Outline

1. Overview

2. Illustration # 1: A PWR core design optimization

3. Illustration # 2: The Encapsulated Nuclear Heat Source (ENHS - A nuclear battery type modular fast reactor) design optimization
1. Overview

- **Typical core design objectives:**
  - Maximize core power density
  - Maximize attainable fuel burnup
  - Minimize Cost Of Electricity (COE)
  
  Subjected to design constraints (many)

- **Typical core design constraints:**
  - Maintain criticality
  - Have negative reactivity coefficients
  - Have sufficient reactivity worth of control system for safe shutdown
  - Do not exceed permissible
    - Fuel temperature
    - Clad (inner) temperature
    - Heat flux from clad
    - Coolant outlet temperature (or void fraction)
    - Coolant pressure loss
    - Coolant speed
  
  - Avoid power oscillations
  - Be able to safely accommodate all accidents
Overview (2)

- Make sure that the fuel clad integrity is not impaired due to
  - Gaseous fission products pressure buildup
  - Fuel swelling
  - Fuel-clad interaction
  - Corrosion/erosion by the coolant
  - Fuel rods vibrations
- Assure mechanical integrity of fuel assembly box and of control elements

- Typical core design variables:
  - Fuel enrichment
  - Fuel rod outer diameter, D
  - Lattice pitch, P
  - Fuel “smear factor”
  - Clad thickness
  - Fission gas plenum volume
  - Core diameter
  - Core height (Effective fuel length)
  - Number/location of control elements
  - Fuel management strategy (number and location of fuel batches)
  - Excess reactivity control strategy
  - Coolant inlet temperature
Illustration 1 – A thermal reactor

Optimization of UO$_2$ Fueled PWR Core Design

NERI 02-189
NERI project participants

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NERI project presentation outline

- Objective
- Approach
- Reference PWR and assumptions
- Neutronics
- Thermal hydraulics
- Clad mechanical integrity
- Vibration and wear
- Economics
- Conclusions
NERI study

**Objective:** Optimize the design of UO$_2$ fueled PWR core using same methodology we adopted for the search of optimal hydride fueled PWR cores

**Approach:** Search for that core design that gives the *minimum Cost of Electricity* (COE) of a retrofitted or a newly built PWR
Study Approach (2)

Design variables:
- Outer fuel rod diameter – D
- Pitch-to-diameter ratio – P/D (square lattice)
- Uranium enrichment – 5%, 7.5%, 10%
- Coolant pressure drop across core – 29 psia (reference) or 60 psia
- Type of fuel rod support – grid spacers or wire wraps

Design constraints:
- $K_\infty > 1.05$
- Negative Doppler, moderator temperature and void $\rho$ coef.
- Minimum Departure from Nucleate Boiling Ratio (MDNBR)
- Peak fuel temperature
- Coolant inlet and outlet temperatures (fixed)
- Coolant pressure drop
Design constraints (cont.)

- Clad internal pressure
- Clad strain
- Clad water-side corrosion
- Constraints imposed by 5 vibration and wear mechanisms:
  - Vortex induced vibration
  - Fluid elastic instability
  - Turbulence induced vibration in cross and axial flow
  - Fretting wear
  - Sliding (or adhesive) wear
Reference PWR and Assumptions

South Texas Project Electric Generating Station

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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</thead>
<tbody>
<tr>
<td>Effective core radius</td>
<td>~1.83 m (72”)</td>
</tr>
<tr>
<td>Active fuel length</td>
<td>4.26 m (168”)</td>
</tr>
<tr>
<td>Fission gas plenum length</td>
<td>17.8 cm (7”)</td>
</tr>
<tr>
<td>Clad outer diameter, D</td>
<td>9.5 mm</td>
</tr>
<tr>
<td>Square lattice pitch, P</td>
<td>12.6 mm</td>
</tr>
<tr>
<td>Pitch-to-diameter ratio</td>
<td>1.326</td>
</tr>
<tr>
<td>Number fuel rods per core</td>
<td>50956</td>
</tr>
<tr>
<td>Power level*</td>
<td>3800 MWt</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inlet temperature</td>
<td>294 C</td>
</tr>
<tr>
<td>Core enthalpy rise</td>
<td>204 kJ/kg</td>
</tr>
<tr>
<td>System pressure</td>
<td>2250 psia</td>
</tr>
<tr>
<td>Radial peaking factor</td>
<td>1.65</td>
</tr>
<tr>
<td>Axial peaking factor</td>
<td>1.55</td>
</tr>
<tr>
<td>Average linear heat rate</td>
<td>174 W/cm</td>
</tr>
<tr>
<td>Average specific power</td>
<td>38.38 W/gU</td>
</tr>
<tr>
<td>Average discharge burnup</td>
<td>60 GWD/tHM</td>
</tr>
</tbody>
</table>

* Parameters in Italics are variables of this study. The other parameters are fixed.

<table>
<thead>
<tr>
<th>Outer diameter (mm)</th>
<th>Clad thickness (mm)</th>
<th>Gap thickness (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>D &lt; 7.747</td>
<td>0.508</td>
<td>0.0635</td>
</tr>
<tr>
<td>D &gt; 7.747</td>
<td>0.508 + (D – 7.747) * 0.0362</td>
<td>0.0635 + (D – 7.747) * 0.0108</td>
</tr>
</tbody>
</table>
Neutronics - methodology

- Unit cell analysis using SAS2H sequence of SCALE4.4
  Good agreement with OECD/NEA MOX benchmarks
- Assuming 3 batch fuel management
  - Same average power density (fractional power) in each batch (simplification)
  - Core \( k_{\infty}(\text{BU}) \) is the average of the \( k_{\infty,i}(\text{BU}) \) of all the fuel batches:
    \[
    \frac{1}{k_{\infty}}(\text{BU}) = \sum_i \left[ f_1/k_{\infty}(\text{BU}_{\text{Batch1}}) + f_2/k_{\infty}(\text{BU}_{\text{Batch2}}) + f_3/k_{\infty}(\text{BU}_{\text{Batch3}}) \right];
    f_i \text{ is the fraction of power delivered by the } i^{\text{th}} \text{ batch}
    \]
  - Similar batch averaging is done for all core characteristics
- \( k_{\infty}(\text{EOC}) = 1.05 \)
- Finding boron concentration in water required to bring \( k_{\infty} \) to 1.05 at any point in time
- Calculating Doppler, MTC and reactivity effect due to 5\% and 90\% voiding – as a function of BU
- Amount of \(^{10}\text{B} \) burnable poison (IFBA) - 0.2D/0.95 mg\(^{10}\text{B}/\text{cm} \)
Neutronics – results; 5% enriched U

- Discharge BU (GWD/tHM)
- MTC (pcm/k)

Moderator Temperature Coefficient of reactivity has to be negative.
Thermal hydraulics - methodology

- Using VIPRE-EPRI subchannel analysis
- Verified against VIPRE full-core analysis
- MATLAB scripts to automate VIPRE execution
- W3-L correlation for MDNBR

**Constraints:**

<table>
<thead>
<tr>
<th>Constraint</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>MDNBR</td>
<td>2.17 (2.35 →)</td>
</tr>
<tr>
<td>Peak/average fuel temperature (°C)</td>
<td>1400/2800</td>
</tr>
<tr>
<td>Present/future Core pressure drop (psia)</td>
<td>29/60</td>
</tr>
</tbody>
</table>
Thermal hydraulics - results

Core Power ($x10^6$ kW$_{th}$)

- MDNBR limit
- Pressure drop limit
- Ref. Geom. 3800 MW$_{th}$
- Max. Power: 4245 MW$_{th}$
- Max. Power: 5458 MW$_{th}$

29 psia

60 psia
Clad integrity - methodology

- Using FRAPCON
- Constraints:
  - Clad corrosion, water side: < 0.1 mm, independent of D
  - Clad strain: < 1% in tension
    - External coolant pressure
    - Thermal expansion (fuel and clad)
    - Fuel swelling
    - Gaseous pressure buildup inside fuel rod
  - Clad internal pressure: < 2500 psia
    - Gaseous fission products
    - Helium from $^{10}$B of IFBA
Clad integrity - results

Weak dependence on coolant pressure

29 psia

60 psia
Fuel rod vibration - methodology

- Vibration mechanisms:
  - Fluid elastic instability
  - Vortex shedding lock-in
  - Turbulence induced vibration in cross and axial flow

- Cladding wear mechanisms:
  - Sliding wear
  - Fretting wear
Fuel rod vibration – results: attainable power

Maximum power at 60 psia is 8% lower
Fuel rod vibration – results: cycle length

29 psia

60 psia
Accidents and transient analysis (limited) – methodology & results

- Using VIPRE-EPRI subchannel analysis
- MATLAB scripts to automate VIPRE execution
- Considering:
  - An overpower transient due to control rod bank withdrawal at full power – DNB should not occur
  - A large break LOCA – peak clad temperature < 2200°F
  - A complete LOFA – DNB should not occur
- Findings:

<table>
<thead>
<tr>
<th>Pressure drop</th>
<th>Peak power MW_{th}</th>
<th>D (mm)</th>
<th>P/D</th>
</tr>
</thead>
<tbody>
<tr>
<td>29 psia</td>
<td>4104 (vs. 4245)</td>
<td>7.1</td>
<td>1.47</td>
</tr>
<tr>
<td>60 psia</td>
<td>4990 (vs. 5045)</td>
<td>6.5</td>
<td>1.39</td>
</tr>
</tbody>
</table>
Economics - methodology

- Option 1: “Major backfit” scenario:
  - Replacement of steam generators
  - Upgrade of high pressure turbine
  - Replacement of pressure vessel head and core internals
- Option 2: New PWR: $1800/kWh
- OECD/NEA cost data and costing methodology
- Fuel assembly fabrication cost:
  - 50% of reference proportional to U loading
  - 50% of reference proportional to # of fuel rods per assembly
- Outage time of reference plant is 20 days:
  - 13 days for refueling – fixed
  - 7 days for maintenance – scales with cycle length (same per year)
Economics – costing assumptions

<table>
<thead>
<tr>
<th>Cost Component</th>
<th>Unit Price</th>
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<tbody>
<tr>
<td>Mining/Ore</td>
<td>$41/kg_{HM}</td>
</tr>
<tr>
<td>Conversion</td>
<td>$8/kg_{HM}</td>
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<tr>
<td>Enrichment</td>
<td>$108/kg_{SWU}</td>
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<tr>
<td>Fabrication</td>
<td>$275/kg_{HM}</td>
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<tr>
<td>Spent Fuel Storage</td>
<td>$250/kg_{HM}</td>
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<tr>
<td>Waste Disposal</td>
<td>1 mill/kWh</td>
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<table>
<thead>
<tr>
<th>Mass Loss Fraction</th>
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<tr>
<td>Mining/Ore</td>
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<td>Conversion</td>
<td>0.005</td>
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<tr>
<td>Enrichment</td>
<td>Varies</td>
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<tr>
<td>Fabrication</td>
<td>0.01</td>
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<table>
<thead>
<tr>
<th>Transaction Time</th>
<th>Value</th>
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<tbody>
<tr>
<td>Fuel Fabrication</td>
<td>1 yr</td>
</tr>
<tr>
<td>Uranium Enrichment</td>
<td>1.5 yr</td>
</tr>
<tr>
<td>Uranium Conversion</td>
<td>1.5 yr</td>
</tr>
<tr>
<td>Uranium Ore Purchase</td>
<td>2 yr</td>
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<tr>
<td>Spent Fuel Storage</td>
<td>$-T_C$*</td>
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</table>

* $T_C$ is the cycle length. A negative sign implies that the storage costs need to be referred back in time to the reference date.

<table>
<thead>
<tr>
<th>O&amp;M function</th>
<th>Variable</th>
<th>Cost</th>
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<tbody>
<tr>
<td>Variable</td>
<td>Cost</td>
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</tr>
<tr>
<td>Refueling Outage</td>
<td>$800,000/day</td>
<td></td>
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<tr>
<td>Forced Outage</td>
<td>$100,000/day</td>
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</tr>
<tr>
<td>Replacement</td>
<td>30 mills/kWh</td>
<td></td>
</tr>
<tr>
<td>Fixed</td>
<td></td>
<td></td>
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<tr>
<td>Personnel</td>
<td>$150,000/person-yr</td>
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<tr>
<td>Number Personnel</td>
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<tr>
<td>Refueling Outage</td>
<td>20 days/cycle</td>
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<tr>
<td>Forced Outage</td>
<td>1%</td>
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<tr>
<td>Availability</td>
<td>99%</td>
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## Economics – costing assumptions

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
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<tbody>
<tr>
<td>Thermal Efficiency</td>
<td>0.33</td>
</tr>
<tr>
<td>Number of Batches</td>
<td>3</td>
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<tr>
<td>Plant Life Extension</td>
<td>20 yrs</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Component</th>
<th>Price ($10^6$)</th>
<th>Scaling Factor</th>
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<tbody>
<tr>
<td>Steam Generators</td>
<td>100</td>
<td>0.6</td>
</tr>
<tr>
<td>Vessel Head</td>
<td>25</td>
<td>-</td>
</tr>
<tr>
<td>Core Internals</td>
<td>25</td>
<td>-</td>
</tr>
<tr>
<td>Turbine Generator</td>
<td>338</td>
<td>0.8</td>
</tr>
<tr>
<td>Existing Fuel Value</td>
<td>67</td>
<td>-</td>
</tr>
</tbody>
</table>
Economics – results for the “major backfit” scenario

Minimum COE vs. P/D

- 5% Enrichment
- 7.5% Enrichment
- 10% Enrichment

29 psia

60 psia
## Summary: Lowest COE designs

### Comparison of Optimal and Reference Geometries

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>29 psia</th>
<th>60 psia</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reference</td>
<td>Optimal</td>
</tr>
<tr>
<td>COE (mills/kWe-hr)</td>
<td>17.75</td>
<td>18.0&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Capital</td>
<td>0</td>
<td>1.3&lt;sup&gt;b&lt;/sup&gt;</td>
</tr>
<tr>
<td>Fuel Cycle</td>
<td>7.65</td>
<td>6.53</td>
</tr>
<tr>
<td>O&amp;M</td>
<td>10.1</td>
<td>10.15</td>
</tr>
<tr>
<td>Power (MWth)</td>
<td>3800</td>
<td>3800</td>
</tr>
<tr>
<td>Geometry: D (mm)</td>
<td>9.5</td>
<td>7.13</td>
</tr>
<tr>
<td>P/D</td>
<td>1.326</td>
<td>1.47</td>
</tr>
<tr>
<td>Number of fuel rods</td>
<td>50,956</td>
<td>73,966</td>
</tr>
<tr>
<td>U inventory (kg_HM)</td>
<td>105,170</td>
<td>81,581</td>
</tr>
<tr>
<td>Linear heat rate (kW/ft)</td>
<td>5.30</td>
<td>3.67</td>
</tr>
<tr>
<td>Power density (kW/liter)</td>
<td>99.85</td>
<td>99.85</td>
</tr>
<tr>
<td>Specific power (kW/kg_HM)</td>
<td>36.1</td>
<td>46.6</td>
</tr>
<tr>
<td>Average burnup (MWd/kg)</td>
<td>45.6</td>
<td>56.55</td>
</tr>
<tr>
<td>Cycle length (yrs)</td>
<td>1.22</td>
<td>1.17</td>
</tr>
<tr>
<td>Capacity factor</td>
<td>0.95</td>
<td>0.94</td>
</tr>
<tr>
<td>MDNBR</td>
<td>2.17</td>
<td>2.17</td>
</tr>
<tr>
<td>Peak fuel temp (F)</td>
<td>2700</td>
<td>1906</td>
</tr>
</tbody>
</table>

<sup>a</sup> Optimal for a newly constructed PWR, not a retrofit; <sup>b</sup> Capital cost component (and total COE) for a retrofitted PWR. <sup>c</sup> 3800 MW from the existing reference PWR plus 1129 MW from newly constructed 3800 MW PWR of the reference design.
Economic analysis - Conclusions

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Pressure drop</th>
<th>Reduction in COE</th>
<th>Increase in power</th>
<th>Optimal D (cm)</th>
<th>Optimal P/D</th>
</tr>
</thead>
<tbody>
<tr>
<td>Retrofit</td>
<td>29 psia</td>
<td>0%</td>
<td>0%</td>
<td>0.95&lt;sup&gt;(a)&lt;/sup&gt;</td>
<td>1.326&lt;sup&gt;(a)&lt;/sup&gt;</td>
</tr>
<tr>
<td>New reactor&lt;sup&gt;(b)&lt;/sup&gt;</td>
<td>29 psia</td>
<td>1.07 mills/kWh</td>
<td>0%</td>
<td>0.713</td>
<td>1.47</td>
</tr>
<tr>
<td>Retrofit &lt;sup&gt;(c)&lt;/sup&gt;</td>
<td>60 psia</td>
<td>24%&lt;sup&gt;(c)&lt;/sup&gt;</td>
<td>30%</td>
<td>0.65</td>
<td>1.39</td>
</tr>
</tbody>
</table>

<sup>(a)</sup> Reference geometry

<sup>(b)</sup> Increase in power

<sup>(c)</sup> The retrofit is compared against 3800 MW from the existing reference PWR plus 1129 MW from newly constructed 3800 MW PWR of the reference design.
Conclusions of PWR study

- It appears possible to reduce COE of newly constructed PWR by ~ 1 mills/kWh by going to thinner fuel rods (D=0.71 cm) of a larger P/D ratio (1.47). Reactor power will be the reference power. This new design will also be more suitable for use of MOX fuel.

- If PWR could be designed to have larger coolant pressure drop, core power density could increase and COE could be reduced. For example, for 60 psia core pressure drop, power density could be increased by ~ 30% and COE drop by ~24%. Optimal geometry: D = 6.5 mm (Vs 9.5 mm), P/D = 1.39 (Vs 1.326).

- If 60 psia coolant pressure drop across core could be accommodated, it may be possible to design new PWR for nearly 2.2 GWe per unit using the same pressure vessel dimensions to be used for the 1.7 GWe EPR.
The Encapsulated Nuclear Heat Source Reactor (ENHS)

Historical background

- **1999** – NERI contract for studying the feasibility of the ENHS was awarded to UCB+LLNL+ANL+Westinghouse. Lasted for 3 years (to 9/2002)
- **2000** – KAERI, KAIST, SNU, CRIEPI join our team
- **2001** – ENHS type reactors selected as one of 6 categories of GEN-IV reactors

Illustration 2 – A fast reactor
DOE has adopted the ENHS vision:

“The LFR battery (one of GEN-IV reactor concepts) is

- a small **factory-built** turnkey plant
- operating on a **closed fuel cycle** with
- **very long refueling interval** (15 to 20 years) cassette core or replaceable reactor module
- meet market opportunities for electricity production on small grids, and for developing countries who may not wish to deploy an indigenous fuel cycle infrastructure to support their nuclear energy systems
- The battery system is designed for **distributed generation of electricity** and other energy products, including hydrogen and potable water”

- Also suitable for developed countries and large grids – installing multiple modules in a single plant
Organizations/researchers participated in the ENHS NERI project

E. Greenspan (PI), A. Barak, D. Barnes, M. Milosovic’, D. Saphier, Z. Shayer, H. Shimada and S. Wang
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Lawrence Livermore National Laboratory

Argonne National Laboratory

M. D. Carelli (co-PI), L. Conway and M. Dzodzo
Westinghouse Electric Company Science & Technology Department

Yeong Il Kim (co-PI) and Ser Gi Hong
Korea Atomic Energy Research Institute (KAERI)

Soon Heung Chang (co-PI) and Kwang Gu Lee
Korea Advanced Institute for Science and Technology (KAIST)

Il Soon Hwang (co-PI), Byung Gi Park and Seung Ho Jeong
Seoul National University

I. Kinoshita (co-PI), A. Minato, Y. Nishi and N. Ueda
Central Research Institute of Electric Power Industry (CRIEPI)
ENHS reactor design objectives

- ~50 MW_e/unit (Can be designed from ~20 MW_e to ~100 MW_e)
- Provide energy security with proliferation resistance
- Factory manufactured and fueled
- Simple to operate and maintain
- Autonomous load following capability
- Superb safety
- Economical
- Environmental friendly
Use lead or Pb-Bi as coolants
First used for USSR “Alpha” submarine reactors

- Chemically inert – Do not vigorously react with air or water/steam
  - Can eliminate secondary coolant (possibly SG)
  - Can simplify plant design

- High boiling temperature of 1740°C for Pb, 1670°C for Pb-Bi eutectic
  - Can accommodate positive void coefficient
  - Might enable high coolant outlet temperature → H₂; high efficiency

- Pb-Bi: ~no change in density upon change of phase

- But: Toxic if ingested and generates radioactive ²¹⁰Po
ENHS reactor layout

Schematic vertical cut through the ENHS reactor

- 30m
- 27m
- 8m
- 2m
- 3m
- 2m

Number of Stacks = 4

Cross Section of Stack
Number of Stacks = 4

Seismic isolators

Steam generators

Underground silo

Reactor pool

ENHS module

Replaceable Reactor module

Reactors Vessel Air Cooling System (RVACS)

- factory fueled
- weld-sealed
- underground silo
- >20 years core
- no fueling on site
- Module is replaced
- shipping cask
- no DHRS but RVACS
- no pumps
- no pipes
- no valves
Expanded view of the ENHS Module and pool

- Steam generator
- Secondary coolant
- Primary coolant
- Heat exchanger
- Peripheral control assembly
- Central control assembly
- Core
- Outer structural wall
- Riser
- Upper slots
- Inner structural wall
- IHX channel
- Downcomer
- Lower slots

ENHS Module
Reactivity control systems

- **Central Safety Assembly (CSA)**
  - Made of B$_4$C & tungsten
  - For shutdown purposes only
  - Can be inserted by gravity due to W

- **Peripheral absorbers**
  - Made of B$_4$C & Tungsten
  - 6 segments forming cylindrical shell
  - 4 segments needed for scram
  - Elevation is adjusted once every 1-3 years
  - Maximum withdrawal speed = 1 mm/s (~ $0.006$/s)
  - $\Delta \rho$ (350°C → full power) ~ $0.6$
  - $\Delta \rho$ (burnup) < $1.0$
ENHS reactor design approach

- Nearly no change in $k_{eff}$ over 20 years by designing core to have breeding ratio $= 1 + \varepsilon$

- Full power heat removal by natural circulation

- No mechanical connections between ENHS module and the energy conversion system

- No fuel handling on site (in host country)

- No access to neutrons
ENHS reactor design approach

- Design core to operate for > 20 years (EFPY)
  - Uniform initial composition
  - No blanket elements (for proliferation resistance)
  - Small burnup reactivity swing → Breeding ratio = 1+ ε
  - Small variation of power distribution over life

Design variables:
- Specific power (W/gHM)
- Fuel rod diameter
- Fission gas plenum volume
- Lattice pitch
- Fissile fuel weight %

Design constraints:
- $\Delta k_{eff} < $1 over core life
- Maintain clad mechanical integrity due to
  - Fission gas pressure buildup
  - Radiation damage (for HT-9 clad=200dpa or $4 \times 10^{23}$ n/cm² of > 0.1 MeV n)
ENHS reactor design approach

- Full power heat removal by natural circulation
  - No pumps in Nuclear Island
  - No valves in Nuclear Island\(^{(a)}\)
  - Reactor Vessel Auxiliary Cooling System (RVACS) is the only decay heat removal system needed

Design variables:
- Fuel rod diameter
- Fission gas plenum length
- Lattice pitch
- IHX length
- Number and dimensions of IHX channels
- Riser length

Design constraints:
- Coolant \(\Delta T\) across core and across IHX < ~ 150°C
- Fuel peak temperature
- Clad peak temperature
- Coolant peak temperature

\(^{(a)}\) Except for a pressure relief valve in pool boundary
Selection of fuel rod diameter and fission gas plenum length is guided by

- Need to reduce pressure drop across core to enable removal of 125 MW\textsubscript{th} using natural circulation \(\rightarrow\) fuel rod outer diameter = 1.56 cm (=1.30+2x0.13 cm)
- Need to achieve burnup corresponding to 200 dpa on HT-9 clad without excessive fission gas pressure buildup \(\rightarrow\) FG plenum volume = fuel volume

Fuel rod integrity analysis
\[P/F = \text{Plenum-to-Fuel volume ratio}; \quad D/t = \text{Fuel outer diameter to clad thickness ratio}\]
Selection of lattice pitch and fissile weight % is guided by need to achieve 20 years core life with < 1% $\Delta k_{\text{eff}}$

- Lattice pitch ($P$) and fissile w/o are adjusted to give
  - $k_{\text{eff}}(\text{BOL})=1+\epsilon$
  - $\Delta k_{\text{eff}}(\text{BU})\leq 0$

$\Rightarrow P/D=1.35$ (D=fuel rod O.D.)
$\Rightarrow$ Pu (from LWR used fuel) $\sim 11.3\text{w/o}$

- Inventory of Pu+MA is kept nearly constant
- Can recycle many times
- Just remove FP, add U_depl and re-fabricate
- Core life constrained by radiation damage to clad

Use $p/d=1.35$
$\text{20EFPY}=55\text{GWd/}$
<table>
<thead>
<tr>
<th>Performance Parameter</th>
<th>BOL</th>
<th>EOL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Peak-to-average power density</td>
<td>1.767</td>
<td>1.778</td>
</tr>
<tr>
<td>Peak linear heat rate (W/cm)</td>
<td>177.8</td>
<td>178.7</td>
</tr>
<tr>
<td>Peak-to-average channel power</td>
<td>1.425</td>
<td>1.435</td>
</tr>
<tr>
<td>Peak-to-average power density in hot channel</td>
<td>1.241</td>
<td>1.235</td>
</tr>
<tr>
<td>Peak burnup after 20 (21.9) EFPY (GWD/tHM)</td>
<td>95.2 (104.4*)</td>
<td>50.7 (55.6*)</td>
</tr>
<tr>
<td>Average burnup after 20 (21.9) EFPY (GWD/tHM)</td>
<td>5.681E+14</td>
<td>5.895E+14</td>
</tr>
<tr>
<td>Peak fast (E&gt;0.1MeV) neutron flux (n/cm²-s)</td>
<td>3.646E+23 (4E+23)</td>
<td></td>
</tr>
<tr>
<td>Conversion ratio</td>
<td>1.164</td>
<td>1.019</td>
</tr>
<tr>
<td>Effective delayed neutron fraction</td>
<td>0.004</td>
<td></td>
</tr>
<tr>
<td>Maximum $\Delta k_{\text{eff}}$ with burnup (%)</td>
<td></td>
<td>0.197</td>
</tr>
<tr>
<td>Doppler effect (dk/kk' °C)</td>
<td>-5.78547E-06</td>
<td>-4.55582E-06</td>
</tr>
<tr>
<td>Axial fuel expansion (dk/kk' °C)</td>
<td>-4.42189E-06</td>
<td>-4.39633E-06</td>
</tr>
<tr>
<td>Coolant expansion (dk/kk' °C)</td>
<td>+5.17092E-07</td>
<td>+9.68071E-07</td>
</tr>
<tr>
<td>Grid-plate radial expansion (dk/kk' °C)</td>
<td>-8.97388E-06</td>
<td>-6.80369E-06</td>
</tr>
<tr>
<td>Cold (350°C) to hot (480°C; fuel: 700°C) $\rho$ swing (dk)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Doppler effect</td>
<td>-2.03373E-03</td>
<td>-1.65610E-03</td>
</tr>
<tr>
<td>Axial fuel expansion</td>
<td>-0.56615E-03</td>
<td>-0.57690E-03</td>
</tr>
<tr>
<td>Coolant expansion</td>
<td>+0.04453E-03</td>
<td>+0.14029E-03</td>
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<tr>
<td>Grid-plate expansion</td>
<td>-1.15292E-03</td>
<td>-0.89738E-03</td>
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<tr>
<td>Total</td>
<td>-3.70827E-03</td>
<td>-2.99013E-03</td>
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<tr>
<td>Void reactivity effect (dk)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Voiding inner 1/3 core</td>
<td>+2.54917E-02</td>
<td>+2.65734E-02</td>
</tr>
<tr>
<td>Voiding middle 1/3 core</td>
<td>+0.88586E-02</td>
<td>+0.94506E-02</td>
</tr>
<tr>
<td>Voiding outer 1/3 core</td>
<td>-0.31642E-02</td>
<td>-0.29761E-02</td>
</tr>
<tr>
<td>Voiding whole core</td>
<td>+3.11058E-02</td>
<td>+3.28106E-02</td>
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<tr>
<td>Tungsten annulus reactivity worth (dk)</td>
<td>0.01130</td>
<td>0.01047</td>
</tr>
<tr>
<td>Central absorber reactivity worth (dk)</td>
<td>42</td>
<td></td>
</tr>
<tr>
<td>Total plutonium mass (kg)</td>
<td>1984.7</td>
<td>2065.8</td>
</tr>
<tr>
<td>Fissile-to-total plutonium ratio</td>
<td>0.7360</td>
<td>0.7292</td>
</tr>
</tbody>
</table>
ENHS response to loss of heat sink

- Negative reactivity feedback is provided by increase in core average temperature.
- Peak fuel temperature does not increase!!!
ENHS autonomous load following capability

- Reactivity feedback is provided by increase in core average temperature
- Peak fuel temperature goes down

Response to 30% load reduction

![Graph showing temperature changes over time]
Outstanding safety and reliability
(objective for GEN-IV reactors)

- No pumps, valves & pipes in primary & secondary loops
- Decay heat removal is via RVACS; depending only on physics laws
- Very large heat capacity
- Very large margin to coolant boiling; > 1000°C
- Maximum possible reactivity insertion < $1

- LOCA & LOFA are inconceivable
- No plant component is damaged under LOHS accident
- Autonomous load following capability
- Very little maintenance
- No emergency planning zone outside NPP fence
High proliferation resistance
(objective for GEN-IV reactors)

- No fuel handling hardware on site (host country)
- No access to fuel inside reactor vessel (vessel is weld sealed)
- Impractical to “steal” ENHS modul
- No access to neutrons
- Can add radiation barrier without interfering with handling of the sealed ENHS module
- Energy security is provided without need of host country to construct front-end and back-end fuel cycle facilities; one fuel loading for 20 y
Economic viability

Preliminary study by LLNL*

Assumptions:

- Annual number of units produced – 50
- Interest rate during construction - 8%
- Construction and testing time - 30 months
- Separative Work Units (SWUs) cost - $85
- $\text{U}_3\text{O}_8$ cost - $30$/kg
- Capacity factor - 90%

Annualized costs by cost category

Case: Base

- **nuclear fuel**: 41.2%
- **site labor**: 29.4%
- **factory labor**: 4.5%
- **turbine purchase**: 13.2%
- **material**: 10.0%
- **consumables**: 0.2%
- **factory overhead**: 0.9%
- **shipping**: 0.1%
- **salvage/disposal**: 0.0%
- **site equipment**: 0.1%
- **Interest during constr**: 0.2%

 LLNL economics study findings
### LLNL economics study findings

<table>
<thead>
<tr>
<th>Total unit capital cost, $/kWe</th>
<th>Base Case</th>
<th>Site Labor 2X</th>
<th>Factory Labor 2X</th>
<th>High SWU Price</th>
<th>High U₃O₈ Price</th>
<th>High Interest Rate</th>
<th>Lower Capacity Factor</th>
<th>Longer Constr Period</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>2,000</td>
<td>2,012</td>
<td>2,007</td>
<td>2,130</td>
<td>2,233</td>
<td>2,044</td>
<td>2,000</td>
<td>2,610</td>
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<tr>
<td>O&amp;M costs, $M/yr</td>
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<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
<td>2.19</td>
<td>4.35</td>
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<td>2.19</td>
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<td>2.19</td>
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<tr>
<td>Busbar costs, ¢/kWh</td>
<td>Capital</td>
<td>1.00</td>
<td>1.02</td>
<td>1.01</td>
<td>1.01</td>
<td>1.02</td>
<td>1.27</td>
<td>1.13</td>
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<tr>
<td></td>
<td>O&amp;M</td>
<td>0.56</td>
<td>1.10</td>
<td>0.56</td>
<td>0.56</td>
<td>0.56</td>
<td>0.63</td>
<td>0.56</td>
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<tr>
<td></td>
<td>Fuel</td>
<td>1.40</td>
<td>1.40</td>
<td>1.40</td>
<td>1.56</td>
<td>1.68</td>
<td>1.62</td>
<td>1.58</td>
</tr>
<tr>
<td>Total</td>
<td>2.96</td>
<td>3.52</td>
<td>2.97</td>
<td>3.13</td>
<td>3.26</td>
<td>3.45</td>
<td>3.33</td>
<td>3.59</td>
</tr>
</tbody>
</table>

Note: Includes the "end of life costs" (removal, dismantlement, etc) for components.

Alternate scenarios: (1) Doubling of site labor cost; (2) Doubling of factory labor cost; (3) High SWU cost—$100; (4) High U₃O₈ cost—$50/kg; (5) High interest rate—10%; (6) Lower capacity factor—80%; and (7) Longer construction period—498 years plus 6 months of testing.
Low financial risk

- Relatively small investment per reactor unit
- Highly modular standard design
- Factory fabrication with good quality assurance
- Cost over-runs during construction are unlikely
- Superb passive safety assures against physical damage