Current reactor technology, a.k.a. GENERATION III:

Radek Škoda
Texas A&M
WNU SI 2011
Christ Church, Oxford
Overview

1. ANALOGY: Old does not mean bad but…

2. GenII/III: How are they different?

3. Resulting types…
FORD Thunderbird 1964

6.4 l 300 bhp (224 kW) V8 petrol engine
0 to 60 mph in 11 seconds
top speed 120 mph (200 km/h)
economy 11 mpg (21 l/100km)
FORD Mondeo 2008

2.2 l 172.6 bhp (128.7 kW) TDCi Common-rail 4l Diesel Engine
8.4 seconds 0 to 100 km/h
Top Speed of about 220 km/h
Economy 45.6 mpg (~ 6.2l / 100km)
### FORD: changes in 45 years

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SAFETY: “In Love With My Car”

5 STAR crash test
ABS
Air bags

ISOFIX
Crash zones
Standard safety belts
Analogy not perfect as Gen II reactors have had many upgrades/uprates which is not true for T-birds...
Back to reactors: Gen III still evolve

First reactors

- Atoms for Peace
- TMI-2
- Chernobyl

Generation I
- Shippingport
- Dresden
- Fermi I
- Magnox

Generation II
- LWR – PWR, BWR
- CANDU
- GCR
- VVER 440, 1000
- RBMK

Generation III
- EPR, AP 1000, AES 2006
- APR 1000, APR 1400

Generation IV
- Shippingport
- Dresden
- Fermi I
- Magnox

Historical development of nuclear power

- 1950
- 1970
- 1990
- 2010
- 2030
- 2050
- 2070
- 2090

Courtesy of J. Misak, NRI Rez
Gen-II typical features

- Power level up to 1000 MWe
- Plant availability ~ 75-80%
- Efficiency ~ 30 %
- Base load operation
- Plant life time 30-40 years
Gen-II typical features

- CDF less than once in 10,000 years
- LERF less than once in 100,000 years
- Resistance to single failure of equipment or human error (redundancy 2x100 %, 3x100 % or 4x50 %)
- Safety systems designed to cope with a set of DBAs
Gen-II typical features

- Limited use of passive systems
- Severe accidents dealt with by means of accident management programmes (absence of dedicated systems)
- Operator grace time minimum 30 minutes, analogue I&C
- Fuel burn-up 30-40 MWd/kg of U, refuelling once a year
Gen II->III progress

• Improved economy
• Innovative technology
• Improved safety, namely:
  • reflect Three Mile Island
  • reflect Chernobyl
  • reflect September 11, 2001
  • reflect Fukushima
Gen-III typical features

• Power level from 1100 to 1700 MWe
• Gross efficiency up to 39%
• Higher availability (from 70-80% up to 95%)
• Load follow capability
• Longer operational life (from 30-40 years to 60 years)
Gen-III typical features

- Reduced frequency of core melt accidents (10-100 times), CDF: 1E-7 – 1E-5/year
- Minimal environmental effects ("practically" no EPZ needed), LERF: 1E-9 – 1E-6/year
- Dedicated systems for mitigation of severe accidents
- Digital I&C
Gen-III typical features

- Extended use of passive systems for some designs
- Increased period without operator actions, sometimes “infinitely”
- Robust double containment (with annulus venting), increased strength, designed against aircraft crash
Gen-III typical features

- Higher burn-up to improve fuel use and reduce waste (from 30-40 MWd/kg to 60-70 MWd/kg in long term up to 100 MWd/kg)
- Standard use of MOX and burnable absorbers
- Fuel cycle 18 – 24 months
- Seismic resistance 0.25 – 0.3 g
2 ways from Gen II to Gen III:

- **ECONOMY**
  - Evolutionary Design
    - Increased Power
- **SAFETY**
  - Passive Design
    - Simplification - Reduced Number of Components
    - Passive Systems
    - Dedicated Systems for Severe Accidents
      - Digital Control, Etc.
Relative Cost Structure of Power Generation

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More $$$ data from Steve Kidd & Adrian Ham in few days.
### Construction cost reduction

**Costs of Construction (est) ABWR**  
(-see Prof. Geo Tolley et al)

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* owners costs, contingency
Capacity factor requirements

Nuclear News

CAPACITY FACTOR REMAINS OVER 90%
The three-year median DER capacity factor of U.S. power reactors remains high, with no sign of adverse effects from work on new reactors.

Full story on page 39

Also in this issue
- Poland moves toward introducing nuclear power ................. p. 49
- Suits filed to stop payments to Nuclear Waste Fund ........ p. 58
Fig. 1: All reactors. The median DER net capacity factor for the 104 operating reactors has effectively leveled off, with the medians of the past three three-year periods within half a percentage point. The chart shows only reactors that are still in service now; there were 20 such reactors in 1974–76, and in each succeeding period there were 43, 53, 60, 77, 97, 102, 103, and 104 in each of the last four. If closed reactors were included in the periods during which they operated, the median would differ by more than one percentage point in only one period: 1980–1982, when it was 57.57.
Safety improvements different for each Gen III design:

**EPR safety features**

Evolutionary design with core catcher. Four active safety trains.
Safety improvements different for each Gen III design: AP1000 passive cooling
Gen II -> III evolution

VVER-440/213 Plant Layout

- 1. Reactor pressure vessel
- 2. Steam generator
- 3. Return line machine
- 4. Steam condenser
- 5. Containment system
- 6. Reactor feed-water system
- 7. Reactor water supply system
- 8. Containment system
- 9. Sparging system
- 10. Check valves
- 11. Intake air vent
- 12. Turbine
- 13. Compressor
- 14. Turbine block
- 15. Feedwater tank with degasifier
- 16. Turbine hall crane
- 17. Electric instrumentation and control compartments

This illustration shows a vertical "cut" through containment with bubble condenser tower.

Internal shell made of prestressed reinforced concrete with hermetically sealed metal lining from the core side for accident localizing.

External shell made of reinforced concrete (non-prestressed)

2011
Gen II -> III evolution

BWR Evolution

Dresden 1 → KRB → Dresden 2 → ABWR
Gen II -> III evolution

Primary Containment Evolution

DRY

MARK I

MARK II

MARK III

ABWR
## Reactors in construction in 2011 (1/5)

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<td>PWR</td>
<td>APR-1400</td>
<td>KHNP</td>
<td>DHICKOPC</td>
<td>2008-10-16</td>
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<td>KR</td>
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## Reactors in construction in 2011(4/5)

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<th>Model</th>
<th>Operator</th>
<th>Reactor Supplier</th>
<th>Const. Date</th>
<th>Grid Date</th>
<th>RUP [MWe]</th>
</tr>
</thead>
<tbody>
<tr>
<td>RU</td>
<td>AKADEMIK LOMONOSOV 1</td>
<td>Under Construction</td>
<td>PWR</td>
<td>KLT-40S 'Floati'</td>
<td>REA</td>
<td>ROSATOM</td>
<td>2007-04-15</td>
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<td>32</td>
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<tr>
<td>RU</td>
<td>AKADEMIK LOMONOSOV 2</td>
<td>Under Construction</td>
<td>PWR</td>
<td>KLT-40S 'Floati'</td>
<td>REA</td>
<td>ROSATOM</td>
<td>2007-04-15</td>
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<td>RU</td>
<td>BELOYARSKY-4 (BN-800)</td>
<td>Under Construction</td>
<td>FBR</td>
<td>BN-800</td>
<td>REA</td>
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<td>2006-07-18</td>
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<td>RU</td>
<td>KALININ-4</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-320</td>
<td>REA</td>
<td>ROSATOM</td>
<td>1986-08-01</td>
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<td>RU</td>
<td>KURSK-5</td>
<td>Under Construction</td>
<td>LWGR</td>
<td>RBMK-1000</td>
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<td>RU</td>
<td>LENINGRAD 2-1</td>
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<td>PWR</td>
<td>VVER V-491</td>
<td>REA</td>
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<td>2008-10-25</td>
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<td>1085</td>
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<td>RU</td>
<td>LENINGRAD 2-2</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-491</td>
<td>REA</td>
<td>ROSATOM</td>
<td>2010-04-15</td>
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<tr>
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<td>NOVOVORONEZH 2-1</td>
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<td>RU</td>
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<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-320</td>
<td>REA</td>
<td>ROSATOM</td>
<td>2009-09-15</td>
<td></td>
<td>1011</td>
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<tr>
<td>RU</td>
<td>ROSTOV-4</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-320</td>
<td>REA</td>
<td>ROSATOM</td>
<td>2010-06-16</td>
<td></td>
<td>1011</td>
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<tr>
<td>SK</td>
<td>MOCHOVCE-3</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-213</td>
<td>EMO</td>
<td>SKODA</td>
<td>1987-01-27</td>
<td>2012-12-30</td>
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<tr>
<td>SK</td>
<td>MOCHOVCE-4</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-213</td>
<td>EMO</td>
<td>SKODA</td>
<td>1987-01-27</td>
<td>2013-09-01</td>
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Reactors in construction in 2011(5/5)

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<tr>
<th>ISO Code</th>
<th>Unit</th>
<th>Current Status</th>
<th>Type</th>
<th>Model</th>
<th>Operator</th>
<th>Reactor Supplier</th>
<th>Const. Date</th>
<th>Grid Date</th>
<th>RUP [MWe]</th>
</tr>
</thead>
<tbody>
<tr>
<td>TW</td>
<td>LUNGSEMEN 1</td>
<td>Under Construction</td>
<td>BWR</td>
<td>ABWR</td>
<td>TPC</td>
<td>GE</td>
<td>1999-03-31</td>
<td>2011-02-15</td>
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<td>TW</td>
<td>LUNGSEMEN 2</td>
<td>Under Construction</td>
<td>BWR</td>
<td>ABWR</td>
<td>TPC</td>
<td>GE</td>
<td>1999-08-30</td>
<td>2012-02-15</td>
<td>1300</td>
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<tr>
<td>UA</td>
<td>KHMELNITASKI-3</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER V-392B</td>
<td>NNEGEC</td>
<td>ASE</td>
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<td>2015-01-01</td>
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<tr>
<td>UA</td>
<td>KHMELNITASKI-4</td>
<td>Under Construction</td>
<td>PWR</td>
<td>VVER</td>
<td>NNEGEC</td>
<td>ASE</td>
<td>1987-02-01</td>
<td>2016-01-01</td>
<td>950</td>
</tr>
<tr>
<td>US</td>
<td>WATTS BAR-2</td>
<td>Under Construction</td>
<td>PWR</td>
<td>W (4-loop) (ICE)</td>
<td>TVA</td>
<td>WH</td>
<td>1972-12-01</td>
<td>2012-08-01</td>
<td>1165</td>
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</table>

COUNTRY               REACTOR TYPE
China    27  38 x PWR (21xCPR/CNP …)
Russia   11  16 x VVER
Korea     5  4 x BWR (GE+Hitachi/GE)
India    5  3 x PHWR
...

*Courtesy of F. Hezoucky, CTU Prague*
Pause
Gen III: not so beautifully different

<table>
<thead>
<tr>
<th>BWR</th>
<th>PHWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR</td>
<td>VVER</td>
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</tbody>
</table>
ABWR
EPR
APWR
Kerena
Atmea1
CPR1000
AP1000
ESBWR
APR1400
EC6
ACR1000
VVER1200
ABWR
Advanced Boiling Water Reactor

- Licensed / Certified in 3 Countries – First Design Certified by NRC under Part 52 – Generation III

- Four operating in Japan

- Several more under construction or planned – was Japan’s BWR for foreseeable future

- Power Level(s):
  - 3,926 MWt (1350 MWe net) US Certified
  - 4,300 MWt (1460 MWe net) FIN Offering
The only GEN-III REACTOR TYPE CURRENTLY IN OPERATION: ABWR
ACR1000/EC6
The CANDU Power Reactor

Qinshan Phase III, China

EC6:
- Generation III
- 700 MWe class
- Heavy water moderated
- Heavy water cooled
- Natural uranium fuel

ACR1000:
- Generation III+
- 1200 MWe class
- Heavy water moderated
- Light water cooled
- LEU fuel
CANDU Design Fundamentals

Natural uranium fuel
Fully automated operation
Computer controlled
Channel-type reactor
On-power refuelling
Heavy water moderator

Fuel bundle

Calandria tube

37-element bundle (20kg / 50cm)
The CANDU Advantage

On-power refuelling
  Allows for high capacity factors (>90%)
  Enhanced load following capability

Heavy Water Moderator/Coolant
  Allows for the use of NU fuel and other advanced fuel cycles
  Thorium, MoX, Spent PWR fuel

Medium reactor size (740Mwe) flexibility for grid stability

Passive safety features and large heat sinks for station blackouts

Extended design life to 60 years

Short construction schedule
APR1400
APWR
Introduction of Mitsubishi APWR Family, GEN-Ⅲ+ Reactors

- Enhanced Safety
- Attractive Economy
- Enhanced Reliability
- More Environmentally Friendly

Comparison of Current 4 Loop and APWR Family (1/2)

<table>
<thead>
<tr>
<th></th>
<th>Current 4 Loop</th>
<th>APWR family</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electric Output</td>
<td>1,180 MWe</td>
<td>1,538 ~ 1,700 class MWe</td>
</tr>
<tr>
<td>Core Thermal Output</td>
<td>3,411 MWt</td>
<td>4,451 MWt</td>
</tr>
<tr>
<td>Steam Generator (hta¹)</td>
<td>4,870 m²</td>
<td>6,500 m² / 8,500 m²</td>
</tr>
<tr>
<td>Reactor Coolant Pump (flow)</td>
<td>20,100 m³/h</td>
<td>25,400 m³/h</td>
</tr>
<tr>
<td>Turbine (LP last-stage blade)</td>
<td>44 inch</td>
<td>54 ~ 70 class inch</td>
</tr>
<tr>
<td>NO. of Fuel Assem.</td>
<td>193</td>
<td>257</td>
</tr>
<tr>
<td>Fuel Lattice</td>
<td>17 x 17</td>
<td>17 x 17</td>
</tr>
<tr>
<td>Active Fuel Length</td>
<td>12 ft</td>
<td>12 ft / 14 ft</td>
</tr>
<tr>
<td>Reactor internals</td>
<td>Baffle/former structure</td>
<td>Neutron Reflector</td>
</tr>
<tr>
<td>In-core Instrumentation</td>
<td>Bottom mounted</td>
<td>Bottom / Top mounted</td>
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</tbody>
</table>

*¹ Heat transfer area
# Introduction of Mitsubishi APWR Family, GEN-III+ Reactors

## Comparison of Current 4 Loop and APWR Family

<table>
<thead>
<tr>
<th>Safety Systems</th>
<th>Current 4 Loop</th>
<th>APWR family</th>
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<tbody>
<tr>
<td>Trains</td>
<td>2 trains</td>
<td>2 / 4 trains</td>
</tr>
<tr>
<td>Electrical</td>
<td>2 trains</td>
<td>4 trains</td>
</tr>
<tr>
<td>Mechanical</td>
<td>2 trains</td>
<td></td>
</tr>
<tr>
<td>HHSI pump</td>
<td>100% x 2</td>
<td>50% x 4 (DVI*1)</td>
</tr>
<tr>
<td>LHSI pump</td>
<td>100% x 2</td>
<td></td>
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<tr>
<td>ACC</td>
<td>4</td>
<td>4 (Advanced)</td>
</tr>
<tr>
<td>RWSP</td>
<td>Outside CV</td>
<td>Inside CV</td>
</tr>
<tr>
<td>Containment Vessel</td>
<td>PCCCV</td>
<td>PCCV</td>
</tr>
<tr>
<td>I &amp; C</td>
<td>Conventional</td>
<td>Full Digital</td>
</tr>
<tr>
<td>Control Room</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety I&amp;C</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Non-Safety I&amp;C</td>
<td></td>
<td></td>
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</tbody>
</table>

*1: Direct Vessel Injection
APWR Family’s advanced technology

**Reactor**
- 1500～1700 MWe class large capacity (thermal 4,451MWt)
- Neutron reflector

**Steam Generator**
- High performance separator
- Increased capacity with compact sizing

**Engineering Safety Features**
- Simplified configuration with 4 mechanical sub-systems
- In-containment RWSP
- Advanced accumulator

**Turbine**
- 54～70 inch-length blades in LP turbine
- Fully integrated LP turbine rotor

**I & C**
- Digital control & protection systems
- Compact console
AP1000
ATMEA1
### ATMEA1 reactor

#### Main technical data

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>3150 MWth</td>
</tr>
<tr>
<td>Net power</td>
<td>1150 MWe</td>
</tr>
<tr>
<td>Thermal efficiency</td>
<td>37%</td>
</tr>
<tr>
<td>Target design availability</td>
<td>92%</td>
</tr>
<tr>
<td># loops</td>
<td>3</td>
</tr>
<tr>
<td>(secondary) Steam pressure</td>
<td>72 bar</td>
</tr>
<tr>
<td>operation cycle length</td>
<td>up to 24 months</td>
</tr>
<tr>
<td>Collective dose</td>
<td>&lt;0.5 manSievert/yr</td>
</tr>
<tr>
<td>Design service life</td>
<td>60 yrs</td>
</tr>
<tr>
<td>I&amp;C</td>
<td>full digital</td>
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ATMEA1™ reactor view
ATMEA1 reactor Layout
EPR
### Main technical data

<table>
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<tr>
<td>Thermal power</td>
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<td>Net power</td>
<td>1650 MWe</td>
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<td>Thermal efficiency</td>
<td>37%</td>
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<tr>
<td>Target design availability</td>
<td>92%</td>
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<tr>
<td># loops</td>
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<tr>
<td>(secondary) Steam pressure</td>
<td>77 bar</td>
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<tr>
<td>Operation cycle length</td>
<td>up to 24 months</td>
</tr>
<tr>
<td>Collective dose</td>
<td>&lt;0.5 manSievvert/yr</td>
</tr>
<tr>
<td>Design service life</td>
<td>60 yrs</td>
</tr>
<tr>
<td>I&amp;C</td>
<td>full digital</td>
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The Quick Tour
March 2009
### Main technical data

<table>
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<th>Parameter</th>
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<tr>
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<td>3370 MWth</td>
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<tr>
<td>Net power</td>
<td>1250 MWe</td>
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<tr>
<td>Thermal efficiency</td>
<td>37%</td>
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<tr>
<td>Target design availability</td>
<td>92%</td>
</tr>
<tr>
<td># loops</td>
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<tr>
<td>(secondary) Steam pressure</td>
<td>75 bar</td>
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<tr>
<td>operation cycle length</td>
<td>up to 24 months</td>
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<tr>
<td>Collective dose</td>
<td>&lt;0.5 manSievert/yr</td>
</tr>
<tr>
<td>Design service life</td>
<td>60 yrs</td>
</tr>
<tr>
<td>I&amp;C</td>
<td>full digital</td>
</tr>
</tbody>
</table>
KERENA reactor Layout

- Circulating water pump building
- Service water pump building
- Emergency diesel generator building
- Switchgear building
- Reactor building
- Turbine building
- Reactor auxiliary building
- Reactor supporting systems building
- Service water pump building
- Emergency diesel generator building
The AREVA Reactor range benefits

**KEY BENEFITS**

**BUSINESS PERFORMANCE**
- Maximized availability: design target >92%
- Short outages
- High thermal efficiency
- Minimized global power generation costs
- Low O&M costs
- Fuel cycle flexibility
- MOX fuel

**OUTSTANDING SAFETY**
- Large commercial airplane crash resistance
- Advanced features for core melt and releases management
- Optimized level of redundancy, diversity of systems and incremental mitigation of abnormal events

**ENVIRONMENTAL PROTECTION**
- Lower volume of final waste
- Reduced collective dose

**ENERGY SUPPLY CERTAINTY**
- Gen III+ based on evolutionary designs
- AREVA integrated supply chain strategy for critical components
- Proven Digital Safety I&C technology
ESBWR
Economic Simplified Boiling Water Reactor (ESBWR) of GE Hitachi

Output Power: 1520 MW net
Emissions: Nearly zero greenhouse gas emissions
Lifetime: 60 years
CDF: 2 E-8, the safest in the industry by a factor of ten
Capacity Factor: 95% with a capacity factor greater than the current BWR fleet's average of 92%
Cycle Length: 24-month cycles

09 March 2011: GE Hitachi Nuclear Energy’s ESBWR Reactor Design Receives NRC’s Final Design Approval, Clearing the Way for Global Sales

RS, Texas A&M WNU SI 2011
CPR1000/+
CPR1000/+ 

- 1,080-MWe PWR reactor for domestic Chinese market
- Based on AREVA three-loop design
- FOAK built in 52 months !!
- CPR1000 Gen II+ designated reactor features
  - digital instrumentation and controls
  - 60-year design life
- CPR1000+ should have all standard Gen III features
- Reportedly cost under 10,000 yuan per kilowatt (US$1,500).
- In accordance with China’s “self-reliance” policies, more than 50% of its equipment was locally sourced
VVER 1200
aka AES 2006
aka MIR 1200
<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
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</thead>
<tbody>
<tr>
<td><strong>1. Lifespan:</strong></td>
<td>60 years</td>
</tr>
<tr>
<td><strong>2. Power unit output, MW:</strong></td>
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<tr>
<td>- electric (gross)</td>
<td>1,158 MW_E</td>
</tr>
<tr>
<td>- electric (net)</td>
<td>1,078 MW_e</td>
</tr>
<tr>
<td>- thermal</td>
<td>3,200 MW_t</td>
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<tr>
<td><strong>3. Minimum coefficient of installed output utilization</strong></td>
<td>&gt; 90%</td>
</tr>
<tr>
<td><strong>4. Efficiency</strong></td>
<td></td>
</tr>
<tr>
<td>- gross</td>
<td>36.2%</td>
</tr>
<tr>
<td>- net</td>
<td>33.7%</td>
</tr>
<tr>
<td><strong>5. Maximum fuel burn-out</strong></td>
<td>60 MWd/kgU</td>
</tr>
<tr>
<td><strong>6. Duration of scheduled outages</strong></td>
<td></td>
</tr>
<tr>
<td>within seven years of operation (period between two large repairs involving dismantling of the turbine equipment is 8 years)</td>
<td>4 x 16 days  2 x 24 days  1 x 30 days</td>
</tr>
</tbody>
</table>
Aircraft crash

Snow and ice loads

Peak (extreme) load due to snow accepted in the design is equal to 4.1 kPa.

The NPP design considers the aspects of aircraft crash.

Seismic loads

The NPP is designed to withstand an earthquake with the maximum horizontal acceleration at a ground level equal to 0.25 g.

Wind loads

The safety related components have been designed taking into account the wind load at a wind velocity of 30 m/s at a height of 10 m. In the course of further design stages, this load can be corrected, taking into considerations the site conditions. It should be mentioned that the determinative wind load is whirlwind effect. The loads accepted in the design are loads induced by whirlwind of class 3.60 on the Fujita scale.

External explosions

The NPP safety related components have been designed to withstand the effect of an air shock wave induced by an external explosion. Pressure in the shock wave front is accepted being equal to 30.0 kPa, compression phase duration 1 s.
Which GEN III reactor is the best?

- Several PERFORMANCE indicators:
  - Availability factor
  - Load factor
  - Capacity factor
  - Operating factor...

- Also SAFETY, SECURITY indicators

- Different COST to build/run...

- Different fuel/enrichment needed...

- Different suppliers...

- Different size

ONE SIZE FITS ALL not applicable!
# Working groups reactor list:

<table>
<thead>
<tr>
<th>Reactor</th>
<th>Working Group</th>
<th>Group</th>
</tr>
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<tbody>
<tr>
<td>ABWR</td>
<td>Working Group</td>
<td>A1</td>
</tr>
<tr>
<td>AP1000</td>
<td>Working Group</td>
<td>A2</td>
</tr>
<tr>
<td>EPR</td>
<td>Working Group</td>
<td>A3</td>
</tr>
<tr>
<td>ESBWR</td>
<td>Working Group</td>
<td>A4</td>
</tr>
<tr>
<td>APWR</td>
<td>Working Group</td>
<td>A5</td>
</tr>
<tr>
<td>APR1400</td>
<td>Working Group</td>
<td>A6</td>
</tr>
<tr>
<td>Kerena</td>
<td>Working Group</td>
<td>A7</td>
</tr>
<tr>
<td>ACR1000</td>
<td>Working Group</td>
<td>A8</td>
</tr>
<tr>
<td>CPR1000</td>
<td>Working Group</td>
<td>A9</td>
</tr>
<tr>
<td>VVER1200</td>
<td>Working Group</td>
<td>A10</td>
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</table>

RS, Texas A&M       WNU SI 2011
Working group action plan:

1. Present in 2 slides what are the strengths of your reactor and why you would build it.

2. Choose another Gen-III reactor from the list of 10 (i.e. one your colleagues promote) and define in 1 slide a situation/scenario in which it would be better than the one you promoted in Step 1.

EXAMPLE:

VVER1200 will be promoted by Working Group A10, and they can choose any reactor of the remaining 9 types for Step 2.
Thank you!

Radek Skoda
Texas A&M University,
317 Olin E. Teague Research Center
3473 TAMU, College Station, Texas 77843-3473, USA
email: rskoda@ne.tamu.edu
## Back up slides

<table>
<thead>
<tr>
<th></th>
<th>AES-2006</th>
<th>APWR 1700</th>
<th>AP1000</th>
<th>EPR</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Thermal/electric power</strong></td>
<td>3200/1150 MW</td>
<td>4451/1700 MW</td>
<td>3415/1200/1117 MW</td>
<td>4250/1655 MW</td>
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<tr>
<td><strong>Design characteristics</strong></td>
<td>Evolutionary design, combination of active and passive features, 4 loops, horizontal SG</td>
<td>Evolutionary design, limited passive features, vertical SG</td>
<td>Extensive passive safety features, simplified construction and operation, 2 steam generators, 4 cold legs, vertical SG</td>
<td>Evolutionary design, limited passive features, 4 loops, vertical SG</td>
</tr>
<tr>
<td><strong>Reactor coolant system</strong></td>
<td>RCS pressure 16.2 MPa, reactor outlet 328.9°C, steam pressure 7 MPa</td>
<td>RCS pressure 15.5 MPa, reactor outlet 325°C</td>
<td>RCS pressure 15.5 MPa, reactor outlet 321°C, steam pressure 5.76 MPa</td>
<td>RCS pressure 15.5 MPa, reactor outlet 330°C, steam pressure 7.63 MPa</td>
</tr>
<tr>
<td><strong>Primary containment</strong></td>
<td>Prestressed concrete with metallic liner, 1100 - 1200 mm, volume 74200 m³, design pressure 0.5 MPa, leak rate 0.2 vol.%/den</td>
<td>Single prestressed concrete with metallic liner, 1120-1320 mm, design pressure 0.57 MPa</td>
<td>Freestanding, cylindrical steel vessel 44.4 mm thick, volume 56600 m³, design pressure 0.5 MPa, leak rate 0.10 vol.%/day</td>
<td>Prestressed concrete with metallic liner, 1300 mm wall, volume 80 000 m³, design pressure 0.55 MPa, leak rate 0.3 vol.%/day</td>
</tr>
<tr>
<td><strong>Secondary containment</strong></td>
<td>Reinforced concrete, 1.8 – 2.2 m, active annulus venting</td>
<td>Vented surrounding structures in the area of penetrations</td>
<td>Reinforced concrete building with conical shell, open to the atmosphere at the top, 0.91 m thick wall</td>
<td>Reinforced concrete, wall thickness 1.8 m, active venting of the annulus</td>
</tr>
<tr>
<td><strong>Plant lifetime</strong></td>
<td>60 years</td>
<td>60 years</td>
<td>60 years</td>
<td>60 years</td>
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</table>

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From J. Misak, NRI REZ
## AES-2006

<table>
<thead>
<tr>
<th>ECCS</th>
<th>APWR 1700</th>
<th>AP1000</th>
<th>EPR</th>
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<tbody>
<tr>
<td>4x100 % high head, 4x100 low head, 4 accumulators, low pressure pumps used for RHR</td>
<td>4 enhanced accumulators, 4x50 % high pressure pumps, no low pressure pumps</td>
<td>Passive system with 3 sources of water: core makeup tanks, 2 accumulators and a in-containment refuelling water storage tank</td>
<td>4x100 % medium head, 4x100 low head, 4 accumulators</td>
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## RHRS

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<tbody>
<tr>
<td>4x50 % active system</td>
<td>4x50% active RHR system from PC, HEX common with spray system</td>
<td>Passive RHR system with full capacity heat exchanger, IRWST located above the primary loops, provides the heat sink for the system</td>
<td>Either through the secondary circuit, or combined with low head safety injection</td>
</tr>
<tr>
<td>4x33 % passive system for severe accidents</td>
<td>4x50% active spray system, pumps and heat exchangers common for RHR system</td>
<td>External-passive through the containment wall to the ambient air, enhanced by gravity drain of water from external tank on the shell, no need for containment spray and fan coolers</td>
<td>For DBAs are ECCS and HVAC system available, no sprays. Dedicated spray system for severe accidents 2x100 %</td>
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## CHRS

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<tr>
<th>Molten corium stabilization</th>
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</thead>
<tbody>
<tr>
<td>Core catcher, passive initiation</td>
<td>Large area flooded reactor cavity, integrity ensured for 24 hours</td>
<td>In-vessel corium retention, flooding from IRWST, passive initiation</td>
<td>Corium spreading compartment 170 m², cooling from IRWST initiated by fusible plugs</td>
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</table>

## Hydrogen control

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<tbody>
<tr>
<td>PARS</td>
<td>Igniters</td>
<td>3 recombines designed for LOCAs, 64 igniters for severe accidents</td>
<td>40 PARs, in combination with igniters in selected locations</td>
</tr>
</tbody>
</table>
Implications of Gen III design features on safety analysis

- **Extended use of passive systems:** low driving forces, in particular in case of natural circulation, therefore more detailed modelling necessary, in particular in case of two-phase flow
- **High reactor thermal power with very flat power profile:** many highly loaded fuel assemblies therefore more vulnerable to damage; exact prediction of a number of damaged fuel rods and source term required
- **Large dimensions of the core:** neutronic and thermal hydraulic space effects and their interrelations more important
- **CDF and LRF reduced by 2 orders,** with large attention put on them; more attention to all components, accuracy, screening-out criteria, etc
Implications of Gen III design features on safety analysis

- Significantly enlarged lifetime of components: limited experience with such long-term processes, monitoring and management of ageing very important.
- Enhanced resistance of containment and other buildings against external hazards, in particular aircraft crashes: harmonization of methodology and improved modelling of impacts needed.
- Severe accidents included in design basis: still several phenomena considered worldwide not known sufficiently, therefore further works necessary on detailed modelling of the processes.
- Corium stabilization by large volume of coolant; resulting containment pressure loading in case of inadequate heat removal to be considered.
Implications of Gen III design features on safety analysis

- Use of dedicated equipment for corium stabilization (core catcher, spreading compartment); adequate information to be provided to scientific community in order to become familiar with their modelling.

- **Management of hydrogen in severe accidents**: production, distribution, combustion and detonation of hydrogen are strongly spatially dependent processes, with potentially locally risky regions; detailed models for production, distribution and management of hydrogen needed.

- **Modified material, geometrical, neutronic and thermal-hydraulic properties of fuel** and the whole core: reliability of heat removal for various plant states needs reconsideration (including experiments).

- **Increased linear dimensions of the main components**: more attention to be paid to 3-D effects and reconsideration of scaling for transfer of results from experiments on the plant.
Implications of Gen III design features on safety analysis

- Large-scale use of computer techniques in control and protection plant systems: the issues connected with verification, validation and diversification of systems to be addressed.

- High plant availability: reduced refuelling period, on line maintenance needs detailed risk modelling, improved risk monitors, use of risk oriented maintenance, etc.

- Load follow operation: operation with reduced power, island mode operation, primary and secondary power control affect plant lifetime, control system reliability, nuclear fuel behaviour, production of waste, etc.

- Significant increase of fuel burn-up, use of burnable absorbers, longer fuel residence time in the core: effects on long-term fuel behaviour in steady-state, transients and accidents, with potential effects on fuel related acceptance criteria.
Implications of Gen III design features on safety analysis

- Higher fuel enrichment, use of MOX fuel, use of fuel from different producers: need to consider different neutronic and thermomechanical properties of fuel, including conditions for manipulations and storage of fuel.

- Enhanced radiological acceptance criteria for operational states and for accidents, including severe accidents: unification of acceptance criteria and methodology for demonstration of compliance without unnecessary conservatism would help.

- Complex assessment of all aspects of accidents: more attention should be paid to all neutronic, thermal-hydraulic, structural and radiological aspects, with clear rules and transparent transfer of information between the codes.
Main international requirements and guidance documents on safety analysis

- IAEA Safety Standards, in particular
- European Utility Requirements for LWR Nuclear Power Plants. Revision C, April 2001
Summary of requirements on safety analysis

Scope of safety analysis

- In accordance with the graded approach, the safety analysis for NPPs shall be of the highest quality.
- The set of events shall be selected using deterministic and probabilistic methods.
- Safety analysis shall take into account all sources of radioactivity in the reactor and in all other places, considering full power, low power, and shutdown regimes, taking into account internal initiating events as well as internal and external hazards.
- Safety analysis shall cover the whole spectrum of the plant states from normal operation through design basis, beyond design basis up to severe accidents, i.e., including highly unlikely events caused by multiple failures of equipment and personnel.
- Initiating events shall be grouped in accordance with frequencies of their occurrence and their safety aspects (related to mechanisms of damage of the barriers), and bounding cases shall be determined for each group using appropriate selection criteria.
Summary of requirements on safety analysis

- **Deterministic analysis**
  - All aspects shall be analysed (neutronic, thermal-hydraulic, structural and radiological) in order to provide for complex evaluation of all acceptance criteria.
  - Safety analyses must demonstrate fulfilment of acceptance criteria with **sufficient margins** including those cases, when best estimate approach is acceptable.
  - If such margins are to be ensured by means of **conservative input data and other assumptions**, these shall be specifically selected in accordance with objectives for each category of events and each acceptance criterion.
  - It is acceptable to use **different approaches to analysis of design basis and beyond design basis events**.
  - Modelling of systems with innovations beyond the usual engineering solutions shall be **adequately supported by research, specific tests or by evaluation of operational experience from similar applications**.
Summary of requirements on safety analysis

- Integration of deterministic and probabilistic analysis
  - Safety analysis shall include complementary deterministic and probabilistic safety analysis in an integrated approach
  - Probabilistic analysis shall be used to balance the design and to identify factors mostly contributing to the risk
  - Broader use of probabilistic methods should also allow for more realistic approach in use of deterministic methods, in particular for determination of scenarios, assumptions for the analysis and for selection of acceptance criteria
  - PSA analysis shall cover all plant states and all significant internal initiating events including internal hazards as well as external hazards
  - Plant specific reliability data should be used to the extent possible (very complicated for new designs)
  - Special attention should be paid to human factor reliability, modelling of common cause failures and modelling of passive systems
Summary of requirements on safety analysis

Criteria for judgment of safety

- For individual groups of events acceptance criteria shall be defined, including both high level criteria limiting radiological consequences as well as derived criteria related to integrity of barriers.
- Criteria shall be graded in accordance with the frequency of the accidents.
- Criteria for design basis accidents should include maximally acceptable level of fuel damage.

Use of computer codes

- Best estimate codes shall be available for the analyses.
- Procedures and computer codes used in safety analysis shall be verified and validated in order to demonstrate that they are capable to predict reliably behaviour of the real systems in the given area of application.
- Scope of the validation should reflect specific implications from design features of new designs.
Summary of requirements on safety analysis

- **Evaluation of uncertainties and sensitivity analysis**
  - It shall be taken into account that there are always uncertainties associated with safety analysis.
  - Whenever the uncertainties are significant for utilization of the results, the uncertainties shall be quantified and sensitivity analysis performed.
  - Quantification of the uncertainties shall be performed using adequately adopted and verified methods.

- **Use of operational experience feedback**
  - Operational data shall include information on operational events and safety relevant characteristics associated with these events.
  - Selection of initiating events as well evaluation of their cause and consequences shall adequately take into account operational experience from similar facilities.
  - Operational data shall also be used for improvements of methods of safety evaluation.
Summary of requirements on safety analysis

- Documentation of safety analysis
  - Sufficient evaluation of results and conclusions shall be documented in safety report
  - Safety documentation must include sufficient demonstration and justification of quality and robustness of analysis
  - Safety report shall contain sufficient details and analysis shall be traceable to allow for independent verification
  - If analyses are performed in sequence by several codes and groups of analysts, transfer of information must be clear and transparent
  - Safety report shall be adequately archived and regularly updated
Summary of requirements on safety analysis

- **Independent verification of safety analysis**
  - Safety analysis shall be independently verified by the operator or any qualified organization on his behalf, prior to submission to the regulatory body.
  - Scope and level of details of the independent verification shall correspond to the associated risks.
  - Independent verification shall contain overall assessment and detailed evaluation of selected parts of the documentation, including independent calculations.
  - Verification shall include adequacy of models and input data.
  - Independent verification by the regulatory body shall be a separate process, taking place after verification of the operator.
Conclusions

- **Generation II and Generation III reactors are essential** for ensuring mid-term security of electricity supply, for sustainability of nuclear power and smooth introduction of future reactor generations.

- Failure to introduce Generation III reactors and operate safely Generation II and III reactors would seriously impact introducing any future use of nuclear power.

- **New Generation III reactors are significantly improved in safety and economy** as compared with existing ones, using new design features for enhanced defence in depth.

- Although majority of design features of Generation III reactors are evolutionary using proven technologies, there are significant challenges that require careful consideration in ensuring and demonstrating safety.

- Continued research is still needed to improve the knowledge in several areas associated with Generation III reactors, most significant ones being issues related to long term operation of reactors, use of advanced fuels, mitigation of severe accidents, and robustness of designs against external hazards.
Conclusions

- It is essential that adequate information on new design features is available not only to plant vendors, but also to operators.
- There are always uncertainties present in safety analysis, these shall be compensated by adequate safety margins, including situations where best estimate approach in analysis is accepted.
- There is maturity of modern best estimate safety analysis and quantification of uncertainties, these should be more broadly used in demonstration of plant safety.
- Sound conservatism is needed in safety analysis in order to demonstrate differential benefit of new design features.
- International cooperation in the research and development is the way for achieving the required results in reasonable period of time.
- IAEA should play its role in harmonization of safety requirements and approaches.