PR&PP Collaborative Study with GIF System Steering Committees

A Compilation of Design Information and Crosscutting Issues Related to PR&PP Characterization

Presented at

Workshop on
PR&PP Evaluation Methodology
for Gen IV Nuclear Energy Systems

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Background

- The Generation IV International Forum has stated four goals to assess performance of GIF nuclear energy systems:
  - proliferation resistance and physical protection
  - sustainability
  - economics
  - safety and reliability

- The PR&PP Methodology developed by the Proliferation Resistance and Physical Protection Working Group provides a comprehensive framework and guidance for carrying out a system evaluation with regard to PR&PP performance.

- In order to facilitate the timely introduction of PR&PP characteristics into the design process, a collaborative effort was initiated in 2007 between the GIF System Steering Committees and the PRPPWG.

Objectives and Scope

- Generate preliminary information about the PR&PP merits of Gen IV systems and recommend directions for optimizing their PR&PP performance
  - Capture the current salient features of the GIF system design concepts that impact their PR&PP performance
  - Identify crosscutting studies to assess PR&PP design or operating features common to various GIF systems
  - Suggest beneficial characteristics of the design of future nuclear energy systems.

Evolution of PRPPWG/SSC Collaborative Study

- Discussions on PRPP/SSC cooperation initiated in 2007
- Workshops – May 2008, July 2009, and January 2010
- PR&PP-focused Gen IV System White Papers
- Compendium Report incorporating the six System White Papers and sections on cross-cutting issues

Organization of White Papers

- Overview of Fuel Cycle Element Options
- Current Technology Status
- Proliferation Resistance Considerations
  - Concealed diversion or production of material
  - Breakout
  - Production in clandestine facilities
- Physical Protection Considerations
  - Theft of material for nuclear explosives
  - Radiological sabotage
- PR&PP Issues, Concerns and Benefits
- References

Overview of Gen IV Systems

System	Neutron spectrum	Coolant	Outlet Temperature °C	Refueling Mode	Fuel cycle	Size (MWe)
VHTR (very-high-temperature reactor)	thermal	helium	900-1000	On-site; Offline batch / Online continuous	open	250-300
SFR (sodium-cooled fast reactor)	fast	sodium	500-550	On-Site; Offline batch / Offline full core (Off-site)	closed	50-150 300-1500 600-1500
SCWR (supercritical water-cooled reactor)	thermal/fast	water	510-625	On-site; Offline batch / Online continuous	open/ closed	300-700 1000-1500
GFR (gas-cooled fast reactor)	fast	helium	850	On-site; Offline batch	closed	1200
LFR (lead-cooled fast reactor)	fast	lead	480-570	On-site; Offline batch / Offline full core (Offsite)	closed	10-100 300-1200 600-1000
MSR (molten salt reactor)	thermal/fast	fluoride salts	700-800	On-site; Online continuous	closed	1000

(Source: GIF 2009 Annual Report, http://www.gen-4.org/PDFs/GIF-2009-Annual-Report.pdf)

Very High-Temperature Reactor (VHTR)

- White paper described two basic design concepts of helium-cooled, graphite-moderated and reflected VHTR by fuel type.
- Prismatic VHTR:
  - Fuel is in the form of TRISO-coated fuel particles embedded in graphite blocks.
  - Five Prismatic VHTR concepts are currently under consideration. Two concepts examined are the GA GT-MHR and the Areva Modular HTR (both 600 MWt).
  - Batch mode refueling; accountancy is by item counting (visual ID of serial no.).
- Pebble Bed VHTR:
  - Fuel is in the form of TRISO-coated fuel particles compacted in small graphite spheres. Fissile content of each fuel sphere is very small.
  - Two example systems (both 200-250 MWt) are the South African PBMR and the Chinese HTR-PM.
  - Onload refueling; accountancy of spent fuel spheres is by bulk accountability.
- Baseline fuel cycle is once-through using LEU but with options to deep-burn TRU/MA and use thorium in fertile fuel particles.
- Intrinsic PR&PP attributes of VHTR:
  - Robust fuel
  - High burnup LEU fuel
- Reprocessing technologies not yet fully developed or demonstrated.

Sodium-Cooled Fast Reactor (SFR)

- The white paper described three conceptual designs
  - Large Loop Configuration SFR – JAEA SFR, 1500 MWe
  - Pool Configuration SFR – KALIMER-600, 600 MWe
  - Small Modular SFR – SMFR, 50 MWe
- Fuel cycle roles:
  - Transmuter (consumes transuranics)
  - Converter (conversion ratio near 1)
  - Breeder
- SFRs are intended for utilization in a closed fuel cycle.
- Fuel types:
  - Oxide (TRU-MOX)
  - Metal (U-TRU-Zr)
  - Nitride (MN) or Carbide (MC)
- Fresh fuel (FF) contains TRU but low in radioactivity.
- Spent fuel contains similar fissile content as FF but has significant heat load and radioactivity.
- Spent blanket (in converter and breeder) has relatively low burnup and high quality Pu.
- Recycling negates need for enrichment technology.
- Increased burnup in SFR reduces frequency of refueling.
- Refueling is complicated and requires inert atmosphere.
- Non-proliferation aspects of the closed fuel cycle (including fuel reprocessing technology and potential for separation of Pu) is a key issue for the SFR.
- Fuel cycle technology R&D is outside Gen-IV SFR scope

Supercritical Water Reactor (SCWR)

- White paper described two reactor concepts, but their development has not progressed sufficiently to enable a meaningful discussion of PR&PP issues, concerns and benefits.
- Pressure vessel concept:
  - European HPLWR is a thermal reactor similar to BWR in PR&PP attributes.
  - Batch refueling limits access to core but more attractive spent fuel (fewer items per SQ and variable burnup).
- Pressure tube concept:
  - Light water cooled and heavy water moderated
  - Onload refueling requires more rigorous accountability of fuel movement but less attractive spent fuel (more items per SQ and more uniform burnup).
- Thermal option:
  - UO_2 fuel in a once through fuel cycle
  - Fuel enrichment up to 6%
  - Exit burnup up to 60 GWd/tHM
- Fast option:
  - MOX fuel with conventional U-Pu fuel cycle
  - Tight lattice core with high power density
  - Up to 120GWd/tHM burnup; small fast SCWR could run for 30 years without refueling
- Thorium option:
  - Considered for the pressure tube concept
  - Self-protection through gamma-emitting ^{232}U
  - Difficult to separate fissile component ^{233}U from non-fissile isotopes (denatured by ^{238}U)

Gas-Cooled Fast Reactor (GFR)

- White paper described one 2400 MWt reference design, an indirect combined cycle with helium turbine (secondary gas cycle) plus a steam generator/turbine (tertiary cycle).
- Two fuel types – pin with carbide fuel (oxide fuel as a backup) and honeycomb plate carbide or nitride fuel.
- GFR will operate in a closed fuel cycle using the GANEX process (separated uranium with all TRU recycled together).
- GFR fuel cycle is similar to that of the SFR with aqueous reprocessing and using depleted uranium and high Pu content MOX fuel.
- For GFR with breeding blankets - potential to produce Pu using uranium targets.
- A guard vessel inside the containment building should provide additional physical protection for the primary system.
- Availability of shutdown cooling system is important for some accidents, in particular in depressurization accidents.

Lead Cooled Fast Reactor (LFR)

- The white paper described two conceptual designs:
  - Small Secure Transportable Autonomous Reactor (SSTAR), 19.8 MWe.
  - European Lead-Cooled SYstem (ELSY), 600 MWe.
- Fuel types and closed fuel cycle operation are similar to the SFR.
- SSTAR has a cassette type core that is replaced at end of core life (15-30 yr) by the reactor supplier.
- ELSY undergoes partial refueling every 15 months.
- By design, introduction of uranium target pins in the core or reflector region is either impossible or easy to detect.
- Simple system and robust design (e.g. low system pressure, high heat capacity of lead coolant, and small footprint) are advantageous for PP.
- R&D needs:
  - System layout (e.g. locating SSTAR underground).
  - Fuels, fuel reprocessing, core replacement (for SSTAR).
  - Detailed safety analysis.

Molten Salt Reactor (MSR)

- White paper discussed the MSR breeder option, the Molten Salt Fast Reactor (MSFR) design concept using the Th/^{233}U fuel cycle with fluoride salts.
- Liquid fuel processing is part of the reactor.
- MSR has low fissile inventory per unit power output.
- Continuous reprocessing of the fuel salt and the introduction of makeup fuel create a fissile inventory outside the core.
- Fuel salt has high freezing temperature and its transfer from the reactor hot cell will be in solid form with strong radiation signature.
- To obtain 1 SQ of fissile material would require a large quantity of fuel salt because of low fuel concentration in the salt.
- Presence of ^{233}U requires additional safeguards (having lower critical mass than ^{235}U).
- Radiation from decay of ^{232}U constrains handling of fuel but produces a visible signature for the detection of fissile material transport.
- Other MSR concepts, such as the MOSART (for actinide burning), still need to be considered.
- More design and safety studies will be needed before starting a PP evaluation.

Cross-cutting Topics [1]

- Fuel type
  - physical form
  - chemical form
  - isotopic composition
- Coolant, Moderator
- Refueling modes
  - Batch (periodic partial core replacement)
  - On-load (continuous)
  - Full core replacement (cassette type core)

Cross-cutting Topics [2]

- Fuel cycle architecture
  - thermal vs. fast neutron spectrum
  - U-Pu vs. Th-U fuel cycle
  - once-through vs. closed fuel cycle
  - burner vs. breeder, with or without blankets
  - co-located vs. centralized reprocessing
- Safeguards
  - Identify MBAs
  - Safeguards approaches for non-LWRs
  - MCP&A – item vs. bulk
- Other GIF topics
  - Safety
  - Economics

Conclusion: Summary

- The SCCs and the PRPPWG have had fruitful engagements through interactions in three major workshops and co-development of PR&PP-focused System White Papers

- The three main objectives of this work are to:
  - capture features of the design concepts that impact their PR&PP performance;
  - identify crosscutting studies that assess PR&PP design or operating features common to various GIF systems;
  - suggest beneficial characteristics of the design of future nuclear energy systems.

- The Compendium Report incorporates the six System White Papers and sections on cross-cutting issues and will be helpful to system designers and program policy makers as they plan for the future maturation of the GIF design concepts.

GEN IV International Forum

Thank you…

Questions?