Fast Neutron Reactors (FNRs)

1. How do FNRs differ from LWRs?

2. Sodium Fast Reactors

3. BN-800, ASTRID, MYHHRA, PFBR

4. Compact propulsion – LFR / SFR
PWR / SFR Comparison

Pressurised Water Reactor
1. Fuel: uranium dioxide (4 - 5% enriched)
2. Fuel Cladding: Zircaloy (98% zirconium, 2% tin)
3. Moderator: light water
4. Loops: 2 – primary & secondary
5. Coolant: light water – light water
   • Core temperature: 300 – 330°C
   • Operating pressure: 150 atm
   • Rankine (steam) cycle: 33% efficiency

Sodium Fast Reactor (eg. BN-800)
1. Fuel: UO₂ (17-26%) Future: MOX, Nitride, Metallic
2. Fuel Cladding: Stainless steel
3. Moderator: none
4. Loops: 3 – primary, secondary, tertiary
5. Coolant: sodium – sodium – light water
   • Core temperature: 354 – 547°C
   • Operating pressure: ~ 1 atm
   • Superheated steam cycle: ~ 40% efficiency

Source: World-Nuclear.org
How many power SFRs are there?

Of the 447 power reactors in the world, only two are fast reactors.

Source: IAEA
SFRs in Russia

Nuclear Power Plants in Russia

<table>
<thead>
<tr>
<th>TYPE</th>
<th>LOCATION</th>
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<tr>
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<td>1011</td>
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</tbody>
</table>

IAEA, 2017.
Spent fuel.

USA total: ~ 72,000 tons
World total: ~ 270,000 tons

Source: http://www.fepc.or.jp/
Framework for Reactor Design

**Coolant**
- Density & viscosity
- Thermal conductivity & heat capacity.
- Heat Transfer characteristics
- Neutron capture

**Materials**
- Neutron capture
- Heat transfer
- Corrosion rate
- dpa damage

**Fuel selection**
- Enrichment
- UO/MOX/nitride/metallic
- Reprocessing option
- irradiation damage

**Neutronics**
- Burn-up
- Shielding
- Core power cosine
- Fuel shuffling

**Safety Analysis**
- Containment
- Cover gas
- pool (integral) design
- Passive safety system
- intermediate loop
- core melt re-criticality
The Bigger Picture
(not forgetting)

Reactor Design

Proliferation resistance
- Country Signatory of NPT?
- IAEA
- DOE

Financing
- Im/Ex bank
- Electricity market
- Contract for difference

Stakeholder analysis
- Community
- Government / election cycle
- lobby groups

Regulator
- Safeguards
- Country’s technical readiness
- Geopolitically stable?
- NRC model or ONR model?

Success?
Sodium Fast Reactors
Fast and Thermal

Incident neutron data / ENDF/B-VII.1 // MT=18 : (α,fission) total fission / Cross section

Pu 241
Pu 239
U 235
U 233

Pu 240

U 238

Thermal

Fast

Th 232
In the early days, SFRs were tasked to breed fuel as uranium resources were thought to be scarce.

Breeders can extend uranium resources by at least 60x.

Since 2012, the BN-600 has been burning weapons-grade plutonium.

Also, SFRs can burn MA in reprocessed spent fuel.

Breeding Ratios
BR > 1 ‘Breeders’.
BR < 1 ‘Burners’.
BR = 1 ‘Iso-breeders’.

Sodium is one of the most neutron transparent materials. 6 times less neutron absorbing than lead.

SFR: BR ~ 1.3, LFR: BR ~ 1.0

Operate at a higher temperature (550°C) than PWRs
SFR Advantages

- 400 reactor-years of experience (2010)
- 20 SFRs have operated since the 1950s.
- IAEA’s INPRO program involving 22 countries is focused on closing the fuel cycle.
- Most Gen IV reactors are FNRs – SFR, GFR, LFR, MSFR
- SFRs currently more expensive than PWRs.
  - BN-800 capital cost 20% more than VVER-1200*
  - BN-800 operational cost 15% more than VVER*
- France studied replacing half of current PWR fleet with SFRs.
- UK DECC studied scenarios of eventually phasing out PWRs with SFRs.

*Source: world-nuclear.org
Fertile and Fissile Materials

Incident neutron data / ENDF/B-VII.1 // MT=18 : (z.fission) total fission / Cross section

- Pu 241
- Pu 239
- U 235
- U 233
- Pu 240
- U 238
- Th 232

Thermal

Fast

Cross-section (b)

Incident Energy (MeV)
Fast Reactors burn plutonium

- FNR uses plutonium as its basic fuel, its fission neutrons are faster than U235 and better at maintaining the fission process.
- Also, the number of neutrons produced per plutonium-239 fission is 25% more than from uranium.
- U-238 transmutates into Pu-239 and Pu-241 fuel.
- Some U-238 directly burnt by 1 MeV neutrons.
- Breeder reactor core surrounded by fertile blanket of U-238 (depleted uranium)
- PWRs have a conversion ratio of 0.6
  LFRs have a ratio ~1.0
  SFRs have a breeding ratio of 1.3.
Choosing a coolant

Number of 1-m-diam. Pipes
Needed to Transport 1000 MW(t)
with 100°C Rise
in Coolant Temperature

<table>
<thead>
<tr>
<th></th>
<th>Water (PWR)</th>
<th>Sodium (LMR)</th>
<th>Helium</th>
<th>Liquid Salt</th>
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<tbody>
<tr>
<td>Pressure (MPa)</td>
<td>15.5</td>
<td>0.69</td>
<td>7.07</td>
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<tr>
<td>Outlet Temp (°C)</td>
<td>320</td>
<td>540</td>
<td>1000</td>
<td>1000</td>
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<tr>
<td>Coolant Velocity (m/s)</td>
<td>6</td>
<td>6</td>
<td>75</td>
<td>6</td>
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</table>
## Coolant properties

<table>
<thead>
<tr>
<th>Property</th>
<th>Lead&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Sodium&lt;sup&gt;b&lt;/sup&gt;</th>
<th>Salt NaCl–KCl–MgCl&lt;sub&gt;2&lt;/sub&gt; (30–20–50)</th>
<th>S-CO&lt;sub&gt;2&lt;/sub&gt;&lt;sup&gt;b&lt;/sup&gt; at 20 MPa</th>
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<tbody>
<tr>
<td>Boiling point (°C)</td>
<td>1737</td>
<td>892</td>
<td>2500</td>
<td>-78</td>
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<tr>
<td>Melting point (°C)</td>
<td>327.4</td>
<td>97.8</td>
<td>396</td>
<td>-58</td>
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<tr>
<td>Density, (\rho) (kg/m&lt;sup&gt;3&lt;/sup&gt;)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>At 450°C</td>
<td>10.536</td>
<td>842</td>
<td>1910</td>
<td>143.75</td>
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<tr>
<td>At 700°C</td>
<td>10.242</td>
<td>780</td>
<td>1715</td>
<td>104.16</td>
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<tr>
<td>Thermal expansion coefficient, (\alpha) (% Vol/K)</td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>At 450°C</td>
<td>0.011</td>
<td>0.029</td>
<td>0.041</td>
<td>0.152</td>
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<tr>
<td>At 700°C</td>
<td>0.012</td>
<td>0.031</td>
<td>0.045</td>
<td>0.104</td>
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<tr>
<td>Dynamic viscosity, (\mu) (kg/(ms))</td>
<td>2.01 \times 10^{-3}</td>
<td>2.59 \times 10^{-4}</td>
<td>3.47 \times 10^{-3}</td>
<td>3.46 \times 10^{-5}</td>
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<td>At 450°C</td>
<td>1.40 \times 10^{-3}</td>
<td>1.81 \times 10^{-4}</td>
<td>1.18 \times 10^{-3}</td>
<td>4.17 \times 10^{-5}</td>
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<td>Thermal conductivity, (k) (W/(mK))</td>
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<tr>
<td>At 450°C</td>
<td>15.4</td>
<td>66.1</td>
<td>~0.39</td>
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<td>At 700°C</td>
<td>17.7</td>
<td>59.1</td>
<td>~0.39</td>
<td>0.072</td>
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<tr>
<td>Specific heat, (c_p) (J/(kgK))</td>
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<tr>
<td>At 450°C</td>
<td>147</td>
<td>1272</td>
<td>~1004</td>
<td>1227.0</td>
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<tr>
<td>At 700°C</td>
<td>147</td>
<td>1276</td>
<td>~1004</td>
<td>1267.9</td>
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<tr>
<td>Density-specific heat product, (\rho c_p) (J/(cm&lt;sup&gt;3&lt;/sup&gt; K))</td>
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<tr>
<td>At 450°C</td>
<td>1.55</td>
<td>1.07</td>
<td>1.92</td>
<td>0.18</td>
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<tr>
<td>At 700°C</td>
<td>1.51</td>
<td>1.00</td>
<td>1.72</td>
<td>0.13</td>
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<td>Pr number</td>
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<tr>
<td>At 450°C</td>
<td>(\nu = \frac{\mu}{\rho})</td>
<td>viscid diffusion rate</td>
<td>(\frac{\mu}{\rho} = \frac{c_p k}{\kappa})</td>
<td>0.0192</td>
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<tr>
<td>At 700°C</td>
<td>(\alpha = \frac{k}{\rho c_p})</td>
<td>thermal diffusion rate</td>
<td>(\frac{k}{\rho c_p} = \kappa)</td>
<td>0.0116</td>
</tr>
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<td>Macropscopic capture cross-section relative to sodium&lt;sup&gt;c&lt;/sup&gt;</td>
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<td>Moderating power ((\Sigma_t))</td>
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<td>1</td>
<td>22</td>
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<td>Moderating ratio ((\Sigma_t/\Sigma_a))</td>
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<td>370</td>
<td>16</td>
<td>525</td>
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<tr>
<td>Transparency</td>
<td>Opaque</td>
<td>Opaque</td>
<td>Transparent</td>
<td></td>
</tr>
</tbody>
</table>

<sup>a</sup> Kutateladze et al., Liquid Metal Coolants, Atomisdat, Moscow, 1976.

<sup>b</sup> NIST Website: www.nist.org.

<sup>c</sup> Calculated for spectrum specific to each reactor type.

Source: P. Hejzlar et al. / Nuclear Engineering and Design 239 (2009) 2672–2691
SFR Design considerations

- To maintain a high neutron flux, SFR cores are smaller than PWRs. Eg. Dourneay FR 65MW_{th} was the size of a tin can.
- High fissile loading 3 x PWR.
- Higher core power density needs superior coolant. (usually liquid metal)
- Passive control is maintained by a strong negative temperature coefficient (Doppler Broadening).
- Containment must be designed to avoid chemical interaction. (sodium + water = boom!)
- Argon cover gas.
- Must look out for positive reactivity void coefficient.

Source: www.nuclear-power.net
SFR design review

**Coolant**
- Density & viscosity
- Thermal conductivity & heat capacity.
- Heat Transfer characteristics
- Neutron capture

**Neutronics**
- Burn-up
- Shielding
- Core power cosine
- Fuel shuffling

**Safety Analysis**
- Containment
- Cover gas
- pool (integral) design
- Passive safety system
- intermediate loop
- core melt re-criticality

**Materials**
- Neutron capture
- Heat transfer
- Corrosion rate
- dpa damage

**Fuel selection**
- Enrichment
- UO/MOX/nitride/metallic
- Reprocessing option
- irradiation damage
SFR History

- Agreement to designed around sodium.
- Material cross section measurements.
- Zero power critical assembly studies.
- Optimal SFR core layouts.
- Safety analysis etc.

Dependence of coolant velocity & pressure drop with pitch-to-diameter ratio.

Judd, A. (2014)
Experimental SFRs

EBR - 1 & 2 (USA)  
BOR / BN-series (Russia)  
CEFR (China)

Dounreay FR (UK)  
Monju (Japan)  
Phénix (France)
Similarities in SFR core design

BOR-60 loading scheme.
ACR  automatic control rod;
BA   radial blanket steel assemblies;
EFA  experimental fuel assembly;
EMA  experimental material testing assembly;
FA   standard fuel assembly;
MCR  manual control rod;
SR   safety rod

Typical core layout of PFR at Dounreay

Izhutov, A et al. (2015)

Jenson, S et al. (1995)
Commercial SFRs

- **GE Hitachi™ PRISM reactor**
- **ROSATOM™ BN-800**
- **TerraPower™ Travelling Wave Reactor**
- Also **ASTRID**
BN-800 Beloyarsk Unit 4

Power
- 2100 MWt, 864 MWe gross, 789 MWe net
- 39.35% thermal efficiency
- 40 year service life

Core
- Number of hexagonal sub-assemblies (FA): 565
- FA width across flats 96 mm
- 127 Fuel pins per sub-assembly
- Active fuel height: 880 mm
- FA cladding: Austenitic / Ferritic steel
- Sodium Plenum Height: 3.0 m
- Upper boron shielding 1.5 m

Source: Rosatom / world-nuclear.org
BN-800 Beloyarsk Unit 4

Core (cont’)

- Enrichment of 17, 21, 26 % for inner, middle and outer core area
- Size and configuration similar to BN-600
- Burn-up 66 MW-day/kg (max. 100 MW-day/kg) (cf. VVER-1200 burnup is 60 MW-D/kg)
- Refuel after 140 days, (max. swap out 229 FA)
- 14 -17 day outage
- fuel flexibility – U+Pu nitride, MOX, or metal
- BN-800 to test these future fuels for BN-1200
- Change over to MOX in 2019
- 40 years of Pu stockpile to burn at a rate of 3 t / yr

Source: Rosatom / world-nuclear.org
BN-800 Beloyarsk Unit 4

Primary circuit (sodium)
- 3 primary loops, 6 intermediate HX.
- Core: inlet 354°C, outlet 547°C
- Reactor coolant inventory: 910 t of sodium
- Primary loop radioactivity $18 \times 10^{11}$ Bq

Secondary circuit (sodium)
- 3 secondary loops
- SG: inlet 309°C, outlet 505°C
- Secondary Na radioactivity $3.7 \times 10^4$ Bq

Tertiary coolant (steam-water)
- 3 tertiary loops powering one turbine
- Superheated steam pressure 14 MPa
- Superheated steam temperature 490°C

Source: Rosatom / world-nuclear.org
Future fuel types

- **Oxide** \((U, Pu)O_2\)
  - low thermal conductivity
  - low density of fissile atoms
  - does not react with lead or sodium.

- **Nitride** \((U, Pu)N\)
  - high thermal conductivity
  - high density of fissile atoms
  - subject to swelling
  - C-14 contamination from N-14 (n,p).
    (would require N-15 enrichment)

- **Carbide** \((U, Pu)C\)
  - high thermal conductivity
  - high density of fissile atoms
  - high swelling
  - poor compatibility with air and water.

The production of both nitride and carbide fuels is more complex than MOX or metal fuels.

(Source: Hareland / Vattenfall, 2011)
MBIR - Multipurpose fast breeder reactor

**Experimental devices:**

- 1 test loop in the core central area
  \[ D \approx 130 \text{ mm}, \Phi n \approx 5 \cdot 10^{15} \text{ n/(cm}^2\text{-s)} \]

- 2 test loops in the reflector
  \[ D \approx 130 \text{ mm}, \Phi n \approx 1.8 \cdot 10^{15} \text{ n/(cm}^2\text{-s)} \]

Loop channels coolants:
- liquid metals
- gas
- molten salts

**Vertical experimental channels (VEC):**
3 channels in the core
\[ D \approx 60 \text{ mm}, \Phi n \approx (3.2-5) \cdot 10^{15} \text{ 1/(cm}^2\text{-s)} \]

**Material test subassemblies:**
up to 15 channels in the core and reflector
\[ D \approx 60 \text{ mm}, \Phi n \approx 5 \cdot 10^{15} \text{ 1/(cm}^2\text{-s)} \]

**Horizontal experimental channels (HEC):**
up to 6 channels outside reactor vessel
\[ D \approx 150-200 \text{ mm}, \Phi n \approx 0.5 \cdot 10^{14} \text{ 1/(cm}^2\text{-s)} \]

**Inclined experimental channels (IEC):**
up to 7 channels outside reactor vessel
\[ D \approx 150-300 \text{ mm}, \Phi n \approx 0.5 \cdot 10^{14} \text{ 1/(cm}^2\text{-s)} \]

Source: Rosatom
Other Fast Reactors
ASTRID
Advanced Sodium Technological Reactor for Industrial Demonstration

- 600MWe prototype of commercial 1500 MWe SFR
- Commercial SFR to be deployed in 2050 to utilise the half a million tons of DU France has accumulated.
- The MOX fuel including MA will have 25-35% plutonium.
- It will have a secondary sodium loop and a nitrogen tertiary loop to run a Brayton cycle (S-CO₂).
- 4 Heat Exchangers.
- Reactor core will leak neutrons, which reduces fissile breeding but gives a negative reactivity coefficient to improve safety.
- Known as a ‘self-generating’ fast reactor to limit net-Pu production.

Source: world-nuclear.org
MYRRHA
Multipurpose Hybrid Research Reactor for High-tech Applications

- 57 MWt accelerator-driven system (ADS)
- Aim: to study transmutation of long-lived radionuclides in nuclear waste
- 600 MeV, 2.5 mA proton beam delivered to a liquid lead-bismuth (Pb-Bi) spallation target
- Pb-Bi cooled, subcritical fast nuclear core.
- To be built by Belgium's SCK.CEN
- Later to run as a critical fast neutron facility without the spallation source.
- Technology pilot for ALFRED, LFR 125MWe.
- A reduced-power model, Guinevere, became operational at Mol, Belgium since March 2010

Source: SCK·CEN
India is focused on an advanced heavy water thorium cycle, based on converting Th-232 to U-233.

First stage is to breed Pu from natural uranium using PHWRs.

Second state is to use the FNRs to burn the Plut surrounded by a thorium and depleted uranium blanket to breed U-233 and more plut.

Stage three uses Adv. PHWRs to use the reprocessed U-233 and plut as driver fuels but utilising thorium as the main fuel, deriving two thirds of the power from thorium.

A 500 MW PFBR is under construction in Kalpakkam, India (first criticality will be Oct 2017)
Compact propulsion – LFR / SFRs
Alfa Class submarine

- Top speed: 41 knots (!!) or 76 km/hr
- Titanium Hull (crush depth 1,300 m)
- 7 subs constructed 1968 – 1975
- Lead-Bismuth Fast Reactor
- OK-550 reactor output 155 MW<sub>th</sub>
- 1.5 times efficiency from higher coolant temp.
- lighter and smaller than water-cooled reactors
- HEU oxide fuel
- Coolant mp 125°C, bp 1670°C (freezing issues)
- 3 steam loops droves 2 x 30MW turbine
- Sea trials saw one reactor lost due to lead oxide formation causing LOCA
Space propulsion Kilowatt Reactor

- 1-10 kWe, 10 yr design life
- solid cast 93% U-235 7% moly alloy core
- 28.4 kg U235
- negligible fuel burnup means minimal fuel swelling
- Boron carbide control rod
- Annular core. approx 4.5” dia x 9.5” tall
- BeO reflector approx 10” dia.
- Single central control rod
- 8 x passive sodium heat pipe
- 1kWe with Stirling power conversion
- Los Alamos designed the reactor, reflector and shielding.

Source: Gibson et al./ NASA
Further reading

Judd, A. (2014) *An introduction to the Engineering of Fast Nuclear Reactors*

Barré, B. (2016) *Nuclear reactor systems*
Hope you enjoyed the talk.
Extra slides
Gen IV reactors

Sodium Cooled Fast Reactor

Very High Temperature Reactor

Molten Salt Reactor

Supercritical Water Reactor

Gas Cooled Fast Reactor

Lead Cooled Fast Reactor
Defense in depth

There are five consecutive protection barriers preventing radioactivity release from the reactor (Defence-in-depth concept).
Ionising Radiation

- Sleeping next to someone (0.05 µSv)
- Living within 50 miles of a nuclear power plant for a year (0.09 µSv)
- Eating one banana (0.1 µSv)
- Living within 50 miles of a coal power plant for a year (0.2 µSv)

1 person emits in 8 hrs = 0.05 µSv
50 miles fr. Nuc-plant = 0.09 µSv/yr
1 banana = 0.10 µSv
50 miles fr. Coal-plant = 0.30 µSv/yr

Airplane flight from New York to LA (40 µSv)
Background dose received by an average person over one normal day (18 µSv)

Natural K in body = 390 µSv/yr

USA background dose = 4,000 µSv
Australian annual dose = 3,500 µSv

1 Oak tree emits in 8 hrs = 0.05 µSv

Source:
[2] Nuclear Regulatory Commission (USA) and others
Ionising Radiation

Approximate total dose at one station at the north-west edge of the Fukushima exclusion zone (40 mSv)

All doses in green chart combined (~75 mSv)

Dose received by two Fukushima plant workers (~180 mSv)

Dose causing symptoms of radiation poisoning, received in a short time (400 mSv, but varies)

Severe radiation poisoning, in some cases fatal (2000 mSv, 2 Sv)

Dose limit for emergency workers protecting valuable property (100 mSv)

Dose limit for emergency workers in lifesaving operations (250 mSv)

Fatal dose, even with treatment (8 Sv)

Radiation worker one-year dose limit (50 mSv)

Lowest one-year dose clearly linked to increased cancer risk (100 mSv)

Fatal dose = 4,000,000 µSv

Lowest dose without increased risk of developing cancer = 100,000 µSv / yr

Radiation worker = 50,000 µSv / yr

Highest dose received by Fukushima workers = 180,000 µSv

= 10,000 µSv

Source:
[2] Nuclear Regulatory Commission (USA) and others