-safety grade passive devices (oxide fuel)
  - Plus an ultimate shutdown system (that can be manually operated) and diverse non-safety grade DHRS paths.
- BDBA/DEC category is often split in two:
  - Typical higher frequency BDBA/DEC events are ATWS (an **AOO** plus failure of reactor shutdown system)
  - Typical lower frequency BDBA/DEC events are unprotected accidents (**DBA** plus failure of reactor shutdown system--but the passive devices are still available)
  - In oxide-fueled designs, unprotected accidents could lead to severe accidents whereas in metal fueled pool-type designs, severe accidents can be pushed into the residual risk category
- Practically eliminated accidents are those against which mitigation measures are not considered

Impact of SFR Neutronics on Safety

- Fast energy spectrum requires for much finer multi-group cross-section structure to resolve neutron reactions
- Fast spectrum leads to ~10× longer neutron mean-free paths
  - Negligible spatial self-shielding
  - Greater sensitivity to minor geometric changes due to enhanced neutron leakage
  - Reactivity perturbations impact the core as a whole, not locally
- Complex reactivity feedback mechanisms (not just Doppler)
- Higher enrichment needed to achieve criticality with uranium cores
  - Core is not in most reactive configuration and design must ensure recriticality (e.g., due to core compaction) does not occur
- Long core life (even no refueling) with breed-and-burn concepts
- Pu-bearing fuels have lower effective delayed neutron fraction (β_{eff})
  - Results in a lower margin to prompt criticality during reactivity transients
  - In breeder concepts (conversion ratio > 1), equilibrium core β_{eff} can be $1/3^{rd}$ of beginning of life core
- Shielding challenges unique to fast neutron spectrum

Thermal-Fluid Design Impact on Safety

- Compact lattice (spacing is typically provided by a thin wire wrapped around each fuel pin) and high core power density (~5X in comparison to an LWR)
- Unpressurized primary and intermediate heat transport systems
  - No LOCA or need for high-pressure injection system (guard vessel--and guard pipes in loop designs--to maintain coolant inventory
- High temperature operation (>500°C core outlet temperature)
  - Material challenges due to thermal creep and fast fluence
- Large thermal inertia with long grace period
- Natural circulation potential
  - ΔT is ~150°C during normal operation (>300°C during accidents) leading to significant sodium inlet/outlet density difference and large buoyancy
- Large margin to sodium boiling
  - Boiling should be avoided (can only be expected only during highly unlikely accidents with large-scale fuel failures)

Thermal-Fluid Design Impact on Safety (cont.)

- High fuel thermal conductivity of metal fuel and high gap conductance from bond sodium help maintain low fuel temperatures and flatter radial temperature profile
- Top-level thermal and fluid design requirements are based on fundamental heating (linear power, heat flux) and cooling (coolant heat transfer and flow) performance characteristics
  - Implications of core configuration, fuel type, material compatibilities and corrosion concerns, pumping power, burnup considerations, thermal and mechanical limits…
    - Most are interdependent factors
- Thermal-fluid design limits
  - Peak centerline temperature, margin to melting
  - Peak cladding temperature, margin to cladding failure
  - Peak coolant temperature, margin to coolant boiling

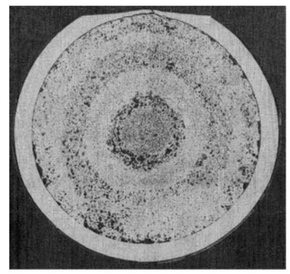
SFR Accidents

- Loss of Coolant (LOC): Reactor vessel or primary piping leak (in a loop-type SFR)
  - Failure to mitigate could cause loss of decay heat removal function or core uncovering
  - Highly unlikely due to reliance on guard vessels and guard piping (in loop-type SFR)
- Loss of Flow (LOF): Possible causes are pump failures or loss of pumping power, which requires flow coast-down enhancement to transition to natural circulation
- Loss of Heat Sink (LOHS): Failures in power conversion system (steam generator upset or turbine trip)
  - SFR designs include auxiliary decay heat removal systems that operate in active mode or based on natural convection (that do not require activation)
- Transient Overpower (TOP): Possible causes are uncontrolled withdrawal of control or shutdown rods/elements, sodium voiding in center of the core

Postulated SFR accidents do not include rod ejection or dropout
Fast neutron spectrum systems do not have Xenon burnout power changes

SFR Accidents (cont.)

- Station blackout: Simultaneous loss of power for primary, intermediate, and energy conversion system pumps
- ATWS: Anticipated transients without scram: An AOO combined with failure of reactor shutdown system
- Unprotected event: A DBA combined with failure of reactor shutdown system
- Local faults: Statistical fuel failures due to fuel fabrication defects, fuel loading or enrichment errors etc.
  - Metallic fuel is compatible with sodium coolant and local faults can be tolerated for an extended period with proper monitoring of fission gas release
  - Demonstrated during the Run Beyond Cladding Breach tests at EBR-II with no fuel loss or significant liquid or solid fission product escape from fuel pin



Metal fuel (12 at-% burnup) after 5 ½ month-long RBCB Test

EBR-II RBCB experiments

An area of cladding was machined down to 25-50 µm (<10% of cladding thickness is left)

After a short period of irradiation, cladding failure occurred at the machined spot

Metal fuel shown ran 169 days after failure (before the PIE was performed)

Figure ref: T. Sofu, "A review of inherent safety characteristics of metal alloy sodium-cooled fast reactor fuel against postulated accidents," *Nuclear Engineering & Technology*, Volume 47, Issue 3, pages 227-239 (August 2015).

Classification of Events

Events	Frequency	Expected Consequences
Anticipated Operational Occurrences (AOOs)	Expected during the lifetime of the plant (>10^{-2} per reactor year)	None. Maintain large margin to fuel failure
Design Basis Accidents (DBAs): Typically failure of one safety-grade system	Not expected to occur during the lifetime of the plant but anticipated in the design (>10^{-4} per reactor year)	Minor fuel damage permissible for lower probability events (<10^{-3} per reactor year). Individual (offsite) exposure below allowable limit
Beyond Design Basis Accidents (BDBAs) or Design Extension Conditions (DEC): Multiple failures of safety-grade systems, including ATWS events	Highly unlikely accidents not expected to occur during the lifetime of the fleet but considered in the design (>10^{-6} per reactor year)	Substantial fuel damage permissible for lower probability events (<10^{-5} per reactor year). Public exposure below allowable limit
Severe Accidents	<10^{-6} per reactor year	Propagation of fuel damage, potentially leading to loss of core integrity and coolable geometry
Early or Large Releases	<10^{-7} per reactor year	Emergency response

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Module 6 – Safety and Accident Analysis

In the U.S., allowable dose limit is 25 rem

Approach for AOO and DBA

- Like LWR, fast reactor safety is first based on utilization of multiple, redundant engineered protection systems to lower the probability of accident occurrence and limit its consequences:
  - Independent reactivity control and shutdown systems
  - Multiple coolant pumps and heat transport loops
  - Diverse decay heat removal systems
  - Multiple barriers to release of radioactive materials
- Unique design features of LMR provide additional measures to protect these reactors during AOOs and DBAs:
  - Superb heat transfer due to high thermal conductivity of liquid metal coolant (70 W/m-K for sodium vs. 0.6 W/m-K for water).
  - Large margin to coolant boiling (~350°C in SFR vs. ~20°C in PWR)
  - Large thermal inertia (long grace period during transients)
- Analyzed using conservative approach or BEPU method

Margin to boiling is even larger for an LFR (boiling can be completely ruled out).

Approach for BDBA (DEC)

- Multiple-failure events that include ATWS (AOO followed by shutdown system failure) or even a much less-likely unprotected event (a DBA followed by shutdown system failure)
- Measures to prevent these occurrences and mitigate their consequences should also be considered in the design
  - Design features that enhance net negative inherent/passive reactivity feedback and passive decay heat removal
- Independence and diversity of preventive design measures in Level 4 of DiD (from those relied in Level 3) are advised
  - Due consideration of potential for common cause failures
- Containment structure to prevent release of radioactivity to the environment as the last barrier (also against external events)
  - Sodium fires that could challenge the containment integrity needs to be specifically addressed
- BDBAs (DECs) are analyzed using best estimate method

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Module 6 – Safety and Accident Analysis

When inherent safety isn't enough, passive reactivity control devices can be used (GEM, Curie point detachment of control rods, hydraulically suspended rods)
The third major bullet needs to be emphasized: Separate measures for BDBA's that we do not take credit for in DBAs is recommended.

Approach for Severe Accidents

- Depending on the design choices and characteristics, severe accidents can be pushed under the residual risk category
  - Inherent/passive safety characteristics and choice of fuel
  - Complex reactivity feedback mechanisms for LMRs
  - Supplementary passive reactivity control devices if needed
  - Proven capabilities during EBR-II inherent safety demonstration and FFTF passive safety testing programs
- If the core damage cannot be prevented, in-vessel retention and core debris coolability need to be assured
  - Reduce the potential impact on the containment function
- Severe accidents that could lead to a significant and sudden radioactivity release has to be practically eliminated:
  - Simultaneous failure of the reactor and guard vessels
  - Complete loss of decay heat removal capability

Reactivity feedback mechanisms include Doppler feedback, fuel axial expansion, core radial expansion, coolant density change, CRDL expansion…
Practical elimination require robust demonstration with very high degree of certainty (not just reduced probability).

Approach for Emergency Planning & Response

- Level 5 covers residual risk events (including the practically eliminated accident sequences)
  - Requires off-site emergency planning and response
- Mechanistic source term (MST) assessments for a range of bounding multiple-failure accidents are recommended:
  - Severe loss of decay heat removal capability
  - Severe loss-off-flow cases (multiple pump seizures)
  - Severe failures in spent fuel storage systems
- MST development process:
  - Identification of radionuclide inventory and sources
  - Modeling of radionuclide transport pathways and phenomena
  - Evaluation of a class of bounding accidents
- Other aspects of Emergency Planning and Response are similar to those employed for LWRs

Inherent/Passive Safety

- Essence of the inherent/passive safety is to rely on intrinsic characteristics of the design to maintain a balance between generated heat and reactor cooling capability to prevent core damage when engineered safety systems fail
- The focus of inherent safety is to avoid:
  - Large uncontrolled increases in core power
  - Insufficient cooling of the reactor core
  - Rearrangement of fuel that could lead to a recriticality
- Inherent/passive safety uses three basic principles:
  - Favorable reactivity feedback (through core physics and structural design)
  - Sufficient natural circulation cooling for decay heat removal
  - Appropriate selection of fuel and cladding materials

Reactivity Feedback Mechanisms

- Doppler feedback: Effect of changes in neutron fission and absorption cross sections due to Doppler broadening
  - Negative at temperatures above normal
- Core radial expansion: Due to thermal expansion, irradiation-induced swelling, and irradiation-enhanced creep
  - Negative at temperatures above normal due to enhanced leakage
- Fuel axial expansion: Effect of thermal expansion and transient swelling of especially the metallic fuels (and cladding)
  - Negative at temperatures above normal due to reduced number density of fissionable isotopes
- Coolant density and void worth: Effect of changes in Na coolant atom numbers at elevated temperatures
  - Can be positive due to reduced Na moderation/absorption, or negative due to enhanced neutron leakage
- Control rod drive line expansion: Due to difference in thermal expansion of control-ride driveline and reactor vessel
  - Can be positive or negative depending on CRDL expansion relative to reactor vessel expansion

In SFR safety analyses, some of these individual reactivity feedback mechanisms are lumped into an integral quantity such as the power coefficient, or the temperature coefficient.
In this slide, we list them based on phenomena they are related to.

What to look for in a Design Review?

Safety analyses are always concept specific and response of a design cannot be easily generalized; however, some fundamental principles apply:

- The design should employ a guard vessel (pool) or guard piping (loop) with enough capacity so that, in case of a leak, core remains covered and decay heat removal systems retain their function
- Reactivity control and shutdown systems should have sufficient reactivity to secure a safe shutdown from the most reactive core state assuming failure of the highest-worth control assembly
- Decay heat removal system(s) should have sufficient capacity to avoid fuel failures and assure integrity of primary coolant boundary (accurate assessment of decay heat level is important)
  - Unless separated by double barriers, residual heat removal system (RHRS) coolant should be compatible with primary sodium coolant and kept at a slightly higher pressure so that leaks result in flow of RHRS coolant into the primary system

If the decay heat removal paths are not open during a RV leak, it may not matter if the core remains covered
Safe-shutdown state is usually defined as the shutdown at a temperature at which refueling (or core unloading) operation van be performed.
If several control elements are connected as a “bank of rods”, failure to insert the whole bank of rods can be considered
Requiring the primary coolant boundary integrity to be maintained as an RHRS function is unique to FRs. In LWRs with LOCA as the bounding DBA, the coolant boundary is already compromised.
Pressure difference is typically achieved via elevation difference and prevents activated primary sodium contaminating the RHRS that bypasses the containment structure.

What to look for in a Design Review? (cont.)

- If a safety-grade RHRS is placed along the IHTS loop path, IHTS should also be a safety grade system
  - Otherwise, ITHS does not provide a safety function other than being a barrier between PHTS and BOP
  - Unless separated by double-layer tubes in IHX, IHTS coolant should be compatible with primary coolant and kept at slightly higher pressure so that IHX leaks result in flow of IHTS coolant into the primary system
- Low pressure and single-phase conditions of the primary coolant system means that SFR containments can act only as a barrier
- But containment structure should have some pressure retaining capability against the heat and pressure from a sodium fire
  - Inert compartments with steel liner are desirable
  - Should not contain any source of water that could ingress into RV
  - Protection against external events can be fulfilled through a hardened reactor building that is not leak-tight
- Since containment isolation valves can interfere with reliability of DHRS and IHTS functions, their use in lines penetrating the containment should be reconsidered through a risk assessment

Containment pressure even from a large sodium fire would be only a fraction of the pressure in an LWR containment after a LBLOCA. Heat from a sodium fire could be a greater source of concern for an SFR containment (usually a steel liner surrounded by a hardened reactor building against external events).

IHTS-Intermediate heat transport system

What to look for in a Design Review? (cont.)

- Core vs. IHX elevation difference should be sufficient to facilitate effective natural circulation
- Core vs. DHRS heat exchanger elevation difference should also be sufficient to allow passive decay heat removal if needed
- Pump coast-down should be sufficiently slow to avoid coolant boiling during the early-phase of a LOF accident (when power-to-flow ratio is > 1) and it needs to be modeled accurately
- If design features in-vessel spent fuel storage, heat load from the stored spent fuel should be included in the analyses
- Interference of active and passive systems could be a source of concern
  - Passive reactivity control systems may not be relied on if the pumps are still running
  - Coolant can freeze if both BOP and DHRS are functional at decay heat levels
- Capturing the impact of passive system reliability in a risk assessment is not trivial (may require dynamic PRA techniques)

Elevation differences are the key parameters that sets the natural circulation flow rates. Designers try to limit that to cut down the commodity cost (for a more compact reactor vessel); so, this is something to be verified via confirmatory analyses.
Pump coast down is also a key parameter. Designers often assume some pump characteristics based on earlier experiences but each pump may be different.
Heat load from in-vessel storage may also impact trace heating capacity assessments (in which case, storage locations should be considered empty).
Interference of active and passive systems is an interesting topic. During conservative safety assessment of DBAs, we assume worst case conditions (power is not available etc). In real life, however, a DBA may proceed at a different sequence (systems designated for different DiD levels may overlap in their function and interfere with each other). A PRA will be key to assess risk of such cases. And capturing the response of a passive system (like decay heat removal that relies on tedious balance between the friction and buoyancy forces) is a tricky business.

What to look for in a Design Review? (cont.)

- Fuel design limits for a given fuel/cladding combination should include the impact of "time-at-temperature"
  - Often captured in terms of "Cumulative Damage Fraction (CDF)"
- Independence, and more ideally, diversity of design features at different levels of DiD is key to a successful design
  - This can be achieved in different ways, but it needs to be carefully evaluated; possible combinations are
    - Reactivity control:
      - Control system (AOO), shutdown system (DBA), inherent safety with ultimate shutdown system (BDBA)
      - Control system (AOO), shutdown system (DBA), self-actuated shutdown system (BDBA)
    - Decay heat removal:
      - BOP (AOO), active mode DRACS (DBA), passive mode DRACS (BDBA)
      - BOP (AOO), active or passive mode DRACS (DBA), RVACS (BDBA)
- Evaluation methodologies should be conservative or BEPU for AOO and DBA, and best estimate methods for BDBA
  - For practically eliminated cases with no mitigation feature, a BEPU is recommended to account for uncertainties against cliff-edge effects

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Simple temperature limits for SAFDL can be both too prescriptive or not sufficiently conservative. In absence of hard limits such as DNBR or CHF and a single bounding event like LOCA for LWRs, SFR fuel forms can maintain their integrity at very high temperature for a short duration while they may fail at a much lower temperature if the cladding is exposed to that temperature sufficiently long enough that time-dependent thermal creep induced failure of the cladding takes place. This "time-at-temperature" phenomena if often captured through a cumulative damage fraction concept that can be leveraged not only for NO/AOO, but also be DBA and BDBA classes, establishing different acceptance criteria at different DiD levels.

BEPU-Best estimate plus uncertainty

How to avoid Core Damage during Unprotected Events?

- When shutdown system fails to scram the reactor, key early measure is to maintain the coolant temperature below its boiling point
- The net negative reactivity feedback (through inherent or passive means) should eventually bring the reactor power into equilibrium with the available heat rejection rate as the system approaches an asymptotic temperature distribution
  - Long-term goal is to keep the asymptotic cladding, reactor vessel, support structure temperatures below creep limits
- Avoiding core damage, therefore, depends on:
  - Providing sufficient negative reactivity feedback to overcome the initial power-to-cooling mismatch, and
  - Reducing the reactivity feedback components (mainly Doppler) that resist the return of the system to equilibrium temperatures

Desired Response to ULOF Events

- Initiator is loss of power to the primary coolant pumps coinciding with failure of the plant protection system
- As core flow decreases, temperature rises and net negative reactivity feedback reduces the power
  - As the power falls, the coolant outlet temperature also begins to decrease with some delay
- With properly designed coast down of the primary coolant pumps, the coolant boiling should be avoided with substantial margin in the short term
- With properly sized passive decay heat removal systems, longer-term transient temperatures should be kept below the levels at which load-stress-induced creep could result in structural failures

Desired Response to UTOP Events

- Typical initiator is an uncompensated withdrawal of a single, maximum-worth control rod (or bank of rods)
- In a metallic-fueled core with a low cycle burnup reactivity swing, the withdrawal of a single rod typically amounts to an insertion of smaller amount of reactivity in comparison to oxide systems
- Reactor power rises above nominal, followed by a heating of the core and the coolant which should introduce sufficient negative reactivity to return the reactor power gradually to equilibrium with the assumed nominal heat rejection at the steam generators
- The low control rod worth in a core with a metallic fuel is an advantage in comparison to oxide fuel core

Desired Response to ULOHS Events

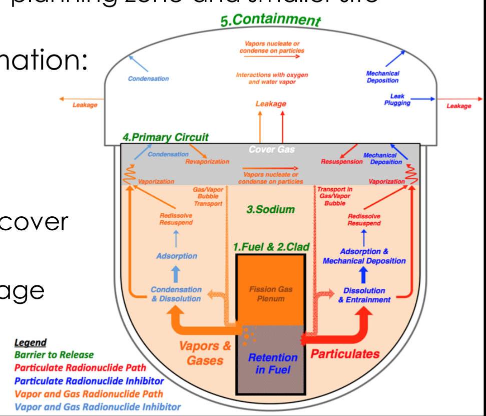
- Feedwater supply to the steam generators is lost with simultaneous failure of the plant protection system, resulting in a gradual heating of the intermediate and primary coolant systems and an increase in the core inlet temperature
- Heating of the core support grid spreads the core radially, introducing key negative reactivity feedback (in addition to Doppler) that should reduce the reactor power
- In the long term, the reactor power should equilibrate with any available heat sink as the inlet temperature remains elevated above its initial steady-state value
  - Peak temperature should be well below boiling point
  - Asymptotic temperature should be below levels at which load-stress-induced creep could result in structural failures

Metal Fuel Performance during Accidents with Fuel Failures

- For metal fuel, scenarios that lead to temperatures sufficient to melt the fuel and/or fail the cladding do not result in blockages
  - Metal fuel has relatively low melting point and it forms eutectic alloys through chemical interaction with the cladding (at temperatures well below the cladding melting point)
  - Failures are predictably near the top of the fuel column
  - Temperature of the above core region is often at or above the melting point of the relocating fuel/steel-eutectic mixture
- Transient over-power experiments at TREAT demonstrate that the fuel/steel-eutectic mixture is carried well above core structure without blockages, resulting in early termination of rapid transient overpower and severe loss-of-heat-sink events
  - Experiments have not yet been performed for severe loss-of-flow conditions, but simulations using phenomenological models predict similar early termination

Mechanistic Source Term Assessments

- For scenarios with core damage, a mechanistic source term (MST) analysis is performed
  - Attempts to realistically assess the transport/retention and release of radionuclides from the plant for specific scenarios
  - Allows for an accurate representation of the many radionuclides barriers present in a metal fuel, pool-type SFR
  - Important for reduced emergency planning zone and smaller site boundary
- MST utilizes scenario-specific information:
  - Burnup level of fuel batches
  - Timing of accident scenario
  - Conditions of fuel pin failures
  - Conditions of the primary sodium, cover gas region, and containment
  - Design information regarding leakage from reactor vessel head and containment



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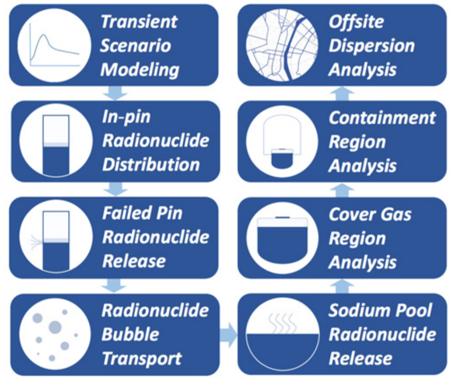
Mechanistic Source Term Assessments



- Calculation involves many steps, which coincide with radionuclide transport pathway
  - Radionuclide inventory in each fuel batch at the time of accident
  - Migration of radionuclides within the fuel pin during irradiation
  - Release of radionuclides from failed fuel pins, including entrainment in bubbles within the pool
  - Removal of aerosols/vapors from bubbles due to "scrubbing" in pool
  - Release of radionuclides to the cover gas region from bubbles and vaporization from sodium pool
  - Radionuclide aerosol behavior in the cover gas region and containment
  - Chemical interactions of sodium vapor/aerosols with O2 and steam
  - Leakage from the cover gas region and containment

Sodium Accidents

- Liquid sodium coolant reacts with air, water and concrete
  - Need be mitigated to avoid their impact on SSCs important to safety
- Sources of sodium leakage inside of containment
  - Sodium from primary loop piping in a loop type SFR
  - Sodium from intermediate loop piping inside the containment
  - Primary sodium from a sodium storage system (if any)
  - Primary sodium from purification system
- Sodium reaction scenarios considered in licensing are those with the potential of leading to radioactive releases
  - Primary sodium fires
  - Low pressure (< 0.5 MPa) intermediate sodium leak
    - Characterized by Na pouring onto the containment floor
  - High pressure (~ 0.5 MPa) intermediate sodium leak
    - Could cause a dispersed sodium spray in the containment atmosphere
  - Steam Generator (SG) tube rupture

Only liquid sodium reacts with air and concrete. It just oxidizes slightly when it is at room temperature.

Identified sources of sodium leakage inside the containment is largely deterministic (non-mechanistic).
There is no specific event sequence associated with such phenomena; so, these events do not have a frequency associated with them.

Sodium Accidents (cont.)

- Implications of sodium fires
  - Impact of elevated temperatures on SSCs including containment
  - Containment atmosphere temperature and pressure
  - Deposit of aerosols from sodium fires onto SSCs
  - Integrity of IHTS from steam generator tube ruptures
- Phenomena involved in sodium leaks and fires
  - Oxygen availability/deficiency (inert cells in small compartments)
  - Phenomena relevant to low-pressure leakage
    - Surface combustion and oxygen transport to surface (often impeded by deposits)
    - Heat transfer from surface to atmosphere and structure (aerosol/smoke formation)
    - Sodium-concrete interaction (usually prevented by use of steel liners)
  - Phenomena relevant to high-pressure leakage (use double-walled piping reduces potential for sodium spray)
    - Jet/spray breakup and spray combustion
    - Heat transfer from spray
    - Aerosol (smoke) formation from spray
  - Heat transfer from atmosphere to structure

Fast Reactor Analysis Codes and Methods

- Despite closure of historical facilities (EBR-II, FFTF), U.S. continues to dedicate R&D efforts to support deployment of novel experimental, prototype, and commercial designs
- Development of codes supporting SFR design and analysis is ongoing over last six decades, with more recent emphasis on high-fidelity approaches
- Current code capabilities are robust and envelope all facets of SFR design and safety analyses
- Functional areas:
  - Core design and steady-state characterization
  - Transient system analyses
  - Fuel performance assessments and transient response analysis
  - Structural design assessments
  - Sodium fire and containment response assessments

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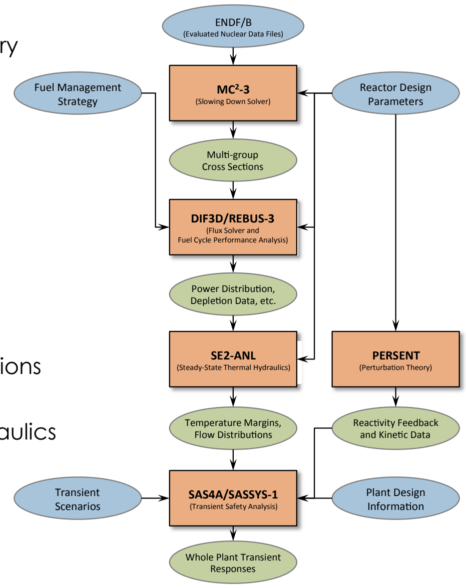
Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety, Sandia National Laboratories, June 2011
Sodium Fast Reactor Safety and Licensing Research Plan — Volume I, Sandia National Laboratories, May 2012
Assessment of Regulatory Technology Gaps for Advanced Small Modular Sodium Fast Reactors, Argonne National Laboratory, 2014
Advanced Reactor Technology — Regulatory Technology Development Plan (RTDP), Idaho National Laboratory, 2015

Codes and Methods Integration

- MC^2-3
  - Multi-group cross-section generation
  - Consistent P1 multi-group transport theory
- DIF3D/VARIANT and PROTEUS
  - Neutron diffusion or transport theory
  - Neutron flux and power distributions
- REBUS
  - Fuel cycle performance analysis
  - Depletion analysis, enrichment search
  - Equilibrium or non-equilibrium states
- VARI3D and PERSENT
  - Transport-based perturbation theory
  - Reactivity feedback coefficient distributions
- SE2-ANL, SAM, and Nek5000
  - Steady-state, sub-channel thermal-hydraulics
  - Peak fuel and cladding temperatures
  - Flow orifice optimization
- SAS4A/SASSYS-1 and SAM
  - Transient safety analysis



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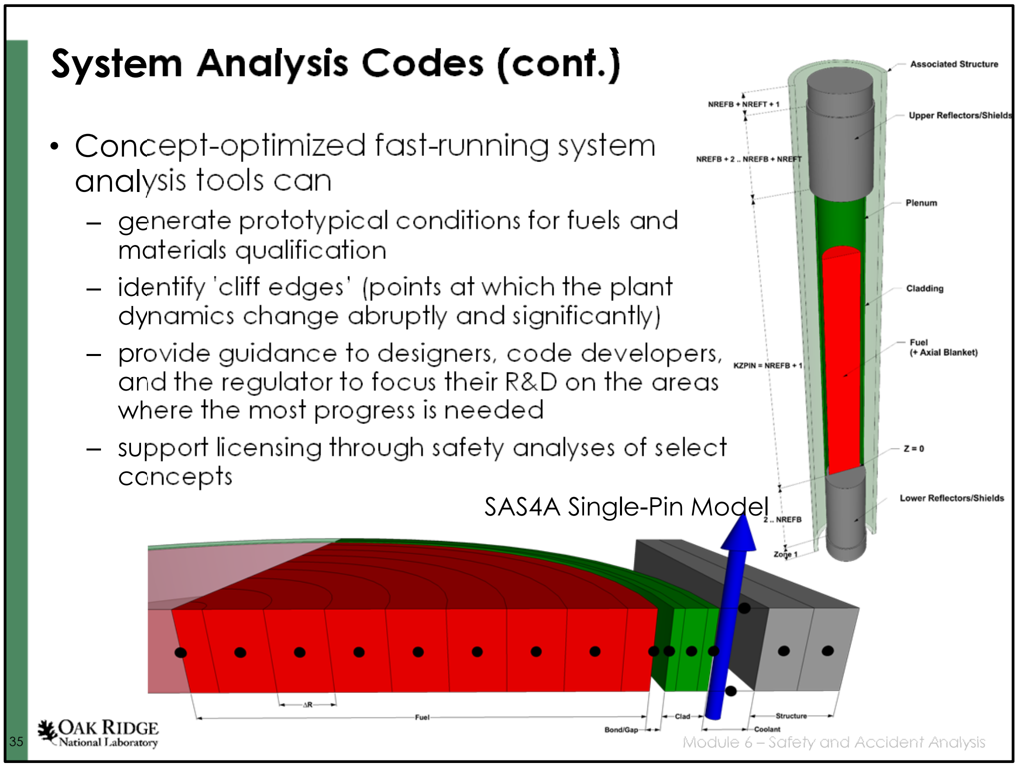
Code in red fonts are modern adaptations, developed under DOE-NE's NEAMS program.

System Analysis Codes

- Traditionally system analysis codes have underpinned most advanced reactor simulation (and licensing) efforts
  - SAS4A/SASSYS-1, CATHARE, MARS-LMR, GRIF, FR-Sdaso
  - Modified versions of LWR systems analysis codes (RELAP, TRACE)
- In a systems analysis, all major physics of the entire plant and integral effects are captured but with some uncertainties
  - Geometry is only coarsely modeled (single-channel or sub-channel approaches for core, 1-D representation of pipes, 0-D representation of volumes)
  - Reliance on correlations with validity for a limited range of conditions
  - Point-kinetics to capture variations in core power level
  - Empirical component models (pumps, valves, heat exchangers, BOP)
- Additional models for fuel behavior (steady-state and transient), fuel/cladding failure models, molten fuel/cladding motion to assess accident progression…

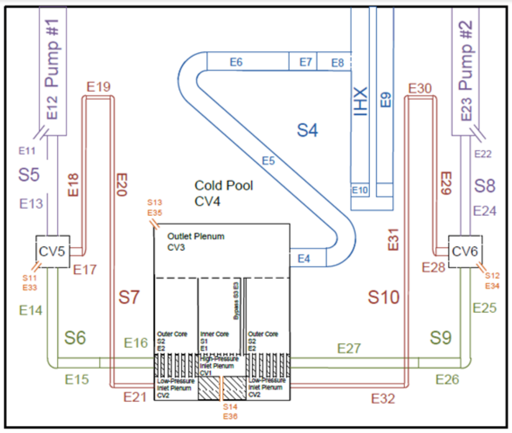
They run fast and can be used on desktop servers and workstations.

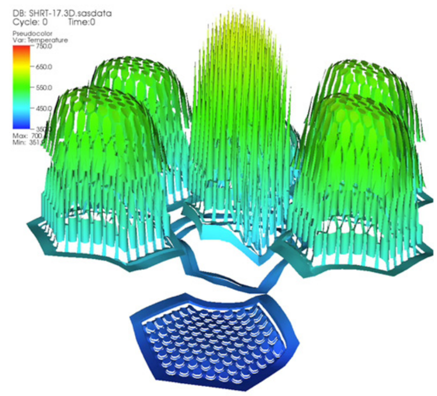


Systems simulations require "concept-optimized" "fast-running" computer codes to increase TRL of advanced reactor designs
High-fidelity T/H (and multi-physics) solutions are affordable only design verification of high-TRL concepts or component design (fuel assemblies, heat exchangers, DHRS, plena/orificing)

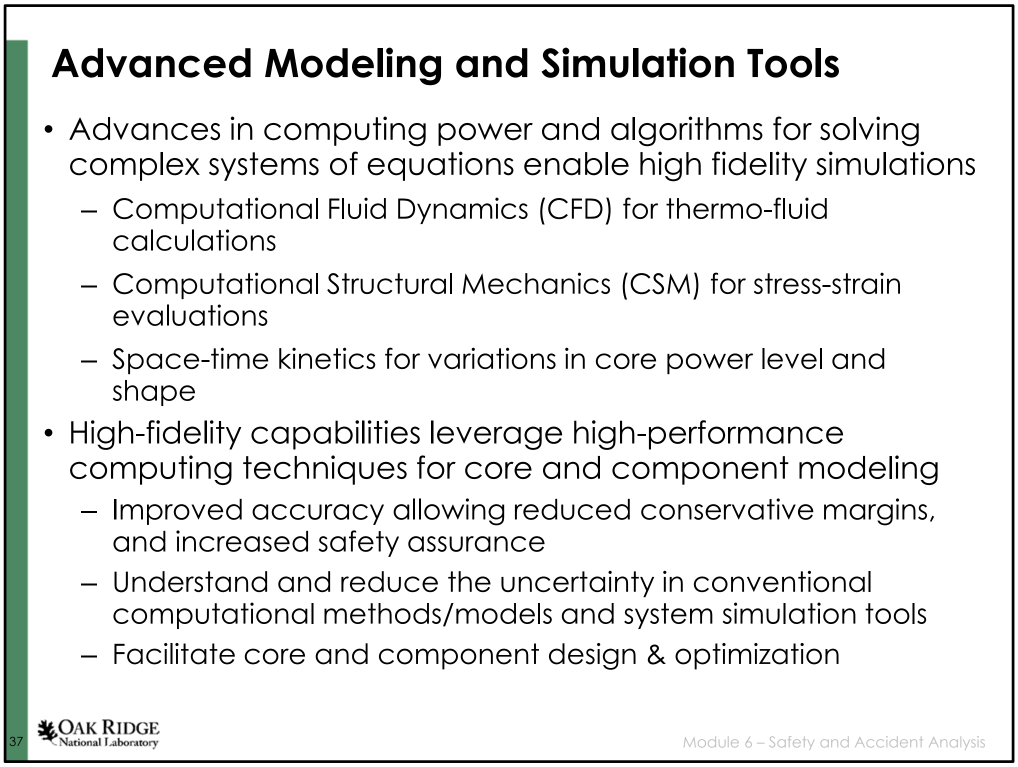
System Analysis Codes Validation Example: EBR-II Benchmark

- Analyses of a protected loss-of-flow and unprotected station blackout tests from full power
  - 4 year IAEA-coordinated research project with 19 participating organizations from 11 countries





SAS4A/SASSYS-1 Model of EBR-II

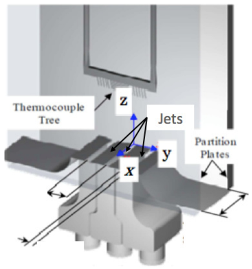


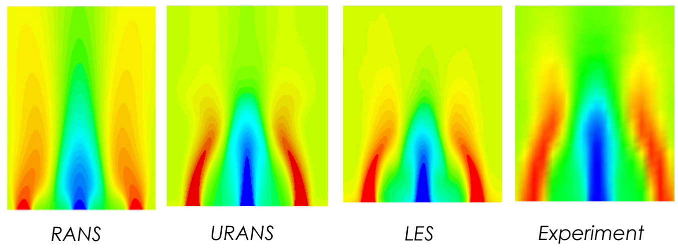
For design optimizations and sensitivity studies, system codes coupled with appropriate subgrid physics or higher-fidelity tools may also provide the information needed for specific phenomena of interest with good accuracy at a reasonable computational cost.

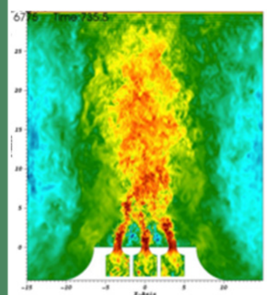
What level of fidelity is needed?

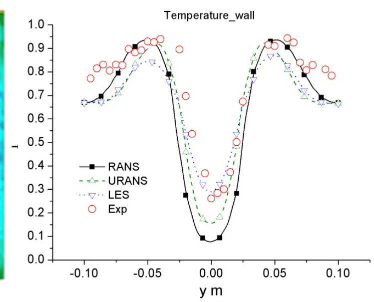
- For bounding case studies in which peak temperatures and profiles are estimated, existing system codes are often sufficient
  - While the results should be considered approximate, they should still represent the major physics and integral effects
  - SFR examples include average and maximum core fuel and coolant temperatures, sodium boiling and multi-phase heat transfer, as-irradiated fuel performance, reactivity feedbacks that contribute to inherent safety
- Higher fidelity is needed if system codes/models are limited in providing information that is known to have a significant effect on plant safety or performance or include a large degree of uncertainty
  - Modeling the mixing and thermal-stratification in large volumes (plena)
  - Thermal-striping of jets at different temperatures leading to thermal fatigue induced failures in upper core structures
  - Stresses on reactor vessel due to sharp thermal and temporal gradients
  - Modeling the influence of detailed geometric design features (e.g., orificing to control core flow distributions, wire-wrap or grid-spacer design on core pressure drop, small bypass flow paths that separate hot and cold pools)

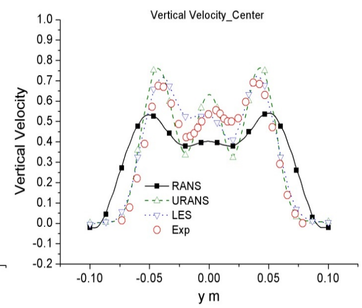
High-fidelity Example: Thermal striping



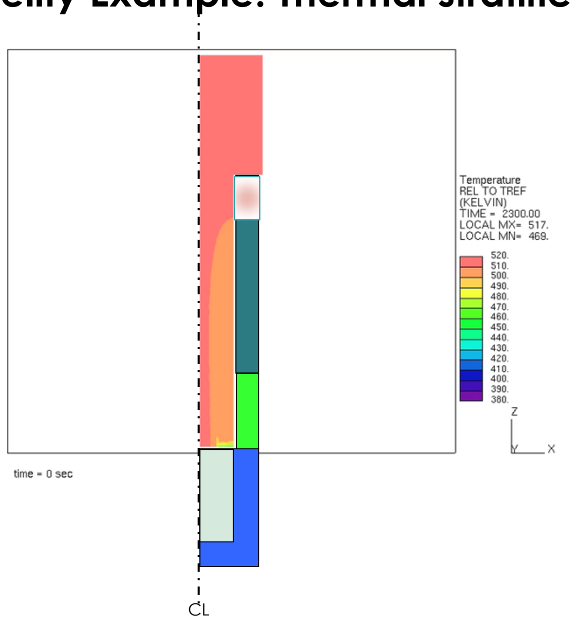








High-fidelity Example: Thermal stratification



Code Validation Basis

- Integral experiments provide average parameters for validation of system codes
  - Monju, Phenix, EBR-II, FFTF and CEFR benchmarks (IAEA-CRPs)
  - Transient fuel failure tests at TREAT, CABRI, EAGLE
- Higher resolution separate effect experiments designed to capture the phenomena that require higher fidelity
  - Natural-convection passive decay heat removal tests at NSTF (ANL), Stella (KAERI) and AtheNa (JAEA)
- Ideally, high-tech instrumentation can be added to integral facilities to capture detailed flow characteristics for multi-physics phenomena
  - Thermal-striping (JAEA) and thermal-stratification tests (CEA)
  - Subassembly flow and inter-assembly heat removal tests at PLANDL facility (JAEA)

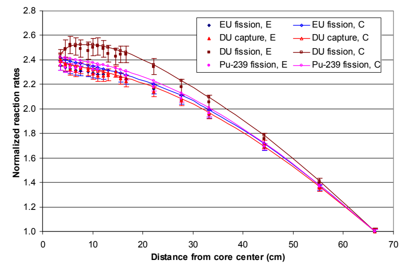
CABRI and EAGLE tests are with oxide fuel forms.
Inter-assembly tests are for teat removal in the space between hex-cans

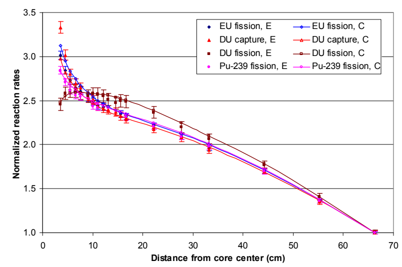
Backup Material:

Fast Reactor Analysis Codes and Methods

MC^2-3: Multigroup XS Generation

- ETOE-2: generates MC^2-3 libraries
- Self-shielding resolved resonances using pointwise cross sections
- Analytic Doppler broadening for temperature change
- Anisotropic inelastic scattering and incident neutron energy dependent fission spectrum
- Ultra fine group 1D or 2D whole-core neutron transport calculation for region dependent cross section generation
- Neutron and gamma libraries with ENDF/B-VII data
- Significant verification and validation efforts using many fast reactor benchmarks and experiments



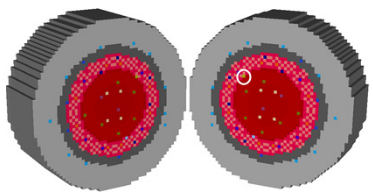


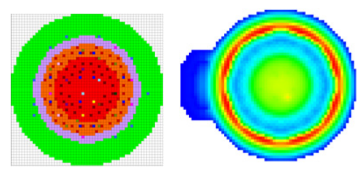
Cross section generation, not self-shielding factor method (lots of iron resonances in key E range)
Resonance self-shielding and Doppler broadening for the specific composition at fine energy group structure
Include spectrum slide as backup
Lots of fast reactor critical for validation of XS and flux transport codes

DIF3D: Neutronics Solver

Steady state neutron transport equation solvers

- DIF3D-FD
  - Finite difference and diffusion theory
  - Integrates diffusion equation over finite volume
  - Eliminates currents with relationships between cell-centered fluxes)
- DIF3D-Nodal
  - Transverse integrated nodal method
  - Diffusion option in Hex or Cartesian geometry
  - Transport (SP_n) option for Cartesian geometry
- DIF3D-VARIANT
  - Variational nodal method solves even parity transport equation
  - Diffusion or transport in Cartesian, Triangular, Hexagonal geometries
  - Flux expanded in spherical harmonics





Diffusion, transport options
FD and nodal options
Use lower order for design option studies, higher order for detailed computation (e.g., reactivity coefficients)

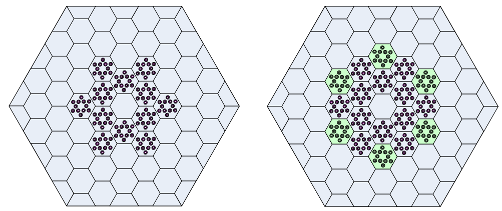
PROTEUS: Neutronics Solver

High-fidelity transport solvers and cross section API

- PROTEUS-Nodal
  - Finite element based nodal transport solver
- Proteus-SN: 2nd order discrete ordinates formulation of the even-parity transport equation
  - Massively parallel solver for fully unstructured finite element mesh
  - Demonstrated to handle >10^{12} degrees of freedom
  - Includes an adiabatic quasi-static kinetics formulation
- PROTEUS-MOC: Method of Characteristics (MOC) solver for unstructured finite element mesh
  - 3D solver practical for small problems due to high memory needs
  - MOCEX that combines 2D MOC method with discontinuous Galerkin finite-element method in axial direction for axially-extruded geometries
- Cross-section API: Generates self-shielded multi-group cross sections on-the-fly
  - Accounts for heterogeneity in geometry, temperature and composition
  - Developed as a functional module and can be easily adapted to other transport codes with fixed source solver

REBUS: Fuel Cycle Analysis/Depletion

- Critical reactors or fixed source problems (ADS)
- Equilibrium cycle analysis of a reactor operating under periodically repeating fuel management
- Non-equilibrium cycle analysis for explicit cycle-by-cycle operation under a specified fuel management program
- Depletion modeling
  - Flexible burn chains
  - Flexible in-core management
- External Cycle modeling
  - Discharge/reprocessing
  - Refabrication/External feed

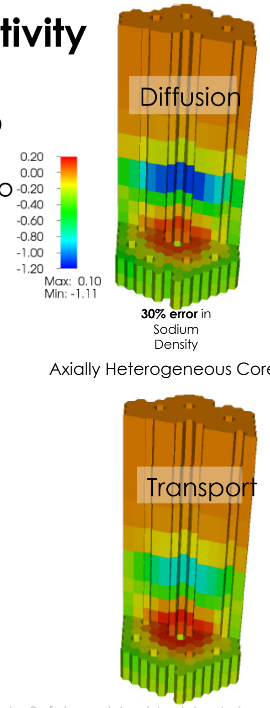


- Search options
  - Fuel enrichment to achieve desired k_{eff} at specified time
  - Burn cycle time to achieve specified discharge burnup or k_{eff}
  - Poison density to achieve k_{eff}
- Validated against EBR-II operational data

Equilibrium cycle with batch averaged compositions
Works well for non-shuffled fast spectrum reactors (minimal local flux perturbations)
Validated against EBR-II DA

VARI3D/PERSENT: Perturbation/Sensitivity

- **VARI3D**
  - Uses finite difference diffusion theory option of DF3D (DIF3D-FD)
  - Calculates effects on reactivity/reacton rates due to changes in material cross sections
  - Reactivity coefficients
  - Dynamics parameters
- **PERSENT** (PERturbation and SENsitivity using Transport)
  - Transport (DIF3D-VARIANT) based perturbation and sensitivity analyses
  - Modern Fortran coding and incorporated basic object oriented design
  - Beta version: common perturbation theory options and beta/lambda sensitivity calculations
  - One of very few 3D transport P/S codes



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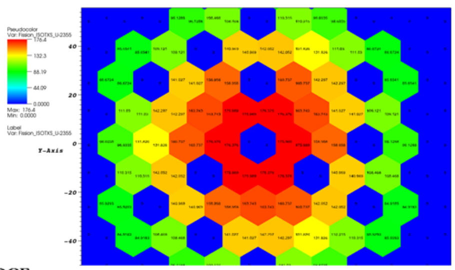
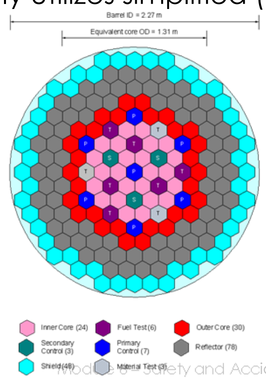
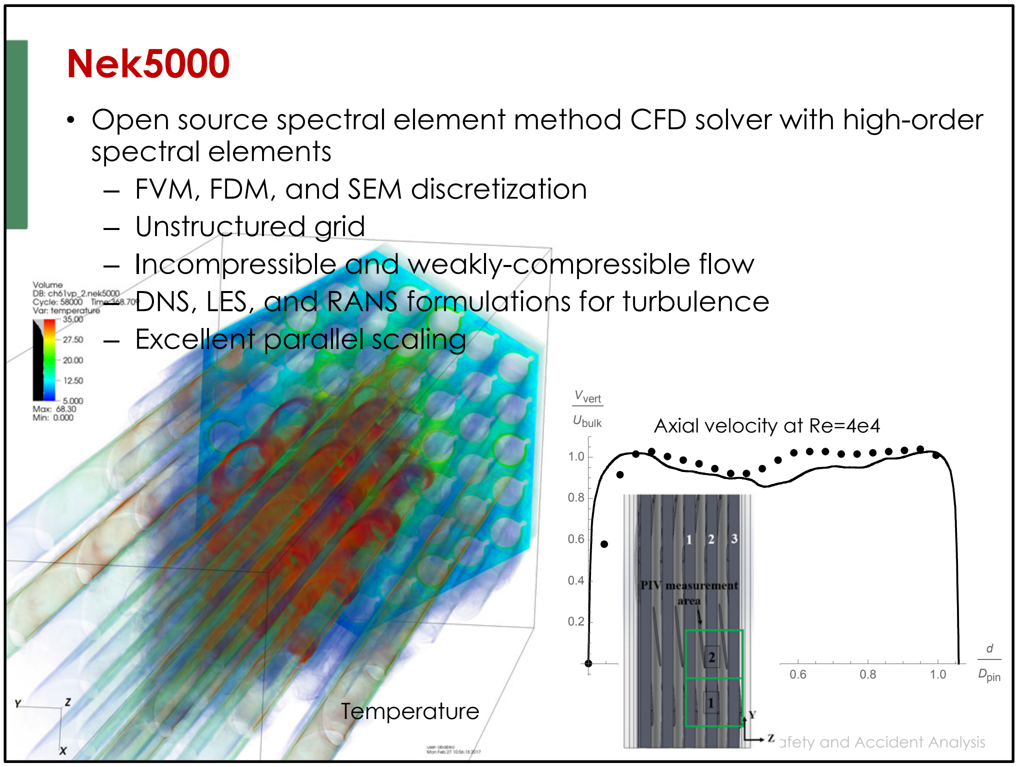
Perturbation theory for reactivity feedbacks (from small changes)
First order or exact for larger (SVW) effects
Modern 3D transport code (nodal) now in common use

SE2-ANL

- ANL version of SuperEnergy-2 code for multi-assembly, sub-channel analysis of wire-wrapped, ducted SFR rod bundles
- Performs orifice zone optimization analyses
- Calculates core-wide temperature profiles
  - Average and two-sigma coolant, clad, and fuel temperatures
- Hot spot analysis, fuel element temperature calculation, allocation of coolant flow
- Subchannel model within each assembly utilizes simplified (porous body) energy mixing model



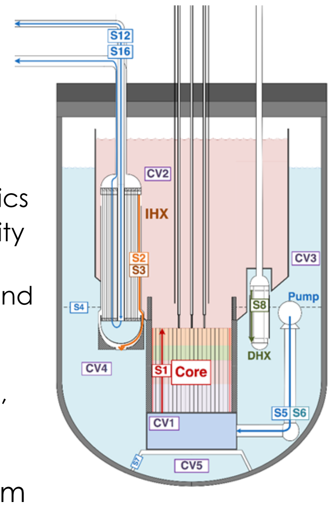




DNS – direct numerical simulation
LES – large eddy simulation
RANS – Reynolds-averaged Navier Stokes

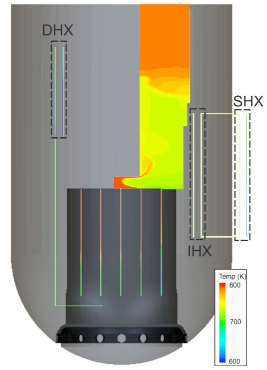
SAS4A/SASSYS-1

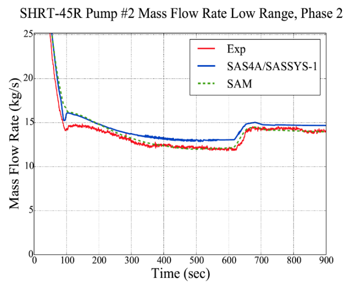
- Originally developed to support CRBR licensing
  - SASSYS-1: DBA/BDBA safety margin assessments
  - SAS4A: Fuel failure consequence assessments
  - Expanded to support metal fuel and passive safety analyses during IFR program
- Modeling features:
  - Single and multi-pin subassembly thermal hydraulics
  - Single and two-phase sodium coolant dynamics
  - Reactor point and spatial kinetics with reactivity feedback
  - Primary/intermediate heat transport systems and components (pumps, pipes, plena, HX, etc.)
  - Decay heat removal systems
  - Steam power cycle with components (turbine, condenser, pumps, etc.)
  - Reactor and plant control systems
- Validated by applications to testing data from EBR-II, FFTF, TREAT, Phenix, and Monju

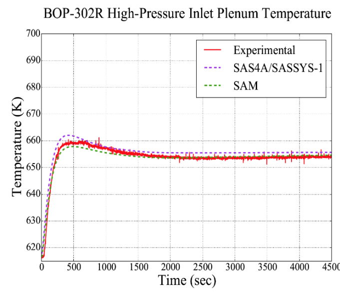


SAM

- System Analysis Module
- MOOSE-based transient analysis capability with a robust high-order FEM model of single-phase fluid flow and heat transfer
- Flexible modeling using single- or multi-channel representation of fuel assemblies with automatically generated core lattice and assembly structures



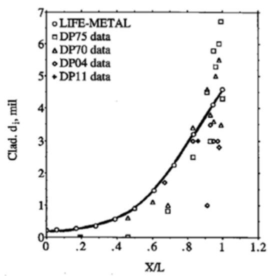


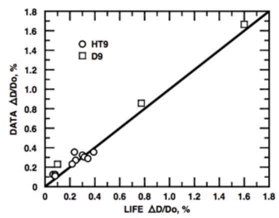


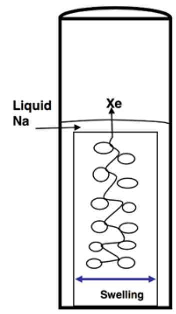
SAM EBR-II Core and Primary Loop Model

LIFE-METAL

- One-dimensional, plane strain analysis of thermal and mechanical behavior of cylindrical fuel elements
- Analytic property correlations used as available
- Assess cladding strain from swelling, creep, and thermal expansion
- Determines stress histories from fuel-cladding interaction and fission gas pressure
- Calculates cladding damage and element lifetime
- Calibrated to thermal/structural benchmark problems with closed analytic solutions

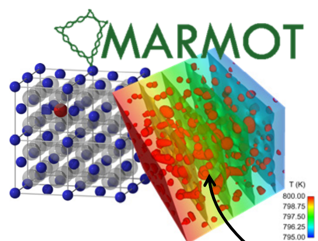


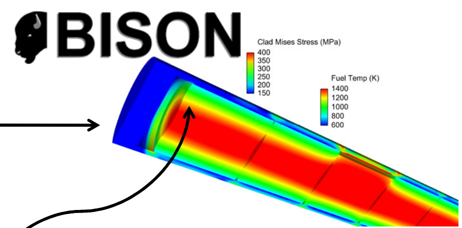




BISON/MARMOT

- The MOOSE-based Bison-Marmot codes provide an advanced multiphysics multiscale fuel performance capability





Atomistic/Mesoscale Material Model Development

- Predicts microstructure evolution in fuel and cladding
- Used with atomistic methods to develop multiscale materials models

Advanced 3D Fuel Performance Code

- Models LWR, TRISO and metal fuels in 1D, 2D and 3D
- Steady and transient reactor operations

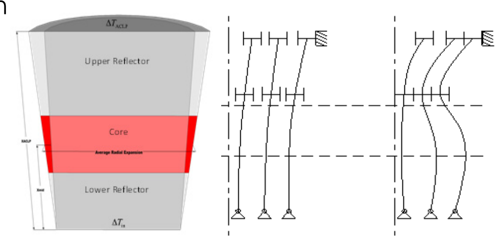
MOOSE

Multiphysics Object-Oriented Simulation Environment

- Simulation framework allowing rapid development of FEM-based applications

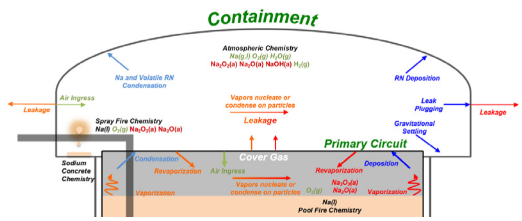
Sodium-Water Interactions and Structural Response

- **SWAAM-II**
  - Assesses pressure transients in secondary system produced by energetic sodium-water chemical reaction
  - Solves coupled phenomena ranging from thermochemical dynamics to propagation of waves through piping system to system rupture
  - Developed to support CRBR licensing, validated against LLTR tests
- **NUBOW-3D**
  - Developed to support design of core restraint systems
  - Predicts transverse displacement of beam elements in 3D core model
  - Includes treatment of inelastic effects of irradiation creep and swelling and duct-to-duct contact, calculates reactivity change due to deformation



Containment Response and Radionuclide Transport

- **MELCOR**
  - Integrated system model tool primarily used for LWR severe accident analyses
  - Robust RN transport model treats transport and dispersion within and release from containment
  - Currently being upgraded to include SFR-specific features (sodium databases, legacy CONTAIN-LMR sodium models)
  - Independent NRC safety evaluation tool
- **CONTAIN-LMR**
  - Originally designed to support ex-vessel severe accident phenomena
    - Updated to treat LMR phenomena
  - Sodium-specific models: spray fire, pool fire, sodium-concrete interactions, debris beds, fission product and aerosol transport/dispersion, RN production from sodium-structure/concrete interactions
  - Spray and pool fire models validated against experiments (e.g. ABCOVE)



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Backup Material:

Accident types, event descriptions and phenomena ranking

Accident Types Comparison

Category	PWR	4S	PRISM (metal)	S-PRISM (metal)	S-PRISM (oxide)
Loss of coolant	Primary system LOCA	NA (no primary piping)	NA (no primary piping)	NA (no primary piping	NA (no primary piping)
Loss of flow	Requires emergency system action to prevent core damage plus scram	No core damage if scram occurs. Flow coast-down assured by inertia power to EM pumps. Automatic natural circulation after that.	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)
Loss of heat sink	Requires emergency action to prevent core damage plus scram	No core damage if scram occurs. Inherent change to naturel circulation	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)

Ref: PRISM Preliminary Safety Information, ML082880369 GEFR-00793 – Vol 1, December, 1987; and ML082880397 GEFR-00793 – Vol 4, December, 1987. Ref: PWR: -Pressurized Water Systems, USNRC Technical Training Center, June, 2003. Ref: Toshiba 4S-ML081440765- Toshiba – Submitted Design Description of 4S – 4S Design Description, May 2008.

Accident Types Comparison (cont.)

Category	PWR	4S	PRISM (metal)	S-PRISM (metal)	S-PRISM (oxide)
Reactivity transients	Requires emergency action to prevent core damage plus scram	No core damage if scram occurs. Two independent shutdown systems (central rod and radial reflector)	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)	No core damage if scram occurs (no action other than scram is required)
Transients without scram	Generic issue	Initially, Doppler and expansion limit power. No fuel damage	Initially, Doppler and axial expansion limits power. Slower acting negative feedbacks return core to new steady state power level	Initially, Doppler and axial expansion limits power. Slower acting negative feedbacks return core to new steady state power level	Initially, Doppler and axial expansion limits power. Slower acting negative feedbacks return core to new steady state power level
Station black out	Emergency diesels required	No emergency diesels required. Natural circulation	No emergency diesels required	No emergency diesels required	No emergency diesels required

SFR Event Descriptions

PROTECTED EVENTS

Loss of Flow and/or Loss of Coolant		
Event Description	**Key Systems Involved**	**Relevant Phenomena**

Equipment Failure
• Electrical faults
• Loss of offsite power
• Controller failures
• Mechanical faults
  • Pump mechanical failure
  • Loss of piping integrity
Operator Error
• Turning off pump power
• Opening breakers to power supplies
External Events
• Seismic, fire, flood, tornado, terrorist
	Component or System
	• Primary pump power supplies
	• Shaft/bearing/impeller
	• Off-site power connection
	• Primary piping and vessel system
	• Core and assembly coolant flow channels
	• Fuel cladding
	• Reactor control and protection systems
	• Shutdown heat removal systems
	• Reactor containment
	• EM pump power leads
		Thermal Fluid
		• Single-phase transient sodium flow
		• Thermal inertia
		• Pump-coast down profiles
		• Sodium stratification
		• Transition to natural convection core cooling
		• Decay heat generation
		Reactivity Effects Prior to Scram
		• Mechanical changes in core structure
		• Fuel/coolant/structure temperatures
		Material Behavior
		• Structure behavior at elevated temperatures
		• Cladding integrity margin
		• Leak-before-break behavior of piping
		• Primary coolant boundary integrity margin
		• Containment building integrity margin
		• Thermal chock to structures

protected events are those where there is no failure to scram

SFR Event Descriptions (cont.)

PROTECTED EVENTS

Reactivity Addition		
Event Description	**Key Systems Involved**	**Relevant Phenomena**
Equipment Failure • Uncontrolled control rod motion • Overcooling from pump speed increase • BOP system pressure loss gas bubble entrainment **Operator Error** • Control rod movement error • Coolant pump control error • Actuation of BOP pressure relief valve **External** • Seismic	**Component or System** • Reactor control system and control rod drives • Primary pumps • BOP heat removal systems • Shutdown heat removal • Primary and intermediate cooling systems • Reactor protection systems • BOP control systems • Reactor containment	**Reactivity Effects Prior to Scram** • Reactivity feedback at high power • End-of-life prediction of reactivity feedback • Burnup control swing/control rod worth • Reactivity effects Of gas bubble entrainment • Integrity of fuel with breached cladding • Integrity of fuel with load following

SFR Event Descriptions (cont.)

PROTECTED EVENTS

Loss of Normal Heat Rejection		
Event Description	**Key Systems Involved**	**Relevant Phenomena**
Equipment Failure • Steam generator failure • Intermediate heat transport system failure • Loss of electric grid load • Flow blockage in heat transfer loop **Operator Error** • Stopping intermediate loop flow • Steam generator blowdown • Isolating plant from the grid **External Events** • Seismic, fire, flood, tornado, terrorist	**Component or System** • Secondary sodium pumps • Secondary system piping • Steam generators • Turbine generators • Shutdown heat removal systems • Intermediate heat exchanger • Reactor protection systems • Reactor containment	**Thermal Fluid Effects** • Sodium steam chemical reactor • Pressure-pulse impacts from chemical reaction • Decay heat generation **Material Behavior** • Long-term performance of structures at elevated temperatures

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What is the sodium – CO_2 heat exchanger? There is no mention of CO_2 in PFR, Phenix, Superphenix, BN-600, or BN -350.

SFR Event Descriptions (cont.)

ANTICIPATED TRANSIENTS WITHOUT SCRAM

ATWS		
Event Description	**Key Systems Involved**	**Relevant Phenomena**
Reactivity Control System Failure Following a Class-2 Component Failure: • Electrical faults • Mechanical faults • Loss of piping integrity	**Component or System** • Primary pump power supplies • Pump mechanicals • Primary piping system • Core and assembly coolant flow channels • Core structure • Fuel and subassemblies • Primary coolant system	**Same as for Protected Events Plus:** **Thermal Fluid Effects** • Thermal inertia • Pump-coast-down profiles • Sodium stratification • Margin to boiling at peak temperatures • Core thermal and structural effects • Heat removal path and capacity **Material Behavior** • Long-term performance of structures at elevated temperatures • Fuel cladding integrity at elevated temperatures

SFR Event Descriptions (cont.)

UNPROTECTED EVENTS

Unprotected Loss of Core Cooling		
Event Description	**Key Systems Involved**	**Relevant Phenomena**

Reactor Shutdown System Failure Following a Class-1 Component Failure:
• Electrical faults
• Mechanical faults
• Loss of site power
• Loss of piping integrity
• Internal flow blockage
	Component or System
	• Primary pump power supplies
	• Pump mechanicals
	• Off-site power
	• Primary piping system
	• Core and assembly coolant flow channels
	• Core structure
	• Fuel and subassemblies
	• Primary coolant system
	• Inherent and passive safety systems
	• Flow coast-down extenders
		Same as for Protected Events Plus:
		Thermal Fluid Effects
		• Thermal inertia
		• Pump-coast-down profiles
		• Sodium stratification
		• Margin to boiling at peak temperatures
		• Core thermal and structural effects
		• Heat removal path and capacity
		Reactivity Effects
		• Core reactivity feedback
		  • Fuel motion in intact fuel pins
		  • Core restraint system performance
		• Reactor shutdown mechanism
		Material Behavior
		• Long-term performance of structures at elevated temperatures
		• Fuel cladding integrity at elevated temperatures

SFR Event Descriptions (cont.)

UNPROTECTED EVENTS

Unprotected Reactivity Addition		
Event Description	**Key Systems Involved**	**Relevant Phenomena**
Reactor Shutdown System Failure with • Uncontrolled withdrawal of a single control rod • Overcooling from pump speed increase	**Component or System** • Reactor shutdown system • Control rod drive system • Fuel and subassemblies • Primary pumps • BOP heat rejection system	**Same as for Protected Events Plus:** **Thermal Fluid Effects** • Heat removal path/capacity **Reactivity Effects** • Reactivity feedback at high power • Coolant heating and margin to boiling • Core reactivity feedback • Core thermal and structural effects **Material Behavior** • Fuel cladding structural integrity at elevated temperatures • Cooling systems structural integrity at elevated temperatures • Containment structural integrity

SFR Event Descriptions (cont.)

UNPROTECTED EVENTS

Unprotected Loss of Normal Heat Rejection		
Event Description	**Key Systems Involved**	**Relevant Phenomena**

Reactor Shutdown System Failure with
• Steam generator failure
• Intermediate heat transport failure
• Decay heat removal system failure
	Component or System
	• Secondary sodium pumps
	• Secondary system piping and IHX
	• Steam generators
	• Decay heat removal systems
	• Sodium-CO_2 heat exchanger
		Same as for Protected Events Plus:
		Thermal Fluid Effects
		• Thermal inertia
		  • Core thermal/structural effects
		Reactivity Effects
		• Core reactivity feedback
		  • Fuel motion in intact fuel pins
		  • Core restraint system performance
		Material Behavior
		• Long-term performance of structures at elevated temperatures
		• Fuel cladding structural integrity at elevated temperatures
		• Containment structure integrity

SFR Event Descriptions (cont.)

SEVERE ACCIDENTS—SUBSTANTIAL CORE DAMAGE

Event Description	Key Systems Involved	Relevant Phenomena

Severe Loss of Core Cooling Event
 	
Severe Reactivity Addition Event
 	
Severe Loss of Heat Rejection Capability
	Component or System
	• Core fuel and assemblies
	• Core grid and restraint structure
	• Primary coolant system
	• Containment building
	• Support structure
	• Seismic isolation
		Same as for Above Plus:
		Fuel and Core Behavior
		• Sodium voiding effects
		  • Temporal and spatial incoherence
		• Fuel pin failure
		• Fuel dispersal, relocation, and coolability
		• Recriticality
		  • Potential for energetic events
		• Primary vessel thermal and structural integrity
		• Radiation release and transport

Evaluation of Phenomena

- Plant responses to accident initiators are calculated using analytical models for which understanding of relevant phenomena is essential. The following viewgraphs indicate the important phenomena as evaluated by a U.S. DOE Technology Gap Team

Ref: J. LaChance, et. al., Sodium Fast Reactor Safety and Licensing Research Plan – Volume II, (Advanced Sodium Fast Reactor Accident Initiators/Sequences Technology Gap Analysis – Fuel Cycle Research and Development (FCRD-REAC-2010-000126)),SAND2012-4259, May 2012.

Evaluation of Phenomena

DBAs and BDBAs not leading to fuel failure

Reactivity Feedbacks in Transients (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
Mechanical changes in core structure	Expansion of core grid structure	High	High	High
	Expansion of control rod drives	High	High	High
	Mechanical changes in core structure over life (swelling, etc.)	High	High	High
	Bowing of fuel assemblies and blanket	High	High	High
	Core restraint system performance	High	High	High
	Axial thermal expansion of fuel and cladding			
	Metal	High	High	High
	Oxide	High	Medium	High

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Module 6 – Safety and Accident Analysis

The importance of phenomena and the state of knowledge are assessed by PIRT (phenomena importance and ranking tables), which are important in identifying areas where phenomena needs to be better understood by appropriate research and development. In the following slides, the importance, modeling, and state of experimental data are ranked as high, medium, or low. It is apparent that most issues are well known.

Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Reactivity Feedbacks in Transients (HIGH IMPORTANCE) (continued)				
Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge Adequacy	
			Modeling	Experimental data
Intact fuel and fuel changes	Fission product impacts on fuel structure and properties	High	High	High
	Doppler feedback as a function of fuel composition	High	High	High
	Cross section information for minor actinides	Low	Medium	Low
	End-of-life power distribution and control rod position	High	High	High
	End-of-life fuel composition	High	High	High
	End-of-life prediction of reactivity feedback	High	Medium	Medium
	Burnup control swing	High	Medium	Medium
	Control rod worth	High	High	High
	Reactivity feedback at high temperature	High	High	High
	Axial growth of fuel with irradiation			
	Metal	High	High	High
	Oxide	Low	High	High

Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Reactivity Feedbacks in Transients (HIGH IMPORTANCE) (continued)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
Sodium density effects	Sodium temperature coefficient of reactivity	High	High	High
	Sodium void coefficients	High	High	High

Margin to Fuel Cladding Failure (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
Fuel cladding failure	Fuel cladding failure mechanisms			
	Metal	High	High	High
	Oxide	High	High	High
	Metal fuel cladding failure time and location	High	High	High
	Oxide fuel cladding failure time and location	High	Medium	Medium

Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Fluid Flow and Heat Transfer (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
Steady state and transient-forced convection	Single phase sodium-forced flow	High	High	High
	Sodium convective heat transfer	High	High	High
	Fuel pin heat removal	High	High	High
Transition to natural convective boiling	Single phase transient sodium flow	High	High	High
	Pump-coast down profiles	High	High	High
	Sodium stratification	High	Medium	High
	Core flow redistribution in transition	High	High	High
	Coolant heat up profile and margin to boiling	High	High	High
Thermal response of structures	Thermal shock to structures	High	High	High
	Thermal striping	High	High	Medium
	Structure heat conduction	High	High	High

Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Fluid Flow and Heat Transfer (HIGH IMPORTANCE) (continued)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
Decay heat rejection	Radiation heat transfer from vessels	High	High	Medium
	Convective heat transfer	High	High	High
	Cooling systems structural integrity over time	High	High	High
	Natural circulation heat removal	High	High	High
Power conversion	Steam-sodium reactions	High	High	High
	Pressure pulse migration	High	High	High
	CO_2-sodium chemical interaction (supercritical CO_2 cycle)	High	Low	Low
	High pressure CO_2 release and impact (advanced cycle)	High	Low	Low

Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Fuel Transient Behavior (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Evolution of fuel and cladding over life	High	High	High
	Cladding structural integrity (margin)	High	High	High
	Length effects on fuel performance during transients			
	Metal	Medium	High	Low
	Oxide	Medium	Medium	Medium
	Fuel pin behavior with breached cladding			
	Metal	Low	High	High
	Oxide	Medium	High	High
	High minor actinide content fuel performance	High	Low	Low
	Source term is different			
	Physics are different			
	Chemistry is different			

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Evaluation of Phenomena (cont.)

DBAs and BDBAs not leading to fuel failure

Material Interactions and Chemistry (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Sodium vapor condensation and plate out (system degradation	High	High	High
	Structural material corrosion	Low	High	High
	Sodium purity control	High	High	High

Structural Mechanics (HIGH IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Seismic response of reactor core and coolant system	High	High	High
	Seismic response of containment	High	High	High

Evaluation of Phenomena (cont.)

DBAs and Beyond DBA Phenomenology with Fuel Pin Failures				
Localized Core Damage (LOW IMPORTANCE)				
Modeling issue	**Underlying phenomenon**	**Importance to safety case**	**Knowledge adequacy**	
			Modeling	**Experimental data**
	Low flow blockage			
	Fission product transport and delayed neutron detection	High	High	High
	Extent of fuel melting within affected subassemblies	High	High	High
	Propagation of fuel melting across subassemblies			
	Metal	High	High	High
	Oxide	High	High	High

Evaluation of Phenomena (cont.)

DBAs and Beyond DBA Phenomenology with Fuel Pin Failures

Severe Core Damage (MEDIUM IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Sodium voiding effects			
	Temporal and spatial incoherence	High	High	High
	Bubble growth at boiling temperature	High	High	High
	Thermal-hydraulic effects	High	High	High
	Fuel failure			
	Failure mode location			
	Metal	High	High	High
	Oxide	High	High	High
	Fuel motion, dispersal, morphology			
	Metal	High	Medium	Medium
	Oxide (including fuel-coolant-interaction)	High	Medium	High

Evaluation of Phenomena (cont.)

DBAs and Beyond DBA Phenomenology with Fuel Pin Failures

Severe Core Damage (MEDIUM IMPORTANCE) (continued)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Pre-existing radionuclide distribution in the pin (ST)			
	Metal (including bond)	High	High	High
	Oxide	High	High	High
	Coolability of rubble/debris bed			
	Metal	High	High	High
	Oxide	High	High	High
	Pressure sources/primary system loads (ST)	High	High	High
	Primary system response to loads (ST)	High	High	High

Evaluation of Phenomena (cont.)

DBAs and Beyond DBA Phenomenology with Fuel Pin Failures

Challenges to Containment (MEDIUM IMPORTANCE)

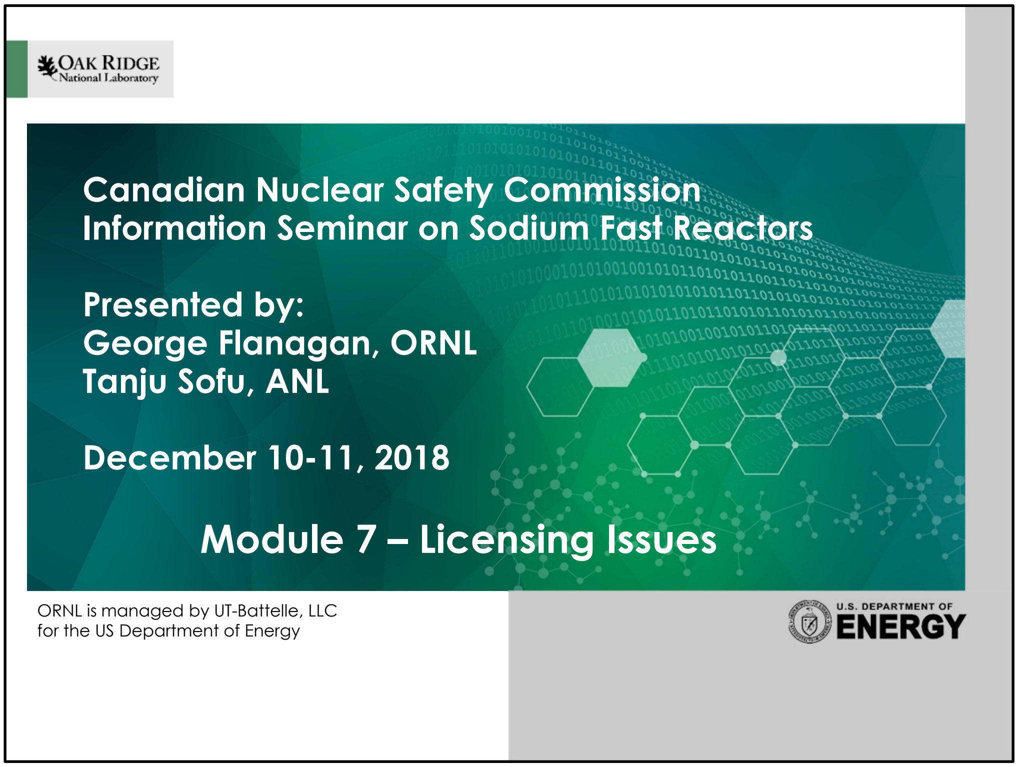
Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Pressure sources/containment loads	High	High	High
	Containment response to loads	High	High	High
	Sodium-concrete interactions (sodium group)			
	Sodium fire with contaminated sodium (ST) (sodium group)			
	Ultimate heat removal path/capacity	High	High	High

Evaluation of Phenomena (cont.)

DBAs and Beyond DBA Phenomenology with Fuel Pin Failures

Hypothetical Core Disruptive Accidents (LOW IMPORTANCE)

Modeling issue	Underlying phenomenon	Importance to safety case	Knowledge adequacy	
			Modeling	Experimental data
	Re-criticality	High	High	High
	Energetic dispersal/reactivity shutdown			
	Sodium voiding timing and coherence	High	Medium	Medium
	Fuel Vaporization	High	Medium	Medium
	Mechanical energy generation	High	Medium	Medium
	Response of primary system to CDA loads	High	Medium	Medium
	Response of containment to CDA loads	High	Medium	Medium
	Ultimate shutdown mechanisms	High	Medium	Medium
	Ultimate heat removal path/capacity	High	Medium	Medium
	Hydrodynamics	High	Medium	Medium



The following set of slide present in detail important SFR safety analysis and licensing issues that are likely to arise in licensing of SFRs.

SFR Safety Analysis and Licensing Concerns

- Modularity (one control room, one steam generator module)
- Opaque coolant
- Complex refueling/spent fuel storage process
- Sodium-bonded pins (metal fuel)—eutectic formation at boundary
- Oxide fuel will react with coolant
- No core damage if reactor scram occurs for TOP, LOC, LOHS, LOF accidents
- Beyond Design Basis Accidents (BDBA) generally do not result in core damage (metal core) even if scram does not occur (for 4S and PRISM)
- Potential for core disruption if very severe event occurs (PRA estimates = $<10^{-6}$/year)

Opaque coolant impacts inservice inspection licensing requirement.
Refueling complicated due to activity taking place under vessel head, no direct visual observation, complex refueling machine. Spent fuel storage will depend on whether the fuel is stored in Na first then water or dry. Also temporary storage in reactor pool requires a second transfer to another location.

Safety Analysis and Licensing Issues

- Thermal Fluid Issues
  - Flow regime transitions, transport properties, channel flow distribution, and sodium boiling
  - Coolant structure interactions
  - Natural convection

Safety Analysis and Licensing Issues (continued)

- Fuel Safety
  - Fuel pin performance under steady state conditions
  - Margin to fuel melting
  - Margin to deterministic fuel failure (under slow power transients)
  - New fuel pin designs and materials
  - Use of minor actinides
  - Target or blanket (for breeding or transmutation) pin performance under steady-state conditions

Safety Analysis and Licensing Issues (continued)

- Severe Accidents
  - Containment of reactor fuel
  - Fuel pin failure
  - Operation of fuel pins with breached cladding
- Sodium Event Issues
  - Sodium spray fires
  - Sodium pool fires
  - Sodium fire aerosol behavior
  - Sodium/containment interactions
  - Sodium/water reactions

Summary of Major Design Issues Related to Licensing of LMRs

- Limited experimental data on prototypical length fuel (for metal fuel)
- Fission product behavior, variation of physical and mechanical properties with burnup
- Fuel/clad eutectic formation and fuel relocation behavior during BDBA accidents
- Positive coolant density feedback with decreasing coolant density—positive void effect (positive for PRISM, negative for 4S)
- Sodium fires and their suppression
- Seismic isolation design of structures
- Severe accident progression—source term, containment behavior, off-site planning
- Residual heat removal systems (passive and active)

6 OAK RIDGE National Laboratory Module 7 – Licensing Issues

Most experience with metal fuels is from EBR-II operation. These fuel pins are significantly shorter that those envisioned for future SFRs.

Detailed Explanation of Safety Analysis and Licensing Issues

- The following are detailed explanations of previous slides showing major topics

Thermal Fluid Issues

1. **Flow Regime Transitions, Transport Properties, Channel Flow Distribution, Sodium Boiling**
   – Affects sodium flow rate and temperature fields
   – Affects possibility of reaching sodium boiling and potential reactivity increase
   – Influences potential for gas entrainment
   – Computational tools developed and experiments performed
   – Future development involves more sophisticated models such as 3D computational fluid dynamics (CFD)

– Thermal hydraulics of the sodium coolant in SFR systems determines the sodium temperature fields inside the different components of different scales (subchannel, subassembly, core, circuits) and in all conditions (nominal, incidental and accidental). Through sodium temperature, the TH conditions influence the core dynamic system.

– In case of loss of flow (potential initiation of a core disruptive accident), the conditions of onset of sodium boiling may be reached (TNa~980° C, 0.2 MPa) and lead to positive reactivity feedback and subsequent power excursion.

– Core thermal hydraulics may involve gas entrainment (for instance, from free surface) with associated risk of gas bubble passage into the core.

– Computational tools have been continuously developed since the start of SFR studies and many experiments have also been performed (CEA, JAEA, FZK,…).

– Future development is now orientated towards more sophisticated approaches (3D-CFD) that need more detailed data for improved and accurate modeling.

Thermal Fluid Issues (continued)

2. **Coolant—Structure Interaction**
   - Fluid structure interaction may influence mechanical loading and behavior of the core structures
   - Due to sodium high temperature in the upper part of the core with possible high-thermal gradients and turbulent flow mixing at high temperature, repeated temperature oscillations may induce thermomechnical stresses on the structures and ultimately cause thermal fatigue failure of the material (thermal striping)
   - Moreover, the presence of a spillway may also induce large vibrations on the thermal baffles as shown during the Super-Phenix commissioning tests
   - Such phenomena occurrence is dependent on core design
   - Prediction of these effects requests detailed evaluation of temperature fluctuations that can result from CFD approaches

Thermal Fluid Issues (continued)

3. Natural Convection

- A main aspect for improved safety of SFR
- EBR-II ran tests from full power to natural convection successfully
- Analyses for Phenix and Super-Phenix (pool types) show successful natural convection
- Effects in large cores need to be demonstrated
- An effective mechanism for cooling by natural convection internal to fuel assemblies at very low power

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- The possibility of generalized natural convection for passive system for residual power removal is one of the main issue for improved safety of SFR.
- Full scale natural convection tests starting from forced convection at full power and various power levels to natural convection heat removal were conducted at EBR-II that provided measured data and demonstrated that peak coolant and fuel temperatures remain low and do not challenge safety limits. Natural convection cooling tests were also performed as part of the passive safety program to demonstrate inherent core cooling capability from refuelling conditions where there is no thermal driving head and from steady state operating conditions. Coolant and fuel temperatures were very low in both cases.
- In reactors such as Phenix and Superphenix (pool type), the possibility of natural convection has been verified for some circuits but no experimental evidence was available for the whole system. This implied that only computational-based demonstration could be obtained.
- The possibility and reliability of generalized natural convection (including the inter-subassemblies sodium flow) has to be evaluated for large cores (and also for the

loop type concept).

– The capability of experimental overall check has to be examined in the future
– For subassemblies with a quite low power (breeder, fuel subassemblies in internal storage positions,..), natural convection inside a subassembly is the way to remove power in case of loss of cooling inside the subassembly; its efficiency depends on the design, on subassembly power and on the power distribution among the pins. Tests have been performed for a limited number of geometries and sophisticated approaches (3D-CFD) and accurate modelling for new fuel assemblies design are required.

Fuel Safety

1. **Fuel Pin Performance Under Steady State Conditions (Irradiation)**
   - Potential for degraded performance after long irradiation
   - Burnup is a major parameter influencing performance under steady state conditions
   - Knowledge base includes irradiation of numerous pins in Phenix, Super-Phenix, JOYO, MONJU, EBR-II and FFTF. Computational tools have been developed
   - There is a lack of data and knowledge concerning the behavior of low-smear density (annular) fuel beyond 6.4 at. % burnup level (well below the target values for new SFR concepts, ~15 at. %)

- Ref: See specific references Phenix, Superphenix, Joyo, Monju, and FFTF in other sections

- During in reactor operation, fuel pins are submitted to thermo-mechanical and physical-chemical phenomena that lead to structural and mechanical changes in both fuel and cladding materials with potential to jeopardize the ability of the fuel pins to withstand DBA.

- The burnup increase during in-reactor stay is the major parameter influencing the pin state under steady-state as it leads to the following effects: reduction of the fuel thermal conductivity and of fuel melting temperature, increase of fission gas retention and release rate, evolution of fuel micro-structure (cracking, restructuring, central hole evolution, linked to high operating fuel temperature), evolution of pellet-clad gap composition (FP compounds) and thickness, internal clad corrosion, clad embrittlement, and swelling.

– Part of knowledge has been gained from the irradiation of numerous pins in the framework of the past R&D on SFR with mixed oxide fuel and of reactor operation of Phenix, Superphenix , JOYO, Monju, FFTF.. and computational tools have been developed on this basis (ie GERMINAL code at CEA, SAS-4A, other…); in some conditions, the use of empirical laws prevents from a reliable prediction capability.

Fuel Safety (continued)

2. Margin-to-Fuel Melting

- Important to evaluate conditions (power level) at which fuel melting could be initiated
- The "power to melt" depends on fuel thermal conductivity, pellet-clad gap conductance, both affected by burnup level
- Influence of pellet design could induce lower thermal conductivity and lower power required to reach melting
- Uncertainty still exists on fuel creep at high temperature, on fission gas-induced fuel swelling, and on impact of higher burnups
- Metallic fuel is characterized by much lower steady-state operating temperature due to its higher thermal conductivity

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- SFR mixed oxide fuel is characterized by a high operating temperature linked to high linear power (range of 400-500 W/cm) and low thermal conductivity. It is thus important to evaluate the power level at which fuel melting could be initiated under transients ("power to melt") as it represents a first step of pin degradation and may affect the subsequent pin mechanical behaviour due to the formation a molten fuel cavity under high pressure (10% fuel volume increase due to melting, fission gases).

- Influence of pellet design (solid, annular, high or low smear density) has also been evidenced through the past R&D work ; in particular, with low smear density irradiated fuel (id annular pellet geometry, 6.4 at%) and under slow power transients, the available data (from IRSN CABRI R&D programs) indicate that porosity increase resulting from fission gas induced swelling and high temperature fuel creep into the free volumes (leading to central hole closure), induce lower thermal conductivity and thus lower power to melt than originally expected.

Fuel Safety (continued)

3. Margin for Deterministic Pin Failure (Under Slow Power Transients)

- It is important to determine and predict the power level at which deterministic pin failure may occur during DBA
- The fuel pin thermomechanical behavior depends on fuel and clad materials mechanical properties, temperature, and burnup level
- Under slow power transients, clad mechanical loading is mainly due to fuel thermal expansion (linked to fuel temperature increase), fission gas-induced swelling (effect of burnup), and molten cavity pressure after fuel melting onset (if any)
- With solid pellets (high-smear density) and high burnup level (12 at. %), pin failure occurs close to fuel melting onset leading to a lower margin to failure

– In relation with the fuel and clad evolution during in reactor operation (cf B1), it is important to determine and predict the power level at which deterministic pin failure may occur during DBA such as slow power transients, in comparison to the operating power.

– The past R&D performed within the IRSN CABRI programs has underlined that pellet geometry (solid or annular) and fuel smear density influence the fuel enthalpy level at failure when fuel melting occurs. Higher pin failure enthalpy thresholds with annular fuel may be expected under slow power transients due to internal molten fuel motion and result in high margin to deterministic failure (for annular fuel at 6.4 at%, Pfail/Pnom >3). Uncertainty still exists for oxide fuel at higher burnup level.

Fuel Safety (continued)

4. New Fuel Pin Designs and Materials (Fuel and Cladding)

- New fuel and cladding materials and new pin designs are anticipated within the on-going and future development of the GEN IV SFR concepts that call for improved safety performances together with economics (reliability, availability of the system) and flexible and robust management of the nuclear materials (waste reduction)
- The new fuel concepts will need R&D work in order to check their behavior with regard to safety aspects and to establish database for safety demonstration and analyses

Fuel Safety (continued)

5. Use of Minor Actinides (MA)

- MA burning is considered in the future SFR concept as it contributes to waste reduction
- The use of MA in the fuel affects the core characteristics and key safety parameters such as power density and distribution, sodium void reactivity, decay heat, source term, and thermal conductivity
- It may also affect the fuel microstructure evolution and fuel pin behavior under irradiation and transient conditions
- The expected impact depends on MA content and type
- Limited data is available
- R&D work is needed on this topic

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Variability of MA loading from reprocessed LWR fuel may result in a wide range of data needs and the impact of various MA loads within one core needs to be studied since MA content will vary between batch loadings

- R&D work is needed on this topic

- and new modelling has to be developed in order to check and quantify the impact of MA use with regards to safety aspects, including fuel pin behaviour under irradiation and accidental transients.

Fuel Safety (continued)

6. Target or Blanket Pin Performance Under Steady State Conditions (Irradiation)

- Phenomena affecting target or blanket pins are different from those occurring in fuel pins
- Part of knowledge has been gained from the irradiation of some target or blanket pins but for a limited irradiation time
- There is a lack of knowledge on the target or blanket and clad materials behavior for long irradiation time

– During in reactor operation, absorber pins are submitted to thermo-mechanical and physical-chemical phenomena that lead to structural and mechanical changes in both absorber and cladding materials. These phenomena are different from those occurring in fuel pins.

– For some absorber materials, the irradiation time leads to the carburation of the cladding with a risk of clad rupture. The absorber material may fracture with associated risk of fragments release inside the sodium in case of clad failure.

Severe Accidents—Containment of Reactor Fuel

Metal Fuel

- Low melting point
- Very good conductivity
- Sodium bonding between fuel and clad—very good gap conductance
- Results with cladding at high temperatures

Oxide Fuel

- High melting point
- Much inferior conductivity compared to metal fuel
- Helium bonding between fuel and clad—poor gap conductance

Severe accidents are those beyond design bases and could lead to fuel melting, and in especially severe cases, fuel relocation, recriticality and energetic core disruption. The extent to which designs need to demonstrate capability to withstand these events will be an important part of NRC reactor regulation.

Severe Accidents—Containment of Reactor Fuel (continued)

- In certain hypothetical accident initiators, the fuel can overheat sufficiently to melt the fuel and fail the cladding, releasing molten fuel into the coolant channels of the fuel assembly
  - The probability of such events is expected to be lower than 10^{-6} per reactor year
- If such fuel pin failure occurs, the course of the accident will depend mainly on when and where the failure happens, and how the fuel relocates
  - This process is affected by the conditions in the coolant channel as well

Severe Accidents—Containment of Reactor Fuel (continued)

- In a fast reactor, the fuel (i.e., the core) is not in the most critical configuration for fission
- Therefore, there is a theoretical potential for fuel relocation to substantially increase core reactivity, even to the extent of exceeding prompt criticality
- The most favorable occurrence under these conditions would be the ability to rapidly remove sufficient fuel from the core region so that the remaining fuel is not capable of maintaining nuclear criticality

Severe Accidents—Fuel Pin Failure

- **Oxide Fuel**
  - High melting point
  - Low conductivity compared to metal fuel
  - Helium bonding between fuel and clad—poor gap conductance
- If the fuel melting takes place followed by cladding breach, the fuel content will be ejected out of the pin into sodium
- Because sodium is much cooler, fuel may refreeze on the cold surfaces potentially leading to a flow blockage
  - Also a concern for inducing new clad failures in neighboring pins, thus leading to propagation of damage
- There is a probability that fuel solidification might result in a local criticality called recriticality condition

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(from Wigeland, Cahalan…)

For the case of oxide fuel, the high melting point of both the fuel (3025 K) and cladding (1700 K) delay fuel pin failure until the sodium coolant is rapidly boiling away (1200-1300 K) or has been completely vaporized in the coolant channel. The initial failure of the fuel occurs only in the relatively few hottest fuel subassemblies and is usually in the upper part of the active (fueled) core region because of the axial coolant temperature profile in the core, although in the case where coolant has already been vaporized or where reactivity addition is very rapid, many dollars per second, the fuel pin failure can occur closer to the code midplane in response to the power profile. In either case, the movement of the core materials, both fuel and steel, usually causing unfavorable changes in core reactivity. In addition, the initial movement of molten fuel or steel into the cooler regions above and below the active core region rapidly results in freezing of the molten core materials, blocking the coolant channels and preventing any further movement of steel out of the core region in that subassembly. Once these events have occurred, the accident inevitably continues with fuel pin failures in other subassemblies until most of, or the entire, core has melted. Analyses of the sequence of events have shown that significant recriticalities will occur, with transient power peaks of several hundreds or thousands of times nominal power. Termination of this accident occurs with an energetic disassembly of the core, potentially damaging or breaking the reactor vessel and threatening the containment building.

Severe Accidents—Fuel Pin Failure (continued)

- **Metallic Fuel**
  - Low melting point
  - Very good conductivity
  - Sodium bonding between fuel and clad—very high gap conductance
- With metallic fuel, the fuel pin failure is observed to take place in the upper core (TREAT experiments), where the sodium temperature is hottest
- When metal fuel contacts cladding, it can form a low-melting-point eutectic—may result in early cladding breach
- Following the cladding breach, molten fuel is sprayed into sodium, resulting in significant decrease in reactivity

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(from Wigeland, Cahalan…)

With metallic fuel, the relatively low melting point of the fuel (1350 K) and cladding (1000 K after alloying with fuel) generally results in fuel pin failure before the sodium coolant boils (1200-1300 K). As with the oxide fuel, the initial fuel pin failure occurs only in the relatively few hottest fuel subassemblies, but the failure is earlier in the transient with metallic fuel due to lower temperatures required for melting and the failure location tends to be fairly high in the upper part of the active (fueled) core region since the core temperatures are still more closely related to the coolant temperature profile than to the power profile.

The relatively low temperature of the molten fuel/steel alloy, at or below the sodium boiling point, does not contribute to vaporization of sodium within the core region and avoids the corresponding introduction of positive reactivity, limiting power rise during this stage of accident. The movement of the core materials, in this case an alloyed mixture of the fuel and cladding steel, into the liquid sodium coolant facilitates movement toward the upper core boundary. However, in this case, the cooler region above the active core region has a temperature above the melting point of the alloyed mixture, preventing freezing such that movement of the fuel/steel alloy away from the core region can occur, introducing such large negative reactivity that the reactor is no longer capable of sustaining fission. Since the coolant channels remain open, the core temperatures can be maintained. In this case, no design modifications are required to achieve such performance, since it is the result of the inherent thermophysical properties of metallic fuel and the interaction with steel cladding. Again, achieving such response is not the result of an active system, but is driven by other inherent phenomena such as gravity or pressure-driven flow.

Severe Accidents—Fuel Pin Failure (continued)

- With metal fuel, failure location is correlated more with the coolant temperature than with the power profile
- It was demonstrated consistently that the failure location is highly predictable
  - Above the centerline, close to the top of the fuel

Severe Accidents—Fuel Pin Failure (continued)

- With the oxide fuel, the opposite is true—failure location is mostly determined by the power profile than the coolant temperature profile, which makes the axial center of the fuel more susceptible to failures
- Experiments did not demonstrate predictability of failure location and consequences

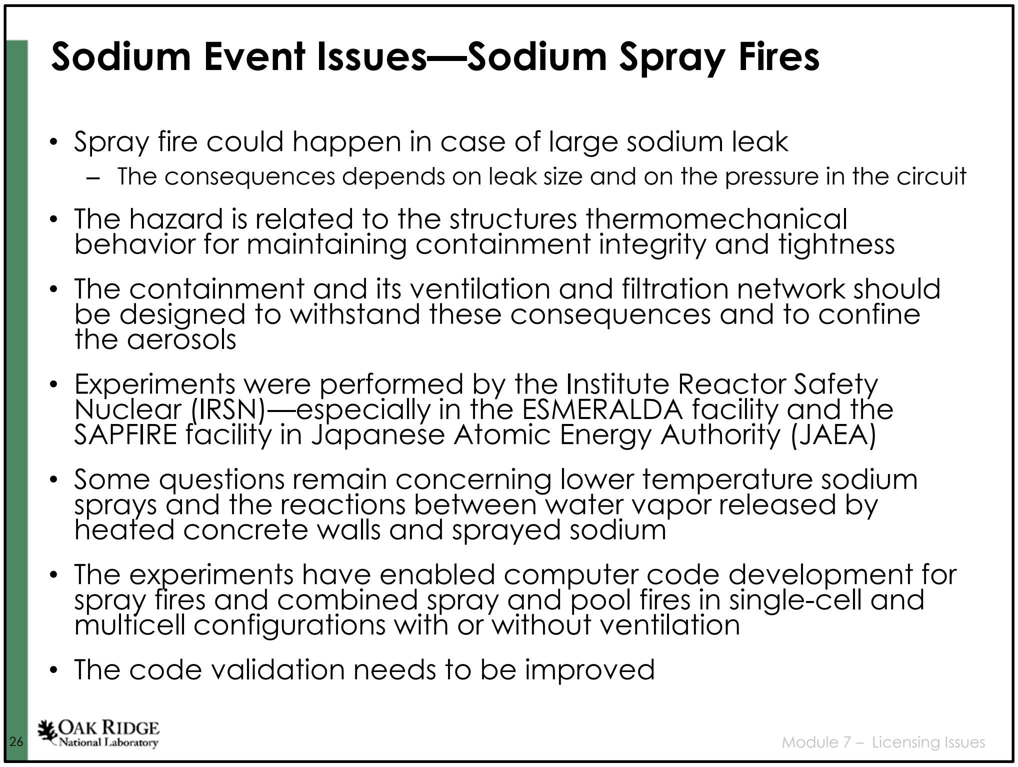
Severe Accidents—Operation of Fuel Pins with Breached Cladding

- Metallic fuel is chemically compatible with sodium reactor coolant but forms a low-melting point eutectic with the cladding at high temperature
  - Primary reason why sodium is used as the thermal bond between fuel and cladding
  - The behavior of failed metallic fuel pins during continued operation of the reactor is, therefore, governed by the properties of the fuel, its fission products, and the cladding not by fuel-sodium chemical reactions

Severe Accidents—Run-Beyond-Cladding-Breach (RBCB) Tests

- RBCB tests performed in EBR-II with U-Fs, U-Zr, and U-Pu-Zr fuel clad with Type 316, D9, and HT9 stainless steel … Purpose:
  - Confirmed the expected benign behavior of metallic fuel pins during operation following cladding breach
  - Characterized the release of fission gas, delayed neutron emitters—and possibly fuel—from breached pins

Ref: Abdellatof, M. Yacut, Long Life Metallic Fuel for the Super Safe, Small and Simple (4s) Reactor, Argonne National Laboratory, June 2008.



Ref: IRSN = Institut de Radioprotection et de Sûreté Nucléaire, ESMERALDA = Ref: J. Sharpenel, DSN/SESTR, Centre de Cadarache, France; and Ref: Y. Sophy, The Esmeralda Project, DSN/SESTR, Centre de Cadarache, France; Ref Sapfire: Yoshiaki Himeno, Current Status of Sodium Fires and Aerosol Research in Japan, Power Reactor and Nuclear Fuel Development Corporation, Japan (iaea.org/inis/collection… 33018342.pdf)

- The sprayed sodium metal ignites and burns during its path in the air before impacting room walls or floor. Combustion kinetic is fast and the consequences consist in pressure and temperature rises in the room. The hazard is related to the structures thermo-mechanical behaviour for maintaining containment integrity and tightness.

Sodium Event Issues—Sodium Pool Fires

- A sodium pool fire may develop in case of sodium spilling on the floor with negligible sodium spray fraction. The consequences are temperature rise for gas and walls, overpressure in the room, and aerosols production with possible release to environment.
- The containment and its ventilation and filtration network are designed to withstand these consequences and to confine the aerosols
- Contact between sodium and concrete could damage the structures and cause water vapor release.
- Experimental studies have been performed for pool fires

Sodium Event Issues—Sodium Fire Aerosol Behavior

- Sodium fires produce sodium monoxide and peroxide aerosols that change to sodium hydroxide particles with water vapor in the air
- In case of release to the environment, atmospheric dispersion must be considered for aerosol size, chemical form, and concentration determination
- IRSN experimental programs and computer code development provide models for prediction effects of sodium fires
- Computer codes exist at IRSN and JAEA for simulating sodium fire aerosols behavior in the room, in the ventilation network, and in open atmosphere in case of release to the environment.

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- Sodium fires produce sodium monoxide and peroxide aerosols that change to sodium hydroxide particles with water vapour in the air. Generally, aerosol concentration is high and makes human intervention impossible in the fire room. Aerosols could damage electric and electronic equipment. If burned, sodium contains radioactive products; hence, sodium fire aerosols could act as vehicle for contamination.
- In case of sodium fire aerosol release outside the building to the environment, sodium hydroxide and sodium carbonate are not equal regarding the toxicity and effect on human health. Atmospheric dispersion has to be considered for aerosol size and concentration determination.
- IRSN experimental programs concerned aerosols physical behaviour characterization, atmospheric dispersion, filtration device development, ventilation driving and equipments failure in aerosols presence.

Sodium Event Issues—Sodium/Containment Interactions

- The direct contact between concrete and hot sodium should be avoided. It causes steam and hydrogen release due to reaction between water vapor released by heated concrete and sodium
- IRSN experimental programs were performed with silico-calcareous concrete for characterizing physical phenomena. Specific concretes less reactive with sodium and metallic liners were tested
- JAEA has experimental data to understand the reaction phenomena of the greywacke-based concrete with sodium
- Computer codes were developed and validated on the basis of available experimental data
- Experimental and computational studies at JAEA are currently in progress to obtain extensive phenomenological information of sodium–concrete reaction and to improve the computational modeling

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- The direct contact between concrete and hot sodium causes steam and hydrogen release due to reaction between water vapour released by heated concrete and sodium. The solid material could be involved in exothermal reactions with sodium. The consequences are pressure and temperature increases in the room and explosion risk. Protection devices are designed for avoiding sodium-concrete interaction.
- IRSN experimental programs were performed with silico-calcareous concrete for characterizing physical phenomena. Specific concretes less reactive with sodium and metallic liners were tested.
- JAEA has experimental data to understand the reaction phenomena of the greywacke-based concrete with sodium.
- Computer codes were developed and validated on the basis of available experimental data. For some of them, validation has to be completed including modelling improvements (sodium-concrete thermal exchange coefficient calculation).
- Experimental and computational studies at JAEA have been currently in progress to obtain extensive phenomenological information of sodium-concrete reaction and to improve the computational modelling.

Sodium Event Issues—Sodium/Water Interactions

- Contact between sodium and water due to steam generator tube failures needs to be prevented. Reaction forces and chemical compounds could cause damage to other parts of the system. Leak detection systems are required and have to be developed and qualified. BN-600 operated with damaged steam generators valved off
- CEA experimental programs provide a lot of results concerning leak flow rate evolution, the pressure waves propagation and the mass transfer within the secondary circuit, and the damages caused on the neighboring exchange tubes and problems (efficiency and rapidity) arising from the sodium–water reaction detection

- Tightness loss of steel wall between sodium and water in a steam generator causes vapour penetration in sodium and sodium-water reactions. The damages could be significant and could affect safety function to confine radioactive materials as the sodium-water reaction propagates in the secondary circuit and may cause damages to the exchange tubes of the intermediate heat exchanger (IHX) which are a part of the second safety barrier. Leak detection systems are required and have to be developed and qualified.

- CEA experimental programs provide a lot of results concerning leak flow rate evolution, the pressure waves propagation and the mass transfer within the secondary circuit, the damages caused on the neighbouring exchange tubes and problems (efficiency and rapidity) arising from the sodium-water reaction detection.

Sodium Event Issues—Sodium/Water Interactions (continued)

- Experimental and analytical studies were carried out using the SWAT facility in JAEA, concerning self-wastage, target-wastage, and overheating of neighboring heat transfer tubes
- Several computer codes were developed, each of them is dedicated to a specific physical phenomenon-like water flow rate evolution, failure propagation, pressure rise within the secondary circuit, and overheating behavior
- Some lack of knowledge and needs of improvements remain, especially concerning the hydrogen detection systems
- The physical phenomena involved and the associated lack of knowledge will be strongly dependent on design options—new coolant fluid in place of water, design of vapor generator or heat exchanger, metallic material …

Summary of Major Design Issues Related to Licensing of LMRs

- Limited experimental data on prototypical length fuel
- Fission product behavior, variation of physical and mechanical properties with burnup
- Fuel/clad chemical interactions (eutectic formation) and behavior during BDBA accidents
- Positive coolant density feedback with decreasing coolant density—positive void effect (positive for PRISM; negative for 4S)
- Sodium fires and their suppression
- Seismic isolation design of structures
- Severe accident progression—source term, containment behavior, off-site planning
- Residual heat removal systems

Backup slides

Safety Analysis and Licensing Issues—Experience

- Design basis accidents
- Reactivity insertion
- Loss-of-cooling events
- Beyond design basis accidents
  - Fundamental changes
  - Unprotected transient over-power accidents
  - Unprotected loss-of-flow
  - Unprotected loss-of-heat sink

Safety Analysis and Licensing Experience—Design Basis Accidents (DBA)

- Design Basis Accidents
  - Reactivity insertion events
  - Loss-of-flow events
- Other topics for regulatory review
  - Natural circulating cooling
  - Piping integrity
  - Emergency power
  - Seismic design
  - Core thermal design and hot channel factors
  - Instrument and control design
  - Quality assurance
  - Radiation protection
  - Waste management
  - Sodium spills
  - Fuel handling
  - External events (e.g., fire, floods, tornados, earthquakes)

Examples of Design Basis Accidents—Reactivity Insertion Events

- Two general classes considered
  - Reactivity insertion events
  - Loss-of-cooling events
- Examples of reactivity insertion events
  - Continuous control rod run-out
  - Single control rod meltdown
  - Loss of hydraulic hold-down
  - Movement of radial core restraint
  - Cold sodium insertion

Safety Analysis and Licensing Experience—Loss-of-Cooling Events

- Loss of offsite electrical power
- Loss of both
  - Offsite electrical power
  - Emergency diesel–electrical power
- Loss of electrical power to one primary pump
- Continuous flow reduction by controllers
- Mechanical seizure of one primary pump
- Loss of air flow in the dump heat exchangers

Safety Analysis and Licensing Issues Experience—Beyond Design Basis Accidents

- Transient Over-Power with Failure to Scram
- Loss of Flow with Failure to Scram
- Loss of Heat Sink with Failure to Scram

- Basic assumptions
  - Protection systems fail to perform their function
  - More than a single active failure occurs
- BDBA are
  - Outside the design basis of the plant
  - Provide mechanistic way to assess potential radioactive release to public

FFTF Unprotected Transient Over-Power Accidents (UTOP)

- 50 cent/s ramp rate assumed as initiator
  - (maximum withdrawal rate of highest worth control rod)

Conservative Bounding Case

- Fuel pins assumed to fail near axial midplane
- Molten fuel flows toward failure location
- Sodium flashes near core midplane

  ➡ Ramp rate at time of disassembly up to $200/s
- With most of the sodium still in the core, disassembly pressure builds up rapidly ("hard" Equation-of-State)
- Maximum energy release ~150 MW-s (Ref: FFTF SAR)

FFTF Unprotected Loss of Flow (ULOF) (Oxide Fuel)

Conservative Bounding Case

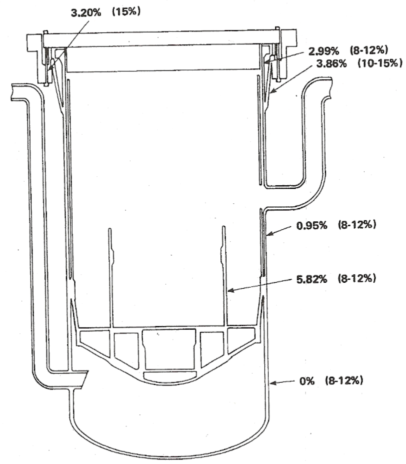
- Coolant boils within ~5 s
- Cladding melting shorting thereafter
- Assume fuel slumping
- Enter transition phase
- Ramp rate at time of disassembly (few $/s)
- Most of core void of sodium at this time
  - (hence, weaker Equation-of-State)
- Maximum energy generated bounded by 150 MW-s
- Containment capacity = 350 MW-s

Find out if this is FFTF oxide fuel

FFTF Unprotected Loss of Heat Sink (ULOHS)

- Analysis showed response similar to ULOF accident

FFTF Vessel Strains for 150 MW-s Energy Release



(Allowable strains to failure in parenthesis)

Ref: Alan E. Waltar, Key Aspects in Conducting Safety Analysis and Addressing Safety Issues Associated with FFTF and CRBR, Prepared for the Nuclear Regulatory Commission Under Arrangements by the Oak Ridge National Laboratory, November 19, 2008.

FFTF FSAR Analysis for the UTOP (Metal Fuel)

Realistic Case

- TREAT tests show pins fail near top of fuel
- Molten fuel is washed out by flowing sodium
- Energetics are very low

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This appears to be metal fuel

FFTF FSAR Analyses for the ULOF (Metal Fuel)

Realistic Case

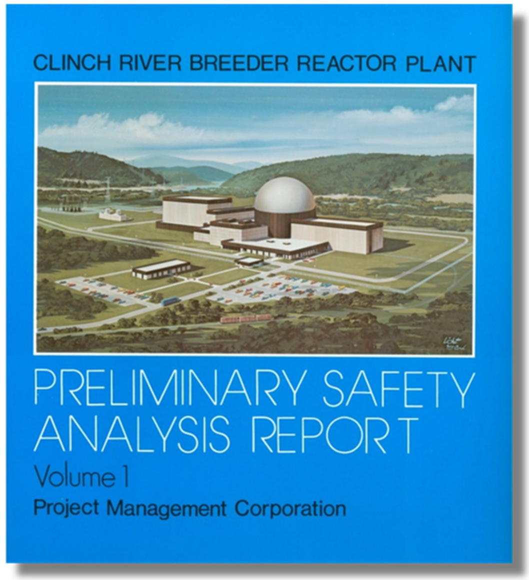
- TREAT tests showed slow boiling and loss of cladding, but difficult to model mechanistically, although SAS4A and SIMMER used to bound energy release
- Also used phenomenological arguments in transition phase to show that natural dispersion tendencies would preclude any breach of containment
- Containment temperature and pressure transients modeled with CACECO code
- Sodium/concrete tests confirmed that debris could be adequately cooled

Refs for SAS4A and CACECO computer codes to be provided in Codes Task

EBR-II Tests

- Experiments were performed with EBR-II where loss of flow was initiated full power
- Neutronic reaction shut down due to core expansion
- Reactor reached stable natural convection conditions without failures

Clinch River Breeder Reactor (CRBR)

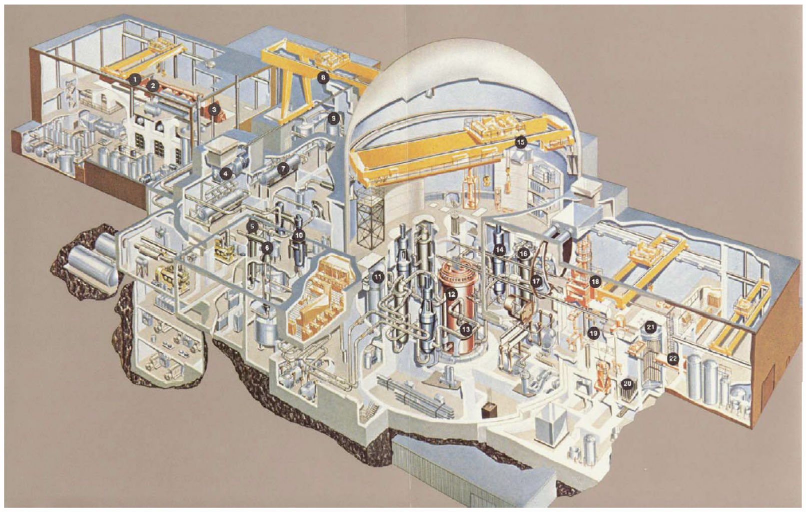


CRBR Design

- Mission: Demonstrate the safe and reliable operation of an LMFBR in a utility environment. Demonstrate LMFBR economics and the transition from technology develop to commercial operation
- 975 MW(t), 380 MW(e) (gross), MOX fuel (19% and 27% plutonium), three-loop primary system, three intermediate sodium loops to steam generators
- Reactor core: 198 fuel assemblies (108 inner/90 outer), 19 control assemblies (15 primary/4 secondary), 150 radial blanket assemblies
- Coolant inlet 388°C, outlet 535°C
- 10 psi steel containment
- Two independent reactor shutdown systems (both by moveable rods)
- Decay heat removal through three independent loops
  - Pony motors on primary and secondary pumps
  - Auxiliary decay heat removal through water side of SG
- Direct decay heat removal system independent of HTS loops
- Core physics and structural design for inherent negative power and temperature reactivity feedbacks

CRBR Plant

Clinch River Breeder Reactor Plant (CRBRP)



CRBR Regulatory Review

- CRBRP was licensed as a commercial power reactor by NRC
  - Project suspended in accord with the Presidential order in 1977
  - Licensing activities continued to obtain the equivalent to a construction permit in 1983
- Site selection in 1972, environmental report early 1975, PSAR April 1975
- Site issues: seismic (0.18 g) and tornado 290 mph rotation)
  - Consistent with other TVA sites
- As for FFTF, HCDAs received much regulatory review attention
  - Early agreement (1976) between NRC and the project that HCDAs would **not** be a design basis for containment
  - However, the role of severe accidents and characterization of their consequences dominated the attention of the interveners, the regulators, and the project
  - Licensing, and treatment of severe accidents, set the critical path for construction

CRBR Regulatory Review (continued)

- PSAR preparation was the responsibility of W-ARD
  - General design criteria
  - Preliminary design
  - Design basis event (DBE) analyses for PSAR Ch. 15
- The LMFBR base program, and particularly ANL, provided significant resources to address design and licensing issues
  - TREAT fuel testing; basic phenomenological test measurements and prototypic TOP and LOF transient tests
  - LOF and TOP accident analyses (SAS3D computer code)
  - Coolant and structural dynamics tests and analyses
  - Post-accident heat removal analyses
- ANL provided direct support to CRBRP licensing
  - Preparation of technical reports of analyses, experiments, and tests for use as PSAR support documents
  - Participation in meetings with NRC staff and ACRS

CRBR Regulatory Review (continued)

- HCDA issues compared to FFTF
  - Bigger reactor [975 MW(t) vs 400 MW(t)], more fuel
  - Positive coolant void reactivity worth (~3$ vs ~0$)
  - LOF sequence bounded energetics as in FFTF, but because of the positive coolant void worth, cladding failures and fuel melting occurred at higher power than in FFTF (~10 P_o vs ~1P_o)
  - Higher power LOF caused other phenomena in accident sequences that raised energy releases in analyses
- CRBRP structural limits (vessel head bolt strength) corresponded to an accident energy release of 661 MW-s: project structural margin beyond the design basis (SMBDB)
  - NRC (LANL performed independent analyses: 1200 MW-S

CRBR Regulatory Review (continued)

- Ultimately —
  - The ASLB ruled against intervenor's contention that HCDAs should be a design basis
  - NRC staff stated: "It is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum."
  - CRBRP project, with ANL support, built a technical case to justify exclusion
  - CRBRP met licensing requirements for construction without inclusion of HCDAs in the design basis

CRBR References

- *Clinch River Breeder Reactor Plant Preliminary Safety Analysis Report*, *Project Management Corporation, Vols. 1–27 (Updated through Amendment 77, May 1983)*
- *1974 Technical Progress Report,* Clinch River Breeder Reactor Project
- L. E. Strawbridge, "Safety Related Criteria and Design Features in the Clinch River Breeder Reactor Plant," *Proceedings of the Fast Reactor Safety Meeting*, CONF-740401-P1, pp. 72–92, American Nuclear Society, Beverly Hills, CA, April 2–4, 1974
- R. J. Slember, "Safety-Related Design Considerations for the Clinch River Breeder Reactor Plant," *Proceedings of the International Meeting Fast Reactor Safety and Related Physics*, CONF-761001, Vol. 1, pp. 112–125, American Nuclear Society, Chicago, IL, October 5–8, 1976
- L. E. Strawbridge and G. H. Clare, "Exclusion of Core Disruptive Accidents from the Design Basis Accident Envelope in CRBRP," *Proceedings of the International Meeting Fast Reactor Safety*, Vol. 1, pp. 317–327, American Nuclear Society, Knoxville, TN, April 21–25, 1985
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- *Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant*, USNRC Report NUREG-0968, Vols. 1 and 2, March 1983
- J. Cahalan, *Sodium Fast Reactors, Safety #2*, DOE/HQ, October 31, 2007; NRC White Flint, November 1, 2007 (Rev. October 2008)

FFTF Regulatory Experience

OAK RIDGE
National Laboratory

FFTF Chronology

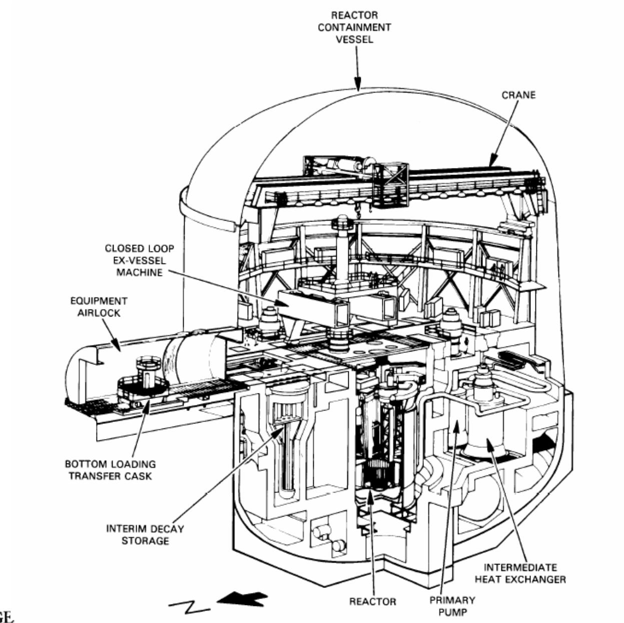
- Conceptual design studies 1966–1969
- PSAR submitted by HEDL in September 1970
- Initial construction authorization in September 1971; full construction authorization in March 1972
- ACRS letter in May 1973
- FSAR submitted by HEDL in December 1975
- Construction complete/sodium fill 1978
- Criticality February 1980; full power October 1980
- Research operations April 1982 to April 1992
- DOE Shutdown order December 1993

FFTF Design

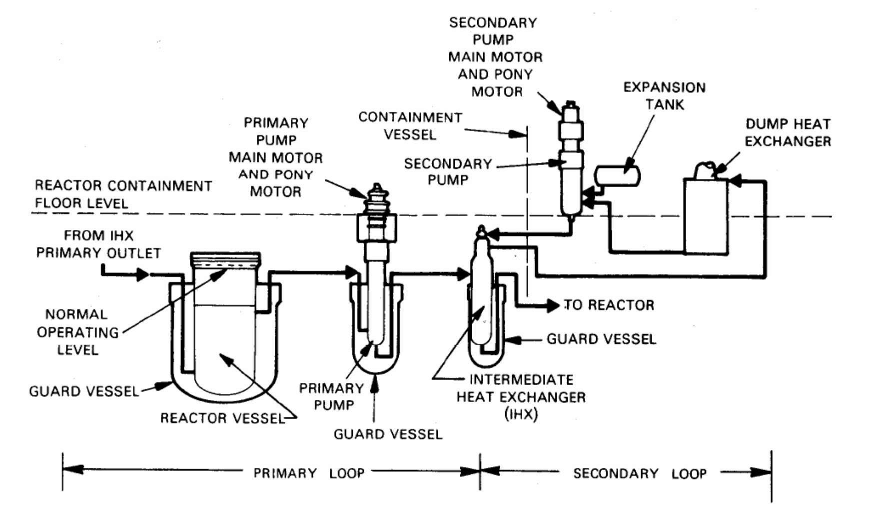
- Mission: Provide a prototypic LMFBR operating environment for testing and development of fuels, materials, and components
  - Secondary: Develop design and construction experience
- 400 MW(t), MOX fuel (22% and 27% plutonium), three-loop primary system, three intermediate sodium loops to air dump heat exchangers
- Reactor core: 73 fuel assemblies, 9 control assemblies, 9 test assemblies
- Coolant inlet 360°C; outlet 527°C
- 10 psi steel containment
- Two independent reactor shutdown systems (both by moveable rods)
- Forced and natural convection decay heat removal through three independent loops
  - Pony motors on primary and secondary pumps
- Core physics and structural design for inherent negative power and temperature reactivity feedbacks

FFTF Site—Hanford, Washington

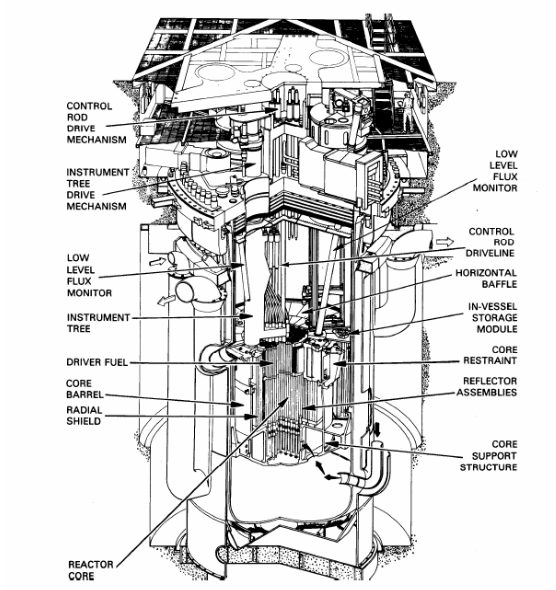
FFTF Containment Building View



FFTF Loop-Type Primary and Secondary Systems



FFTF Reactor and Vessel Design



FFTF Regulatory Review

- As an AEC project, FFTF did not require licensing as for commercial LWR plants, but technical review by NRC was requested
  - The depth and detail of the NRC review was similar to full licensing
  - Construction and operation permission; ACRS letters
- Site selection in 1968; site evaluation report prepared in 1969 and submitted for review in July 1970 (See "Chronology"). PSAR September 1970. First meeting with NRC staff November 1970; first ACRS meeting December 1970
- Site issues: seismic (0.25 g) and tornado (150 mph rotation)
  - Studies and analyses submitted to NRC and ACRS for review
- Hypothetical Core Disruptive Accidents (HCDA) received the greatest regulatory attention and review emphasis
  - Basis for evaluation of containment margins (10 CFR 100 offsite dose)
  - Project position: HCDA was not a design basis (15 MW-s margin evaluation basis)
  - NRC requested further study; response by HEDL and ANL

FFTF Regulatory Review (continued)

- A major part of the LMFBR safety base program was oriented to support FFTF regulatory review
- At HEDL
  - Transient Overpower (TOP) accident analysis (MELT computer code)
  - TOP fuel testing (TREAT)
- At ANL
  - Loss-of-Flow (LOF) accident analysis (SAS3A computer code)
  - LOF fuel testing (TREAT)
  - Post-Accident Heat Removal (PAHR) analyses and experiments
  - Structural dynamics analysis and testing
  - Fuel Element Failure Propagation (FEFP) studies and experiments
  - Coolant dynamics analyses and experiments
  - Fuel dynamics analyses and experiments (OPERA)
  - High-temperature materials properties
  - Fuel coolant interactions (FCI) analyses and experiments (OPERA)

FFTF Regulatory Review (continued)

- LMFBR safety base program activities also performed at ORNL, AI, GE, and W-ARD
  - Activities coordinated under HEDL technical direction
- ANL provided direct support to FFTF licensing
  - Preparation of technical reports of analyses, experiments, and tests for use as FSAR support documents
  - Participation in meetings with NRC staff and ACRS
- Regulatory review for construction nominally concluded with the May 1973 ACRS letter, but open issues continued to receive attention
  - HCDA energetics
  - Design fallbacks, including sealing the reactor head compartment and ex-vessel core melt retention
  - Piping integrity; provision for pipe break mediation design, and surveillance and in-service inspection
  - Natural convection cooling and emergency power

FFTF Regulatory Review (continued)

- Through 1976, HEDL and ANL continued to meet with NRC staff and ACRS
- ANL supplied technical support for resolution of the HCDA energetics and core melt retention issues
  - In 1974, NRC concurred with the ANL assessment that HCDA energetics would not exceed FFTF capability. Also, NRC concurred that sealing of the head compartment would not significantly improve containment margins
  - In 1975, NRC recommended that construction could be completed without addition of an ex-vessel core catcher
- The FFTF FSAR was issued in March 1976 followed by an NRC staff review
- The NRC staff review continued, and the Final Safety Evaluation Report was issued in August 1978. The SER stated that the major unresolved issues were natural convection verification, control room habitability, piping integrity, and containment margins
- Natural convection verification testing was performed during start-up

FFTF Regulatory Review (continued)

- A safety-grade system to provide control room isolation upon detection of unacceptable levels of sodium aerosols or radioactivity was added
- The piping integrity and containment margin issues were resolved without design changes by additional analyses and information submittals to NRC
- ACRS concurred with NRC findings in a November 1978 letter
- Coolant filling was accomplished in 1979, and fuel loading began
- First criticality was in February 1980, and power operation began in October 1980

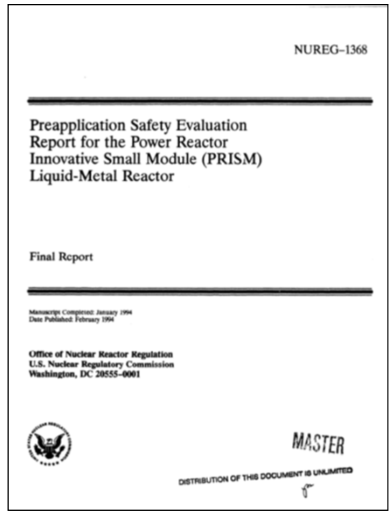
FFTF References

- *Fast Flux Test Facility Preliminary Safety Analysis Report, Vols. 1 and 2*, September 1970
- C. P. Campbell, *A Summary Description of the Fast Flux Test Facility*, HEDL-400, December 1980
- D. E. Simpson, "FFTF Design for Safety," *Proceedings of the Fast Reactor Safety Meeting*, CONF-740401-P2, pp. 1041–1060, American Nuclear Society, Beverly Hills, CA, April 2–4, 1974
- D. E. Simpson, "Resolution of Key Safety-Related Issues in FFTF Regulatory Review, *Proceedings of the International Meeting on Fast Reactor Safety and Related Physics*, CONF-761001, Vol. II, pp. 400–410, American Nuclear Society, Chicago, IL, October 5–8, 1976
- A. R. Schade and D. E. Simpson, "FFTF Regulatory Review for Operating Authorization," *Proceedings of the International Meeting on Fast Reactor Safety Technology*, Vol. 5, pp. 2425–2430, American Nuclear Society, Seattle, WA, August 19–23, 1979
- *Safety Evaluation Report by the Office of Nuclear Reactor Regulation*, USNRC, for the Fast Flux Test Facility, USNRC Report NUREG-0358, August 1978 (Supplement No. 1, May 1979)
- J. Cahalan, *Sodium Fast Reactors, Safety #2*, DOE/HQ, October 31, 2007; NRC White Flint, November 1, 2007 (Rev. October 2008)

NUREG-1368 on PRISM (ALMR)

8 Design Features Different From LWRs

- Accident evaluation
- Calculation of source term
- Containment
- Emergency planning
- Staffing
- Heat removal
- Positive void coefficient
- Control room design


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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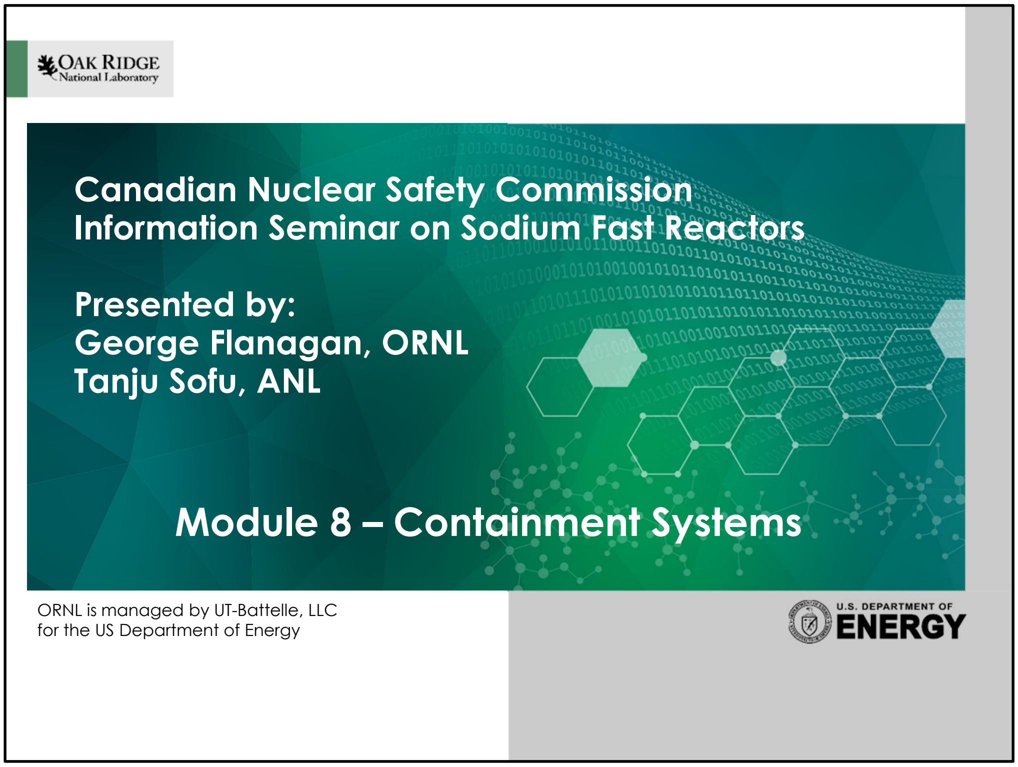
PRISM References

- Pre-application Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Reactor, USNRC Report NUREG-1368, February, 1994
- P. M. Magee, U.S. ALMR Licensing Status, Proc. Intl. Topical Mtg. Advanced Reactor Safety, pp. 1011–1017, American Nuclear Society, Pittsburg, Pennsylvania, April 17–21, 1994
- Eric Loewen, "Advanced Reactors: NUREG-1368 Applicability to Global Nuclear Energy Partnership," NRC Regulatory Conference, March 15, 2007
- J. Cahalan, *Sodium Fast Reactors, Safety #2*, DOE/HQ, October 31, 2007; NRC White Flint, November 1, 2007 (Rev. October 2008)



SFR containments need to be able to accommodate some events that do not exist for LWRs These include energetic core disruption, sodium fires and sodium (oxide and hydroxide) aerosol dispersion. On the other hand, new SFR containments can be designed to withstand terrorist attacks and other external events without the legacy issues of existing LWR designs.

SFR Containment

- LMR containment systems have evolved
  - Early systems were required to contain very high pressures and temperatures resulting from a hypothetical core disruptive accident (HCDA) with large energy releases
  - HCDAs involved core melting and fuel compaction followed by violent fuel-coolant and fuel-concrete interactions (CRBRP containment)[1]
  - More recent analyses indicate that the energy releases from HCDAs are less than early analyses indicated
  - Sodium aerosol analyses and experiments indicate that agglomeration is expected along with plate out in the systems inside containment, thereby reducing analyzed releases to the environment[2,3]

[1]Alan E. Waltar and Albert B. Reynolds, *Fast Breeder Reactors*, Pergamon Press, 1981, and H. A. Bethe and J. H. Tait, *An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses*, RHM-567-113, April 1956.

[2]A. B. Reynolds and T. S. Kress, "Aerosol Source Considerations for LMFBR Core Disruptive Accidents," in *Proceedings of the SCNI Specialists' Meeting on Nuclear Aerosols in Reactor Safety*, Gatlinburg, TN, April 1980

[3]M. Silberberg, Chairman, *Nuclear Aerosols in Reactor Safety*, A State-of-the-Art Report by a Group of Experts of the OECD NEA Committee on the Safety of Nuclear Installations, June 1979.

1. Ref: Waltar and Reynolds, op. cit., and H. A. Bethe and J. H. Tait, An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses, RHM-56-113, April, 1956
2. A. B. Reynolds and T. S. Kress, "Aerosol Source Considertions for LMFBR Core Disruptive Accidents", Proc. CSNI Spec Mtg on Nuclear Aerosols in Reactor Safety, Gatlinburg, TN, April 1980.
3. M. Silberberg, Chairman, "Nuclear Aerosols in Reactor Safety" A State of the Art Report by a Group of Experts of the OECD NEA Committee on the Safety of Nuclear Installations, June 1979.

4. Early HCDA analyses assumed coherent reassembly of core into most reactive configuration and explosive neutronic reaction. Subsequent analyses and experiments indicate that core expansion and fuel sweep out preclude the energetic recriticality assumptions inherent in the Bethe Tait and Hicks Menzies analyses. See subsequent slide.

SFR Containment (cont.)

- Accidents with fuel failures:
  - In pool design, combination of reactor vessel and guard vessel provide containment function
  - In the loop design, all primary piping is double walled to provide containment function
- Recent designs (PRISM) and 4S proposed an underground reactor with a dome over the reactor vessel
- 4S assumptions of operations with 1% fuel failure and low-power level reduce need for robust containment

SFR Containment (cont.)

Comparison of PWR, 4S, and PRISM Containments

Category	PWR	4S (metal)	PRISM (metal)
Material	Steel-lined prestressed or poststressed concrete	Material not stated	Cr–1 Mo steel alloy (guard vessel) top dome
Size	~150-ft diam × ~250-ft high (large dry type) (see Fig. P5)	Slightly large than reactor vessel	19-ft, 10-in. OD 59-ft, 6-in. length
Thickness	~3.5 ft	1 in.	1 in.
Location	(Mostly) Above ground	Guard vessel—below ground (top dome above ground)	Guard vessel—below ground (top dome above ground)

4 OAK RIDGE National Laboratory Module 8 – Containment Systems

Ref: PRISM Preliminary Safety Information, ML082880369 GEFR-00793 – Vol 1, December, 1987; and ML082880397 GEFR-00793 – Vol 4, December, 1987. Ref: PWR: -Pressurized Water Systems, USNRC Technical Training Center, June, 2003. Ref: Toshiba 4S- ML081440765- Toshiba – Submitted Design Description of 4S – 4S Design Description, May 2008.

Containment Systems Used in Selected SFR Plants

- Containment type—single containment
  - Open head compartment and low-leakage outer compartment
  - Reactors—FFTF, EBR-II, JOYO
- Double containment
  - Sealed, inert high-pressure inner containment barrier, surrounded by a low-leakage out containment building
  - Reactors—FERMI, SEFOR

Ref: For descriptions of FFTF, EBR-II, JOYO, FERMI, and SEFOR, see IAEA TECDOC-1531, Fast Reactor Data Base 2006, International Atomic Energy Agency, December 2006

Containment Systems Used in Selected SFR Plants (cont.)

- Containment/confinement
  - Sealed, low-leakage inner containment barrier, surrounded by ventilated low-pressure outer confinement building with discharge to stack via an air cleaning system
  - Reactors—PFR, CRBRP, Super-Phenix, BN-350, BN-600

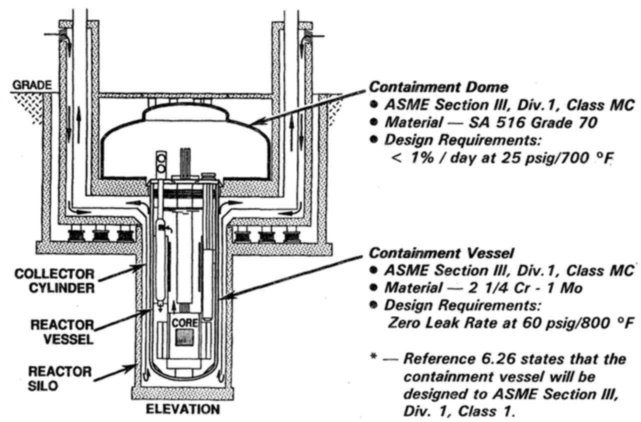
Ref: For descriptions of PFR, CRBRP, Superphenix, BN-350, and BN-600, see IAEA TECDOC-1531, Fast Reactor Data Base 2006, International Atomic Energy Agency, December, 2006

Containment Systems Used in Selected SFR Plants (cont.)

- Multiple containment with pumpback
  - Sealed, high-pressure inner containment surrounded by one or more outer barriers. A negative pressure zone is maintained in the out space by pumping back to the inner containment space. Eventual venting to a stack via the air cleaning system is provided
  - Reactors—SNR-300

Ref: Waltar and Reynolds, op. cit., p 688. and S. E. Seeman and G. R. Armstrong, “Comparisons of Containment Systems for Large Sodium-Cooled Breeder Reactors”. HEDL-TME 78-35, Hanford Engineering Development Laboratory, April 1978.

PRISM "Containment"



Ref: U. S. Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid Metal Reactor", Final Report, NUREG-1368, February 1994.

Decay Heat Removal

- SFRs have used a number of systems to remove decay heat
  - Prior to the PRISM design most used the existing power train with pumping power provided by the external grid or diesel generators
  - No Emergency Core Cooling Systems
  - PRISM introduced the use of Reactor Vessel Auxiliary Cooling (RVAC)
    - First introduced by the mHTGR
    - Passive - relied on internal convection in primary pool, conduction to and through the vessel, and radiation and convective cooling using air- no reliance on AC power
    - Supplemented by passive auxiliary cooling of steam generator by natural convection of air.

Decay Heat Removal (cont.)

- Large Pool reactors use an additional heat exchanger submerged in the primary sodium pool which dumps the energy to an air draft heat exchanger-Direct Reactor Auxiliary Cooling (DRAC)
  - Passive - does not rely on AC power
  - Relays on convective heat transfer via natural circulation to remove heat from primary sodium via an intermediate loop to the air draft heat exchanger
- Variations can be found in different SFRs depending whether the system is a loop or pool design.
  - Some loop designs use air cooling of the intermediate heat exchanger

OAK RIDGE National Laboratory

Canadian Nuclear Safety Commission Information Seminar on Sodium Fast Reactors

Presented by:
George Flanagan, ORNL
Tanju Sofu, ANL

December 10-11, 2018

Module 9 – Selected SFR Operating Experience

ORNL is managed by UT-Battelle, LLC
for the US Department of Energy

U.S. DEPARTMENT OF ENERGY

This module focuses on five plants that have operated for a significant time and have produced commercial electricity as well as closing the fuel cycle, thus demonstrating breeding capability.

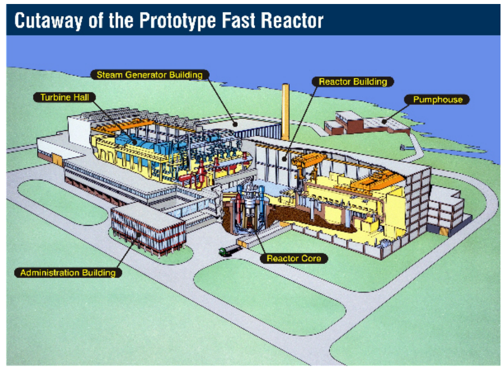
Operating Experience

- Focus on plants that have operated and generated electricity
  - PFR
  - Phenix
  - Super-Phenix
  - BN-600
  - BN-350
  - MONJU
  - Fermi-1

All information for these plants, except for Monju is from IAEA TECDOC 1569 Liquid Metal Cooled Reactors, Design and Operation. Monju information is from Monju and FBR Development in Japan, Fast Breeder Research and Development Office, JAEA, February 11, 2011., from Wikipedia March 14, 2011, and from http://www.jaea.go.jp/04/monju/EnglishSite/contents02.html

PFR Prototype Fast Reactor (British)

- 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



PFR Operation

- Steam generator leaks
- Sodium aerosol deposits on control rods drives
- Major oil leak into primary circuit
- Cracks in air heat exchangers of decay heat rejection loops (10 tonnes of NaK in these loops)
- Sodium mixing and vibration problems

PFR Operation (cont.)

- Steam generator—single wall tubes
- U tubes—all welds above sodium level
- A total of 37 gas-space leaks were experienced in PFR SG units in the period 1974 to 1984
  - 33 of these occurring in evaporators
  - 3 in superheaters
  - 1 in a reheater
  - All the gas-space leaks originated at the welds between the tubes and the tubeplates

PFR Operation (cont.)

- In the period 1984–1987, all the six austenitic tube bundles were replaced by new tube bundles. The design benefited from the early experience of caustic stress corrosion following the leaks in the austenitic units
- Complete replacement of superheater and reheater tubes by 9 Cr–1 Mo ferritic steel
- Considerable effect on plant availability while fixing leaks

PFR Operation (cont.)

- Steam generator tube cracks were all on sodium side
- Large under-sodium leak caused failure of rupture disc and plant shutdown
- 150 kg of water penetrated the sodium system
- Possible for large number of tubes to fail simultaneously (40 guillotine breaks in 10 s)
- Flow-induced vibrations were major factor in under-sodium breaks

PFR Operation (cont.)

- Sodium aerosol problems
  - Caused sticking of rotating refueling plugs in BN-350 and BN-600
  - Caused deposition control rod cracks, causing difficulty in insertion in Phenix and Super-Phenix
  - PFR still had argon purge gas problems. Needs to be free of hydrogen, oxygen, or methanes
  - Oil leak problems. Black tarry deposits

PFR Operation (cont.)

- Fuel
- Fuel assembly bowing was an issue
  - 14 mm allowed
  - Needed to rotate assemblies to stay below limit
  - 50 dpa maximum limit

PFR Operation (cont.)

- Fuel cycle
  - Fuel cycle closed in June 1982 when plutonium loaded into core
  - Reprocessing plant had treated 23 tonnes of oxide fuel
  - Recovered over 3.5 tonnes of plutonium
  - Highest burnup 17.6%
  - PUREX cycle used, recovered over 99.5% of plutonium

Phenix (France)

- Nominal power May 18, 1973, 255 MW(e)
- 400°C in, 550°C out of core
- Pool type
- Average burnup 13.5% at periphery of core
- Peak thermal efficiency ~45%
- Closed fuel cycle PUREX
- Breeding ratio 1.16

Phenix Milestones

- 1973–1990 — Demonstration of fast reactor technology and closed 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
- Slab on ball and socket pads
- Pumps and heat exchanger on movable supports, below seals
- Sodium in vessel—2 zones, cooler zone near vessel walls
- Guard vessel
- Three primary sodium pumps—variable speed rotating centrifugal

Phenix Operation

- **Fuel**
  - UO_2 PuO_2 fuel 217-pin bundles
  - 103 subassemblies
  - 5.5-mm diam pellets in stainless steel cladding
  - First 2 years, a few fuel failures detected by "wet" sipping at subassembly outlets

Phenix Operation (cont.)

- The following major incidents or unforeseen events
  - Series of leaks of secondary circuit sodium 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 secondary pipes and some components, particularly in austenitic steel like 321 stabilized with titanium over about 30 years operation

Phenix Operation (cont.)

- Pump vibration due to faulty construction
- Negative reactivity shutdowns probably due to gas bubbles passing through core
- Sodium aerosol deposits—jamming shutdown rod drives
  - Two shutdown systems
  - One protected with bellows—never a problem with these
- Extensive post-shutdown examination. Thermal striping cracks in 304 SS steam generators

Phenix Fire Protection

- During the test and renovation, new studies and significant works on protection of the steam generator building against large sodium fires have been undertaken, including
  - The separation by steel insulated walls and doors of firebreak zones to limit the spreading of a large fire
  - Two separation steam generator cells reconstructed to resist a major sodium fire with temperature of about 1000°C for 30 min
- The partitioning or the housing of cable trays and building steel structures
- The ventilation and the smoke cleaning circuits
- Installation of multisampling detection circuit

Phenix Main Production Data

Phenix Main Production Data

Characteristic	Value
Effective full power days, EFPD	3900
Gross electrical energy production, GHh	22,424,087
Load factor (since commissioning in July 1974), %	~50
Number of irradiated subassemblies	829
Number of irradiated pins	166,521
Burnup (maximal) (heavy atoms)	17%

Data from from IAEA TECDOC 1569

Super-Phenix (France)

- Pool type
- Derived from Phenix
- Changes from Phenix
  - Primary sodium purification units within vessel
  - Four helical coil steam generators, 750 MW(t) each
  - Larger fuel subassemblies to allow more burnup
  - Simplified design of main vessel and roof
  - Dome over upper part of vessel to provide containment

Super-Phenix 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

Super-Phenix Characteristics (cont.)

- Sodium flow rate (primary/secondary), tonnes/second—4 × 4.24/4 × 3.27
- 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

Super-Phenix Characteristics (cont.)

- Steam flow rate—4 × 340 kg/s
- Feed water temperature—237°C
- Type of steam-water cycle—steam reheating system
- Number of SG per loop—1 once-through SG, no reheat
- Total mass—194 tons
- Cost of power—2.7X that of Paluel PWR

Cost data from: M. RAPIN, Fast breeder reactor economics, presented in the Royal Society Meeting on the Fast-neutronbreeder
fission reactor, London, U.K., 24-25 May 1989; R. CARLE, Detailed design studies demonstrate major
improvements in economics. Nucl. Eng. Int., February 1988

Super-Phenix Shutdown

- 11 years existence, operation for 4.5 years—rest of time was for fixing things
- Shutdown July 1996
- 100 events, 16 were sodium related
- Major factor: sodium leak from used fuel storage drum

Super-Phenix Shutdown (cont.)

- Since December 24, 1996, the reactor scheduled shutdown followed by legal cancellation of its operation license. It was made very clear that the reason of the shutdown was in no way with safety problems
- With economy, the government said that when uranium appears now durably cheap, there is no need today to operate an industrial fast reactor prototype
- "Cost more than expected"
- Shut down by government directive after Chernobyl accident. Very few issues were caused by sodium

Super-Phenix Operation

- Very little problems with single wall steam generators
- Fuel handling drum
- Drum holds fuel in sodium until power down from 25 kW to 7.5 kW, where fuel can be transferred for water cleaning and storage
- Storage drum 9-m diam, 13-m high with safety vessel outside (150-mm spacing)
- Failure of drum caused concern because safety vessel is made of the same material and a leak could not be tolerated in it

Super-Phenix Operation (cont.)

- The destructive examination samples taken at the beginning of 1988 showed that cracking was very probably due to
  - The existence of start site (microcracking) in zones of high hardness
  - Embrittlement by hydrogen
  - The cracks developed disruptive zones under the influence of residual welding stresses close to the elastic limit of the material
- Replaced by the fuel transfer station "FTS"

BN-600 (Russia)

Pool Type

- Advantages of pool type
  - Pipelines with high-temperature coolant, operating under stress are excluded
  - Cumbersome electric heating cables and the sealed concrete cells for location of the primary equipment are excluded
  - Less metal is used for the components, and the amount of construction work is greatly reduced

USSR experience indicates distinct preferences for pool type layouts.

BN-600 Characteristics

- Advantages of pool type (cont.)
  - The surface area of load-bearing walls separating radioactive sodium from the external environment is largely reduced
  - Absolute leak-tightness of the main primary circuit pipes is not required as leaks are confined within the reactor vessel
  - The reactor vessel is of simple cylindrical shape—12.8 m in diameter and 12.6-m high, with no nozzles below the sodium level
  - The low cover gas pressure in the reactor (~0.4 kg/cm^2) enables the large-sized reactor vessel to be made with a small wall thickness (30–40 mm)
  - As experience has shown, this kind of vessel can be assembled on-site from individual factory-produced parts with minimal problems

BN-600 Characteristics (cont.)

- The distinctive feature of the BN-600 reactor is bottom support of the reactor vessel which gives, in the designers' opinion, certain structural and technological advantages compared with alternative option of top-suspended reactors. Through a support ring welded at the point where the cylindrical wall joins the base, the vessel is seated on foundation roller supports

BN-600 Characteristics (cont.)

- Rotating plug top
- IHXs are shielded so little secondary sodium activation
- Centrifugal pumps
- Core run 450 days—150 days at full power

BN-600 Characteristics (cont.)

Fuel Handling

- Three of 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-600 Characteristics (cont.)

Fuel Handling (cont.)

- Two inclined loading–unloading elevators, which transport the assemblies from the reactor to the transfer box of the handling and transport channel and back
- An assembly transfer mechanism situated in the transfer box, which transfers assemblies from the elevators to the handling and transport channel and back
- The sodium level in the reactor vessel is such that transportation of spent assemblies inside the vessel take place under sodium
- 40 reactor reloads have been carried out satisfactorily

BN-600 Operation

- Up to January 1, 2004, the reactor plant total on-power operation time amounted to
164,000 h, and about 88,000 GWh of electricity was generated
- Base load operation
- Control system is such that plant can continue to operate at 67% power with one loop shut down
- Load factor would have been ~7% if could not run with loops out for maintenance

BN-600 Operation (cont.)

- BN-600 is operating as a base load plant with the reactor not participating in the regulation of load and frequency in the power system
- The reactor power was, therefore, nominal [600 MW(e)]
- 2/3 of nominal during operation without one of the heat transfer loops is not available or zero during shutdown
- The time spent by the BN-600 is characterized by the following data
  - Nominal [600 MW(e)]: 68%
  - 2/3 power: 11%
  - Zero power: 21%

BN-600 Operation (cont.)

Steam Generators

- Leaks are intolerable, so BN-600 has multiple units that can be shut off while the plant continues to run on other SGs
- BN-600 has three secondary sodium circuits each with
  - Eight separate steam generator modules and each of these consists of separate evaporator, superheater, and reheater sections making a total of 72 separate heat exchangers
- At least, partly because of this, availability of BN-600 has been consistently high

BN-600 Operation (cont.)

Steam Generators (cont.)

- Studies showed that the most likely cause of the water-into-sodium leak was manufacturing faults
- This situation occurred in the superheater modules (7 leaks) and reheater modules
  (4 leaks), which are made of austenitic steel

BN-600 Operation (cont.)

Steam Generators (cont.)

- The monitoring system worked satisfactorily under these conditions and, in most cases, enabled the time of the leak, the section it was in, and even the module to be determined promptly
- The confinement system also operated satisfactorily. It ensured a controlled dumping of argon–hydrogen mixture without the maximum design pressure being exceeded
- Problems were encountered in implementing the algorithm for rapid creation of nitrogen counter pressure on the tertiary circuit side following evaporation of the steam-and-water mixture from the affected module

BN-600 Operation (cont.)

Steam Generators (cont.)

- Operating experience revealed the following possibilities
  - Continuing operation of the loop with the 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 (cont.)

Sodium Leaks

System	Number	Quantity, L	Number of sodium burnings
Reactor	0	—	—
Intermediate heat exchanger	0	—	—
Storage drum	0	—	—
Primary auxiliary systems	5		—
– Gas purification	1	0.1	—
– Sodium purification system	4	0.3; 3; 0.2; 1000	1
Secondary circuit	18	—	—
– Main pipelines	0	—	—
– SG valve seals	4	1; 300; 30; 10	3
– SG leak detection system	1	2.0	1
– Drain and blow-off lines	10	0.2; 1; 10; 600; 300;	6
		100; 0; 0; 1; 0.0	
– Sodium storage	3	1.0; 0; 0	—

BN-600 Operation (cont.)

Main Causes of Sodium Leaks

- Poor quality repair: 8 events
- Latent defects of manufacturing and mounting: 6 events
- Depletion of equipment lifetime due to inadequacy of the design: 7 events
- Equipment design imperfections: 4 events
- Human errors during operation: 2 events

BN-600 Operation (cont.)

Sodium Leaks (cont.)

- On October 7, 1993, the largest leak happened on the pipeline for removal of sodium from the cold trap. The total amount of sodium escaped during the event was assessed to be approximately 800 kg. Failure due to thermal stresses
- The largest leakage of secondary sodium happened in May 1994 in a drain pipeline (ID 48 mm). Approximately 600 L of sodium were lost but only about 30 kg burned
- The remaining sodium was retained and smothered with extinguishing powder in the catch system. In both leaks, the protective systems were effective. The damage was not extensive, and repairs were affected quickly
- Of the 27 cases of leaks, in only one case was it necessary to shut down the reactor

BN-600 Operation (cont.)

Reactor Core Fuel

- Fuel failures were due to stress-induced corrosion of stainless steel cladding, particularly in peripheral sections of core
  - Because of fuel shuffling and rotation, fuel on peripheries reached 54 kW/m and 710°C
- Changes in operation
  - The core active height was increased from 75 to 100 cm, decreasing the fuel rod maximum linear heat rating to 47.2 kW/m
  - Reshuffling and rotation of the fuel assemblies were eliminated
  - Swelling-proof, cold-worked austenitic steel was used for the cladding and for the fuel assembly ducts

BN-600 Operation (cont.)

Reactor Core Fuel (cont.)

- Ferritic steel was used in the new duct design and boron-modified, cold-worked austenitic steel for cladding
- Fuel burnup in the core has reached 10% heavy atoms with a fuel cycle length of 160 effective power days (EFPD)
- Development of advanced radiation-resistant steels was (and is) the main problem in the attainment of higher fuel burn up

BN-350 Characteristics (Kazakhstan)

BN-350 Loop-Type Plant

- First commercial plant; on shore of Caspian sea (now Kazakhstan)
- Fast sodium-cooled reactor, six primary loops
  - Six intermediate (secondary loops)
  - Steam generators
  - Refueling complex (integrated mechanical system)
  - Automated process control system, including the reactor control and protection system
  - Diagnostic systems for monitoring the operating state of the safety-related components and systems
- EM pumps used in part of secondary system

BN-350 Characteristics (cont.)

BN-350 Basic Operating Parameters

1. Reactor thermal output, 750 MW
2. Primary sodium temperature at reactor inlet/outlet, 288°C/437°C
3. Sodium flow through reactor, 141,000 tonnes/hour
4. Secondary coolant temperature at SG inlet/outlet, 420°C/260°C
5. Sodium flow in secondary loop, 340 tonnes/hour
6. Number of operating loops (plus one reserve), 5
7. Main steam temperature/pressure, 405°C/4.5 MPa
8. Steam flow, 1070 tonnes/hour

BN-350 Characteristics (cont.)

- The BN-350 plant history is as follows
  - 1965–1971: construction period
  - November 29, 1972: first criticality of the reactor
  - July 16, 1973: power startup of the reactor. The extended startup was 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
  - End of 1973–February 1975: SG repairs; 1973–1975: operation at power levels up to 300 MW(t)
  - From March 1975: operation at 650–750 MW(t) for electrical power [~150 MW(e)]
  - Generation and sea water desalination (~100,000 tons of desalinated water per day)
  - From January 1996 to June 1998: operation at 420 MW(t), 50 MW(e) producing 45,000 tons of distilled water per day
  - April 1999: final shutdown

BN-350 Operation

- The BN-350 steam generators consist of two superheaters with U-shaped tubes and two evaporators with re-entrant tubes inside which water flows under natural convection and partial evaporation conditions
- Metallographic examination of a great number of re-entrant tubes showed the presence of microcracks in the tube-to-bottom weld joints

BN-350 Operation (cont.)

- The reactor was designed for thermal output of ~1000 MW, but in the early periods of operation, its power level was limited by unsatisfactory operation of the steam generators
- Power was subsequently restricted to 750 MW
- AGV load factor based on restricted operation was 85%
- Numerous cladding failures
  - Activity in primary sodium gamma dose on surface of sodium equipment 8.9 microsieverts, 80% of which was from cesium
- Reactor vessel highly radioactive during shutdown, so reactor vessel pit is inaccessible

BN-600 and BN-350 Experience

Some knowledge gained about pool vs. loop

- Pool vessel supported from top (all designs except BN-600)
- BN-600 vessel is supported from bottom. Allows thinner vessel wall. Bottom support is better regarding seismic resistance
- Pool has advantage that most components operate at about the same temperature

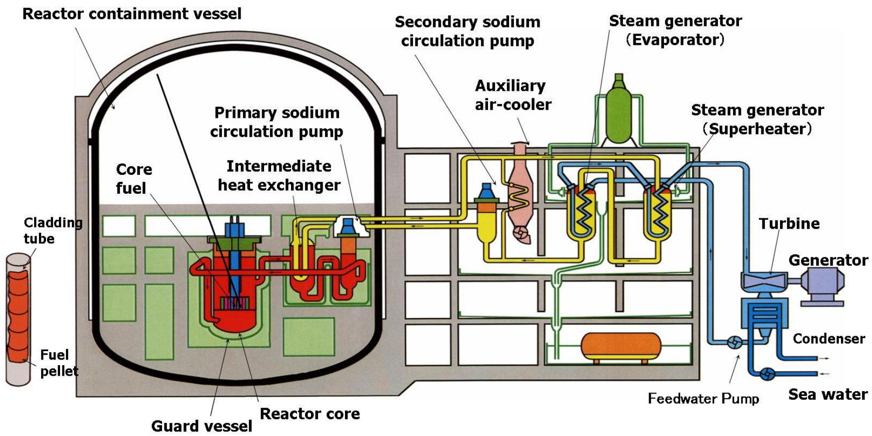
BN-600 and BN-350 Experience (cont.)

- Pool has inner vessel separating hot and cold sodium
- Pool has large sodium inventory and more thermal inertia
- Loop, in large systems, has large pipes and long pipe runs—more susceptible to leaks
- Components grow at different amounts due to thermal expansions—major issue with loop designs
- Operator experience indicates preference for pool-type configuration

MONJU History (Japan)

- Location: Tsuruga, Fukui Prefecture, Japan
- Construction: began May 10, 1986
- First criticality: April 1994
- Sodium-cooled, MOX-fueled loop reactor with three loops
- Power: 714 MW(t), 289 MW(e)
- Shut down: December 1995 after major sodium leak
- Restart: May 6, 2010
- Dropped fuel handling machine onto reactor August 2010

MONJU Plant Summary



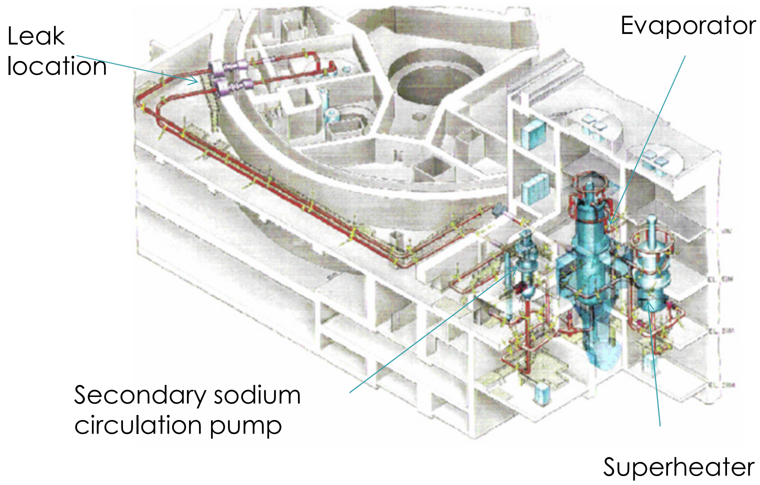
M. Lind, H. Yoshikawa, S. Jorgensen, M. Yang, K. Tamayama, K. Okusa, “Modeling Operating Modes for the Monju Nuclear Power Plant,” January 2012.

Figure reference: M. Lind, H. Yoshikawa, S. Jorgensen, M. Yang, K. Tamayama, K. Okusa, “Modeling Operating Modes for the Monju Nuclear Power Plant,” January 2012.

MONJU Sodium Leak and Fire

- December 8, 1995: Vibration caused failure of thermowell in secondary sodium pipe 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
- PRDC accused of covering up accident

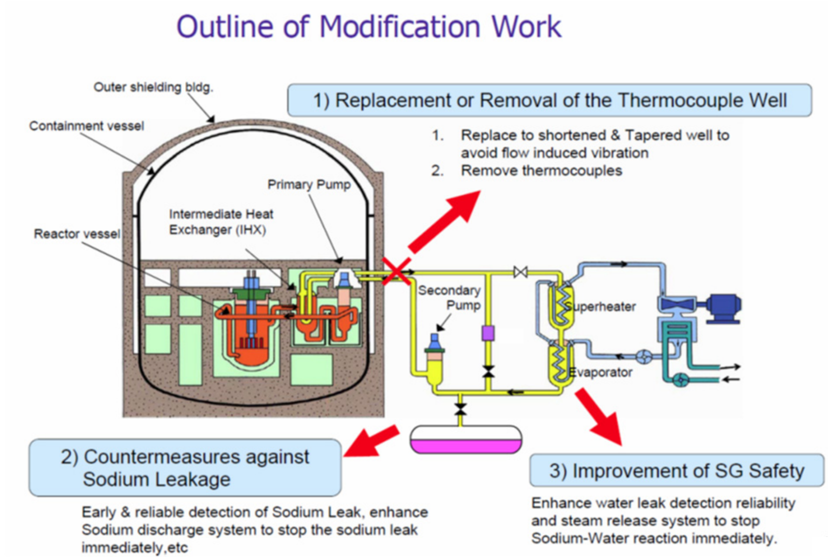
MONJU Sodium Leak



MONJU Fire Sequence

- December 8, 1995
  - 19:47—Accident origin: a fire alarm alerted
  - 19:48—A sodium leak detector alerted. Smoke confirmation around the leakage area
  - 20:00—Normal reactor shut down procedure was taken against small sodium leakage
  - 20:50—Additional fire alerts in rapid succession. Increased white fume observed
  - 21:20—Manual reactor trip
  - 22:55—Sodium drain began from secondary C-loop
  - 23:13—Automatic ventilation stop at the steam generator room
- December 9, 1995
  - 00:15—Sodium drain completion

MONJU Post-Fire Modification



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MONJU Machine Drop Accident

- **“In-Vessel Transfer Machine” falling accident**
- On August 26, 2010, 1 3.3-tonne “In-Vessel Transfer Machine” fell into the reactor vessel when being removed after a scheduled fuel replacement operation
- On October 13, 2010, an unsuccessful attempt was made to retrieve the machine. The JAEA tried to recover the device used in fuel exchange but failed as it had become misshapen and prevented its retrieval through the upper lid.
- The JAEA began preparatory engineering work on May 24, 2011, to set up equipment to be used to retrieve the IVTM that fell inside the vessel
- The fallen device was successfully retrieved from the reactor vessel on June 23, 2011

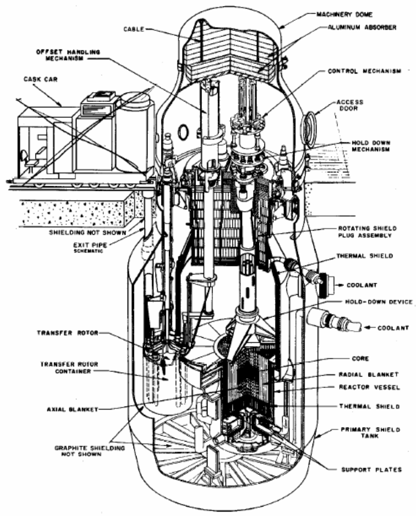
MONJU Restart in May 2010

- Operators started withdrawing control rods on May 6, 2010, marking the restart of the plant
- Tests were to continue until 2013
- Shutdown by Japanese government 2017

Fermi - 1

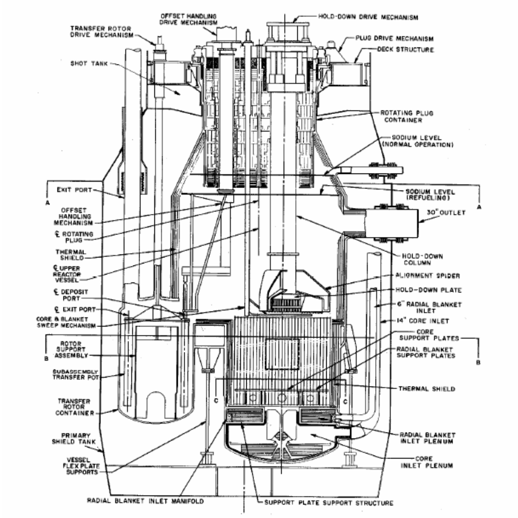
Fermi-1 Vessel and Containment Dome

Fermi-1 Primary System



PERSPECTIVE VIEW OF REACTOR

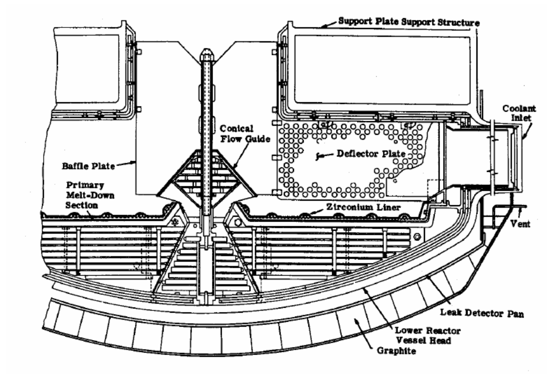
Fermi-1 Reactor Vessel



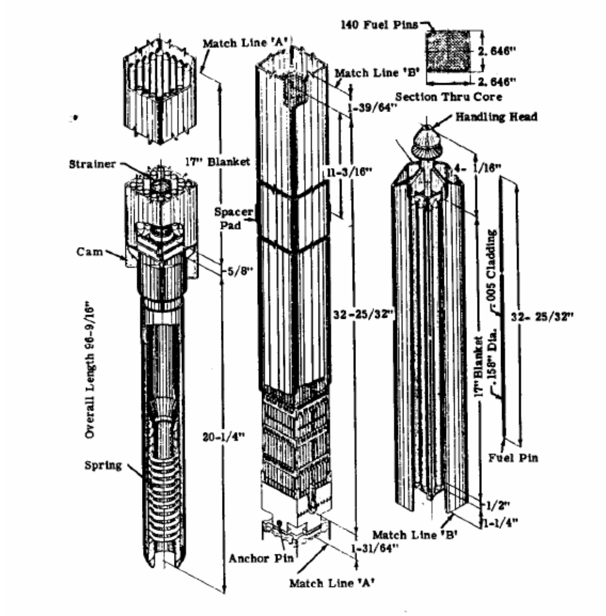
REACTOR VESSEL (ELEVATION)

Fermi-1 Inlet Area with Core Catcher

Fermi-1 Core Inlet Plenum and Melt-Down Section



Fermi-1 Fuel Assembly



Fermi-1 Fuel Melting Accident

- Prior to October 5, high fuel assembly outlet thermocouple readings had been observed during low-power operations
  - Assemblies with abnormal temperature readings were relocated to positions under different thermocouples
  - The location(s) of the high temperature readings changed on each start-up but not in correlation with the assembly movements
- On October 5, during a power ascension at 34 MW(t), building radiation alarms sounded, indicating fuel damage
  - The reactor had previously operated at 100 MW(t) without problems
- 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), in the inlet plenum design (coolant flow distribution and assurance of assembly supply), in the instrumentation design (detection by multiple thermocouples, delayed neutron detectors, gas tags), and in fuel handling equipment design (casks)
- Internationally, in some countries, the fuel assembly blockage scenario has become a design basis accident

Fermi-1 References

- *Technical Information and Hazards Summary Report*, Enrico-Fermi Atomic Power Plant, Power Reactor Development Company, Detroit, MI, June 1961
- W. J. McCarthy and W. H. Jens, "A Review of the Fermi Reactor Fuel Damage Incident and A Preliminary Assessment of Its Significance to the Design and Operation of Sodium-Cooled Fast Reactors," *Proceedings of the International Conference on the Safety of Fast Reactors, Aix-en-Provence, September 19–22, 1967,* pp. Va-1-1 Va-1-23, Commissariat a l'Energie Atomique, 1967

Canadian Nuclear Safety Commission
Information Seminar on Sodium Fast Reactors

Presented by:
George Flanagan, ORNL
Tanju Sofu, ANL

December 10-11, 2018

Module 10 – SFR Testing Experience

ORNL is managed by UT-Battelle, LLC
for the US Department of Energy

U.S. SFR Experience

- SFRs have been extensively studied and operated by DOE and its predecessor, AEC
  - Experience with EBR-I, EBR-II, FFTF, and CRBR project
- After CRBR project was canceled, DOE continued on with Advanced Liquid Metal Reactor (ALMR) and Integral Fast Reactor (IFR) programs
  - Emphasis on a pool-type SFRs with metal alloy fuel to address regulatory concerns related to severe accidents
- Under the ALMR program, PRISM (GE) and SAFR (Rockwell/WEC) concepts submitted their Preliminary Safety Information Document to NRC in 1986
- ALMR and IFR programs continued safety testing program at EBR-II FFTF and TREAT reactors

1932: Chadwick's discovery of neutrons; 1942: CP-1; 1951: First nuclear electricity in EBR-I; 1964: First criticality in EBR-II (all in quick succession)

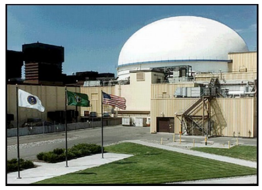
Severe accident concerns impeded CRBR licensing. Even though U.S. NRC Atomic Safety and Licensing Board (ASLB) eventually excluded HCDAs from the licensing basis, it stated that "probability of core melt and disruptive accidents must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum". That was one of the main motivations of the ALMR and IFR programs.

U.S. SFR Experience (cont.)

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

TREAT

- Transient Reactor Test Facility designed for the transient testing of nuclear fuels and materials under off-normal and accident conditions
- 35 years of operation for over 800 testing for numerous fuel forms

Bullets show the topics for performance assessments.

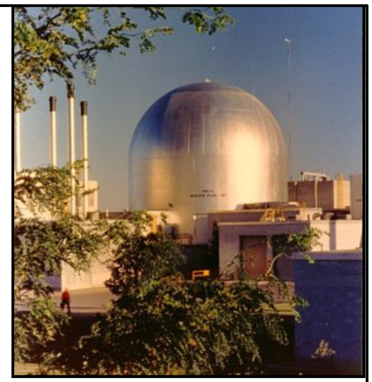
U.S. SFR Safety Testing Program

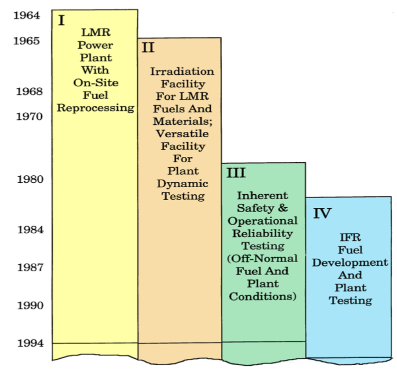
- In the U.S., past SFR R&D programs focused on development and demonstration by testing of the concepts with inherent and passive safety features that lead to no serious consequences even during unprotected (without scram) accidents
  - EBR-II Shutdown Heat Removal Tests
    - Includes landmark EBR-II inherent safety demonstration test
  - FFTF passive safety tests without scram
  - Transient fuel behavior tests:
    - Mild transients on whole fuel assemblies in EBR-II and FFTF
    - Pin disruptive tests on one or a few whole fuel pins in TREAT
    - Lab-tests on segments of fuel pins in the Fuel Behavior Test Apparatus (FBTA) and on whole fuel pins in the Whole-Pin Furnace (WPF) facility

EBR-II

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 structures and components in the event of loss of active cooling
- 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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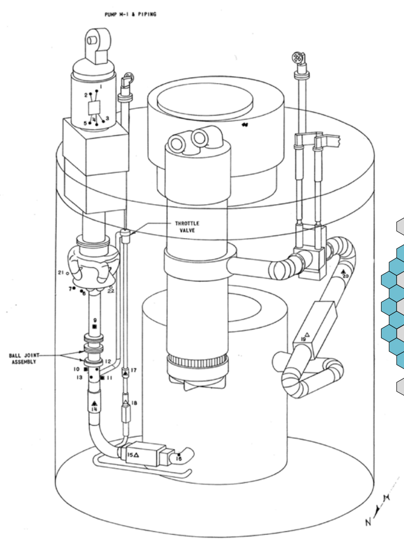
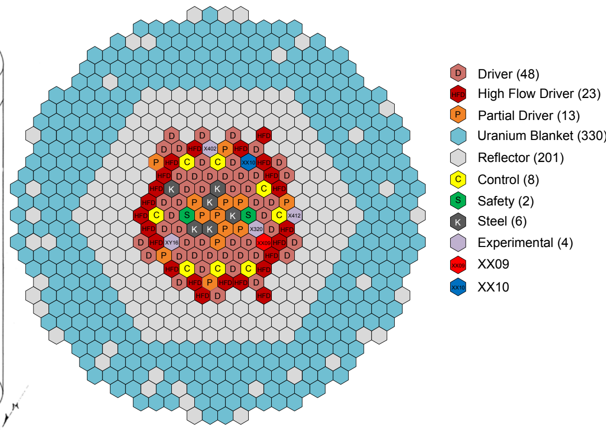
PHTS – primary heat transport system

EBR-II (cont.)
PHTS layout and reactor core

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





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EBR-II Transient Tests

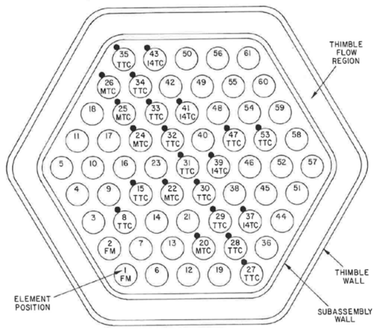
- EBR-II testing program, initially aimed at verifying safe and reliable operation of EBR-II, evolved into an experimental program to support design and performance assessment of ALMRs with special emphasis on inherent safety
  - Testing program started from mild steady-state natural circulation tests and culminated with unprotected transients (no scram)
  - These collective efforts were aimed at understanding EBR-II response to a wide variety of upset conditions and validating computer codes for application to new plant designs
- Initial emphasis was on phenomena for reactor and primary heat transport system. Later shift in focus to whole-plant dynamic behavior
  - Plant instrumentation was upgraded so that flow rates and temperatures in the primary, secondary, and steam systems could be measured and collected by a data acquisition system
  - Additional control system functions were added to facilitate the conduct of whole-plant dynamic testing

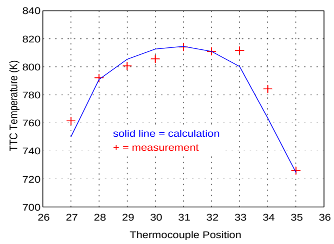
EBR-II Transient Tests (cont.)

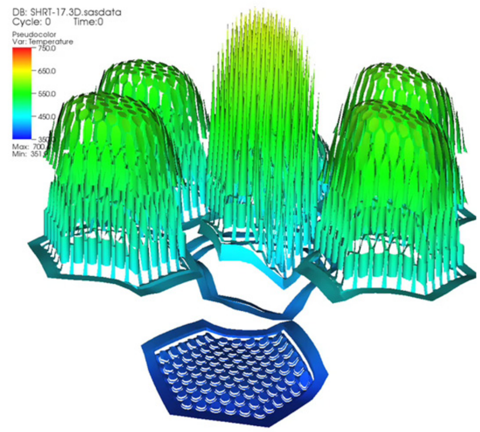
- Over 100 transient tests conducted during 1984-1986 period can be arranged into several categories:
  - Loss of flow with scram to natural circulation
  - Scram with delayed LOF to natural circulation
  - Loss of flow without reactor scram at different levels of severity
    - Includes landmark inherent safety demonstration test (station blackout without scram)
  - Reactivity feedback characterization
  - Dynamic frequency response tests
    - Reactivity perturbation and rod-drop tests
    - Multi-frequency control rod and secondary flow oscillations
  - Loss-of-heat-sink tests (with or without scram)
  - Steam drum pressure reduction
  - Plant inherent control tests (to demonstrate "load-following" features of the reactor)

Emphasize that unprotected tests were BDBAs.

EBR-II Loss of Flow Test







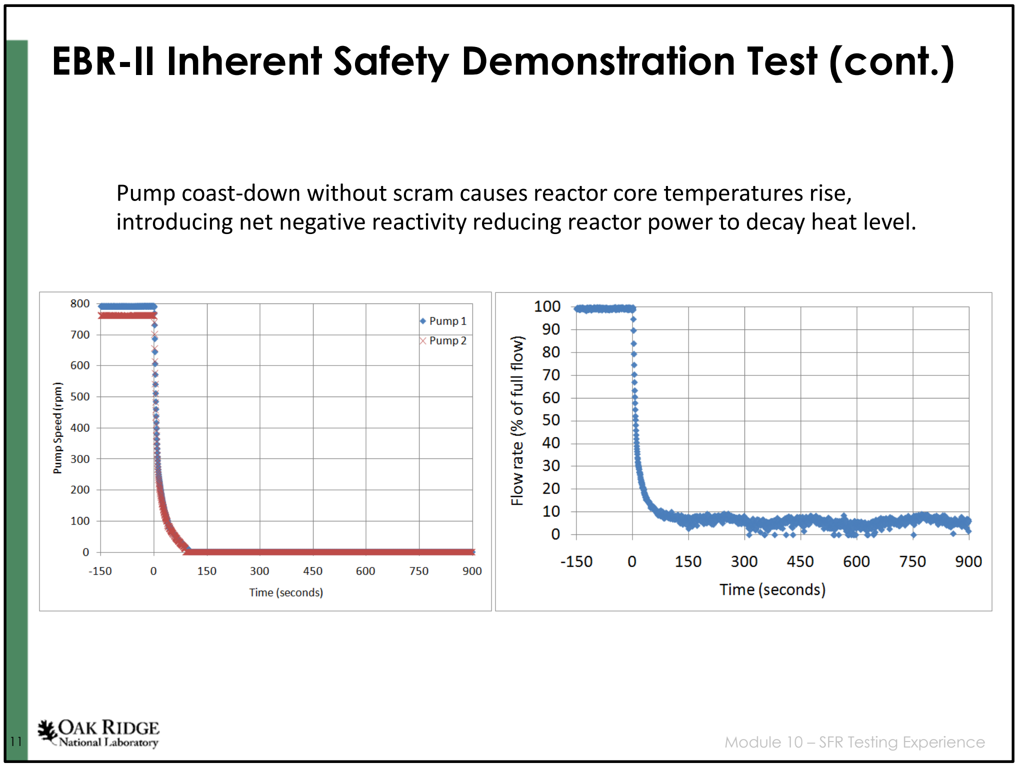
Mesh represents sodium temp. distribution at top of the XX09 and surrounding assemblies (elevation and color represents temperature)

EBR-II Inherent Safety Demonstration Test

- Potential of SFRs to survive severe accident initiators with no core damage was demonstrated during Landmark EBR-II Inherent Safety Demonstration Test
- SHRT-45R: Station blackout without scram on April 3, 1986
  - From full power and full flow
  - Simultaneous trip of all sodium pumps
    - Only auxiliary sodium coolant pump on battery power continued to operate
  - Reactor control system was manipulated to avoid scram
  - Decay heat removal system continued to operate at its rated capacity as a passive device

The accident is assumed to be initiated by a total loss of offsite power, causing the electrical power to be lost to all primary pumps, intermediate loop pumps, and feedwater pumps. There is also a total failure to scram the reactor so that the reactor power changes only in response to the reactivity feedbacks.

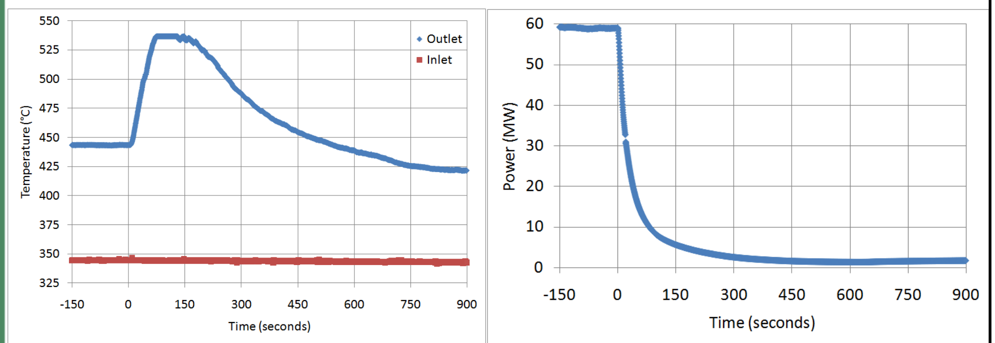
The primary and intermediate pumps coast down according to their internal characteristics only. The loss of the feedwater pumps causes the loss of heat rejection to water side at the steam generators. The decay heat removal system continues to operate at rated total capacity.



The accident is assumed to be initiated by a total loss of offsite power, causing the electrical power to be lost to all primary pumps, intermediate loop pumps, and feedwater pumps. There is also a total failure to scram the reactor so that the reactor power changes only in response to the reactivity feedbacks.

The primary and intermediate pumps coast down according to their internal characteristics only. The loss of the feedwater pumps causes the loss of heat rejection to water side at the steam generators. The decay heat removal system continues to operate at rated total capacity and small auxiliary pump on battery power remains operational.

EBR-II Inherent Safety Demonstration Test (cont.)



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The change in core flow is accompanied by a change in power, with the flow dropping faster than the power. The power-to-flow ratio reaches 2.0 at 12 seconds after the start of the transient, and peaks at 2.3 at about 30 seconds.

Consequently, the peak coolant temperature reaches a maximum at about 70 seconds. The coolant temperature rises to 533°C at this point while the coolant saturation temperature is > 900°C. The minimum margin to coolant boiling of > 366°C.

Based on the analysis of this test, after the start of the transient, the negative reactivity contributions from core expansion, Doppler, and to the lesser degree, core axial expansion and control rod driveline expansion makes the core subcritical and net reactivity remains negative for the remainder of the transient.

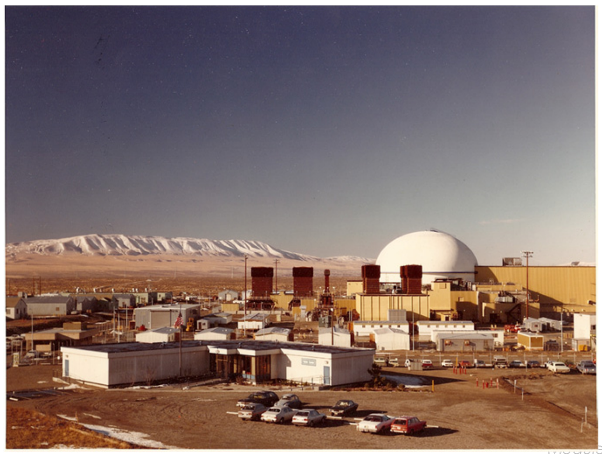
All reactivity feedbacks tend to return to zero as power-to-flow ratio returns to 1.0 with slightly negative net reactivity. The temperatures stabilize at about nominal core-outlet coolant temperatures in the long term.

Measured and predicted peak XX09 instrumented assembly temperatures suggest that the eutectic limit was exceeded for ~ 90 seconds in the hottest driver assembly before natural convection fully compensated for the lack of pumping power (DAS records suggest no fuel failures)

FFTF

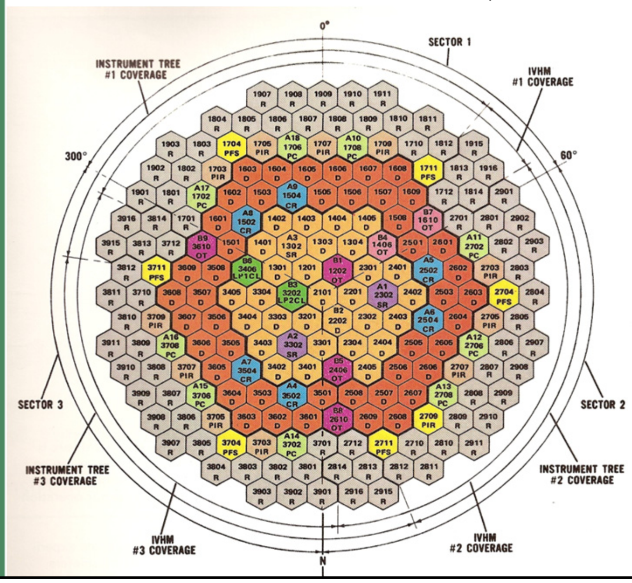
Operated at DOE’s Hanford site as a test facility

- 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}=360C and T_{out}=527C

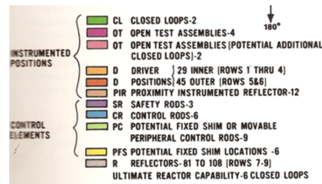


FFTF (cont.)

- 217 pins/assembly
- ~150 fuel pellets per pin in 316 SS cladding
- Average discharge burnup: 45 MWd/kg
- Limiting peak burnup: 80 MWd/kg
- Power density: 0.39 MW/lt



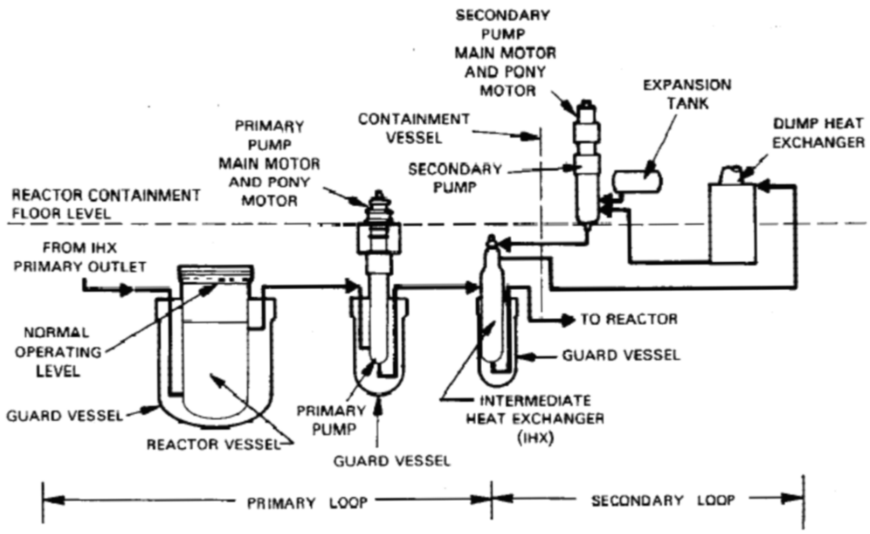
- Rows 1 through 4 enriched to approximately 20% heavy metal fissile content



- Rows 5 and 6 enriched to about 25% fissile
- Rows 7, 8 and 9 contained radial reflectors

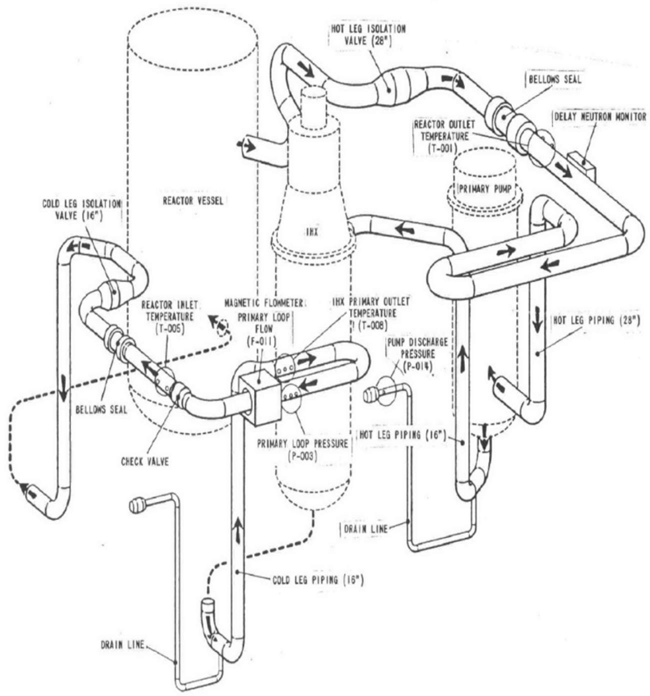
FFTF (cont.):

Heat Transport Systems



FFTF (cont.):

Primary Loop Schematic



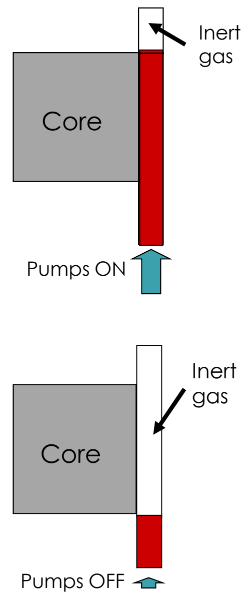
Oak Ridge National Laboratory

FFTF Transient Tests

- In late 1980's, a series of passive safety tests were conducted:
  - to demonstrate the safety margins of SFR designs
  - to provide data for validation of computational models
- Of particular interest was a series of Loss of Flow Without Scram tests from power levels up to 50%
  - Due to large Doppler feedback and stored heat, oxide-fueled SFRs have smaller margins to coolant boiling and large-scale fuel failures than metal-fueled SFRs
  - To overcome this deficiency, a reactor self-shutdown device called the Gas Expansion Module (GEM) was introduced into the core design to mitigate the consequences of an unprotected (without scram) loss of flow event

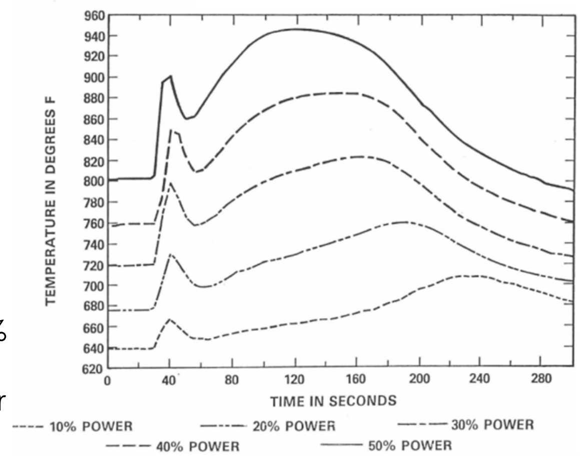
FFTF Transient Tests: Gas Expansion Modules

- GEM is essentially an empty assembly, sealed at the top but open at the bottom, fitted with FFTF core compatible hardware at both ends to permit insertion into the inner row of the reactor radial reflector
  - During normal operation, sodium level in the device rises until the core inlet pressure equals the compressed argon gas pressure, about 12-16" above the active core height
- It provides a mechanism for automatic removal of reactivity if primary flow is lost
  - A passive protective feature against a reduction in inlet plenum pressure caused by a loss of primary flow
  - The loss of pressure causes the trapped argon gas to expand, forcing the sodium in the internal volume back down below the core level
  - Displacement of sodium increases the neutron leakage from the core, introduced -$1.50 reactivity



FFTF Loss-of-Flow Without Scram Tests

- First series of ULOF tests were conducted with the primary pump pony motors on throughout the transient so that the minimum flow reached in each test was ~9%
- Peak coolant temperature for the test series was approximately 493 C
- ULOF tests were then repeated with the same initial conditions, except the primary pony motors were turned off
- A direct transition to natural circulation flow in the primary system was observed
- Tests were repeated from 10, 20, 30, 40, 45, and 50% power
- The peak temperature for this series was 509 C



Transient Fuel Behavior Tests

- Database developed from various types of tests is considerable and it provides a significant basis for the current understanding of the transient behavior of fast reactor fuels for a range of off-normal conditions
- Experiments performed with metallic fuels focused on the key issues:
  - transient-induced changes in fuel morphology
  - fuel-cladding chemical interactions (FCCI)
  - fission-gas release behavior
  - cladding failure margins
  - fuel motions before and after cladding breach
- Fuel Behavior Test Apparatus was used for heating short (~1 cm) segments of irradiated fuel pins
  - Test temperatures ranged from 670-850°C, test duration ranged from 5 minutes to 4 hours
- Whole Pin Furnace tests on whole (intact) irradiated fuel pins
  - Peak test temperatures varied from 650 to 820°C and test duration ranged from few minutes to 36 hours

Slower heating capability in the WPF and FBTA allowed for the contribution of cladding creep effects on cladding failure

Duration of the TREAT tests which were performed was too short to allow creep effects to become significant.

In-pile TREAT tests allowed demonstration of fission-gas release effects on pre-failure fuel motion, as well as post-failure fuel expulsion from the cladding and molten-fuel/coolant dynamics in the coolant channel.

TREAT Metallic Fuel Tests

- Transient overpower tests with both oxide and metal fuel providing valuable info on failure modes/location/timing
  - Estimates for margin to cladding failure and insight into accident progression
- Seven tests (M1 through M7) investigated the response of a variety of metallic fuel designs to overpower transients
  - Tests M1-M4 tested U-5Fs fuel in 316-SS cladding
  - Tests M5-M7 tested U-Zr and U-Pu-Zr fuels in D9 and HT9 clad
  - Designed to be sufficiently severe to cause fuel damage
  - In-pin fuel motions were made with a neutron hodoscope
- Metal fuel tests at TREAT demonstrated that:
  - Metal-alloy fuel slugs axially expand in the cladding tube during overpower transients before the fuel melting and cladding breach
  - Combined with in-pin and/or ex-pin molten fuel motion, this contributes termination of accidents before propagation of fuel failures

No cliff edge effects!

TREAT Metallic Fuel Tests (cont.)

- In test M1, open-ended segments of an irradiated fuel were tested in inert gas to determine if extensive solid-fuel axial extrusion occurs
- Tests M2 to M7 subjected two or three intact pins to an overpower transient having a power rise with 8 s period, in flowing sodium loop
  - Nominal conditions were for 40 kW/m axial peak, 360°C inlet temperature, and 150°C coolant temperature rise along the core
  - Peak transient power was ~ 4 x Nominal
  - Nine U-5Fs fuel pins in 316-SS cladding were tested in M2 through M4
  - Five pins of D9-clad U-19Pu-10Zr fuel at peak burnups between 1 and 10 at.% were tested in M5 through M7
  - One U-10Zr fuel pin, having HT-9 cladding and 3 at.% peak burnup, was also tested in M7
- These transient overpower tests aimed for measurements of margin to failure and pre-failure elongation of metal fuel
  - Designed to be sufficiently severe to cause extensive fuel damage
  - In-pin fuel motions were detected with a neutron hodoscope

No cliff edge effects!

TREAT Metallic Fuel Tests (cont.)

- Tests M5-M6 were the first transient overpower tests of margin to failure and pre-failure elongation of the reference ternary (U-Pu-Zr) alloy fuel of the IFR concept with D9 cladding
- Test M7 extended the results to a higher burnup (9.8 at.%) ternary fuel and initiated testing of binary (U-Zr) fuel with HT9 cladding
- Available experimental information include flow tube temperatures, cladding failure time and location, fuel axial expansion, and melt fraction

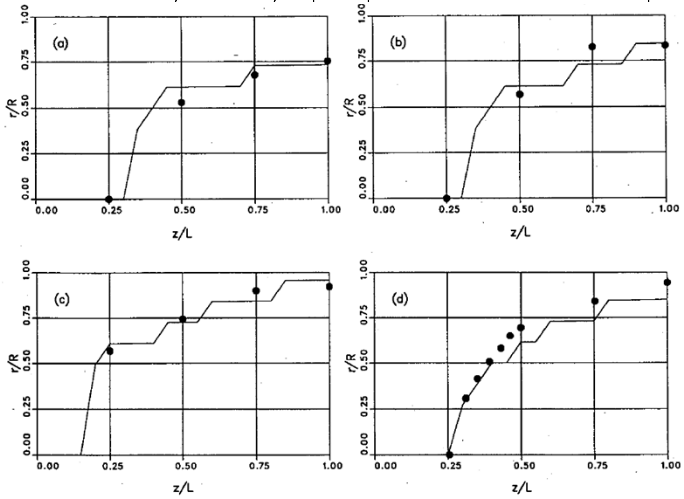
TREAT Test		Burnup (at.%)	Measured Failure	
			Time (s)	Location
M5F1	Pin 1	0.8	No Failure	
	Pin 2	1.9	No Failure	
M5F2	Pin 1	0.8	No Failure	
	Pin 2	1.9	No Failure	
M6	Pin 1	1.9	No Failure	
	Pin 2	5.3	13.24	Top of fuel column
M7	Pin 1	9.8	17.72	Top of fuel column
	Pin 2	2.9	No Failure	

No cliff edge effects!

TREAT Metallic Fuel Tests (cont.)

- PIE results show near complete melting (radially) of the unfailed pins near the upper half of the fuel slug.
- The moment the molten fuel comes into contact with the clad, however, it leads to rapid eutectic penetration with immediate cladding failure

Molten fuel cavity boundary at peak power for unfailed metal fuel pins



. Comparison of measured and calculated molten fuel cavity boundary at peak power for the TREAT test: (a) M5—Pin 1, (b) M5—Pin 2, (c) M6—Pin 1, and (d) M7—Pin 2. Solid lines are the calculated results, and data points are the measurements.

No cliff edge effects!

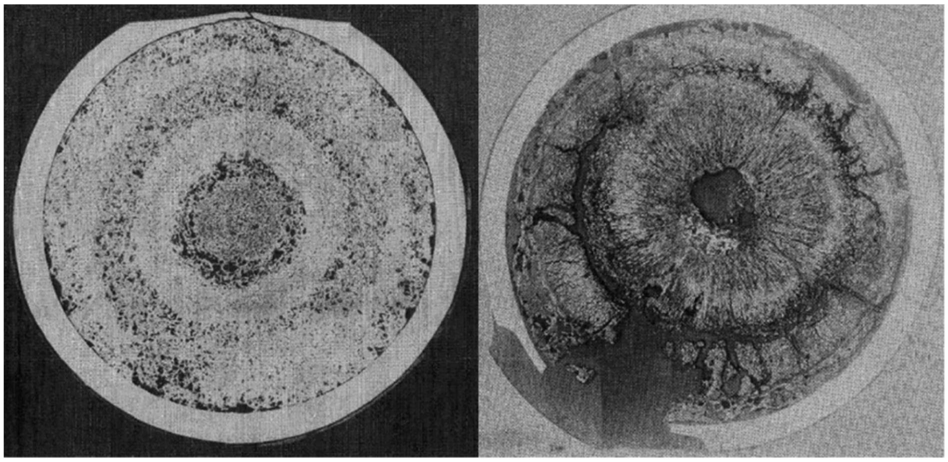
Tests at Fuel Behavior Test Apparatus (FBTA)

- A computer-controlled radiant furnace capable of heating short (about 1 cm long) segments of irradiated fuel pins
  - >50 fuel-cladding compatibility tests for irradiated fuel pin segments (U-10Zr or U-Pu-Zr fuels in 316SS, D9, and HT9 claddings)
  - Segments were cut at various axial locations between x/L = 0.20 and 0.93 of the fuel column from fuel pins having 3 to 17 at.% peak burnup
  - Test temperatures ranged from 670-850°C, test duration ranged from 5 minutes to 4 hours
  - Yielded information regarding
    - fuel melting
    - cladding penetration by low-melting-point liquid phases or by matrix dissolution
    - time variation of the cladding penetration rate
    - reaction zone composition
  - No liquid-phase fuel-cladding interaction was observed in any of the tests performed below ~700-725°C
  - Above that range, liquid-phase grain boundary penetration was found to occur predominantly in high-burnup fuel

Whole Pin Furnace Tests

- WPF was also an in-cell, computer-controlled radiant furnace capable of accommodating intact fuel pins
  - Flat radial temperature distribution it provided was considered representative of profile in LOF accidents at decay heat levels
  - Six metal fuel tests were performed with U-Zr and U-Pu-Zr pins, all in HT9 cladding in a burnup range of 2.2 to 11.4 at.%.
  - Peak test temperatures varied from 650 to 820°C and test duration ranged from few minutes to 36 hours
  - Differences in fuel pins and test conditions affected the relative roles of cladding thinning by the formation of low-melting-point "eutectic" phases at the fuel-cladding interface and cladding creep strain due to pressure in pin plenum
  - Tests provided data for comparison with results of fuel behavior models that described modes, mechanisms, and thresholds of cladding failure

EBR-II Run-Beyond-Cladding-Breach Tests

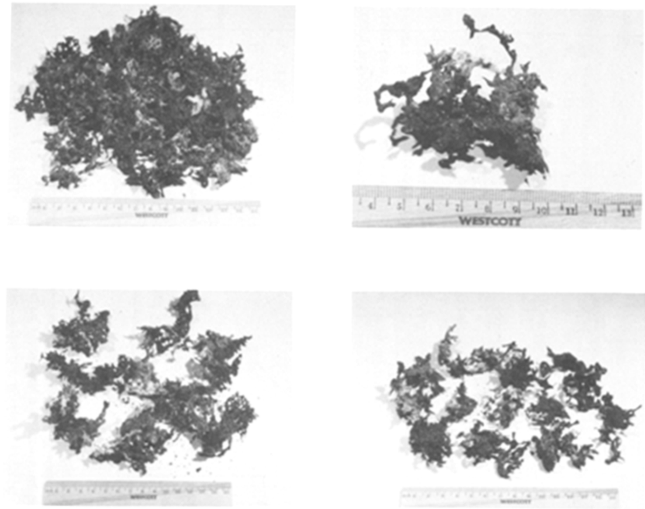


Metal Fuel (12% burnup) RBCB Test **Oxide Fuel (9% burnup) RBCB Test**

T. Sofu, "A review of inherent safety characteristics of metal alloy sodium-cooled fast reactor fuel against postulated accidents," *Nuclear Engineering & Technology*, Volume 47, Issue 3, pages 227-239 (August 2015).

Molten Fuel Quenching Tests

- Examined fragmentation to verify core-debris coolability and in-vessel retention



Debris bed formed from breakup of 3-kg U melt jet (400°C superheat, 25 mm dia., 2 m/sec velocity) interacting with a 0.6 m deep sodium pool at 600°C.