PWR and GFR Design Challenges

Course 22.39

9/26/05

Professor Neil Todreas
<table>
<thead>
<tr>
<th>Major Design Choices</th>
<th>PWR</th>
<th>GFR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant</td>
<td>Water</td>
<td>He or SCO$_2$</td>
</tr>
<tr>
<td>Neutron Spectrum Fuel</td>
<td>Thermal</td>
<td>Fast</td>
</tr>
<tr>
<td>Fuel</td>
<td>UO$_2$</td>
<td>Dispersion in Matrix CERCER (U-TRU) C/SiC</td>
</tr>
<tr>
<td>Decay Heat Removal System</td>
<td>• Active (Gen II) &lt;br&gt; • Passive (AP1000 and ESBWR)</td>
<td>Active or Passive</td>
</tr>
<tr>
<td>Power Conversion Cycle</td>
<td>Rankine</td>
<td>Brayton with Supercritical CO$_2$ Or Helium</td>
</tr>
</tbody>
</table>
# Principle PWR Design Challenges

## #1 Reduction of Capital Cost

### Design Approaches:

- **Constructability**
  - Modularity, Informatics, Construction Techniques

- **Design Approach**
  - Safety by Natural Phenomena
  - Unique Approaches
    - Filtered, Vented Containment
    - Containment in Cooling Tower
    - Steam Generators outside Containment
    - Rapid Refueling Technology
#2 Reduction in O&M Cost

Design and Management Objectives:

- Reduce Operator Burden
- Reduce Plant Operating Staff
#3 Reduce Spent Fuel Inventory (holding fuel cycle cost level)

Design Approaches

• Increase Fuel Burnup
• Increase Plant Thermal Efficiency
• Separation of Actinides
• Reprocessing of Actinides
IRIS Reactor as Design Example

Integral, modular, 335 MWe PWR

48 Month refueling and maintenance cycle

Safety-by-design approach
IRIS Pressurizer

IRIS Spherical Steel Containment Arrangement

# IMPLEMENTATION OF SAFETY-BY-DESIGN™

<table>
<thead>
<tr>
<th>IRIS Design Characteristic</th>
<th>Safety Implication</th>
<th>Accidents Affected</th>
</tr>
</thead>
<tbody>
<tr>
<td>Integral layout</td>
<td>No large primary piping</td>
<td>• LOCAs</td>
</tr>
<tr>
<td>Large, tall vessel</td>
<td>Increased water inventory</td>
<td>• LOCAs</td>
</tr>
<tr>
<td></td>
<td>Increased natural circulation</td>
<td>• Decrease in heat removal</td>
</tr>
<tr>
<td></td>
<td>Accommodates internal CRDMs</td>
<td>• Various events</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• RCCA ejection</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Vessel head penetrations issues</td>
</tr>
<tr>
<td>Heat removal from inside</td>
<td>Depressurizes primary system by condensation and not by loss of mass</td>
<td>• LOCAs</td>
</tr>
<tr>
<td>the vessel</td>
<td>Effective heat removal by SG/EHRS</td>
<td>• LOCAs</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• All events for which effective cooldown is required</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• ATWS</td>
</tr>
<tr>
<td>Reduced size, higher</td>
<td>Reduced driving force through primary opening</td>
<td>• LOCAs</td>
</tr>
<tr>
<td>design pressure containment</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Multiple coolant pumps</td>
<td>Decreased importance of single pump failure</td>
<td>• Locked rotor, shaft seizure/break</td>
</tr>
<tr>
<td>High design pressure</td>
<td>Primary system cannot over-pressure secondary system</td>
<td>• Steam generator tube rupture</td>
</tr>
<tr>
<td>steam generator system</td>
<td>No SG safety valves</td>
<td>• Steam line break</td>
</tr>
<tr>
<td></td>
<td>Feed/Steam System Piping designed for full RCS pressure reduces piping failure</td>
<td>• Feed line break</td>
</tr>
<tr>
<td></td>
<td>probability</td>
<td></td>
</tr>
<tr>
<td>Once through steam</td>
<td>Limited water inventory</td>
<td>• Steam line break</td>
</tr>
<tr>
<td>generator</td>
<td></td>
<td>• {Feed line break}</td>
</tr>
<tr>
<td>Integral pressurizer</td>
<td>Large pressurizer volume/reactor power</td>
<td>• Overheating events, including feed line break.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• ATWS</td>
</tr>
</tbody>
</table>

{ } – Only accident where effect is potentially negative


# TYPICAL PWR CLASS IV ACCIDENTS AND THEIR RESOLUTION IN IRIS DESIGN

<table>
<thead>
<tr>
<th>Condition IV Design Basis Events</th>
<th>IRIS Design Characteristic</th>
<th>Results of IRIS Safety-by-Design™</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Large Break LOCA</td>
<td>Integral RV Layout – No loop piping</td>
<td>Eliminated by design</td>
</tr>
<tr>
<td>2 Steam Generator Tube Rupture</td>
<td>High design pressure once-through SGs, piping, and isolation valves</td>
<td>Reduced consequences, simplified mitigation</td>
</tr>
<tr>
<td>3 Steam System Piping Failure</td>
<td>High design pressure SGs, piping, and isolation valves. SGs have small water inventory</td>
<td>Reduced probability, reduced (limited containment effect, limited cooldown) or eliminated (no potential for return to critical power) consequences</td>
</tr>
<tr>
<td>4 Feedwater System Pipe Break</td>
<td>High design pressure SGs, piping, and isolation valves. Integral RV has large primary water heat capacity.</td>
<td>Reduced probability, reduced consequences (no high pressure relief from reactor coolant system)</td>
</tr>
<tr>
<td>5 Reactor Coolant Pump Shaft Break</td>
<td>Spool pumps have no shaft</td>
<td>Eliminated by design</td>
</tr>
<tr>
<td>6 Reactor Coolant Pump Seizure</td>
<td>No DNB for failure of 1 out of 8 RCPs</td>
<td>Reduced consequences</td>
</tr>
<tr>
<td>7 Spectrum of RCCA ejection accidents</td>
<td>With internal CRDMs there is no ejection driving force</td>
<td>Eliminated by design</td>
</tr>
<tr>
<td>8 Design Basis Fuel Handling Accidents</td>
<td>No IRIS specific design feature</td>
<td>No impact</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Rank</th>
<th>Year</th>
<th>Plant</th>
<th>Accident Precursor</th>
<th>IRIS</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1979</td>
<td>Three Mile Island</td>
<td>Pressurizer Power Operated Relief Valve stuck open</td>
<td>Same accident cannot occur: IRIS has integral pressurizer and no power operated relief valve. Similar accidents (any small break LOCA) have intrinsic mitigation (core always covered)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Partial Core Meltdown occurred</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>1985</td>
<td>Davis Besse</td>
<td>Total Loss of Feedwater (main and auxiliary)</td>
<td>Cannot occur: IRIS safety grade decay heat removal system (EHRS) does not require any source of water injection to the steam generators; also, increased primary side thermal inertia inherently mitigate loss of main feedwater events</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Core Damage Probability = 7*10^-2</strong></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>1981</td>
<td>Brunswick</td>
<td>Residual Heat Removal (RHR) U-tubes Heat Exchanger Failure due to blockage (oyster shells)</td>
<td>BWR Event; eliminated by design and operational procedures for RHR, inherent mitigating features</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Core Damage Probability = 9*10^-3</strong></td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>1991</td>
<td>Shearon Harris</td>
<td>Unavailability of high pressure safety injection (HPSI) pump</td>
<td>Cannot occur: IRIS does not need, thus does not have safety related HPS pumps</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Core Damage Probability = 6*10^-3</strong></td>
<td></td>
</tr>
<tr>
<td>5</td>
<td>2002</td>
<td>Davis Besse</td>
<td>Degraded vessel head; unqualified coatings and debris in containment; potential HPSI pump failure during recirculation</td>
<td>Cannot occur: IRIS has no vessel head penetrations by adoption of internal CRDMs and has no HPSI pumps</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Core Damage Probability = 6*10^-3</strong></td>
<td></td>
</tr>
</tbody>
</table>

Implications of the IRIS Safety-by-Design™ on the 5 Most Severe Accident Precursors since 1979 as Ranked by NRC

Courtesy of M. D. Carelli. Used with permission.

Gas Cooled Fast Reactor (GFR)

Characteristics

• He coolant (SCO2 backup)
• 850°C outlet temperature
• direct gas-turbine conversion cycle – 48% efficiency
• 600 MW_{th}/288 MW_{e}
• Several fuel options and core configurations

Benefits

• Waste minimization and efficient use of uranium resources
• Hydrogen

Focus of this presentation
**Introduction - Why fast reactors?**

**Traditional view**

**Thermal**
- Only ~1% of resource usable

**Fast**
- ~100% of resource usable

- Natural U contains 0.7% U235
- U238 fissionable only with fast neutrons, but fission cross section small
- Neutron capture in U238 converts U238 to fissile Pu239 – fissions in both thermal and fast reactors
- Efficient breeding requires high neutron yield per neutron absorbed in Pu239 – possible only for high neutron energies
- Fast breeder reactors

**Fast reactors make utilization of ~all natural resource possible**
Why consider FRs now?

- Actinide management (waste issue)
  - Actinides in spent LWR fuel – source of long term radiotoxicity
  - If recycled and fissioned, radiotoxicity of waste will reduce to levels comparable to that of nat. U in less than 1000 years
  - Recycling in LWRs possible, but with build up of higher minor actinides
  - Reprocessing and manufacturing difficulties from large neutron doses from spontaneous fission of Cm and Cf

FRs fission minor actinides more effectively
Why consider FRs now? (Cont’)

• Expansion of nuclear into transportation sector (hydrogen economy)
  
    ▪ Need for higher temperature
      • LWRs - temperature limited to ~ 330°C
      • Other coolants – fast reactors offer more options
        * gases, liquid salts – thermal or fast reactors
        * lead alloys, sodium – fast reactors

    ▪ Substituting for gasoline with hydrogen or nuclear electricity for hybrid or electric cars requires enormous expansion of nuclear power

    ▪ Concern over long-term resource availability (cost) returns
## Comparative FBR and LWR characteristics affecting safety

<table>
<thead>
<tr>
<th>LWR</th>
<th>GFR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Optimal reactivity geometry</td>
<td>Core not arranged in most reactive configuration</td>
</tr>
<tr>
<td>High stored energy in coolant</td>
<td>Modest stored energy in coolant</td>
</tr>
<tr>
<td>Reactivity loss on lost of coolant</td>
<td>Reactivity gain* on loss of coolant</td>
</tr>
<tr>
<td>$\beta_{\text{eff}}=0.0065$</td>
<td>$\beta_{\text{eff}}=0.004$</td>
</tr>
<tr>
<td>$\Lambda \sim 2 \times 10^{-5}\text{s}$</td>
<td>$\Lambda \sim 2 \times 10^{-7}\text{s}$</td>
</tr>
<tr>
<td>Thermal time constants</td>
<td></td>
</tr>
<tr>
<td>Fuel</td>
<td>Comparable or longer</td>
</tr>
<tr>
<td>Structures</td>
<td>Seconds</td>
</tr>
<tr>
<td>Coolant</td>
<td>Smaller</td>
</tr>
</tbody>
</table>

*design dependent

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New challenges for FR

FR for resource utilization and actinide management

• Small effective delayed neutron fraction
  – Consequence of TRU – some TRUs have small $\beta$

• Increased positive coolant void worth
  – Consequence of Np237 and Am241 – exhibit threshold effect on fission cross section

• Achievement of BR=1 difficult in the absence of blankets
  – Large cores with low leakage preferred for good neutron economy
  – Less freedom to reduce coolant void worth by increased leakage than in actinide burner core

GFR will be given as example of FR choice
Example of GFR: Fast reactor for resource utilization and actinide management

- **Objectives**
  - 1\textsuperscript{st} core to be loaded with TRUs from spent LWR fuel and uranium
  - Ultimately a conversion ratio $= 1.0$, self-sustaining core with recycling of its own fuel (only natural uranium feed for subsequent core loadings)
  - Temperatures compatible with high efficiency hydrogen generation
  - Excellent safety
  - Attractive economy
Key safety issues

• Safety was the key issue in the GCFR development in the 70’s
• Significant difference from liquid metal cooled FRs – coolant is pressurized and can be lost rapidly, as in LWRs
• Gas has very small heat capacity and heat transport capability
• Hence, primary issue is post LOCA decay heat removal
• Another issue is reactivity increase from coolant depressurization
Issue of coolant void worth

- Wide-spread belief – gas cooled reactors do not pose coolant void reactivity problem (coolant neutronically transparent)
- Does not hold for gas-cooled GFRs with hard spectrum
- Currently two coolant candidates – supercritical CO$_2$ and helium
- $\text{SCO}_2$ core-average density at 20MPa = 0.137g/cc (1/5th of water) – significant moderation and increase of fission to capture ratio in HMs upon LOCA, moreover large scattering cross section
- How about He at 8MPa (density=0.01g/cc) – less challenging
- Can void $\Delta k$ be decreased through leakage enhancement as in LMRs? Yes - SCO$_2$; No - Helium
Issue of post LOCA cooling

Can a GFR be designed with passive decay heat removal?

Approaches investigated

• Solid matrix core with decay heat removal by conduction and radiation
• Solid matrix core with heat pipes
• Low pressure drop core cooled by natural circulation loop with emergency cooling heat exchangers + containment at elevated pressure

Note: Design exploration in progress in France, Japan and US

Coolants: helium, CO2, supercritical CO2
Approach 1: Long life, low power density design

Low power density design – possible candidate?

Addresses LOCA

Coolant holes

CERMET or METMET fuel in a metallic matrix

Replaceable low-density reflector or void

Permanent side reflector

Active core

Synergistic twin to thermal MHR-GT

Same power density as MHR-GT - 8kW/l
Long core life – 50 years

Decay heat after depressurization removed by conduction and radiation through vessel wall


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Approach 1: Very long life core but high fuel cycle cost

Pu from spent LWR fuel

50-year core life feasible with MET-MET fuel

Within allowable fluence and burnup limits

Pu composition and mass at discharge same as at BOL - provides fuel for next reactor

\[ f_{HM} = \text{weight fraction of HM in matrix} \]


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Approach 1: Decay heat removal

Decay heat from 300MWe-core can be dissipated at $T_{\text{max}}=1200^\circ\text{C}$
How to increase conduction/radiation-limited power density?

Diagram removed for copyright reasons.
Approach 2: Distributed Heat Sinks

Two possible options:

1. **Use coolant channels available for normal operation cooling**
   - Simplicity, no additional structures, better neutron economy
   - Dictates natural convection cooling, low HT rates, hence elevated pressure
   - Loss of containment pressure leads to core damage (different from PWRs)

2. **Introduce additional heat sink (HPs) independent of coolant channels**
   - Independent of state of coolant, no need for higher pressure
   - Rugged and simple devices, passive initiation of operation
   - **But**, Impaired neutronic performance (parasitic absorption)
   - Exposure to high fluence, challenge for materials
   - Introduction of additional modes of reactivity change
   - Interference with refueling, additional vessel penetration
1. Additional Heat Sinks

- Power density 50W/cc feasible with 37 heat pipes in block cores
- Significant neutronic penalties
- Issues of heat pipe failure, containment penetration
- Materials, which meet mechanical, thermal, neutronic and economic targets are difficult to find
2. Convective cooling at elevated pressure

- 3x 50% or 4x50% cooling loops
- low pressure drop core
- after depressurization of primary system, containment pressure increases and provides elevated pressure needed for natural circulation

But
- Relatively large pipes needed
- Potential for vapor ingress to core in case of HX tube failure
- containment must reliably maintain elevated pressure

- This approach selected as the most promising
Natural circulation performance - CO$_2$ and He

Post LOCA core temperature profiles

CO$_2$
- Limits – peak cladding temperature $= 1200^\circ$C, maximum core-average outlet $T = 850^\circ$C
- 2% decay heat can be removed by natural circulation
- CO$_2$ much better than He – requires backup pressure of 5 bars versus 13 bars for He
- Helium – issue of excursion type instabilities

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Design selections to achieve high safety

- Use combination of active and passive safety systems
  - 3x50% (or 4x50%) emergency/shutdown cooling loops with diverse power supply (Diesels, fuel cells, microturbines)
  - Active blowers are first line of defense
  - Passive emergency cooling by natural circulation is a backup
- MIT - CO$_2$ coolant at Tout=650°C is primary choice. It can remove 2% decay power at containment pressure of 5bars – typical PWR containment design pressure, but not compatible with carbide fuel, hence innovative tube-in-duct assembly (ODS duct) with VIPAC (U,TRU)O$_2$ investigated
- France- Helium coolant is current primary choice – use active systems to operate for first 24 hours until decay heat is reduced to 0.5% when natural circulation can take over at reasonably low backup pressure (cost of containment is key). Fully active version is also being looked at.
Example of GFR design by CEA

- Jet pump
- DHR loop
- Steel Guard confinement
- Gas tank for additional circulation system (First 24 hrs)
- Plate low-dp core
- (U-TRU)C fuel
- SiC matrix

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