GE Advanced Liquid Metal Reactor

S-PRISM

by

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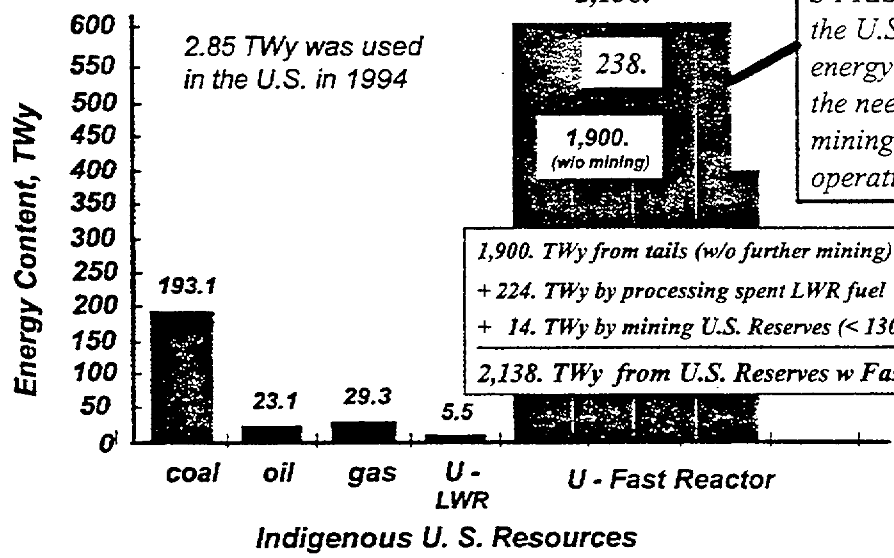
Topics

- Incentive for developing S-PRISM
- Design and safety approach
- Design description and competitive potential
- Previous Licensing interactions
- Planned approach to Licensing S-PRISM
- What, if any, additional initiatives are needed?

United States Energy Resources

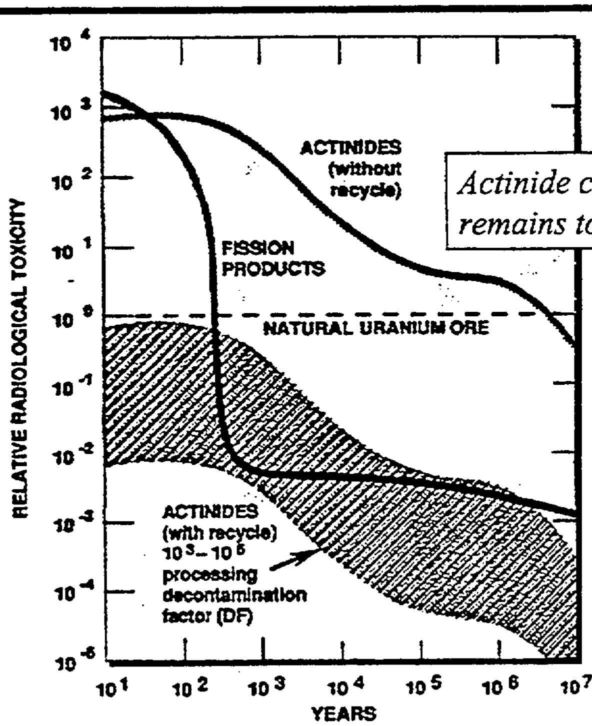


S-PRISM would provide the U.S. with a long term energy source without the need for additional mining or enrichment operations.

1,900. TWy from tails (w/o further mining)
+ 224. TWy by processing spent LWR fuel
+ 14. TWy by mining U.S. Reserves (< 130\$/kg)
2,138. TWy from U.S. Reserves w Fast Reactor

Energy estimates for fossil fuels are based on "International Energy Outlook 1995", DOE/EIA-0484(95). The amount of depleted uranium in the US includes existing stockpile and that expected to result from enrichment of uranium to fuel existing LWRs operated over their 40-y design life. The amount of uranium available for LWR/Once Through is assumed to be the reasonably assured resource less than \$130/kg in the US taken from the uranium "Red Book".

Time Phased Relative Waste Toxicity (LWR Spent Fuel)



Actinide containing LWR spent fuel remains toxic for millions of years

- *Processing to remove the fission products (~3% of LWR spent fuel), uranium (95%), and transuranics prior to disposal shortens the period that the "waste" remains toxic to less than 500 years.*

- *The recovered U and TRU would then be used as fuel and burned.*

Relative Decay Heat Loads of LWR and LMR Spent Fuel

Decay Heat Load	***Decay Heat (Watts per kg HM)***	
	LWR	***S-PRISM***
Spent Fuel at Discharge	*2.3*	*11.8*
Normal Process Product After Processing Spent Fuel • ***Pu from PUREX Process for LWR*** • ***Pu + Actinides from PYRO Process***	*9.62*	*25.31*
Weapons Grade Pu-239	*1.93*	

During all stages in the S-PRISM fuel cycle the fissile material is in a highly radioactive state that always exceeds the "LWR spent fuel standard".
<u>Diversions</u>
would be extremely difficult.

Stage of the Fuel Cycle	Material Barriers: Isotopic	Radiological	Chemical	Mass and Bulk	Detectability	Technical Barriers: Facility Unattractiveness	Facility Access	Available Mass	Diversion, Detectability	Skills, Knowledge, Expertise	Time
Co-Located Fuel Cycle Facility [illegible]											
Phase 3: Equilibrium Operations											
Fuel handling	VL	VL	L	L	VL	VL	VL	I	VL	M	L
Spent fuel storage				L			M	I		M	L
Head-end processing				M			VL	I		I	L
Fuel processing				M			VL	I		I	L
Fuel fabrication				L			VL	I		I	L
Reactor operations				L			VL	I		M	L
Waste conditioning				L			VL	VL		I	VL
Waste shipment				VL			VL	VL		I	VL

<u>Phase 1</u>
These opportunities for proliferation are not required for S-PRISM.

<u>Phase 2</u>
All operations are performed within heavily shielded enclosures or hot cells at the S-PRISM site.

<u>Phase 3</u>
All operations are performed within heavily shielded and inerted hot cells at the co-located S-PRISM/IFR site.

Key Non-Proliferation Attributes of S-PRISM

1.) The ability to create S-PRISM startup cores by processing spent LWR fuel at co-located Spent Fuel Recycle Facilities eliminates opportunity for diversion within:

- *Phase I (mining, milling, conversion, and uranium enrichment phases) since these processes are not required.*

and

- *Phase II and III (on-site remote processing of highly radioactive spent LWR and LMR fuel eliminates the transportation vulnerabilities associated with the shipment of Pu)*

2.) The fissile material is always in an intensely radioactive form. It is difficult to modify a heavily shielded facility designed for remote operation in an inert atmosphere without detection.

3.) The co-located molten salt electro-refining system removes the uranium, Pu, and the minor actinides from the waste stream thereby avoiding the creation of a uranium/Pu mine at the repository.

Incentive for Developing S-PRISM

- *Supports geological repository program:*
  - *deployment of one new S-PRISM plant per year for 30 years would eliminate the 86,000 metric tons of spent LWR fuel that will be discharged by the present fleet of LWRs during their operating life.*
  - *reduces required repository volume by a factor of four to fifty*
  - *All spent fuel processing and waste conditioning operations would be paid for through the sale of electricity.*
  - *limits interim storage to 30 years*
- *Reduces environmental and diversion risks*
  - *repository mission reduced from >> 10,000 to <500 years*
  - *facilitates long term CO_2 reduction*
  - *resource conservation (fossil <u>and</u> uranium)*
  - *allows Pu production and utilization to be balanced*
  - *utilizes a highly diversion resistant reprocessing technology*

Topics

- *Incentive for developing S-PRISM*
- *Design and safety approach*
- *Design description and competitive potential*
- *Previous Licensing interactions*
- *Planned approach to Licensing S-PRISM*
- *What, if any, additional initiatives are needed?*

S-PRISM Safety Approach

Exploits Natural Phenomena and Intrinsic Characteristics

- *Low System Pressure*
- *Large heat capacity*
- *Natural circulation*
- *Negative temperature coefficients of reactivity*

Key Features of S-PRISM

- *Compact pool-type reactor modules sized for factory fabrication and an affordable full-scale prototype test for design certification*
- *Passive shutdown heat removal*
- *Passive accommodation of ATWS events*
- *Passive post-accident containment cooling*
- *Nuclear safety-related envelope limited to the nuclear steam supply system located in the reactor building*
- *Horizontal seismic isolation of the complete NSSS*
- *Accommodation of postulated severe accidents such that a a formal public evacuation plan is not required*
- *Can achieve conversion ratio's less than or greater than one*

S-PRISM Design Approach

Simple Conservative Design

- *Passive decay heat removal*
- *Passive accommodation of ATWS Events*
- *Automated safety grade actions are limited to:*
  - *containment isolation*
  - *reactor scram*
  - *steam side isolation and blow-down*

Operation and Maintenance

- *Safety grade envelope confined to NSSS*
- *Simple compact primary system boundary*
- *Low personnel radiation exposure levels*

Capital and Investment Risk Reduction

- *Conservative Low Temperature Design*
- *Modular Construction and seismic isolation*
- *Factory fabrication of components and facility modules*
- *Modularity reduces the need for spinning reserve*
- *Certification via prototype testing of a single 380 MWe module*

S-PRISM Features Contribute to:

- *Simplicity of Operation*
- *Reliability*
- *Maintainability*
- *Reduced Risk of Investment Loss*
- *Low Cost Commercialization Path*

S-PRISM Design Approach *(continued)*

1. ***Design basis events (DBEs)***
   - *Equipment and structures design and life basis*
   - *Bounding events that end with a reactor scram*
   - *Example, all rod run out to a reactor scram*

2. ***Accommodated anticipated transients without scram (A-ATWS)***
   - *In prior reactors, highest probability events that led to boiling and Hypothetical Core Disassembly Accidents were ATWS events*
   - *In S-PRISM, ATWS events are passively accommodated within ASME Level D damage limits, without boiling*
   - *Loss of primary flow without scram (ULOF)*
   - *Loss of heat sink without scram (ULOHS)*
   - *Loss of flow and heat sink without scram (ULOF/LOHS)*
   - *All control rod run out to rod stops without scram (UTOP)*
   - *Safe shutdown earthquake without scram (USSE)*

3. ***Residual risk events***
   - *Very low probability events not normally used in design*
   - *In S-PRISM, residual events are used to assess performance margins*

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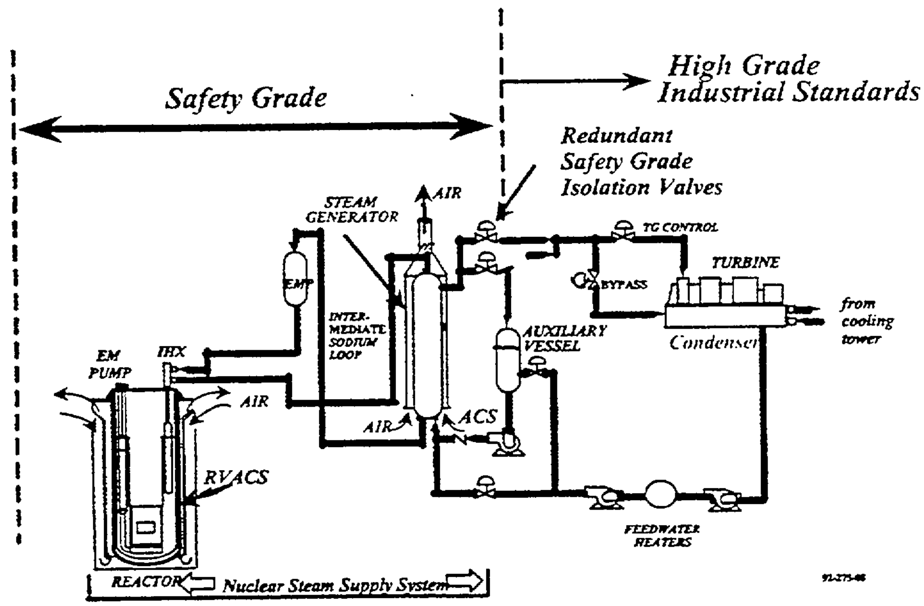
Power Train



RVACS	ACS	Condenser

Shutdown Heat Removal Systems

S-PRISM - Principal Design Parameters

<u>Reactor Module</u>

- Core Thermal Power, MWt	1,000
- Primary Inlet/Outlet Temp., C	363/510
- Secondary Inlet/Outlet Temp., C	321/496

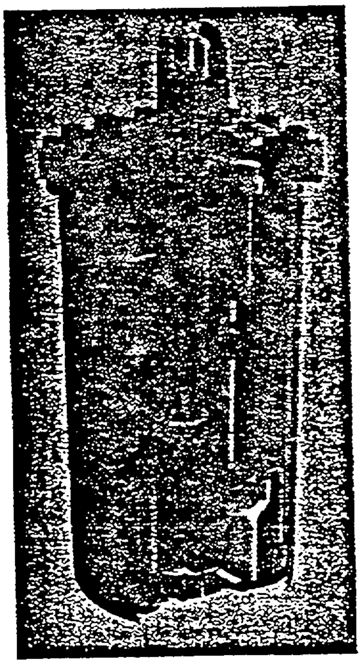
<u>Power Block</u>

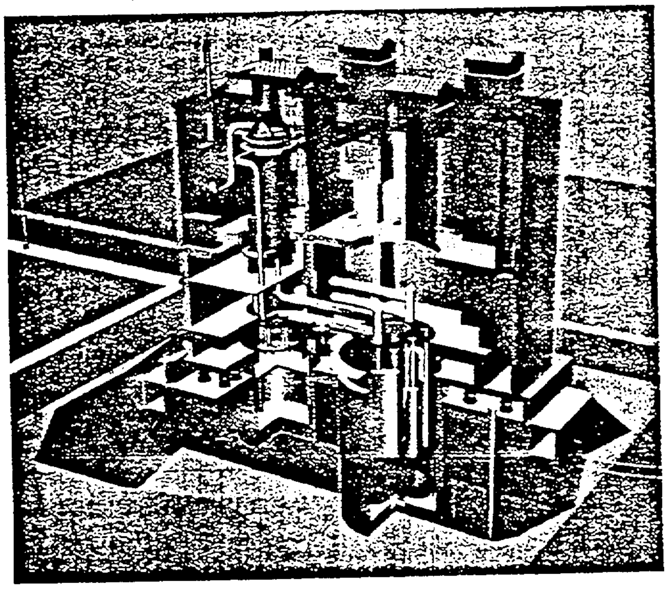
- Number of Reactors Modules	2
- Gross/Net Electrical, MWe	825/760
- Type of Steam Generator	Helical Coil
- Turbine Type	TC-4F 3600 rpm
- Throttle Conditions, atg/C	171/468
- Feedwater Temperature, C	215

<u>Overall Plant</u>

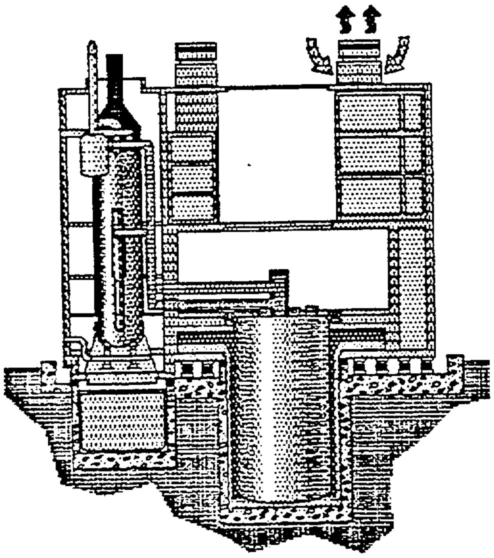
- Gross/Net Electrical, MWe	2475/2280
- Gross/Net Cycle Efficiency, %	41.2/38.0
- Number of Power Blocks	3
- Plant Availability, %	93

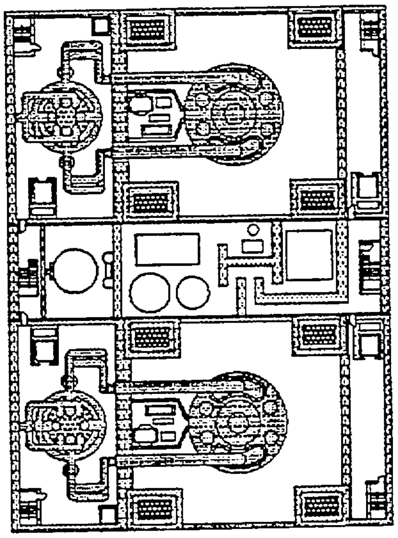
Super PRISM





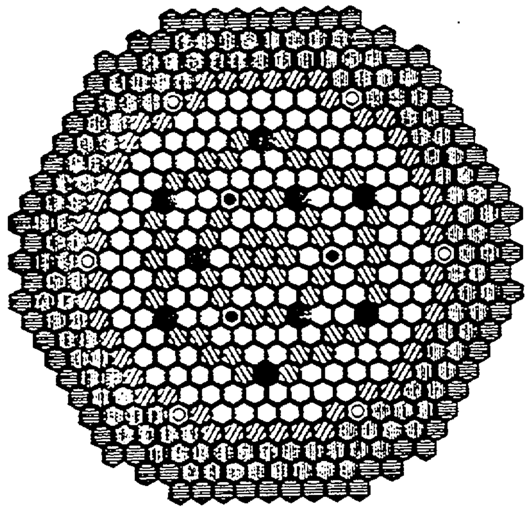
S-PRISM Power Block (760 MWe net)





Two 380 MWe NSSS per Power Block

Metal Core Layout



Number of Assemblies

Driver Fuel	138	Fuel: 23 month x 3 cycles
Internal Blanket	49	
Radial Blanket	48	Blkt: 23 month x 4 cycles
Primary Control	9	
Secondary Control	3	
Gas Expansion Module	6	
Reflector	126	
Shield	72	
Total	451	

Oxide vs. Metal Fuel

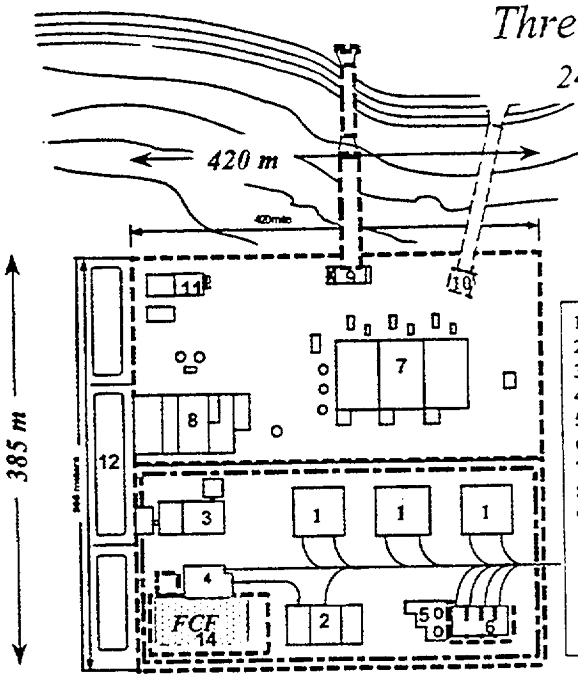
- *Attractive features of metal core include:*
  - *fuel is denser and has a harder neutron spectrum*
  - *compatible with coolant, RBCB demonstrated at EBR-II*
  - *axial blankets are not required for break even core*
  - *high thermal conductivity (low fuel temp.)*
  - *lower Doppler and harder spectrum reduce the need for GEMs for ULOF (6 versus 18)*
- *Metal fuel pyro-processing is diversion resistant, compact, less complex, and has fewer waste streams than conventional aqueous (PUREX) process*
- *However, an "advanced" aqueous process may be competitive and diversion resistant.*

S-PRISM can meet all requirements with either fuel type.

S-PRISM - Three Power Block Plot Plan

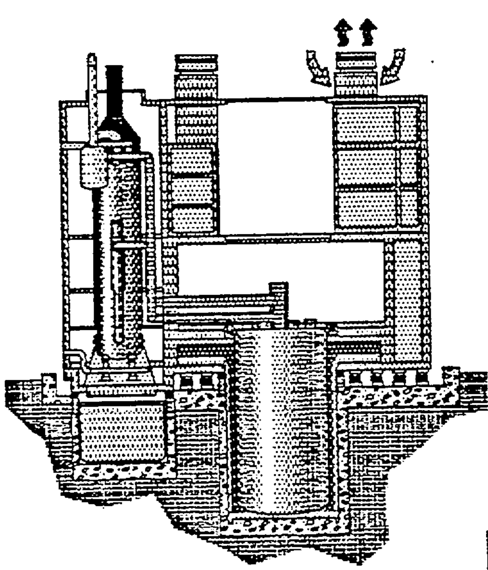
Three Power Block Plant

2475 MWe (2280 MWe net)



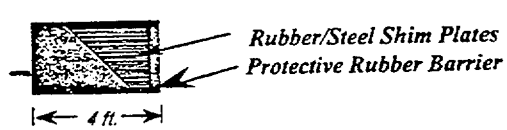
1 Reactor Building (2 NSSS/block)
2 Reactor Maintenance Facility
3 Control Facility
4 New and Spent Fuel Handling Facility
5 Assembly Facility
6 Cask Storage Facility
7 Turbine-Generator Facility
8 Maintenance Facility
9 Circulating Water Inlet Pump Station
10 Circulating Water Discharge
11 Waste Treatment
12 Parking Lot
13 Switch Yard
14 Fuel Cycle Facility

S-PRISM - Seismic Isolation System



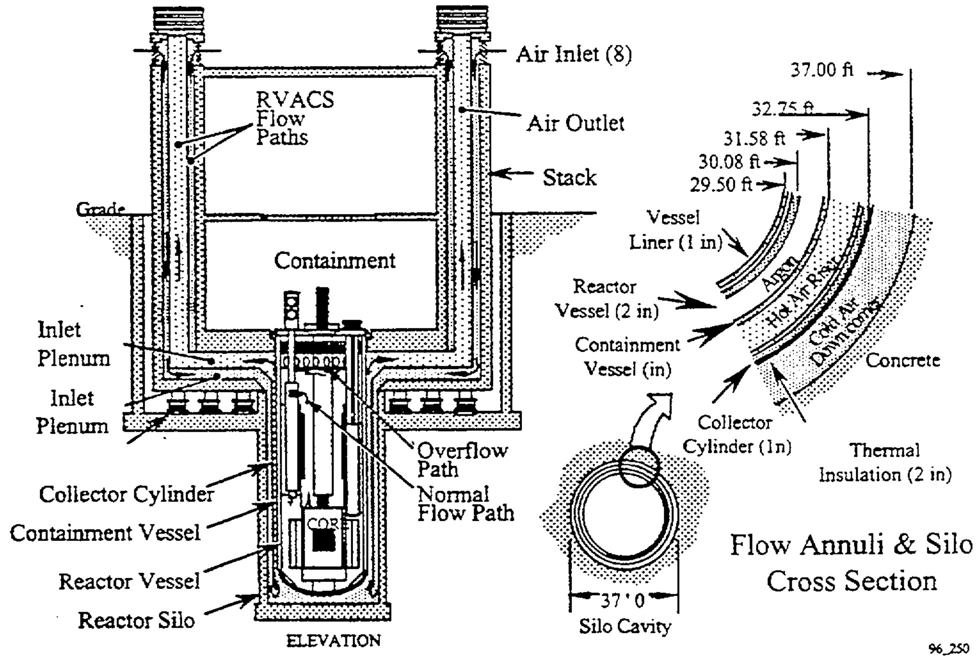
Characteristics of Seismic Isolation System

- *Safe Shutdown Earthquake*
  - *Licensing Basis* 0.3g (ZPA)
  - *Design Requirement* 0.5g
- *Lateral Displacement*
  - *at 0.3g* 7.5 inch.
  - *Space Allowance*
    - *Reactor Cavity* 20 inch.
    - *Reactor Bldg.* 28 inch.
- *Natural Frequency*
  - *Horizontal* 0.70 Hz
  - *Vertical* 21 Hz
- *Lateral Load Reduction* > 3

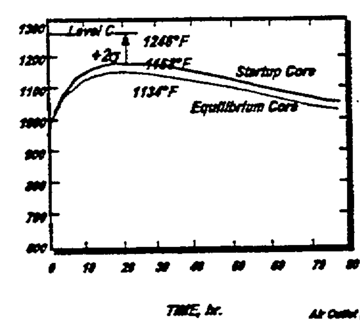


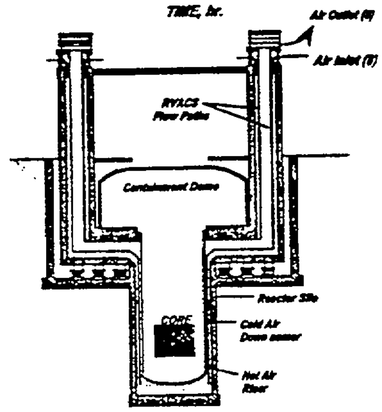
Seismic Isolators (66)

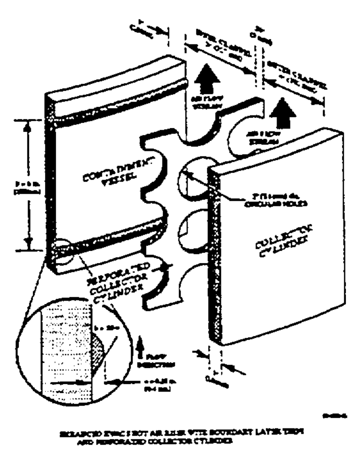
Reactor Vessel Auxiliary Cooling System *(RVACS)*



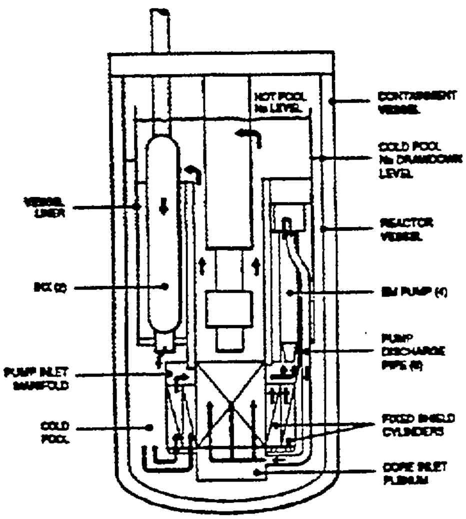
Passive Shutdown Heat Removal (RVACS)



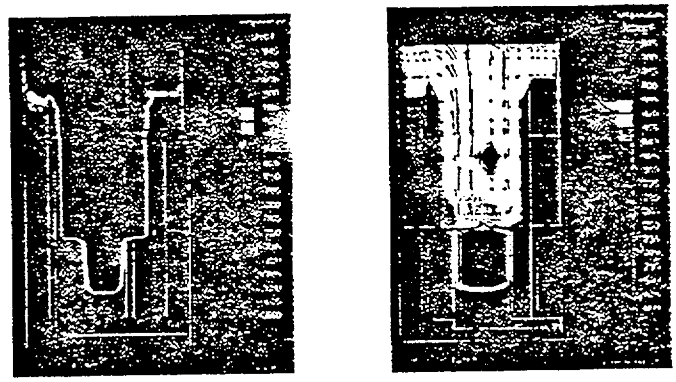




Natural Circulation Confirmed by 3 Dimensional T/H Analysis



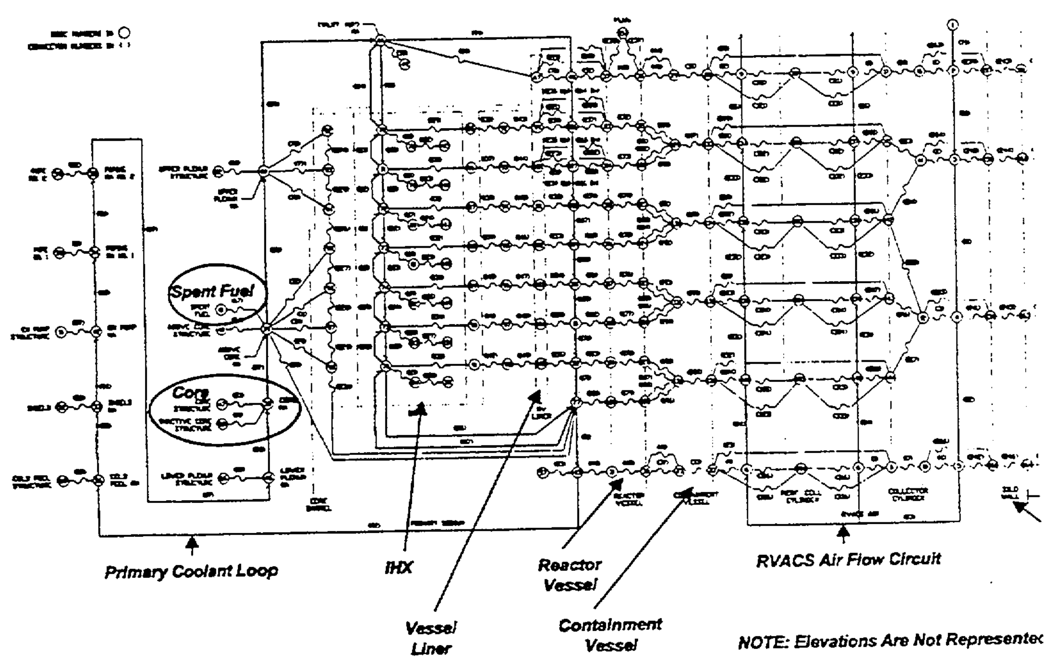
Normal Operation



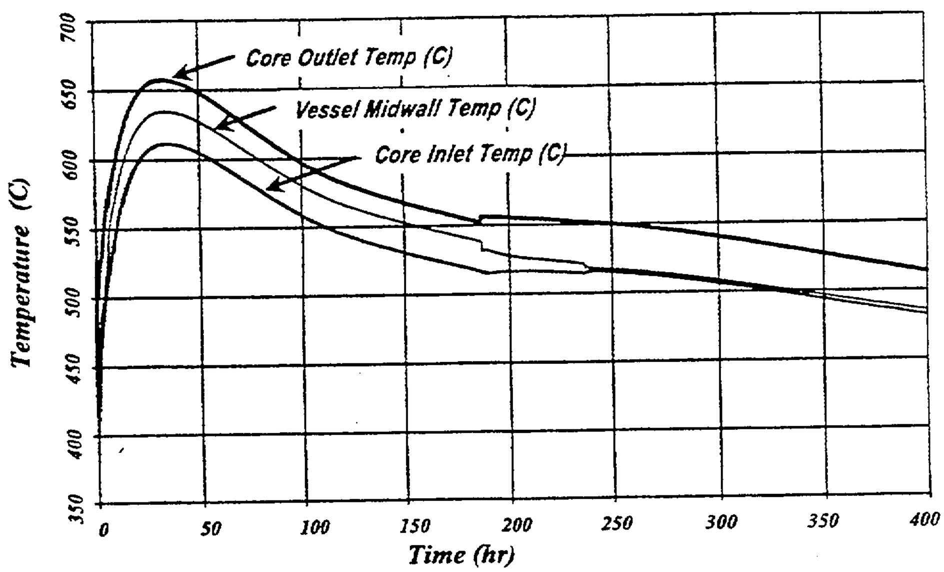
Examples

Temperature and velocity distribution at 4 and 20 minutes after loss of heat sink

Decay Heat Removal Analysis Model

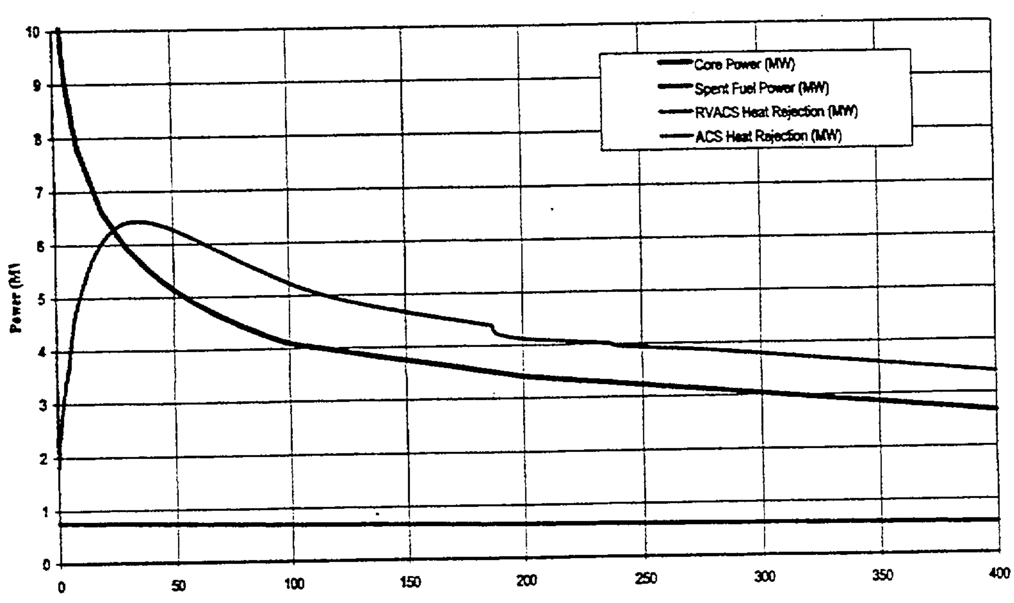


RVACS Cooling - Nominal System Temperatures

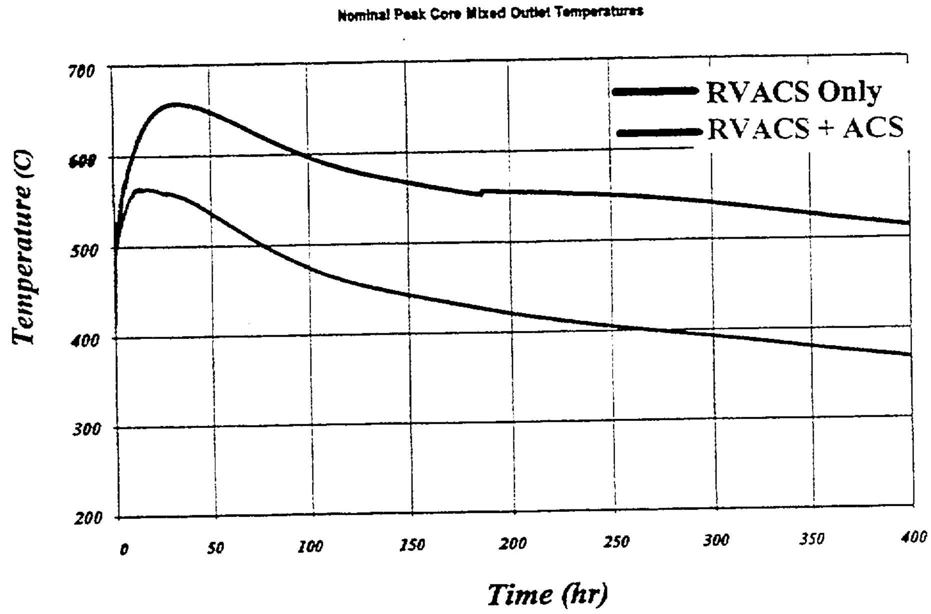


RVACS Transients Are Slow Quasi Steady State Events

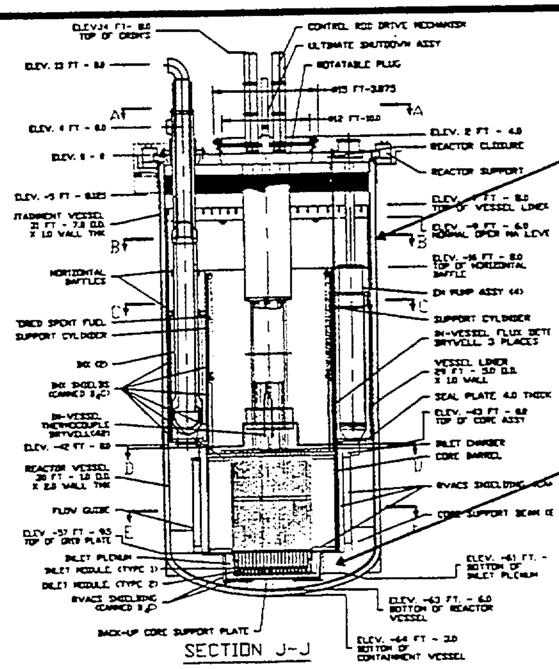
RVACS Heat Rejection and Heat Load versus Time



RVACS Cooling - Nominal Mixed Core Outlet Temperature



Damage Fraction from Six RVACS Transients



Peak Temperature & Damage Fraction at Vessel Mid Wall (nominal / 2-sigma)

Temperature °C	Damage Fraction
635 / 683	<0.002 / 0.002

Peak Temperature & Damage Fraction at Core Support (nominal / 2-sigma)

Temperature (°C)	Damage Fraction
612 / 658	<0.002 / 0.002

Damage from RVACS Transients Is Negligible

S-PRISM Approach to ATWS

Negative temperature coefficients of reactivity are used to accommodate ATWS events.

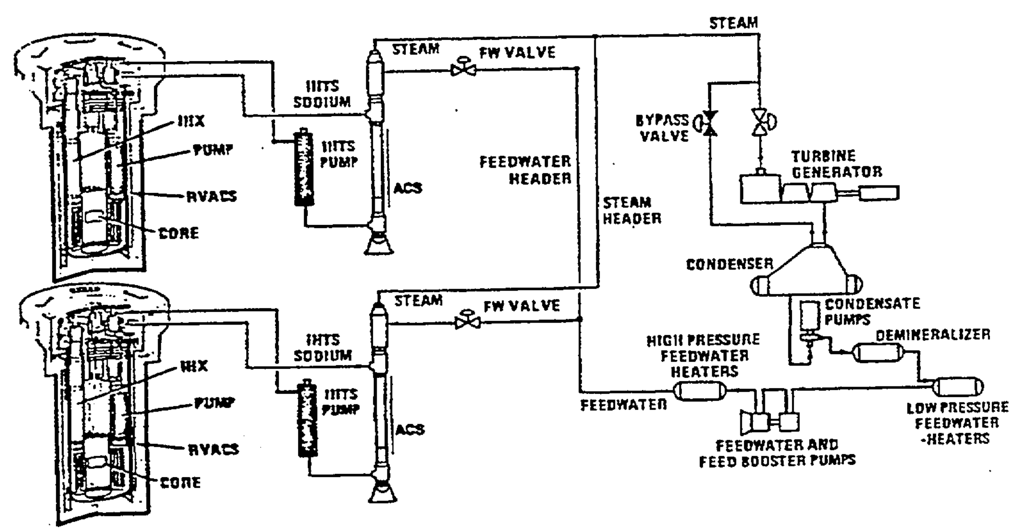
- *Loss of Normal Heat Sink*
- *Loss of Forced Flow*
- *Loss of Flow and Heat Sink*
- *Transient Overpower w/o Scram*

These events have, in priorLMR designs, led to rapid coolant boiling, fuel melting, and core disassembly.

S-PRISM Requirement:

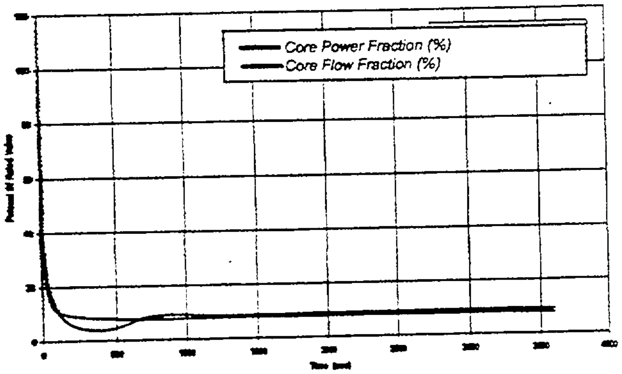
Accommodate the above subset of events w/o loss of reactor integrity or radiological release using passive or inherent natural processes. A loss of functionality or component life-termination is acceptable.

ARIES-P Power Block Transient Model



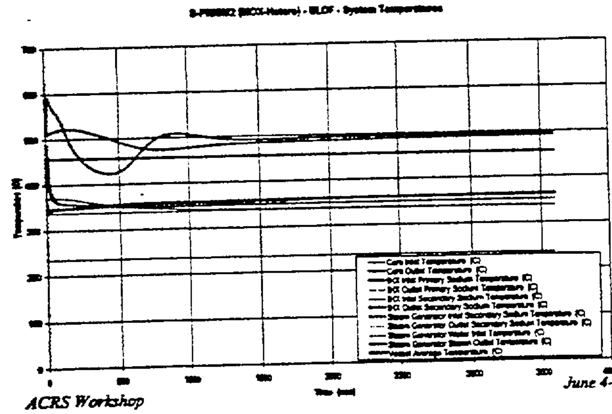
- ***Two-Reactors Coupled to a Single TG***
- ***One Group Prompt Jump Core Physics with Multi-Group Decay Heat***
- ***RVACS/ACS***
- ***Once-through Superheat***
- ***Control Systems:***
  - *- Plant control system (global and local controllers)*
  - *- Reactivity control system (RCS)*
  - *- Reactor protection system (RPS)*
  - *- EM pump control system and synchronous machines*

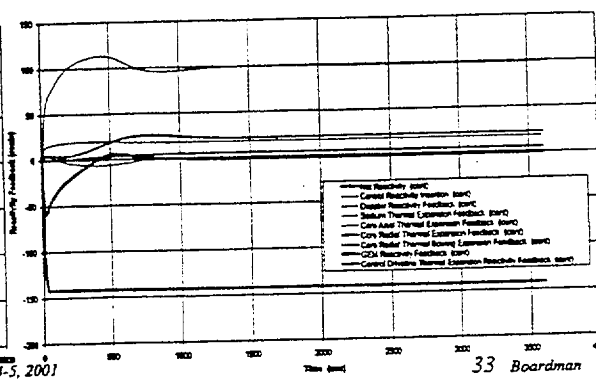
Example ATWS - Loss Of Flow Without Scram



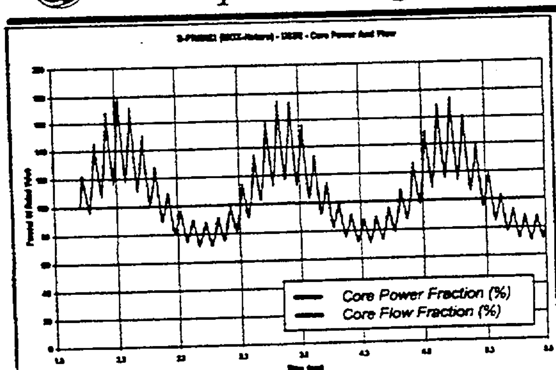
Loss of Primary Pump Power w/o Scram

- *Loss of pump pressure allows GEM feedback and fission shutdown*
- *Continuation of IHTS flow and feed water water enhance primary natural circulation to 10%*
- *Excess cooling of core outlet shortens CR drivelines and pulls control rods slightly to balance fission power with heat removal*





Example - 0.5 g ZPA Seismic Event Without Scram

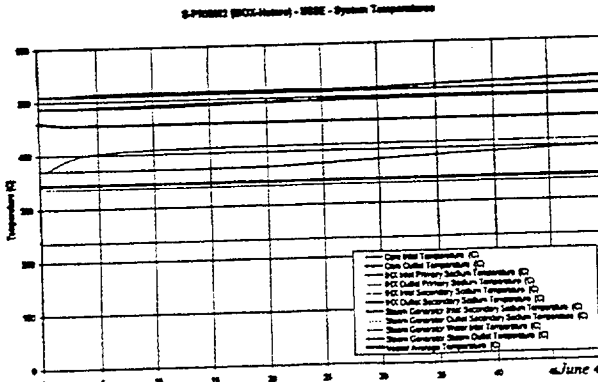


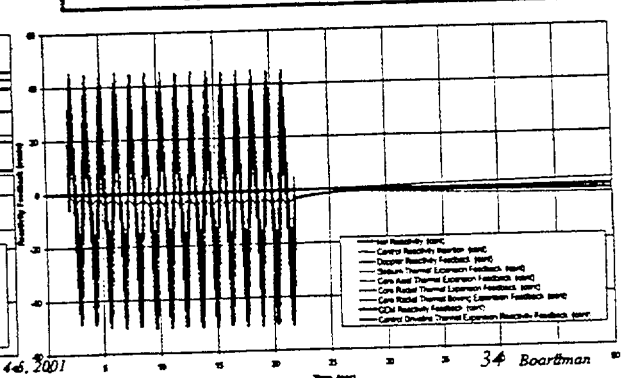
- *Reactivity:*

  +- 0.30$ at 3/4 Hz (horizontal core compaction)

  +- 0.16$ at 10 Hz (vertical CR-core motion with opposite phases)

- *Power oscillations to 180%, short duration, not supercritical*
- *Fuel heat capacity absorbs power oscillation without melting*
- *Fuel releases heat to structures slowly and gives small Doppler feedback to reduce power peaks*



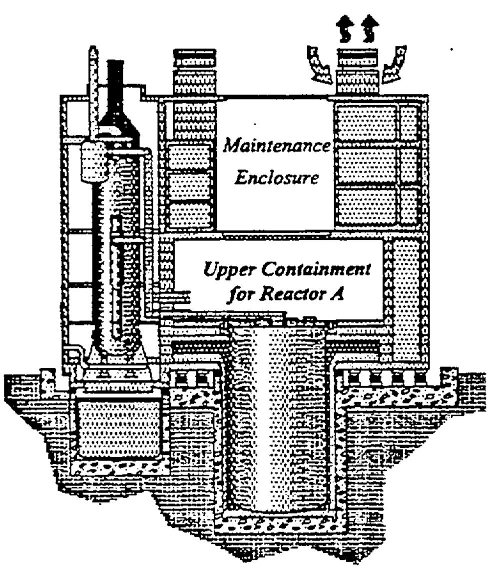


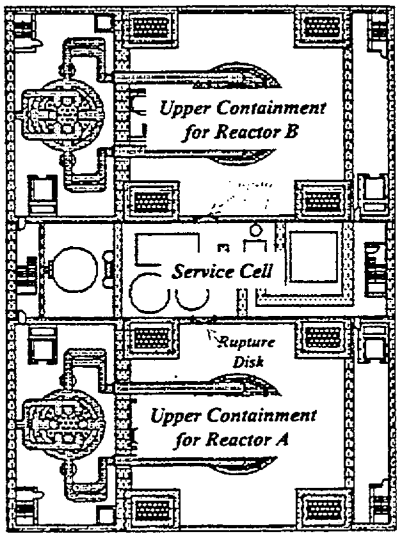
S-PRISM Transient Performance Conclusions

S-PRISM tolerates ATWS events within the safety performance limits

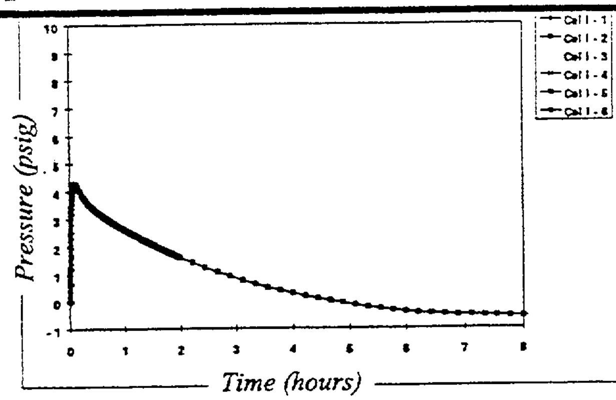
The passive safety performance of S-PRISM is consistent with the earlier ALMR program

S-PRISM Containment System





Example - Large Pool Fire



Beyond Design Basis (Residual Risk)
events have been used to assess containment margins

This event assumes that the reactor closure disappears at time zero initiating a large pool fire

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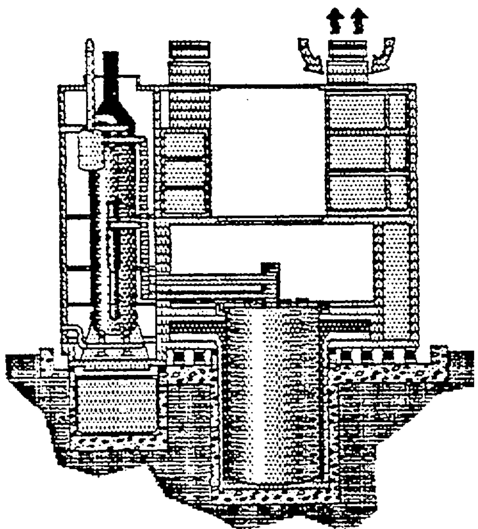
Note that the containment pressure peaks at less than 5 psig and drops below atmospheric pressure in less than 6 hours

Comparison of Emergency Power Requirements

Function	S-PRISM	Generation III LWRs
• Shutdown Heat Removal	Completely Passive	Redundant and Diverse Systems
• Post Accident Containment Cooling	Passive Air Cooling of Upper Containment	Redundant and Diverse Systems
• Coolant Injection/Core Flooding	N/A	Redundant and Diverse Systems
• Shutdown System	3/9 Primary or 2/3 Secondary Rods Self Actuated Scram on Secondary Rods Passive Accommodation of ATWS Events	Most Rods Must Function Boron injection N/A
Emergency AC Power	< 200 kWe from Batteries	~ 10,000 kWe

Layers of Defense



All Safety Grade Systems Are Located within the Reactor/NSSS Building

- Containment
  (passive post accident heat removal)
- Coolant Boundary (Reactor Vessel
  (simple vessel with no penetrations below the Na level)
- Passive Shutdown Heat Removal
  (RVACS + ACS)
- Passive Core Shutdown
  (inherent negative feedback's)
- RPS Scram of Scram Rods
  (magnetic Self Actuaed Latch backs up RPS)
- RPS Scram of Control Rods
  (RPS is independent and close coupled)
- Automatic Power Run Back
  (by automated non safety grade Plant Control System

Increasing Challenge

Normal Operating Range

- Maintained by Fault Tolerant Tri-Redundant Control System

Adjustments Since End of DOE Program In 1995

Parameter or Feature	1995 ALMR	S-PRISM
Core Power, MWt	840.	1000.
Core Outlet Temp, °C	499	510
Main Steam, °C / kg/cm^2	454/153	468/177
Net Electrical, MWe (two power blocks)	1243.	1520
Net Electrical, MWe (three power blocks)	1866	2280
Seismic Isolation	Yes. Each NSSS placed on a separate isolated platform	Yes. A single platform supports two NSSSs
Above Reactor Containment	Low leakage steel machinery dome	Low leakage steel lined compartments above the reactor closure

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Optimizing the Plant Size

1988 PRISM → S-PRISM

1988 PRISM

1263 MWe (net) from 3 blocks
9 NSSS (425 MWt each)
3 421 MWe TG Units
9 primary Na containing vessels
9 SG units/eighteen IHTS loops

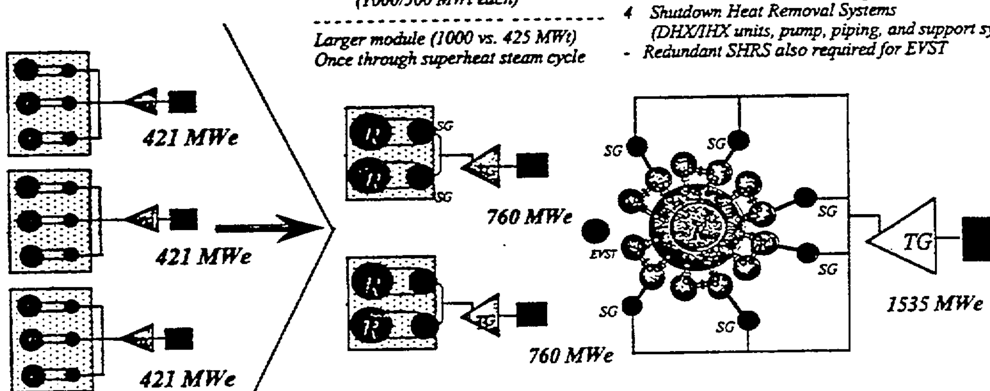
S-PRISM

1,520 MWe (net) from two blocks
4 NSSS (1000 MWt each)
2 825 MWe (gross) TG Units
4 primary Na containing vessels
4 SG units and eight IHTS loops (1000/500 MWt each)

Larger module (1000 vs. 425 MWt)
Once through superheat steam cycle

Large Commercial Design

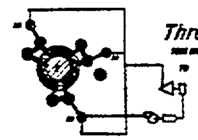
1,535 MWe Monolithic LMR
1 NSSS (4000 MWt)
1 1535 MWe TG Unit
*14 primary Na containing vessels**
(12 primary component vessels, reactor, and EVST)
6 SG units and 6 IHTS loops (667 MWt each)
4 Shutdown Heat Removal Systems (DHX/IHX units, pump, piping, and support systems)
- Redundant SHRS also required for EVST



Simplicity allows Reduction in Commodities and Building Size

Scale Up -- LWR versus Fast Reactor

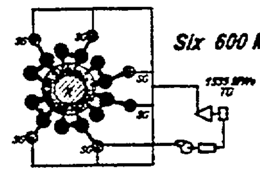
1600 MWt Sodium Cooled Fast Reactor | **1600 MWt Light Water Cooled Reactor**



Three 533 MWt Loops

Two 800 MWt Loops

3600 MWt FR



Six 600 MWt Loops

Rating Limited by:
IHTS Piping: < 1 m diameter

3600 MWt PWR

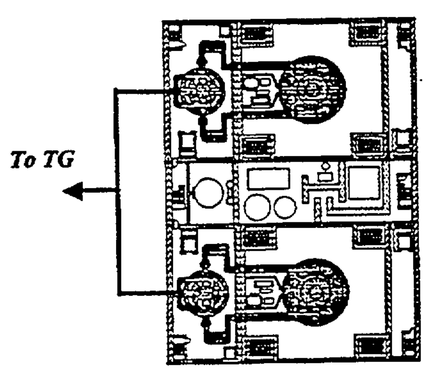
Two 1800 MWt Loops

Two Loops Viable Because:
Specific heat of water 5 x sodium at operating temperatures

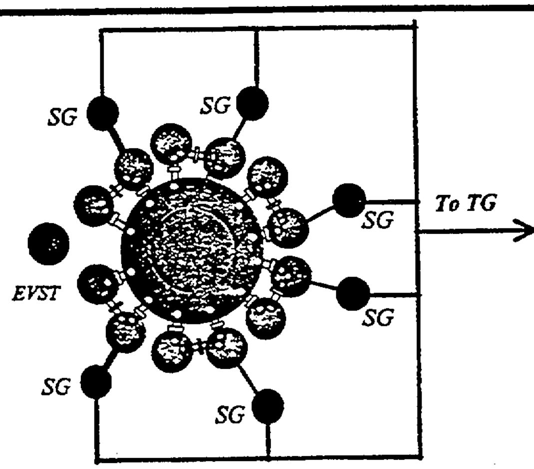
- ***The complexity and availability of a PWR is essentially constant with size***
- ***Due to the lower specific heat of sodium, six or more loops are required in a large FR.***

The Economy of Scale is Much Larger for LWRs then FBRs

Modular versus Monolithic (Fast Reactors)



Modular (S-PRISM)

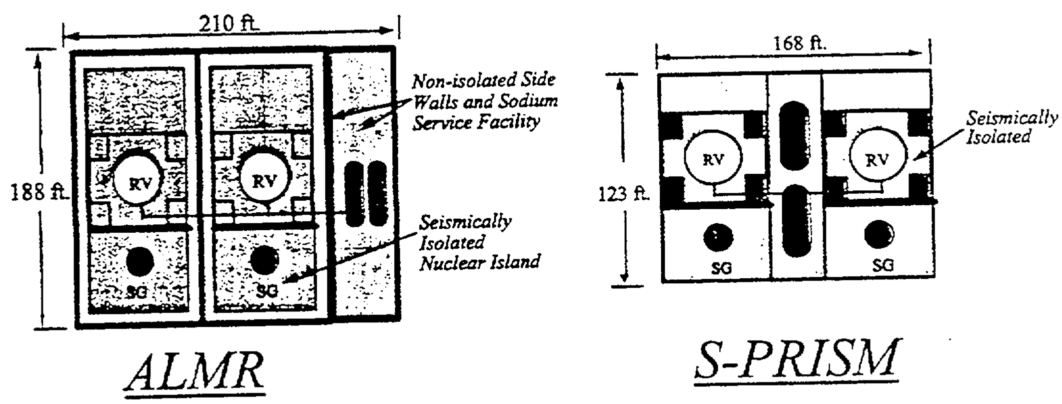


Monolithic Fast Reactor

The one-on-one arrangement:

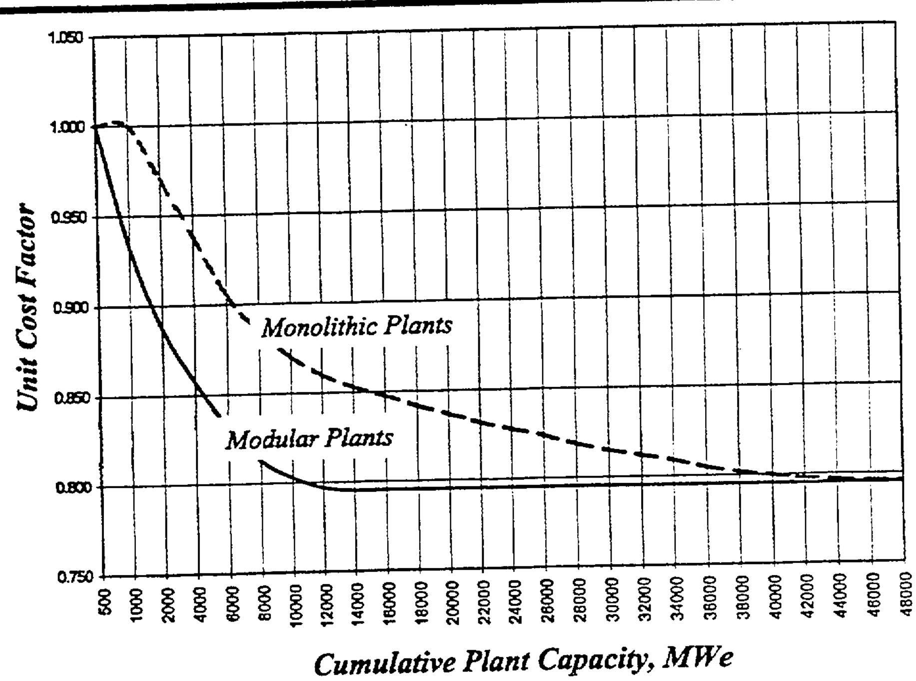
- *simplifies operation,*
- *minimizes the size of the reactor building*
- *improves the plant capacity factor*
- *reduced the need for backup spinning reserve*

NSSS Size, ALMR verses S-PRISM

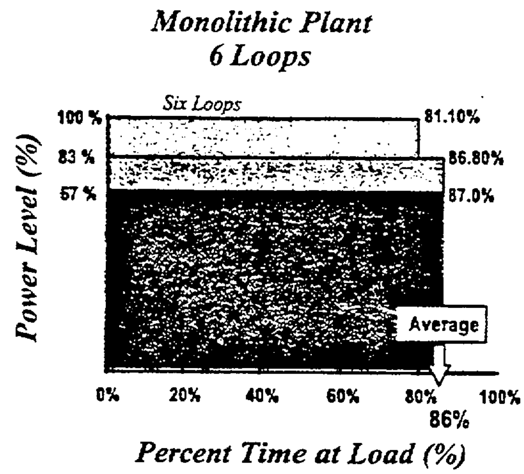


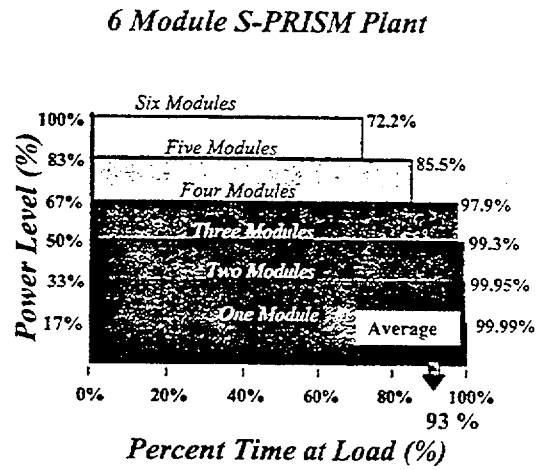
22 % More Power from Smaller NI

Learning Effect Favors Modular Plant Designs



Modular vs. Monolithic Availability and Spinning Reserve

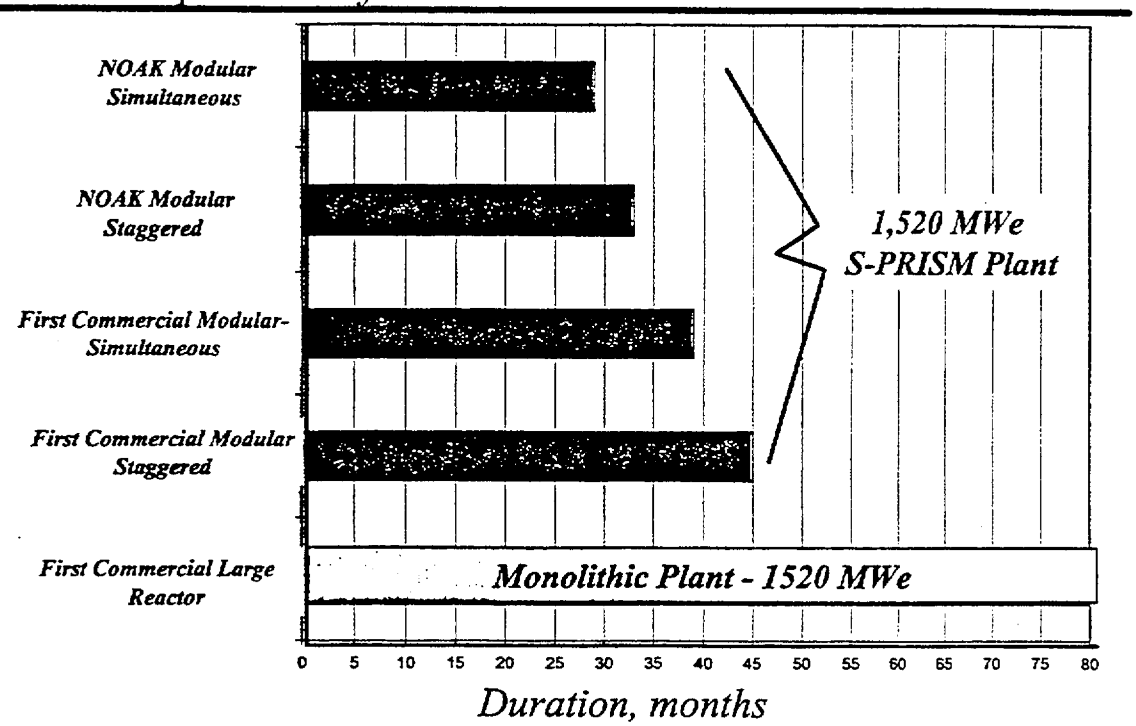




Seven point advantage caused by:

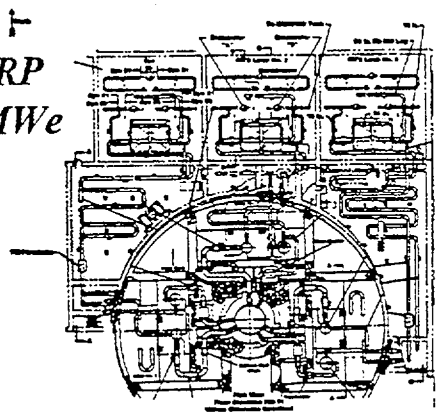
- *Relative simplicity of each NSSS (one SG System rather than 6)*
- *Ability to operate each NSSS independently of the others*

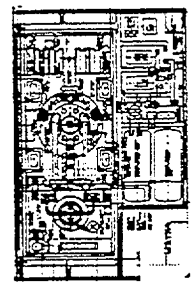
Comparison of Plant Construction Schedules



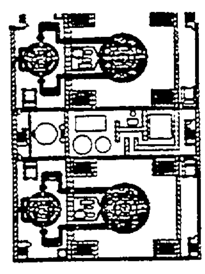
NSSS Size, CRBRP/ALMR /S-PRISM

CRBRP
350 MWe





ALMR
311 MWe



S-PRISM
760 MWe

Topics

- Incentive for developing S-PRISM
- Design and safety approach
- Design description and competitive potential
- Previous licensing interactions
- Planned approach to licensing S-PRISM
- What, if any, additional initiatives are needed?

ALMR Design and Licensing History

199[redacted] →
S-PRISM
GE Funded

1989 - [redacted]995
A[redacted]ROGRAM
- $ 42 M
- Advanced. Conceptual & Preliminary Design
- ***Regulatory Review***
- Economic Reviews
- Commercialization Studies
- Technology Development ($107 M Additional)

1988 -
PRDA
- $ 5 M
Continue Trade Studies

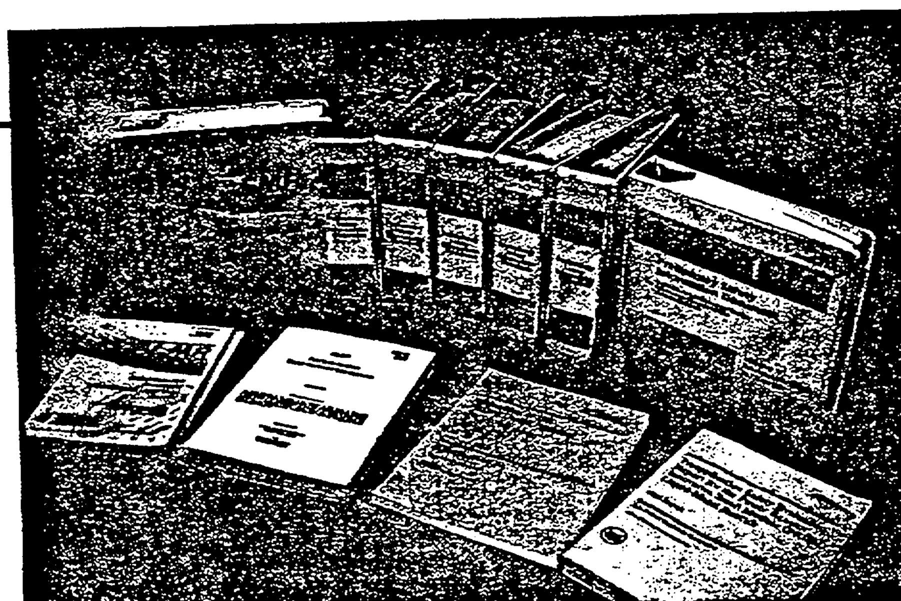
1985 - [redacted]987
[redacted]M
- 30 $ M
Competition for National Program

1981 [redacted]84
G[redacted]SM Program

GE Funded
Innovative Design Studies

$ 77 Million DOE Program

S-PRISM is supported by a 100 million dollar Data Base



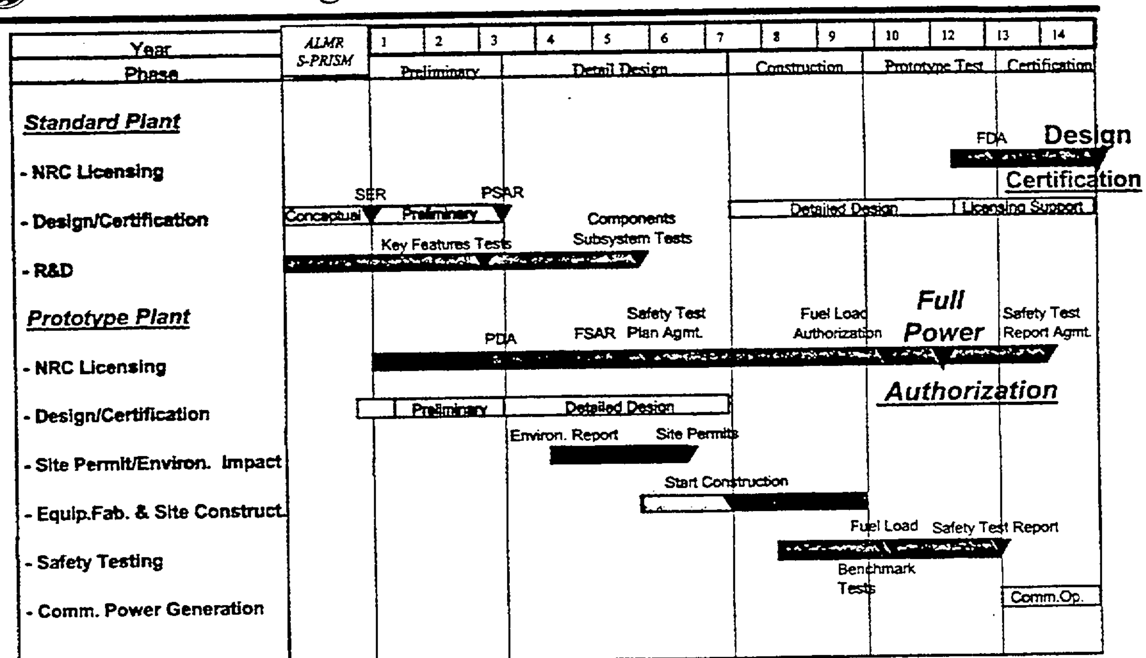
The NRC's Pre-application Safety Evaluation of the ALMR (NUREG-1368) concluded:

"the staff, with the ACRS in agreement, concludes that no obvious impediments to licensing the PRISM (ALMR) design have been identified."

Topics

- *Incentive for developing S-PRISM*
- *Design and safety approach*
- *Design description and competitive potential*
- *Previous Licensing interactions*
- *Planned approach to Licensing S-PRISM*
- *What, if any, additional initiatives are needed?*

Detailed Design, Construction, and Prototype Testing



Design Certification would be obtained through the construction and testing of a single 380 MWe module

Topics

- *Incentive for developing S-PRISM*
- *Design and safety approach*
- *Design description and competitive potential*
- *Previous Licensing interactions*
- *Planned approach to Licensing S-PRISM*
- *What, if any, additional initiatives are needed?*

Safety Review/Key Issues

NAME	LOCATION							
France								
Rapsodie	Cadarache							
Phenix	Marcoule							
SuperPhenix	Creys Malville							
INDIA								
FBTR	Kalpakkam							
ITALY								
PEC	Brasimone							
JAPAN								
Joyo	Oarai							
Monju	Ibaraki							
UK								
DFR	Dounreay							
PFR	Dounreay							
USA								
Clementine	Los Alamos							
EBR-1	Idaho							
Lampre	Los Alamos							
EBR-2	Idaho							
Enrico Fermi	Michigan							
SEFOR	Arkansas							
FFTF	Richland							
Clinch River	Oak Ridge							
USSR								
BR-2	Obninsk	Research	1956	–	0.1	–	Pu	Hg
BR-5	Obninsk							
BOR-60	Melekess							
BN-350	Shevchenk							
BN-600	Beloyarsk							
BN-800	–							
BN-1600	–							
W. Germany								
KNK	Karlruhe							
SNR-300	Kalkar							
SNR-2	Kalkar	demonstration	–	–	3420	1460	U02/Pu02	Na

Safety Methods

- *Containment*
- *Core energetic potential*
- *Analysis of Design Basis SG Leaks*
- *PRA*
- *Nuclear Methods*
- *T/H Methods*

Fuels

- *Validation of fuels data base (metal/oxide)*

Waste

- *Fission Product Treatment and Disposal*

More than 20 Sodium cooled Fast Reactors have been built

Most have operated as expected (EBR-II and FFTF for example)

The next one must be commercially viable

Component Verification and Prototype Testing

Final component performance verification can be performed during a graduated prototype testing program.

Example: The performance of the passive decay heat removal system can be verified prior to start up by using the Electromagnetic Pumps that add a measurable amount of heat to the reactor system

> *Licensing through the testing of a prototypical reactor module should be an efficient approach to obtaining the data needed for design certification.*
>
> *Defining the T/H and component tests needed to proceed with the the construction and testing of the prototype as well as defining the prototype test program will require considerable interaction with the NRC*

G. Apostolakis, ACRS Chairman: What are the most dangerous mines in the mine field that you feel we ought to be working on?

A. Rao, General Electric Nuclear Energy: Our experience on the last go round was a time and material effort. There tends to be no closure when you're having NRC review of the licensing submittals, whether it's with the national labs which are consultants to the NRC staff or the NRC staff. So there is a minimum incentive for closure of some of the items. That was our experience with the SBWR in the past.

We don't think there are any technical issues that are there because we've had -- I haven't emphasized the international part of our meetings. Typically we meet twice a year and have 30 or 40 people from national labs and people from all different parts of industry. So we don't think there's any technical issues. It's just bringing the NRC staff up to the same state where we are. That's one thing.

The other question is do the people who reviewed the SBWR in the NRC staff, are they still there? I think some of them are still there. That would make it go faster. The process of someone else coming up to the same level of understanding as those who worked on it is, I think, one of the major challenges we faced in the SBWR. I remember -- I don't know whether it was Ivan Catton or someone on the ACRS. It took several years before we got people to appreciate how simple our passive containment cooling system was, for example. It was actually not a natural circulation system. It was a -- circulation system. And so if the same members of the NRC staff are not there, we might have to go through that same process again. So it's those kind of institutional issues, I think, which will be a harder challenge for us.

Summary of NRR Future Licensing Activities

In response to a renewed interest in building nuclear power plants, the NRC has created organizations within its major program offices to prepare the NRC staff for new applications (early site permits [ESPs], design certifications, and combined licenses) and to manage special task groups and pre-application reviews of new reactor designs. Activities planned in FY2001 and FY2002 include: (1) evaluating the ability of the NRC staff to support future application reviews under 10 CFR Parts 50 and 52; (2) performing pre-application reviews of the AP1000 (a light-water reactor design with passive safety systems), Pebble Bed Modular Reactor (PBMR - a high-temperature gas-cooled reactor design), ESPs, IRIS (an advanced light-water reactor design), and GT-MHR (a high-temperature gas-cooled reactor design); (3) initiating and/or performing related rulemakings that will update 10 CFR Part 52 to reflect lessons learned from certifying three nuclear plant designs, update Tables S-3 and S-4 of 10 CFR Part 51 to address higher burnup fuel considerations and non-LWR advanced designs, and address alternative siting considerations; (4) reactivating the construction inspection program; and, (5) interacting with stakeholders to ensure there is a clear understanding of upcoming activities related to future applications and to solicit stakeholder input.

In FY2002 and FY2003, activities are expected to include: (1) managing the reviews of fine new applications resulting from the pre-application reviews (including one design certification, one combined license, and three ESP reviews); (2) managing two pre-application reviews (IRIS and GT-MHR); (3) updating regulatory and review guidance for new applications, i.e., Standard Review Plans (SRPs), Regulatory Guides, and referenced codes and standards, and identifying where enhancements are needed; (4) developing independent codes to analyze the safety of non-LWR designs, with supporting validation testing; and, (5) addressing regulatory infrastructure issues, including NEI's proposed New Plant Regulatory Framework initiative, and NRC regulations governing financial issues and operator staffing.

ACRS WORKSHOP ON ADVANCED REACTORS
JUNE 4, 2001

NRR FUTURE LICENSING ACTIVITIES

INTRODUCTION: M. Gamberoni

FUTURE LICENSING AND INSPECTION READINESS: N. Gilles

EARLY SITE PERMITS: T. Kenyon

ITAAC/CONSTRUCTION: T. Kenyon

AP1000: A. Rae

REGULATORY INFRASTRUCTURE: E. Benner

FUTURE LICENSING ORGANIZATION

- William Borchardt — Associate Director for Inspection and Programs
  - Richard Barrett — SES Manager
    - Marsha Gamberoni — Section Chief
      - J. N. Wilson — Sr. Policy Analyst
      - A. Rae — AP1000 PM
      - E. Benner — Regulatory Infrastructure
      - A. Cubbage/D. Jackson — PBMR/GT-MHR/IRIS PMs
      - T. Kenyon — Siting PM
      - N. Gilles — FLIRA Lead
      - J. Sebrosky — ITAAC/Construction PM
      - J. Williams — Senior PM

3

FUTURE LICENSING AND INSPECTION READINESS ASSESSMENT (FLIRA)

- Evaluate Full Range of Licensing Scenarios
- Assess Readiness to Review Applications & Perform Inspections
  - Staff Capabilities
  - Schedule and Resources
  - External Support
  - Regulatory Infrastructure
- Recommendations:
  - Staffing
  - Training
  - Contractor Support
  - Schedules
  - Rulemakings & Guidance Documents
- Complete Assessment by September 28, 2001

EARLY SITE PERMITS

- Early Site Permits (ESP)
  - Site Safety
  - Environmental Protection
  - Emergency Planning
- 10 CFR Part 52, Subpart A
  - Regulatory Guides
  - Environmental SRP
  - Experience with Environmental Reviews on License Renewal
- Initial efforts
  - Coordinate Preparations for ESP Reviews
  - Interact with Stakeholders
  - Recent Meetings with NEI ESP Task Force
- Applications
  - One in 2002, Two in 2003, Three in 2004

5

ITAAC/CONSTRUCTION

- Construction Inspection Program Re-activation
  - Develop Guidance for Inspection of Critical Attributes
  - Include Inspections for Plant Components & Modules at Fabrication Site
  - Initiate Development of Training for Inspection Staff
- Reactivation of Construction Permit (WNP-1)
- Resolution of "Programmatic" ITAAC

AP1000 PRE-APPLICATION REVIEW

- Phase 1 Complete
  - July 27, 2000 Letter Identified 6 Issues that Could Impact Cost and Schedule of Design Certification
- Phase 2 Scope
  - Applicability of AP600 Test Program to AP1000 Design
  - Applicability of AP600 Analyses Codes to AP1000 Design
  - Acceptability of Design Acceptance Criteria in Selected Areas
  - Applicability of Exemptions Granted to AP600 Design
- Phase 2 Schedule
  - Receipt of Analyses Codes Will "Officially" Start Phase 2
  - Estimated Duration of Review - 9 Months
- Phase 3 - Westinghouse Application 2002?

REGULATORY INFRASTRUCTURE

Current Activities:

- Rulemaking to Update 10 CFR Part 52
  - Incorporate Previous Design Certification Rulemaking Experience
  - Update Licensing Processes to Prepare for Future Applications
  - Proposed Rule Package (9/01)
- Rulemaking on Alternative Site Reviews
  - Amend Requirements in 10 CFR Parts 51 and 52 for NEPA Review of Alternative Sites for New Power Plants
  - Initiation of Rulemaking - Mid-FY2002
- Rulemaking on 10 CFR Part 51, Tables S3 and S4
  - Amend Part 51 Tables S-3 & S-4 for Fuel Performance Considerations and Other Issues to Reflect Current and Emerging Conditions in the Various Stages of the Nuclear Fuel Cycle

REGULATORY INFRASTRUCTURE

- Financial-Related Regulations
  - NRC Antitrust Review Requirements
  - Decommissioning Funding Requirements
  - Modular Plant Requirements (Price-Anderson)

Future Activities:

- NEI Petition for Generic Regulatory Framework
  - NEI Intends to Propose Risk-Informed GDC, GOC and Regulations
  - Petition Anticipated in December 2001
  - NEI Proposal May Be Similar to Option 3 of RIP50
- Licensing of New Technologies
  - Short-Term: Address via Existing Regulations, License Conditions and Exemptions
  - Long-Term: Address via Rulemaking

9

Office of Nuclear Regulatory Research
Summary of Advanced Reactors Activities
June 4, 2001

John H.Flack (Branch Chief) and Stuart D.Rubin (Senior Advisor) from the Division of Systems Analysis and Regulatory Effectiveness (RES), Regulatory Effectiveness and Human Factors Branch (REAHFB), provided an overview of the historical and current role of RES in pre-application reviews of advanced reactors. Pre-application interactions with potential licensee applicants will help NRC prepare for future submittals, through the development of the infrastructure necessary for licensing application reviews. RES has the lead for non-LWR advanced reactor pre-application initiatives and longer-range new technology initiatives. An advanced reactor group has been formed in REAHFB, and is currently performing a pre-application review of Exelon's Pebble Bed Modular Reactor. Recent industry requests for future pre-application interaction include General Atomics' Gas Turbine-Modular Helium Reactor (GT-MHR) and Westinghouse International Reactor Innovative and Secure (IRIS) design. RES advanced reactors activities also include participation as an observer in DOE's Generation IV initiative.

Pre-Application review objectives include the development of regulatory guidance, licensing approach, and technology-basis expectations for licensing advanced designs, including identifying significant technology, design, safety, licensing and policy issues that would need to be addressed in the licensing process. In addition, the pre-application review will help to develop necessary analytical tools, obtain contractor support, train staff to achieve fully the capacity and the capability to review advanced reactor license applications.

The presentation described the pre-application process for the Exelon PBMR. NRC first identifies additional information following topical meetings with Exelon, and Exelon formally documents and submits required topical Information. The staff then develops a preliminary assessment and drafts a response which is followed by stakeholder input and comments at a public workshop. Preliminary assessments are discussed with ACRS and ACNW, and Commission papers are written which provide staff positions and recommendations on proposed policy decisions. Some of the significant areas for the PBMR include:

Process Issues, Legal & Financial Issues
Regulatory Framework
Fuel Performance and Qualification
Traditional Engineering Design (e.g, Nuclear, Thermal-Fluid, Materials)
Fuel Cycle Safety Areas
PRA, SSC Safety Classification
PBMR Prototype Testing

Sources of expertise for the PBMR include, RES, NRR, NMSS, OGC technical expertise and associated regulatory experience, contractor support from National Labs,

prior NRC Modular HTGR pre-application review experience, design, operating and safety review experience for Fort St. Vrain HTGR, International HTGR experience including IAEA, Japan, China, Germany, UK, and external stakeholder comments, ACRS and ACNW advice and insights.

United States
Nuclear Regulatory Commission

Office of Nuclear Regulatory Research
Advanced Reactors Activities
June 4, 2001

John H.Flack
Stuart D.Rubin

Introduction

- Historical role of RES in preapplication reviews
- Preapplication review of advanced reactors
- Current role of RES in advanced reactor reviews
- Advanced reactor group in Division of Systems Analysis and Regulatory Effectiveness (RES)

Advanced Reactor Activities

- Advanced reactors have greater reliance on new technology and safety features.
- Preapplication interactions and reviews will help NRC prepare for licensing application
- NRR has lead with RES support for LWR advanced reactor preapplication initiatives and licensing application reviews
- NMSS has lead for fuel cycle, transportation and safeguards
- RES has lead for non-LWR advanced reactor preapplication initiatives and longer-range new technology initiatives
- Recent industry requests for preapplication interactions:

  Westinghouse: AP1000 (5/4/00)
  Exelon: Pebble Bed Modular Reactor (12/5/00)
  General Atomics: Gas Turbine-Modular Helium Reactor (3/22/01)
  Westinghouse: International Reactor Innovative and Secure (4/06/01)

- NEI Risk-Informed framework for Advanced Reactor Licensing

RES Advanced Reactors Activities

- PBMR:
  - Request for pre-application interactions received from Exelon
  - NRC response
  - Plan developed (SECY-01-0070)
  - Pre-application work underway (FY2001-2002)
  - Objective - identify issues, infrastructure needs and framework for PBMR licensing
  - Develop nucleus of staff familiar with HTGR technology
- GT-MHR
  - Request for pre-application interactions received from General Atomic
  - NRC Response

RES Advanced Reactors Activities (cont.)

- IRIS
  - Developed under DOE-NERI program
  - Initial meeting on 05/07/01
- Generation IV
  - International activity coordinated by DOE
  - Longer term
  - NRC participating as an observer
- Generic Framework:
  - NEI developing proposal
  - Need for NRC to establish an effective and efficient risk-informed,and where appropriate, performance-based licensing framework

Significant Technology Issues:

- Unique, First of a Kind Major Components
- Fuel Design, Performance, Qualification, & Manufacture
- Source Term
- Thermal-Fluid Flow Design
- Hi-Temperature Performance
- Containment
- Fuel Cycle Safety & Safeguards
- Prototype Testing and Experiments
- Human Performance and I&C
- Probabilistic Risk Assessment Methodology and Data
- Emergency Planning
- Regulations Framework
  - design basis accident selection
  - safety classification
  - acceptance criteria
  - GDC,
  - use of PRA
  - Safety Goals

PBMR Pre-Application Review Objectives

- To develop guidance on the regulatory process, regulations framework and the technology-basis expectations for licensing a PBMR, including identifying significant technology, design, safety, licensing and policy issues that would need to be addressed in licensing a PBMR.

- To develop a core infrastructure of analytical tools, contractor support, staff training and NRC staff expertise needed for NRC to fully achieve the capacity and the capability to review a modular HTGR license application.

PBMR Pre-Application Review Guidance

- Commission Advanced Reactor Policy Statement
- NUREG-1226 on the Development And Utilization of the Policy Statement
- Previous Experience with MHTGR Pre-Application Review
- Identify Safety, Technology, Research, Regulatory & Policy Issues

PBMR Pre-Application Review Scope

Selected Design, Technology and Regulatory Review Areas:

- Fuel Design, Performance and Qualification
- Nuclear Design
- Thermal-Fluid Design
- Hi-Temp Materials Performance
- Source Term
- Containment Design
- PBMR Regulatory Framework
- Human Performance and Digital I&C
- Prototype Testing Program
- Probabilistic Risk Assessment
- Postulated Licensing-Basis Events
- Fuel Cycle Safety
- Emergency Planning
- SSC Safety Classifications

PBMR Pre-Application Review Process

- Conduct Periodic Public Meetings on Selected Topics:
  - Process Issues, Legal & Financial Issues, Regulatory Framework (4/30)
  - Fuel Performance and Qualification (6/12-13)
  - Traditional Engineering Design (e.g., Nuclear, Thermal-Fluid, Materials)
  - Fuel Cycle Safety Areas
  - PRA, SSC Safety Classification
  - PBMR Prototype Testing
- NRC Identifies Additional Information Following Topical Meetings
- Exelon/DOE Formally Documents and Submits Topical Information
- NRC Develops Preliminary Assessment and Drafts Documented Response
- Obtain Stakeholder Input and Comments at a Public Workshop
- Discuss Preliminary Assessments With ACRS and ACNW
- Commission Papers Provide Staff Positions and Recommend Policy Decisions
- Commission Provides Policy Guidance and Decisions
- NRC Staff Formally Responds to Exelon with Positions and Policy Decisions

PBMR Pre-Application Review Sources of Expertise

- RES, NRR, NMSS, OGC Technical Expertise and Regulatory Experience
- Contractor Support From National Labs and Design/Technology Experts
- Prior NRC Modular HTGR Pre-Application Review Experience
- Design, Operating and Safety Review Experience for Fort St. Vrain HTGR
- International HTGR Experience: IAEA, Japan, China, Germany, UK
- Exelon and DOE Design, Technology and Safety Assessments
- External Stakeholder Comments
- ACRS and ACNW Advice and Insights

PBMR Safety Significant Review Issues/Topics

- Fuel Performance and Qualification
- High Temperature Material Issues
- Passive Design and Safety Characteristics
- Accident Source Term and Basis*
- Postulated Licensing Basis Events*
- Prototype Testing Scope and Regulatory Credit
- Containment Functional Design Basis*
- Emergency Planning Basis*
- Risk-Informed Regulatory Framework*
- Probabilistic Risk Assessment

* Commission Policy Decision Likely Is Needed

PBMR Pre-Application Review Schedule

- About 18 months to Complete
- Monthly Public Meetings To Discuss Topics
- Feedback on Legal, Financial and Licensing Process Issues (~9/01)
- Feedback on Regulatory Framework (~12/01)
- Feedback on Design, Safety, Technology & Research Issues (~6/02)
- Feedback on Policy Issues (~10/02)

Regulatory Infrastructure Development Needs

- Staff Training Course for HTGR Technology
- Analytical Codes and Methods for Advanced Reactor Licensing Reviews
- Regulatory Framework for Advanced Reactor Licensing Reviews
- Core Staff Capabilities for Advanced Reactor Licensing Reviews
- Contractor Technical Support Capabilities
- Possible RES Confirmatory Testing and Experiments
- Possible Codes and Standards for Advanced Reactor Design and Technology

G. Apostolakis, Chairman, ACRS: If someone comes to you using Part 52, is there anything there that says that you need the risk-informed, performance-based system?

J. Flack, RES: There's nothing in Part 52 that says that we need to have a risk-informed, performance-based licensing approach.

G. Apostolakis, Chairman, ACRS: So they could approach the licensing issue without using risk information. Could they?

J. Flack, RES: Yes, I would expect that would be the case.

G. Apostolakis, Chairman, ACRS: Is there anything that gives you the authority to request risk information?

J. Wilson, NRR: The Part 52 licensing process is just that, it's a licensing process, and so it references back to parts 20, 50, 70 and 100 for the actual safety requirements. So whether or not those safety requirements remain as they are or change as a result of some risk-informed process, it will use whatever is the requirement that's currently in place.

G. Apostolakis, Chairman, ACRS: What if the industry doesn't want to use risk information? What if they just want to use existing regulations with exemptions or changes and maybe they feel that going to a risk-informed system adds an impediment because we have to understand it and do it. It's new. And try to go with the existing system and maybe a PRA would be an assessment at the end if you guys request it but maybe it will be a good idea not to bring it up at all. Why is that the need?

J. Flack, RES I think it would be to their advantage to come in that way.

G. Apostolakis, Chairman, ACRS: We heard today from several speakers, I think, that they're trying to reduce involvement of the humans. Do you think that the human performance issue will be as important here as in the current reactors?

J. Flack, RES: I've discussed this at length. I don't know whether we can say it's going to be less important. I mean it's going to be a different environment which that human operates in, and one has to understand that environment and what's changing in that environment. So it's something that one has to look at very carefully. So it's hard to say.

D. Powers, ACRS Member: It seems to me that the change is really entertaining and in the direction that's most difficult for us because as they design the plants to be less and less dependent on the human operator intervening. We become more and more worried about the fact that the operators are not going to sit there and do nothing. They will intervene and the potential for them to intervene incorrectly in a system that's designed to operate with rather minor low head forces operating on it. So you get into

the problem of errors of commission that we are most incapable of addressing. It's a subtle problem.

J. Flack, RES: Yes. The environment changes and you don't really have as much data as you wish you'd have to go on.

J. Garrick, Chairman, ACNW: This is probably the question that I was half asleep on when George asked the question about the risk assessment. But you mentioned that on the PBMR you're going to get a risk assessment. What's the nature of that? Has that been requested?

S. Rubin, RES: We have urged Exelon to provide as much information on the current risk assessment that they've done for the plan to support our review of this risk-informed framework for making licensing decisions. I wouldn't call it a risk-informed regulations framework as the extent of wholly replacing Part 50 but we think we now understand that this framework is not quite going to do that but will through risk insights be able to identify systems requirements for mitigation, prevention, the level of redundancy in those systems, which systems should be designated as safety significant and also things like what are the special treatment requirements on the system. But we're not talking about a regulations framework which covers all of Part 50.

But to answer your question, we have asked for that and we've also asked, to the extent possible, that we get information on the design itself. We have not yet, except for these kinds of viewgraphs that we've seen today, gotten what I would call a significant design description and principles of operation document from Exelon. I think the staff would very much like to get both a PRA and a design description so we have a context for reviewing this framework. It is on our schedule. We talked about that. It's not now but it is later.

Nuclear Energy Institute (NEI) Summary

Prepared by ACRS Staff for R. Simard

Mr. Ron Simard of the Nuclear Energy Institute (NEI) provided a brief presentation on the state of energy demand in the United States and discussed the improving economics for new nuclear power plants. He discussed the consolidation of companies under deregulation and the ability of these larger companies to undertake large capital projects such as nuclear power plant construction. He discussed efforts under way to support a new generation of plants but noted that there needs to be greater certainty in the licensing process. He discussed infrastructure challenges in terms of people, hardware, and services to support new and current plants. He stated that there needs to be fair and equitable licensing fees and decommissioning funding assurance for innovative modular designs such as the PBMR. He concluded that NRC challenges will include resolving 10 CFR Part 52 implementation issues, establishing an efficient and predictable process for siting, COL permits and inspection, and an increasing regulatory workload.

Licensing needs for future plants

ACRS workshop on Regulatory Challenges
for Future Nuclear Power Plants

Ron Simard
Senior Director, Business Services
Nuclear Energy Institute

June 5, 2001

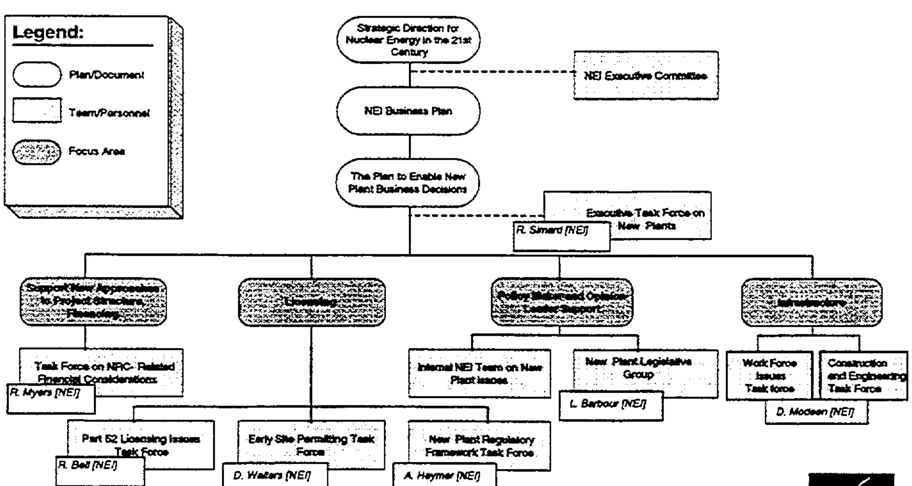
New Nuclear Power Plants - New Momentum

- Growing electricity demand, need for new generating capacity
- Fossil fuel price volatility, clean air constraints
- Improving economics of new nuclear power plants
- Industry consolidation = companies large enough to undertake large capital projects
- Significant public and political support
- Potential for greater certainty in the licensing process

Focus of efforts to pave the way for new plants

- Policy, legislative, regulatory changes needed to support new approaches to ownership, risk sharing and project financing
- Policymaker support (Administration, Congress and others)
- Infrastructure (people, hardware, services) to support new and current plants
- Licensing, licensing, licensing

Activities in support of the plan to enable new plant business decisions



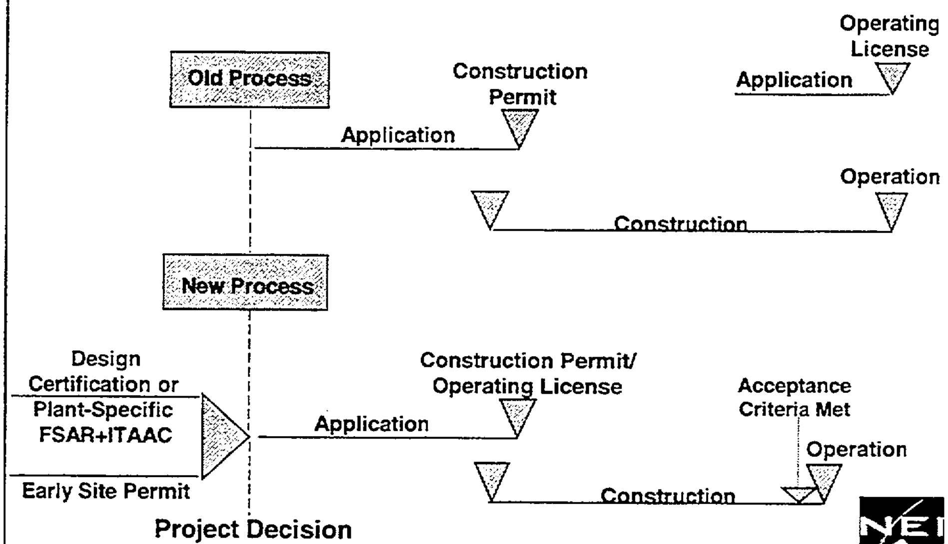
Licensing needs with respect to ...

- working out the Part 52 implementation details
- assuring safety and equitable application of regulations to new types of designs
- clarifying how financial related requirements apply in the new business environment

Examples of Part 52 licensing needs

- a timely and efficient ESP process (e.g., focusing on the incremental impacts of additional reactors at existing sites)
- a timely and efficient process for COL applications and reviews
- an efficient process for construction inspection and ITAAC verification

New Licensing Process Significantly Reduces Project Risk



"New design" licensing needs (in addition to safety determinations)

- For modular designs, clarification of
    - number of licenses per facility
    - application of Price Anderson requirements
    - basis for Part 171 annual fees
    - basis for control room staffing
- For gas cooled designs, clarification of
    - decommissioning funding levels
    - generic environmental impacts (Tables S-3, S-4)
    - basis for EP action levels, reporting requirements, implementation of NUREG-0654

Licensing needs for the new business environment

- Clarification of how financial related requirements apply to merchant nuclear plants
  - no need for an NRC antitrust review
  - nature of financial qualifications
  - appropriate mechanisms for decommissioning funding assurance

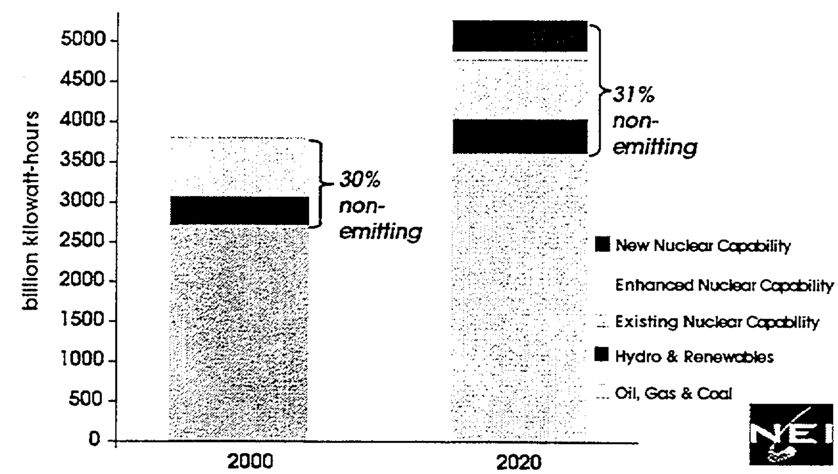
NEI

The nuclear energy imperative

- DOE projects 400,000 MW of additional capacity needed by 2020 (to replace existing plants that reach end of life and to meet new demand)
- 30% of our current generation is non emitting (nuclear, hydro, renewables)
- maintaining that contribution to clean air will require 50,000 Mwe of new nuclear

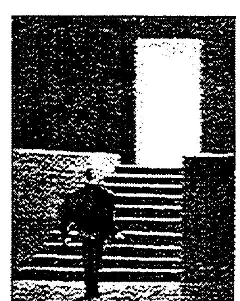
NEI

Vision 2020



The Future isn't what it used to be because ...

- Electricity demand will continue to grow
- New nuclear generation is no longer an option
  - it is an imperative
- The business case for new nuclear plants will be clear
- The cost and schedule drivers must be known and manageable to much more certainty than in the past



The Future isn't what it used to be because ...

NRC will be challenged to

- ▶ resolve Part 52 implementation issues in a timely manner
- ▶ establish efficient and predictable processes for siting, COL license applications, construction inspection
- ▶ respond to an increasing workload with new focus, discipline and efficiency

D. Powers, ACRS Member: There seems to be a body of opinion that maybe we don't need that much electrical energy, and that we, in fact, can achieve the necessary energy supplied by conservation.

R. Simard, NEI: No question that conservation and efficiency are important, but it's folly to think that you're going to conserve your way out of having to add almost a 50 percent increase. The gains that we have made in conservation have been impressive, at times, and efficiency has really helped quite a bit, but there's no way to conserve your way out of the low end of this projection without disruptive impacts on the economy.

There are still some people who will question the need to have that much electricity and they might even go so far as to say that we can keep our current demand steady. The future isn't what it used to be because I think the consensus is here now that the demand will grow, and we used to talk about the nuclear option. It's not an option anymore. It's an imperative.

The business case for new plants is pretty clear, but we have to have cost and schedules known to a greater degree of certainty than we ever had before, which leads us into the challenge for the NRC because, the ability to bring this plant to make depends upon being able to work out these Part 52 implementation issues in a timely manner, and having in place efficient and, Commissioner Diaz's word, "scrutable" processes for early siting and licensing and construction inspection. And what's emerging here from this day and a half is the challenge for NRC to be able to respond to this with a whole new focus and discipline and efficiency.

D. Powers, ACRS Member: One of the persistent problems that we encounter when new things are brought to this particular body is the documentation is incomplete; the documentation is not rigorous. Those kinds of things slow the process substantially.
Is the industry doing anything to try to address those kinds of questions?

R. Simard, NEI: I think the challenge on our side is to bring in an unprecedented quality of application. On our side, we need to bring to the NRC the highest quality of information and application. What you're seeing both with the Westinghouse and PBMR North America International with NRC, is an effort early on to really clearly identify exactly what the staff needs are going to be to be able to do their review.

G. Apostolakis, ACRS Chairman: In one of your earlier slides, it says "acceptance criteria met," do we have those criteria?

R. Simard, NEI: Yes, in the three designs that have been certified, a key feature and a high level of detail in those certifications are the ITAAC. So they're clearly specified. In the ABWR, for example, the high pressure core flooder system, there were 31 separate ITAAC that clearly focus on the performance of a pump. For example, what inspections or tests will be done on that pump and what acceptance criteria will be necessary to show that, in fact, that pump is going to deliver the amount of water you need at the time you need it? So in the design certification, a key feature of them has been these ITAAC. We need to add a few more that are site specific when the licensee brings the application.

T. Quinn, Consultant, General Atomics: The reason for success in the license renewal process to a large extent was the project management role that was put in place with a lot of work by NEI, with a lot of work by the NRC, and a suite of documents that became part of the

process, (e.g., the GALL report, and the NEI guideline). Have you considered working with NRC on a similar type of suite of documents to help us make this a more stable framework?

R. Simard, NEI: Yes, I think you're right, Ted. That's been a good model in the past. By bringing to bear the range of industry resources and expertise on an area and combining that with the NRC, I think we've wound up with a better quality product in the end and improved the efficiency of the process.

So building on our success with license renewal, maintenance rule or other things like that, it is our intent to put a lot of thought from our side into how -- for example, the format of an early site permit application, and that's something we actually have underway, or with respect to construction inspection at ITAAC verification, it's our intent to bring together the folks who still have construction experience in the industry, if we can find them, and again, drawing upon their expertise and our knowledge of how Part 52 -- the basic principles of Part 52.

Again, it would be our intent in cases like that to bring in a document and ask the NRC for its review and reactions and use that as the framework for these productive discussions.

ACRS WORKSHOP
Regulatory Challenges for Future Nuclear Power Plants
"Safety Goals for Future Nuclear Power Plants"
Neil E. Todreas
KEPCO Professor of Nuclear Engineering
Massachusetts Institute of Technology

This talk presents technology goals developed for Generation IV nuclear energy systems that can be made available to the market by 2030 or earlier. These goals are defined in the broad areas of sustainability, safety and reliability, and economics. Sustainability goals focus on fuel utilization, waste management, and proliferation resistance. Safety and reliability goals focus on safe and reliable operation, investment protection, and essentially eliminating the need for emergency response. Economics goals focus on competitive life cycle and energy production costs and financial risk.

The goals have three purposes: First, they define and guide the development and design of Generation IV systems. Second, they are challenging and will stimulate the search for innovative nuclear energy systems—both fuel cycles and reactor technologies. Third, they serve as the basis for developing criteria to assess and compare the systems in a technology roadmap.

The Generation IV technology goals derive from a set of guiding principles:

- Technology goals for Generation IV systems must be challenging and stimulate innovation.
- Generation IV systems must be responsive to energy needs worldwide.
- Generation IV concepts must define complete nuclear energy systems, not simply reactor technologies.
- All candidates should be evaluated against the goals on the basis of their benefits, costs, risks, and uncertainties, with no technologies excluded at the outset.

The Generation IV technology goals are intended to stretch the envelope of current technologies. Hence, the following caveats are important to note:

- The goals will guide the development of new nuclear energy systems. The objective of Generation IV systems is to meet as many goals as possible.
- The goals are not overly specific because the social, regulatory, economic, and technological conditions of 2030 and beyond are uncertain.
- The goals must not be construed as regulatory requirements.

Future designs will likely (but not necessarily) involve new fuel cycles and the capability to produce a broader range of energy products. For these reasons and

to enhance the economic performance of electricity-only producing systems, I anticipate:

- New Fuel Materials
- Higher Burnups
- Longer Operating Cycles
- Higher Temperature Operation

These trends will be driven by the Sustainability (SU 1, 2, +3) and the Economic (EC 1+2) Goals.

Since these trends involve significant safety issues, all the goals should be considered as relevant contributors to the safety profile of future Generation IV energy systems.

Each of the eight goals is presented and the key issues debated and decided upon in their formulation will be discussed. The illumination of this debate is reflected in the Viewgraphs by highlighting the wording in the Goals Statements that best embodies the deliberations.

For the Sustainability Goals the following observations are relevant:

- Fuel cycle development offers the only way to address objectives of availability, waste management and nonproliferation in an integrated manner.

- Hence, for the US, R+D on fuel cycle options needs to be reinvigorated.

- The once -thru fuel cycle will likely be hard to beat considering that the objective of effective fuel utilization involves the following elements:

    - Economics (fuel cycle plus effect on O+M cost).

    - Nonproliferation concerns - challenge remains on cross-rating individual intrinsic and extrinsic barriers.

    - Environmental concerns - to what degree are externalities to be internalized in the nuclear fuel cycle and in other competing energy supply systems.

For the Economic Goals the following observations are relevant:

- Legitimate differing views exist on whether "clear" life-cycle cost advantage will be needed over the 30 year horizon for introduction of GENIV systems or whether breakeven will suffice because of recognition/credit for environmental benefits derived from nuclear systems.

- A judgement has been made that the history of deployment of nuclear systems has so raised the specter of risk and uncertainty of deployment cost that a "clear" advantage will be necessary to induce a commercial

commitment to GENIV (or any nuclear system i.e. NTD) systems. Further, although allowable financial risk is limited by the need to achieve the life cycle cost advantage, the nuclear deployment history has also so raised the issues of risk & uncertainty of deployment that the separate and specific EC-2 Goal on "level of financial" risk was deemed necessary.

Anticipating enhanced interest in this audience in the three Safety and Reliability Goals, the text of the discussion which follows and supports each of these goals is also presented with the relevant wording similarly highlighted.

The latest statement of these Goals which was presented to NERAC on May 1, 2001 and subsequently accepted by DOE for final presentation to GIF is appended.

Conclusion:

Future reactors fall in three categories - those which are:

- Certified or derivatives of certified designs.
- Designed to a reasonable extent and based on available technology.
- In Conceptual form only with potential to most fully satisfy the GENIV goals.

My focus has been on goals for the third category.

It will be desirable to develop a range of design options in this third category to enable response to a range of possible market demands such as:

- cheap versus expensive uranium
- small versus large power ratings
- significant reduction of greenhouse emissions
- new fuel cycles to achieve a significant response to the sustainability goals

Considerable R+D activity will be required to achieve these goals among which fuels, materials, and coolant corrosion research are the most intensive and long term.

Consequently it is important that while an early dialogue between designers and regulators occur, the dialogue be framed to encourage & promote fundamental design directions which inherently promote safety. Development of a new regulatory process using risk-based principles is an important element of this dialogue. Interactions which frame the dialogue around the current regulatory framework can have the undesirable intent of discouraging the necessary and desirable exploration of technology and design alternatives.

ACRS WORKSHOP
Regulatory Challenges for Future Nuclear Power Plants

Safety Goals for Future Nuclear Power Plants

Neil E. Todreas
KEPCO Professor of Nuclear Engineering
Massachusetts Institute of Technology

AM June 5, 2001

M.I.T. Dept. of Nuclear Engineering

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HOW TO MISCONSTRUE THIS TALK

I am not talking about:

- NRC Safety Goals - Quantitative Health Objectives - CDF and LERF.
- Suggested Regulatory Requirements for Future Power Plants.
- Soley about Future Power Reactors.
- Goals for Near Term Deployment* Plants (by 2010).

I am talking about:

- DOE and GIF Generation IV Technology Goals.
- Technology Goals formulated to
  - stimulate innovation.
  - suggest metrics for downselection which specifically are not to be construed as regulatory requirements.
- Nuclear Energy Systems Including
  - Fuel Cycles
- Goals for Systems to be Deployed from 2011 to 2030.

* Deployment: Manufacture, construction, and startup of certified plants ready to produce energy in their chosen market.

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HOW TO MISCONSTRUE THE GOALS

- Assume that new nuclear energy systems must meet every new goal
  - Tradeoffs among goal parameters must be made for each design. Future markets may value different parameters.

    Desirable outcome is a spectrum of designs each best suiting different market conditions hence different goals.
  - Some goals presently appear unattainable (S+R 3).
  - Most goals are not overly specific because the social regulatory, economic and technological conditions of 2030 and beyond are uncertain.

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HOW TO MISCONSTRUE THE GOALS (cont.)

- Assume that all safety considerations are encompassed in the Safety and Reliability Goal grouping (S+R 1, 2, +3)
  - Future designs will likely (but not necessarily) involve new fuel cycles and the capability to produce a broader range of energy products. For these reasons and to enhance the economic performance of electricity-only producing systems, I anticipate:
    - New Fuel Materials
    - Higher Burnups
    - Longer Operating Cycles
    - Higher Temperature Operation
  - These trends will be driven by the Sustainability (SU 1, 2, +3) and the Economic (EC 1+2) Goals.

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4

SUSTAINABILITY

Sustainability is the ability to meet the needs of present generations while enhancing and not jeopardizing the ability of future generations to meet society's needs indefinitely into the future.

Sustainability–1.

Generation IV nuclear energy systems including fuel cycles will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.

Sustainability–2.

Generation IV nuclear energy systems including fuel cycles will minimize and manage their nuclear waste and notably reduce the long term stewardship burden in the future, thereby improving protection for the public health and the environment.

Sustainability–3.

Generation IV nuclear energy systems including fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials.

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SAFETY AND RELIABILITY

Safety and reliability are essential priorities in the development and operation of nuclear energy systems.

Safety and Reliability –1.

Generation IV nuclear energy systems operations will excel in safety and reliability.

Safety and Reliability–2.

Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

Safety and Reliability–3.

Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

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Safety and Reliability –1. Generation IV nuclear energy systems operations will excel in safety and reliability.

This goal aims at increasing operational safety by reducing the number of events, equipment problems, and human performance issues that can initiate accidents or cause them to deteriorate into more severe accidents. It also aims at achieving increased nuclear energy systems reliability that will benefit their economics. Appropriate requirements and robust designs are needed to advance such operational objectives and to support the demonstration of safety that enhances public confidence.

During the last two decades, operating nuclear power plants have improved their safety levels significantly, as tracked by the World Association of Nuclear Power Operators (WANO). At the same time, design requirements have been developed to simplify their design, enhance their defense-in-depth in nuclear safety, and improve their constructability, operability, maintainability, and economics. Increased emphasis is being put on preventing abnormal events and on improving human performance by using advanced instrumentation and digital systems. Also, the demonstration of safety is being strengthened through prototype demonstration that is supported by validated analysis tools and testing, or by showing that the design relies on proven technology supported by ample analysis, testing, and research results. Radiation protection is being maintained over the total system lifetime by operating within the applicable standards and regulations. The concept of keeping radiation exposure as low as reasonably achievable (ALARA) is being successfully employed to lower radiation exposure.

Generation IV nuclear energy systems must continue to promote the highest levels of safety and reliability by adopting established principles and best practices developed by the industry and regulators to enhance public confidence, and by employing future technological advances. The continued and judicious pursuit of excellence in safety and reliability is important to improving economics.

Safety and Reliability–2. Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.

This goal is vital to achieve investment protection for the owner/operators and to preserve the plant's ability to return to power. There has been a strong trend over the years to reduce the possibility of reactor core damage. Probabilistic risk assessment (PRA) identifies and helps prevent accident sequences that could result in core damage and off-site radiation releases and reduces the uncertainties associated with them. For example, the U.S. Advanced Light Water Reactor (ALWR) Utility Requirements Document requires the plant designer to demonstrate a core damage frequency of less than 10^{-5} per reactor year by PRA. This is a factor of about 10 lower in frequency by comparison to the previous generation of light water reactor energy systems. Additional means, such as passive features to provide cooling of the fuel and reducing the need for uninterrupted electrical power, have been valuable factors in establishing this trend. The evaluation of passive safety should be continued and passive safety features incorporated into Generation IV nuclear energy systems whenever appropriate.

Safety and Reliability-3. Generation IV nuclear energy systems will eliminate the need for offsite emergency response.

The intent of this goal is, through design and application of advanced technology, to eliminate the need for offsite emergency response. Although its demonstration may eventually prove to be unachievable, this goal is intended to stimulate innovation, leading to the development of designs that could meet it. The strategy is to identify severe accidents that lead to offsite radioactive releases, and then to evaluate the effectiveness and impact on economics of design features that eliminate the need for offsite emergency response.

The need for offsite emergency response has been interpreted as a safety weakness by the public and especially by people living near nuclear facilities. Hence, for Generation IV systems a design effort focused on elimination of the need for offsite emergency response is warranted. This effort is in addition to actions which will be taken to reduce the likelihood and degree of core damage required by the previous goal.

ECONOMICS

Economic competitiveness is a requirement of the marketplace and is essential for Generation IV nuclear energy systems.

Economics–1.

Generation IV nuclear energy systems will have a clear life-cycle cost advantage over other energy sources.

Economics–2.

Generation IV nuclear energy systems will have a level of financial risk comparable to other energy projects.

CONCLUSIONS

- Future reactors fall in three categories - those which are:
  - Certified or derivatives of certified designs.
  - Designed to a reasonable extent and based on available technology.
  - In Conceptual form only with potential to most fully satisfy the GENIV goals.

  My focus has been on goals for the third category.
- It will be desirable to develop a range of design options in this third category to enable response to a range of marketing demands such as:
  - cheap versus expensive uranium.
  - small versus large power ratings.
  - significant reduction of greenhouse emissions.
  - new fuel cycles to achieve a significant response to the sustainability goals.

  Considerable R+D activity will be required to achieve these goals among which fuels, materials, and coolant corrosion research are the most intensive and long term.
- Consequently it is important that while an early dialogue between designers and regulators occur, the dialogue be framed to encourage & promote fundamental design directions which inherently promote safety. Development of a new regulatory process using risk-based principles is an important element of this dialogue. Interactions which frame the dialogue around the current regulatory framework can have the undesirable intent of discouraging the necessary and desirable exploration of technology and design alternatives.

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D. Powers, ACRS Member: One of the questions that comes to mind, especially after the previous speaker portrayed something of a crisis appearing, I wonder if in looking at these goals and looking at new systems that you compare the more modern or the existing plants against them to see if we really need all new concepts, and the 94 new concepts that were portrayed to us yesterday or, in fact, how well do the existing plants meet these various goals that you've laid out?

N. Todreas, MIT: The answer to that is on the metrics that we're going to develop to assess these new concepts. We've picked a standard, and the evaluation process will measure these new concepts against the standard. Is it better, much better, et cetera, worse, much worse, and the standard we picked is the advanced LWR with once through fuel cycle. The rest of your question asked me what's the answer going to be, and I don't know that yet.

D. Powers, ACRS Member: I find that a peculiar standard to pick because we don't have a whole lot of experience with advanced LWR. With existing machines, we have a lot of experience, and that experience, at least my friends at NEI certainly provide metrics that suggest that experience, is outstanding right now.

N. Todreas, MIT: I can see thinking about that, but if we're going to develop advanced systems, I would say from the vendor community and the development community, we've got ABWR experience to an extent, and we have some degree of real respect for what the designs have accomplished in the ALWR. As a minimum you'd include both, but I certainly wouldn't go back just to the operating reactors as the standard for the future. I wouldn't ignore the 15 years of ALWR development.

G. Wallis, ACRS Member: I think that while you're being innovative, you should not use -- you seem to be here really talking about core damage frequency, and that just may get you in a box, and I think to be innovative, get away from these terms of the past and be more general.

T. Kress, Chairman, Future Reactors Subcommittee: I think it is fission products we're worried about.

N. Todreas, MIT: That's a reasonable point. If you are saying that we ought to get away from terms of the past which will lock us into certain design directions and means of dialogue, that is really my whole message, too. If you're offering me a suggestion that says what I wrote doesn't go that way; I should go a different way, then I'd perfectly accept it.

J. Garrick, Chairman, ACNW: We have to be a little careful not to unduly focus on fission products because for many of the most important scenarios it is not the fission products that's driving the long-term performance of Yucca Mountain. It's mainly, technetium and Iodine 129 certainly are in there, but depending on the scenario and depending on how you look at it, Neptunium 237 is the principal driver.

And also, in most low level waste situations, you find that much to our surprise most of the low level waste is uranium contaminated. So, again, the fission products are not driving the long-term stewardship or management of a lot of the low level waste, but rather it's actinides. The same thing is true in WIPP for transuranic waste. Again, it's not fission products, but it's plutonium.

N. Todreas, MIT: But that also refers back to the sustainability goal. It really doesn't obviate the suggestion relative to Safety and Reliability (S&R) grouping 2 relative to core damage. I say that because what Garrick's comment really impacts on is the waste issue, not effectively the immediate release through core damage.

T. Kress, Chairman, Future Reactors Subcommittee: In particular, do you have some sort of criteria on what it would take to eliminate this need? And if so, does that criteria encompass some sort of measure of defense-in-depth also?

N. Todreas, MIT: That's how the ACRS ought to look at it. From the point of view of a regulator or a group advising a regulator. These are technology goals. These are goals we want to drive the designers into thinking about.

T. Kress, Chairman, Future Reactors Subcommittee: How would you know if you met that goal? That was my question. What is the measure that you're going to use to say, "Okay. The technology we have here meets that goal." Whether or not it actually comes about or not is another thing.

N. Todreas, MIT: The measure has got to be release of fission products or radioactivity of a certain amount past the boundary.

D. Powers, ACRS Member: I can always find a way to get fission products out. Any design you come up with I can find a mechanism to get the fission products out to the point that it violates some emergency planning guide.

E. Lyman, Nuclear Control Institute: There are a few goals that are really missing from this whole formulation. First of all, under sustainability you refer to one that minimizes, that a goal is minimizing and managing nuclear waste, but at the same time, you really should impose a requirement that the routine emissions from the entire fuel cycle, as well as, occupational exposures are also minimized because one of the concerns with fuel cycles that involve reprocessing are these additional routine emissions, and you have to balance whether the reduced risk in a repository is justified by increased short-term emission. So that's really something you have to keep to minimize at the same time or it doesn't make sense.

Second of all, under the financial goals issue, you didn't really dwell on the one that requires or suggests that the financial risks should be comparable to other energy projects, and I was wondering if in that context, you would also have a requirement then, that Price Anderson protection not be extended to Generation IV plants because other energy projects don't require that kind of protection.

N. Todreas, MIT: Yes, on the first point you brought up, the specifics of that have been recognized and will come up in Safety and Reliability group 1 because there we are talking about across the whole fuel cycle, and those routine emissions are picked up there. They could be picked up either place, but that's where they come up.

And on Price Anderson, we didn't get into the specific item within the structure of the goal that can be picked up and debated. It's been debated to some extent, but we didn't pin it down and resolve it specifically. I know that's coming up legislatively.

R. Barrett, NRC: My question relates to the methods that we use for estimating the likelihood of core damage and the likelihood of release of radioactivity.

If NEI is correct and we have 50,000 new megawatts of capacity out there, and those are modular reactors -- that's 500 cores, and in an environment like that you find yourself striving for lower and lower core damage frequencies, and as you do that, you begin to put more and more stress on the current methods of estimating core damage frequency, and you begin to get to the point where many people think you're beyond the capability and the limitations of the method and the ability to have a complete model.

In addition, as you move to different types of reactors, you find that you're depending less and less on highly reliable, redundant, and diverse systems and more and more on the intrinsic capability of the core itself to withstand these accidents, and to withstand them either indefinitely or for long periods of time. And, again, the methods that we have today really don't deal very well with this kind of intrinsic, passive capability.

So my question to you is the stated purpose of your effort is to stimulate innovation in the design of the reactors, and my question is: could you also complement that with trying to stimulate innovation in the methods that we use for analyzing the risk associated with these reactors?

N. Todreas, MIT: Yes, I would answer that two ways. First, it's a good suggestion and a fair suggestion. There's nothing implicit in my statement that precludes risk methods development -- what's going to come out of this fundamentally it is a spectrum of concepts to focus on, but much more than that, an R&D road map of activities to flesh out those concepts and the methods associated with the concept development is certainly part and parcel of that. So we could do that.

The other thing though that I'd say is if we were to develop the methods we're going to have to reduce core damage frequencies further to get a desired output. So that really leads you to say that if you go with concepts now that are clones or like -- I'm talking about 20, 30 years down the road -- existing concepts are like these, you're going to reach a point where the methods can only go so far based on the existing design approaches, and so that's a clarion call to change those approaches and go toward -- well, first, you go toward situations that avoid core melt, but that's very limited in a sense that what you really want to do is do what Dana Powers was talking about. It's not core melt. It's the fission products, and it's the radioactivity in the dose from that, and that's what you've got to get after. So I would say we certainly would accept and develop methods, but what we are trying to do is stimulate. I'm talking about real innovation, beyond that, to try to open up approaches that really change the playing field.

L. E. Hochreiter, Penn State University: It's not clear to me why in your conclusions you have to have small versus large power ratings. It seems like you're biasing yourself already towards a particular class of designs.

N. Todreas, MIT: Yes. Yesterday I presumed the whole layout of this program was announced or was explained as an international program with eight to nine countries now, and one of the goals of the program is to come up with design solutions or concepts that meet markets internationally, and there are some international markets. Also if you listen in the United States, too, depending on the grid size, there are some markets that have a priority toward low rated systems. And so you have some of those, and then you also have the

traditional, if you talk about Asia, Japan, Korea, Taiwan, large systems. So inherent in the whole program, since it's looking at worldwide markets, we're going to have this dichotomy, these two parts, and no one reactor thrust or direction is going to meet them both. So you're going to have to come up with systems in both directions. Now, your point may be fine, but they're not going to be sellable in the United States or the industrialized world. That's fine, but we'll have a product for that. We just may not use the other product.

N. P. Kadambi, NRC: If I understand the rules by which the South Africans are trying to license their plant, one of their goals is that in the long term the concepts employed should be amenable for society to make a decision that higher levels of safety need to be obtained from these energy systems.

And therefore, one of their goals, as I read it, and if I should be corrected, I'd like somebody to point this out; one of their goals is the design should be amenable for society to demand higher levels of safety at some future time if we take, you know, these systems as operating for many decades.

Where does such a concept fit into the kinds of goals that you have articulated?

N. Todreas, MIT: Okay. On this let me give you a brief answer and ask for some help because I am not knowledgeable about a specific or the specific South African drive that you're talking about. I just haven't interacted with them specifically. I would say that even though these are general goals, we are going to have some kind of constraint because we're going to come up with a set of specific metrics that go with each of these goals. They're going to be as we go on a year or two -- there's going to be some numbers and some specificity here . So there's going to be a little bit of a lock-in with your desire to accommodate future societal wishes for enhanced safety. The way I interpret what you're saying is you come up with a design. Society decides they want more safety, and so this design has somehow got to be expandable or have margin or a way to capture more safety. That's how I understand it.

So I don't know the answer. These goals have been pushed in through a discussion with the so-called GIF countries, of which South Africa is a part, and we didn't get any effective comment back from them that's relevant to what you said. But if Andy Kadak or somebody else can speak specifically to that, that would help me.

J. Slabber, PBMR: In the South African concept, the baseline was to use existing technology as far as possible, existing technology that has been qualified and tested and proven to be acceptable for use in the PBMRs, and with a basis that the fuel is the central point of focus. And within that framework, we do the system design. Imbedded in the design is the requirement to be fulfilled that no reliance is placed on immediate operator action to bring the reactor to a safe state, and I again say, in inverted commas, inherent safety and small units, and usable for not only producing nuclear power, but also some other usable byproducts such as possibly desalination specific for South Africa.

N. Todreas, MIT: Can I build on that maybe in answer to his question? I would say with that focus and the ability, as you went to successive improvements in fuel fabrication and fuel reliability, you could actually enhance your safety profile if the key focus is fuel, and that would be an answer back to how you reflect the future, the fuel.

J. Slabber, PBMR: I think the objective of any new innovative system should be to improve, but there is a limit because it's also costly. So improvement, the improvement for public acceptance, improvement of safety, that at the boundary you do not have to shelter and evacuate. These are all factored in to provide a facility which is still affordable and reliable.

Establishing a Safety and Licensing Basis for Generation IV Advanced Reactors

"License By Test"

Andrew C. Kadak
Massachusetts Institute of Technology

Workshop on Advanced Reactors
Advisory Committee on Reactor Safeguards
June 4-5, 2001

Background

The regulations for nuclear power plants (NPPs), as codified in 10 CFR Part 50 and the general design criteria (GDC), were formulated more than thirty years ago. The objectives were to protect public health and safety by making the probabilities of accidents very low and providing mitigating capabilities in case such accidents occurred. Since methods for quantifying these probabilities were not available at the time, a conservative safety philosophy emerged that was called defense in depth. Much has been learned since that time. The development of probabilistic risk assessment (PRA) has allowed the quantification of the probabilities of accident sequences, thus providing analytical tools for the evaluation of individual defense-in-depth measures. PRA applications have shown that many of these measures do not contribute significantly to safety and have been termed an "unnecessary burden." At the same time, the integrated PRA approach to reactor safety has identified the need for additional requirements in some instances that were not foreseen by the traditional, defense-in-depth based, regulatory system.

To gain maximum benefit from design innovation of advanced reactors, engineers must utilize these advances in our ability to assess the safety of NPPs. Unfortunately, the current regulations have not been changed to a significant degree to reflect PRA insights and capabilities. One of the major impediments to the Electric Power Research Institute's advanced light water reactor (ALWR) initiatives that led to the design criteria for ALWRs was the objective of trying to comply with existing regulations. This effort, while making some improvements in design, resulted in plants that were still limited in terms of innovation. The ALWRs are considered to be too expensive for the US market due to these requirements that are arguably still too prescriptive.

As the United States proceeds to develop a new generation of nuclear energy plants commonly referred to as "Generation IV," a new regulatory approach is needed that will allow for innovation and that captures the important features of safety. The existing regulations have been developed for water-based reactors and are not easily adaptable to other technologies such as gas cooled reactors. At the present time, the NRC is working on "risk informing" their regulations. This

process is still largely focused on light water reactors is very difficult; and will take many years to complete implementation (should that ever occur). Once concluded, the result is expected to be a mix of deterministic, probabilistic, and subjective defense- in-depth criteria that will be an improvement, but still water based.

There is a recognized need by the International Atomic Energy Agency[1] and by the technical community worldwide that for true innovation in reactor design, a new approach to establishing the safety of advanced nuclear power plants is required. This new approach is required not only for innovation in design but also for a rational process of licensing and deployment in a timely way.

For the next generation of advanced nuclear power plants, it is likely to be necessary for safety regulation to proceed under some modified version of the current NRC safety regulatory system. The reason for this is that replacement of the current system is likely to demand such time and resources that the new system may not become available in a timely fashion. The existing regulatory system may, in fact, deter deployment of innovative reactor designs. Thus, it is important to formulate a safety basis for a licensing approach for advanced reactors to be pursued under the current system. In doing this we wish to identify areas where current regulations do not address the safety issues of new reactor concepts;; to utilize our risk-based regulatory approach for development of proposed treatments of these latter areas; and to formulate ways of utilizing a demonstration plant in conjunction with a testing program in future licensing.

Structuralist vs Rationalist Approach to Defense in Depth

Current regulations and standards are based, in large part, on the principles of defense-in-depth (DID) and safety margins, which have evolved since the first reactors were designed in the 1940s. The NRC's Advisory Committee on Reactor Safeguards (ACRS)[2] and Senior Fellow John Sorensen et al[3]. discuss this evolution, identify two schools of thought on DID, labeled "structuralist" and "rationalist," and recommend an approach for risk-informed regulation.

The two schools differ in the process used to deal with uncertainty in reaching an acceptable level of safety. The structuralist approach has evolved from the early days of nuclear power with a process of accumulating DID features until a

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[1] M. Gasparini, "Verification of Defense in Depth for Operating Nuclear Power Plants," draft September, 1999.

[2] Letter to Shirley Ann Jackson, Chairman, U.S. Nuclear Regulatory Commission, from D.A. Powers, Chairman, Advisory Committee on Reactor Safeguards, Subject: The Role of Defense in Depth in a Risk-Informed Regulatory System, May 19, 1999.

[3] Sorensen, J.N., Apostolakis, G.E., Kress, T.S., and Powers D.A., On the Role of Defense-in-Depth in Risk-Informed Regulation. Proceedings of The International Topical Meeting on Probabilistic Safety Assessment, Washington, DC, pp. 408-413, 1999.

judgment was made that sufficient protection against uncertainty in performance had been achieved. With the development of PRA methods, the rationalist approach uses these tools to quantify uncertainty and to explicitly account for DID features in reducing uncertainties to acceptable levels. The main difference is that the structuralist accepts DID as a fundamental principle, while the rationalist would place DID in a subsidiary role. Additionally, the structuralist does not deal with uncertainties in a quantitative manner, while the rationalist takes advantage of the fact that advances in PRA allow the quantitative estimation of some of these uncertainties. For new plants, the rationalist approach to DID, employed within the context of PRA, is preferred to more effectively develop a body of regulations that eliminates requirements that do not contribute significantly to safety.

The rationalist relies on PRA methods to provide an integrated and systematic analysis of the plant that explicitly addresses sources of uncertainty. The process envisioned by the rationalist is: establish quantitative safety goals, such as health objectives, core damage frequency, and large release frequency; design and analyze the plant using PRA methods to establish that the safety goals are met; evaluate the uncertainties in the analysis, including those due to model inadequacies, system performance and reliability, and lack of knowledge; and determine what steps (i.e., DID, new design features) to take to address those uncertainties. The quantification of uncertainties in terms of probability distribution functions provides a means for determining how much redundancy and diversity (i.e., DID) is sufficient.

Discussion of the Risk-Based Framework

The framework we propose for risk-based regulation and design is illustrated in Figure 1. A top-down hierarchy, indicated on the left side of Figure 1, is being used to define the goal, establish an overall approach, and develop and implement appropriate strategies and tactics. The framework is based on an application of PRA methods and reflects a rationalist approach to DID.

Regulations for NPPs are required to ensure adequate protection to the health and safety of the public. Accordingly, the *goal* of this effort is to provide a framework for developing and implementing risk-based regulations that meet this requirement. An *approach* based on evaluating risk against quantitative safety goals is proposed to achieve the stated goal. With respect to adequate protection, the NRC has established safety goals including quantitative health objectives (QHOs) that state the Commission expectations with respect to how safe is safe enough. Although the NRC safety goals are not considered quantitative measures of adequate protection, for new plants, we will consider the determination of adequate protection using increased reliance on comparisons of PRA results to quantitative risk measures. The safety goals we are using for the framework, indicated in the gray boxes in Figure 1, have been adapted from the NRC goals.

The *strategies* for developing and evaluating compliance with requirements for risk-based regulation and design are based on the use of PRA to quantify risk and uncertainties. High confidence is achieved through explicit consideration of uncertainties, including modeling adequacy and equipment design and performance. These strategies include consideration of the risk information available from Level 1, Level 2, and Level 3 PRA analyses. Level 1 PRA evaluates the potential for accident initiators and the system response to prevent core damage. An estimate of core damage frequency is compared to the corresponding goal. Level 2 PRA encompasses the response to and mitigation of core damage, including containment of fission products. Risk estimates here can be compared to goals for conditional probability of large release, both early and late. Level 3 PRA encompasses the response to and mitigation of radionuclide releases, including emergency response. These risk estimates can be directly compared to the QHOs or to subsidiary goals for conditional probability of early fatalities and latent cancer risks.

To develop risk-based regulations, *implementation* of the framework is achieved by defining functional system characteristics, within the context of how PRA is performed, to determine what areas need to be regulated to assure safety. Implementation for design is achieved by specifying design configurations and using PRA to evaluate the design, then iterating with subsequent design changes. A master logic diagram (MLD), illustrated in Figure 2, is used to take a top-down approach to identify the safety functions, and systems, structures, and components (SSCs) that are required to maintain safety and to identify the accident initiators and system response failures that could compromise safety[4]. The top event is stated in terms of risk exceeding the safety goals. The gray shaded events correspond to the Level 1, Level 2, and Level 3 PRA strategies, respectively, in Figure 1. The sixth level of the MLD defines the system functions that are required to assure safety. The next level down indicates that initiating events and failure of mitigating systems, containment, and emergency response can compromise safety functions. The last level of the MLD indicates that internal initiators for all operating modes and external initiators will be considered for completeness. Further development of the MLD will determine the "regulatory risk space" for which regulatory and design requirements are needed.

Various *tactics* (e.g., design criteria, procedures, redundancy, emergency response, etc.) are applied to support the PRA strategies and implementation. Once the SSCs required to achieve safety have been identified, then decisions on appropriate tactics for regulation and design can be made. The specification of these tactics will be based on a systematic evaluation of the areas that need to be regulated for the purposes of assuring safety and will also evolve from this

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[4] Apostolakis, G.E., Some Issues Related to Goal Allocation and Performance Criteria. Proceedings of the 8th International Conference on Structural Mechanics in Reactor Technology, Brussels, Belgium, Paper M2 4/3, 1985

process.

Work on the practical implementation of the risk-based regulatory approach has proceeded focusing upon how to formulate design basis accidents (DBAs) within a risk-based system[5] In this work, it was concluded that DBAs are not necessary within such a system, and rather are replaced by the set of risk-dominant accident event sequences as the focus of judging whether a nuclear power plant design is adequate, and for negotiating between the license applicant and regulator concerning what changes in the design application are required before a license can be issued. These risk dominant accident sequences will be used in the development of a "license by test" approach to safety verification and licensing.

The thinking on how to approach establishing a safety basis and then licensing non-traditional nuclear technologies was advanced in 2000 by ESKOM, the South African utility proposing to build a pebble bed reactor at their Koeberg nuclear plant site in Capetown. ESKOM, through its Pebble Bed Modular Reactor Development Company, PBMR Pty, issued "PBMR Safety Case Philosophy" - PNL-001 Rev 1, (10) and the Safety Case Support Document - PNL-009 (11), that identifies the approach to safety being proposed by ESKOM for eventual licensing in South Africa. This approach calls for the use of PRA to establish general design criteria for the pebble bed reactor. Their approach is consistent with South African's National Nuclear Regulator's approach to safety that is illustrated on Table 1 which is safety goal based and uses a combination of deterministic and probabilistic techniques to establish general design criteria and tests that need to be performed to demonstrate safety. What is proposed by ESKOM is whenever there is inadequate information or large uncertainty to bound that uncertainty and perform tests on the prototype reactor to assess impact. It is judged that these guidelines can form the basis of a workable risk informed safety basis that will allow a license by test approach.

License by Test (LBT)

Both this effort and the work being performed internationally are general and could be applied to non-traditional technologies. What is needed is an extension of this process to specific reactor technologies, such as the gas reactor, and to include the concept of "license by test." The aim would be to improve safety and reduce the time and effort to certify new designs while not compromising safety.

This weakness in the generic applicability of the existing regulatory system is not easily addressed. An approach that historically was provided in the regulations

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[5] B. C. Beer, G.E. Apostolakis, and, M.W. Golay, "Methods for Formulation of Design Basis Accidents Within a Risk-Informed Approach to Safety Regulation of New Nuclear Power Plants," PSAM 5 Conference, Osaka Japan (November, 2000). [Included as Appendix YYY]

was "license by test." In this concept, a reactor prototype could be built and tested to demonstrate the safety characteristics and on that basis granted a license. This approach has to our knowledge never been applied to an entire nuclear plant but only subsystems which were tested to calibrate performance.

With the advent of new safer plants that derive their safety from inherent deterministic safety features as opposed to active or passive safety systems that must work, we have an opportunity to apply the license by test concept on a real plant on an integrated basis. The challenge is to develop the test envelope to validate safety in a licensing sense.

What is desired is to avoid years of paper analyses and simulations which can be costly and still leave doubts in the regulator's and the public's mind about the real safety of the plant. As an alternative to spending the huge amounts on paper analyses, separate effects tests, it is proposed to design and build a prototype of a plant that meets the fundamental safety standards relative to public health risk. Using a combination of deterministic and probabilistic tools as described above to develop subsystem and integral tests that would be conducted with this prototype (full scale if the plant is small enough) to demonstrate the safety of the plant. In this case, we would not only have a certified plant design and a license at the conclusion of the process, but also a nuclear power plant that the public could see met the safety standard and that could also be used to produce electricity.

This licensing concept has been suggested as being especially applicable to the PBMR.The basic idea of LBT is that a set of integral tests of a full size reactor and associated systems be used to demonstrate the ability of the overall system to mitigate successfully a set of "bounding" accident cases. Should the set be sufficiently comprehensive (e.g., loss of coolant inventory and loss of coolant flow), and should they be sufficiently more demanding than expected accident situations. The argument goes that successful performance in these tests should be sufficient proof of the safety of the nuclear power plant concept.

These tests, even if partially successful, can also serve another very important purpose, namely, to validate the computer codes that have been developed for design and accident analysis. The validation of such codes is always an issue when a new design, even water-based, is submitted to the NRC. For example, a recent ACRS letter[6] recommended that the validity of predictive codes such as NOTRUMP that have been approved for the design certification of AP600 must be demonstrated before they are used in the certification process for AP1000. What is hoped is that the full scale tests will be able to answer remaining questions for the validation of codes. This approach can only be possible for reactors that have significant safety margins which is a criteria for new advanced reactors.

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[6] Report from D.A. Powers, ACRS Chairman, to R.A. Meserve, NRC Chairman. Subject: "Pre-Application Review of the AP1000 Standard Plant Design – Phase I."

The Challenge of Licensing by Test

The challenge is to define a process that would identify what range of testing would be required for the granting of a license. For example, what design information needs to be available to allow for the construction of the prototype and the extent of review by the regulator required for the test program? What criteria constitute success or failure of the test program leading to the granting or denial of the license?

It is our hope to work with the NRC to develop the guidance to allow such a process to work such that advanced non water (or water based) technologies could be used without spending years on inconclusive (from the standpoint of the public perception) analyses.

We need to develop this concept in more detail; establish clear performance indicators for success; identify review requirements for the design and construction of the prototype facility; and specify the test program that would be used to certify the design for mass production.

Reference Plant Selection

At the present time, there are many advanced reactor projects under development. The three most significant are the high temperature pebble bed gas reactor, the light water IRIS project and several variants using lead bismuth. All new reactor concepts would benefit from the application of the safety framework approach using PRA technology in a methodology that identifies important safety features and requirements.

The modular pebble bed reactor has been selected as be the reference plant used as a demonstration for the new safety and licensing approach since it provides some unique challenges to conventional wisdom about safety and traditional licensing. As result of the leadership and development of this technology for commercial application by ESKOM, the South African utility and MIT's three year conceptual development project on the PBMR, there is sufficient design and knowledge information to allow this plant to be the advanced reference design. Eventually, this plant will be constructed in the US since PECO and BNFL, investors in the South African PBMR, have indicated that they would like to use the South African plant as the prototype for the US plants. In addition, I have been proposing the construction of a "reactor research facility"site which can be used to test the concepts proposed by this project. In either case, a new approach to the safety basis of advanced plants will be required to allow for timely introduction of new nuclear plants in the US. The licensing strategy for the research facility relies upon "license by test" that needs a workable risk-informed regulatory framework in which to succeed.

This is a novel approach to safety and licensing that is made possible by the integrated approach to reactor safety that PRA provides. Whether this approach

is applied to the South African prototype for US application or for a new "reactor research facility" built on a DOE site, it is important to develop the safety framework upon which to license the plant.

Establishing a Safety and Licensing Basis for the Plant Within the Current Regulatory System

In order for license by test to be successful in the short term, it must fall within the guidance of the current regulatory system. There are several steps that need to be taken to assess whether this will work. These steps are outlined below.

1. Review literature on safety bases used by the IAEA, NRC, and the South African regulator. Focusing on risk-based safety goal performance criteria addressing defense-in-depth issues.[7], [8] The NRC regulations should be reviewed as would the GDC for applicability. This will also include understanding the relationships for normal operation, severe accidents, containment of fission products and public health and safety objectives in the context of fundamental design objectives as outlined by IAEA and others. It will be necessary to identify areas in the current NRC regulatory system, utilizing available precedents, which currently address PBMR-relevant regulatory questions. NRC's regulations will be reviewed to identify opportunities for establishing license by test approach to demonstrate safety.

2. Develop risk-based approach to address safety basis and regulatory questions which are not treated by existing regulations, and then to use risk-based logic for refining the existing regulatory structure.

3. Within the context of the existing high level regulatory design objectives, develop a license by test process that can demonstrate functional compliance to these design objectives. Identify those areas where a "Licensing by Test" approach could answer unresolved questions safely within the existing regulatory structure and seeking applications of it in the overall PBMR regulatory strategy.

The great safety licensing problem faced by the PBMR in the United States is that no body of regulations exists for licensing it comparable in detail than that available for LWRs. Much of the current regulatory literature and policies have been concerned with LWR-specific problems or features (e.g., metal-water interactions following onset of critical heat flux; pressurized thermal shock). This emphasis creates the need to ensure that LWR-influenced treatments of safety issues will not be incorrectly applied to PBMR safety questions and

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[7] Clappisson, Metcalf, Mysen"PBMR-SA Licensing Project Organization", November, 1999, Beijing, China.
[8] Kadak,"Risk Based Regulation – The Time is Now", PSA-99, International Topical Meeting on Probabilistic Safety Assessment, August, 1999.

requirements, also. With diligence and effort this pitfall can be avoided, but recognizing it is the start of doing so.

The precedents created by the Ft. St. Vrain and Peach Bottom gas-cooled reactors will be useful in PBMR licensing, but the overall regulatory structure has changed considerably since the licenses for the reactors were issued (both prior to the 1979 Three Mile Island reactor accident), and they are silent concerning such PBMR aspects as reliance upon highly reliable fuels and the stochastic nature of the individual fuel pebble histories.

License by Test Approach

1. Use of the risk-based regulatory approach to fill identified regulatory gaps

Using the risk-based framework outlined above, establish the detailed steps and information required to use the logic of the risk based framework for developing proposed treatments of the regulatory gaps identified. A review of accident scenarios leading to release of fission products needs to be performed. Deterministic analysis will be performed using the VSOP code and other transient analysis code packages to evaluate the expected plant performance in accident scenarios for the pebble bed plant.

This task is critical. On the one hand, we recognize that the risk-based framework that we have discussed is not part of the regulations at this time, although it is consistent with the NRC's Option 3 for risk-informing 10 CFR Part 50. On the other hand, the existing regulations are LWR-based and will have to be modified for application to the PBMR. Appendix A to 10 CFR Part 50 states: "The General Design Criteria are also considered to be generally applicable to other types of nuclear power plants and are intended to provide guidance in establishing the principal design criteria for such other units." This step will investigate how the GDCs and other regulations could be utilized in a risk-based framework to actually provide a basis for design and licensing of a gas reactor.

2. Perform a PRA on the Reference Plant

Perform a PRA on the MIT design or obtain a PRA of the PBMR from ESKOM to identify major accident sequences that affect safety goal attainment to identify critical systems requiring test. This PRA should be at least a level 2 if not level 3 PRA in order to support the public health and safety goal philosophy.

3. Use of the risk-based approach for development of traditional deterministic elements of the current regulatory approach

Develop a risk-based technical basis for establishing risk dominant accident sequences (RDAS) that could form what are traditionally referred to as Design Basis Accidents (DBAs). The RDAS are used to formulate acceptable essential

safety function unavailabilities and for judging defense in depth. Defense-in-depth is a general and somewhat subjective philosophy used by regulators and designers to cover unquantified uncertainties. IAEA and the ACRS have recently begun to address this issue again. The objective of this step is to better define this term given the advances in analytical methods to assess risk and to allow for innovative new designs may have inherent physical attributes that are demonstrably, rather than analytically, safe. This step will examine the basis of defense in depth relative to gas reactors and how it can be generically applied to all advanced reactor designs. Our risk-based approach offers a method for doing this, by evaluating acceptable risks at a conservative confidence level rather than in terms only of expected outcomes.

4. Develop Test Plan for Certification

Based on the risk assessment identifying critical safety component, systems and structures, develop a series of subsystem and integral tests to confirm the performance of the components and systems as required to validate performance and computer codes. These tests will form the basis of the safety case for the technology. This test plan is expected to include the loss of coolant test, air and water ingress, reactivity feedback tests, control rod withdrawal, turbine trips, overspeed, to name a few. What will be necessary is to demonstrate how a safety and licensing basis could be established.

5. License by Test Certification

Once the test program is completed and reviewed, a general certification for the design can be issued. It is recognized that not all areas of performance can be tested using a real reactor, but as a result of the tests, the computer codes that are used to demonstrate performance can be validated and by extension justified for broader application. This area needs to be reviewed to assure that the tests provide an adequate justification for application over the range of conditions expected.

Conclusions:

The license by test approach to licensing is a novel method of licensing reactors. It provides an opportunity to deal with innovative non-water reactors in a direct way on a time scale that could permit early certification based on tests of a demonstration reactor. The uncertainties in the design and significant contributors to risk would be identified in the PRA during the design. Deterministic analysis computer codes could be tested on a real reactor. Scaling effects and associated uncertainties would be minimized. License by test is an approach that has sufficient merit to be developed and tested.

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2. F.A. Duran, A.L. Camp, G.E. Apostolakis, M.W. Golay, "Framework for Regulatory Requirements and Industry Standards for New Nuclear Power Plants", PSAM 5 Conference, Osaka, Japan, November, 2000.

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7. Report from D.A. Powers, ACRS Chairman, to R.A. Meserve, NRC Chairman. Subject: "Pre-Application Review of the AP1000 Standard Plant Design – Phase I."

8. Clappisson, Metcalf, Mysen "PBMR-SA Licensing Project Organization", November, 1999, Beijing, China

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11. ESKOM, "Safety Assessment Philosophy – Support Document", PNL-009, November 2000.

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Figure 1. Framework for risk-based regulation and design.

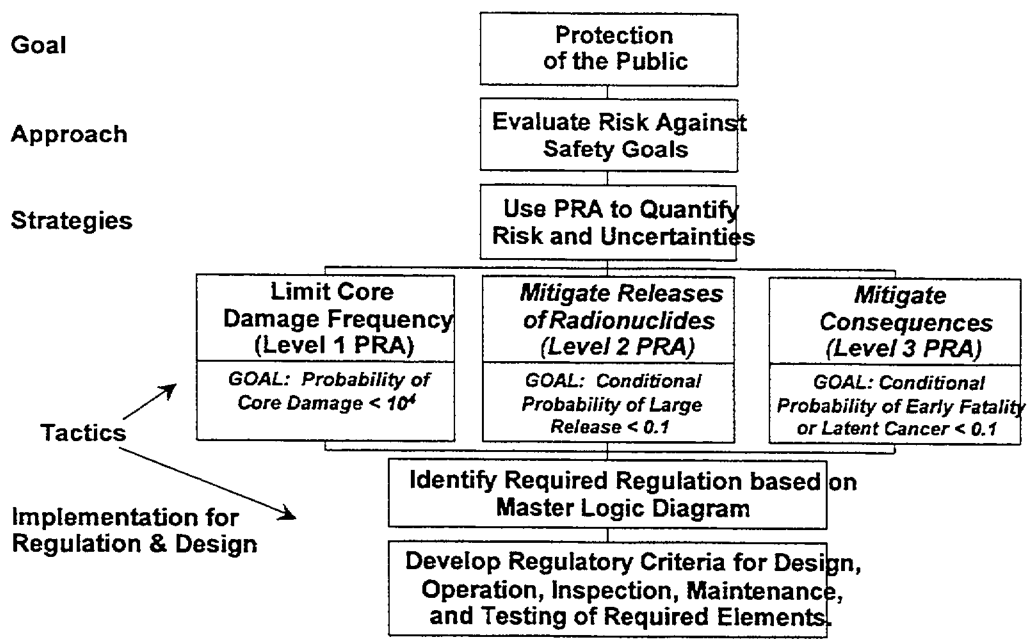


Figure 2. Example master logic diagram for framework implementation. (for light water reactors)

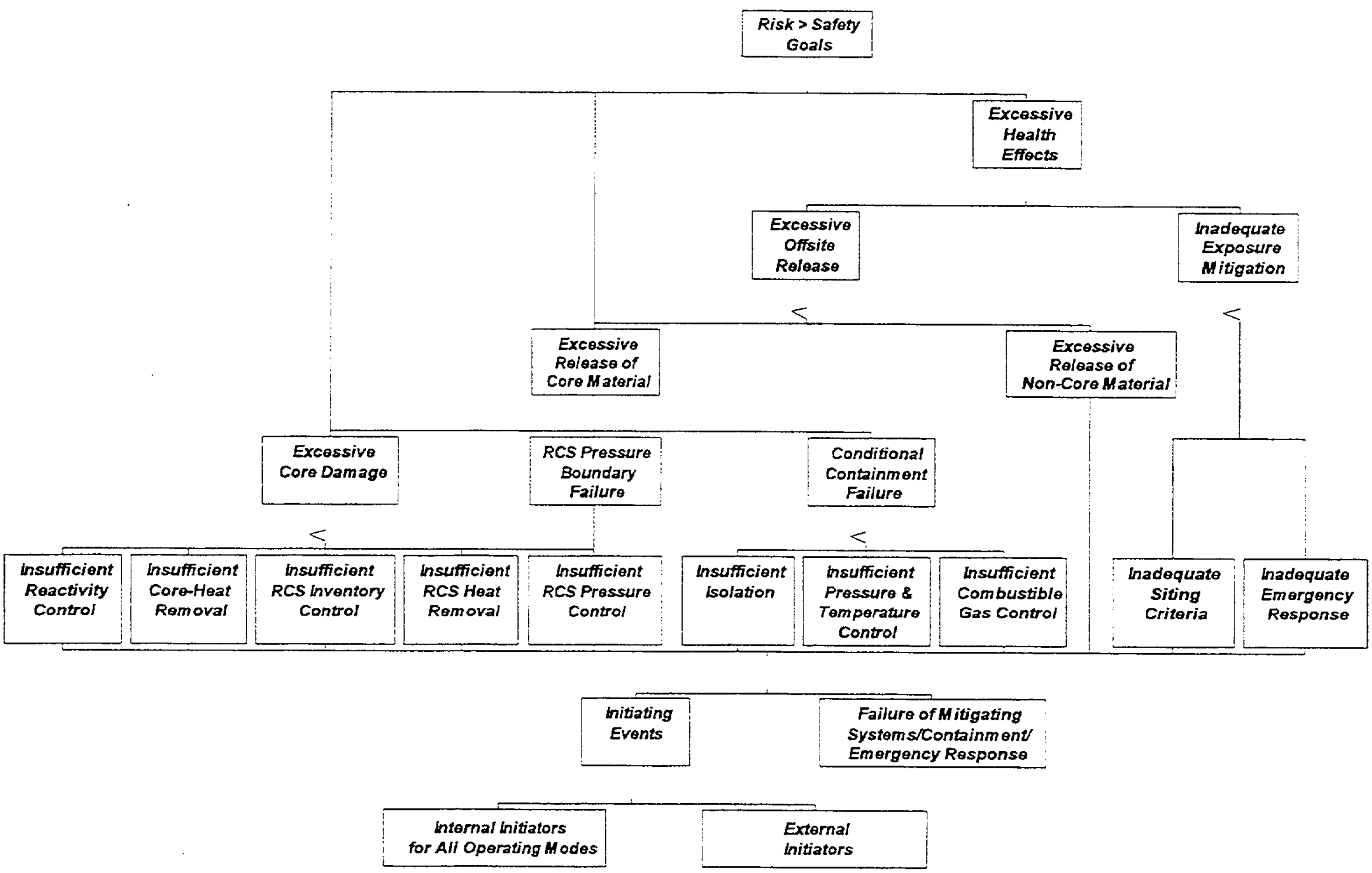


Table 1

Council for Nuclear Safety Licensing Approach
For the Pebble Bed Modular Reactor (PBMR)

	SAFETY REQUIREMENTS	EVENT FREQUENCY	SAFETY CRITERIA
a	The design shall be such to Ensure that under anticipated Conditions of normal operation There shall be no radiation hazard To the workforce and members of The public. This must be Demonstrated by conservative deterministic analysis.	Normal operational conditions shall be those which may occur with a frequency up to but not exceeding 10^{-2} per annum.	Individual radiation dose limits per annum of 20 mSv to workers and 250 μSv to members of the public shall not be exceeded. +ALARA+ Defense in depth criteria
b	Design to be such to prevent and mitigate potential equipment failure Or withstand externally or internally originating events which could give Rise to plant damage leading to Radiation hazards to workers or the public. This must be demonstrated By conservative deterministic Analysis.	Events with a frequency in the range 10^{-2} to 10^{-6} per annum shall be considered.	Radiation doses of 500 mSv to workers and 50 mSv to members of the public shall not be exceeded. +ALARA+ Defense in depth criteria
c	The design shall be demonstrated To respect the CNS risk criteria. This must be demonstrated by probabilistic risk assessment using Best estimate + uncertainty analysis.	Consideration shall be given to all possible event sequences.	CNS risk criteria apply. $5X10^{-6}$ Individual risk 10^{-8} Population risk Bias against larger accidents. +ALARA

(CNS is the former name of the National Nuclear Regulator)

Licensing Approach for Generation IV Technologies

"License By Test"

Andrew C. Kadak
Massachusetts Institute of Technology
June 5, 2001

Challenges

- Regulations focused on water
- Knowledge of technology lacking
- Regulatory System Rigid
- Infrastructure to Support New Technology Not Developed
- Changes in System take a long time

How to Introduce New Technology in Less than a Lifetime ?

- Go Back to Basic Safety Fundamentals
- Work Within Existing Regulatory High Level Objectives
- Use Risk Informed - Risk Based and Deterministic Analysis
- Assess Gaps in Knowledge
- Prioritize (risk assess)
- License by Test

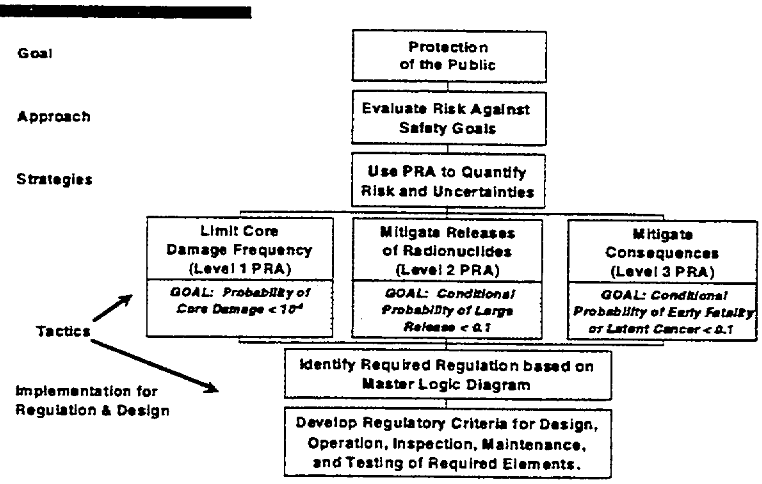
Establish a Safety Basis

- Use Public Health & Safety Goal
- Define Plant Risks:
  - Normal Operating Plant
  - Transients
  - Accident Scenarios
- Identify Safety Margins
- Quantify Risks
- Show Defense in Depth

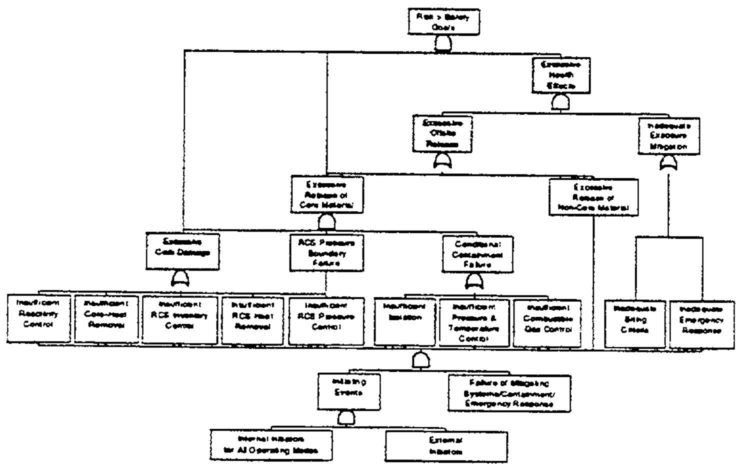
Risk Informed Approach

- Establish a public health and safety goal
- Demonstrate by a combination of deterministic and probabilistic techniques that the safety goal is met.
- Using risk based techniques identify dominant accident scenarios, critical systems and components that need to be tested as a functional system

Risk Informed Approach



Master Logic Diagram for Water Reactors



Council for Nuclear Safety Licensing Approach
For the Pebble Bed Modular Reactor (PBMR)

	SAFETY REQUIREMENTS	EVENT FREQUENCY	SAFETY CRITERIA
a	The design shall be such to Ensure that under anticipated Conditions of normal operation There shall be no radiation hazard To the workforce and members of The public. This must be Demonstrated by conservative deterministic analysis.	Normal operational conditions shall be those which may occur with a frequency up to but not exceeding 10^{-2} per annum.	Individual radiation dose limits per annum of 20 mSv to workers and 250 µSv to members of the public shall not be exceeded. +ALARA+ Defense in depth criteria
b	Design to be such to prevent and mitigate potential equipment failure Or withstand externally or internally originating events which could give Rise to plant damage leading to Radiation hazards to workers or the public. This must be demonstrated By conservative deterministic Analysis.	Events with a frequency in the range 10^{-2} to 10^{-6} per annum shall be considered.	Radiation doses of 500 mSv to workers and 50 mSv to members of the public shall not be exceeded. +ALARA+ Defense in depth criteria
c	The design shall be demonstrated To respect the CNS risk criteria. This must be demonstrated by probabilistic risk assessment using Best estimate + uncertainty analysis.	Consideration shall be given to all possible event sequences.	CNS risk criteria apply. $5X10^{-6}$ individual risk 10^{-8} Population risk Bias against larger accidents. +ALARA

(CNS is the former name of the National Nuclear Regulator)

Review Existing Regulatory Structure for Gaps

- Based on plant specific safety basis:
  - Identify existing regulations that apply.
  - Use risk based regulatory approach to fill in gaps for areas not covered.
  - Develop implementation approach to General Design Criteria.

Develop Traditional Deterministic Regulatory Approach

- Establish Design Basis Accidents using risk based techniques
- Develop Defense in Depth Basis Using natural physical attributes of designs
- Establish confidence levels for analysis using risk assessment methods

License By Test

- Build Full Size Demonstration Plant
- Perform Critical Tests on components and systems identified using risk informed techniques
- If Successful, Certify Design

Why License By Test ?

- Needs:
  - To validate analyses
  - To shorten time for paper reviews
  - To "prove" what is debatable
  - To reduce uncertainty
  - Show Public and NRC that plant is safe

Tests Required

- Traditional Performance tests of equipment still required for reliability
- Use Risk Based Techniques to identify:
  - Accident Scenarios of Importance
  - Critical Systems
  - Critical Components
- Conduct Integrated System Tests

Examples of Tests

- Loss of Coolant
- Reactor Depressurization
- Natural Circulation
- Rod Withdrawal
- Reactivity Shutdown Mechanisms
- Reactor Cavity Heat Up and Removal
- Selected Component Key Component Failures

Additional Tests

- Balance of Plant Failures - turbine overspeed, loss of heat sink, compressor failures, etc
- Control Rod Ejection (rapid withdrawal)
- Reactor Cavity Heat Up
- Validate Core Physics Models
- Validate Safety Analysis Codes and Methods
- Xenon Transients

Tests Leading Up to Demonstration Facility Tests

- Fuel Performance - Irradiation, post accident heat up, cycling
- Air Ingress - validate chimney model for air ingress potential
- Water Ingress - assess reactivity effect and fuel damage

Reactor Research Facility

- Pebble Bed Reactor as a prototype for this licensing approach.
- Built in Idaho - Full Size w/Containment
- Implement Structured Test Program
- Develop Regulatory Process as Part of Certification of Technology using RRF.
- Research Reactor Continues as facility to innovate and test new technologies for fleet of standard designs.

Will License By Test Be Able to Answer All Questions ?

- No...
- In combination of subtier component tests described and the risk informed analysis, it should provide high confidence of critical safety performance.

Will License by test instill public confidence ?

- Yes,
- By having the public and the media observe these tests, the confidence in the technology and the regulatory will be enhanced.
- $10^{-\text{(pick a number)}}$ is not understandable or effective in safety discussions.
- It will encourage development of naturally safe reactors.

Traditional Regulatory Approach

- Ask General Atomics for MHTGR
- Ask Canadians for Candu
- Ask W about AP-600 - 1000
- Costly - Time Consuming - Risky
- Answers Not always possible to Satisfy NRC staff - Ask Licensees.
- Need An Alternative to the "Bring me a Rock" Process.
- This may be it...

Summary

- For Non-traditional technologies, a new licensing approach is needed for timely deployment.
- Risk Informed Techniques with Safety Goals Appear to meet the Need.
- License By Test is the most direct means of answering difficult questions.
- LBT should increase public confidence.

D. Powers, ACRS Member: Why do you focus on fatalities?

A. Kadak, MIT: It's an easy measure. You could talk about injuries, if you like, as a separate measure.

D. Powers, ACRS Member: If we're going to learn something from accidents that have occurred, the most transparent consequence of Chernobyl has been radiation injuries rather than fatalities. Land contamination could arguably be the other thing that we've learned. Why not change the measures in response to things we've learned?

A. Kadak, MIT: We could do that. I'm not limiting it. I'm just saying establish something that everybody is comfortable with, and I mean societal comfort. And if it talks to land, if it talks to injuries or if it talks to fatalities, fatalities is the one that we now have.

G. Wallis, ACRS Member: You seem to be applying what we do today to what we might do tomorrow, and did you question whether we really need design basis accidents in their present form? Or would it be replaced by something else which might be less plant specific and be more effective?

A. Kadak, MIT: The process that I would recommend is developing dominant accident sequences as part of the regulatory process, and don't call them design basis accident.

T. Kress, Chairman, Future Reactor Subcommittee: What we attribute to integral tests are two purposes: one, to see if there's something going on that we hadn't thought of; two, to validate our computerized analytical tools so that they can be used in an extrapolatory sense to cover the things we can't do in the test. Would that be your view of what this test might do for you?

A. Kadak, MIT: The needs. Why? To validate analysis. Okay? To shorten the time for paper reviews; to try to prove in quotes what's debatable; to reduce uncertainty, and this is very important; to show the public and the NRC, and I include them as the public in this case, that the plan is, in fact, safe. And that's what it's all about. Can we do the -- you know, can we try to melt the core? If we believe that we can do it without melting the core, yes.

G. Wallis, ACRS Member: So what you should do is you should give an operator carte blanche to try to melt the core, and he or she will fail. Is that your test?

A. Kadak, MIT: Depending upon the design, yes. I mean, theoretically that would be the test, but I would structure it more carefully than that.

G. Wallis, ACRS Member: Are you asking for a kind of full scale LOFT test?

A. Kadak, MIT: Full scale LOFT test, I suppose in the sense of a LOCA. There will be others on a facility, and one of the things it avoids is to remember the scaling issue that you've had to fight over? I mean, clearly the scaling issue sort of goes away if you do a full scale plan or a large enough scale to be able to say scaling is not a factor.

D. Powers, ACRS Member: Let's look at that control rod ejection because it's a fun one to look at. The scenario that we're now worried about is one where the fuel had extremely high burn-up. How are you going to do that in your test?

A. Kadak, MIT: That would have to be outside of the reactor. There would be a whole series of fuel tests as part of this program.

D. Powers, ACRS Member: Well, and the problem that plagues the rod ejection accident is an argument over how it propagates within the whole course. So if you do this test at the CABRI facility with one rod, that doesn't answer the question. You need a whole bunch of rods.

A. Kadak, MIT: Well, I think we could do like I said, a rapid withdrawal, and we could model it from the standpoint of what we expect as a reactivity transfer and to see whether those codes, in fact can model the event.

D. Powers, ACRS Member: I mean, that's where the argument is, is whether the codes are right or not, and whether they give you the right amount of heat going into the clad and not into damaging fuel.

A. Kadak, MIT: Well, the first is the reactivity. Then we can go to heat, right?

D. Powers, ACRS Member: No. This is a time scale where those two are very coupled together. I think you ought to look at the experience they had at the Phoebus facility, which was doing an experiment, which amounted to melting down 21 fuel rods, two of which were fresh fuel and the rest of them were irradiated, and the public response prior to the first test there, and how eager they were to watch that particular test.

A. Kadak, MIT: I'll look it up. I'm not familiar with it.

E. Lyman, NCI: So you're proposing that the test facility go with a containment which is not the same containment that the pebble bed is planned to have?

A. Kadak, MIT: Only because it's a research facility.

E. Lyman, NCI: So I've heard the argument that the passive cooling of the pebble bed is incompatible with a leak tight containment and it would interfere with, for instance, the design basis LOCA heat removal.

A. Kadak, MIT: Well, we'd have to look at that to see whether or not and how we could make it compatible for this particular facility. We'd have to look at whether, in fact, we need to make additional modifications to the facility to accommodate the passive cooling feature.

T. Fabian, Nuclear Waste News: It's not as exciting as melting down the core, but I'm wondering if as part of your conceptual design process you've done the sort of things that the fusion materials program has done, is looking forward to end of plant life and looking at lower activation materials that are easier to dispose of, possibly easier to resmelt and reuse in a nuclear facility, designing the plant for decommissioning using robotics and remote technology; is any of this played a part in the design process?

A. Kadak, MIT: Not at this stage, although we are following what's going on in Germany as they're decommissioning their AVR reactor. Clearly, one of our initial objectives was to design a plant with decommissioning in mind, also having a lot of personal experience about decommissioning the Yankee Rowe plant. So I'm very sensitive to that issue. We haven't really looked at it, and we're not really at that level of detail yet.

L.E. Hochreiter, Penn State University: As an AP600 design certification survivor, I'm familiar with the testing that we had done and a number of questions that we got from the NRC, which were large. When you structure a test program, usually you build on separate effects tests to try to identify and create a model that you then put into an integral code, and then you use integral tests for verification of that model. I think one of the problems that we have in the water reactor technology world is that we don't have very good integral systems tests. The loft tests, which are the largest integral systems tests, that we've all used for a code validation, there's a lot of questions on the accuracy of the instrumentation, which are really measuring versus what you think you're measuring, and so forth. There may be a lot of potential problems for that in this type of a program unless it's structured very carefully, and then if you add the instrumentation that you want to add, you can start to distort the things that you're trying to measure.

So I think that you're -- I like the idea. I think that you really have a background of tests that you're going to have to provide in addition to a large, full scale test where you build the technology so that you can have confidence then in the code that you'll use to predict the test, which you'll then try to run in the facility. Otherwise you may have some unpleasant surprises. You'll have conflicting objectives in the design of the plant versus the measurements that you want to make. I mean, that's the problem that LOFT had.

A. Kadak, MIT: I think a lot of that stuff that we're talking about, some of which at least I should say has been done in Germany, we don't know. I don't know, first of all, and like it's sort of the code of record which essentially is based on no experience in the United States. We are learning how to use it, and it's got a lot of models built into it and has been benchmarked against some of the tests that they've done in Germany.

We would hopefully use that data, disrupt your test, but I think your point earlier is exactly right. This is a research facility. In order to be effective, it's got to be well instrumented, and that is going to cost much more money than just building a straight power plan.

W. Hauter, Public Citizen: Who should assume liability for this test? How does Price Anderson play into this? What kind of radiation releases is it appropriate to expose the public to? Should there be a public process, public hearings and so forth to determine if this is something that the public would want to buy into?

A. Kadak, MIT: Let me answer the last question first. I think clearly the public has to buy into this process, and relative to the public hearings, I'm not all that familiar with how that would occur. My sense is it would have a licensing proceeding, become a licensed and experimental facility, and if successful, probably another licensing facility would be ready for operation. The Price Anderson question, I'm not an expert on Price Anderson, but, you know, depending upon who ultimately ends up being the builder, whether it's DOE or some private government partnership, those people would obviously have to pay the insurance costs for that.

In terms of releases, you would design the test such that it would essentially address this.

J. Slabber, PBMR: I'm not claiming or proposing that part of the PBMR demonstration unit in South Africa be used as supplying all of the information to Andy Kadak, but part of our objective as a demo unit, and it's not a prototype; it's a demonstration unit; it will be instrumented to such an extent that critical parameters during transients, like load rejection, may be loss of coolant,

could be measured, and this is not making an open statement. We've got quite a good technological base for proposing something like this because in an AVR, they have done loss of coolant simulations, as well as reactivity excursion experiments. It is documented, and they found, and this is, again, coming back to the integrity and the quality of the fuel, that they did not observe any significant increase in releases, although the core was filled with fuel, with a variable degree of quality and burn-ups, and they've also substantiated the reactivity predictions, the temperature coefficient predictions. In fact, there is a base where we can stand on to claim that some of the tests that are proposed in such a reactor have got some supporting evidence in Germany.

A. Kadak, MIT: To the extent that it's appropriate and doable, I think many of these tests could be done on the south African demonstration facility. So the concept is a generic concept suitable for, I believe, any type of advanced reactor that has certain characteristics.

L.E. Hochreiter, Penn State University: One of the things that we dealt with a lot in the AP 600 was looking at uncertainty, uncertainty in the predictions, uncertainty in the analysis. Do you know if they've done that with these codes for the pebble bed in Germany?

A. Kadak, MIT: I don't know. I have not been able to get at some of the qualifications.

A NEW RISK-INFORMED DESIGN AND REGULATORY PROCESS

by

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Regulatory Challenges for Future Nuclear Power Plants
Advisory Committee on Reactor Safeguards
US Nuclear Regulatory Commission
Washington, DC

OVERVIEW

In a project funded by the U.S. Department of Energy (USDOE) in its Nuclear Energy Research Initiative Program, the authors have been involved in formulating a new risk-informed approach for nuclear safety regulation. We believe that this work is important because a new regulatory treatment is needed both for the licensing of new non-light water reactors (LWRs), and to rationalize the regulation of LWRs. It is common today for the plans for new reactor concepts to include proposals for how they should be licensed. The existence of such proposals is implicit evidence that the existing regulatory structure is inadequate for this purpose. Similarly, attempts to "risk inform" the regulations governing LWRs have made only small progress because of the complexity and inconsistency of the existing structure. Thus, we have concluded that a fresh start in formulating a regulatory structure is worth attempting. This paper describes the fundamental concepts of that attempt.

The overall purpose of the new approach, termed Risk-Informed Regulation, is to formulate a method of regulation that is logically consistent and devised so that both the reactor designer and regulator can work together in obtaining systems able to produce economical electricity safely. In this new system the traditional tools (deterministic and probabilistic analyses, tests and expert judgement) and treatments (defense-in-depth, conservatism) of safety regulation would still be employed, but the logic governing their use would be reversed from the current treatment. In the new treatment, probabilistic risk analysis (PRA) would be used as the paramount decision support tool, taking advantage of its ability to integrate all of the elements of system performance and to represent the uncertainties in the results. The latter is the most important reason for this choice, as the most difficult part of safety regulation is the treatment of uncertainties, not the assurance of expected performance.

STRUCTURE OF THE NEW REGULATORY APPROACH

The scope of the PRA would be made as large as that of the reactor system, including all of its performance phenomena. The models and data of the PRA would be supported by deterministic analytical results, and data to the extent feasible. However, as in the current regulatory system, the models and data of the PRA would require being complemented by subjective judgements where the former were inadequate. All of these elements play important roles in the current decision-making structure; the main departure from current practice would be making all of these treatments explicit within the PRA, therefore, decreasing the frequency of sometimes arbitrary judgments.

In the intended sense the PRA would be used as a vehicle for stating the beliefs of the designer and regulatory decision-maker; the foundation of their decisions. Thus, the PRA should be viewed as a Bayesian decision tool, and be used in order to take advantage of its capabilities in integration and inclusion of

uncertainties. In order to do this, all regulations must be formulated in terms of acceptable levels of unavailability of essential functions, including an acceptable level of uncertainty (e.g., the acceptability of system performance could be evaluated at a stated confidence level rather than in terms of the mean value as is typical currently).

Implied in this treatment is a hierarchy of acceptable performance goals. At the highest level societal Safety Goals would be used, supported by subgoals formulated at increasingly fine levels of detail as the hierarchical level of the goal would decrease (see Figure1).

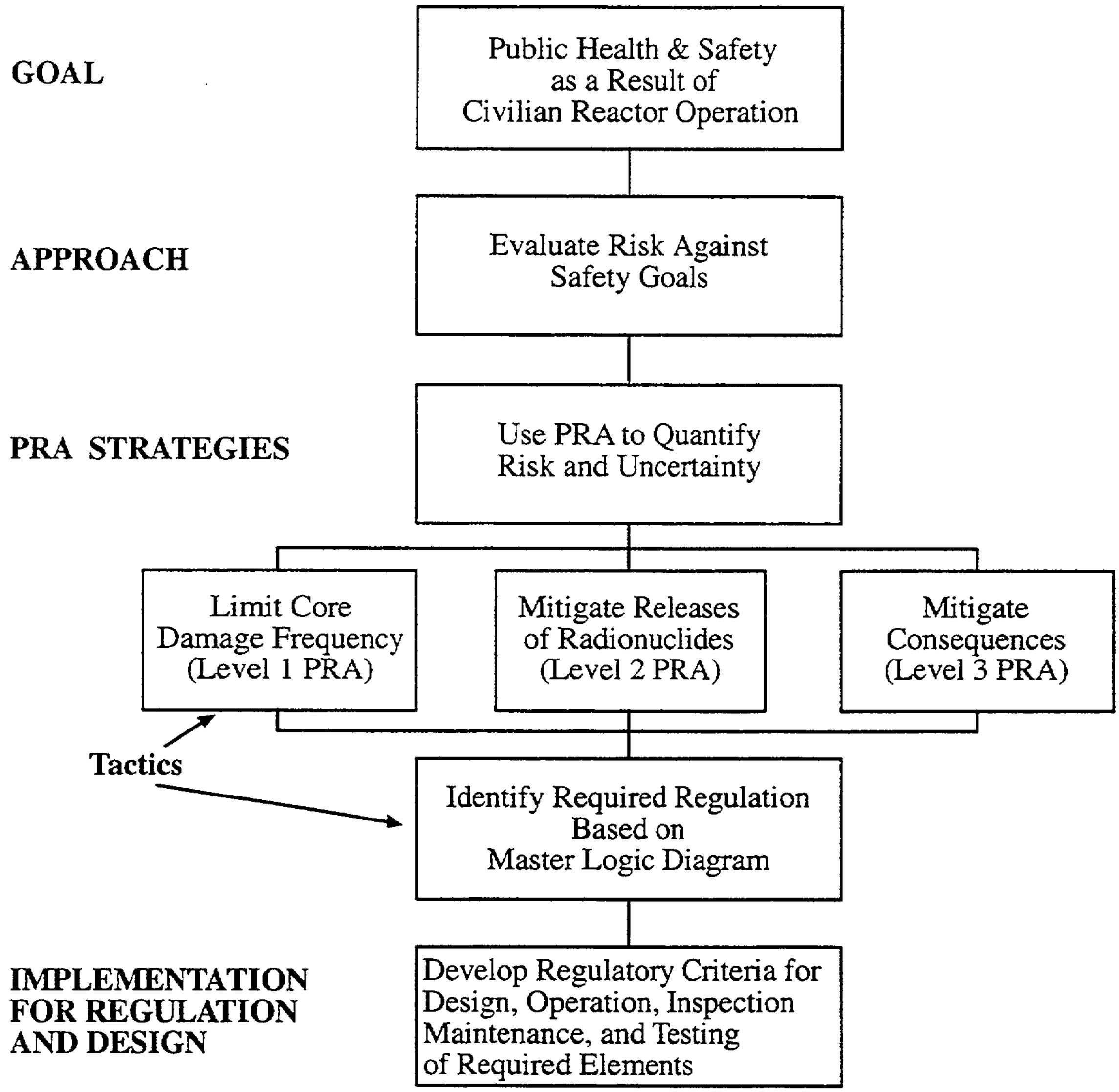


Figure 1. Framework for Risk-Based Regulation and Design

The differences between the proposed treatment and current practices are illustrated in Figure 2, which shows that the use of defense-in depth and requiring performance margins would remain. However, the current practice of permitting such features to be required without justification would be abandoned; rather, wherever such a requirement were to be made it would also be necessary for the regulator to provide evidence concerning the value of the requirement and to reflect that value in the master PRA (i.e., if a redundancy is to be worth including in a system, its safety value should also be stated in the overall system performance analysis).

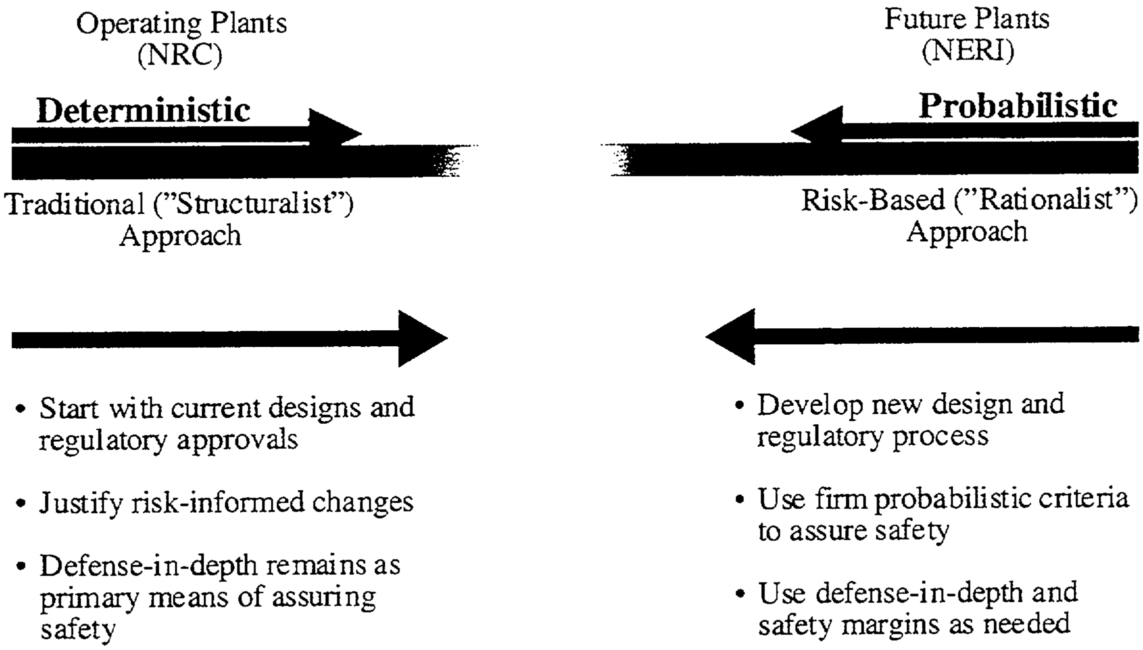


Figure 2. Comparison of NRC and NERI Risk-Informed Regulatory Processes

IMPLEMENTATION

In the licensing of any new reactor concept the degree of detail that the regulatory system may require will increase with the maturity of the concept (see Figure 3). When viewed from this perspective, it is seen that many aspects of the current LWR-focused system of safety regulation (e.g., general design criteria, design basis accidents) may not be applicable as the body of knowledge and experience needed for the formulation of new concepts will likely be unavailable in the earlier stages of their maturation. It is important to realize this in order that un-critical application of current requirements (e.g., a reactor containment building) not lead to impaired system performance or economically inefficient uses of resources. We suggest that some aspects of LWR-based regulation should not be applied to new reactor concepts without careful study.

As far as we can tell, the proposed regulatory approach can be applied to all areas of nuclear safety regulation (see Figure 4), including the "cornerstones" of the NRC's revised reactor oversight process. In the work of our project, we have focused upon the traditional areas of reactor licensing: determination of initiating events and requirements for mitigating systems, but nothing that we have done indicates an inability to extend the ideas being developed to all areas of regulation.

Determination of acceptable unavailability standards for a reactor's essential performance functions must be done on both combined general (high level) and reactor concept-specific bases (see Figure 5). The Master Logic Diagram (MLD) of Figure 5 is developed for the example of the pebble bed modular gas-cooled

reactor (PBMR). At each level of the MLD a set of performance goals must be formulated which are required to be consistent with those of the MLD levels immediately above and below the level of interest.

Development Stage	Goals and Acceptance Criteria	Evaluation Tools	Relevant Evidence
Initial Concept	High level - qualitative	Qualitative, simple, deterministic	Experiences of other concepts, deterministic analyses
Initial detailed design	High level - quantitative	Quantitative – probabilistic, deterministic	Prior quantitative analyses
Final detailed design	Detailed – quantitative (design-specific subgoals)	Detailed – quantitative – probabilistic, deterministic	Prior quantitative analyses
N-th of a kind for a given plant type	Very detailed – quantitative (design specific criteria – DBAs, GDCs,....)	Very detailed – quantitative, probabilistic, deterministic, tests	Prior quantitative analyses, tests, field experience

Figure 3. Stages of Nuclear Power Plant Concept Development

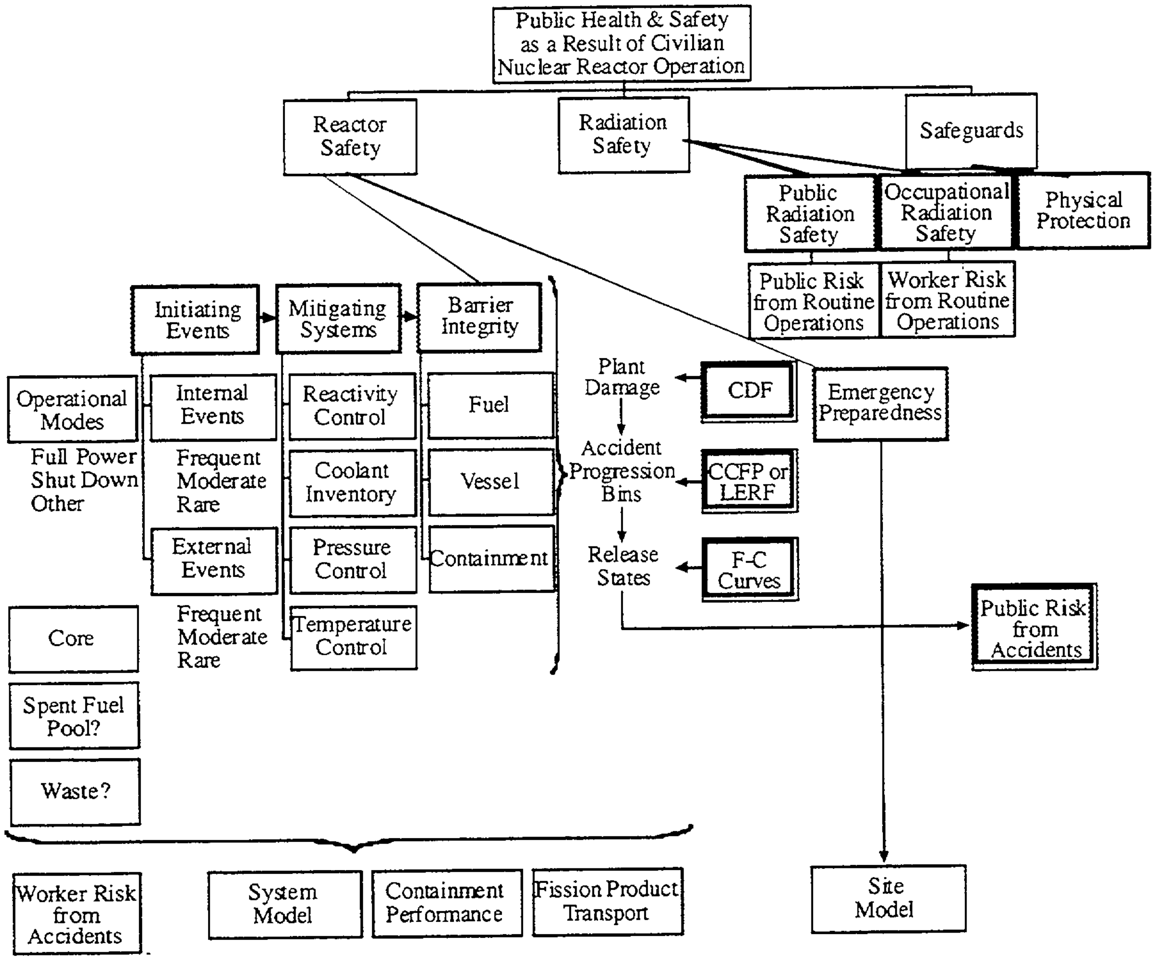


Figure 4. Scope of New Regulatory Scheme

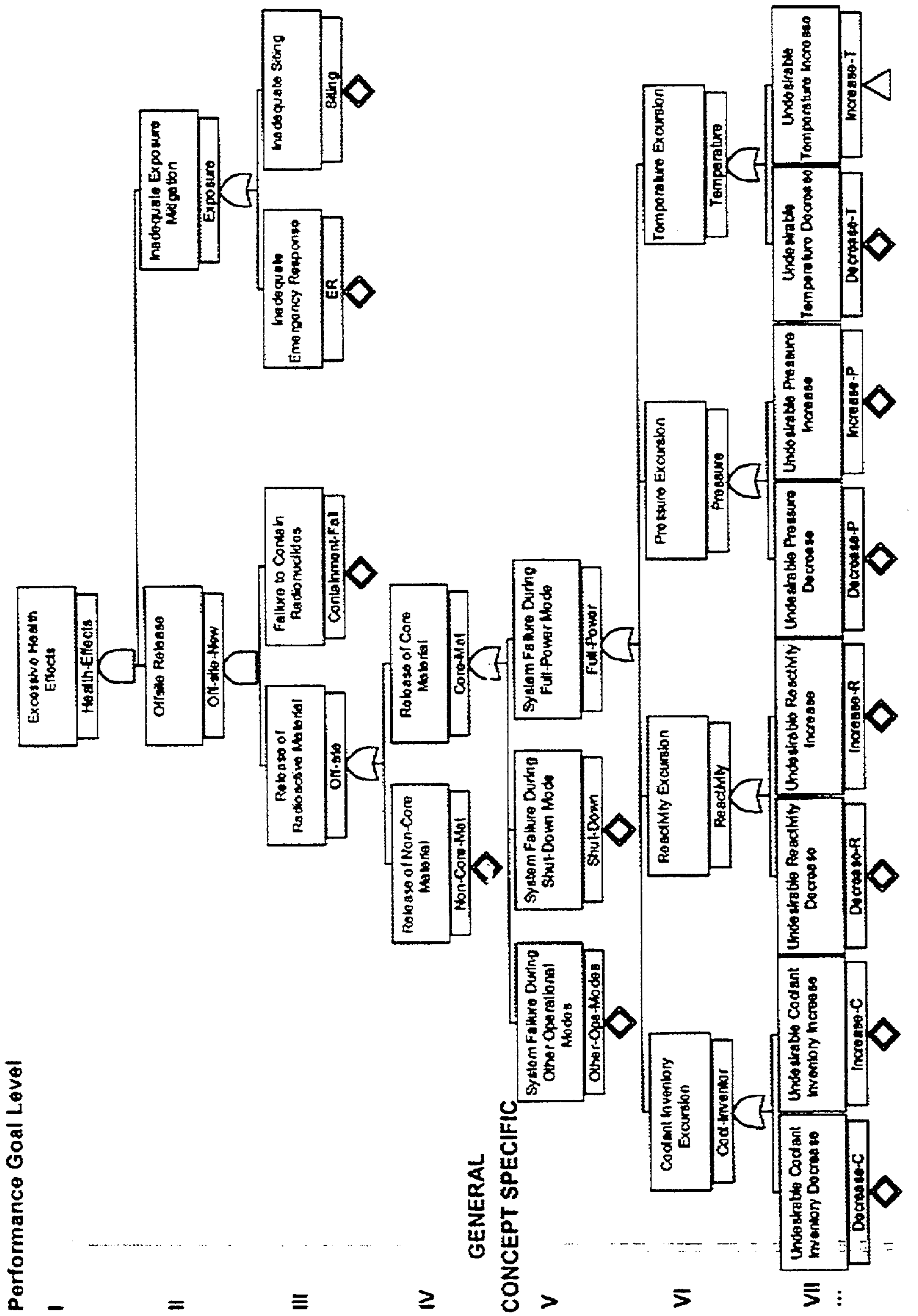


Figure 5. Illustrative Logic Diagram for Pebble Bed Modular Gas-Cooled Reactor

In the regulatory example used subsequently to illustrate the practicality of the ideas presented here an acceptable performance goal for all loss of coolant accidents (LOCAs) was formulated to be that

$$(0.75*\text{CDF-50}) + (0.25*\text{CDF-95}) < 7\text{ E-7 (per reactor year), where} \qquad (1)$$

CDF-50 is the median core damage frequency for all LOCAs, and
CDF-95 is the 95% confidence level value of the core damage frequency for all LOCAs.

This value and its formulation are used merely for purposes of illustration. A method for determination of the various performance goals must be developed. Doing this will likely be an iterative process exploring what is feasible balancing ideals and practicality.

Because new reactor regulation (i.e., licensing) must be able to address the performance vector of different reactor concepts and to accommodate their respectively differing levels of knowledge, the probabilistically-based treatment suggested here appears to be appropriate. For regulation of actual construction and operations it appears to be more feasible to utilize deterministic decision rules, based upon the plant's PRA, and revised as needed via use of the PRA.

DESIGN AND LICENSING NEGOTIATIONS

In any licensing regulatory process the plant's designer develops a design which he/she considers to be adequate for producing electricity safely. In areas where performance uncertainties are large or where potential accident consequences so large that risk aversion is justified, the designer would have obvious incentives to utilize defense in depth and performance margins in the design, and to reflect the effects of these tactics in the evaluated performance of the plant systems. When this design is submitted for regulatory approval, a negotiation follows which leads to any design changes required for regulatory approval. Currently, this negotiation is conducted focusing upon how adequately the design basis accidents are mitigated, with some background consideration being given to the important risk contributors and risk sensitivities of the plant. In our new design and regulatory concept, this negotiation would be conducted using the PRA as the primary discussion vehicle. The important questions would concern whether the relevant functional performance goals were satisfied with sufficient confidence.

Once the goals were specified, the remaining questions would concern the models and data used in evaluation of the un-availabilities (including uncertainties) associated with performance of these functions. Disagreements between the licensee and regulator would be focused upon the adequacy of models and data used in the PRA. A response to such a disagreement could include further defense in depth or design conservatism, but it could also include defense and improvements of the relevant models and databases.

An additional feature of this approach is that the burden upon the regulator to justify his challenges to the adequacy of the design would be made explicitly. Any design changes that the regulator thinks necessary would also be required to be reflected in the PRA, and the reasons for disagreement about the adequacy of the design would have to be formulated in terms of the adequacy of the PRA. Unavoidably, some of these disagreements would involve factors of subjective judgement. Such judgements would be required to be integrated into the results of the PRA, and their bases stated explicitly. This requirement would be an important departure from current practice where the regulator is not required to justify changes demanded of a license applicant.

For example, in the recent Design Certification licensing of the AP-600 PWR concept, the Certification was held up by the NRC until the designers agreed to add an active containment spray system which is redundant to the passive containment cooling system of the original design. Neither the PRA nor the deterministic design analysis of the plant indicated the need for the active system, but the regulator was able to require that it be added (presumably because of concern that the passive system might display unanticipated modes of behavior) without explicit justification (it was deemed to be the "prudent" thing to do).

As an illustration of how the new negotiation process would work, the designer before application submission would follow the process illustrated in Figure 6. In this process the designer would be guided by the PRA in identifying the set of marginally most valuable design changes to reduce functional unavailability values to being lower than those specified in regulations to be acceptable. The method of doing this would be to search for event sequences where design modifications would best reduce risks and/or their associated uncertainties. Then, once an adequate design is developed it would be submitted for licensing approval.

An illustration of this process is shown in Figure 7. In this illustration, a design thought to be adequate by the designer is rejected by the regulator who disagrees with data and models used to evaluate the risks of high pressure LOCA event sequences in the PRA. Rather than defend the models and data of that portion of the PRA the designer investigates further design changes as summarized in Table 1 and Figure 8. It is seen that addition of greater depressurization capability (used to transform the high pressure LOCA into a low pressure one, for which adequate mitigation systems exist in the design) is inadequate to meet the specified performance goal because of the remaining risk contributions of common cause failures in the emergency diesel generator and cooling water systems. Only when design changes to reduce the risks contributed by the common cause failures does the design become satisfactory to the regulator.

In this illustration, both the designer and regulator become focused upon ways to reduce risks and uncertainties, all of which are stated explicitly. Both parties have incentives to utilize good design practices, high quality components and

redundancy and conservatism in order to ensure that the specified performance goals will be satisfied.

From this examination it is not apparent that tools of current regulation such as design basis accidents and general design criteria are required. They may be retained in regulation for purposes of convenience, but their necessity is not apparent.

Rather, the needs of the new regulatory process are more concerned with ways of formulating a consistent set of performance goals and sub-goals, of ensuring that data bases and models will be of high and uniform quality, of formulating methods for the reproducible integration of subjective judgments into PRAs and for formulation of a risk-based Standard Review Plan for use by the regulatory staff. The tactics for creating some of these needed elements is not obvious as the problems involved are complex and subtle.

The best way of satisfying the new regulatory needs appears to be investigation of a set of example regulatory examples, where needed improvements in a general approach can be revealed via inadequacies in the application. Doing this is time consuming and expensive. Thus, the program for such investigations must be initiated well in advance of the time of anticipated license applications for new reactors and be sustained financially. These requirements imply the need for a program of risk-based regulatory development to be an essential component of any national effort to provide new nuclear power technology options.

The question facing energy technology planners is not that of whether to include a regulatory research component in future nuclear technology development efforts, but rather is one of how to make such an element sufficiently effective that it will permit the creation of the logically consistent and economically efficient licensing process required for the success of future generations of nuclear power technologies. The active participation of the NRC in this process is also essential for its success.

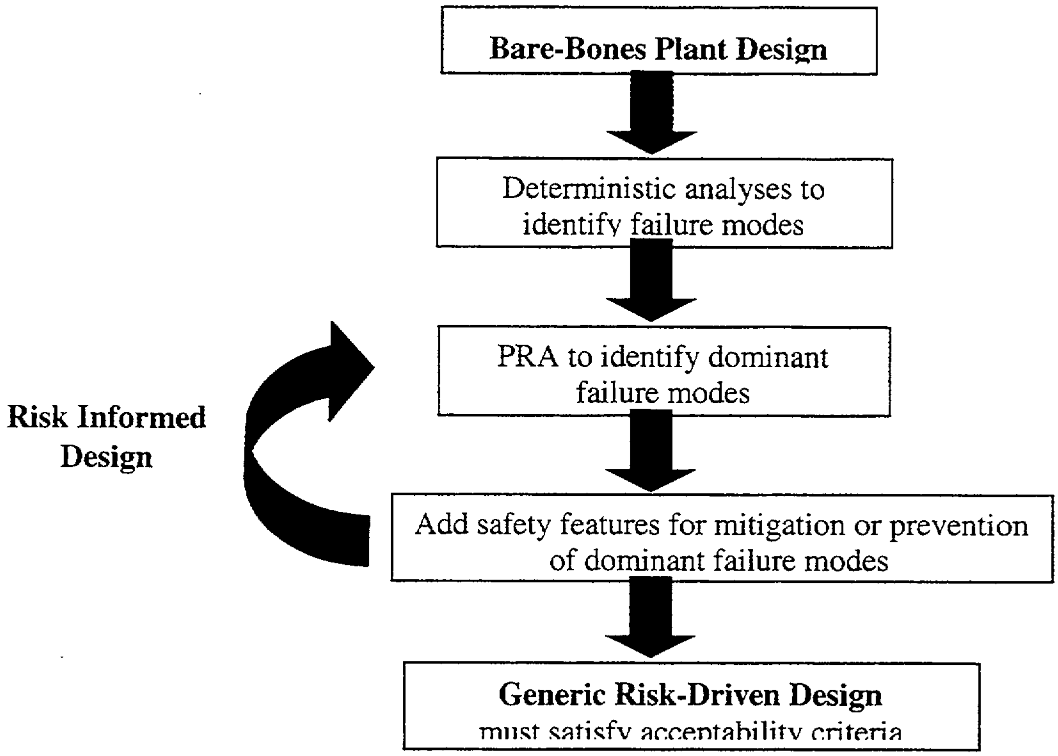


Figure 6. Schematic Diagram of the Risk-Driven Generic Design—Builds Upon A Bare-Bones Design, Using an Iterative Process

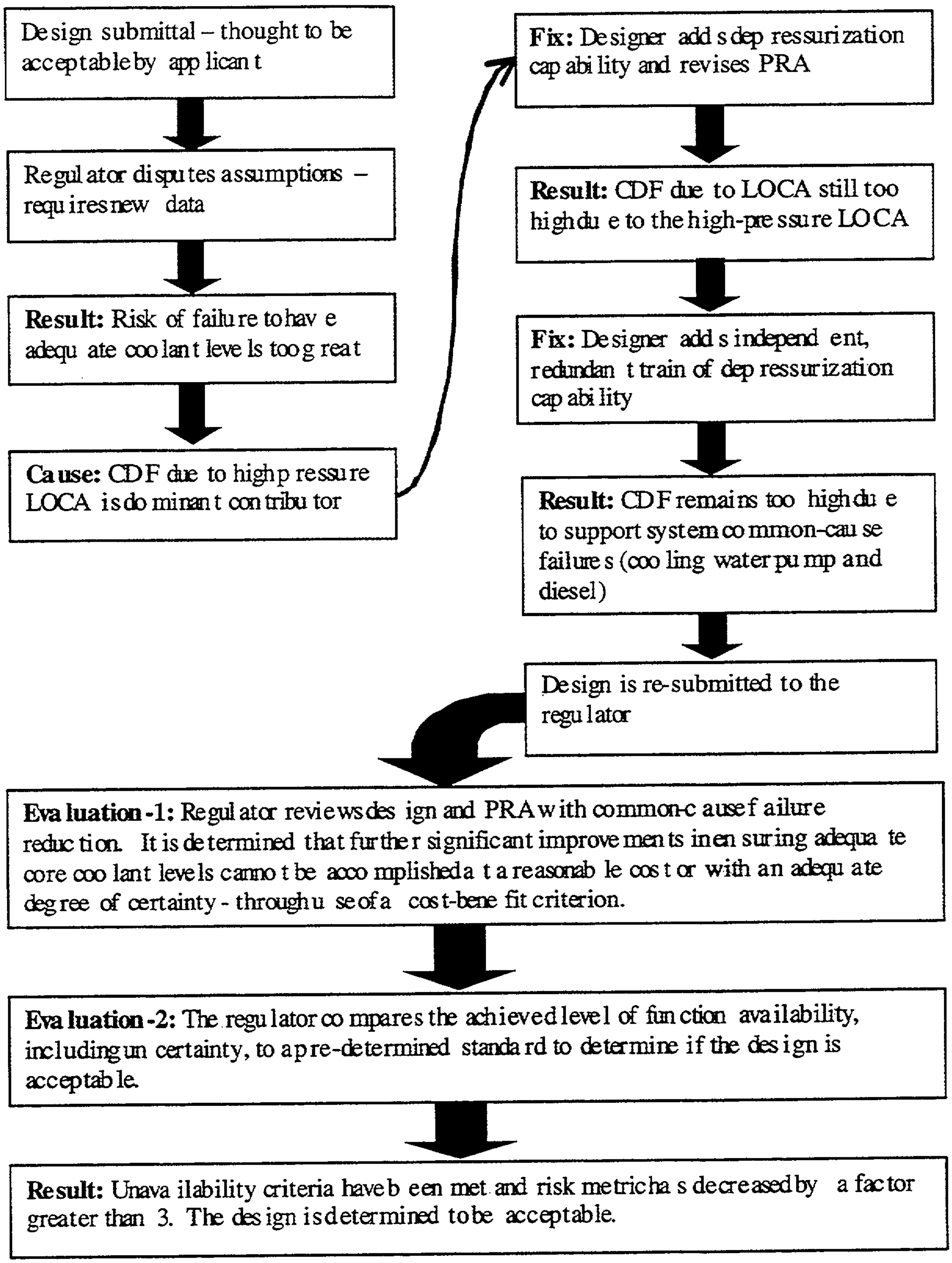


Figure 7. Example of Negotiation Between Applicant and Regulator

Table 1. Following the Effect of Design Modifications Upon Important Risk Metric Values

Plant Configuration	Median-CDF	5% Conf.	95% Conf.	Risk Metric*
No Depressurization	*1.528E-06*	*3.093E-07*	*4.278E-06*	*2.216E-06*
One Division of Depressurization	*7.086E-07*	*1.226E-07*	*1.890E-06*	*1.004E-06*
Two Divisions of Depressurization	*7.055E-07*	*1.445E-07*	*1.980E-06*	*1.024E-06*
Depressurization and reduced CW CC Failure**	*4.970E-07*	*1.008E-07*	*1.432E-06*	*7.308E-07*
Depressurization and reduced Diesel CC Failure	*6.120E-07*	*1.211E-07*	*1.718E-06*	*8.885E-07*
Depress with reduced CW and Diesel CC Failure	*4.020E-07*	*7.960E-08*	*1.290E-06*	***6.24E-07***

* Risk metric selected = (0.75 □ Median CDF) + (0.25 □ 95% confidence CDF)
** CW = Cooling Water; CC = Common Cause

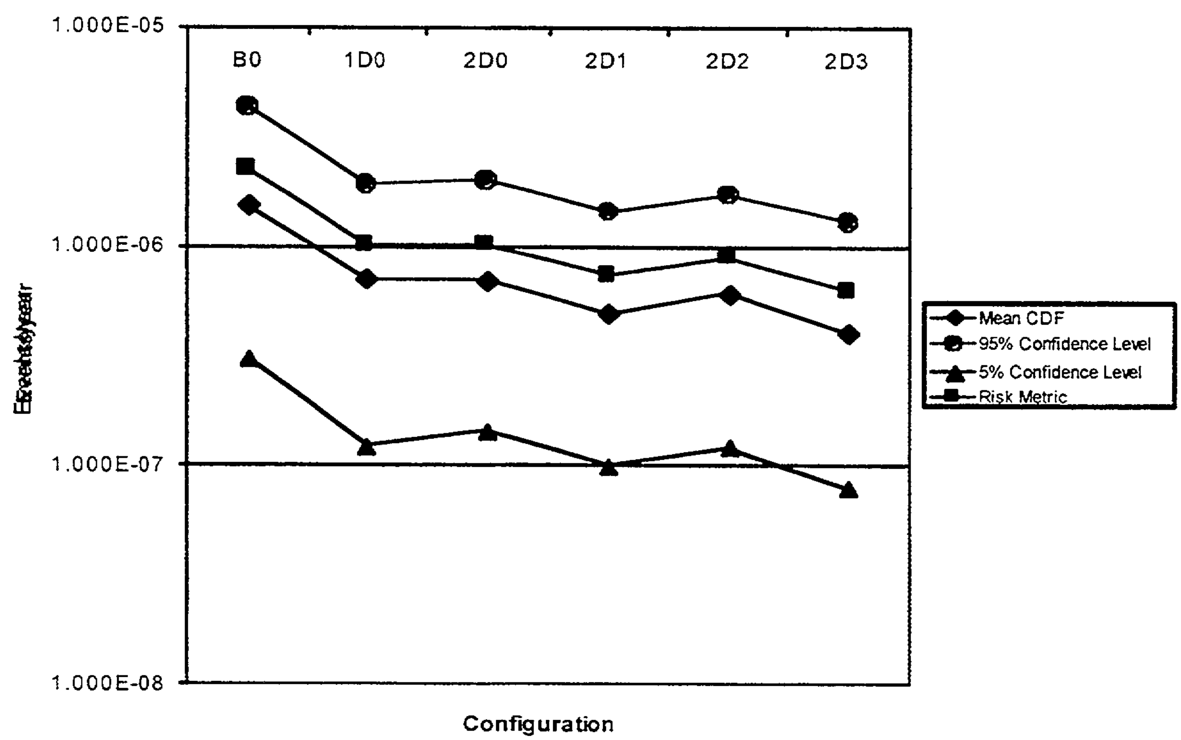


Figure 8. Effects of Design Modifications on CDF

ACRS Workshop on Regulatory Challenges for Future Nuclear Power Plants

NERI Project on Risk-Informed Regulation

June 5, 2001

Mr. George Davis - Westinghouse
Professor Michael Golay - MIT

Presentation Breakdown

- Mr. George Davis
  - Purpose and Overview
  - Expectations for the Future
- Professor Michael Golay
  - A New Risk-Informed Design and Regulatory Process
  - Example Problem

Westinghouse — INEEL — Massachusetts Institute of Technology
Duke Engineering & Services — Sandia National Laboratories — NC STATE UNIVERSITY
EGAN & ASSOCIATES, P.C. Counselors at Law

Purpose of Presentation

- Describe our project and its vision of a new design and regulatory process
  - provide a "work-in-progress" illustrative example
- Explain the need for continuing the development of a new design and regulatory process
  - keep pace with the development and licensing of new reactor design concepts.

Substantial Reductions in Capital Costs and Schedule Will be Needed for New Plants

- Production costs (Fuel plus O&M) for operating plants approaching 1 cent/KW-hr
  - not much room for further improvement
- Future investors likely to require payback of capital costs within 20 years of operation, or less
- Capital costs must be reduced by 35% or more relative to large ALWRs
  - overnight capital cost below $1,000/KWe
  - construction schedule of about 3 years (or less)

Three NERI Proposals Aimed at New Processes to Lower Plant Capital Costs

Program	Basic Objective
Risk-Informed Assessment of Regulatory and Design Requirements	Development of methods for a new design and regulatory process.
"Smart" Equipment and Systems to Improve Reliability and Safety in Future Nuclear Power Plants	Development of methods for demonstrating improved component and system reliability; including on-line health monitoring systems.
Development of Advanced Technologies for Design, Fabrication, and Construction of Future Nuclear Power Plants	Development of methods and procedures for collaborative, internet-based engineering, integrated design analyses, and improved construction schedules.

Comparison of NRC and NERI Risk-Informed Regulatory Processes

The new design and regulatory process must be developed further to support new plant license applications - including Generation IV design concepts.

Risk-Informed Assessment - Interactions With Other Programs

- NERI framework development activities are being coordinated with NEI
  - NEI will emphasize the development of regulations
  - The NERI project will address the overall risk-informed design and regulatory process
  - Westinghouse will be an NEI Task Force member
- It is anticipated that a new risk-informed design and regulatory process will be an input to new plant license applications, including Generation IV reactor concepts.

A New Risk-Informed Design and Regulatory Process

Massachusetts Institute of Technology

George Apostolakis, Michael Golay

Sandia National Laboratories

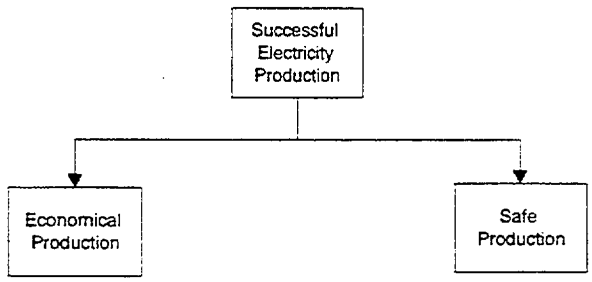
Allen Camp, Felicia Durán

Westinghouse Electric Company

David Finnicum, Stanley Ritterbusch

Overall Goal of Safety-Regulatory Reform

- Create methods to assure consistency of nuclear power plant applicant and regulator in performance/ goals for producing safe, economical power plants

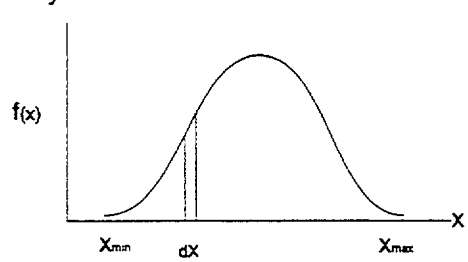


Major Elements:
- Acceptance Criteria
- Comprehensive, consistent assessment methods
- Designers, operators

Major Elements:
- Acceptance Criteria
- Comprehensive, consistent assessment methods
- Regulators, designers, operators

Risk-Informed Regulatory Approach - Fundamental Ideas

- Regulatory decisions are founded upon the informed beliefs of decision-makers.
- Any regulatory belief can and should be stated in a probabilistic format.

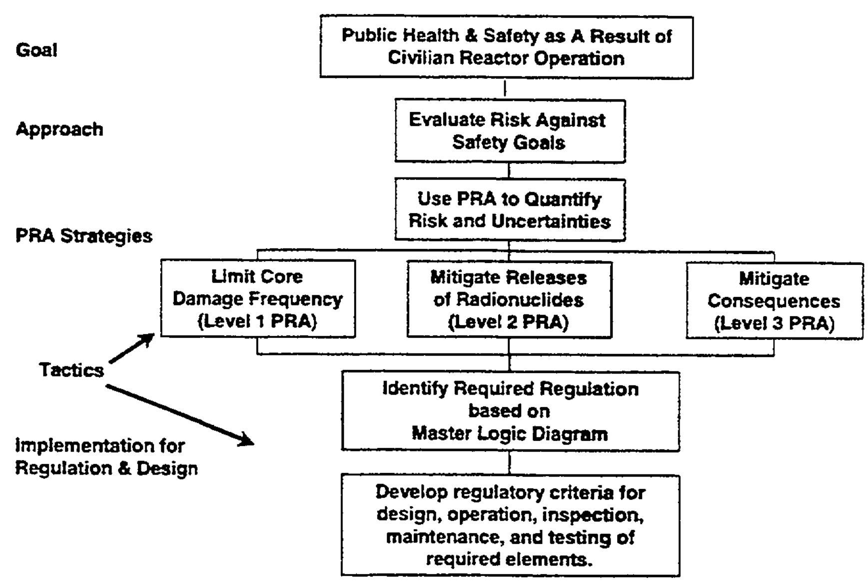


Probability $(x < X < x+dx) = f(x)dx$

- Regulatory acceptance criteria must reflect acceptable best-estimate performance expectations and uncertainties.

Risk-Informed Regulatory Approach - Fundamental Ideas....

- Regulatory questions and acceptance criteria should also be stated within a probabilistic framework.
- The probabilistic framework should be as comprehensive as possible:
  - utilize probabilistic and deterministic models and data where feasible - and use subjective treatments where not feasible,
  - state all subjective judgments probabilistically and incorporate into the PRA,
  - require both license applicant and regulatory staff to justify their decisions explicitly, and
  - initiate resolution process to resolve applicant-regulator disagreements.



Framework for Risk-Based Regulation and Design

Comparison of NRC and NERI Risk-Informed Regulatory Processes

Operating Plants (NRC/NEI)	Future Plants (NERI/New NEI Task Force)
Deterministic →	← Probabilistic
Traditional ("Structuralist") Approach	Risk-Based ("Rationalist") Approach

Operating Plants:

- Start with current designs and regulatory approvals.
- Justify risk-informed changes.
- Defense-in-depth remains as primary means of assuring safety.

Future Plants:

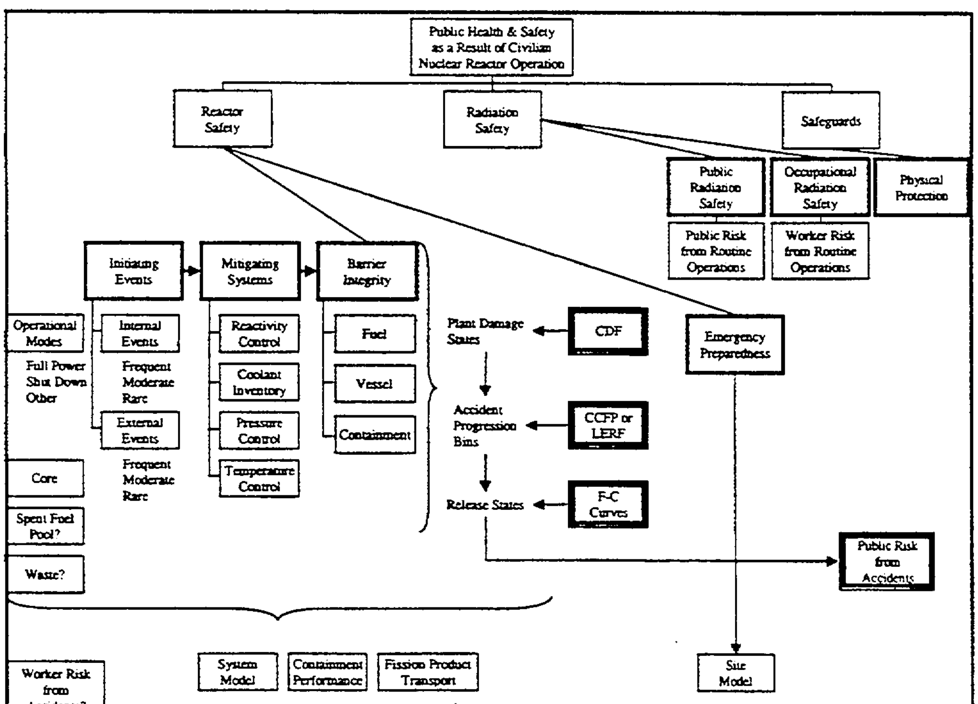
- Develop new design and regulatory process.
- Use firm probabilistic criteria to assure safety.
- Use defense-in-depth and safety margins as needed.

Risk-Informed Regulatory Approach....

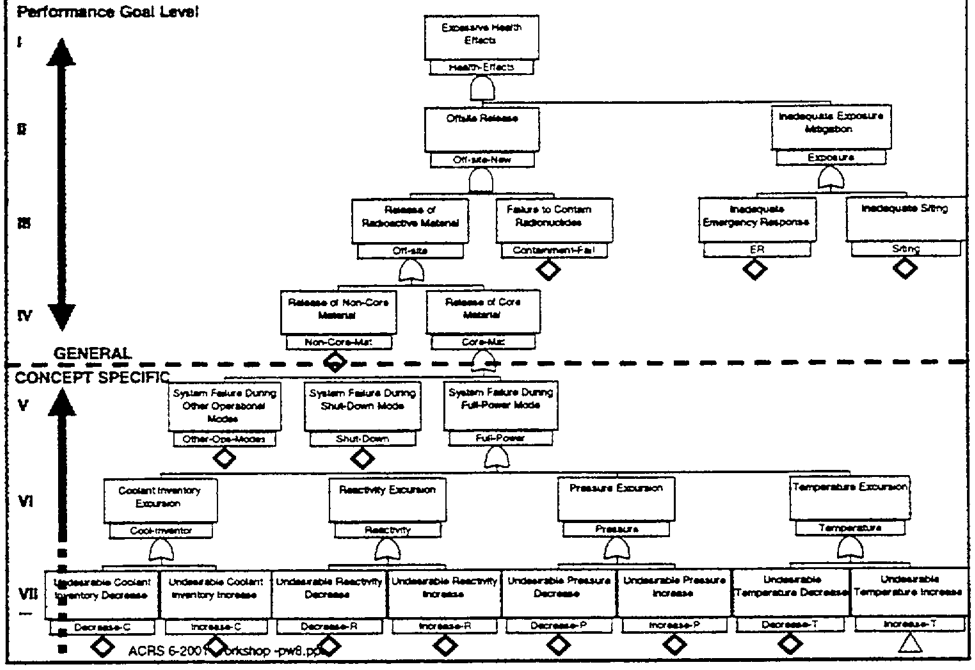
- At all conceptual stages of development, nuclear power plant evaluation is performed probabilistically and is supported by deterministic analyses, tests, experience, and judgements.
- Safety results of defense-in-depth, performance margins, best-estimate performance, and subjective judgements are all incorporated into a comprehensive PRA
  - PRA is used as a vehicle for stating evaluator beliefs concerning system performance
- The level of detail of acceptance criteria becomes finer as the level of concept development increases
  - many LWR-based regulatory constructs (e.g., DBAs, GDCs) are not applicable to less mature

Stages of Nuclear Power Plant Concept Development

Development Stage	Goals and Acceptance Criteria	Evaluation Tools	Relevant Evidence
Initial Concept	High level - qualitative	Qualitative, simple, deterministic	Experiences of other concepts, deterministic analyses
Initial detailed design	High level - quantitative	Quantitative – probabilistic, deterministic	Prior quantitative analyses
Final detailed design	Detailed – quantitative (design-specific subgoals)	Detailed – quantitative – probabilistic, deterministic	Prior quantitative analyses
N-th of a kind for a given plant type	Very detailed – quantitative (design specific criteria – DBAs, GDCs,....)	Very detailed – quantitative, probabilistic, deterministic, tests	Prior quantitative analyses, tests, field experience



Master Logic Diagram



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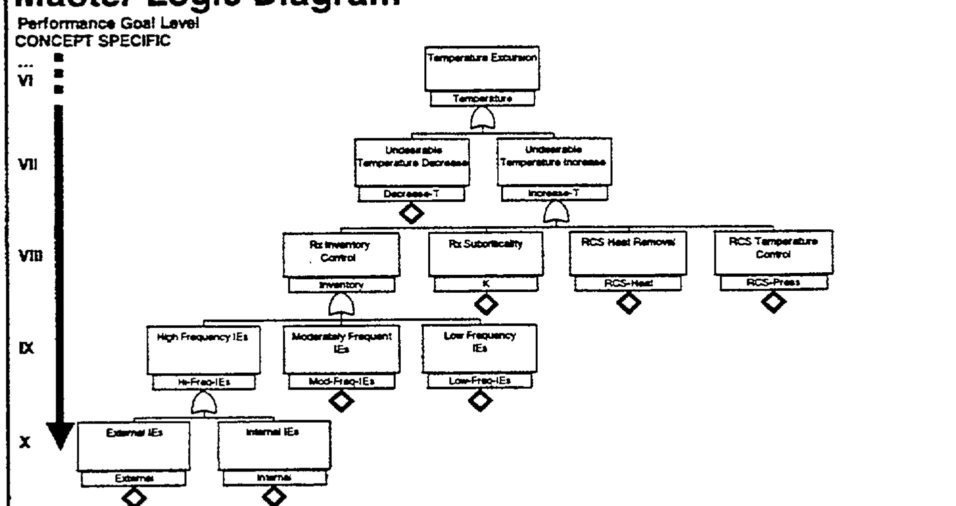
Master Logic Diagram



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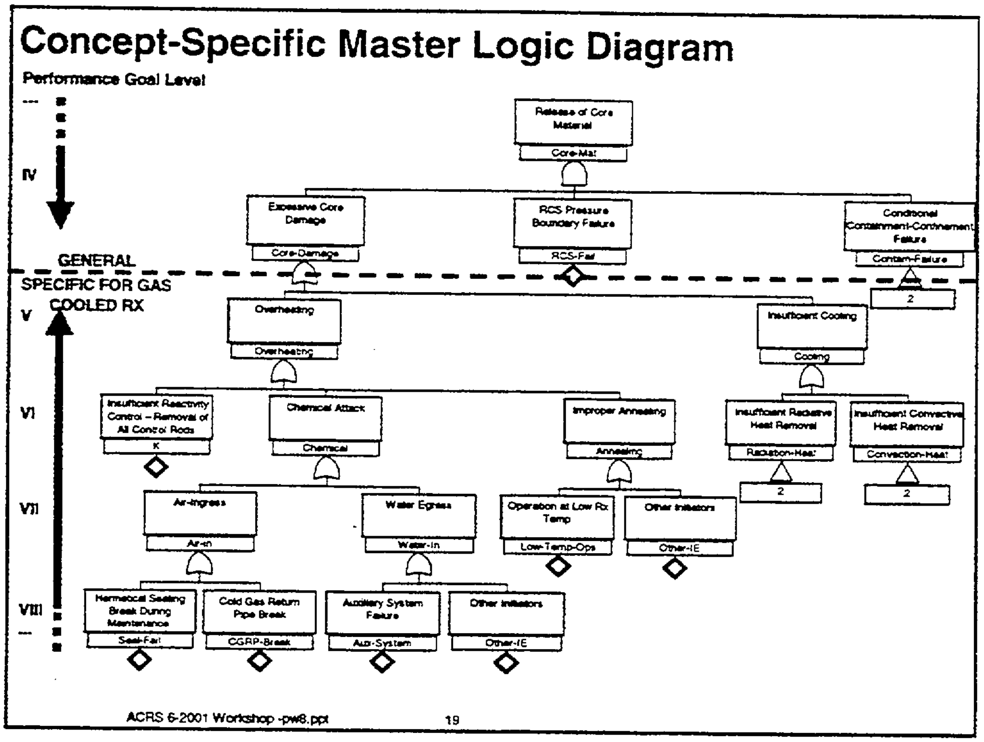
Concept-Specific Master Logic Diagram
Performance Goal Level
IV
Release of Core Material
Core-Mat
Excessive Core Damage
Core-Damage
RCS Pressure Boundary Failure
RCS-Fail
Conditional Containment-Confinement Failure
Contam-Failure
GENERAL
SPECIFIC FOR GAS COOLED RX
2
V
Overheating
Overheating
Insufficient Cooling
Cooling
VI
Insufficient Reactivity Control -- Removal of All Control Rods
K
Chemical Attack
Chemical
Improper Annealing
Annealing
Insufficient Radiative Heat Removal
Radiation-Heat
Insufficient Convective Heat Removal
Convection-Heat
2
2
VII
Air-Ingress
Air-In
Water Egress
Water-In
Operation at Low Rx Temp
Low-Temp-Ops
Other Initiators
Other-IE
VIII
Hermetical Sealing Break During Maintenance
Seal-Fail
Cold Gas Return Pipe Break
CGRP-Break
Auxiliary System Failure
Aux-System
Other Initiators
Other-IE
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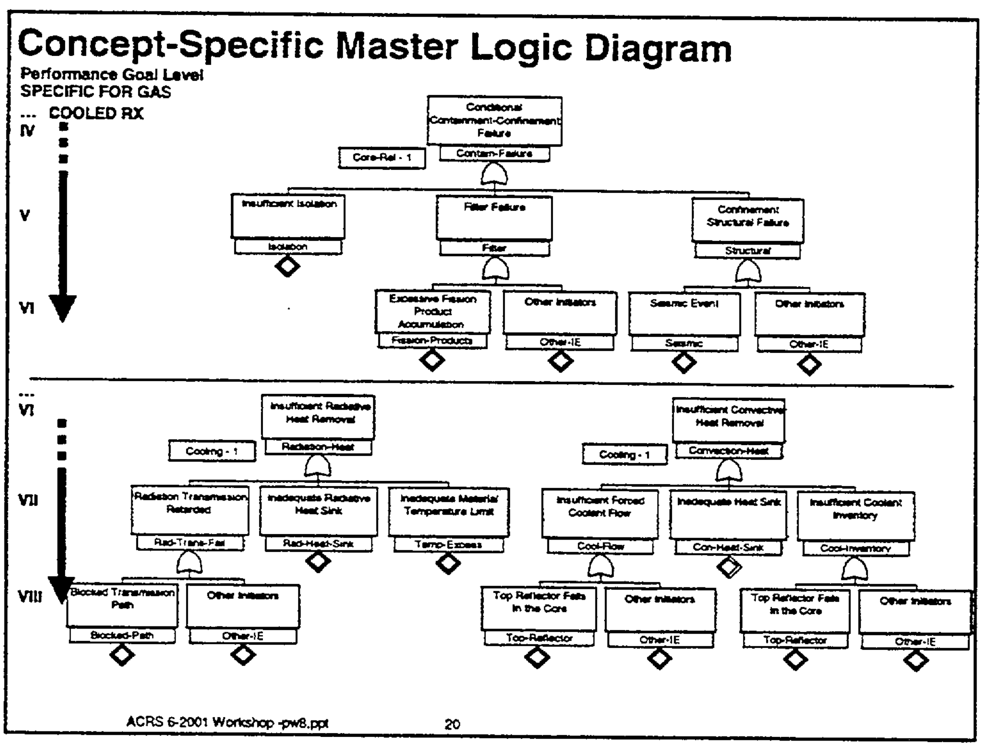
Concept-Specific Master Logic Diagram
Performance Goal Level
SPECIFIC FOR GAS COOLED RX
IV
Conditional Containment-Confinement Failure
Core-Rel - 1
Contam-Failure
V
Insufficient Isolation
Isolation
Filter Failure
Filter
Confinement Structural Failure
Structural
VI
Excessive Fission Product Accumulation
Fission-Products
Other Initiators
Other-IE
Seismic Event
Seismic
Other Initiators
Other-IE
VI
Insufficient Radiative Heat Removal
Cooling - 1
Radiation-Heat
Insufficient Convective Heat Removal
Cooling - 1
Convection-Heat
VII
Radiation Transmission Retarded
Rad-Trans-Fail
Inadequate Radiative Heat Sink
Rad-Heat-Sink
Inadequate Material Temperature Limit
Temp-Exceed
Insufficient Forced Coolant Flow
Cool-Flow
Inadequate Heat Sink
Con-Heat-Sink
Insufficient Coolant Inventory
Cool-Inventory
VIII
Blocked Transmission Path
Blocked-Path
Other Initiators
Other-IE
Top Reflector Fails In the Core
Top-Reflector
Other Initiators
Other-IE
Top Reflector Fails In the Core
Top-Reflector
Other Initiators
Other-IE
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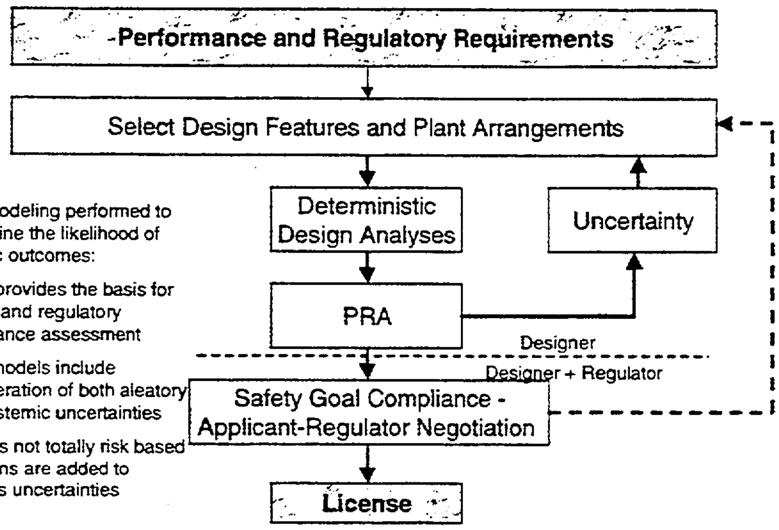
Fundamental Interactions Between License Applicant (or Licensee) and Regulator

- Should be formulated with probabilistic methods
- Acceptability negotiation for new license application or license revision
  - currently is deterministic
  - should be risk-based; completion of procedures, tools, and termination criteria is needed
- Plant construction oversight
  - can be deterministic, subject to risk-based oversight
- Plant operation oversight
  - can be deterministic, subject to risk-based oversight

Basic Design and Regulatory Process - Employed Traditionally, Remains Valid Today

- Designer develops a plant design that both produces power reliably and operates safely
  - responsible for plant safety, using high level regulatory criteria and policies as inputs
- Regulator reviews the design
- Designer and regulator engage in a dialog
  - specific safety features, their performance criteria, and methods of design and analysis
- Documentation is developed throughout the process
  - designer documents the design basis
  - regulator documents the safety evaluation, policies established, and criteria for future reviews (e.g., Reg. Guides and Standard Review Plans, and possibly regulations)

Risk-Informed Design and Regulatory Process - PRA Decision Making



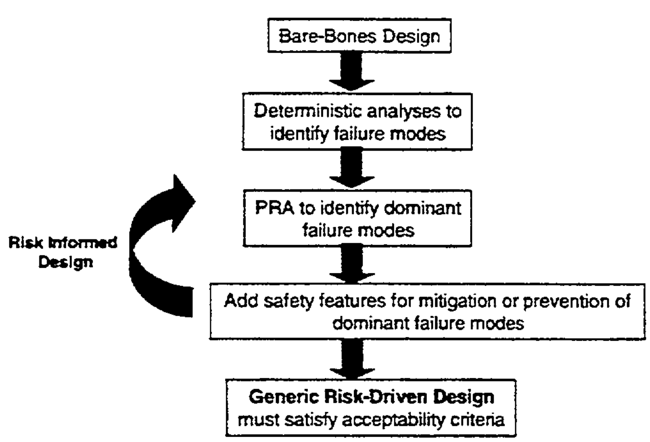
PSA Modeling performed to determine the likelihood of specific outcomes:

- PSA provides the basis for design and regulatory compliance assessment

-PSA models include consideration of both aleatory and systemic uncertainties

- PSA is not totally risk based - margins are added to address uncertainties

Schematic Diagram of the Risk-Driven Generic Design - Builds Upon A Bare-Bones Design, Using an Iterative Process



Classification of Event Sequences Within the Risk-Informed DBA Approach

Initial Sequences
Very Small Leak
Safety Relief Valve Stuck Open
Small Pipe Break LOCA
Pilot Operated Relief Valve Stuck Open
RC Pump Seal Failure
Medium Pipe Break LOCA
Large Pipe Break LOCA

Shared Functional Challenges
Insufficient RCS Inventory Control
Insufficient RCS Pressure Control
Insufficient RCS/Core Heat Removal

Classes	Response Required
Very Small Leak SRV Stuck Open	Normal Coolant Make-Up
Small Pipe Break LOCA PORV Stuck Open RC Pump Seal Failure	Emergency High Pressure Coolant Injection
Medium Pipe Break LOCA Large Pipe Break LOCA	Depressurization and Emergency Low Pressure Coolant Injection

Apportionment of a Performance Goal Into Subgoals

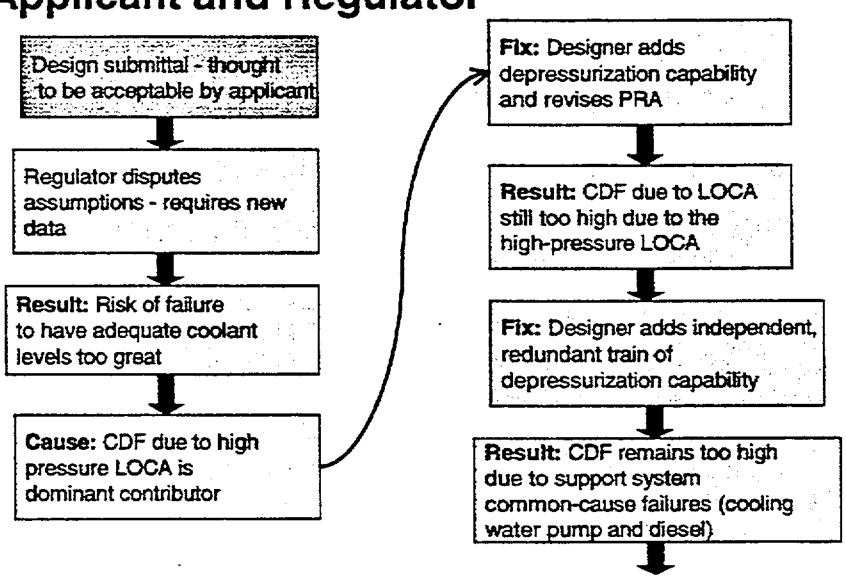
- Designer proposes apportionment - then negotiates with regulator
- Apportionment must reflect what is feasible in the design
- Example shows that the reliability/availability of mitigation systems reflects feasibility of the design

Initiating Event	Initiating Event Frequency	Mitigation Unavailability	Core Damage Frequency
Very Small LOCA	4E-3 /yr	1E-4	4E-7/yr
Small LOCA	2E-4 /yr	1E-3	2E-7/yr
Large LOCA	4E-5 /yr	1E-2	4E-7/yr
Example Acceptability Criterion: Achieved Total CDF due to LOCAs must be less than or equal to **2E-6 /yr**			Achieved Total CDF due to LOCAs: **1E-6 /yr**

Example of Designer's Initial Risk-Informed Submittal to the Regulator

- Two safety system divisions - each contains:
  - two active high-pressure injection trains
  - one active low-pressure injection train
  - cooling water (component cooling, service water, HVAC)
  - two diesel generators
  - DC (battery) power
- Shared support systems
  - chemical volume control system
  - off-site power
- PRA Includes:
  - deterministic analyses, data, models,
  - uncertainties, inter-dependencies, and common-cause failures
  - initiator data are from documented sources (NUREG/CR-5750)
  - component failure frequencies are estimated from existing PRA studies (for this LWR example problem)

Example of Negotiation Between Applicant and Regulator



Example of Negotiation Between Applicant and Regulator....

Design is re-submitted to the regulator

Evaluation-1: Regulator reviews design and PRA with common-cause failure reduction. It is determined that further significant improvements in ensuring adequate core coolant levels cannot be accomplished at a reasonable cost or with an adequate degree of certainty - through use of a cost-benefit criterion.

Evaluation-2: The regulator compares the achieved level of function availability, including uncertainty, to a pre-determined standard to determine if the design is acceptable.

Result: Unavailability criteria have been met and risk metric has decreased by a factor greater than 3. The design is determined to be acceptable.

Following the Effects of Design Modifications Upon Important Risk Metric Values

Plant Configuration	Median-CDF	5% Conf.	95% Conf.	Risk Metric*
No Depressurization	*1.528E-06*	*3.093E-07*	*4.278E-06*	*2.216E-06*
One Division of Depressurization	*7.086E-07*	*1.226E-07*	*1.890E-06*	*1.004E-06*
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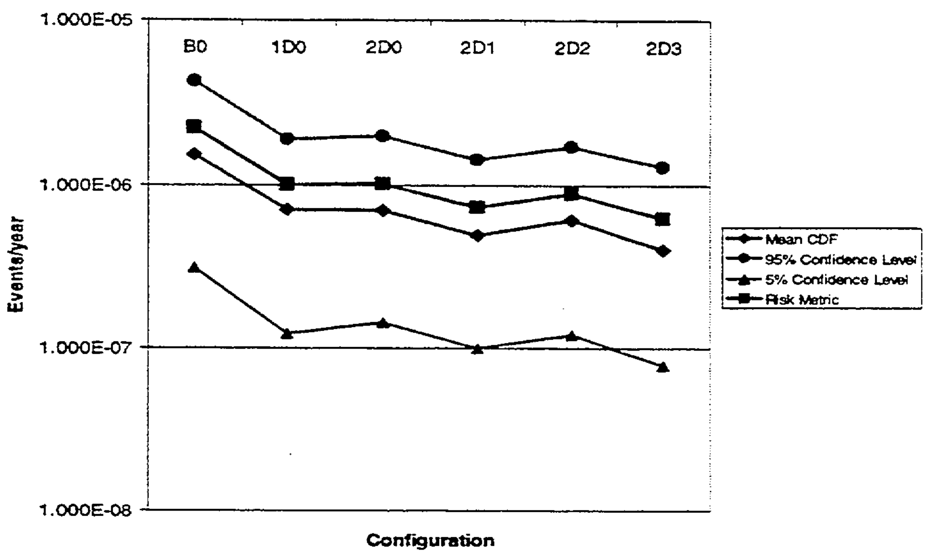
* Risk metric selected = (0.75 * Median CDF) + (0.25 * 95% confidence CDF)

** CW = Cooling Water; CC = Common Cause

Effects of Design Modifications on CDF



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Example Problem - Results & Questions

- Concerns about common cause failures and large uncertainties would lead designers and regulators to conservative design approaches
  - defense-in-depth, safety margins
- Guidelines are needed for consistently reflecting model weaknesses in the probabilistic database
- Consistent acceptance criteria are needed for negotiation guidance and termination
- Practical implementation requires more work
  - more trial examples
  - standardized models, methods, databases
  - methods for treatment of subjective judgements
  - replacements for:
    - GDCs
    - DBAs (risk-dominant event sequences)
    - Standard Review Plan

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Summary

- The favored approach for a new design and regulatory process would:
  - use risk-based methods to the extent possible
  - use defense-in-depth when necessary to address model and data uncertainty.
- A new risk-informed design and regulatory process would:
  - provide a rational method for both design activities and applicant-regulator negotiations
  - provide a method for an integrated assessment of uncertainties in design and regulation
  - provide a process that is applicable to non-LWR technologies
- Development of a new design and regulatory process should be continued to support new reactor license applications.

T. Kress, Future Reactors Subcommittee Chairman: How would you deal with the issue that the PRAs are traditionally very incomplete? They don't deal with shutdown conditions very well. They don't include fires very well, and seismic even is often not treated very well -- would you incorporate those kinds of missing ingredients into the uncertainty of distribution?

M. Golay: Yes, basically the way you would incorporate them is through a statement of the subjective judgment of those who have to assess what practice is to be used.

D. Powers,ACRS Member: You're going to expand the capability of PRA to carry this out. One of the areas you're going to expand it to carry it out is in the shutdown risk. I presume that you have a plant here that you say is going to have some history, and during that history it's going to have various kinds of shutdowns, those that it planned, to do a variety of activities that are going to be quite different, and it's going to have an occasional unscheduled shutdown. And you can prognosticate all of those things, all of the different configurations of the plant that go on during a shutdown, a scheduled shutdown for refueling and what not. But now we don't try to quantify, those times and configurations, and yet you want us to do that. How is this possible?

M. Golay, MIT: I would say that your task in those areas has not changed from that people have today; that when you consider a license application, you try to consider the spectrum of conditions under which the plant will be operated, and using evidence appropriate for each condition, judge whether it will be operated successfully.

The development of shutdown risk analysis provides an illustration of how you do that in, say, a non-power state, and when you're comparing operations between those states, you, as T. Kress just brought out, you inevitably come to situations where the available objective evidence is not sufficient for you to determine which practice is better. Do you do perform maintenance while you're shut down or do you do it on line, for example? Again, subjective judgment has to come into the process. What I'm submitting is that we use that subjective judgment today. We simply don't spell out loud the factors the way that we're weighing the factors. What's changed with the approach that we're suggesting is that we state everything in probabilistic terms and incorporate it into the PRA.

T. Kress, Future Reactors Subcommittee Chairman: What I'm interested in is the risk associated over the full lifetime of the plant. That means shutdown number e.g; 85 is going to take place "n" years from now and I need to incorporate into my risk assessment. Now, since I don't know what that shutdown consists of, what planned maintenance they're going to have because it hasn't even come about yet, it may even be an unplanned shutdown. How do I know how to incorporate the short time during shutdown, short compared to other things? That risk, how do I put that risk component into my risk assessment when I don't even know what it is. We're dealing with a change, a variable configuration in time rather than a fixed configuration, which is what PRAs usually deal with. How do I deal with that in a PRA? Is that something that needs a new PRA methodology?

M. Golay, MIT: I would submit not, but let me go to why. The first question that may arise is why do you need research on regulatory reform. Why can't you just get a few people to go off and think in the corner for a time and come up with some proposals and then try them out?

My experience has been that you don't know what is a good idea until you've gone through some feasibility attempts. That there's an iterative process at work here, and that's the heart of

regulatory reform research, to find out what's feasible and then from that find a good blend of feasible approaches consistent with an over arching logical framework. In terms of the question you've asked, I would suspect, without having tried to do the analysis, that, first of all, the level of detail that you indicate as being required is probably not necessary; that approaching it from the point of view of looking at safety during shutdown and trying to anticipate a range of conditions that you think are reasonably plausible, which is the approach we have today, will likely work. What I would try and do is turn the question around and try and use a real probabilistic treatment of the safety, but not to try and anticipate the fine detail the history of a plant that might occur or might not occur.

G. Wallis, ACRS Member: Do you have a good measure of safety margin in a probabilistic sense?

M. Golay, MIT: Yes. If you're using margin on let us say concerning the approach to melting temperature or something of that kind, what that would translate into would be to formulate your acceptance criterion from the design point of view at a very, very high confidence level so that you ensure satisfaction.

G. Wallis, ACRS Member: But once you start saying there's a failure point, you are making things deterministic, which really are not.

M. Golay, MIT: Well, I'm trying to relate it to the current design process.

G. Wallis, ACRS Member: That's right, but I think it would be interesting to see what you could do with a definition of margin which got away from these ideas of having a point or --

M. Golay, MIT: Right, and what you would do, as you're hinting, is really to use a distribution on all of the performance limits, and that would be a natural evolution that I think we would go to and probably quicker than I'm anticipating.

G. Wallis, ACRS Member: You would look at the probabilities of all of those and the consequences of all of those.

M. Golay, MIT: Right. That's right. So what you expect is that if people are using the approach we're suggesting well, they would have natural incentives to put defense-in-depth into their designs partly because they could see a benefit for doing it when they make a regulatory submittal. The same thing would be the case with incorporating performance margin.

T. Kress, Advanced Reactor Subcommittee Chairman: How do I decide what confidence level constitutes an acceptable margin?

M. Golay, MIT: My short answer is you have to work on it. It's partly a social policy and has to be worked out in an iterative manner.

G. Wallis, ACRS Member: It's an interesting idea, but it seems to me that as you learn more about a plant, you might actually get less detail than any kind of plan. You might really know what you have to worry about and you don't need all of this detail.

M. Golay, MIT: Conceivably, and we've seen that, for example. The evolution of the passively based water-cooled reactors could be an illustration of that. But one reason for putting this

figure together is to address this question of where do the design basis accidents and general design criteria come into the picture. I would say that it's a tentative conclusion, not a firm one, that those really play a role when you get to the detailed design and later stages of evolution because when you try to formulate design basis accidents, you have to have a design. You have to have a concept in terms of which to think about and have some seasoning in terms of your understanding of its weaknesses, things of that kind. if you look at what we've done with light water reactors, we've gone through that process.

G. Wallis, ACRS Member: Let's try to think about this. The method of design and analysis is going to be in probabilistic terms. You mean that every time you put a correlation in a code, you have to do something probabilistic with it?

M. Golay, MIT: Only if it propagated through into your risk evaluation.

G. Wallis, ACRS Member: It probably does.

M. Golay, MIT: Yes. For example, if your new correlation had a different uncertainty treatment, you would expect that to be propagated through. That's right.

G. Wallis, ACRS Member: Why do you need subgoals? It seems to me that if you had a plant that had no LOCA probability at all because of its design, then you might trade this off and be allowed to have more probability somewhere else if all you care about is the total.

M. Golay, MIT: But you care about the uncertainty associated with the total as well.

G. Wallis, ACRS Member: Yes, you do, but the total, the bottom line is the thing, not really how it breaks up in all these pieces.

M. Golay, MIT: Well, I would say that another reason why you want to do this is that in the long run for regulatory convenience and efficiency, you probably want to formulate risk-based deterministic decision rules as you reach a high stage of maturity. So there will be sort of natural incentives to formulate subgoals as the concept matures. And that's the reason we have this in here, simply to illustrate that you have to go through this iterative process.

L.E. Hochreiter, Penn State University: You talk about using best estimate performance, expectations and uncertainties. And you really have two kinds of uncertainties. You can have the plant uncertainties, but you can have the uncertainties in the model that you use to do the predictions, and with a light water reactor, we've got 40 years of a database, experimental database so that we can quantify the models and the model uncertainty so that we have a good handle on that. I don't know how you address that for a new design like we've been talking about for these Gen. IV designs where you really don't have much of a database at all.

Mike Golay, MIT: Yes, with any concept, regardless of its level of maturity, I'll submit that as you try to do a risk analysis of comparing alternatives, you ultimately end up at a point where the available objective data reach their limits. You can find this with plenty of light water examples as well, that what you're really into is a situation where you -- I think always -- that's too strong a word because I don't have the basis for saying "always," but my experience has been so -- that you end up with a combination of objectively based evidence and you have to supplement that by your judgment. So the only suggestion that we're making is that you should state that in probabilistic terms and incorporate it into the PRA so that with the new concept,

you reach that limit much sooner than with the mature one, but that the general structure holds up for both.

Larry Parme, General Atomics: You mentioned possibly replacing the DBAs with the risk dominant events, and overall I'm supportive of your approach, but in the licensing risk based approach that we did for the MHTGR, one of the things we were looking at that sort of approach, and we immediately ran into the problem that when you go and say that the risk dominant events replace DBAs, you find that certain non-risk dominant events are the only challenges, if you will, to certain key equipment or safety functions, and the risk dominant events may not demonstrate to the regulator the various ways that your safety functions are done. And I hope you follow what I'm saying. My question to you is: did you think about this?

We had thought about this in the '80s, found that risk dominant events weren't a true substitute for DBAs and had to also use the PRA, but had to find from our event trees events that challenged each of the safety functions regardless of their risk dominance.

Mike Golay, MIT: Right. Let me try and translate it though. What I think you're really saying is that there's a concern about the level of uncertainty associated with your risk based analysis, such that if you went in and claimed that you were doing very, very well, it wouldn't be a credible claim, and that it was necessary to, in effect, show that you could handle something tougher, is in some way a defense- in-depth kind of capability.

Summary
Advanced High-Temperature Reactor for Hydrogen and Electricity Production

Dr. Charles W. Forsberg
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Historically, the production of electricity has been assumed to be the primary application of nuclear energy. That may change. The production of hydrogen (H_2) may become a significant application. The technology to produce H_2 using nuclear energy imposes different requirements on the reactor, which, in turn, may require development of new types of reactors. This alternative application of nuclear energy may necessitate changes in the regulatory structure.

Alternative Applications of Nuclear Energy—H_2 Production

World consumption of H_2 for the production of chemicals (e.g., CH_3OH and NH_3) and the refining of crude oil into transport fuels is growing rapidly. Hydrogen is added to heavy crude oils to (1) produce lighter fuels such as gasoline and (2) remove impurities such as sulfur. As resources of high-quality light crude oils are exhausted, more H_2 is required to produce an equivalent amount of gasoline per barrel of lower-grade crude oil. Because much of the H_2 is produced from lower-value refinery streams, an economical outside source of H_2 would allow the conversion of these hydrocarbons into gasoline rather than require their use for H_2 production. As a result, the output of liquid fuel per barrel of crude oil could significantly increase, thereby reducing crude oil imports. Nonfossil H_2 would also substantially decrease the quantity of natural gas that is used to produce H_2, thus reducing carbon dioxide emissions.

Currently it is estimated that 5% of natural gas is used to manufacture H_2 for chemical and refinery use. Hydrogen consumption is increasing rapidly, and some projections indicate that by 2010 the energy value of the hydrocarbons used to manufacture H_2 will exceed the energy output of all nuclear reactors in the United States. Hydrogen has also been proposed as a future transport and distributed-power fuel. These advanced applications would increase the H_2 demand by one to two orders of magnitude. The development of economic nonfossil H_2 would also protect the domestic chemical and refinery from high natural gas prices that could increase H_2 costs sufficiently to cause parts of these industries to move offshore for lower cost sources of natural gas.

Hydrogen and electricity represent the only large potential markets for nuclear energy. Therefore, if the uses of nuclear power are to expand, reactors must be designed to efficiently produce H_2. Many direct thermochemical methods are possible for producing H_2 with the input of heat and water. High temperatures (800 to 1000°C) are required to ensure rapid chemical kinetics (small plant size with low capital costs) and high conversion efficiencies (~50% thermal energy converted to H_2). A low-pressure reactor coolant is desired to couple to the low-pressure chemical plant. The development of such a reactor would also make possible better methods of electricity production: indirect Brayton cycles and direct thermal-to-electric conversion techniques. Efficient technologies for the latter process do not exist at present.

Advanced High-Temperature Reactor (AHTR)

If nuclear energy is to be used for production of H_2 or similar applications, reactors that can meet the unique high-temperature requirements (800 to 1000°C) are required. One such

reactor—the AHTR—is described herein. The high-temperature operations also create the potential for very-high efficiency methods for the production of electricity.

The AHTR would generate up to 600 MW(t) with an outlet temperatures of >1000°C. The reactor core contains a graphite-matrix fuel and core that has the same general characteristics as that developed for modular high-temperature gas cooled reactors (MHTGRs). Such fuels have been demonstrated at temperatures up to 1200°C. The AHTR fuel cycle would be similar to that for the MHTGR. The liquid coolant would be a molten fluoride salt ($2LiF\text{-}BeF_2$) developed for molten-salt-fueled fission reactors and proposed as a coolant for fusion reactors. The coolant would transfer heat from the coated-particle graphite fuel to the H_2 chemical plant. This particular salt has a boiling point of ~1400°C. Several other candidate salts exist such as FLiNaK (a eutectic mixture of LiF, KF, and NaF). Fluoride salts are fully compatible with graphite (the aluminum industry has electrolyzed aluminum fluoride salts in graphite furnaces for over a century to produce aluminum metal).

The combination of the graphite fuel form and the molten salt coolant makes possible the very high temperatures. The low-pressure coolant reduces the need for high-temperature, *high-strength* materials in the external heat exchangers, compared with those required in reactors that use high-pressure helium or other high-pressure fluids to transfer heat. The maximum salt outlet temperature can be significantly higher than that for a gas-cooled reactor with the same graphite fuel and same peak fuel-temperature limits. This is a consequence of the heat-transfer properties of molten salt (similar to water) compared to helium. The improved heat transfer lowers temperature drops between (1) fuel and coolant and (2) coolant and the H_2 plant.

The AHTR reactor has some safety systems in common with other reactors, as well as some unique features. Reactor power is limited by the high-temperature Doppler effect within the fuel. Because the molten salt expands upon heating, an additional negative moderator temperature coefficient is associated with coolant expansion. The reactor physics are similar to those of the MHTGR. In an accident, the decay heat would be conducted directly from the reactor core, through the reactor vessel, and then to the environment. This is similar to the emergency decay-heat-removal system in an MHTGR.

The liquid coolant lowers the potential for radionuclide release by several mechanisms: (1) atmospheric pressure eliminates a primary driving force for radionuclide releases, reduces the forces that can destroy the containment or confinement system, and simplifies isolation of the reactor from the environment, (2) the difference (at least 400°C) between the operating temperature and boiling point of the salt provides a large margin before boiling occurs, (3) the physical properties of the coolant allow natural circulation of the coolant to provide decay-heat cooling, and (4) most fission products and actinides dissolve into the coolant. Significant work is required before the full safety implications of this type of reactor are understood and before such a reactor could be built.

Regulatory Implications

The production of alternative products using nuclear energy encompasses different safety considerations involving both the reactor and the energy conversion facility. The impacts of the reactor on the chemical plant and the impacts of the chemical plant on the reactor must both be considered. It implies ownership—and possibly operation—by non-utility corporations. The different products (H_2) may require reactors with non-traditional coolants such as molten salts.

Advanced High-Temperature Reactor for Hydrogen and Electricity Production
(Joint ORNL–Sandia Activity)

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ACRS Workshop: Regulatory Challenges For Future Nuclear Power Plants
Advisory Committee on Reactor Safety
U.S. Nuclear Regulatory Commission
Washington D. C.
June 5, 2001

Outline

- Is a nuclear-based hydrogen economy in our future?
- The Advanced High-Temperature Reactor (AHTR)
  - An option for hydrogen production
  - An option for electric production
- Regulatory implications

Is a Hydrogen Economy in our Future?

(It may already be here)

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Rapid Growth Is Expected in Industrial Hydrogen (H_2) Demand

- Rapidly growing H_2 demand
  - Production uses 5% of U.S. natural gas plus refinery by-products
  - If projected rapid growth in H_2 consumption continues, the energy value of fuel used to produce H_2 will exceed the energy output of all nuclear power plants after 2010
- The chemical industry (NH_3 & CH_3OH) is a large consumer
- Changing refinery conditions are driving up the H_2 demand
  - More heavy crude oils (limited supplies of high-quality crude)
  - Demand for clean fuels (low sulfur, low nitrogen, non-toxic fuels)
  - Changing product demand (less heating oil and more gasoline)
- If nonfossil sources of hydrogen are used, lower-value refinery streams can be used to make gasoline rather than hydrogen—reduced oil imports

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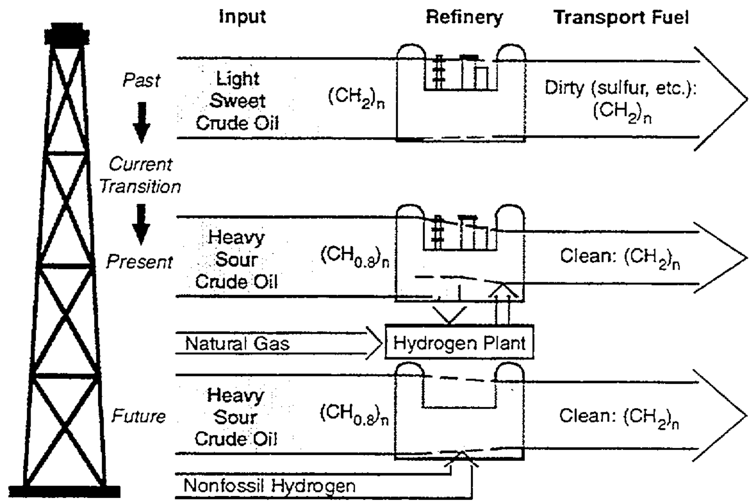
Increased Use of More Abundant Heavy Crude Oils Reduces Refinery Yields, Unless Nonfossil Hydrogen Is Used



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ORNL DWG 2001-107R

Multiple Benefits with Economic Nonfossil Sources of Hydrogen

- **Increased transport fuel yields per barrel**
  - **Lower-value oil components converted to transport fuel rather than to hydrogen (current practice)**
  - **Reduced imports of crude oil and natural gas**
- **Greater use of heavy crude oils**
  - **More abundant with lower costs**
  - **Western Hemisphere suppliers (Venezuela, Canada, and the United States)**
- **Competitive chemical and refinery industry**
  - **Natural gas price increases are increasing H_2 costs**
  - **Risk of parts of the industry moving offshore**
- **Lower carbon dioxide emissions**

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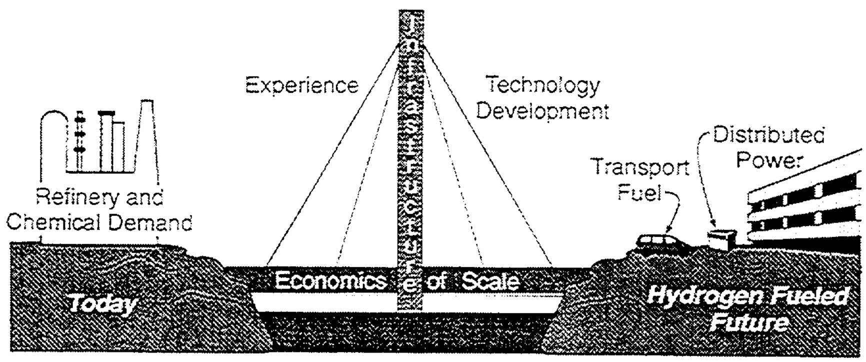
The Growing Industrial Demand for Hydrogen Creates a Bridge to the Hydrogen Economy



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Hydrogen Can Be Produced with Heat from a Nuclear Reactor

- Heat + water ⇛ hydrogen (H_2) + oxygen (O_2)
- Nuclear energy would compete with natural gas for H_2 production
  - Rising natural gas prices
  - Constant (level load) H_2 demand matches nuclear output
- Characteristics of hydrogen from water
  - Projected efficiencies of >50%
  - High-temperature heat is required: 800 to 1000ºC
  - Existing commercial reactors can not produce heat at these high temperatures
  - An alternative reactor concept is required

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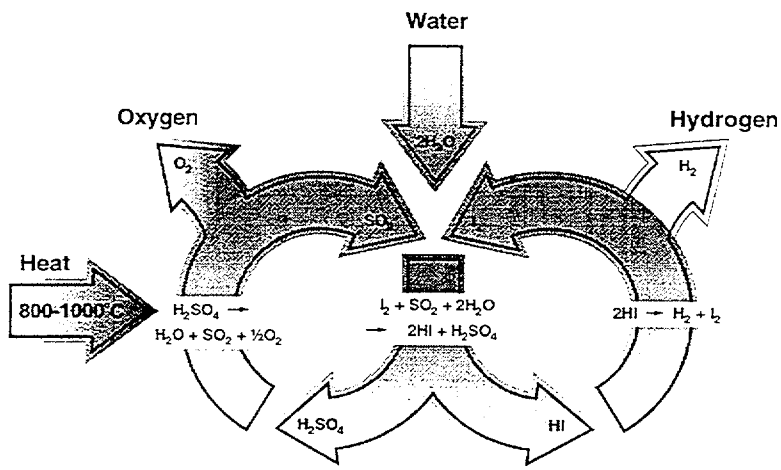
Chemical Processes Convert High-Temperature Heat and Water to Hydrogen and Oxygen

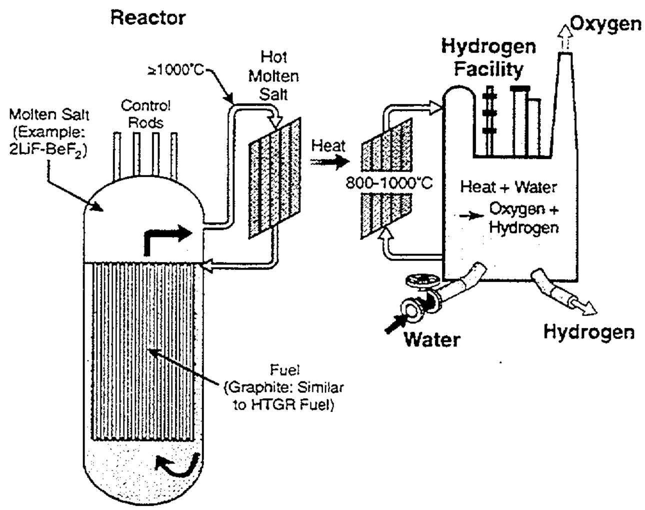
(Example: Iodine–Sulfur Process)



An Advanced High-Temperature Reactor (AHTR)—A Reactor Concept for Hydrogen Production

(Different products may require different reactors)

Advanced High Temperature Reactor Coupled to a Hydrogen Production Facility



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Desired Reactor Characteristics to Produce High-Temperature Heat

- Low-pressure system (atmospheric)
  - Metals become weaker at higher temperatures
  - Low pressures minimize strength requirements
  - Match chemical plant pressures (atmospheric)
- Efficient heat transfer
  - Need to minimize temperature drops between the nuclear fuel and application to deliver the highest-temperature heat
  - Liquid coolant

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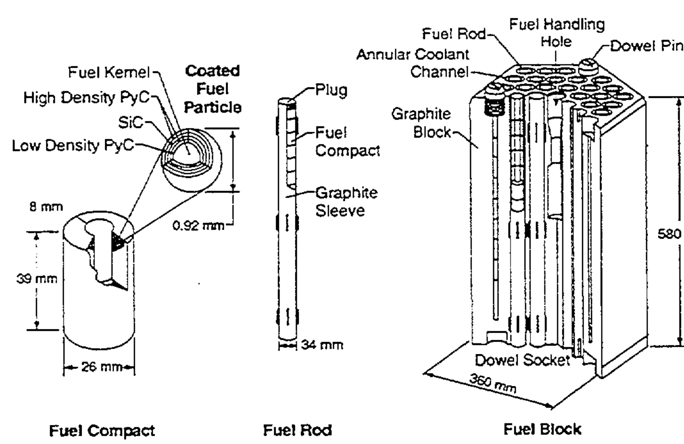
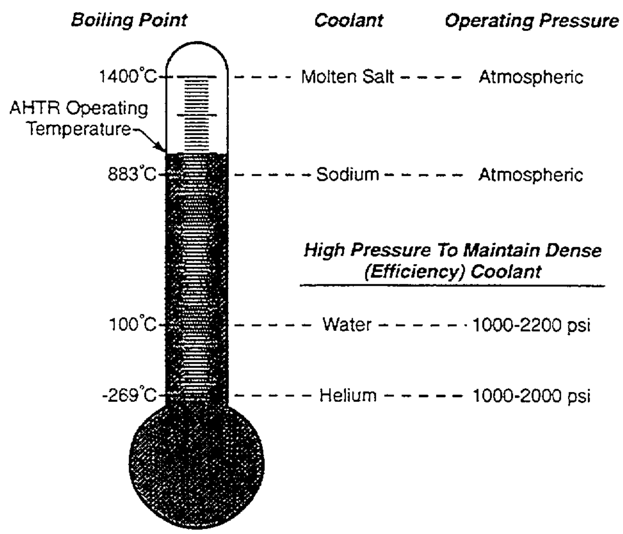
The AHTR Combines Two Different Technologies To Create an Advanced High-Temperature Reactor Option

- Graphite-matrix fuel
  - Demonstrated operation at an operating limit of ~1200ºC
  - Same fuel technology planned for modular high-temperature gas-cooled reactors
  - Fuel geometry/dimensions would be different for molten salt
- Molten salt coolant ($2LiF\text{-}BeF_2$)
  - Very low pressure (boils at ~1400ºC)
  - Efficient heat transfer (similar to that of water, except it works at high temperatures)
  - Proposed for fusion energy machines

Japanese High-Temperature Engineering Test Reactor Fuel for 950ºC Helium Exit Temperatures



Molten Salt Coolants Allow Low-Pressure Operations at High Temperatures Compared With Traditional Reactor Coolants



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The Safety Case for the AHTR

- Low-pressure (subatmospheric) coolant
  - Escaping pressurized fluids provide a mechanism for radioactivity to escape from a reactor during an accident
  - Low-pressure (<1 atm) salt coolant minimizes accident potential for radioactivity transport to the environment
  - Minimize chemical plant pressurization issues
- Good coolant characteristics provide added safety margins for many upset conditions
- Passive decay-heat-removal system similar to that proposed for other advanced reactors
  - Heat conducts outward from fuel to pressure vessel to passive vessel-cooling system
  - Power limited to ~600 MW(t)

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High Temperatures Also Create New Options For Production of Electricity

- **High-efficiency helium gas-turbine cycles**
  - Conversion efficiency >50% at 1000ºC
  - Provide isolation of power cycle from the reactor using low-temperature-drop heat exchangers
  - Use advanced gas-turbine technology
- **Direct thermal to electric production**
  - No moving parts (solid-state) methods to produce electricity from high-temperature heat
  - Radically simplified power plant
  - Potential for major cost reductions
  - Longer-term option—solid-state technology is in an earlier stage of development

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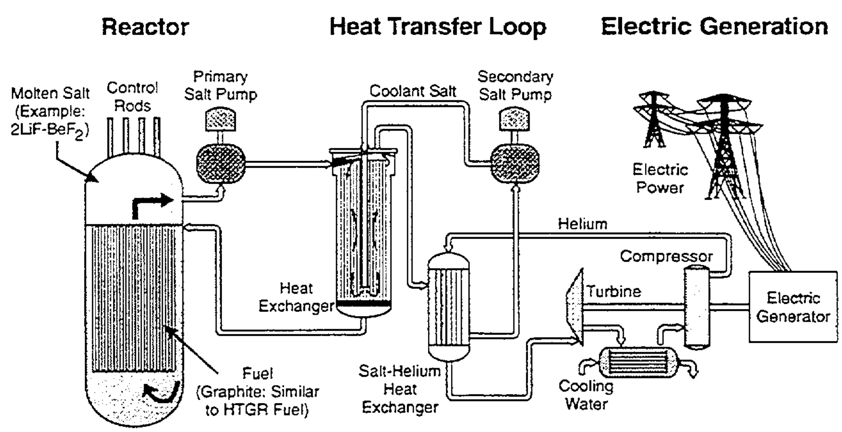
Advanced High Temperature Reactor With Brayton Cycle For Electricity Production



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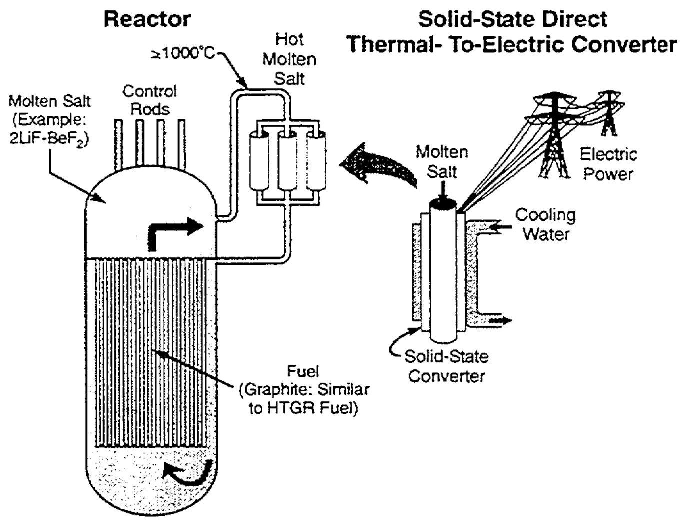
The AHTR May Enable the Longer-Term Option of Direct Conversion of Thermal Energy to Electricity



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High Temperatures Create Development Challenges

- AHTR uses some demonstrated technologies
  - Fuels (modified HTGR fuel)
  - Coolant
- AHTR requires advanced technology
  - High-temperature materials of construction
  - Optimized system design
  - Heat exchangers
  - Hydrogen and energy conversion systems

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Regulatory Implications of Hydrogen Production

- Different owners: oil & chemical companies
  - Larger than traditional utilities
  - Different perspectives
- Both chemical and nuclear safety must be considered (it is not clear where the primary hazard is)
  - Chemical plant must not impact nuclear plant
  - Nuclear plant must not impact chemical plant
- Non traditional (non-water, non-liquid-metal, non-gas) reactors may be preferred

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Conclusions

- Economic methods to produce hydrogen from nuclear power may provide multiple benefits
  - Increased gasoline and diesel fuel yields per barrel of crude oil with reduced dependence on foreign oil
  - Long-term pathway to a hydrogen economy
- High-temperature heat allows for new, more-efficient methods to produce electricity
- Reactors with different characteristics may be preferred for such different uses
  - Very high temperatures
  - Low pressures

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Added Information

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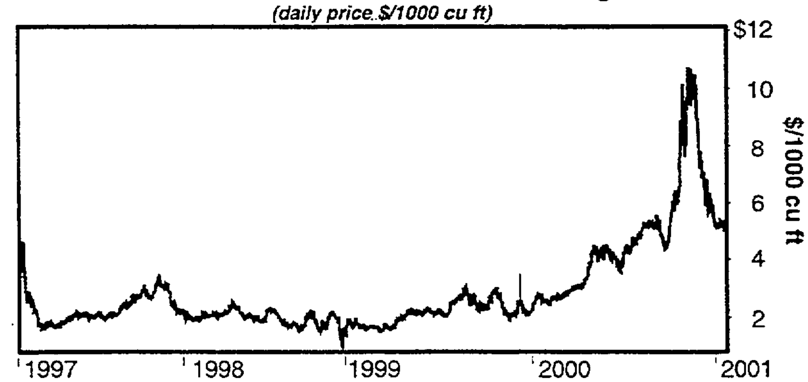
Hydrogen is Made From Natural Gas—If Gas Prices Remain High, a Significant Fraction of the Chemical and Refinery Industry May Move Offshore

U.S. Natural Gas Prices are Rising

(daily price $/1000 cu ft)



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There Has Been Extensive Development of Molten Salt Technologies For High-Temperature Nuclear Applications

- Initial development was for the Aircraft Nuclear Propulsion Program
  - Heat transferred from the solid-fueled reactor to the heat exchanger in the aircraft jet engine
  - Molten salts were chosen based on physical (pressure <1 atm.) and nuclear properties
- Molten salts are being considered for cooling fusion reactors (both types)
- Russian studies on molten-salt-cooled reactors

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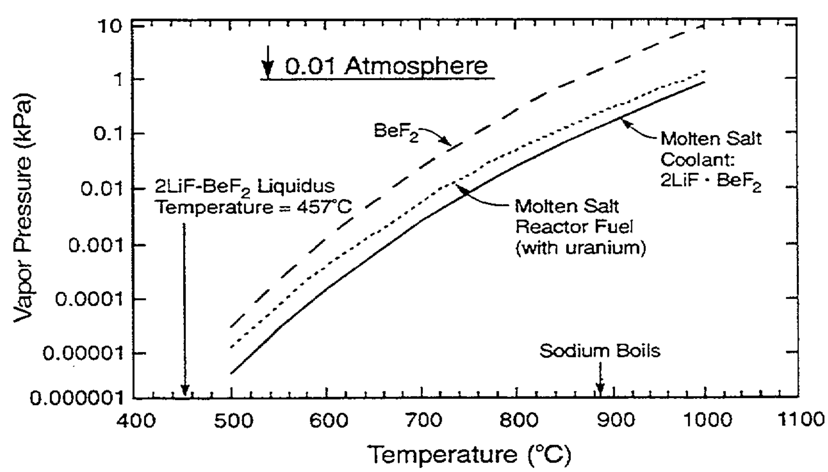
Vapor Pressure of $2LiF\text{-}BeF_2$ Is Low Compared To Other Reactor Coolants



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Characteristics of Molten Salts

- For the proposed $2LiF\text{-}BeF_2$ salt, the temperature rise from the AHTR operating point to the boiling point is ~400ºC
- Several other fluoride salts could be used
- Natural circulation cooling is an option
- Fluoride salts dissolve most fission products and actinides (basis for molten salt fueled reactor)
- Freeze point is ~457ºC
- Large industrial experience with other fluoride salts (aluminum production)

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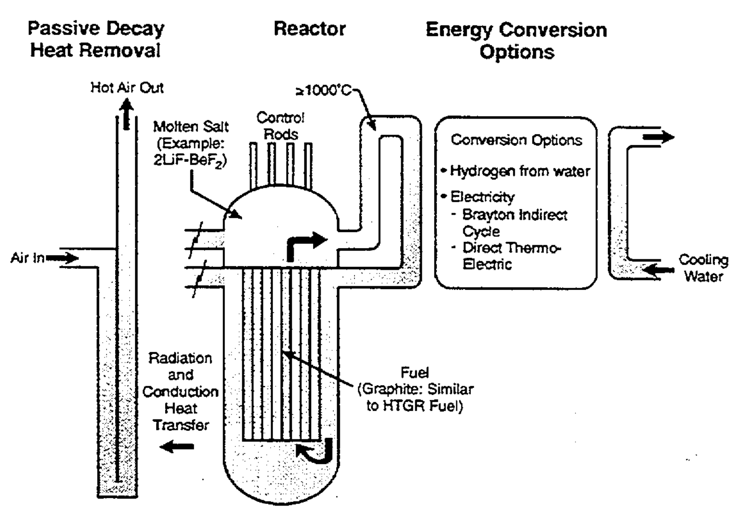
Advanced High-Temperature Reactor



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D. Powers, ACRS Member: It even goes beyond that because by taking out the aromatics you reduce the octane level -- octane rating of it, and so now you have to do more processing on the octanes.

C. Forsberg, ORNL: Yes. This type of refinery has about 95 percent efficiency. That is for every 100 BTUs going in here you get 95. This type of refinery for every 100 BTUs you get about 80 BTUs out. So the refinery efficiency is dropping.

D. Powers, ACRS Member: And those particular salts that you've got there, just about everything dissolves, even the things we think are nominally metals.

C. Forsberg, ORNL: I know. This is an unusual coolant. But it's a different approach to safety also, and that's why I mention it because we normally don't think of coolants as fission product absorbers. And in this case the coolant is a fission product absorber.

D. Powers, ACRS Member: We saw this in TMI, that you blow fission products through water. They stay in the water. And here all you're doing is magnifying that with a coolant that has a higher dynamic range than water does.

C. Forsberg, ORNL: Yes. I think it's an important issue though because there are different approaches to safety also that you can think about when you go to these high temperatures and when you go to other coolants.

D. Powers, ACRS Member: I think it has some interesting safety issues that are peculiar to itself. This is the classic problem of over-cooling accidents. Start-up is interesting. Start-up and shutdown, both are interesting events in this reactor.

C. Forsberg, ORNL: What D. Powers means by start-up is that this material thaws, becomes a liquid at about 400 C., molten salt. So you have a system that is, on start-up when it turns to liquid, is already moderately warm. In fact, it's hotter than any light water reactor on start-up, which is not your normal way of thinking about things.

J. Sieber, ACRS Member: I presume you pumped this molten salt around the surface. Are there pumps that can actually do that at these temperatures?

C. Forsberg, ORNL: Yes. Well, we haven't done anything at this temperature. The molten salt reactor experiment at Oak Ridge operated at 700 C. Now, the difference is in that reactor the uranium was dissolved in the salt. There was not a solid fuel element. But that operated about a much lower temperature of 700 C., and of course, nobody has operated a salt system at these temperatures.

J. Garrick, ACNW Chairman: Are you going to say anything about performance characteristics other than temperature and pressure?

C. Forsberg, ORNL: We're very early in the game, and I wouldn't make any promises that we have any information that would be considered credible. It's very, very early in the game.

J. Garrick, ACNW Chairman: Just cycle times?

C. Forsberg, ORNL: That's right. We started this effort about six or eight months ago, so we're very early in the game. Starting with the observation that there some -- maybe some demands for a very high temperature reactors, and if you have very high temperatures, how do you get there with the materials that may exist, and obviously you throw out water; you throw out sodium.

J. Sieber, ACRS Member: To maintain the pressure, how does it accommodate power swings that could be pretty sever in some accident situations.

C. Forsberg, ORNL: Yes. We're not at the point where we've investigated the details of how you're going to handle these types of events.

D. Powers, ACRS Member: Have you thought about what your primary pressure boundary is going to be?

C. Forsberg, ORNL: There are three obvious choices. One is a molybdenum alloy. Then there is some oxide dispersion stainless steels that may have the capability, and then there are also graphites. But we're very, very early. And all of those things are cases where people have shown in the laboratory that the materials are capable of doing something, but nobody knows whether or not they could be made on a large scale or whether you could fabricate them or whether you could convert this into a practical reactor design.

So what we have is materials that are used -- we have -- there are a number of high temperature materials that are used in research applications that operate at these conditions normally, in a research environment, but have not been used in a production environment. So what you have is materials that, yes, some of them have been used for 40 years, but only in a research environment. There's a big difference between research and production.

D. Powers, ACRS Member: There's a big difference between research environments and flowing, high velocity flows. The problem here is interesting. It's not carbon extraction, it's alloying-agent extraction.

C. Forsberg, ORNL: That's right. That's exactly right. There is a fair amount of experience based up to about 700, 800 C. Above 800 C., the databases begin to get very sparse.

T. Kress, Future Reactor Subcommittee Chairman: There wasn't any way to get the fission products out to the atmosphere or there didn't seem to be. The reason I say that is why wouldn't this be an attractive concept for just electricity generation? Because you don't have these extra hazards then of the chemical plant and so forth. And just by itself it looks like would be a pretty safe, inherently safe concept.

C. Forsberg, ORNL: I think it has potential attractiveness. And that's worth considering, but I think an important other consideration is that in this particular case you may also have multiple markets. And it's those multiple markets that may make it much more attractive for a serious consideration as an advanced reactor concept.
But clearly if you develop this, one will take a very hard look at it as a electric power producing reactor because those safety benefits apply to any other application as long as it doesn't have interface issues.

D. Carlson, NRC: Lithium 6 is a strong neutron absorber and produces copious amounts of tritium.

C. Forsberg, ORNL: It's isotopically separated lithium. Lithium 7. If we're looking at several coolants, some with lithium and some without lithium. The ones that include lithium have Lithium 7 because otherwise the neutronics doesn't work.

D. Carlson, NRC Staff: Well, even impurity levels of Lithium 6 would give you lots of tritium. In fact, in the pebble bed reactor work in Germany, where they were considering processed heat applications, the very small amounts of tritium on the order of 1,000 Curies per year were a concern in terms of getting the tritium into the product gas.

C. Forsberg, ORNL: Yes. That's why one of the reasons we consider multiple coolants. Each coolant has particular advantages and disadvantages. Neutronically the lithium beryllium fluoride is a tremendous advantage. But the disadvantages include tritium and a couple of other issues.

The sodium potassium, sodium potassium zirconium fluoride avoids that problem. It has a little more activity in the coolant, and has some other issues. So one of the issues in a molten salt reactor is which coolant you want. They all have the same general characteristics, but that's where the tradeoff comes on, coolant A versus coolant B.
You're absolutely right. That's why the coolant decision has not been made and why several coolants are being considered. All fluoride salts, but they have different benefits.

Nuclear Energy Institute (NEI) Summary

Prepared by ACRS Staff for A. Heymer

A. Heymer of NEI provided a brief discussion on the benefits of establishing a new regulatory framework. He suggested that a new paradigm in regulatory thinking is needed and stated that the reactor oversight process (ROP) serves as the appropriate basis for starting these discussions. He suggested that the ROP cornerstones of safety be used as the starting point for developing a new set of General Design Criteria (10 CFR Part 50, Appendix A). He suggested that new operating criteria, generic risk-informed and performance-based regulations be developed with associated design-specific and regulation-specific regulatory guides

New Plant Regulatory Framework

NRC ACRS Workshop on Advanced Reactors

New Regulatory Framework

Adrian Heymer, NEI

(aph@nei.org, 202-739-8094)

NEI

Benefits of Establishing New Framework

- **Helps establish a new paradigm of thinking**
  - Not burdened by current requirements or interpretations
  - Provides a standard against which to set requirements
- **Provide a platform for agreement on principles and objectives**
  - Ensures issues are focused on safety and are tied to defined safety objectives

NEI

Benefits of Establishing New Framework

- **Provides basis for NRC & industry positions**
- **Improves regulatory consistency**
  - Aligns regulations and oversight process
- **Use Reactor Oversight Framework as basis for starting industry & regulatory interactions**
  - Avoids "re-invention" of framework already accepted by NRC
  - Cultural change burden eased

NEI

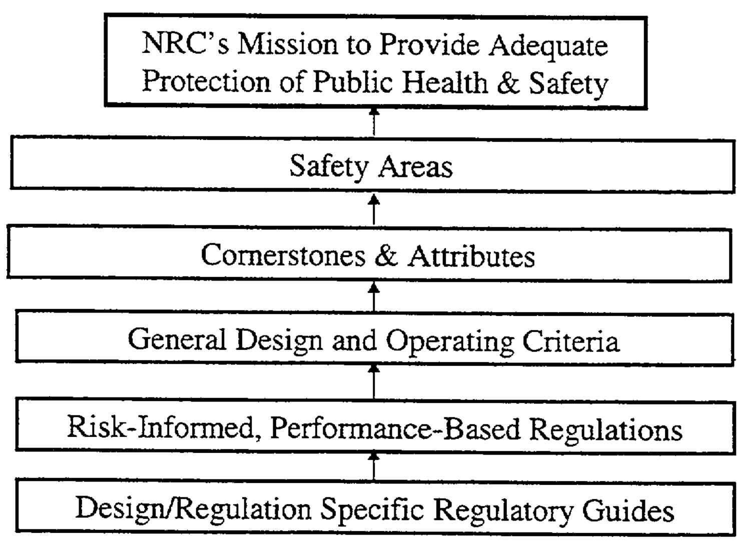
New Plant Regulatory Framework

- **Generic to all types of reactor**
- **Top-down approach based on NRC mission**
  - Adequate protection of public health & safety
- **Based on NRC oversight cornerstones**
- **New General Design Criteria**
- **Introduce General Operating Criteria**
- **Develop a new set of generic, risk-informed, performance-based regulations**
- **Develop design-specific and regulation specific regulatory guides**

NEI

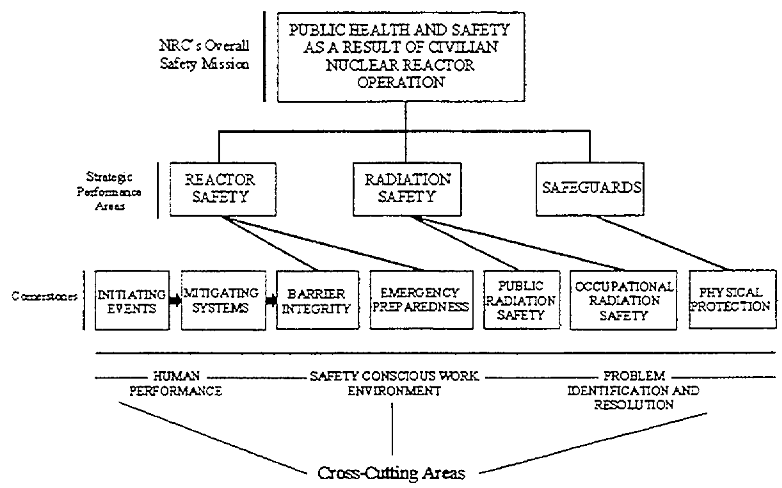
Establishing a New Regulatory Framework for New Plants

- **Concept -- Risk-Informed, Performance-Based Licensing and Regulatory Regime**
- **Proof-of-concept application(s)**
  - Use License Renewal and Option 2 models
  - Minimizes hypothetical discussions
  - Definitive schedule to drive resolution process
- **Industry effort consolidates lessons learned from proof-of-concept activities**
  - Vehicle for supporting proof-of-concept positions



DRAFT

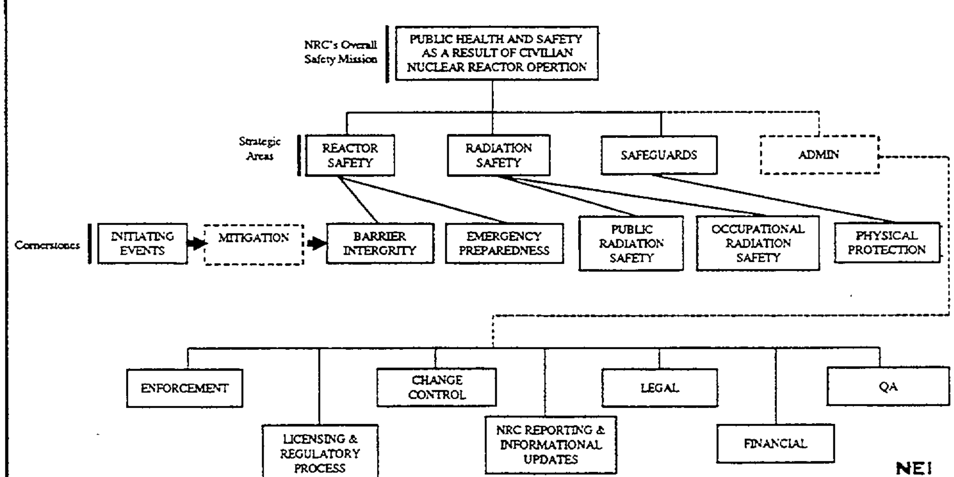
REGULATORY OVERSIGHT FRAMEWORK



NEI

DRAFT

REGULATORY FRAMEWORK FOR NEW PLANTS



NEI

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Cornerstones
10 CFR Part 50

- **160 GDCs, Regulations & Appendices**
  - Initiating Events -- 16
  - Mitigation (*Systems*) -- 46
  - Barriers -- 27
  - EP -- 3
  - Pub. Radiation Safety -- 9
  - Occupational Safety -- 4
  - Safeguards -- 4
  - Administrative -- 68
  - Financial -- 6
  - *Operational -- 23*

NEI

Example of New Regulation

XX.63 Plant configuration management

Licensee shall assess and manage changes in risk that result from maintenance, modifications and operational activities that could degrade safety-significant functions.

DRAFT

NEI

Example of New Design Criteria

Protection against natural phenomena

Safety-significant structures, systems, and components shall be designed to withstand, or be protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design and protective features shall reflect the most severe natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for uncertainty related to the limited accuracy, quantity, and period of time in which the data have been accumulated.

DRAFT

NEI

G. Apostolakis, Chairman, ACRS: Everybody keeps saying risk informed performance based, but can licensing really be performance based?

A. Heymer, NEI: I think in the context of purely the licensing action, no, but what follows on afterwards is.

G. Apostolakis, Chairman, ACRS: Oh, the regulatory.

A. Heymer, NEI: Yes.

G. Apostolakis, Chairman, ACRS: The oversight, sure. We are not dealing with that now. You are dealing with licensing, aren't you?

A. Heymer, NEI: Well, we think that if you put a new Part 53 in place that there should be some element dealing with operational aspects, and so that's where we see that coming in, and there's also a probability that if you look at the Part 52 process in ITAAC, that is akin to a performance based element to a certain extent.

G. Apostolakis, Chairman, ACRS: What I'm saying is that you are overplaying it a little bit, unintentionally, the significance of the fact that this framework has been used in the oversight process. The fundamental issues are there. If you look at the report the staff developed on Option 3, essentially they follow the same approach, but they dare go beyond that, and I think you guys are a little cool towards the other stuff they did. If you look at what Golay did, well, it's buried in there. It's the same idea. So I think this is a good starting point, but I wouldn't overplay the connection to the oversight process. It's a very different regulatory problem. That's my impression.

A. Heymer, NEI: That's good insight. It's good input. I'm going to take that.

G. Apostolakis, Chairman, ACRS: What we're seeing now on the screen is the NRC oversight process. When you go to yours, you are adding a fourth element in the second tier, but how about the bottom? What happened to human performance, safety conscious work environment, and problem identification or resolution? Are you going to handle those in a different way?

A. Heymer, NEI: Problem identification and resolution is in the quality assurance element.

G. Apostolakis, Chairman, ACRS: I thought your -- the emphasis of your talk was going to be on licensing of the new concepts. But yours seems to be attacking the whole thing.

A. Heymer, NEI: It's a regulatory --

G. Apostolakis, Chairman, ACRS: Does Exelon really worry about how the NRC will regulate the pebble bed after they get the license? They worry about it right now?

A. Heymer, NEI: They worry about it right now, but if you're dealing with -- and that's why I said when you develop the framework, you have people like Exelon moving out and testing the process on a pebble bed, and there's a feedback process that comes in and you can adjust.

M. Bonaca, ACRS Member: Although I must say that I still am confused about what's different in this from the previous system. I mean I could take the previous -- the existing system and then put it on --

A. Heymer, NEI: From a framework perspective, not much. It's when you get down to specific regulations you begin to see --

M. Bonaca, ACRS Member: Okay. Well, I can understand that. I don't quite understand from the examples where the differences may be, and I really couldn't figure it out. But I understand your intent. I mean, clearly you said before that it has to be risk informed and you're looking. The reason why I bring it up is that we saw a number of innovative processes this morning, and the concern I have is that you can put in a licensing framework now that may stifle, in fact, the credibility of some of the innovative cultures as much as the old system stifles.

A. Heymer, NEI: Well, when you look at the framework and you see the current regulations and requirements, I would agree with you. If you look at the framework and say there are alternative regulations or a different set of regulations, a different set of design criteria, I think that gives you the flexibility.

G. Apostolakis, Chairman, ACRS: No, there is a slight problem here, I think, in the sense that I cannot determine what is risk significant or safety significant until I have a PRA which will tell me when the PRA will be based on the actual design, but now I'm supposed to use the results of that PRA, in fact, to create the knowledge base for the PRA.

A. Heymer, NEI: Well, it's an iterative process.

G. Apostolakis, Chairman, ACRS: So you start with one and do it and do it again?

A. Heymer, NEI: Yes, and there is experience. You just don't say, "Well, I'm starting with a new design. What have I got?" I mean, there's --

G. Apostolakis, Chairman, ACRS: I must say overall though, Adrian, maybe it's too early in the process, but I, frankly, thought you were going to come up with something that's a little more daring. You are really sticking to the existing regulations which you have blasted in the past. We must be doing something right.

A. Kadak, MIT: Let me suggest something a little more daring, and it's reestablishing the regulatory compact between what the regulator's job is, what the licensee's job is in terms of how they deal with the future protection of public health and safety from a system that is quite prescriptive in terms of its requirements to something that more fully puts the burden on the operator to meet some high level goals.

And I'm not sure what that new relationship is, but clearly if we go to 1,000 plants, let's just say, in trying to build on G. Apostolakis's ten times whatever the probability is and it gets to be a large number, that you can't continue doing it the same way, and what new regime might be appropriate to protect the public health and safety in the sense of a risk informed and performance based system. So that addresses the inspection and addresses the enforcement action, as well as the standards that you apply to new technology. So that's kind of the comment to the NEI people as well as to the rest of us, and that is how can we improve the

overall process not only for design and construction and operation, but also regulation. If there was a question on that, you can try to answer it, but it's a new regulatory paradigm.

A. Heymer, NEI: Yes. It's thinking ahead and saying, like just challenging the NRC relative to how are they going to do license renewals for 80 plants in the next five years or ten years. They can't. Something has to change, some trust, some new relationship, and we have to figure out how that will work in a legal way.

Dana Powers ACRS Member: Well, I think they came up with a fairly effective solution. I mean, they've gone through the catalog to a variety of data on the age degradation, a huge number of topical reports, they run four or five pilots, established a template, and people were following the template. Based on what we saw from ANO, you follow the template and you put out a pretty good product, and it goes very quickly. You're not going to have 80 new concepts in five years. We haven't got the same problem.

ACRS WORKSHOP

Key Regulatory Challenges for Future Nuclear Power Plants

Neil E. Todreas
KEPCO Professor of Nuclear Engineering
Massachusetts Institute of Technology

PM June 5, 2001

CHALLENGES

FUEL AND CLAD MATERIALS - TAKEN TO HIGHER BURNUPS AND OPERATED AT HIGHER TEMPERATURES.

Drivers: Longer Operating Cycles.
Higher Temperature Primary Systems.

Particular Challenges:
1) Reductions in Waste Toxicity and Volume.
2) Understanding and Control of Coolant Corrosion, particularly role of coolant impurities.
★ 3) Qualification of Core Loads of Billions of Fuel Particles.
★ 4) New Maintenance Practices.

3) Questions Regarding Particle Fuel Qualification

- How many particles, if failed at the most limiting time in core life released, would be required to exceed the following conditions:
  - Dose limits for plant workers?
  - The lowest condition on the IAEA scale of plant incidents?
  - Protective action guidelines for the general public?

- If the fuel particle specification is product based:
  a. What are the individual particle attributes which are controlled by the specification, and for each, to what levels, and allowable variation to prevent particle failure?
  b. What is the allowable variation in related individual particle attributes which must be maintained to prevent particle failure?

- If the fuel particle specification is process based:
  a. What are the individual process variables which are controlled by the specification, and for each, to what levels, and allowable variation to prevent particle failure?
  b. What are the individual allowable variations in process variables which are sufficient to prevent particle failure?
  c. What is the allowable variation in related individual process variations which must be maintained to prevent particle failure?

Particle Fuel - Consequences of a Process Specification

- **Critical Operator Actions now become located in the fuel fabrication facility. The fuel fabricator is the de facto control room operator.**

- **Innovation in particle fuel design & fabrication processing is likely more costly and hence inhibited.**

4) Maintenance Practices

Driver: - Longer Operating Cycles

Frequencies - Extended

Plant Mode - More on-line.

Practice - Example: Relief Valve Testing

M.I.T. Dept. of Nuclear Engineering

5

Why are these items Challenges?

- New Technologies - require development of
  - NRC staff expertise
  - NRC confirmatory research basis

- Design Solutions are aimed at precluding historic initiators
  - Establishment of a new risk-based regulatory framework will be needed.

M.I.T. Dept. of Nuclear Engineering

6

REGULATORY CHALLENGES FOR THE LICENSING OF FUTURE NUCLEAR PLANTS: A PUBLIC INTEREST PERSPECTIVE

Edwin S. Lyman
Scientific Director
Nuclear Control Institute

ACRS Advanced Reactor Workshop
June 5, 2001

REGULATORY CHALLENGES

- NRC licensing of advanced plants must ensure that these economic imperatives do not have adverse impacts on
  - Safety
  - Risk of radiological sabotage
  - Waste management and disposal
  - Non-proliferation
  - Full opportunity for public participation

EXAMPLE: PBMR

- PBMR characteristics **fundamental to its economic viability** represent significant deviation from traditional "defense-in-depth"
  - Lack of pressure containment
  - Significant reduction in safety-related SSCs
  - Reduction in EPZ radius by a factor of 40 (exploits regulatory exemption for HTGRs)
  - Greatly increased reliance on fuel integrity under accident conditions for protection of public health
- ACRS (1988): "unusually persuasive argument" required to justify "major safety tradeoff"

PBMR FUEL PERFORMANCE AND SAFETY GOALS

- Source terms must be accurately determined for a full range of potential accidents
  - Pebble performance very sensitive to initial conditions ---- relationship poorly understood
  - Robustness of PBMR fuel is being oversold --- significant fission product release (several % of Cs inventory) can occur at 1700-1800°C) --- hundreds of degrees below fuel degradation temperature
  - Quality control is paramount --- BNFL involvement in South African fuel fabrication plant suggests that a fuel quality control programmatic ITAAC is necessary

PBMR SAFETY GOALS

- Safety goals need to be reexamined for advanced reactors
  - Current goals not conservative enough --- could still be met by reactors today with containments removed!
  - "Large release fraction" if EPZs are reduced
- Accident frequencies that could result in LR must be accurately calculated
  - Design-basis LOCA --- safety margin may be too small
  - Air or water ingress
- System upgrades may be necessary to meet goals
  - secondary coolant system (MIT vs. Eskom)
  - advanced fuel coating materials (i.e. ZrC)

RADIOLOGICAL SABOTAGE --- THE "SHOW-STOPPER"?

- Providing adequate physical protection to defend plants against sabotage has proven to be a major challenge:
  - 50% of U.S. nuclear plants failed force-on-force (OSRE) testing of plant security in 2000
  - At Exelon's Quad Cities plant, "deficiencies in the licensee's protective strategy enabled the mock adversaries to challenge the … ability to maintain core cooling and containment" (NRC, October 18, 2000)

RADIOLOGICAL SABOTAGE (cont.)

- No nuclear system can be rendered "inherently safe" from radiological sabotage
  - Deliberate graphite fire in PBMR remains possible even if accidental fire is incredible
  - Reduction in security staffing requirements for PBMRs not technically justifiable
  - Systems with in-situ reprocessing plants (S-PRISM) would be especially attractive targets
- ACRS (1988) recommended that NRC develop guidance for incorporating sabotage resistance into advanced designs --- need early involvement of Reactor Safeguards staff

PBMR WASTE DISPOSAL

- Final waste disposal may be the single largest obstacle to nuclear power expansion
- Spent pebbles create a huge waste problem: per MWD, compared to spent LWR fuel:
  - Volume and weight are about 10 times greater– with proportionate increase in storage and transport requirements
  - Carbon-14 inventory is 10-20 times greater --- problem for unsaturated repository like Yucca Mountain

PUBLIC ACCEPTANCE

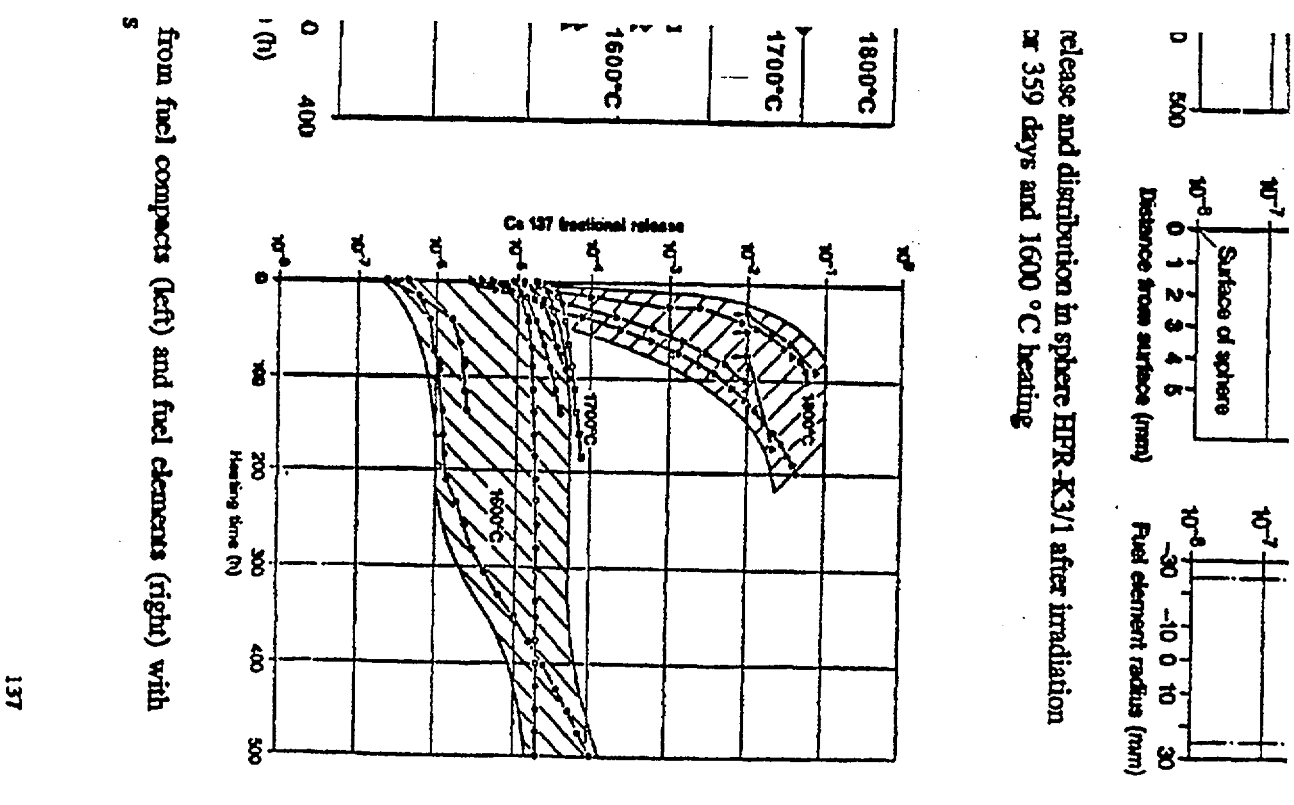
- New facility siting is a great challenge:
  - Favors new plants at existing sites in areas of broad public support
  - Trying to greatly increase number of nuclear plant sites is a losing strategy --- but there is little advantage in modularity if available sites remain highly limited
  - Favors minimization of transport of nuclear materials
- Public opposition may only be deterred with a clear commitment to maximize safety:
  - Favors "gold-plating" nuclear plants
  - Inconsistent with attempts to eliminate containment, reduce emergency planning, etc

PUBLIC ACCEPTANCE (cont.)

- Aggressive licensing schedule proposed by Exelon for PBMR (construction to begin in 2004, operation in 2007) will only antagonize antinuclear groups now mobilizing
- "License by test" is just a PR move --- unlikely to be adequate to resolve all safety issues to NRC satisfaction
- Better to proceed more cautiously and make sure that full resolution of all technical concerns is achieved

THE FUNDAMENTAL DILEMMA OF NUCLEAR POWER EXPANSION

- Without ratepayer or taxpayer subsidy, no new nuclear plants will be built unless they can successfully mimic the desirable economic features of gas turbines:
  - low capital cost
  - short construction time
  - modularity and ease of distribution
- Can this be done safely? Or is nuclear technology incompatible with these objectives?



was higher than that of ^{137}Cs, which was consistent with the previous work.[10–13] Although the inventory is small, the release of ^{110m}Ag would be troublesome in mainte-

ticles. To compare the irradiation performance of the individual particles, activity ratios, not activities, were used to account for variations in kernel size and to minimize

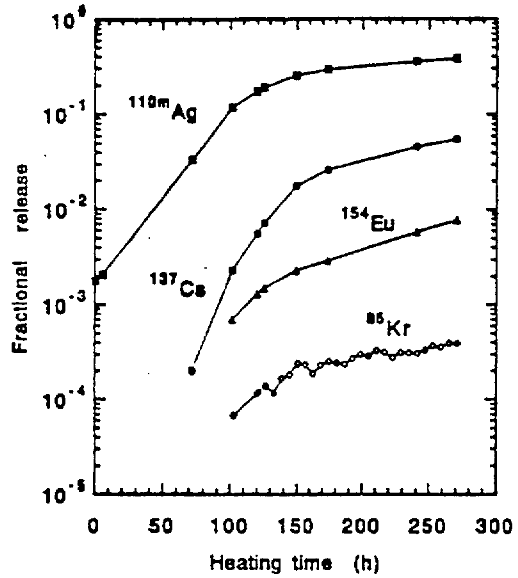


Fig. 2. Time-dependent fractional releases of fission products during the ACT3 heating test at 1700°C for 270 h, obtained by the on-line measurements of fission gas release and intermittent measurements of metallic fission product release.

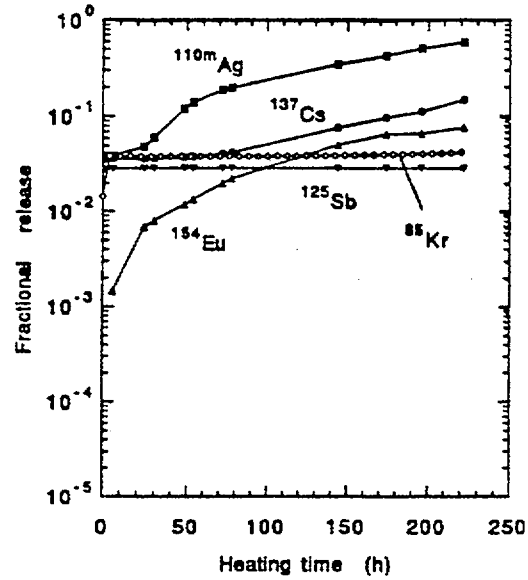


Fig. 3. Time-dependent fractional releases of fission products during the ACT4 heating test at 1800°C for 222 h, obtained by the on-line measurements of fission gas release and intermittent measurements of metallic fission product release.

L. E. Hochreiter, Penn State University: Some of these designs are looking at not having a containment, and then I think you have issues. Today in the light water area, really failed fuel is a utility or an operator concern, and it's a vendor concern, and you're very, very careful about it because obviously if you want to sell fuel, you don't want it to fail. So it's a problem that solves itself. But you've got a containment around the plant. In some of these designs you don't have a containment, and I think it could be more of a problem.

N. Todreas, MIT: Okay. First, let me answer I'm not promoting either a process or a product specification. What I am doing is asking whether it is going to be a process or a product specification, and then developing a line of questioning along each.

L. E. Hochreiter, Penn State University: Neil, on your process control, are you envisioning a control process where you can try to control each, on these particles, each layer in this thickness within a specified amount or the total product as it comes out?

Because I don't see how you control each layer, and if you control on the total product that comes out, if it doesn't come out right, and you won't find that out probably until you operate, then you've got a problem.

However, now, in addition though the way you ask the words, a process specification means that you control the process of every manufacturing step. So you may have a process where you're doing the coating, but you don't go and measure the coating or sample the coating. What you do is you control the attributes of the fabrication process. How do you know you meet your criteria if you don't go and measure?

N. Todreas, MIT: No, no, because what you do in the qualification stage, you take the product that comes out; you put it in the reactor; and you'd better make well sure it can take the burn-up with a failure criteria over whatever your design lifetime is.

L. E. Hochreiter, Penn State University: At some point you're going to have to have gone through and verified that whatever your process is gave you the product that you wanted.

Neil Todreas, MIT: There's a tremendous amount of radiation data on this particle fuel. If you can pin down the process that it was made to and link it to the data, then you can say you identified the process, and then you can basically duplicate it and keep going. That's the burden the applicant is going to have.

ENDNOTES

The ACRS Subcommittee on Future Reactors met on June 4-5, 2001, at 11545 Rockville Pike. The purpose of this meeting was to discuss regulatory challenges for future nuclear power plants.

The Subcommittee received no written comments or requests for time to make oral statements from members of the public regarding the meeting. The entire meeting was open to public attendance. M. Markley was the cognizant ACRS staff engineer and Designated Federal Official for this meeting. The meeting was convened at 9:00 a.m. and recessed at 7:15 p.m. on June 4. The meeting was reconvened at 8:30 a.m. and adjourned at 5:50 p.m. on June 5. During the course of the meeting, ACRS members Apostolakis, Leitch, Powers, and Sieber and ACNW member Garrick announced that they have conflicts with certain presentations made to the Subcommittee.

PARTICIPANTS

ACRS/ACNW

T. Kress, Subcommittee Chairman
G. Apostolakis, ACRS Chairman
M. Bonaca, ACRS Member
P. Ford, ACRS Member
G. Leitch, ACRS Member
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A. Rae, NRR
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Principal Presenters and Speakers

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R. Versluis, DOE

NRR	Office of Nuclear Reactor Regulation
RES	Office of Nuclear Regulatory Research
DOE	U.S. Department of Energy
GE	General Electric
MIT	Massachusetts Institute of Technology
NCI	Nuclear Control Institute
NEI	Nuclear Energy Institute
ORNL	Oak Ridge National Laboratory
PBMR	Pebble Bed Modular Reactor

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG/CP-0175

2. TITLE AND SUBTITLE

Proceedings of the Advisory Committee on Reactor Safeguards
Workshop on Future Reactors

June 4 - 5 , 2001

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	2001

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Because of the large amount of regulatory activity that is anticipated for licensing future reactor concepts, the ACRS decided to hold this workshop on "Regulatory Challenges for Future Reactor Designs." The workshop was held primarily for the benefit of the Committee C to acquaint the members with the various design concepts and to identify potential regulatory and policy issues for which ACRS may be called upon to give advice to the Commission. It was also believed that the workshop would be of benefit to the NRC staff as well as to the industry in getting an early dialogue started on the possible regulatory approaches to licensing future reactor designs. These designs are expected to be significantly different from the LWRs which are the primary focus of the current regulations and regulatory system.

The primary purpose of the workshop, as indicated by its title, was to identify the regulatory challenges. A list of such challenges identified by the workshop was developed from the workshop notes, the various presentations, the panel discussions, and the question and answer sessions.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

AP100
Department of Energy
ESBWR
Future reactor designs
Generation IV
IRIS
LWRs
MHTGR
PBMR

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300