Next Steps Generation III+/IV

Presentation at ACRS Workshop "Regulatory Challenges for Future Nuclear Power Plants"

June 4, 2001

R. Shane Johnson, Associate Director
Office of Technology
and International Cooperation

Near-Term Deployment of Advanced Reactors

Near-Term Actions

- Complete report on recommended DOE activities
  - Report will reflect generic and design specific issues
  - Report to be issued by September 30, 2001
- Significant activities expected to include:
  - Development of Regulatory Framework for Gas Reactor Technologies
  - Early Site Permit Demonstration
  - Combined Construction/Operating License Demonstration
  - Design Certification of Advanced Reactors

Generation IV Technology Roadmap

Near-Term Actions

- Evaluate the most viable concepts
- Compare concept performance to technology goals
- Identify technology gaps
- Identify R&D needed to close technology gaps
- Prepare comprehensive report on most promising concepts including detailed R&D plan

Safety Design Aspects and U.S. Licensing Challenges of the PBMR

Ward Sproat - Exelon Generation
Dr. Johan Slabber – PBMR Pty.

Agenda

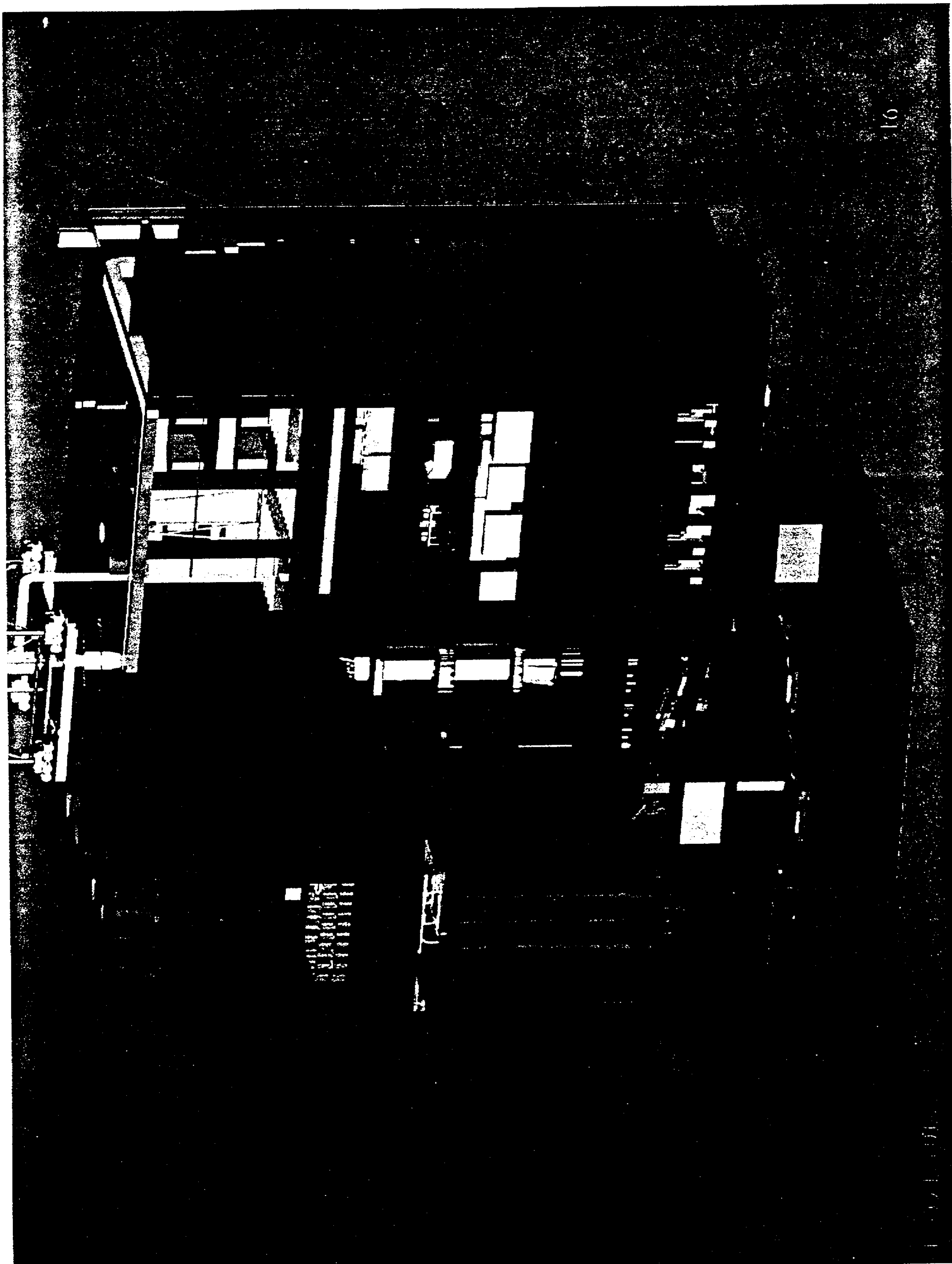
- Project Overview
- PBMR Safety Design Features
- U.S. Licensing Challenges

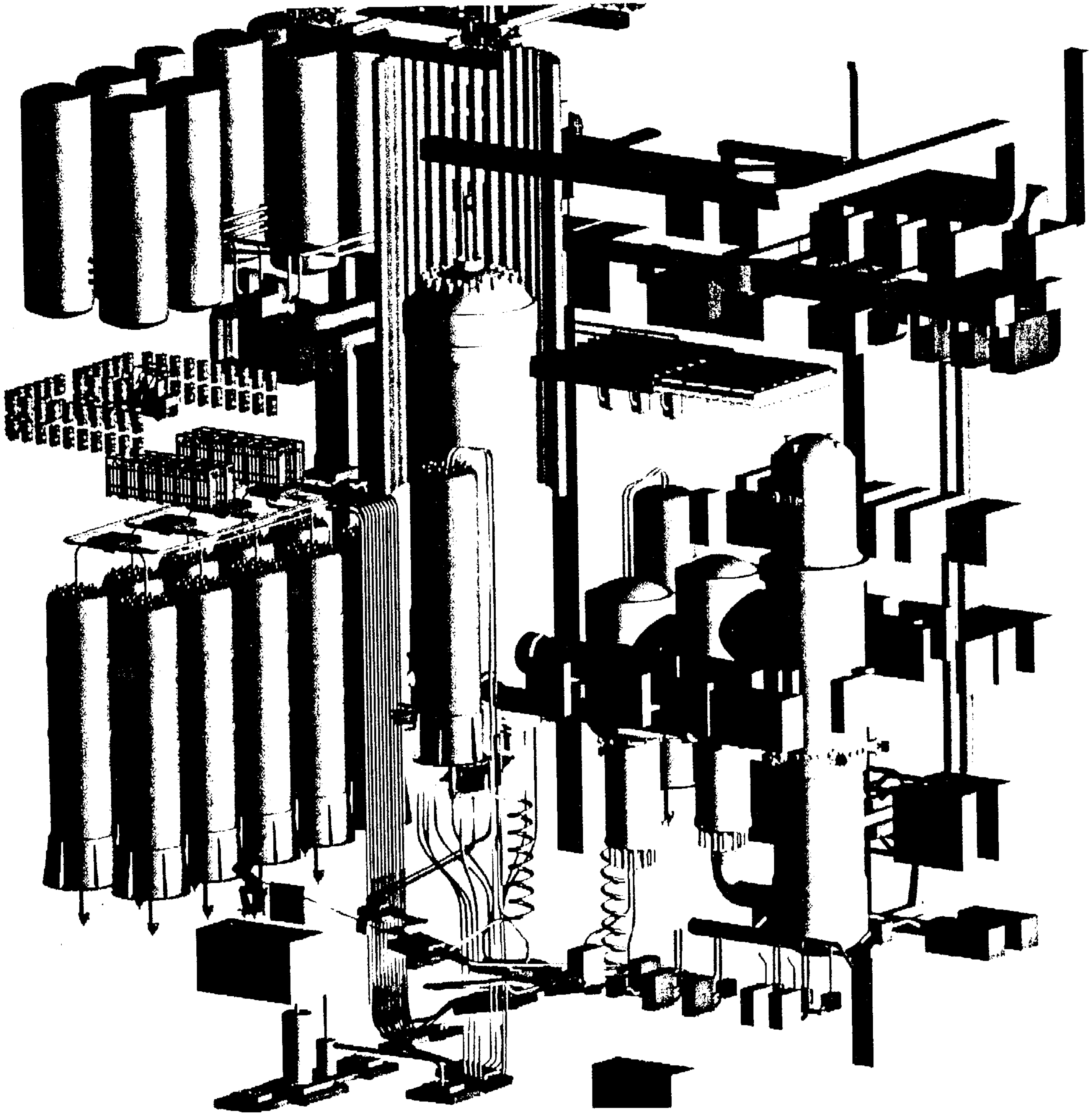
PBMR Project Overview

- Ending Preliminary Design Phase
- Feasibility Study in preparation
- Investors' decisions by end of year
- RSA demonstration plant construction start in late 2002 pending approvals
- Exelon decisions hinge on economics and technical risks

Design Philosophy

- Employ passive and active engineered features
- Provide prevention and mitigation capability
- Reduce dependence on operator actions

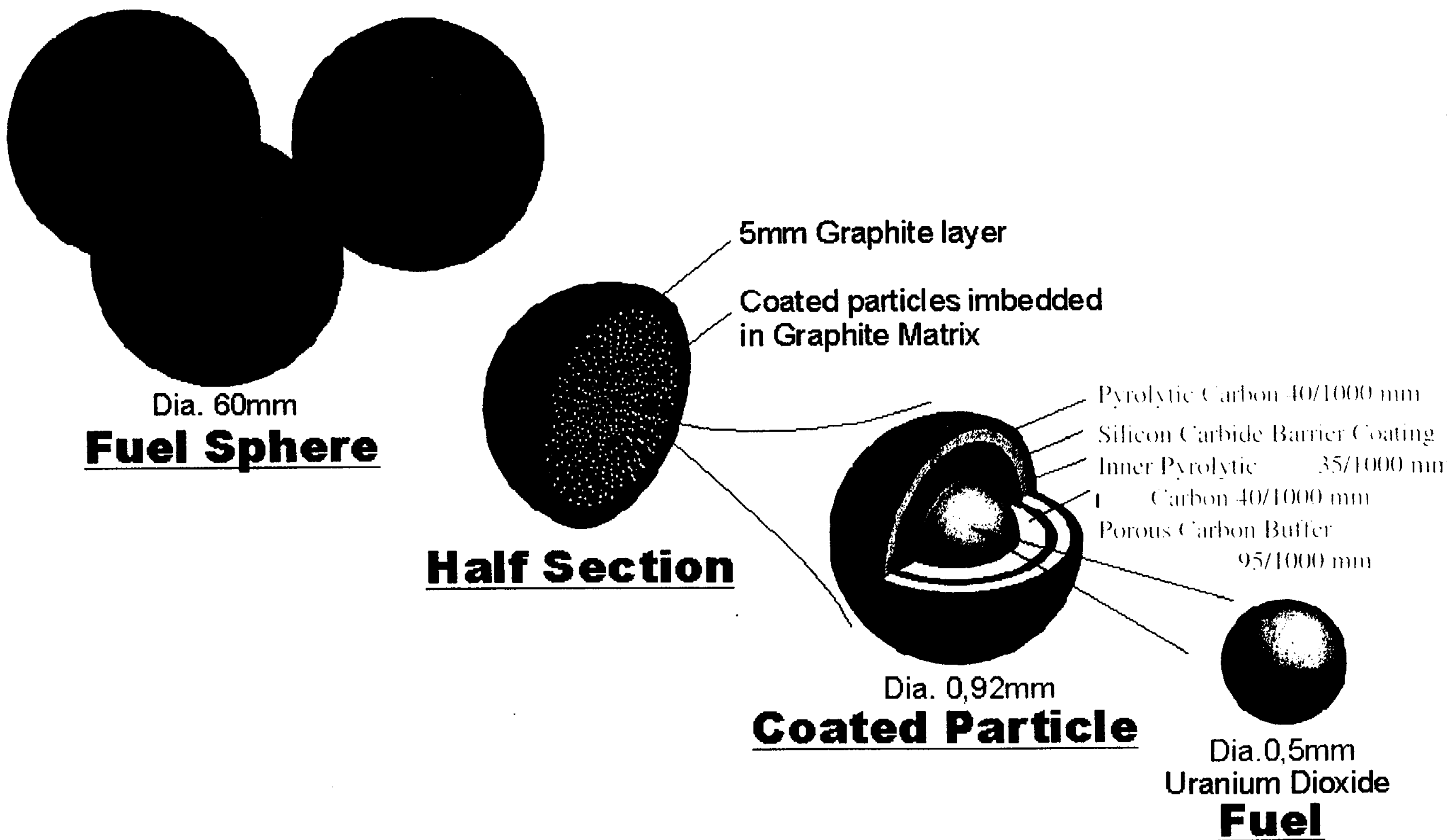




Reactor Safety Design Principles

- Assure fuel integrity
- Multiple fission product barriers to the environment
- Nuclear material proliferation safeguards

FUEL ELEMENT DESIGN FOR PBMR



Jan 31 2001

Reactor Design Principles

- Assure Fuel Integrity
  - Assure Fuel Quality
  - Control Excess Reactivity
  - Assure Heat Removal from Fuel
  - Prevention of Chemical Attack
  - Prevent Excess Burnup

Assure Fuel Integrity

- Assure Fuel Quality
  - Fuel Design has been proven internationally
  - Fuel Qualification Program
    - Fuel Performance Testing Program
    - Fuel Fabrication Quality Assurance Program
  - Operational fuel integrity assurance by monitoring primary coolant activity online

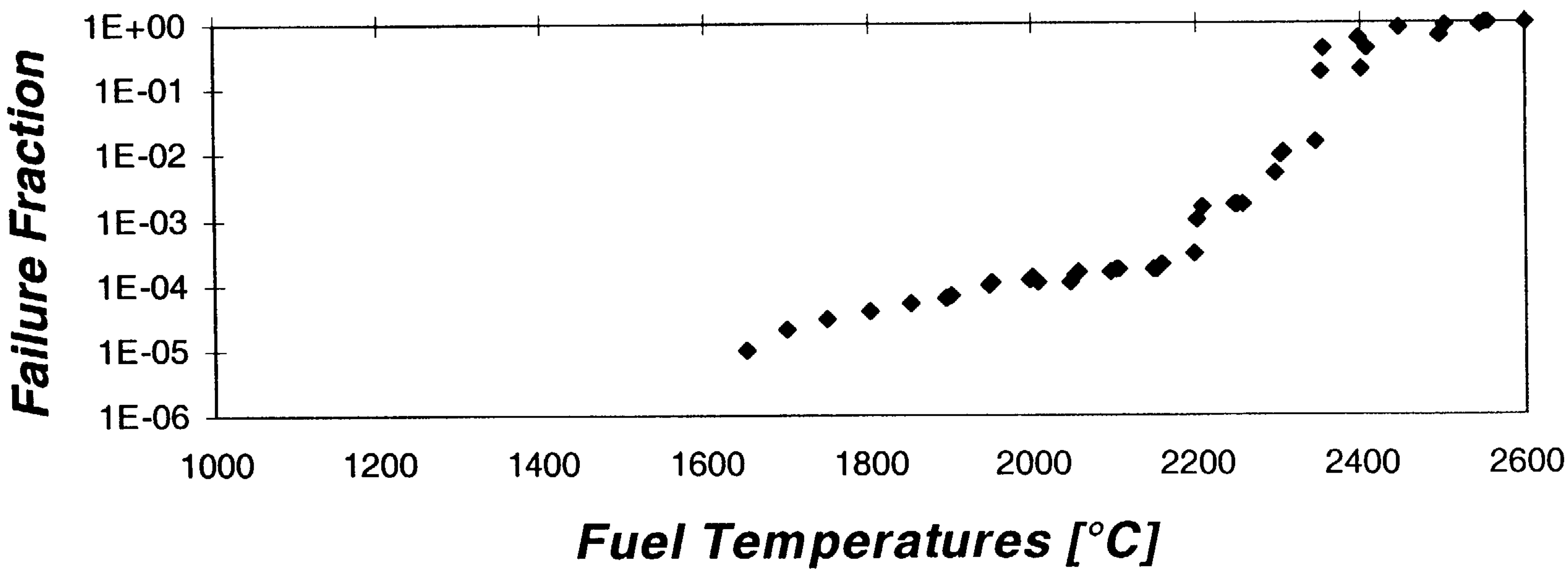
Assure Fuel Integrity (cont'd)

- **Control of Excess Reactivity**
  - Low Excess Reactivity = 1.3% delta k effective
  - Core geometry maintained by design for all credible events
  - PBMR core design precludes Xenon oscillations
  - Demonstrable large Negative Temperature Coefficient of Reactivity
  - Criticality safety assured for spent and used fuel

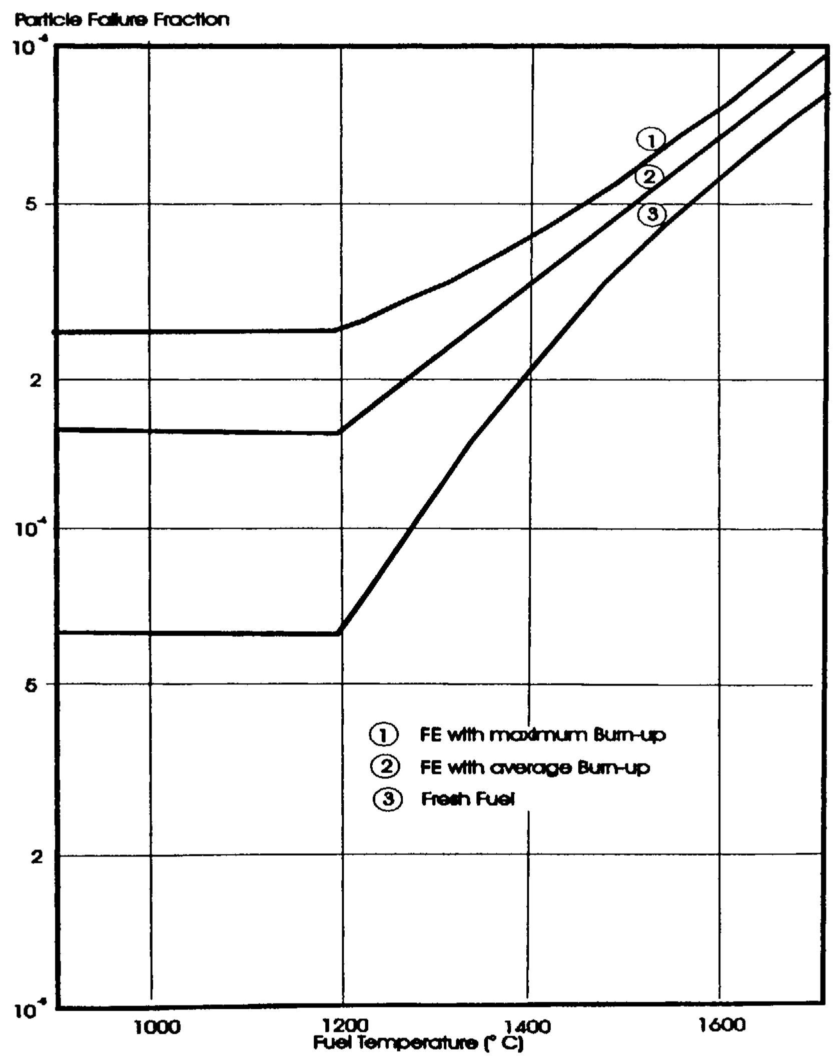
Assure Fuel Integrity (cont'd)

- **Assure Heat Removal From Fuel**
  - Materials properties and design features assure heat transfer from fuel to RPV
  - Passive heat sink provided by the Reactor Cavity Cooling System for extended period
  - The reactor cavity including its structures will maintain geometry during all credible events.

Fuel Performance at Elevated Temperatures



Nominal Fuel Performance



Assure Fuel Integrity (cont'd)

- **Prevention of Chemical Attack**
  - Water systems at a lower pressure than that of the primary coolant system during operation
  - Water ingress to reactor when depressurized prevented by physical design
  - Primary coolant system monitored to detect, and cleaned to remove moisture and air
  - Graphite oxidation due to air ingress prevented by physical design of reactor, gas manifold and citadel

Assure Fuel Integrity (cont'd)

- Prevention of Excess Burn-up
  - Physical core design
  - On-Line gamma spectrometric system to measure fuel burn-up

Fission Product Barriers to Environment

- Individual fuel kernels with 3 layers
- High integrity primary pressure boundary
- Containment (Confinement)
  - Reinforced concrete structure
  - Filtered vent path
  - Hold up of fission products
  - Plate out
  - Auto-close blowout panels
  - Late release

Nuclear Material Proliferation Safeguards

- International Atomic Energy Agency (IAEA) / Government of the Republic of South Africa Safeguards Agreement
- Non-Proliferation attributes inherent in fuel design

Key Technical Licensing Challenges

- Lack of gas reactor technical licensing framework
- Fuel qualification and fabrication process licensing (South African Fuel)
- Source Term: Mechanistic or Deterministic
- Containment performance requirements
- Computer code V&V
- PRA - Uncertainties, Initiators and End States
- Regulatory treatment of non-safety systems
- Classification of SSC's
- Lack of technical expertise on gas reactors

Key Legal Licensing Challenges

- Price Anderson indemnity
- NRC operational fees
- Decommissioning trust funding
- Untested Part 52 process
- Potential number of exemptions

IRIS

International Reactor Innovative and Secure

M. D. Carelli

Westinghouse Science & Technology

ACRS Subcommittee Workshop on Advanced Reactors

June 4, 2001

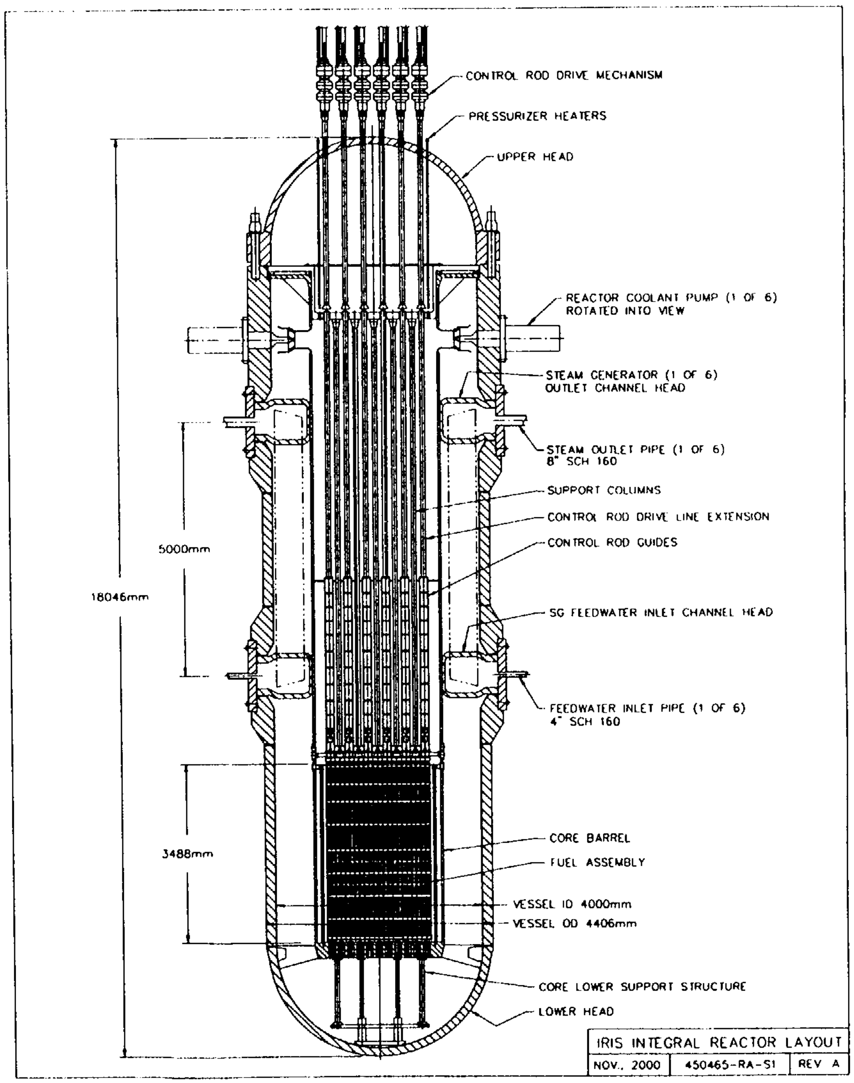
OUTLINE

- Overview
  - Team Partnership
  - Funding
  - Schedular Objectives
- Fuel Designs
- Configuration (Integral vessel, internal shield, steam generators)
- Enhanced Safety Approach (Safety by Design)
- Maintenance Optimization
- Issues
- Conclusions

OVERVIEW

IRIS is a Modular LWR, with Emphasis on Proliferation Resistance and Enhanced Safety

- Small-to-medium (100-300 MWe) power module
- Integral primary system
- 5- and 8-year straight burn core
- Utilizes LWR technology, newly engineered for improved performance
- Most accident initiators are prevented by design
- Potential to be cost competitive with other options
- Development, construction and deployment by international team
- First module projected deployment in 2010-2015



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IRIS AND GENERATION IV GOALS

Design feature	GOAL		
	Sustainable development	Safety and Reliability	Economics
Modular design		✓	✓
Long core life (single burn, no shuffling)	✓		✓
Extended fuel burnup	✓		✓
Integral primary circuit	✓	✓	✓
High degree of natural circulation		✓	
High pressure containment with inside-the-vessel heat removal		✓	✓
Optimized maintenance	✓	✓	✓

∴ **Attractive Commercial Market Entry**

Innata atomica vis Renovata intaque Ingenti necessitati Sine proliferatione succurrit
WESTINGHOUSE - MIT - POLIMI - UCB - ANSALDO - BECHTEL - BNFL - ININ - JAPC - MHI - NUCLEP - TIT - UNIPI
IRIS
ional Reactor Innovative &

IRIS Consortium Members Functions

Separate file -

IRIS Consortium Members for VG ACRS 60401.doc

FUNDING

DOE NERI	~ $1.6M over 3 years (9/99 - 8/02)		
Consortium Members	~ $4M	in	2000
	~ $8M	in	2001
	$10-12M anticipated in 2002		

IRIS SCHEDULAR OBJECTIVES

• Assess key technical & economic feasibilities (completed)	End 2000
• Perform conceptual design, preliminary cost estimate	End 2001
• Perform preliminary design	End 2002
• Pre-application submitted	?
• *Decision to proceed to commercialization*	*End 2002*
• Complete SAR	2005
• Obtain design certification	2007
• First-of-a-kind deployment	2010-2015

IRIS FUEL DESIGN OPTIONS

IRIS 5-YEAR DESIGN — *FIRST CORE*

- CURRENT FUEL TECHNOLOGY
- PROVIDES MINIMUM-RISK PATH FORWARD
- (DETAILED CORE DESIGN IN PROGRESS)

IRIS 8-YEAR DESIGN — *RELOADS*

- BOTH UO_2 and MOX MAY BE USED
- EMPHASIZES PROLIFERATION RESISTANCE
- (SCOPED INTERCHANGEABLE CORE DESIGN)

CONFIGURATION

335 MWe LAYOUT

Separate File -

335 MWe Layout LEC 450475-RA-S2

INTERNAL SHIELDS

- A "gift" of integral configuration
- Dose rate outside vessel surface as low as 10^{-6} mSv/h
- No restrictions to workers in containment
- Simplified decommissioning
- Vessel (minus fuel) acts as sarcophagus

ANSALDO PHOTO

HELICAL STEAM GENERATOR

- LWR and LMFBR experience
- Fabricated and tested
- Test confirmed performance (thermal, pressure losses, vibration, stability)
- 8 SGs practically identical to Ansaldo modules will be installed in IRIS

ENHANCED SAFETY APPROACH
(Safety by Design)

SAFETY PHILOSOPHY

- Generation II reactors cope with accidents via active means
- Generation III reactors cope with accidents via passive means
- Generation IV reactors (IRIS) emphasize prevention of accidents through "safety by design"

IRIS SAFETY BY DESIGN APPROACH

Exploit to the fullest what is offered by IRIS design characteristics (chiefly, integral configuration and long life core) to:

- Physically eliminate possibility for accident(s) to occur
- Lessen consequences
- Decrease probability of occurrence

IMPLEMENTATION OF IRIS SAFETY BY DESIGN

Separate file -

Implementation of IRIS Safety by Design 52401 ACRS & Cairo

AP600 CLASS IV ACCIDENTS AND IRIS RESOLUTION

	Accident	IRIS Safety by Design	IRIS Resolution
1.	Steam system piping failure (major)	Reduced probability Reduced consequences	Can be reclassified as Class III
2.	Feedwater system pipe break		
3.	Reactor coolant pump shaft seizure or locked rotor	Reduced consequences	Can be reclassified as Class III
4.	Reactor coolant pump shaft break		
5.	Spectrum of RCCA ejection accidents	Can be eliminated	Not applicable (with internal CRDMs)
6.	Steam generator tube rupture	Reduced consequences	Can be reclassified as Class III
7.	Large LOCAs	Eliminated	Not applicable
8.	Design basis fuel handling accidents	Reduced probability	Still Class IV 1/3-1/5 lower probability

IRIS CONTAINMENT

- It performs containment function

<u>plus</u>

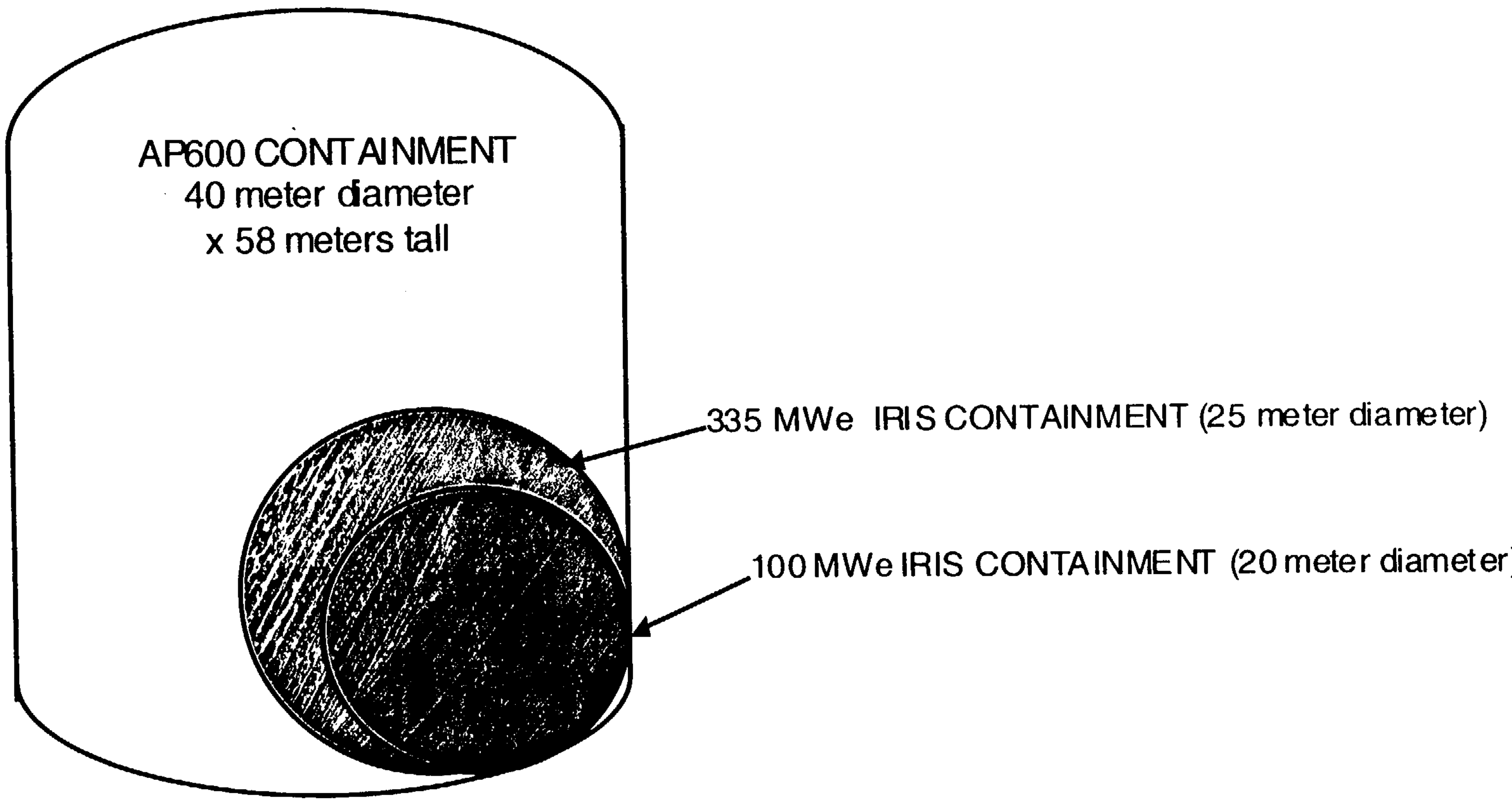
- In concert with integral vessel, it practically eliminates LOCAs as a safety concern

<u>On first principles</u>

Pressure differential (driving force through rupture) is lower in IRIS because

- Containment pressure higher (lower volume, higher allowable pressure)
- Vessel pressure lower (internal heat removal)

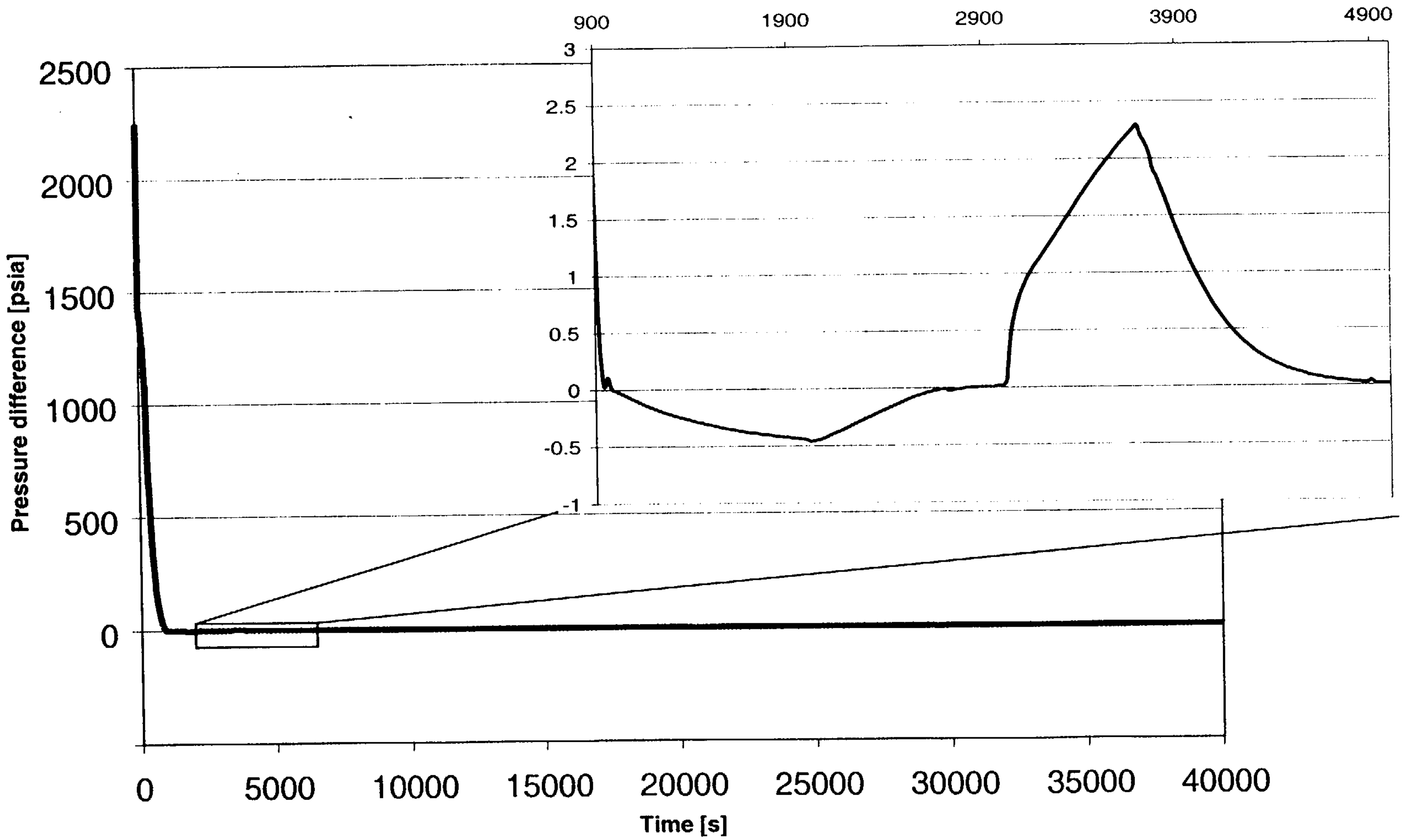
AP600/IRIS Containment Size Comparison



ANALYSES PERFORMED

- **Break size: 1, 2, 4"**
- **Elevation: Bottom of vessel, above core (inside and outside cavity), 12.5 m above bottom**
- **No water makeup or safety injection**
- **Three codes provided consistent results**
  - Proprietary (POLIMI)
  - GOTHIC (Westinghouse)
  - FUMO (Univ. Pisa)

REACTOR VESSEL/CONTAINMENT PRESSURE DIFFERENTIAL EQUALIZES QUICKLY



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Viewgraph 24

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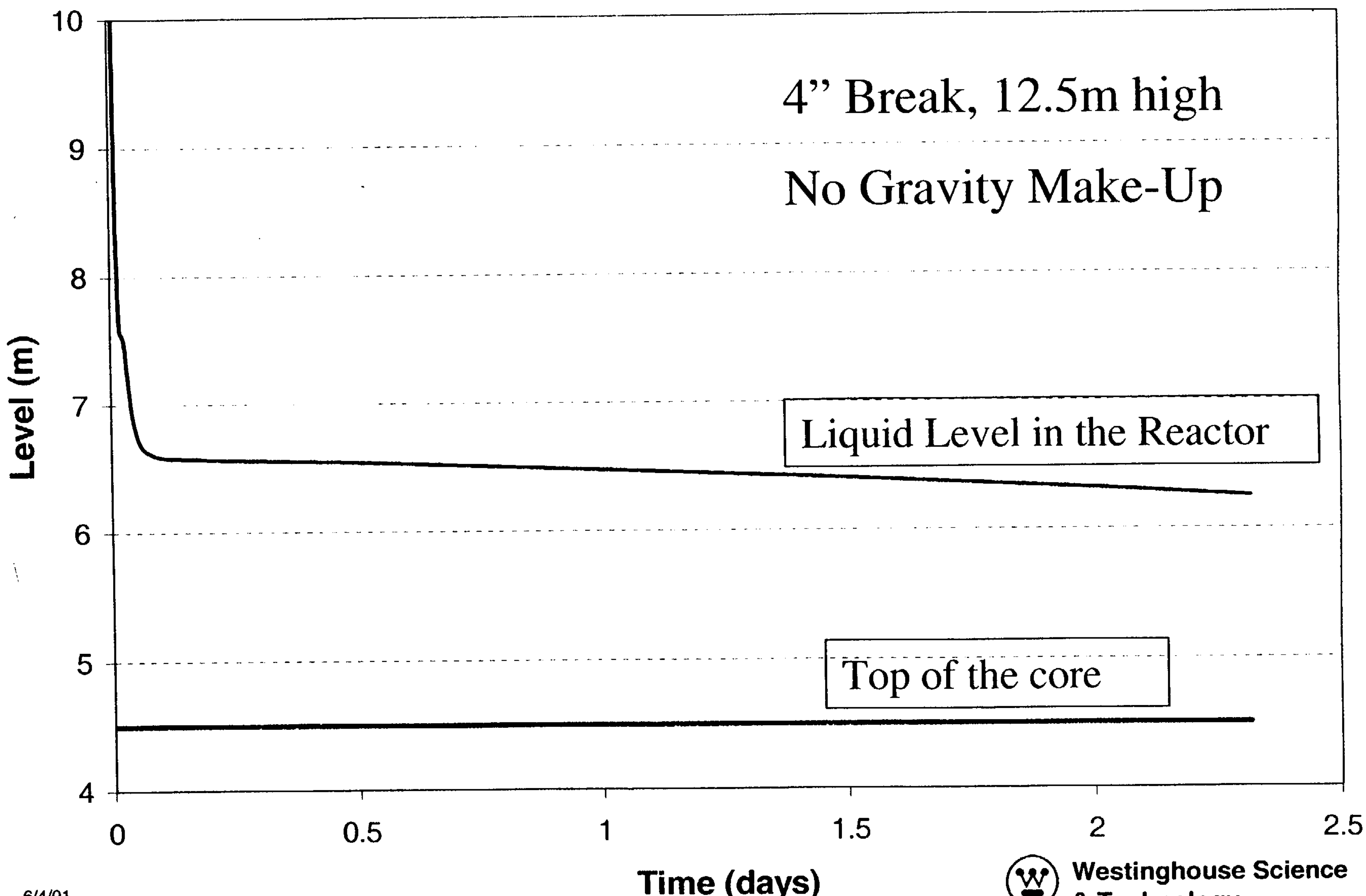
CORE STILL UNDER 2 METERS OF WATER AFTER 2 DAYS



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A LICENSING CHALLENGE

"…..simultaneous loss-of-coolant accident, loss of residual heat removal system, and loss of emergency core cooling…..PMBR can meet that challenge…..but "you can't assume that sequence for any LWR" even advanced units….."

Nucleonics Week 5/10/01 Pg. 10

IRIS CAN MEET THAT CHALLENGE

• Loss of coolant accident	Safety by design
• Loss of residual heat removal system	Three independent diverse systems
• Loss of emergency core cooling	Not needed (gravity makeup available anyway)

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MAINTENANCE OPTIMIZATION

GOAL

- Perform maintenance shutdowns no sooner than 48 months

SURVEILLANCE STRATEGY

"defer if practical, perform on-line when possible, and eliminate by design where necessary"

Design where necessary:

- **Utilize existing components**
- **Utilize existing technologies**
- **Request rule changes**
- **Develop new components/systems**
- **Develop new technologies**

Direction of increasing cost, design effort, and risk

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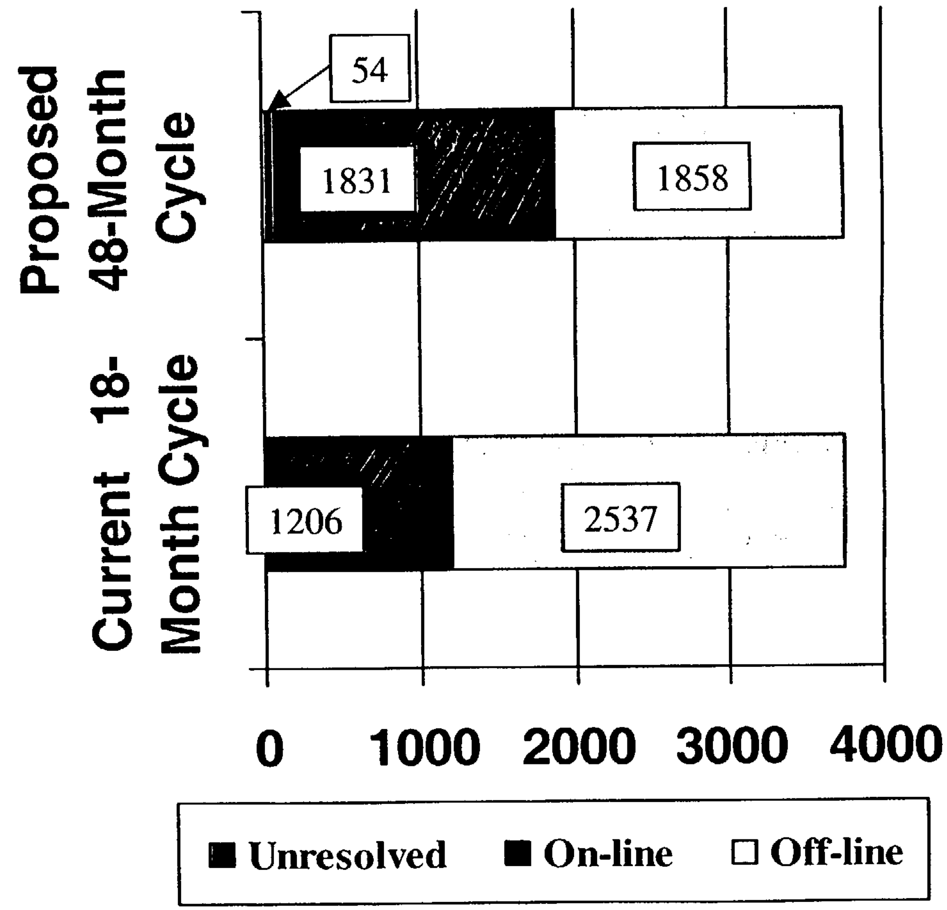
THE BOTTOM LINE

- IRIS must utilize components and systems which are either *accessible on-line* for maintenance or *do not require any off-line* maintenance for the duration of the operating cycle
- IRIS must utilize *high reliability* components and systems to minimize the probability of failure leading to unplanned down-time during the operating cycle

EXTENDED FUEL CYCLE PROJECT

- Study completed in 1996 investigated extending PWR to 48 month cycle
- Recategorized all off-line maintenance as either:
  - Defer to 48 months
  - Perform on-line
  - Unresolved

PWR Surveillance Program Comparison



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ISSUES

DEVELOPMENT APPROACH

- No need for prototype since no major technology development is required

- First-of-a-kind IRIS module can be deployed in 2010 or soon after

- Future improvements can be implemented in later modules (Nth-of-a-kind)

LICENSING CHALLENGES AND OPPORTUNITIES VS. GEN II REACTORS

- First core fuel well within current state of the art
- Reload, higher enrichment fuel (post 2015) handled through licensing extension
- IRIS does have containment which in addition to its classic function is thermal-hydraulically coupled with integral vessel to choke small/medium LOCAs
- Safety by design approach eliminates some accident scenarios and significantly diminishes consequences of others. Simplification and streamlining possible.
- Risk informed regulation will be coupled with safety by design to show lower accidents and damage probabilities
- How can we translate IRIS improved safety into licensing opportunity, e.g., site requirements relaxation?
- Are regulatory changes necessary to accommodate extended maintenance?
- Multiple modules plants with common functions, e.g., control room

IRIS APPROACH TO LICENSING, CONSTRUCTION AND OPERATION VS. GEN II REACTORS

- *Licensing*
  - No unique major changes identified at this time
  - Testing to confirm IRIS unique traits (safety by design, integral components, maintenance optimizations, inspections)
- *Construction*
  - Modular fabrication and assembly
  - Use of advanced EPC tool sets (Bechtel)
  - Multiple, parallel suppliers
  - Staggered modules construction
- *Operation*
  - Extended cycle length straight burn
  - Maintenance shutdown intervals no shorter than 48 months
  - Refueling shutdowns every 5 to 10 years
  - Reduced number of plant personnel
  - Multiple modules operation

DO SCHEDULES SUPPORT PLANNED LICENSE APPLICATIONS/DEPLOYMENT?

Achieving 2007 design certification requires:

- Lead testing (safety by design) be initiated in 2002
- IRIS Consortium members decision by end 2002 to pursue commercial effort
- Continuous NRC interaction beginning late 2001/early 2002

Achieving early deployment (2010 or soon after) requires US generator interested by 2005

SUMMARY AND CONCLUSIONS

- IRIS specifically designed to address Gen IV requirements
- Modularity and flexibility address utility needs
- Enhanced safety through safety by design and simplicity
- IRIS is based on proven LWR technology, newly engineered for improved performance
- Testing program needs to start in 2002 on selected high priority tests. Early interaction with NRC and ACRS will be extremely beneficial.

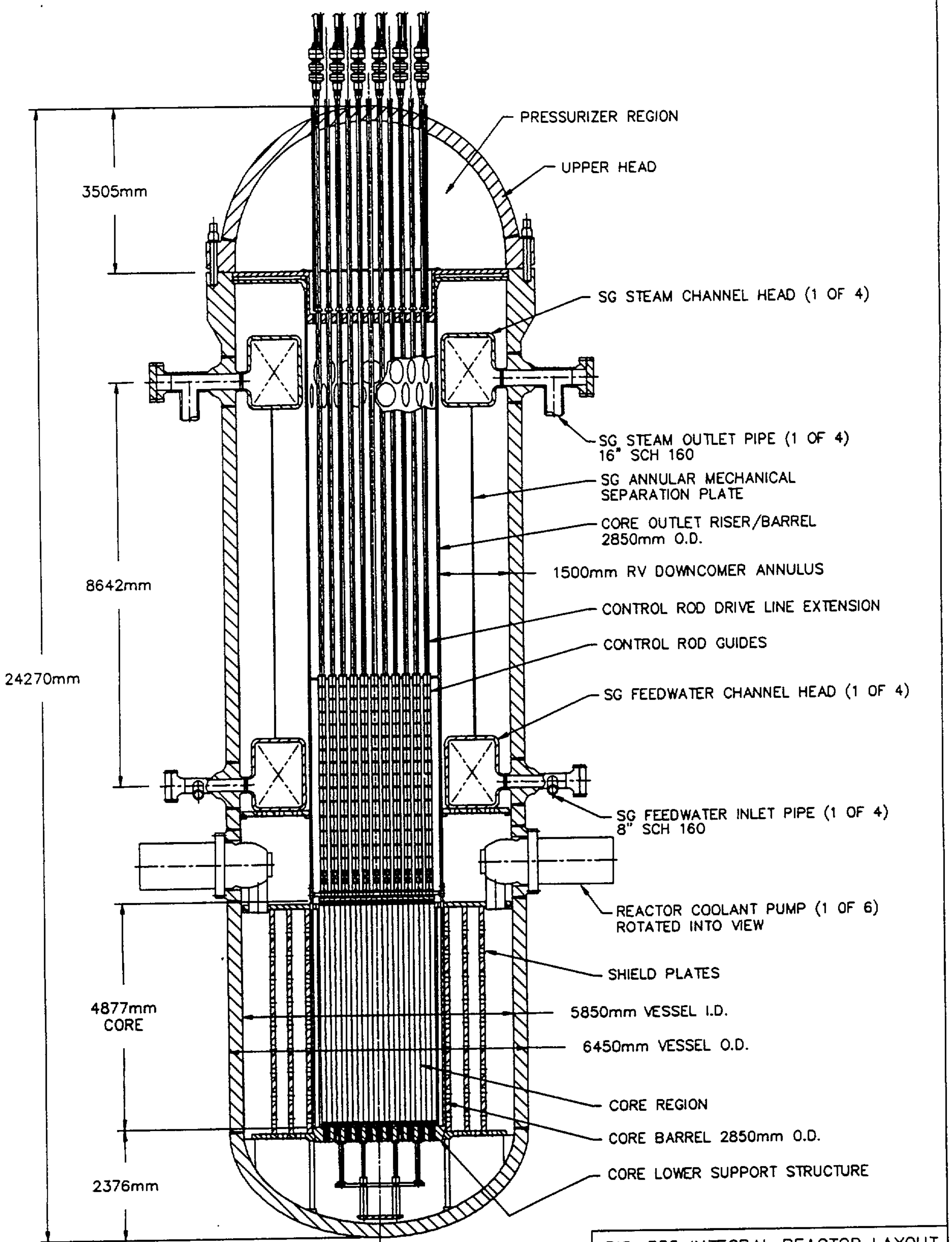
IMPLEMENTATION OF IRIS SAFETY BY DESIGN

Design Characteristic	Safety Implication	Related Accident	Disposition
Integral reactor configuration	No external loop piping	Large LOCAs	Eliminated
Tall vessel with elevated steam generators	Can accommodate internal control rod drives	Reactivity insertion due to control rod ejection	Can be eliminated
	High degree of natural circulation	LOFAs (e.g., pump seizure or shaft break)	Either eliminated (full natural circulation) or mitigated consequences (high partial natural circulation)
Low pressure drop flow path and multiple RCPs	N-1 pumps keep core flow above DNB limit, no core damage occurs		
High pressure steam generator system	Primary system cannot over-pressure secondary system	SGTR	Automatic isolation, accident terminates quickly
	No SG safety valves required	Steam and feed line breaks	Reduced probability Reduced consequences
Once through SG design	Low water inventory		
Long life core	No partial refueling	Refueling accidents	Reduced probability
Large water inventory inside vessel	Slows transient evolution Helps to keep core covered	Small-medium LOCAs	Core remains covered with no safety injection
Reduced size, higher pressure containment	Reduced driving force through primary opening		
Inside the vessel heat removal			

IRIS Consortium Members

Team Member	Function			Scope
	Engineering	Supplier	Development	
Westinghouse Electric LLC, USA	*		*	Overall coordination, leadership and interfacing, licensing
Polytechnic Institute of Milan, Italy (POLIMI)			*	Core design, in-vessel thermal hydraulics, steam generators, containment
Massachusetts Institute of Technology, USA (MIT)			*	Core thermal hydraulics, novel fuel rod geometries, safety, maintenance
University of California at Berkeley, USA (UCB)			*	Core neutronics design
Japan Atomic Power Company, Japan (JAPC)	*		*	Maintenance, utility feedback
Mitsubishi Heavy Industries, Japan (MHI)	*	*	*	Steam generators, modularization
British Nuclear Fuels plc, UK (BNFL)	*	*	*	Fuel and fuel cycle, economic evaluation
Tokyo Institute of Technology, Japan (TIT)			*	Novel fuel rod geometries, detailed 3D T&H subchannel characterization, PSA
Bechtel Power Corp., USA (Bechtel)	*	*	*	Balance of plant, cost evaluation, construction
University of Pisa, Italy (UNIPI)			*	Containment analyses, transient analyses
Ansaldo, Italy	*	*	*	Steam generators, reactor systems
National Institute Nuclear Studies, Mexico (ININ)			*	Core neutronics
NUCLEP, Brazil	*	*		Containment, vessel, pressurizer
ENSA, Spain	*	*		Reactor internals, steam generators, vessel
Oak Ridge National Laboratory, USA (ORNL)	*		*	Core analyses, safety, cost evaluation, testing
Nuclear Energy Commission, Brazil (CNEN)	*		*	Transient, structural analyses, testing
<u>Associates</u>				
University of Tennessee, USA			*	Modularization, transportability
Ohio State University, USA			*	Novel In-Core Power Monitor

335 MWe Vessel Layout



IRIS-335 INTEGRAL REACTOR LAYOUT		
APRIL, 2001	450475-RA-S4	REV. A

ACRS WORKSHOP
Regulatory Challenges for Future Nuclear Power Plants

Gas Turbine - Modular Helium Reactor

4 - 5 June 2001

Laurence L Parme
Manager: Safety & Licensing
Power Reactor Division

GENERAL ATOMICS

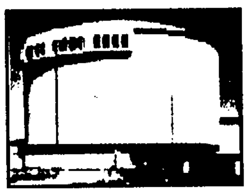
Presentation Outline

- Background and design description
- Key safety features
- Licensing approach
- Design status and deployment schedule
- Conclusions

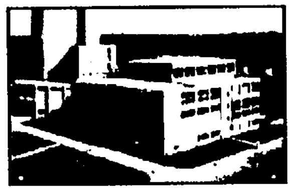
U.S. AND EUROPEAN TECHNOLOGY BASES FOR MODULAR HIGH TEMPERATURE REACTORS

BROAD FOUNDATION OF HELIUM REACTOR TECHNOLOGY

EXPERIMENTAL REACTORS

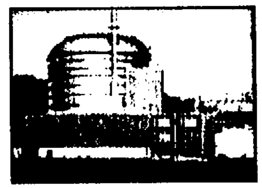


DRAGON
(U.K.)
1963 - 76

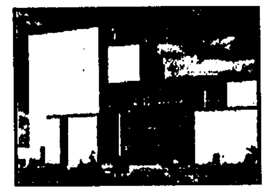


AVR
(FRG)
1967 - 1988

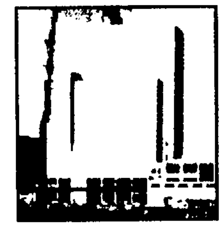
DEMONSTRATION OF
BASIC HTGR TECHNOLOGY



PEACH BOTTOM 1
(U.S.A.)
1967 - 1974



FORT ST. VRAIN
(U.S.A.)
1976 - 1989

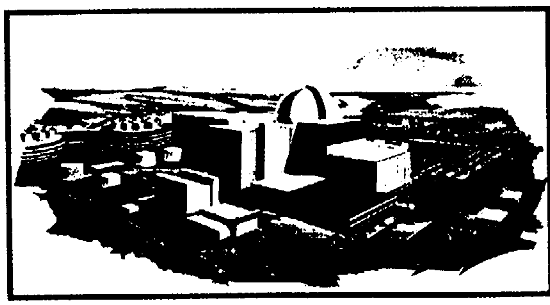


THTR
(FRG)
1986 - 1989

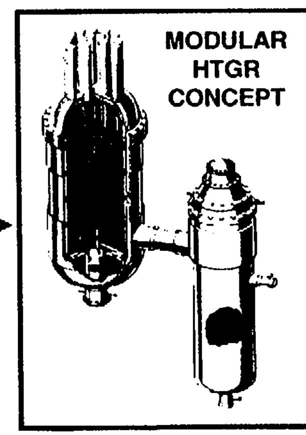
HTGR TECHNOLOGY PROGRAM

- MATERIALS
- COMPONENTS
- FUEL
- CORE
- PLANT TECHNOLOGY

LARGE HTGR PLANTS

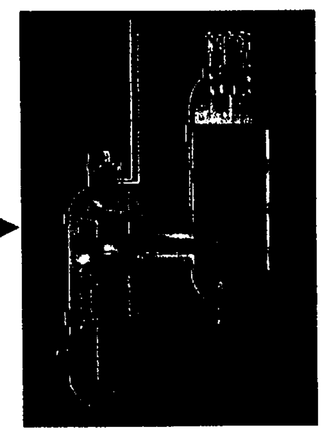


MHTGR



Steam Cycle

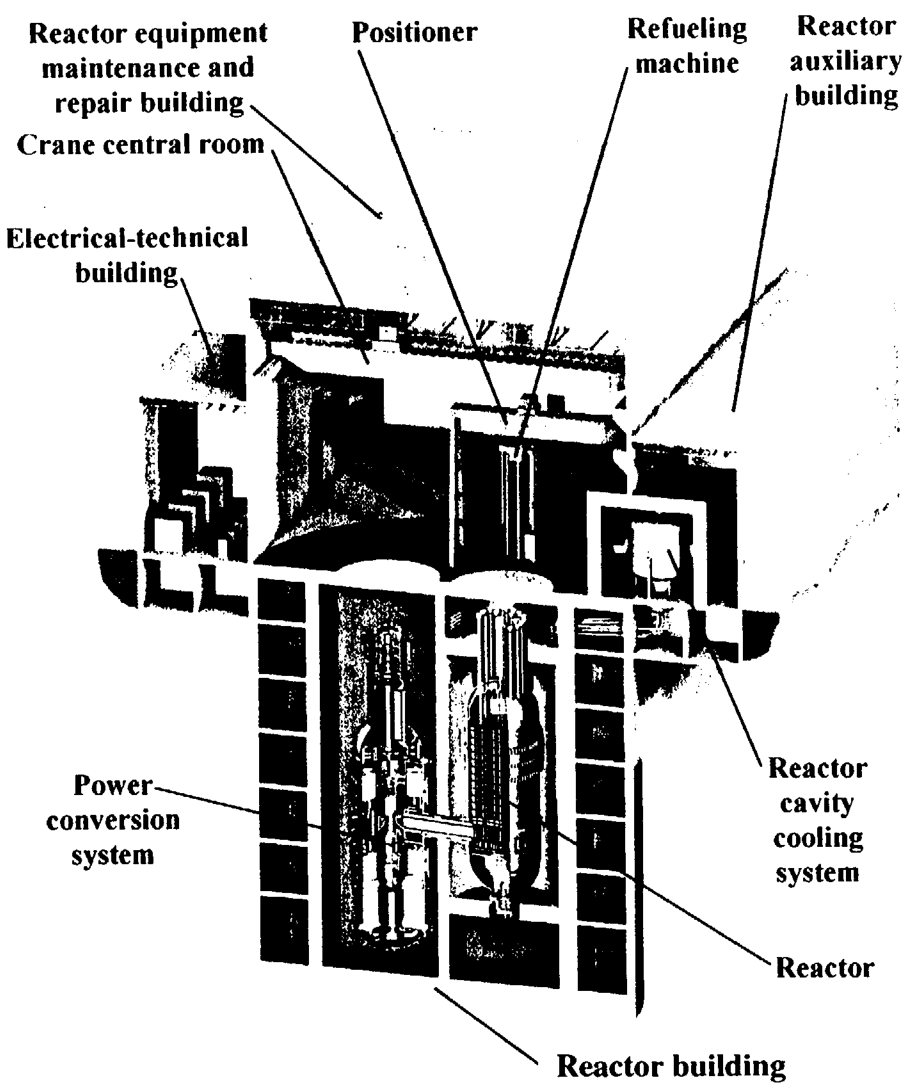
GT-MHR



Gas Turbine Cycle

GENERAL ATOMICS

3D Arrangement of Plant

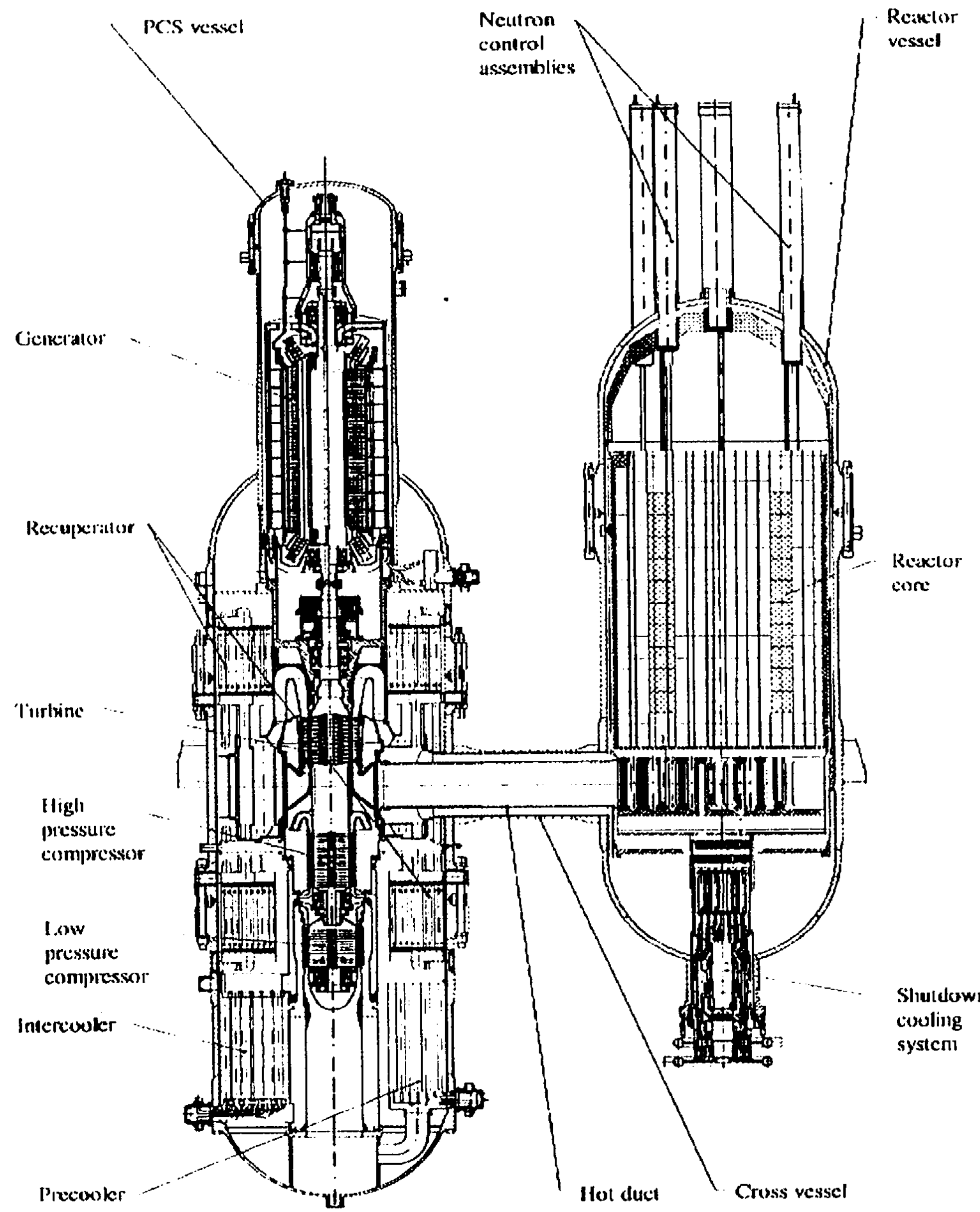


- 600 MW(t) - 285 MW(e)
- Power conversion system integrated in single vessel
- Vented, below grade reactor building
- Continuously operating, natural circulating, air cooled reactor cavity cooling

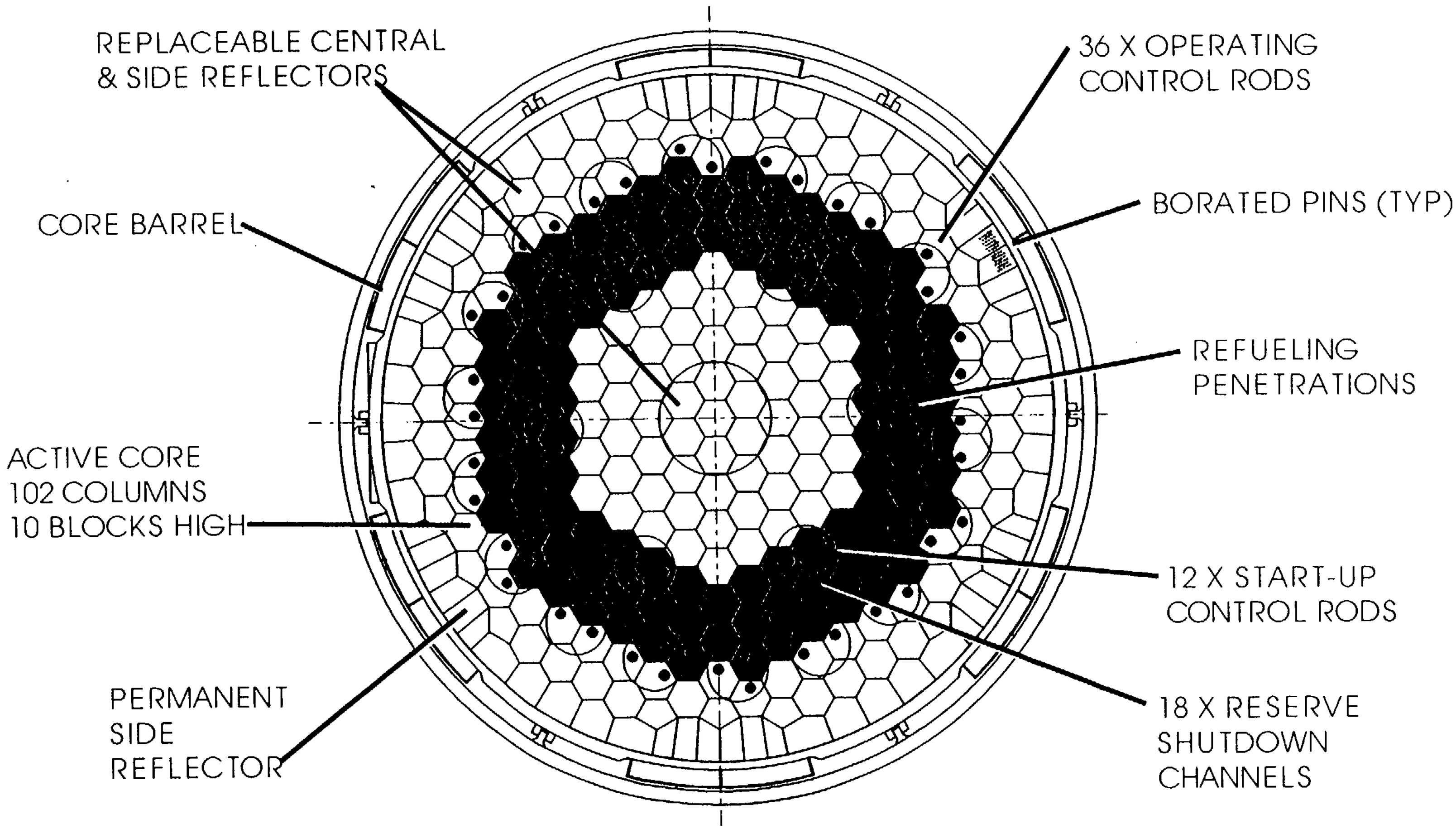
GENERAL ATOMICS

GT-MHR COMBINES MELTDOWN-PROOF ADVANCED REACTOR AND GAS TURBINE BASED POWER CONVERSION SYSTEM

GENERAL ATOMICS



ANNULAR REACTOR CORE LIMITS FUEL TEMPERATURE DURING ACCIDENTS



. . . ANNULAR CORE USES EXISTING TECHNOLOGY

GENERAL ATOMICS

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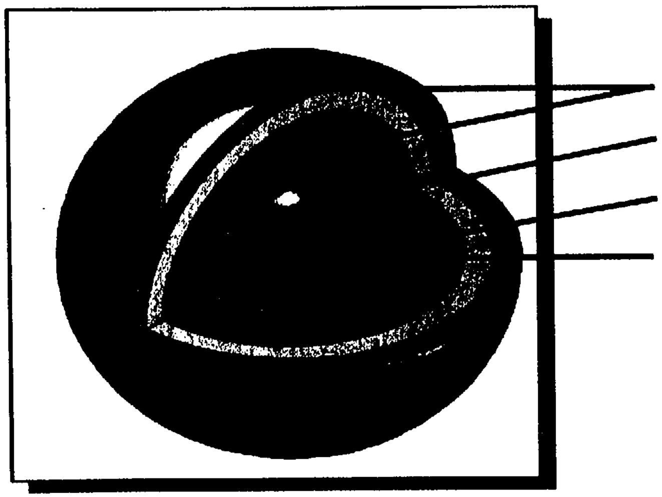
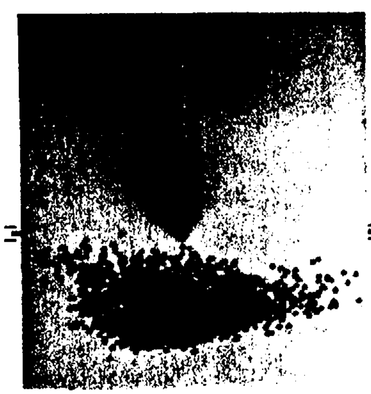
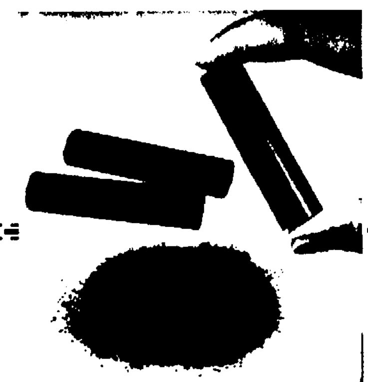
CERAMIC COATED FUEL IS KEY TO GT-MHR SAFETY AND ECONOMICS



TRISO Coated fuel particles (left) are formed into fuel rods (center) and inserted into graphite fuel elements (right).



PARTICLES

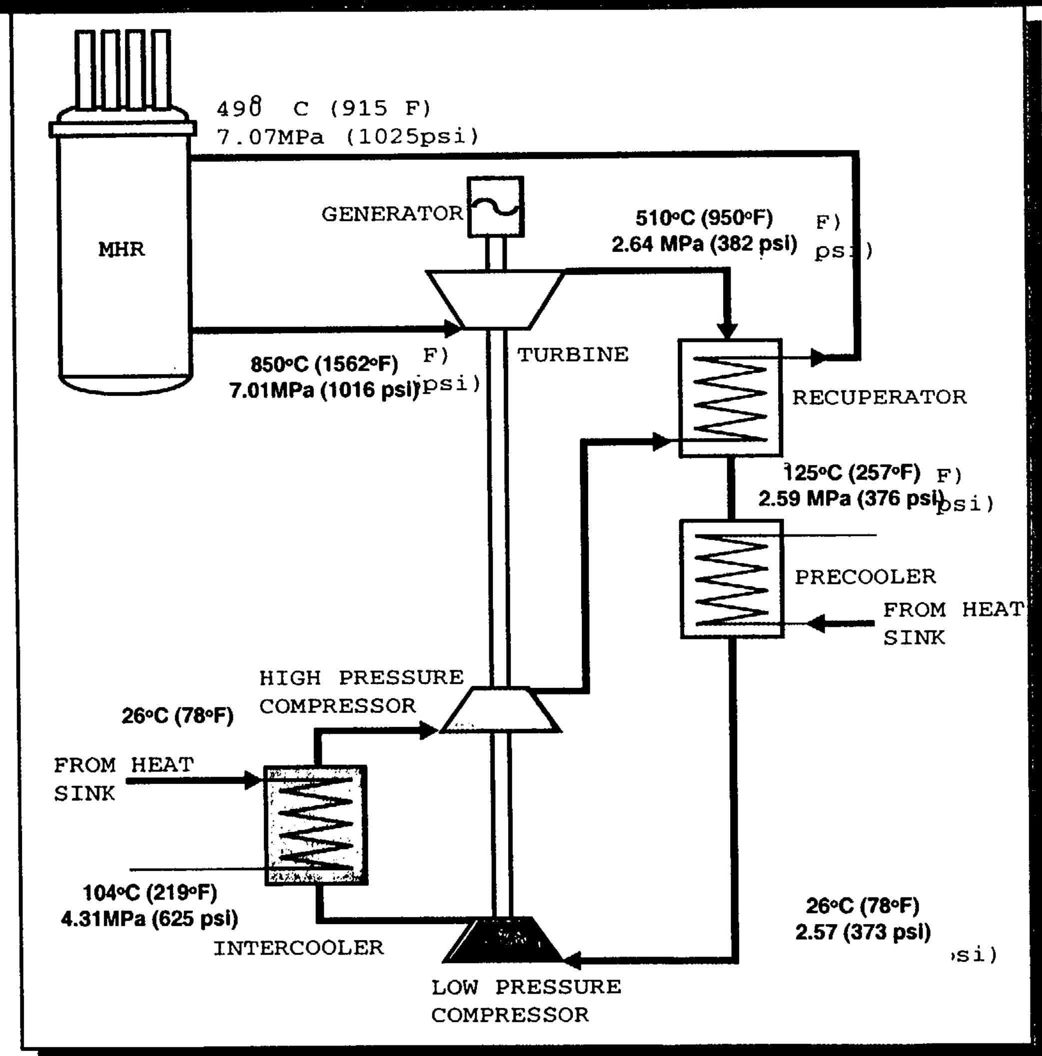


COMPACTS

FUEL ELEMENTS

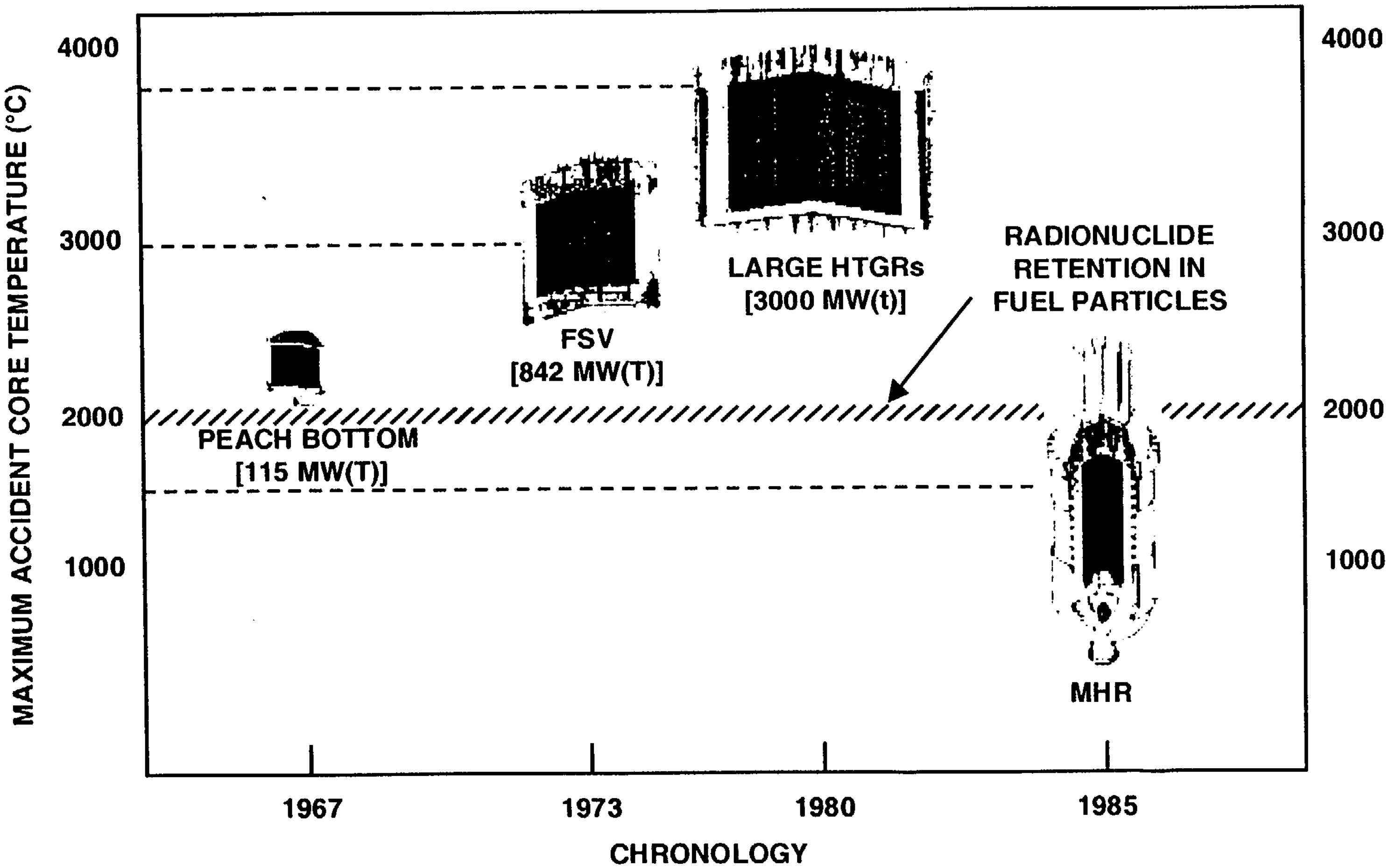
GENERAL ATOMICS

GT-MHR FLOW SCHEMATIC



GENERAL ATOMICS

MODULAR HELIUM REACTOR REPRESENTS A FUNDAMENTAL CHANGE IN REACTOR DESIGN AND SAFETY PHILOSOPHY

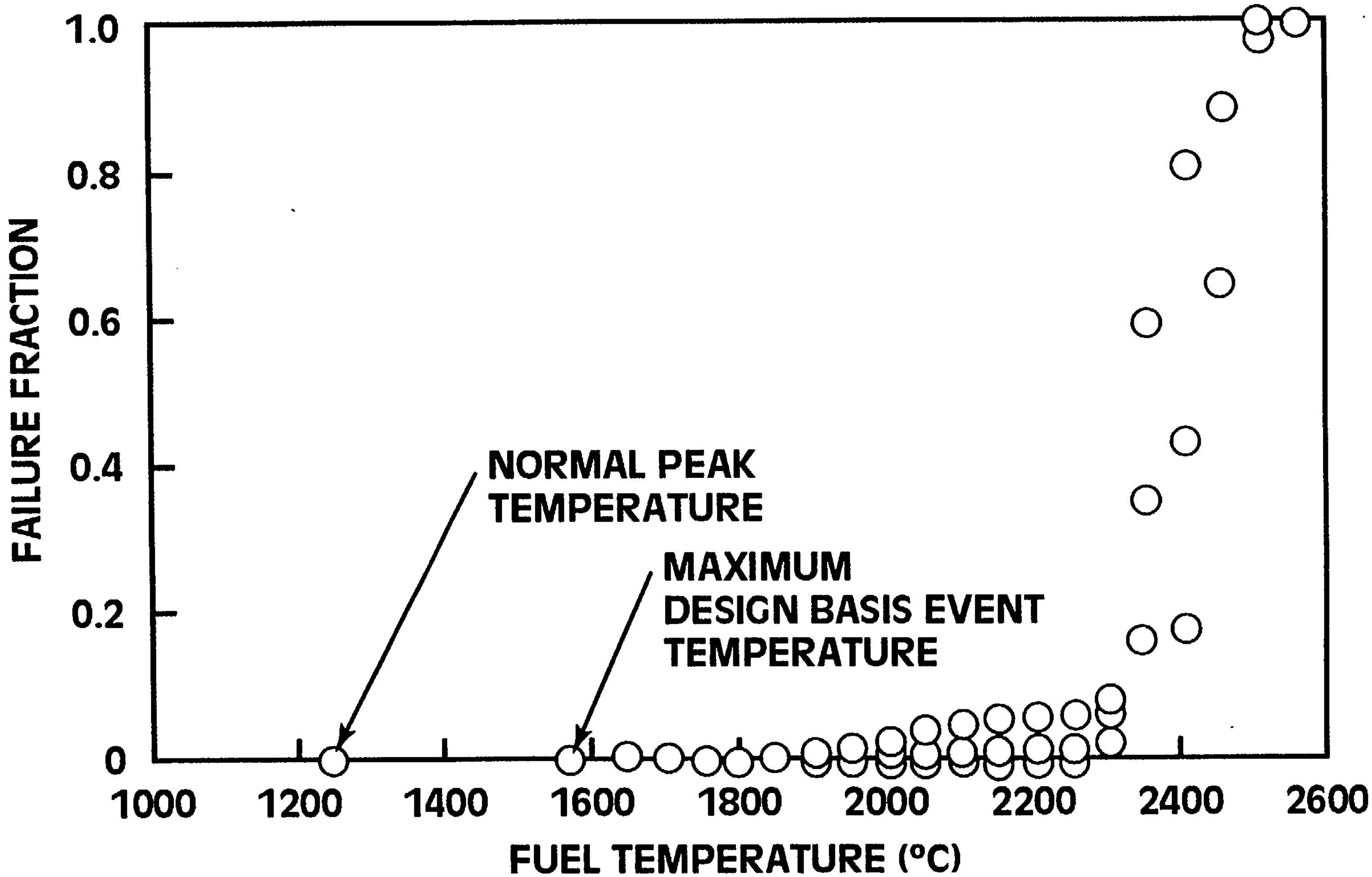


...SIZED AND CONFIGURED TO TOLERATE EVEN A SEVERE ACCIDENT

GENERAL ATOMICS

I-222(1)
1-12-96

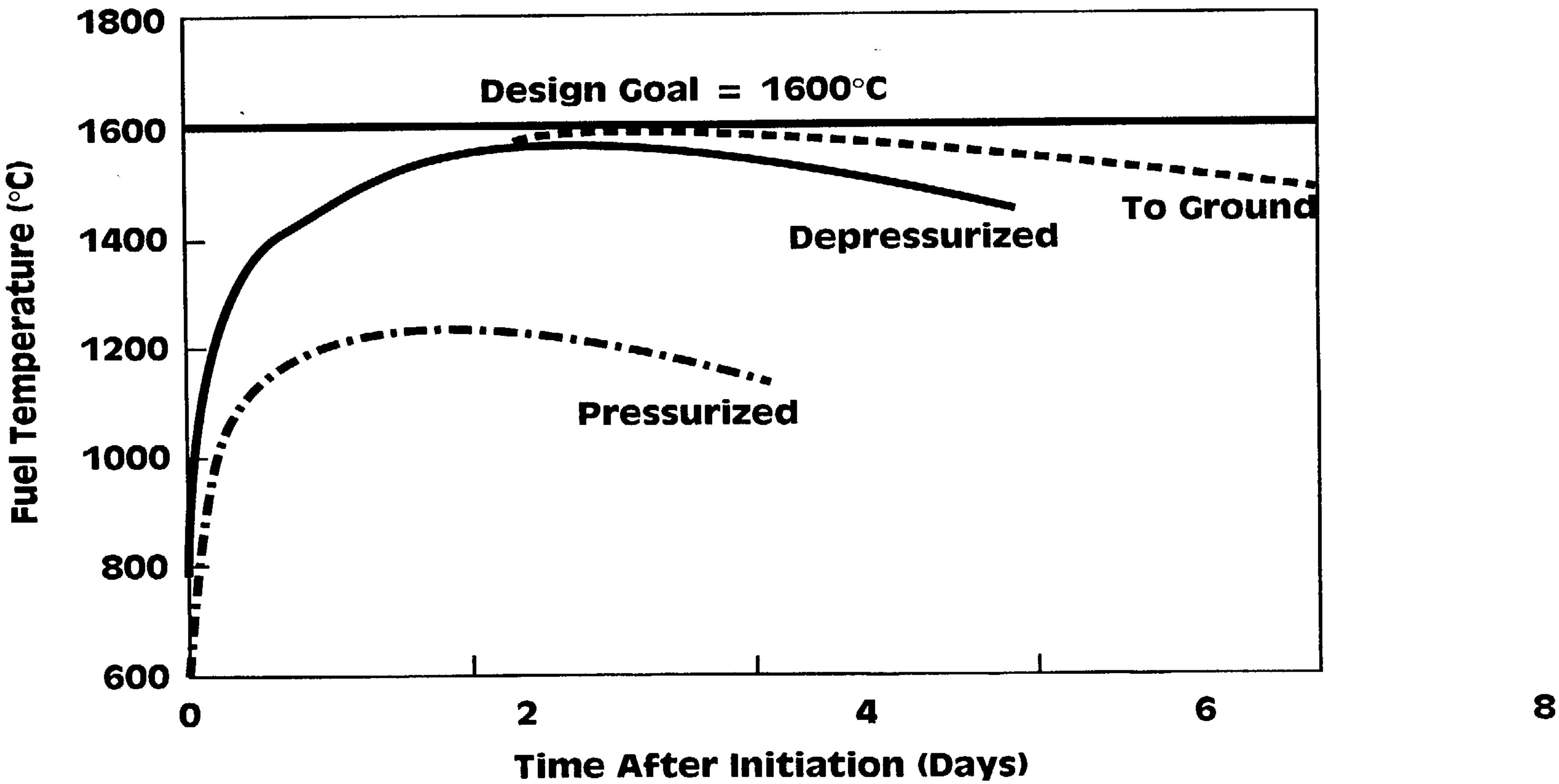
COATED PARTICLES STABLE TO BEYOND MAXIMUM ACCIDENT TEMPERATURES



GENERAL ATOMICS

L-266(1)
7-28-94
W-9

FUEL TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING LOSS OF COOLING EVENTS



. . . PASSIVE DESIGN FEATURES ENSURE FUEL REMAINS BELOW 1600°C

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L-340(3)
11-16-94

PASSIVE SAFETY BY DESIGN

- Fission Products Retained in Coated Particles
  - *High temperature stability materials*
  - *Refractory coated fuel*
  - *Graphite moderator*

- Worst case fuel temperature limited by design features
  - *Low power density*
  - *Low thermal rating per module*
  - *Annular Core*
  - *Passive heat removal* *....CORE CAN'T MELT*

- Core Shuts Down Without Rod Motion

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Licensing Approach Builds on Mid-80s Submittal to NRC

- The DOE MHTGR program in the mid-80's utilized a "clean sheet of paper" integrated approach to the conceptual design
  - utilized participant experience in PRA's of HTGRs
  - approach underwent a preapplication review by the NRC/ACRS

- Provided risk-informed MHTGR Licensing Bases
  - Top Level Regulatory Criteria
  - Licensing Bases Events
  - Equipment Safety Classification
  - Safety Related Design Conditions
  - Basis design criteria

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Bases for Top Level Regulatory Criteria

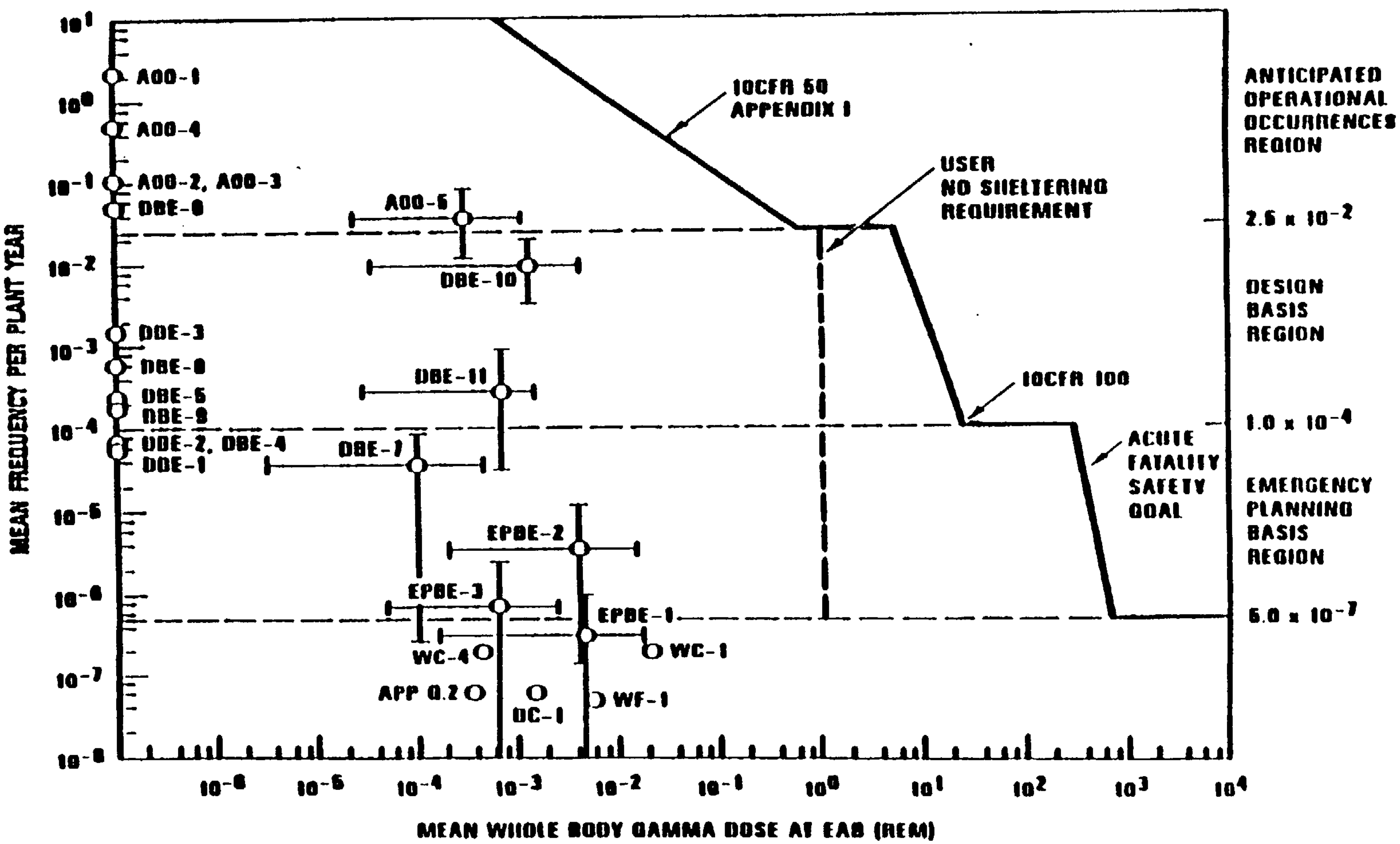
- Direct statements of acceptable consequences or risks to the public or the environment
- Quantifiable statements
- Independent of plant design

- Top Level criteria include
  - 51FR130 individual acute and latent fatality risks
    $5x10^{-7}$/yr and $2x10^{-6}$/yr, respectively
  - 10CFR50 Appendix I annualized offsite dose guidelines
    5 mrem/yr whole body
  - 10CFR100 accident offsite doses
    25 rem whole body and 300 rem thyroid
  - EPA-520/1-75-001 protective action guideline doses
    1 rem whole body and 5 rem thyroid

Licensing Basis Events

- Off-normal or accident events used for demonstrating design compliance with the Top Level Regulatory Criteria
- Collectively, analyzed in PRAs for demonstrating compliance with the 51FR130 safety goals
- Encompass following event categories
  - Anticipated Operational Occurrences
  - Design Basis Events
  - Emergency Planning Basis Events

Ranges of Top Level Regulatory Criteria and MHTGR Licensing Basis Events



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Equipment Safety Classification

- Safety related systems, structures, and components (SSC) are those performing required functions to meet 10CFR100 doses for DBEs

Retain Radionuclides in Fuel

- Control Heat Generation
- Remove Core Heat
- Control Chemical Attack

MHTGR functions for 10CFR100 focus on retention within fuel particles

Licensing Bases Application to GT-MHR

- The above process is generic and should be directly applicable to the GT-MHR
- Prior application to the MHTGR did not reveal a large sensitivity to the power conversion system
- GT-MHR would be expected to have some different LBEs and therefore some differences in safety related SSC
  - potential for new initiating events with rotating equipment in primary system
  - potential for different consequences with higher core rating
  - LBEs involving water ingress very unlikely---no SGs

GT-MHR NOW BEING DEVELOPED IN INTERNATIONAL PROGRAM

- In Russia under joint US/RF agreement for destruction of surplus weapons Plutonium

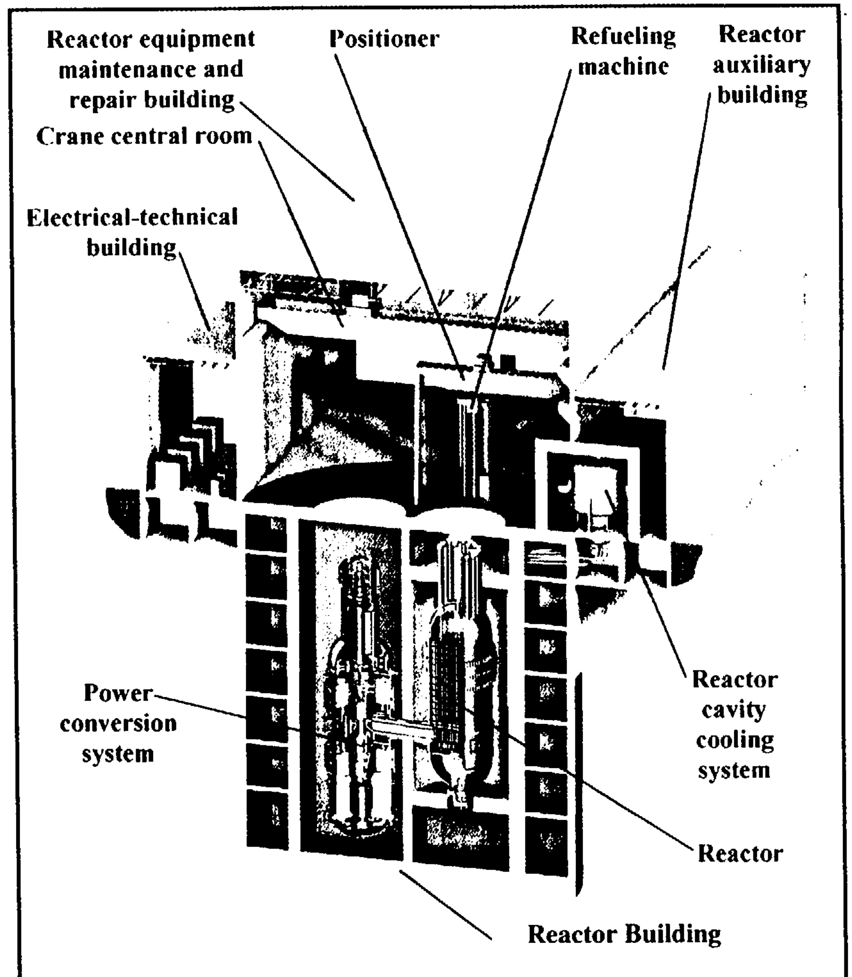
- Sponsored jointly by US (DOE) and RF (Minatom); supported by Japan and EU

- Conceptual design completed; preliminary design complete early 2002

INTERNATIONAL GT-MHR PROGRAM

- Design, construct and operate a prototype GT-MHR module by 2009 at Tomsk, Russia
- Design, construct, and license a GT-MHR Pu fuel fabrication facility in Russia
- Operate first 4-module GT-MHR by 2015 with a 250 kg plutonium/ year/module disposition rate

....Fuel contains Pu only
......No fertile component



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COMMERCIALIZATION PROGRAM

COMMERCIAL PROGRAM	=	INTERNATIONAL PROGRAM TECHNOLOGY	+	URANIUM FUEL RATHER THAN Pu FUEL

Plant construction can start in 5 years

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LIMITED ENGINEERING WORK REQUIRED

COMMERCIAL PLANT ENGINEERING

- Define Commercial Plant Requirements
- Safety and Licensing
- Transfer International Program Technology
- Performance Assessments
- Prepare Incremental Design Items

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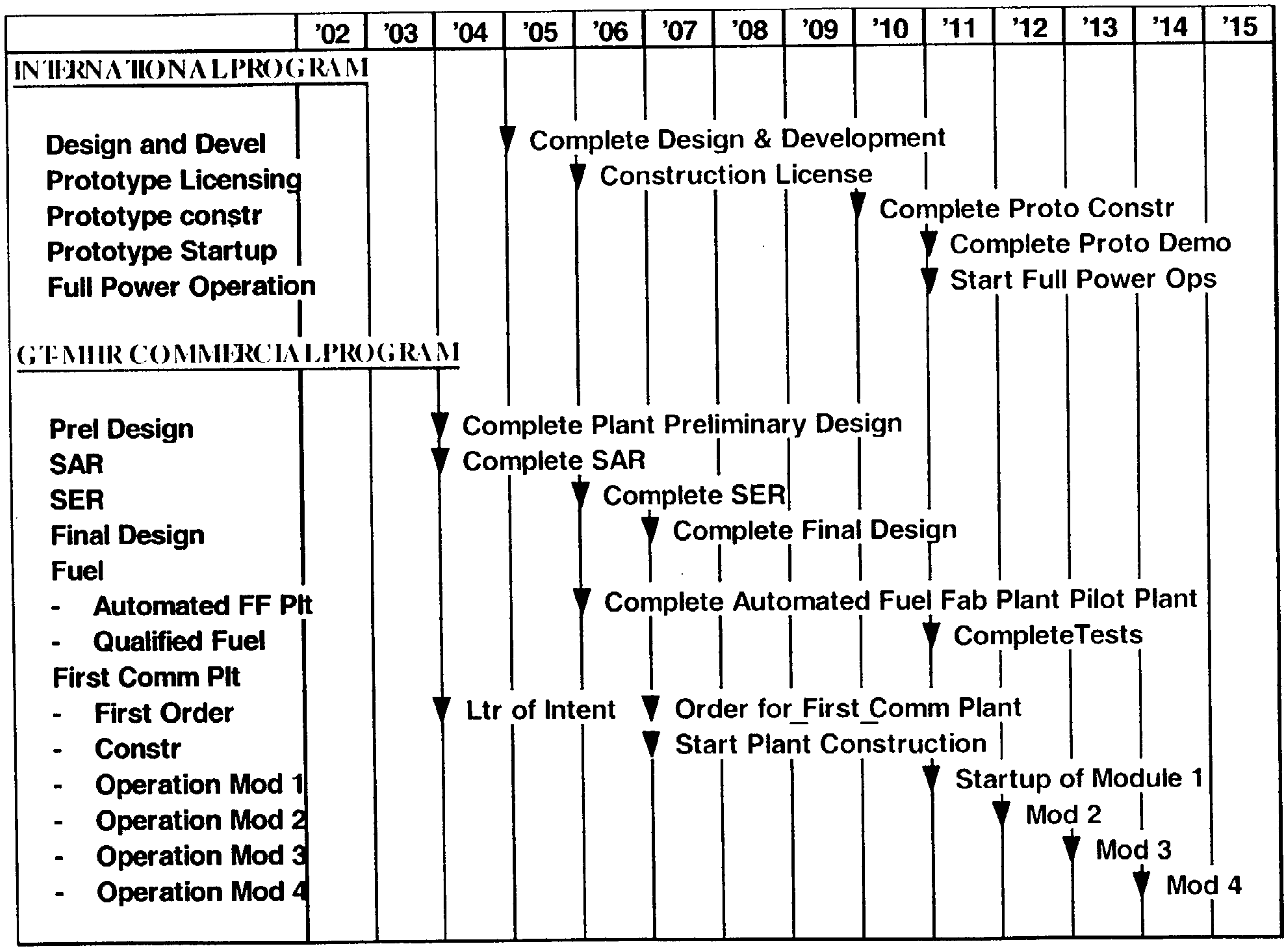
COMMERCIAL PROGRAM FOLLOWS INTERNATIONAL PROGRAM



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SUMMARY

- GT-MHR
  - Rooted in decades of international HTGR technology
  - Builds on 1980's (MHTGR) experience
- Optimization of inherent gas-reactor features provides
  - High thermal efficiency
  - Easily understood, assured safety
- International program facilitates near term deployment

GE Nuclear Energy

ESBWR Program and Regulatory Challenges

Atam Rao
GE Nuclear Energy, USA

ACRS Workshop – Regulatory Challenges for Future Nuclear Plants
June 4/5, 2001, Rockville, Maryland

Outline

- **Design is based on SBWR and ABWR components**

  Natural Circulation, ABWR Fuel, Vessel, CRD – just less

  Passive safety systems – based on NRC reviewed SBWR

  Optimized buildings/structures – economics/construction

  8 year international design and technology program

  Goal was to improve performance/safety and economics

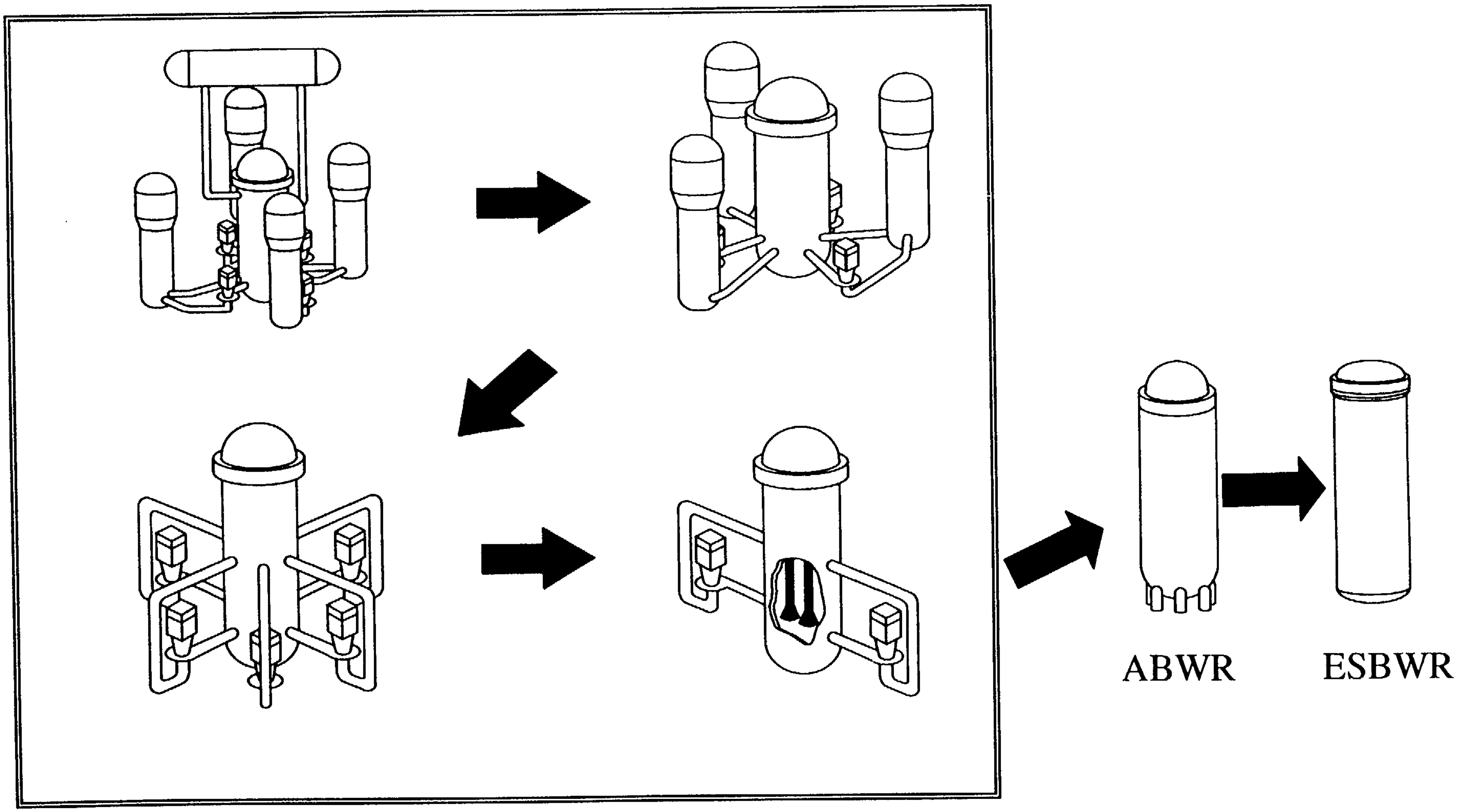
- **Regulatory Issues**

  How much use can be made of SBWR review by NRC?

  Extensive new testing completed - Is it enough?

  Is the regulatory hurdle too high for new plants?

Evolution of the BWR Reactor Design



Evolution Towards Simplicity

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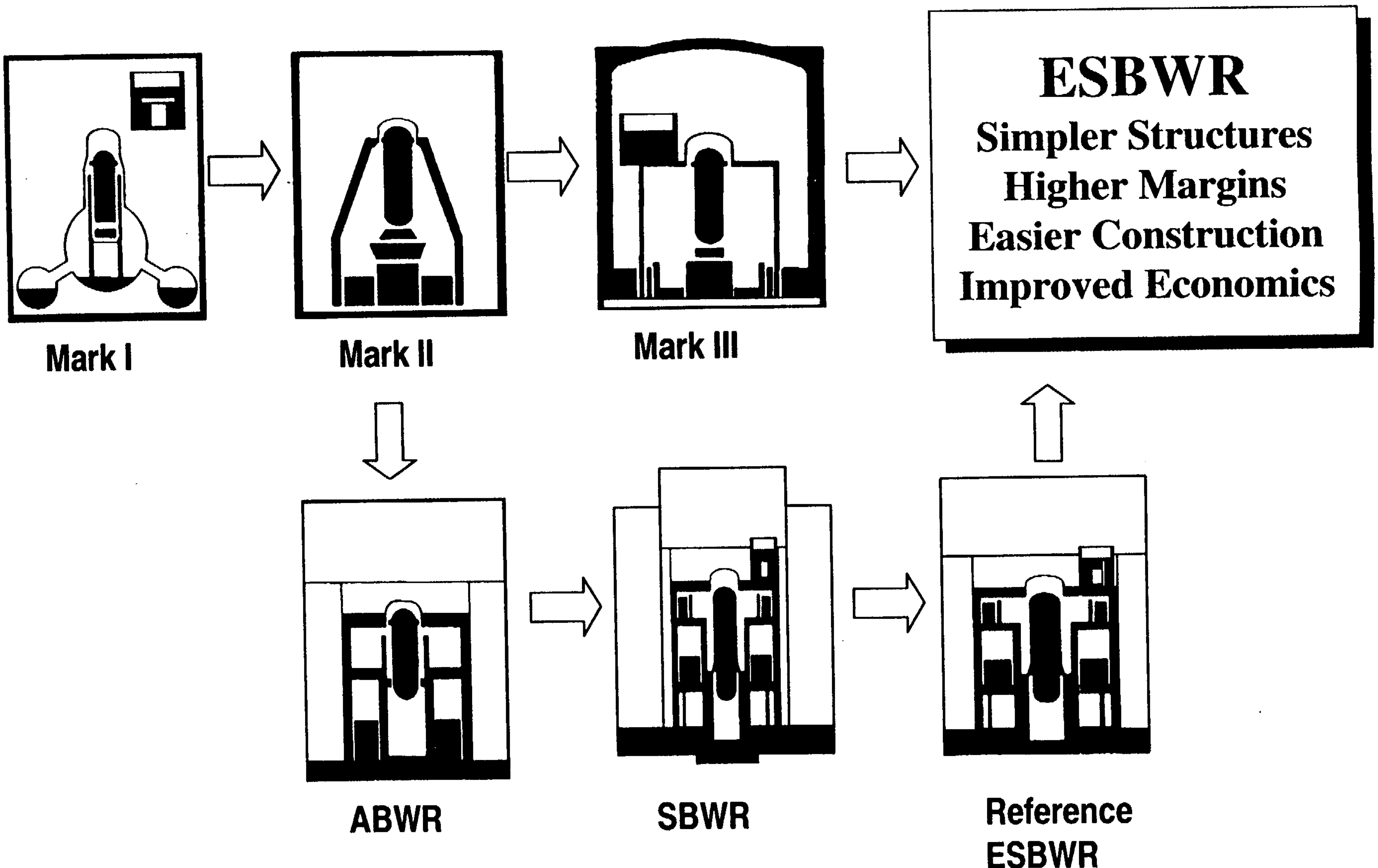
Evolution of BWR Containments



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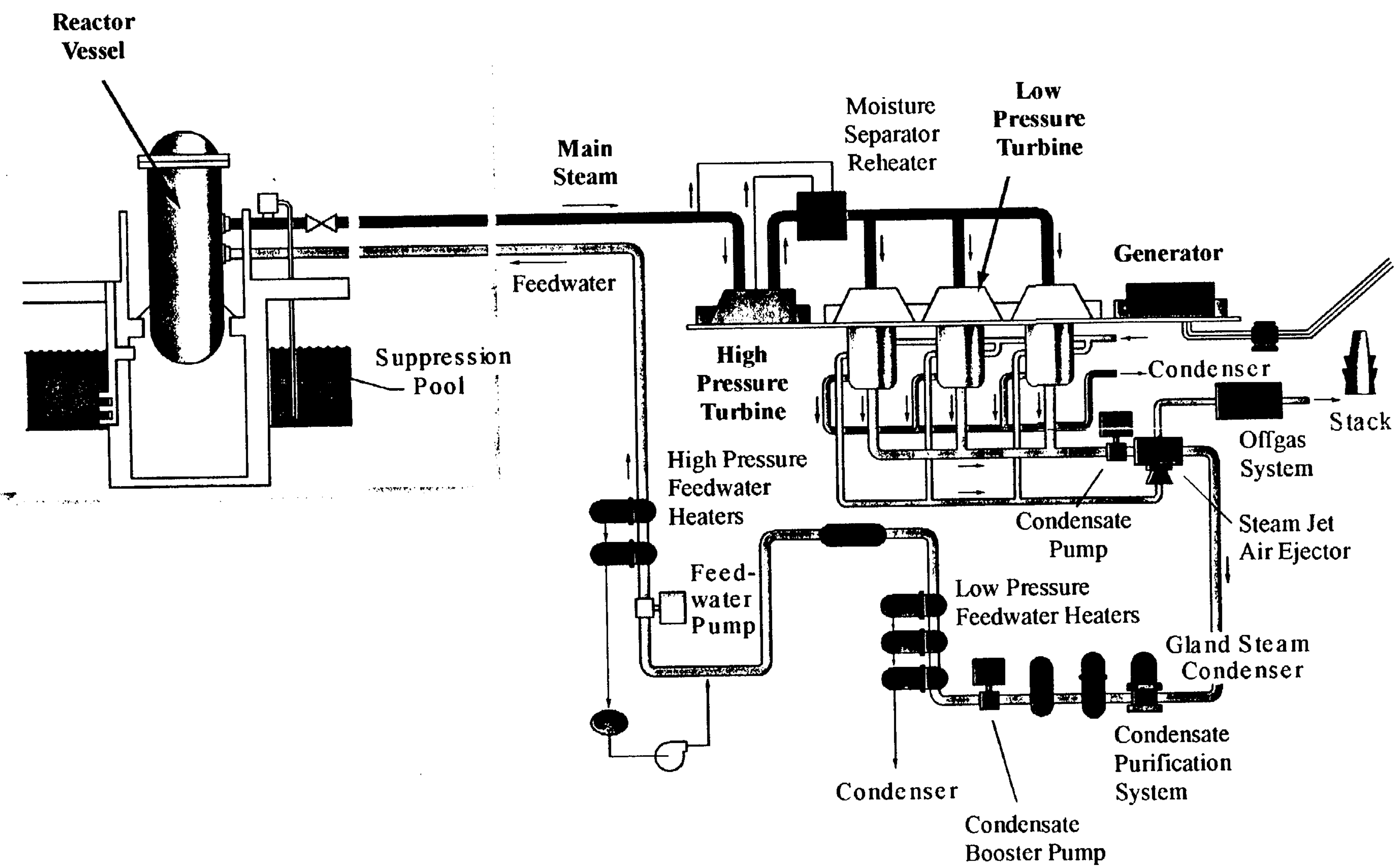
ESBWR Plant Schematic



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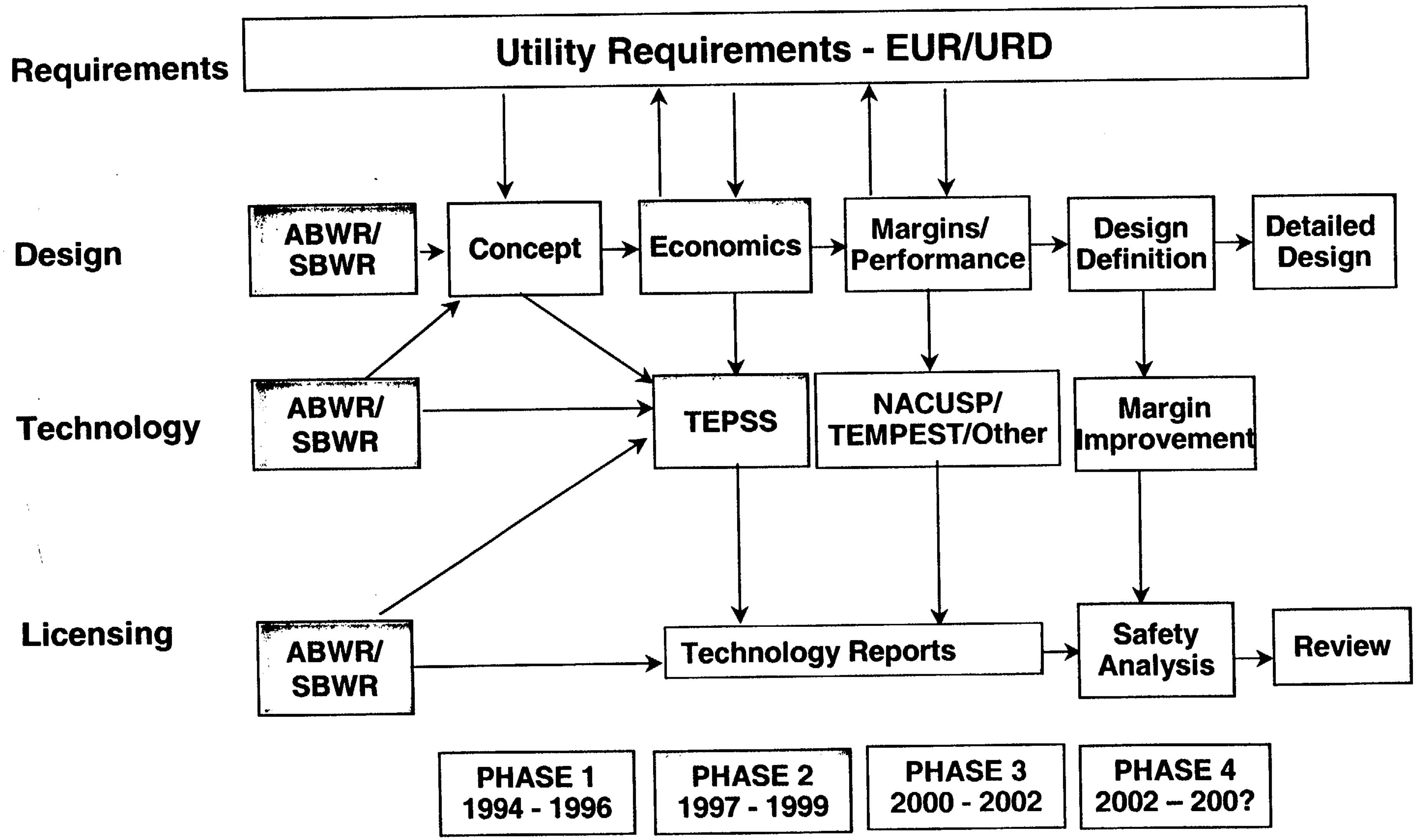
Comparison of Key Parameters

Parameter	ABWR	SBWR	ESBWR
▪ Power (MWt)	3926	2000	4000
▪ Power (MWe)	1350	670	1380
▪ Vessel height (m)	21.1	24.6	27.7
▪ Vessel diameter (m)	7.1	6.0	7.1
▪ Fuel bundles, number	872	732	1020
▪ Active fuel height (m)	3.7	2.7	3.0
▪ Power density(kw/l)	51	42	54
▪ Number of CRDs	205	177	121
▪ Building Size (m^3/MWe)	195	350	140

ESBWR Program Plan



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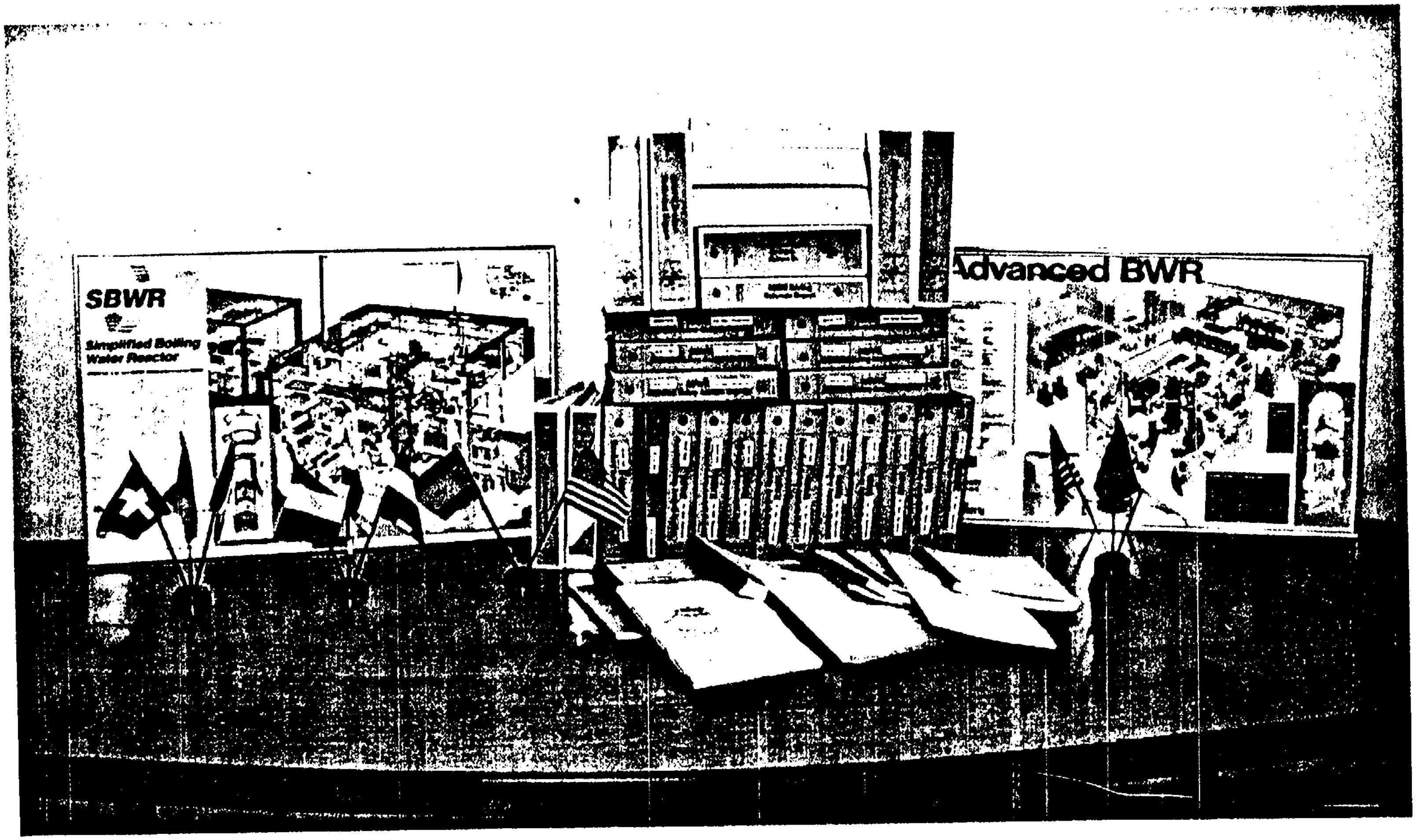
SBWR Simplifies ESBWR Challenges

- ABWR certification provides many inputs/bases
- SBWR program provides a solid base for ESBWR
  - SBWR program was a $200 – 300 million program
  - Completed a complete SAR with technology reports
  - Completed extensive testing and code qualification
  - Completed a multi-year NRC/ACRS review
- 8 year ESBWR program expanded the SBWR base
  - Used essentially the same design features
  - Completed extensive new testing and analysis
  - Improved the overall economics
- SBWR reviewers/developers still around

Increased performance and safety margins

ESBWR Design/Technology based on SBWR and ABWR

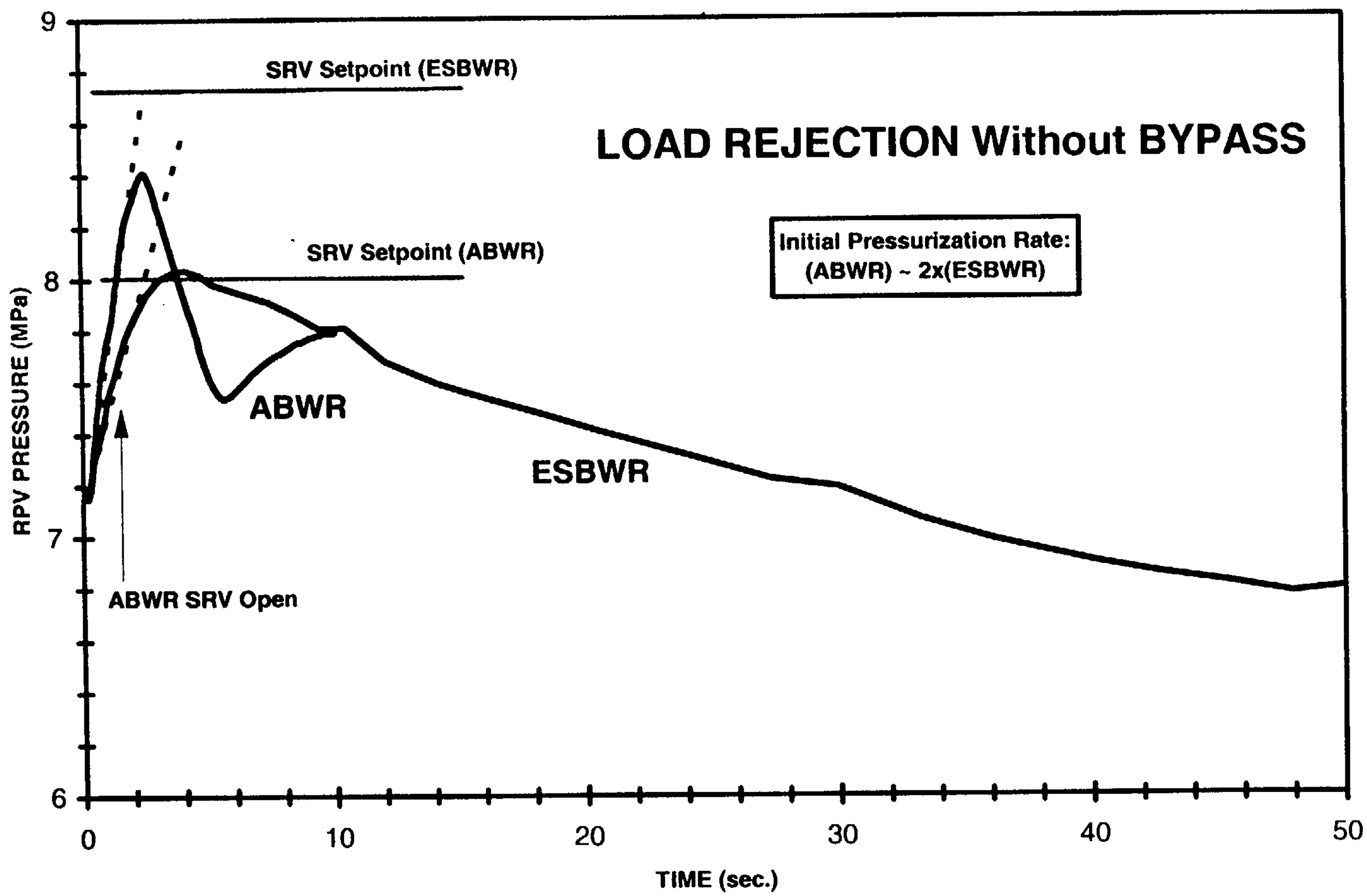


Comparison of Plant Performance

Parameter	Typical BWR	Passive BWR SBWR	Passive BWR ESBWR
Natural Circulation flow/bundle, kg/s	3.5 - 5	8.5	10.6
Power/Flow Ratio, MW/(kg/s)	0.25	0.31	0.26
Transient pressure rate, MPa/s	0.8	0.4	0.4
Margin to SRV setpoint during isolation transient, MPa	valve opens	0.52	0.32
Minimum water level after accident, m above top of fuel	0.0	1.5	2.8
Post accident containment pressure margin, KPa below design pressure	40	100	200

ESBWR Performance is Better Than or Equal to Most Plants

Fast pressurization transient

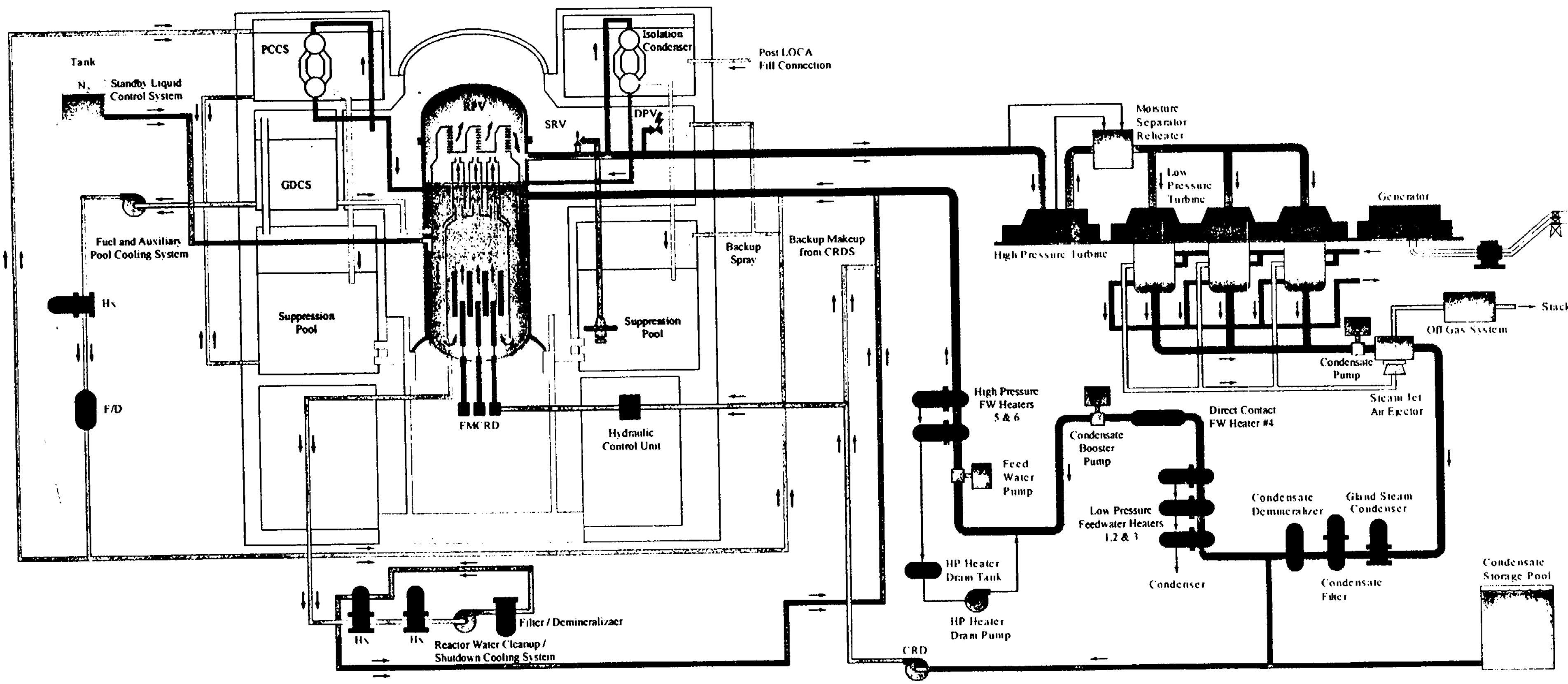


ESBWR: slower pressurization due to large steam volume in chimney; adequate margin to prevent SRV from opening

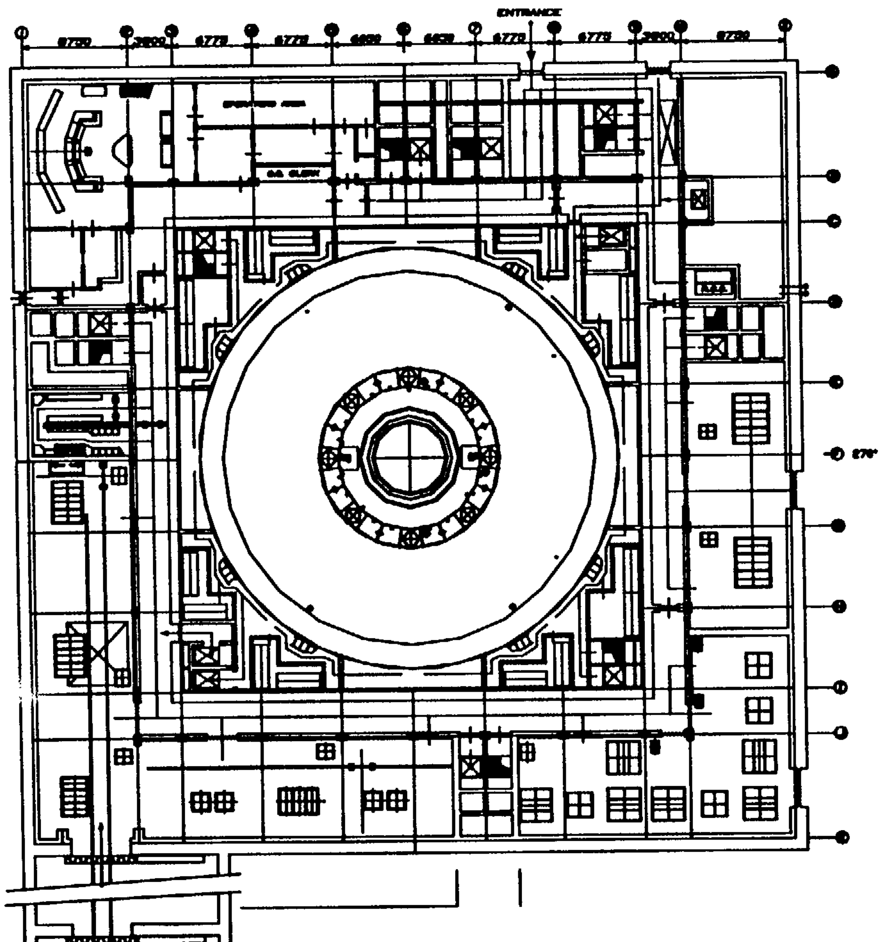
Factors that Resulted in Improved Economics

- **Economy of Scale**
  - Higher Power Density
  - Higher Plant Power
  - Use of Modular Passive Safety Systems
- **Design Features That Enhanced Economy of Scale**
  - Made GDCS Pool As Part of Wetwell
  - Modular Safety Systems With Little Dependence on Power Level
  - Smaller PCCS Pools and Larger Heat Exchangers
- **Improved the Overall Design**
  - Large Blade Control Rods
  - Simpler Reactor Internals
  - Improved Plant Arrangements
    - Moved Non Safety Systems, Stacked Spent Fuel
    - Flexible Building Embedment - External Cask Hatch

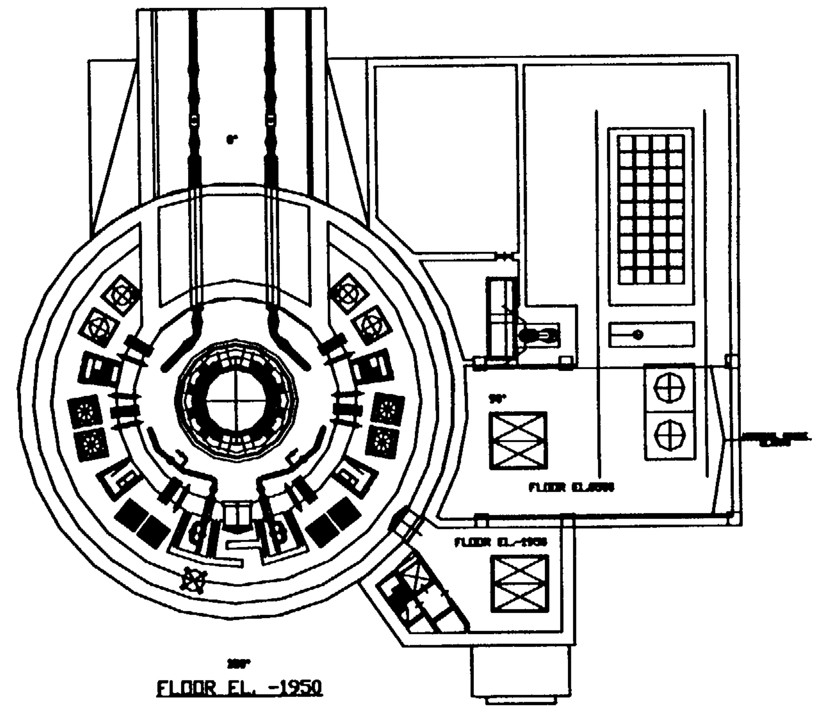
ESBWR Nuclear and Turbine Island Schematic



Comparison of SBWR/ESBWR Buildings



SBWR (670 MWe)



ESBWR (1380 MWe)

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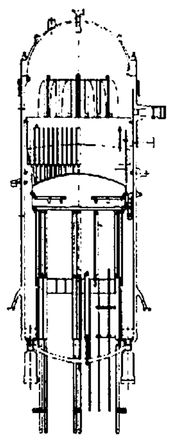
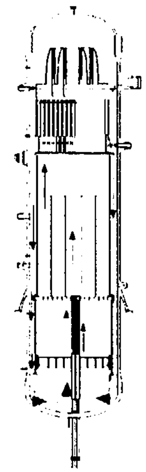
Core Design Evolution

ABWR
3926 MWt
872 bundles
7.1m / 21.4m



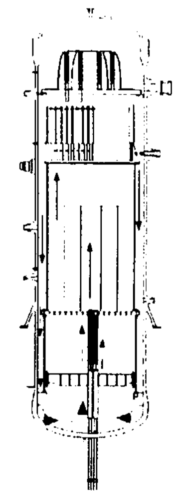
Eliminating pumps, shorten fuel

SBWR
2000 MWt
732 bundles
6.0m / 24.5m



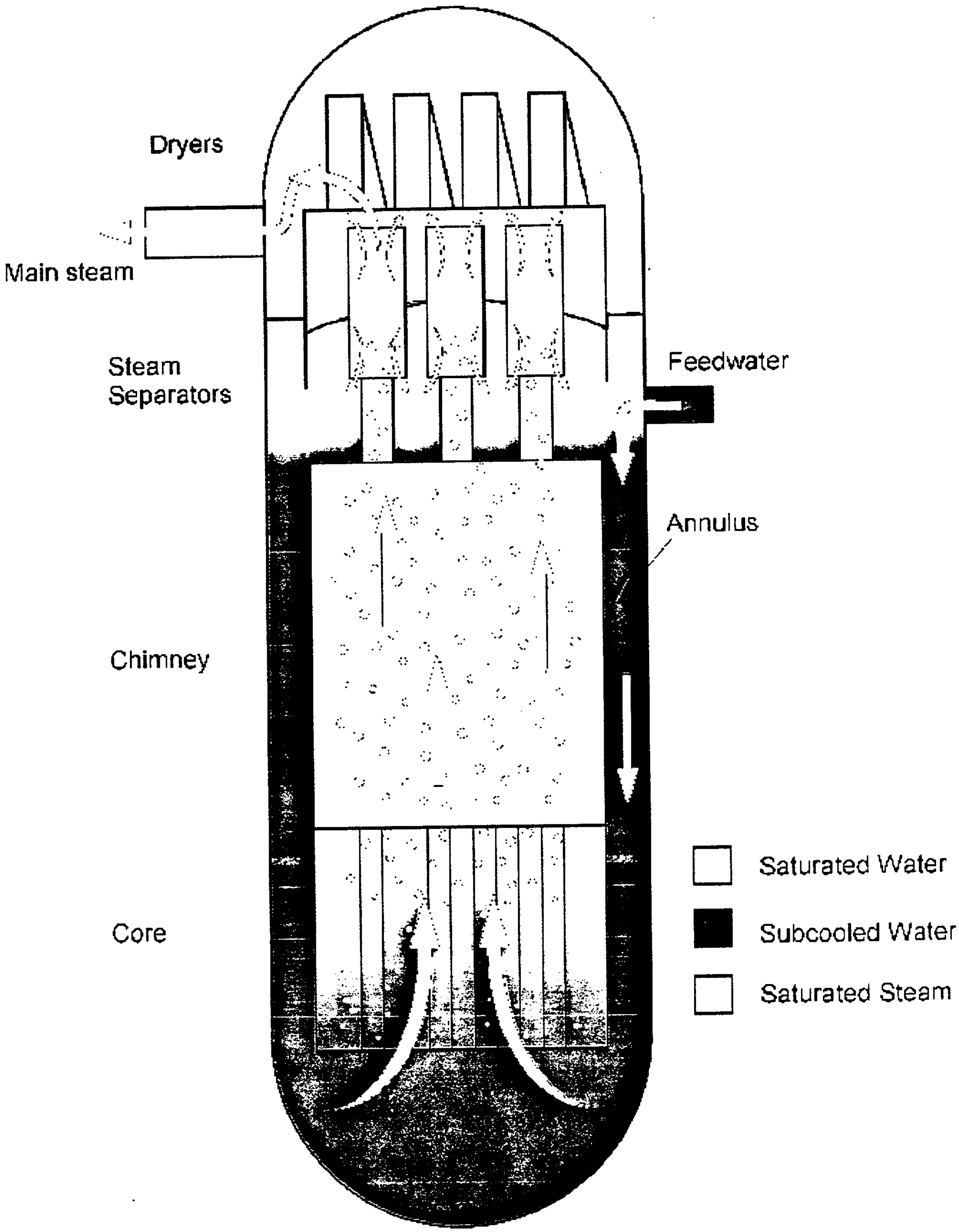
Taller vessel, improved internals

ESBWR
4000 MWt
1020 bundles
7.1m / 27.7m



ESBWR Design Evolution - Core

	ABWR	SBWR	ESBWR – Phase 1	ESBWR – Phase 2	ESBWR – Phase 3
Power (MWt)	3926	2000	3613	4000	4000
RPV Height (m)	21.4	24.5	25.4	25.9	27.7
RPV ID (m)	7.1	6.0	7.1	7.1	7.1
# of bundles	872	732	1132	1132	1020
Active fuel length (m)	3.67	2.74	2.74	2.74	3.05
Power Density (kw/l)	51.0	41.5	48.5	53.7	53.7


Dryers
Main steam
Steam
Separators
Feedwater
Annulus
Chimney
Core
Saturated Water
Subcooled Water
Saturated Steam

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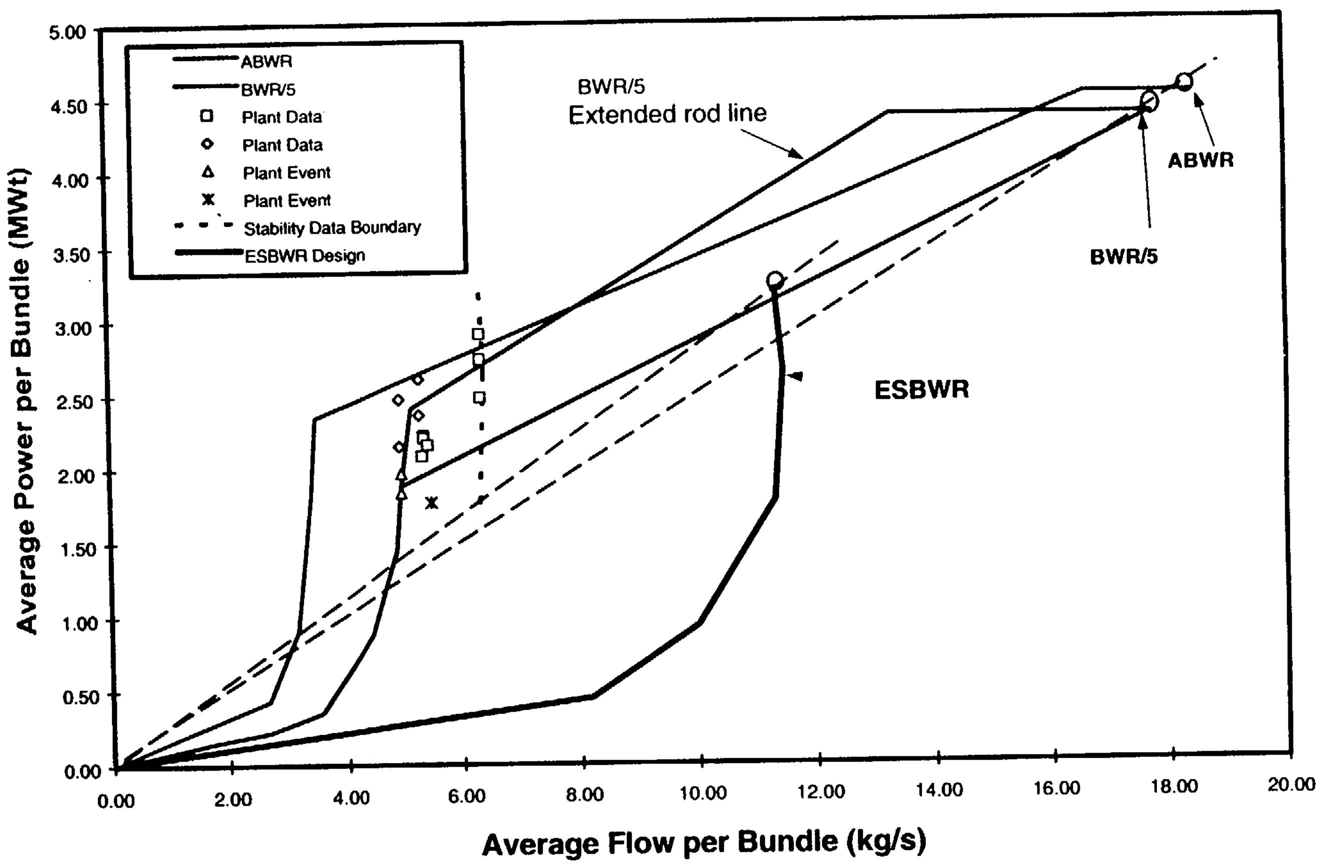
Bundle Power vs. Flow for various BWRs



POWFLO-2.xls chart 9

ESBWR has 100% flow margin to stability data boundary

Natural Circulation Technology Program

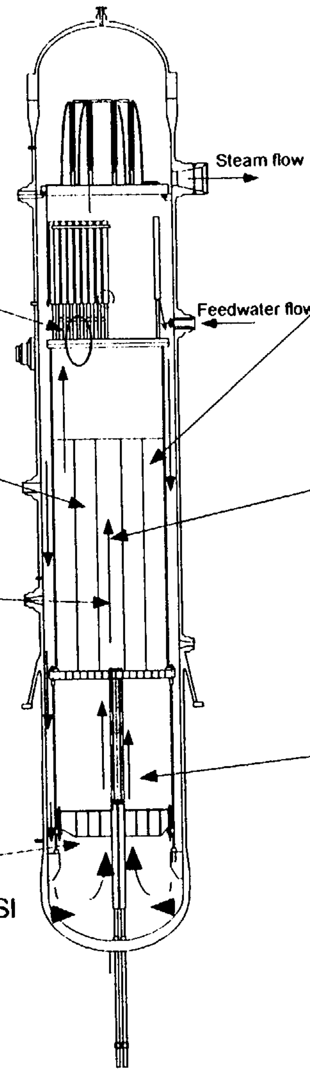
SBWR	ESBWR Phase 1	ESBWR Phase 2	ESBWR Phase 3

Separator Performance
ATLAS Tests - AS2B
- smooth inlet geometry
- reduced pitch
  (305 mm -> 292 mm)

Chimney Void Fraction
Ontario Hydro Tests
- transient test (pump induced)
- round pipe (0.518 m ID)
- relatively flat void distribution

Startup Flow Oscillation
CRIEPI Tests
- single chimney
- SBWR conditions
- large margin to oscillation regime

Core Flow Optimizaton
- studies performed by PSI
- supported by
  Swiss Utilities



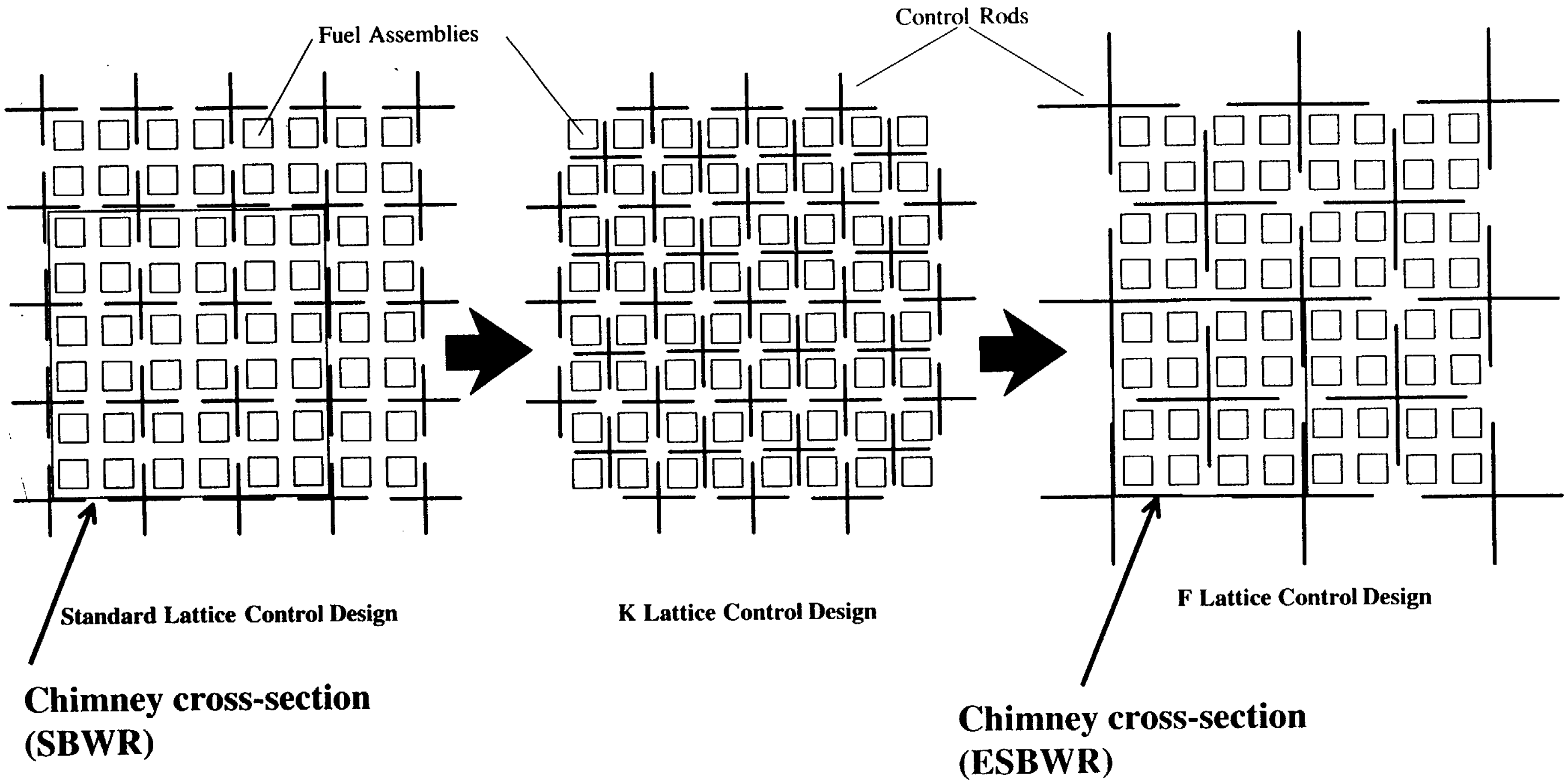
Chimney Void Fraction
CEA Chimney Tests
- scaled ESBWR conditions
- 3-D void distributions
- FIV on chimney partition
- supported by EdF

Startup Flow Oscillation
PSI / IRI Testing
- full range parameters
- ESBWR conditons
- scaling and other effects

Regional Oscillation
IRI / ETH Projects
- code development
  and analyses
- chimney effect
- core size effect

Control Rod Drive Design Evolution

- The "F" lattice is an extrapolation of earlier "K" lattice design

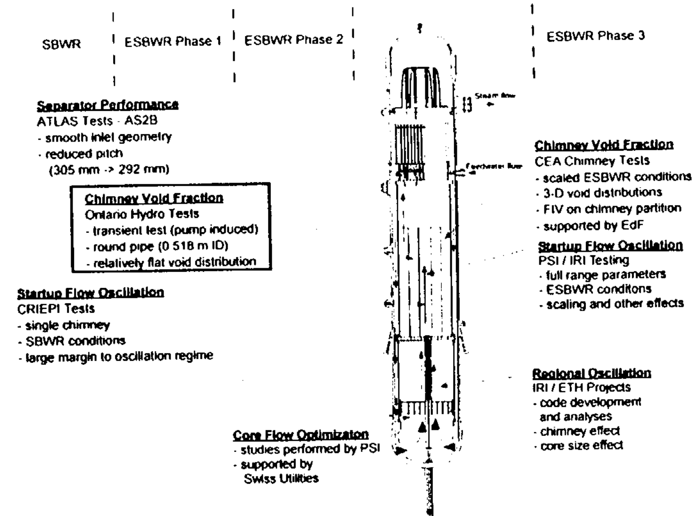


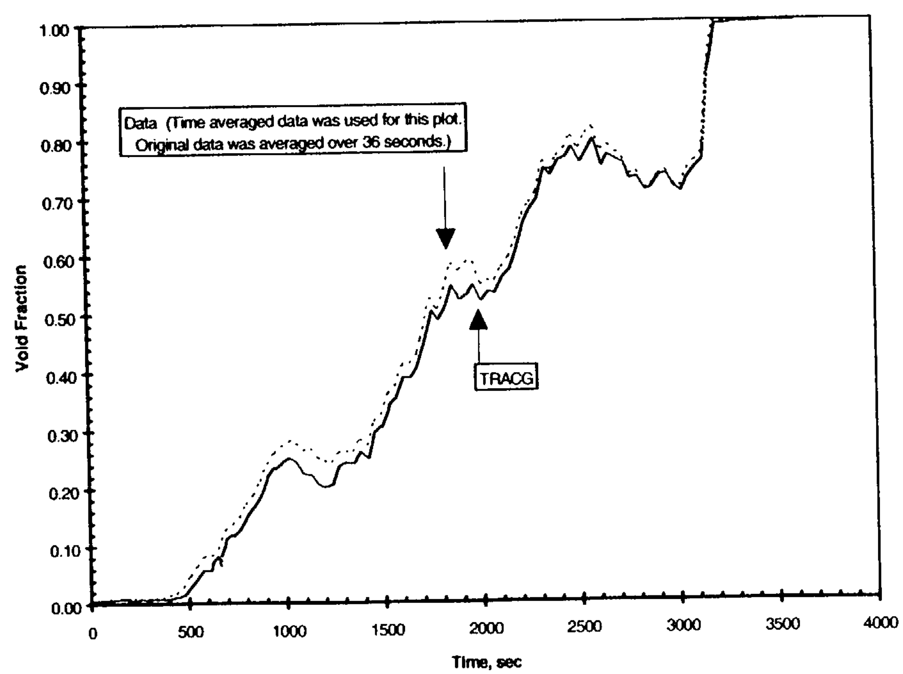
Chimney and Technology Programs

- Chimney provides the driving head for the natural circulation flow
- Flow rate is sensitive to the chimney void fraction
- Test programs to evaluate void fraction profile and to access flow induced vibration on chimney partition

Chimney Void Fraction

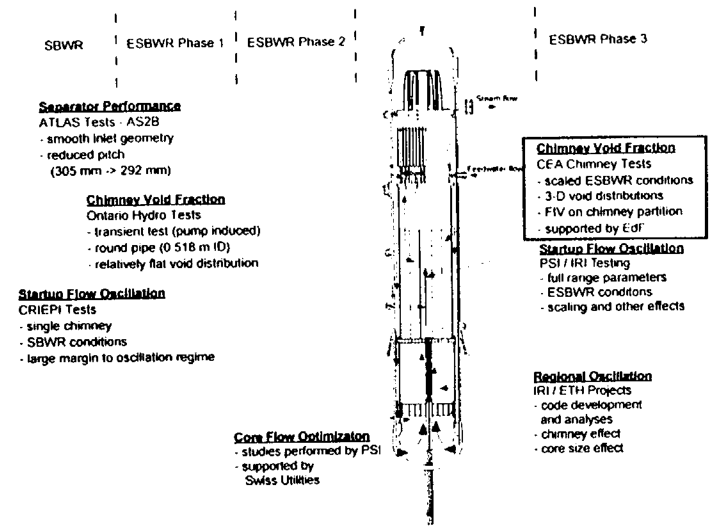
- Ontario Hydro Tests
  - Large pipe void fraction data
  - 0.51 m diameter, 6.4 and 2.8 MPa
- Relatively flat void profile across the pipe section
- Pump induced transient tests





Chimney Void Fraction

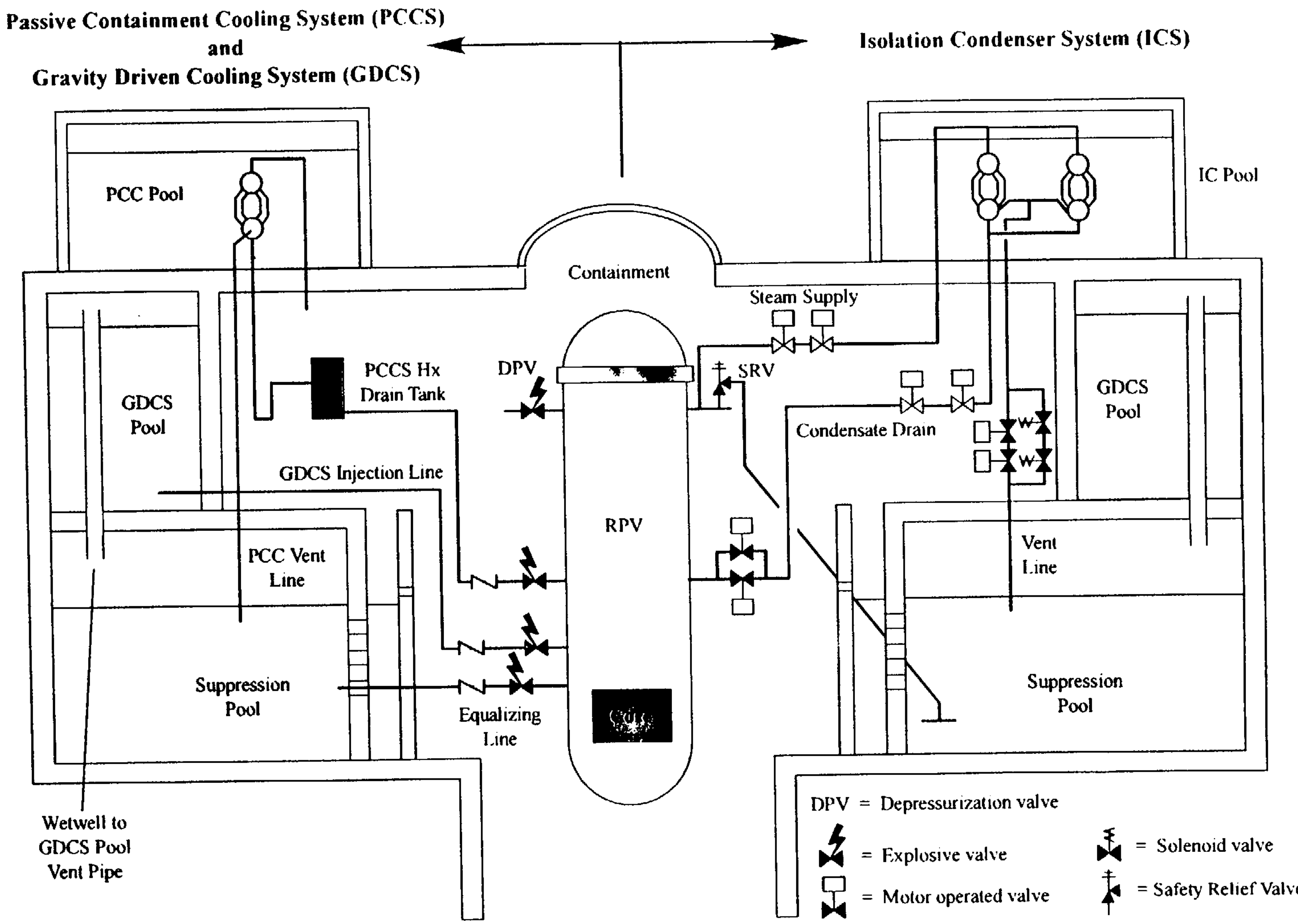
- CEA Chimney Tests
  - scale ESBWR geometry and conditions
  - measure 3-D void distributions
  - evaluate FIV on chimney partition
  - tests supported by EdF



Passive Safety Systems - Simplify the Plant

- Reactivity Control
  Electro-hydraulic control rod drive system
  Accumulator driven backup boron injection system
- Inventory Control
  Large vessel with additional inventory
  High pressure isolation condensers (IC)
  Depressurization and gravity driven cooling system (GDCS)
- Decay Heat Removal
  Isolation condensers for transients
  Passive Containment Cooling System (PCCS) condensers for pipe breaks
- Fission Product Control and Plant Accident Release
  Passive condensers
  Retention and holdup with multiple barriers

Simplified Systems Extending Operating Plant Technology



Passive Containment Cooling System (PCCS)
and
Gravity Driven Cooling System (GDCS)
Isolation Condenser System (ICS)
PCC Pool
IC Pool
Containment
Steam Supply
GDCS
Pool
PCCS Hx
Drain Tank
DPV
SRV
Condensate Drain
GDCS
Pool
GDCS Injection Line
RPV
PCC Vent
Line
Vent
Line
Suppression
Pool
Equalizing
Line
Suppression
Pool
Wetwell to
GDCS Pool
Vent Pipe
DPV = Depressurization valve
= Explosive valve
= Solenoid valve
= Motor operated valve
= Safety Relief Valve

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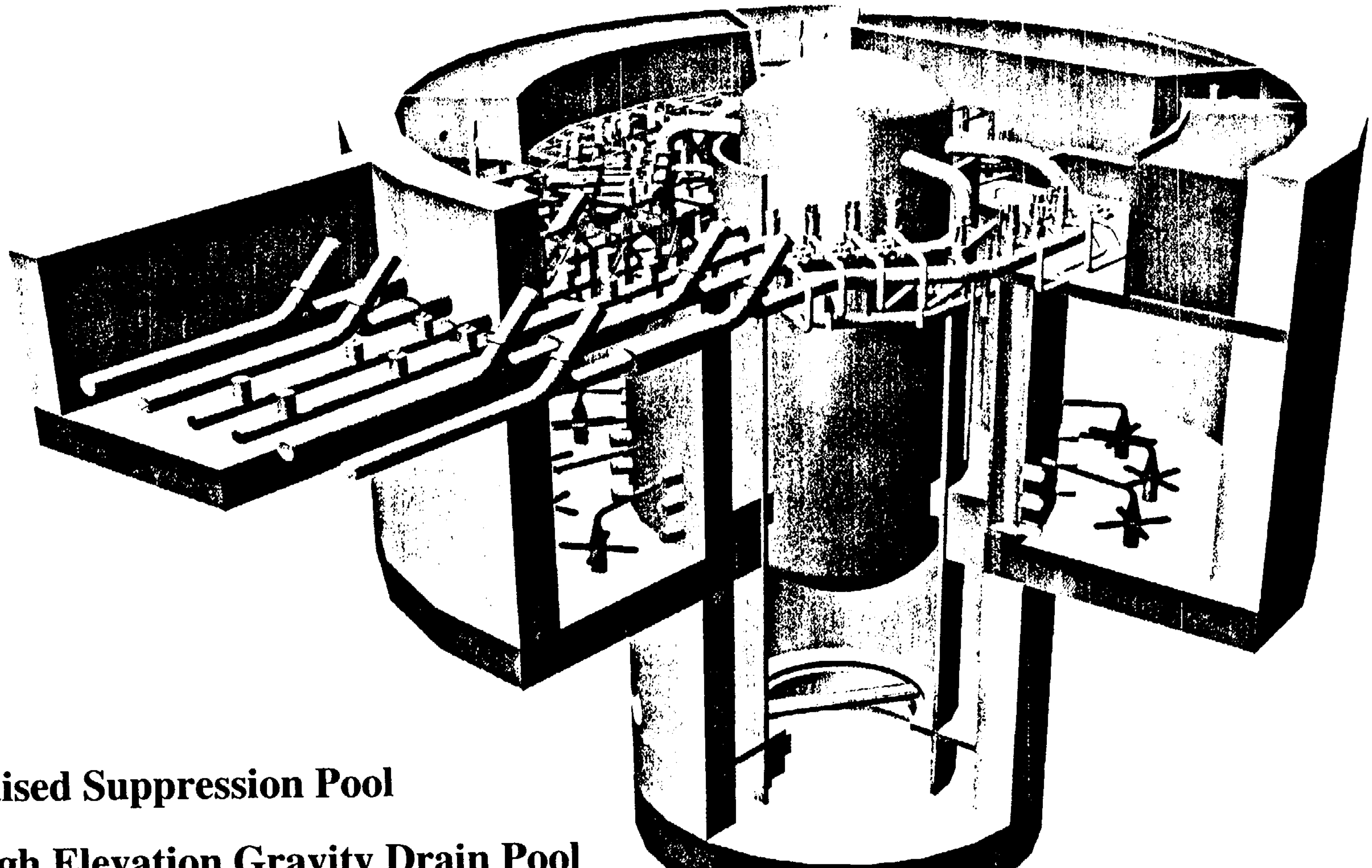
Design Philosophy for the Safety Systems

- Meet all Regulatory Requirements with Simple Passive Systems
  - Emphasis on simplification
  - No operator actions needed for 72 hours for design basis events
- Active Systems Modified Slightly to Enhance Overall Safety
  - Active systems are non safety-grade
  - Minor changes made to improve PSA results
- Plant Shutdown and Accident Recovery
  - Use active systems

Safety Systems Inside Containment Envelope



- Raised Suppression Pool
- High Elevation Gravity Drain Pool
- All Pipes/Valves Inside Containment
- Decay Heat Condensers Above Drywell

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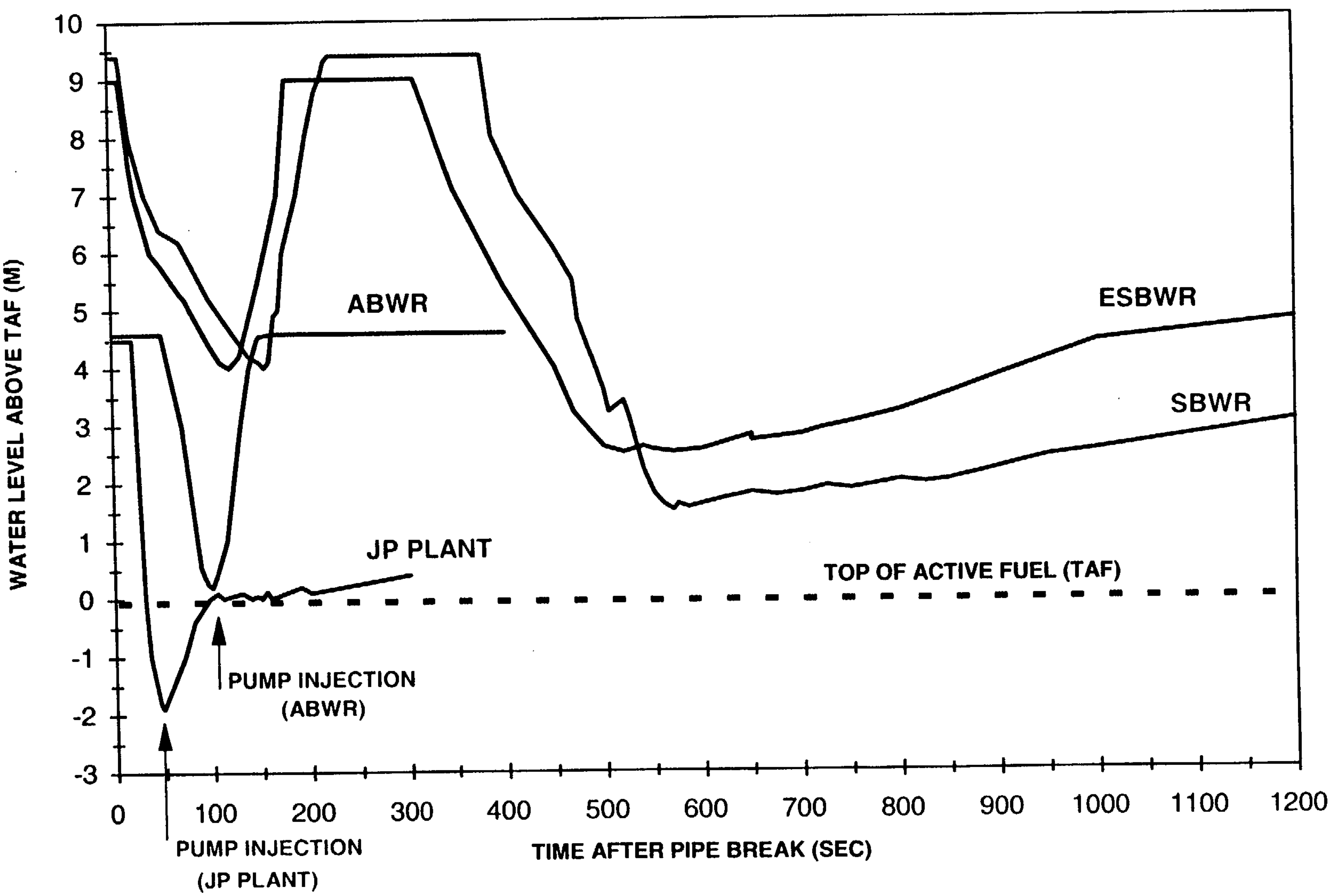
Water Level in Shroud Following a Pipe Break



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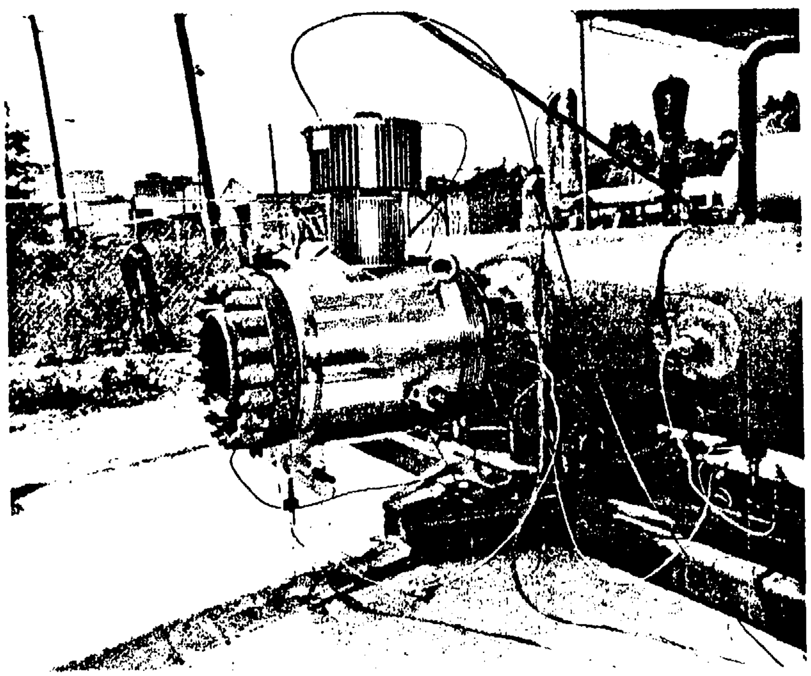
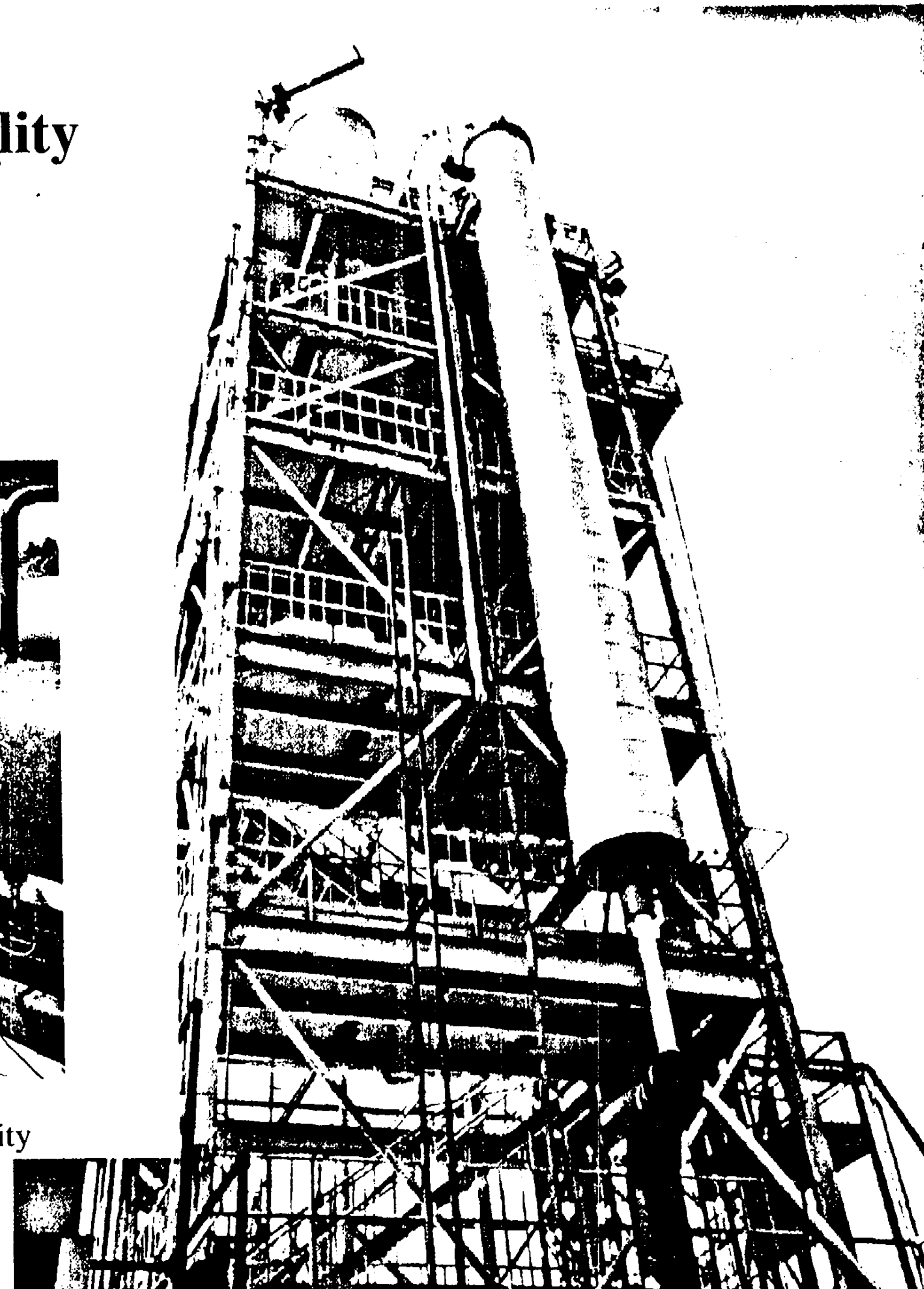
Safety System (GIST) Test Facility and Depressurization Valve



Reactor Depressurization Valve in the Test Facility



Decay Heat Removal/Containment Features and Technology

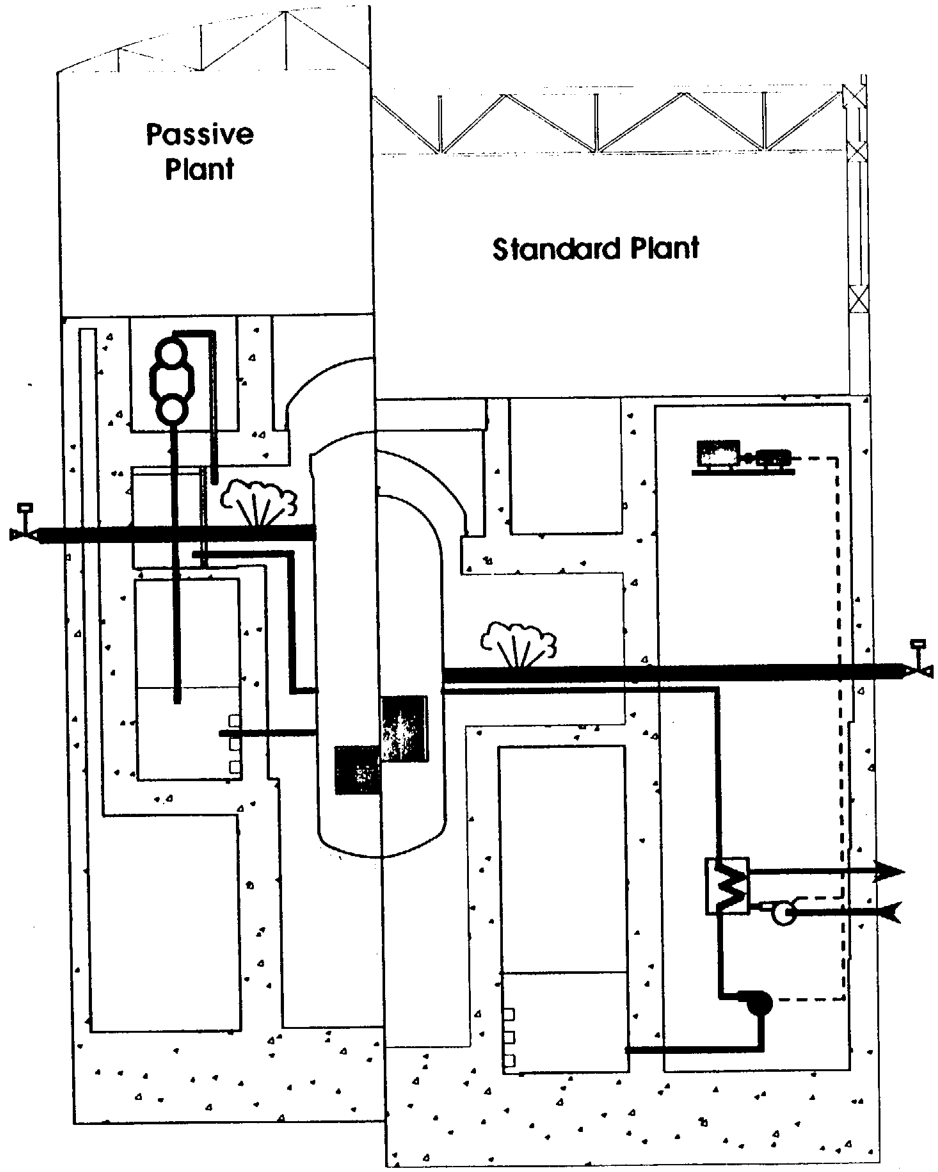
- **Decay Heat Removal Design Features**
- Past Technology Program - SBWR
- ESBWR System Modifications from SBWR
- ESBWR Technology Program
- Conclusions

ESBWR Decay Heat Removal

- Remove Decay Heat From Vessel
  - Main Condenser
  - Normal shutdown cooling system
  - Isolation condensers
  - Remove vessel heat through valve opening

- If Needed, Remove Heat From Containment
  - Suppression pool cooling
  - Containment sprays
  - Passive containment cooling (PCCS) condensers

Several Diverse Means of Decay Heat Removal



Containment Heat
Removal System

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Decay Heat Removal/Containment Features and Technology

- Decay Heat Removal Design Features
- **Past Technology Program - SBWR**
- ESBWR System Modifications from SBWR
- ESBWR Technology Program
- Conclusions

Extensive Technology Program to Qualify Features New to SBWR

- Component and Integral tests as part of the SBWR program
  - Full scale components tests - condensers, valves
  - Integral tests at different scales, with the largest test at PANDA
- Testing extended to incorporate European requirements
  - Large hydrogen releases and severe accidents
  - Improvements in the plant design
- Ongoing programs will further quantify margins
  - Natural circulation in the vessel
  - Severe accident performance/features for passive systems
- Testing used to qualify computer codes
- Extensive international cooperation

A Complete and Thorough Technology Program Supports the Design

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Containment Technology Overview

SBWR and ESBWR Phase 1

Condensation with N/C
MIT - external condensation
UCB - single tube tests
GIRAFFE - component tests
PANTHERS - component tests
PANDA - steady state tests

PCCS Performance
Steady-state: PANDA, GIRAFFE, PANTHERS
Start-up: PANDA, GIRAFFE
Secondary Side ht: PANDA, PANTHERS, GIRAFFE
N/C Buildup: PANDA, PANTHERS, GIRAFFE
Interactions: PANDA
Unit interactions PANDA

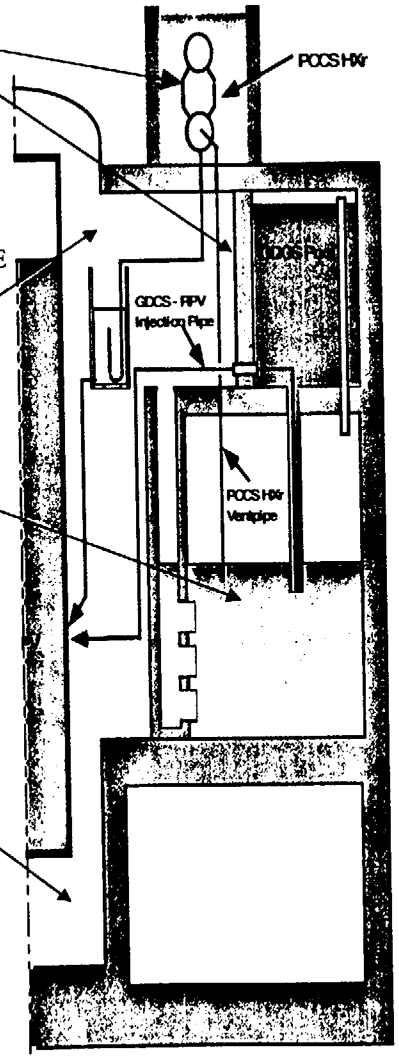
System Interactions
PANDA
GIRAFFE

SP Stratification
LINX (ALPHA)
PANDA

DW Stratification and Hideout
PANDA
GIRAFFE

Steam Quenching
Main Vent:
- Horiz. Vent Test/MK III tests (PSTF)
PCC Vent:
- PSI theoretical work (Coddington et al)
- UCB Sparger/Vent chimney
- PANDA

Heat/Mass Leakage DW to WW
Finite Element Analysis
VB Testing



ESBWR Phase 2

PCCS Performance
PANDA (TEPSS)
- startup
- interactions
- secondary side ht
- N/C Buildup
- Unit interactions

ESBWR Configuration
PANDA (TEPSS)
- reduced cont. volume
- GDCS in WW
- PCCS Condensate to RPV
VTT
- Modeling of larger PCC

DW Stratification and Hideout
PANDA(TEPSS)
- Asymmetric loading
- hydrogen

SP Stratification
LINX (TEPSS)

ESBWR Phase 3

PCC Hydrogen Distribution
PANDA + CFD (FFWP)
VTT - CFD

DW Stratification and Hideout
UCB + CFD (FFWP)
PANDA + CFD (FFWP)

WW Gas Stratification
UCB + CFD (FFWP)
KALI + CFD (FFWP)

PANTHERS

- Demonstrate that prototype heat exchanger is capable of meeting design requirements
- Provide database for TRACG (code) qualification to predict heat exchanger performance spanning the range of conditions expected in the SBWR (i.e. steam flow, air flow, pressure, temperature)
- Investigate the difference between lighter-than-steam and heavier-than-steam noncondensibles
- Structural component qualification

PANDA-M

- Objectives

  Demonstrate steady-state, startup and long-term operation of the PCCS system

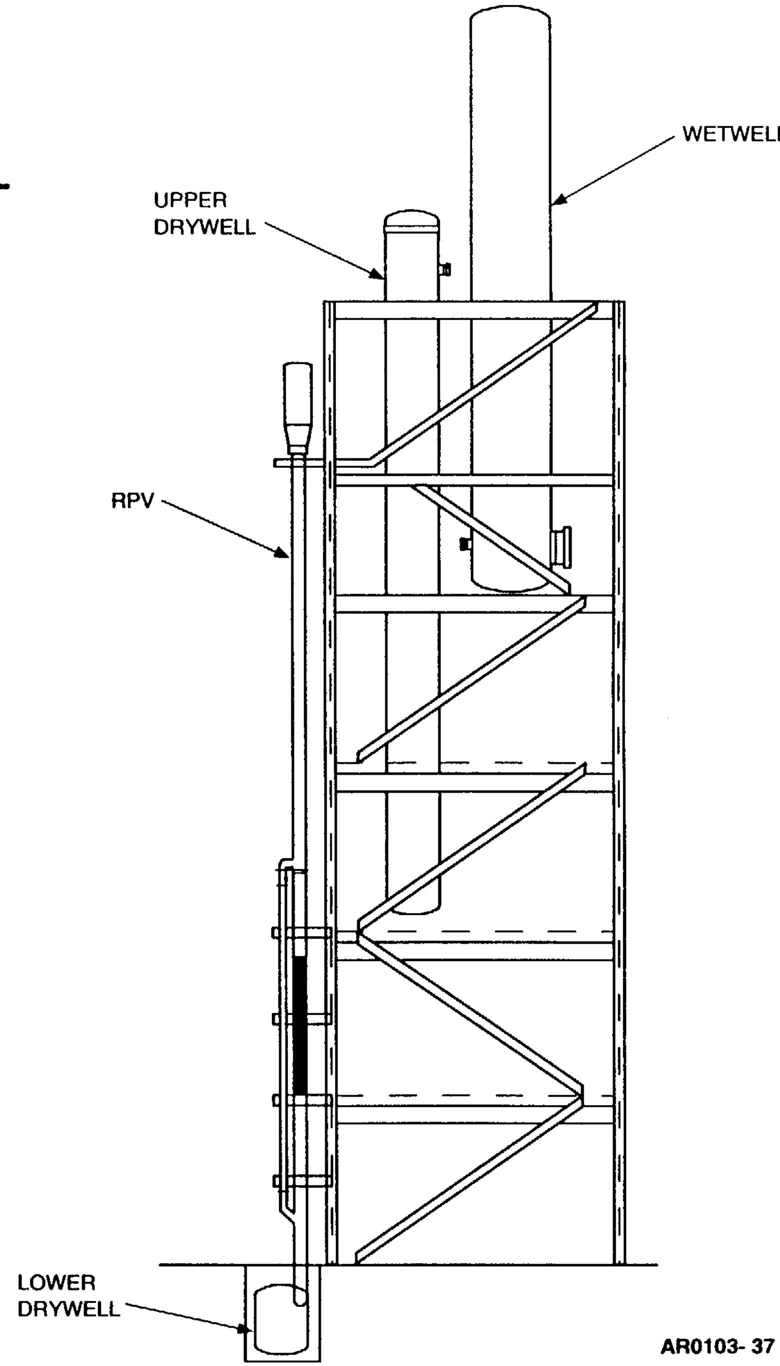
  Demonstrate effects of scale on PCC performance

  Data for TRACG (code) qualification to predict SBWR containment system performance including potential system interactions

- 10 steady state PCC component tests over a wide range of steam and air flow rates
- 12 transient tests representative of post-loca conditions with different configurations

GIST

- Objectives
  - Demonstrate technical feasibility of GDCS concept
  - Database for qualification of TRACG (codes) to predict GDCS initiation times, flow rates and RPV water levels
- 26 tests representing a range of conditions encompassing 3 LOCA's and a no break condition



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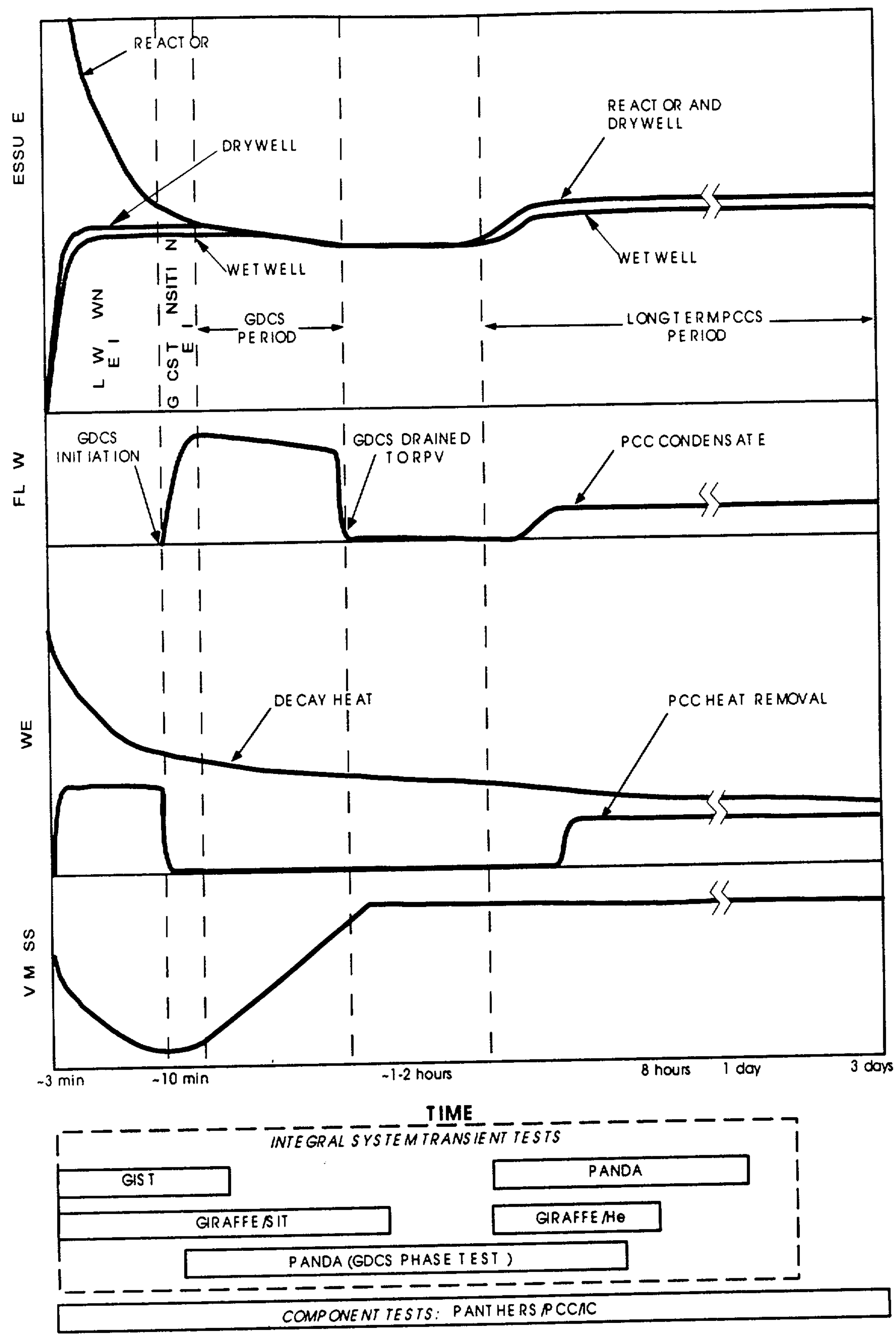
GIRAFFE

- 3 Test series:
  - GIRAFFE/Helium
    - Demonstrate system operation with lighter-than-steam noncondensibles including purging noncondensibles from the PCC
    - Data for TRACG (code) qualification to predict SBWR containment system performance including potential system interactions with l-t-s gas
  - GIRAFFE/SIT
    - Data for TRACG (code) qualification to predict SBWR ECCS performance during late blowdown/early GDCS phase of a LOCA - specific focus on system interactions
  - GIRAFFE/Step 1 and 3
    - Steady state performance of PCCS
    - System performance



Key Variables and Test Coverage

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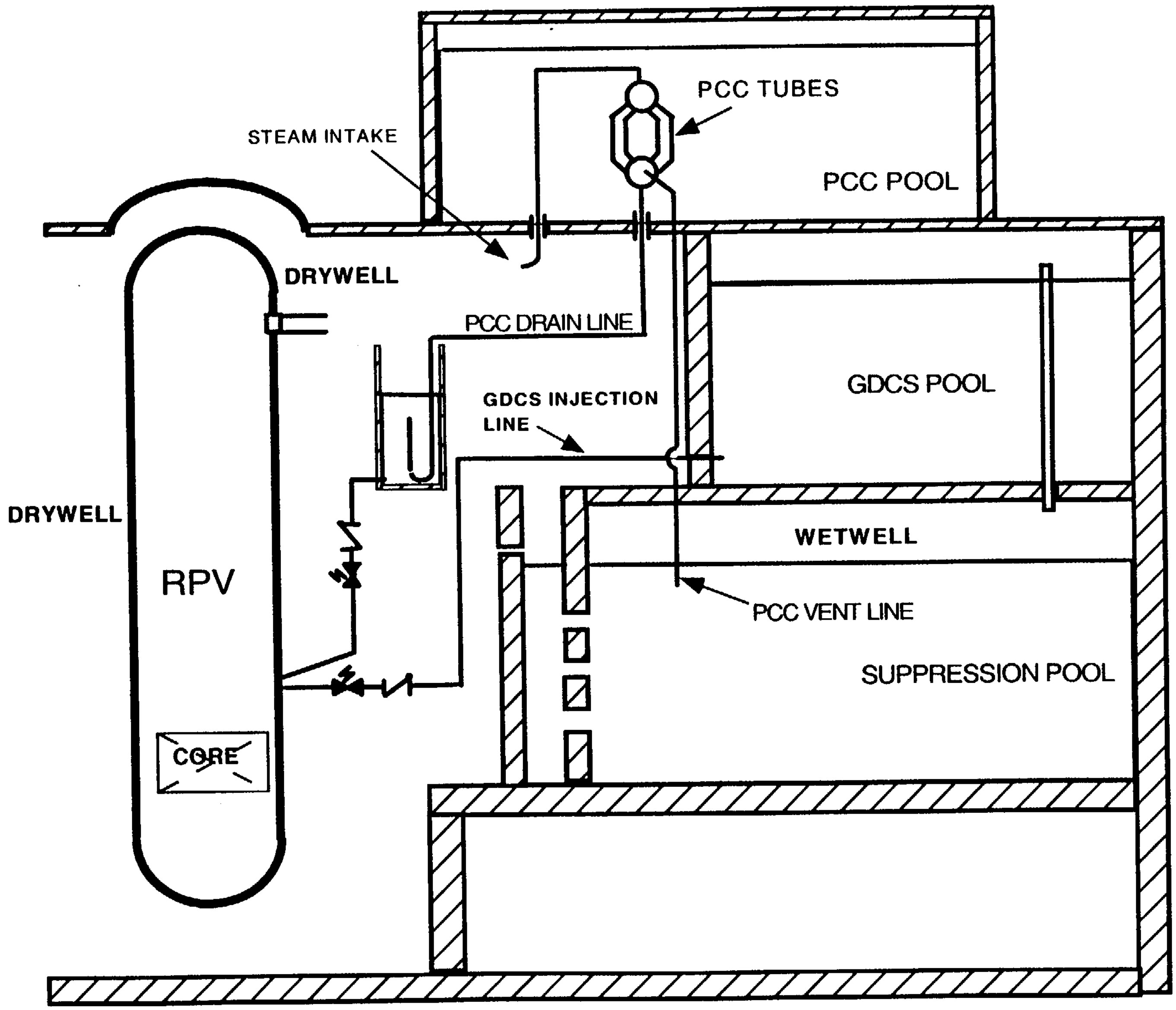
Decay Heat Removal/Containment Features and Technology

- Decay Heat Removal Design Features
- Past Technology Program - SBWR
- **ESBWR System Modifications from SBWR**
- ESBWR Technology Program
- Conclusions

ESBWR System Modifications

- Containment Configuration Optimized
  - Utilize GDCS pool draindown space to provide increased wetwell volume for severe accident (GDCS moved from DW to WW)
  - PCCS Condensate Tank added in DW
- Increased Power
  - Number of bundles, bundle length and power density increased
  - Additional PCC and IC added
  - Increased number of PCCS tubes per unit by 35%

ESBWR System Modifications



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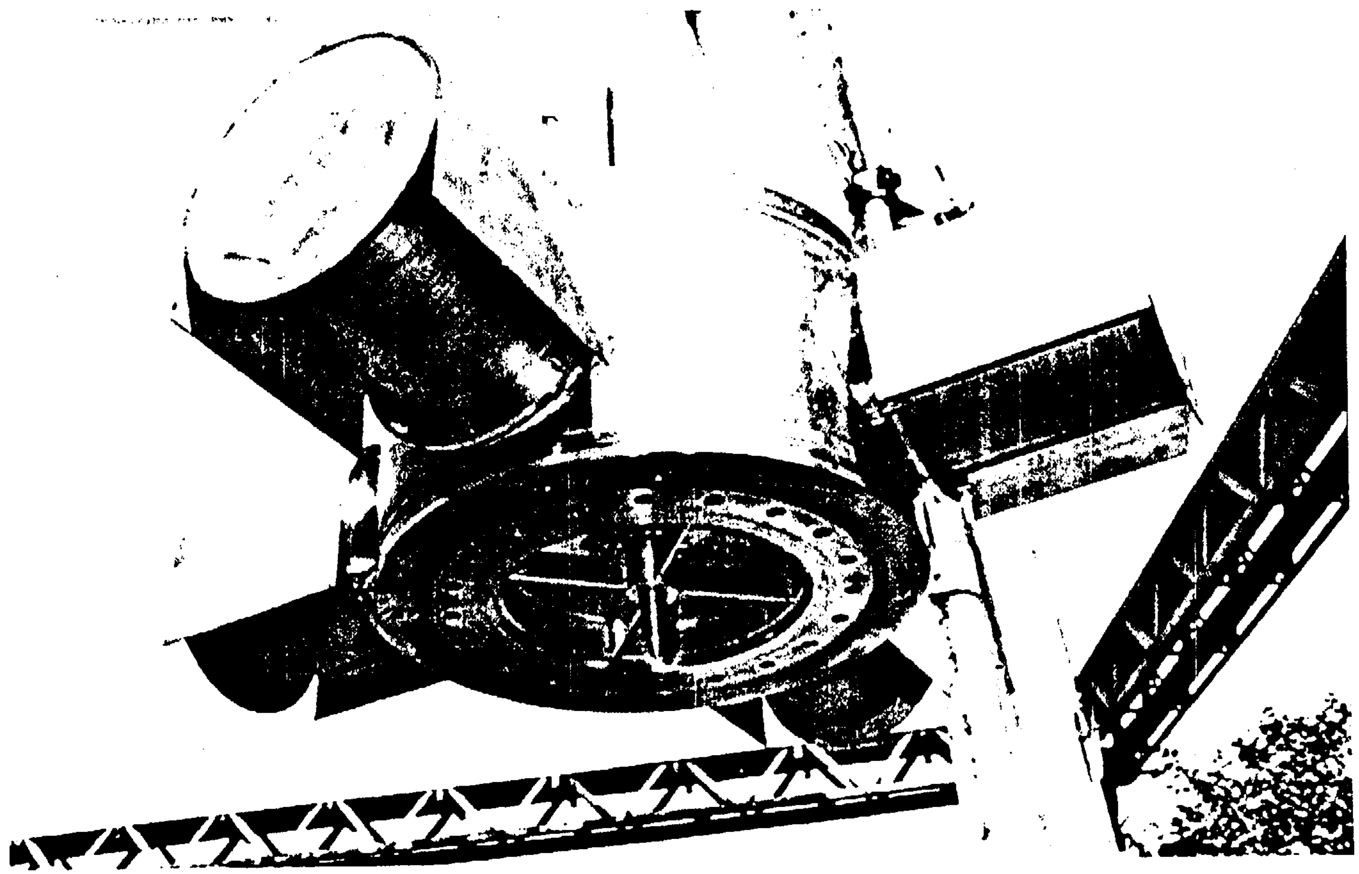
Prototype Vacuum Breaker



Decay Heat Removal/Containment Features and Technology

- Decay Heat Removal Design Features
- Past Technology Program - SBWR
- ESBWR System Modifications from SBWR
- **ESBWR Technology Program**
- Conclusions

TEPSS Program

3 Part program to extend the SBWR database to the ESBWR

- Suppression Pool stratification and mixing
  - 9+ tests with flow visualization in LINX
  - CFD analysis using CFX
- Passive Decay Heat Removal
  - 8 Integrated system tests run in PANDA
  - Pre- and post-test predictions using TRACG, TRAC-BF1, RELAP5 and MELCOR
- Passive Aerosol Removal
  - PCCS testing in AIDA
  - Analysis with MELCOR
  - Demonstrate PCCS as fission product aerosol filter
  - Demonstrate ability of PCC to remove decay heat with aerosol build-up

Suppression Pool Stratification/Mixing (LINX)

- Objectives
  - Improved countermeasures against pool stratification
  - Database for pool mixing models
- Conclusions
  - Steam bypass not expected for ESBWR
    - Bypass onset only at very high pool temperature (very low sub-cooling)
    - Limitations on test vent flow rate so that bypass for worst case ESBWR flow could not be completely excluded
  - Good pool mixing observed
    - Strong mixing for steam-air mixtures
    - Good mixing for steam only flow (less than 4 ºC for worst case)
    - Results may not be scalable
  - Analytical model validated against published plume spreading data

Passive Decay Heat Removal (PANDA-P)

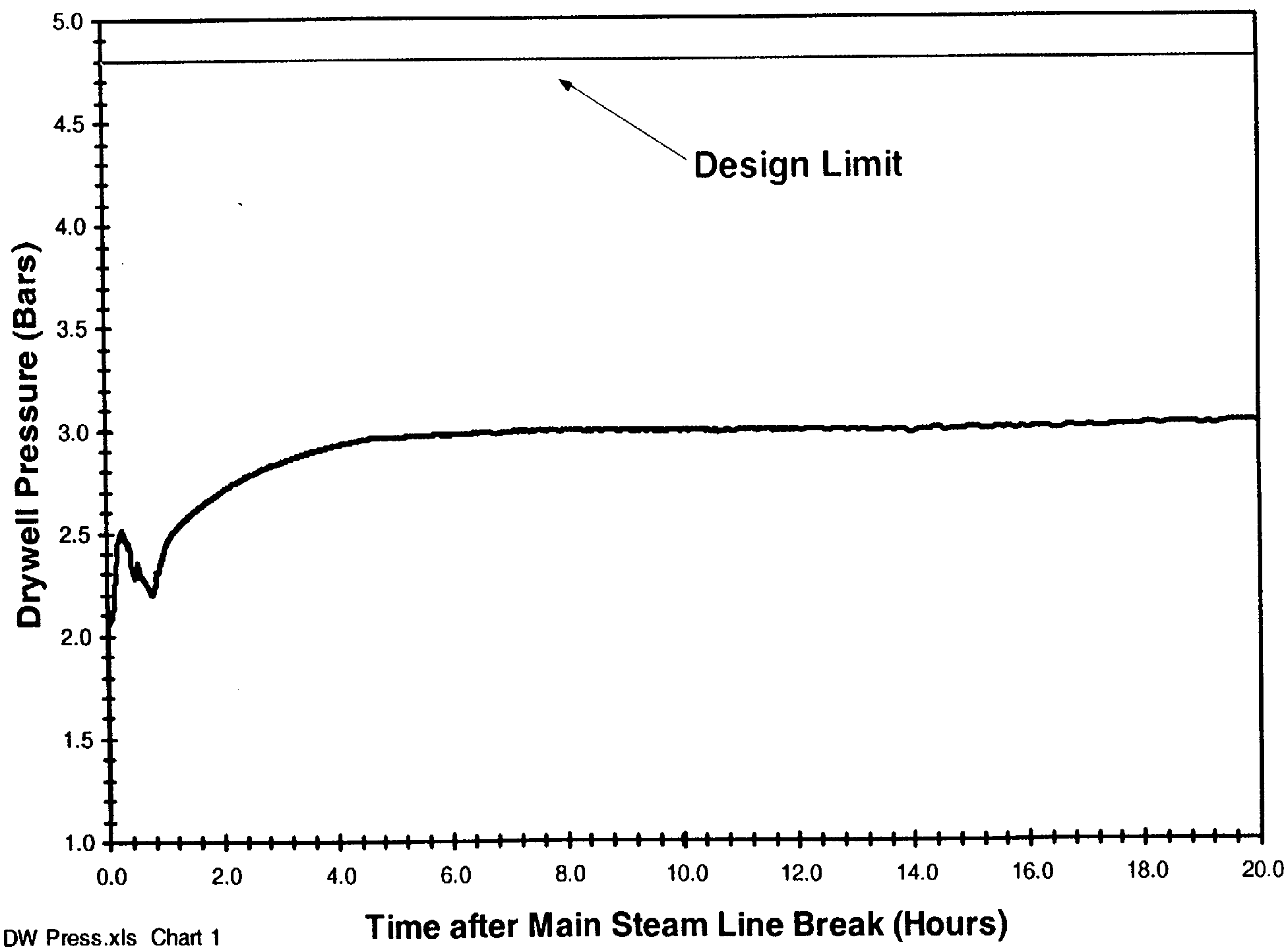
- Objectives
  - Testing of new containment features with respect to: PCCS long-term performance, PCCS start-up and systems interaction and distribution of steam and gases within the containment
  - Database to confirm the capability of TRACG to predict ESBWR containment system performance, including potential systems interaction effects
  - Effect of lighter-than-steam gas on system behavior
- Conclusions
  - Containment system operated robustly over all conditions tested
  - TRAC-BF1, RELAP5 and MELCOR benchmarked against test data
  - Some remaining uncertainties related to hydrogen behavior

TRACG has been benchmarked against the new test data

PCCS Extension

- Objectives
  - Analytical program to investigate the ability to scale up the PCC from 10 MW to 13. 5 MW without adverse effects
  - Investigation of secondary side heat transfer
- Conclusions
  - The PCC heat removal scales approximately linearly with number of tubes
  - Secondary side heat transfer does not limit the condenser performance

Substantial Margin for DBA Containment Pressure



MSLB DW Press.xls Chart 1

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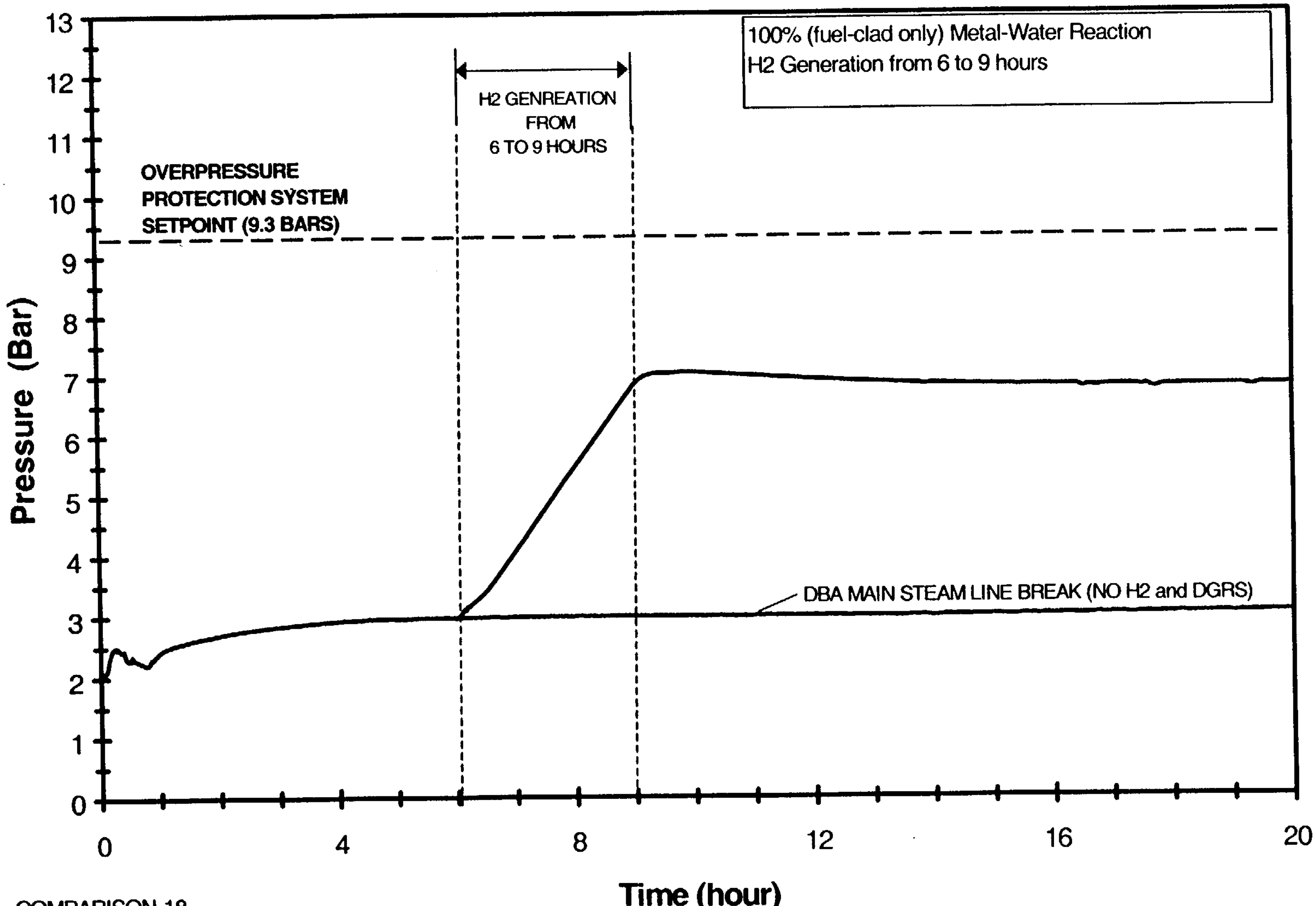
100% Clad Metal Water Reaction Results



COMPARISON-18

AR0103-50

Decay Heat Conclusions

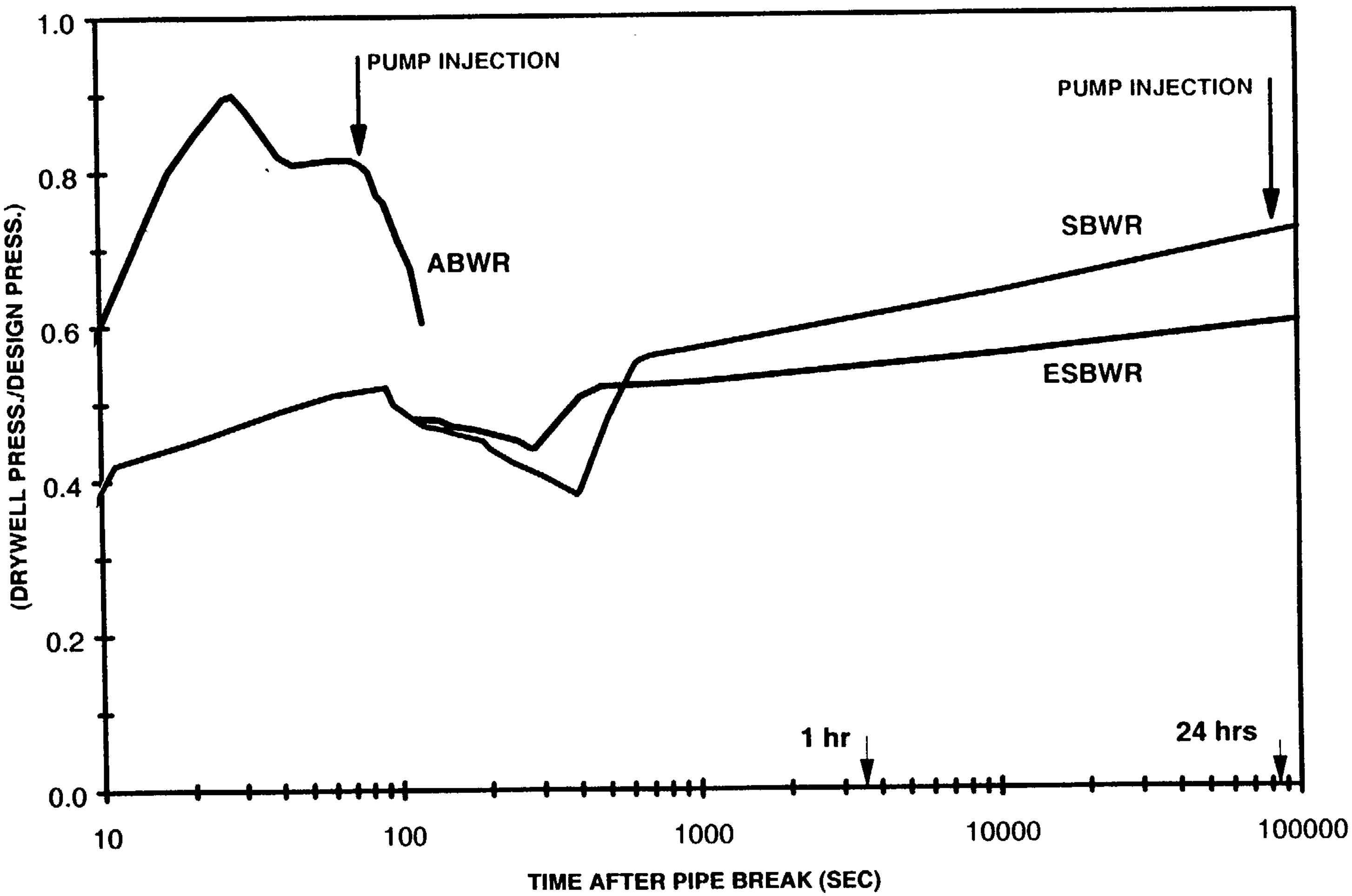
- Robust behavior of ESBWR containment demonstrated
  - ESBWR containment modifications improve pressure performance
  - Significant margins for Design Basis Accidents
  - Asymmetry effects not important
  - System interactions do not adversely effect performance
- PCCS capabilities confirmed
  - Start-up and long-term operation with noncondensibles confirmed
  - Heat removal capability sufficient over the range of conditions expected in ESBWR
  - Good performance with both light and heavy noncondensibles
  - Scalable technology

Decay Heat Conclusions (Cont'd)

- Suppression Pool Performance Good
  - Very little stratification in Suppression Pool
  - No steam PCCS vent bypass expected in ESBWR

Issues related to decay heat removal resolved through extensive testing and analysis programs

Containment Pressure Following a Pipe Break



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Ongoing Simplification Studies

- **Reduce Fuel Bundles, CRD, Vessel - COMPLETE**

  Increase Fuel Length

- **Improve Plant Availability - 5%**

  Refueling and Outage Plan and System Improvements

- **Reduce Buildings and Structures - 30%**

  Reduce Basemat Thickness

  Reduce Containment Design Pressure

  Move Spent Fuel Pool to Grade Elevation/Separate Building

  Separate Reactor Building From Containment

Normal performance margins maintained while reducing excessive conservatisms in other areas

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Fuel, Vessel and CRD optimization

- **Optimization of Fuel Length**

  0.3m Increase in Fuel Length Gives Significant Benefit

  Performance Margins Are Sufficient

  Design Options Being Explored to Increase Margins

  Further Studies Expected to Confirm Margins

- **Reduction in Key Components**

  Control Rod Drives and Fuel Bundles Reduced 10%

  Significant Simplification in Vessel and Internals

- **Impact on Building Height Minimal**

  Other Changes Will Have a Bigger Impact

Selected key parameters to simplify the design

Building/Structures & Refueling Optimization

- **What Controls Building Size**
  - Wetwell, PCCS Parameters and MSIV Access Control Building Height
  - Vessel Height Does Not Control Building Height
  - Refueling Floor Size and Dimensions Control Footprint
  - Refueling Schemes Are Very Important for Optimization
- **What Controls Structures**
  - Containment Design Pressure
  - Plant Seismic Design Basis
- **What is the Impact on the Construction Schedule**

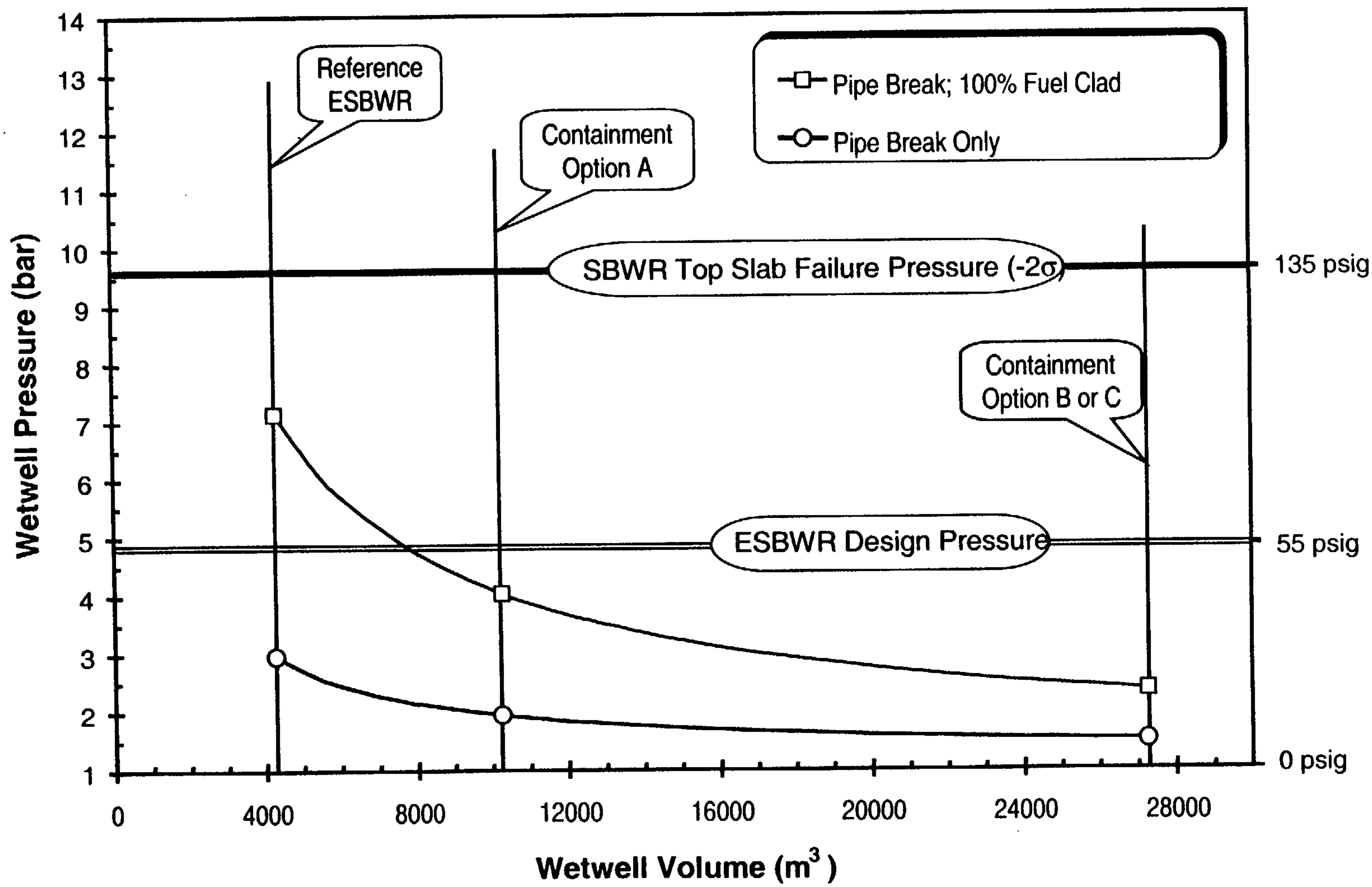
Several interesting options have been identified

Key parameters in Various Options

- **Ways to Reduce Containment Design Pressure**
- **Spent Fuel in Containment or Reactor Building**
  - Horizontal or Inclined Fuel Transfer
  - Stacked Spent Fuel Option
  - Cask Transfer Schemes
  - Size of Spent Fuel Pool
- **Refueling Floor Arrangement**
- **Location of Steam Line**

Several promising choices
All improve margins and reduce building cost

Calculated ESBWR Wetwell Pressures vs. Wetwell Volume



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Key Technology Results and Design Impact

- **Effect of ESBWR Containment Configuration Changes**
  - Allowed Scaleup of Power Without Containment Size Increase
  - Tests Showed Significantly Lower Pressure
- **Effect of Reduced Water Levels in the PCCS Pool**
  - Allowed the Use of a Smaller PCCS Pool, Which Then Kept the Refueling Floor and Building Reasonably Sized
  - Tests Showed That Pool Level (up to a Limit) Has No Effect on Containment Heat Removal and Containment Pressure
- **Effect of Hydrogen on Decay Heat Removal**
  - Allowed the Use a Smaller Containment, Even When Considering Severe Accident Conditions
  - Results Show No Overall Heat Transfer Degradation When Hydrogen Is Present

Technology programs provide confidence in plant design/performance and help reduce costs

Ongoing Technology Programs

- **Quantify Natural Circulation Performance Margins**
  NACUSP Programs at IRI, NRG, CEA and PSI
  Additional Testing at IRI and CRIEPI
  Independent Stability Assessment at ETH, IRI
- **Reduce Uncertainty in Natural Circulation Parameters**
  Chimney Tests at CEA
- **Develop Confidence in Safety System Performance**
  TEMPEST Programs at PSI, VTT, NRG, CEA
- **Develop Back-up Systems to Provide Additional Margin**
  TEMPEST Programs at PSI
- **Provide Additional Data for Code Qualification**

Technology programs to confirm that design is robust

Program Summary and Conclusion

- **8 year ESBWR program**

  Reduced Components and Systems - simplify

  Reduced the Structures and Buildings - simplify

- **8 year Technology Studies**

  Large margins confirmed – increased over SBWR

  Qualified codes for incremental changes for ESBWR

- **Challenges for the Coming Years**

  Crossing the regulatory minefield? hurdles? resources?

Improved Safety/Performance and Economics
Completed Extensive Technology Program
SBWR and ABWR Programs ease Regulatory Challenges

GE Advanced Liquid Metal Reactor

S-PRISM

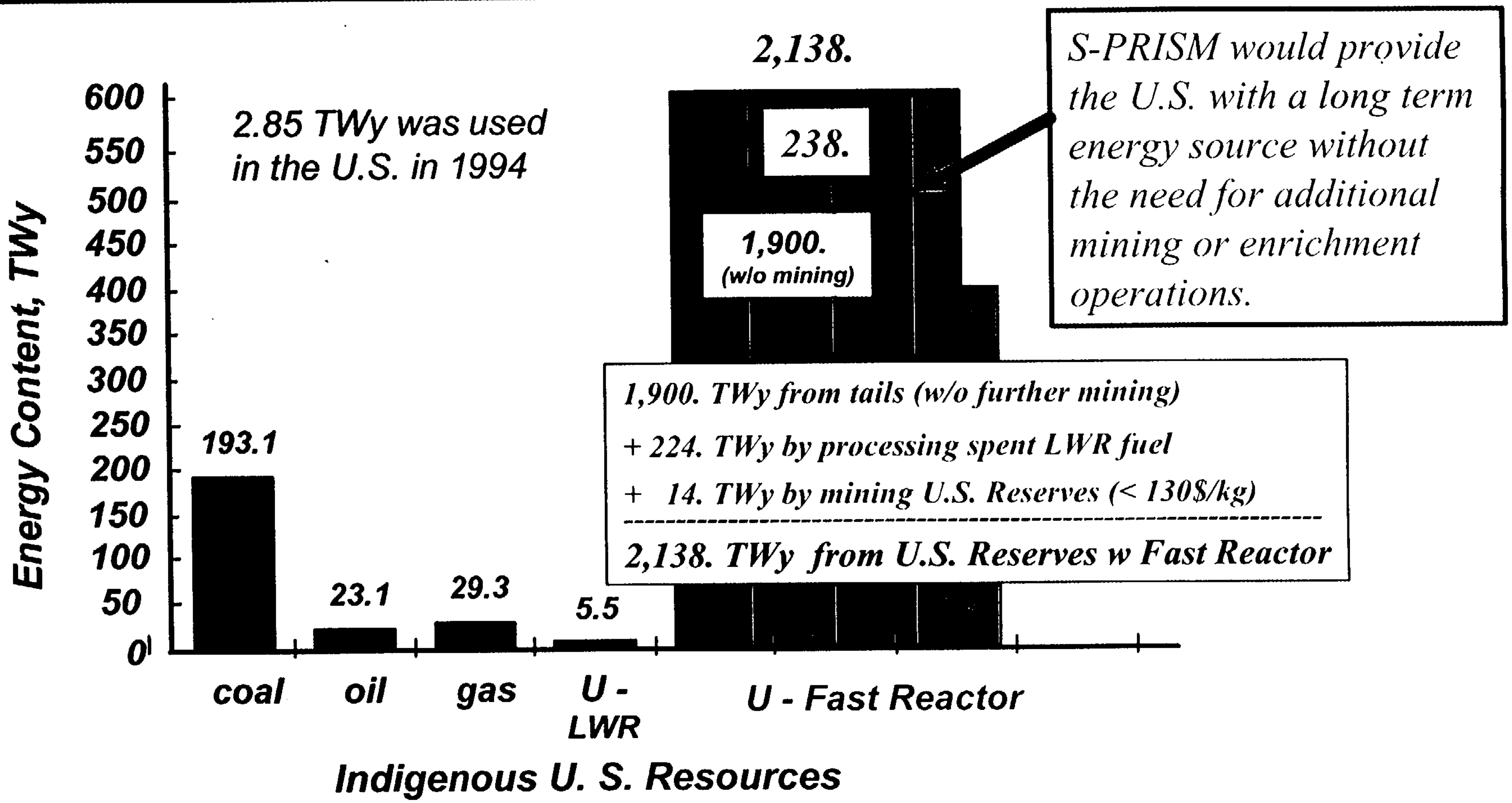
by

C. Boardman
GE Nuclear
San Jose, CA

Topics

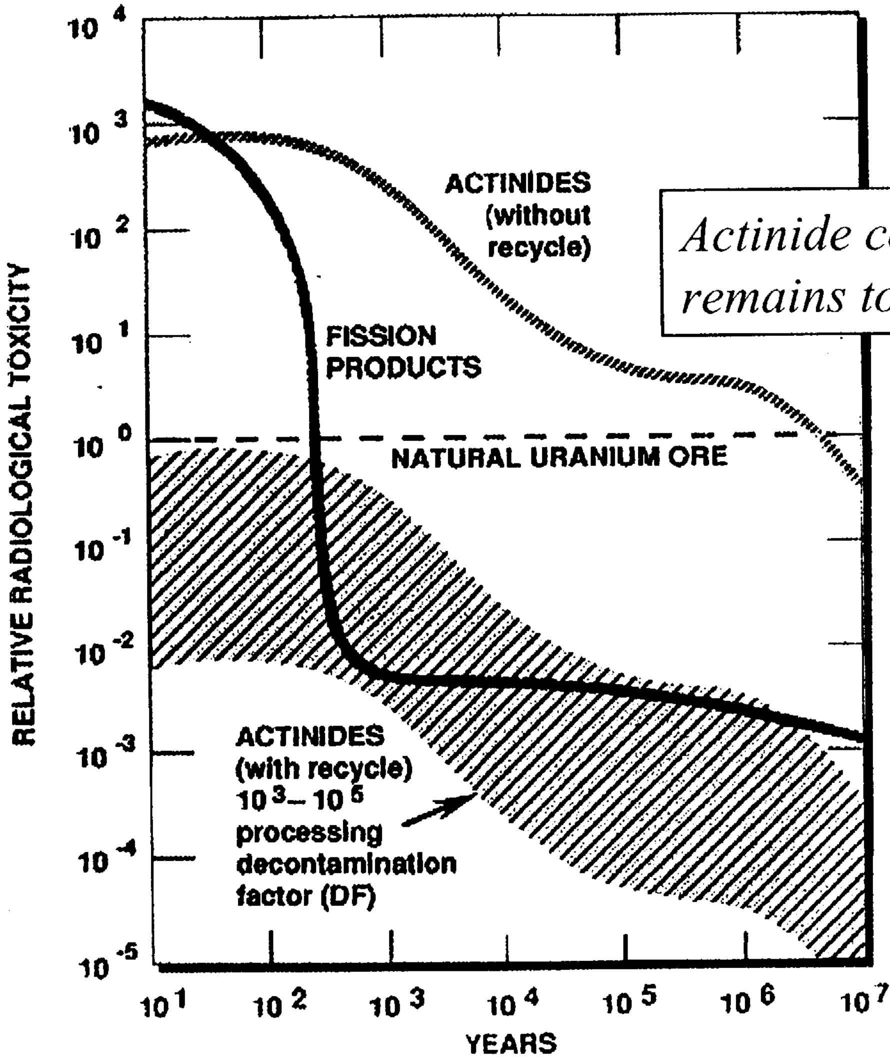
- *Incentive for developing S-PRISM*
- *Design and safety approach*
- *Design description and competitive potential*
- *Previous Licensing interactions*
- *Planned approach to Licensing S-PRISM*
- *What, if any, additional initiatives are needed?*

United States Energy Resources



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".

Diversions would be extremely difficult.

Stage of the Fuel Cycle	Material Barriers					Technical Barriers					
	Isotopic	Radiological	Chemical	Mass and Bulk	Detectability	Facility Unattractiveness	Facility Access	Available Mass	Diversion, Detectability	Skills,Knowledge, Expertise	Time
Phase 1: Fresh fuel fabrication											
Mining											
Milling		Not required						Not required			
Conversion											
Uranium enrichment											
Plutonium storage								Not required			
Transport											
Fuel fabrication		Not required									
Storage											
Transport											
Phase 2: Initial core loading											
Storage of fresh fuel											
Fuel handling		Not required						Not required			
Reactor irradiation											
Phase 3: Equilibrium Operations											
Fuel handling				L			VL	I		M	L
Spent fuel storage				L			M	I		M	L
Head-end processing				M			VL	I		I	L
Fuel processing	VL	VL	L	M	VL	VL	VL	I	VL	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

Co-Located Fuel Cycle Facility

Phase 1
These opportunities for proliferation are not required for S-PRISM.

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

Phase 3
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 and 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*

Topics

- *Incentive for developing S-PRISM*
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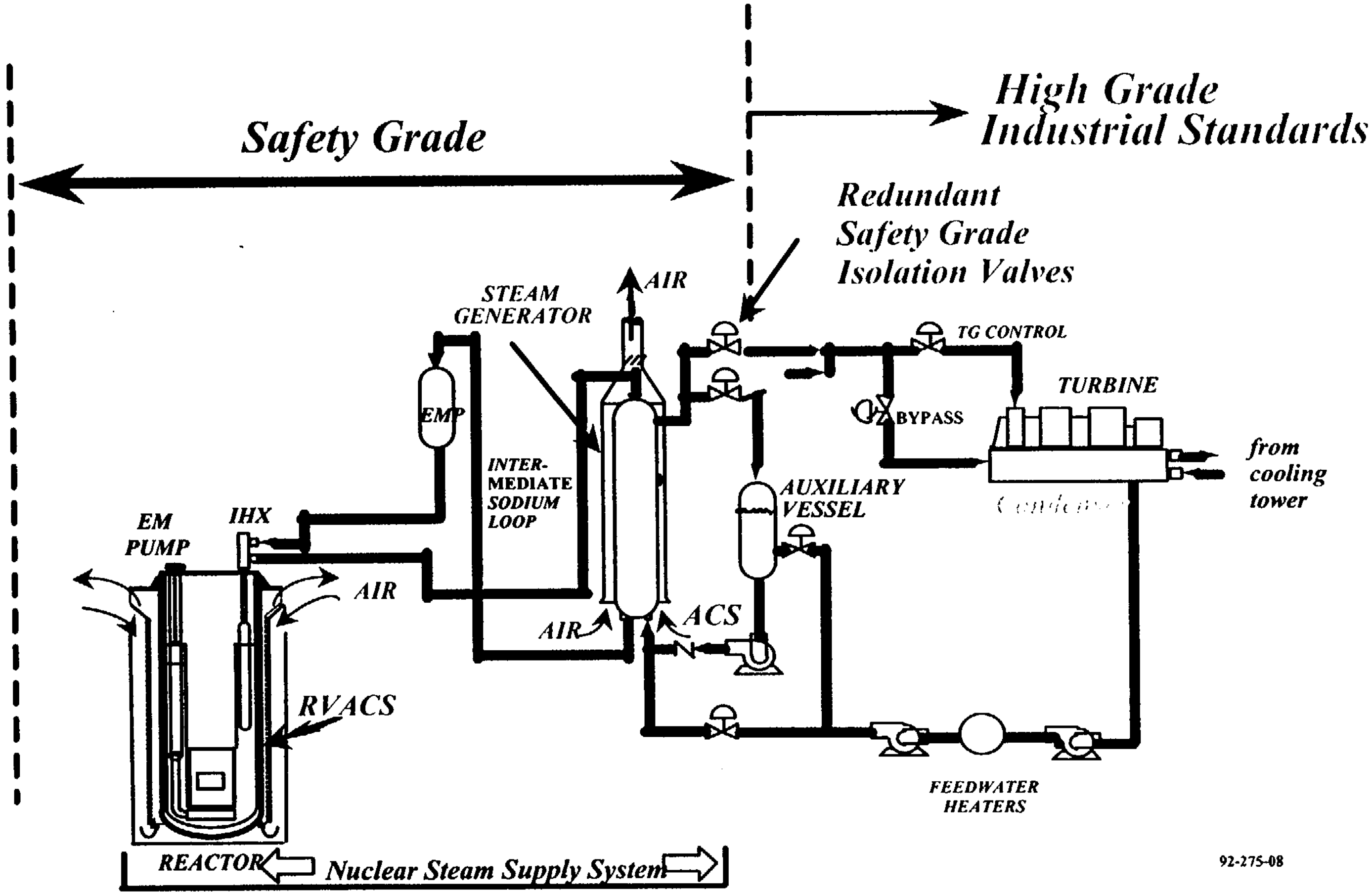
Power Train



RVACS	ACS	Condenser

Shutdown Heat Removal Systems

S-PRISM - Principal Design Parameters

Reactor Module

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

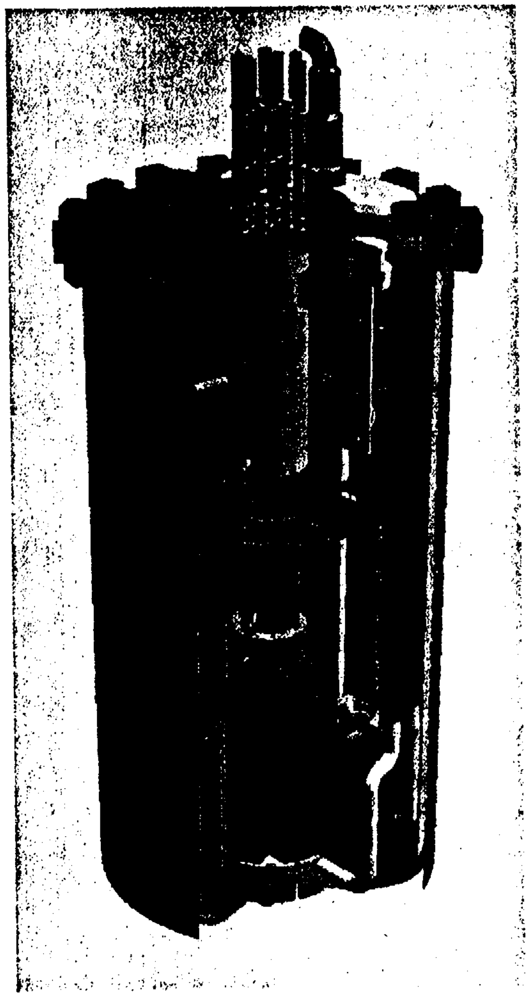
Power Block

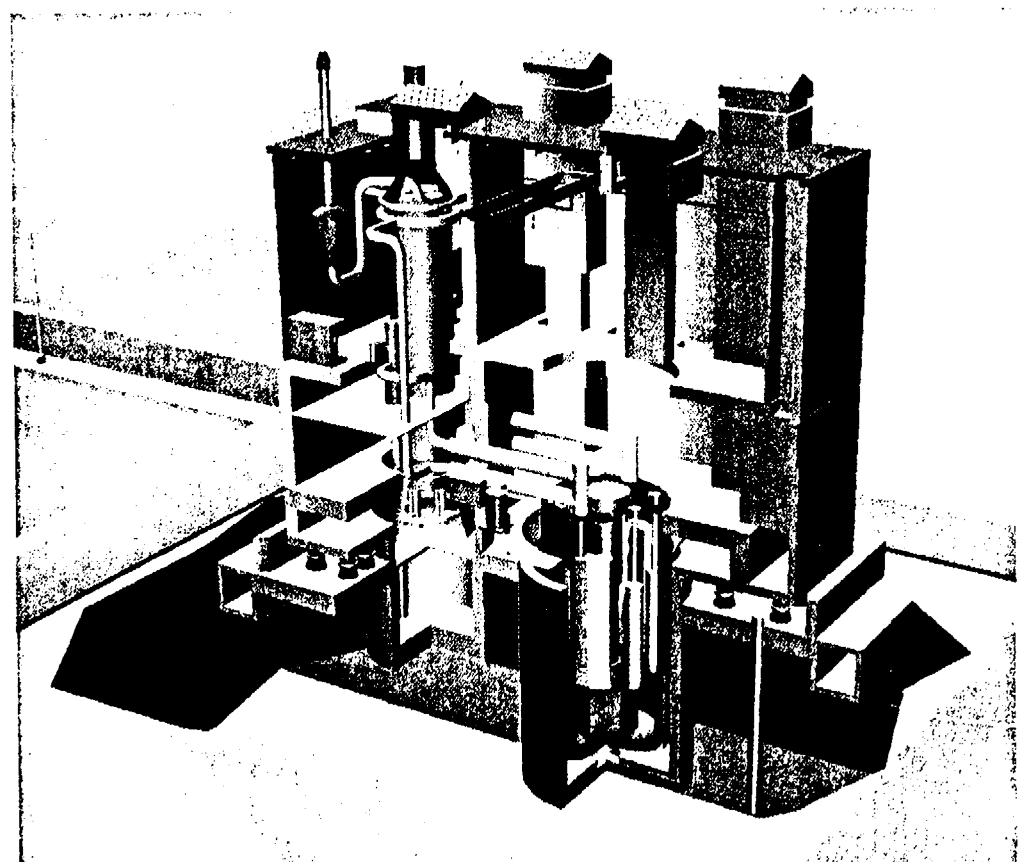
- Number of Reactors Modules	2
- Gross/Net Electrical, MWe	825/760
- Type of Steam Generator	Helical Coil
- Turbine Type	TC-4F 3600 rpm
- Throttle Conditions, atg/C	171/468
- Feedwater Temperature, C	215

Overall Plant

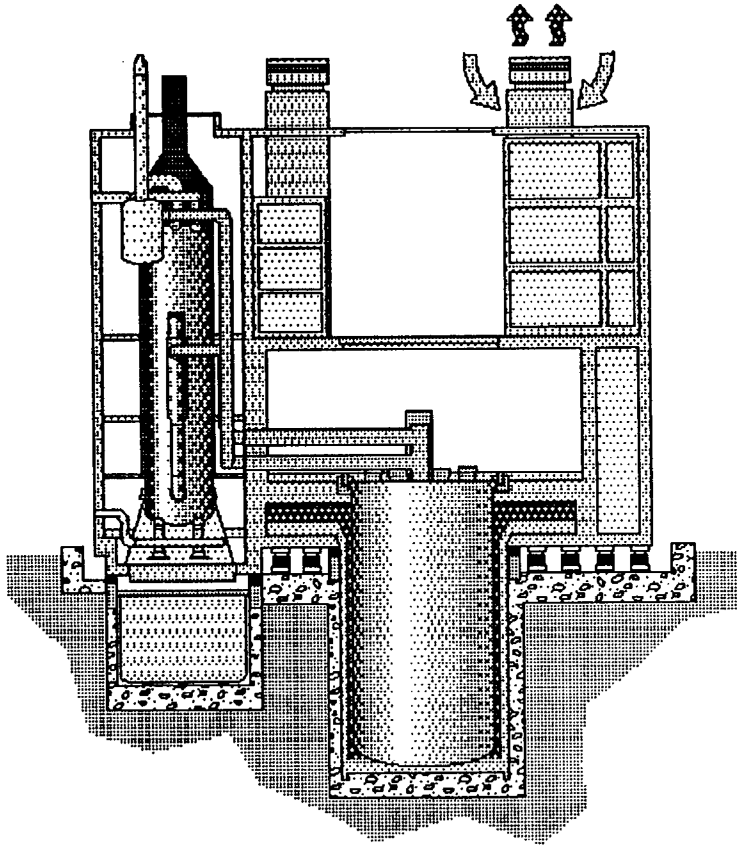
- Gross/Net Electrical, MWe	2475/2280
- Gross/Net Cycle Efficiency, %	41.2/38.0
- Number of Power Blocks	3
- Plant Availability, %	93

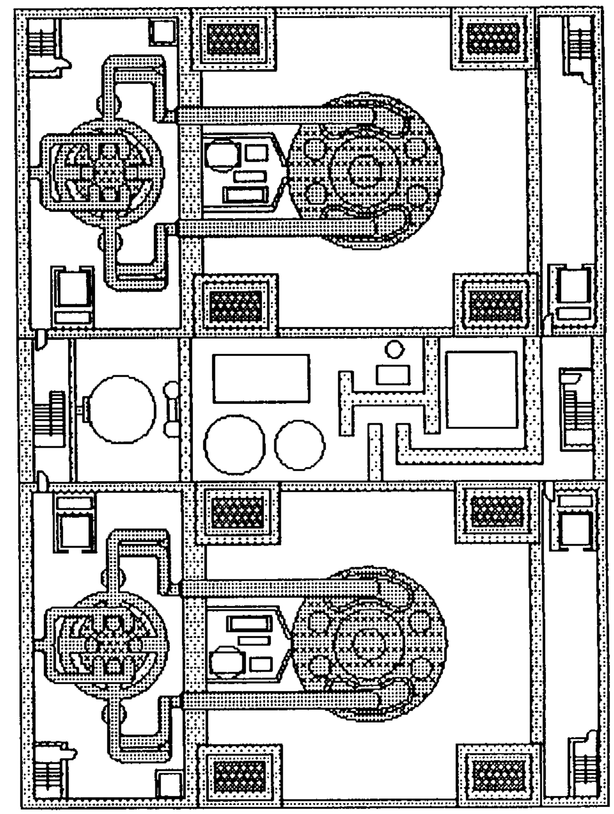
Super PRISM





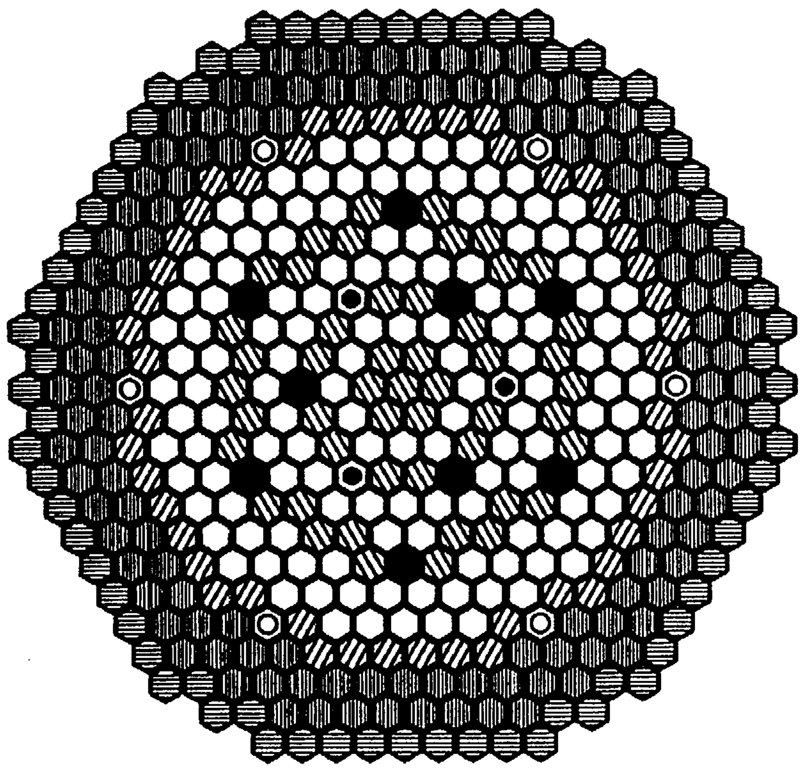
S-PRISM Power Block (760 MWe net)





Two 380 MWe NSSS per Power Block

Metal Core Layout



Number of Assemblies

Driver Fuel	138
Internal Blanket	49
Radial Blanket	48
Primary Control	9
Secondary Control	3
Gas Expansion Module	6
Reflector	126
Shield	72
Total	451

Fuel: 23 month x 3 cycles

Blkt: 23 month x 4 cycles

Oxide vs. Metal Fuel

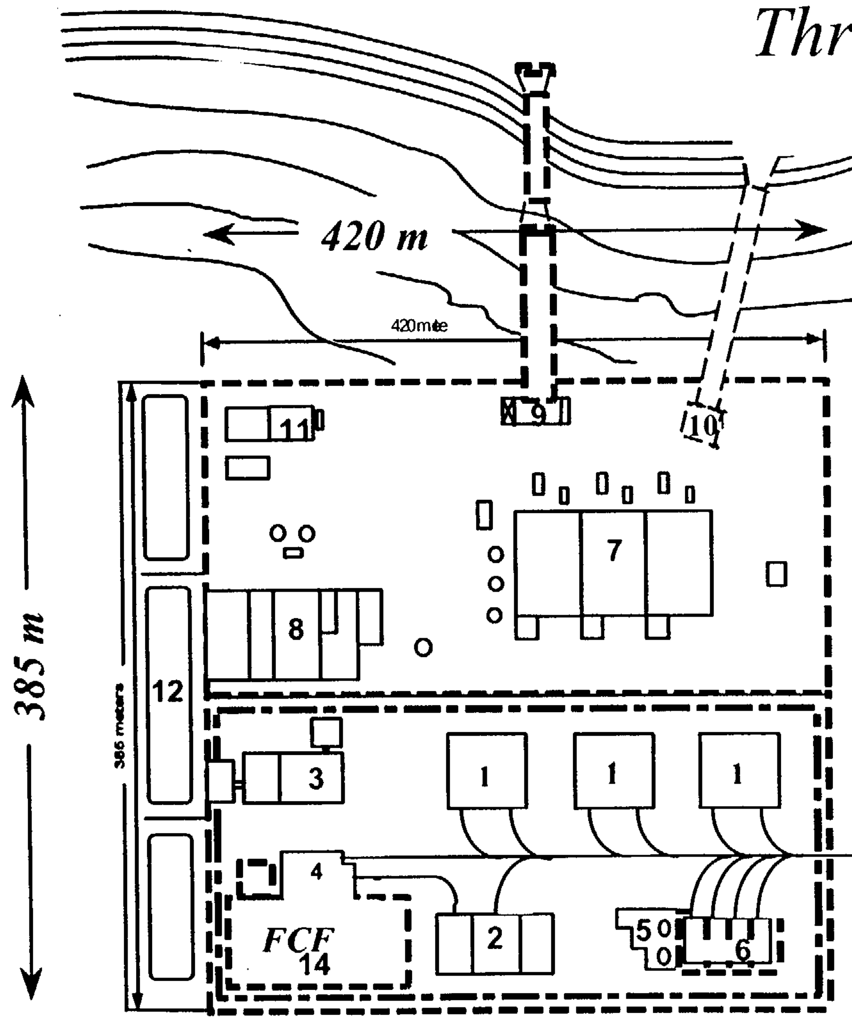
- *Attractive features of metal core include:*
  - *fuel is denser and has a harder neutron spectrum*
  - *compatible with coolant, RBCB demonstrated at EBR-II*
  - *axial blankets are not required for break even core*
  - *high thermal conductivity (low fuel temp.)*
  - *lower Doppler and harder spectrum 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

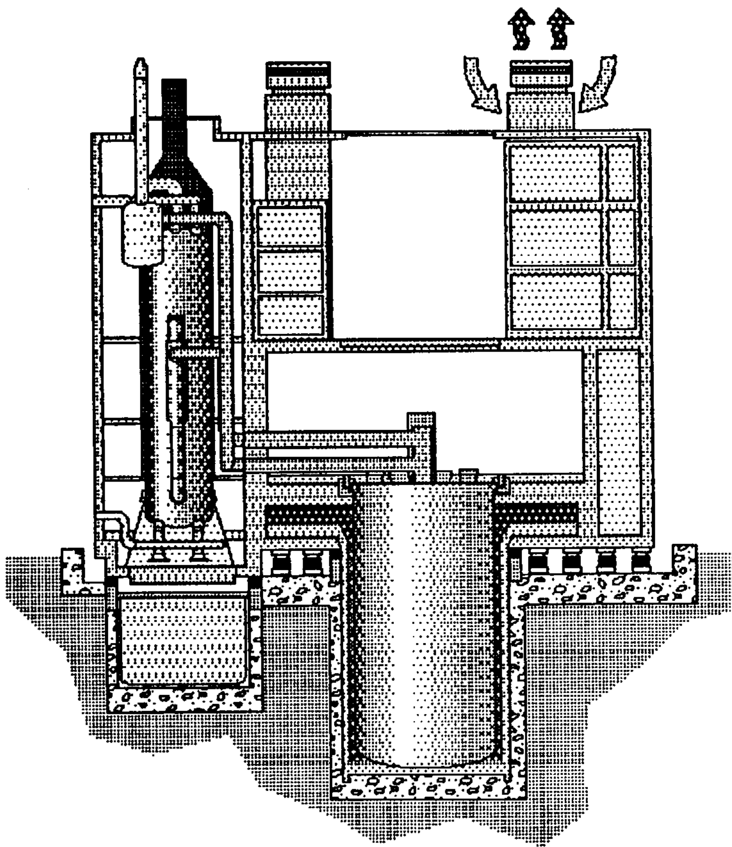
Three Power Block Plant

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



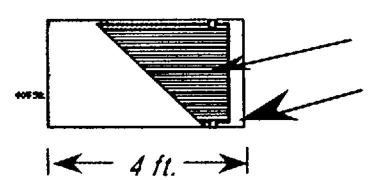
Characteristics of Seismic Isolation System

- ***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.***

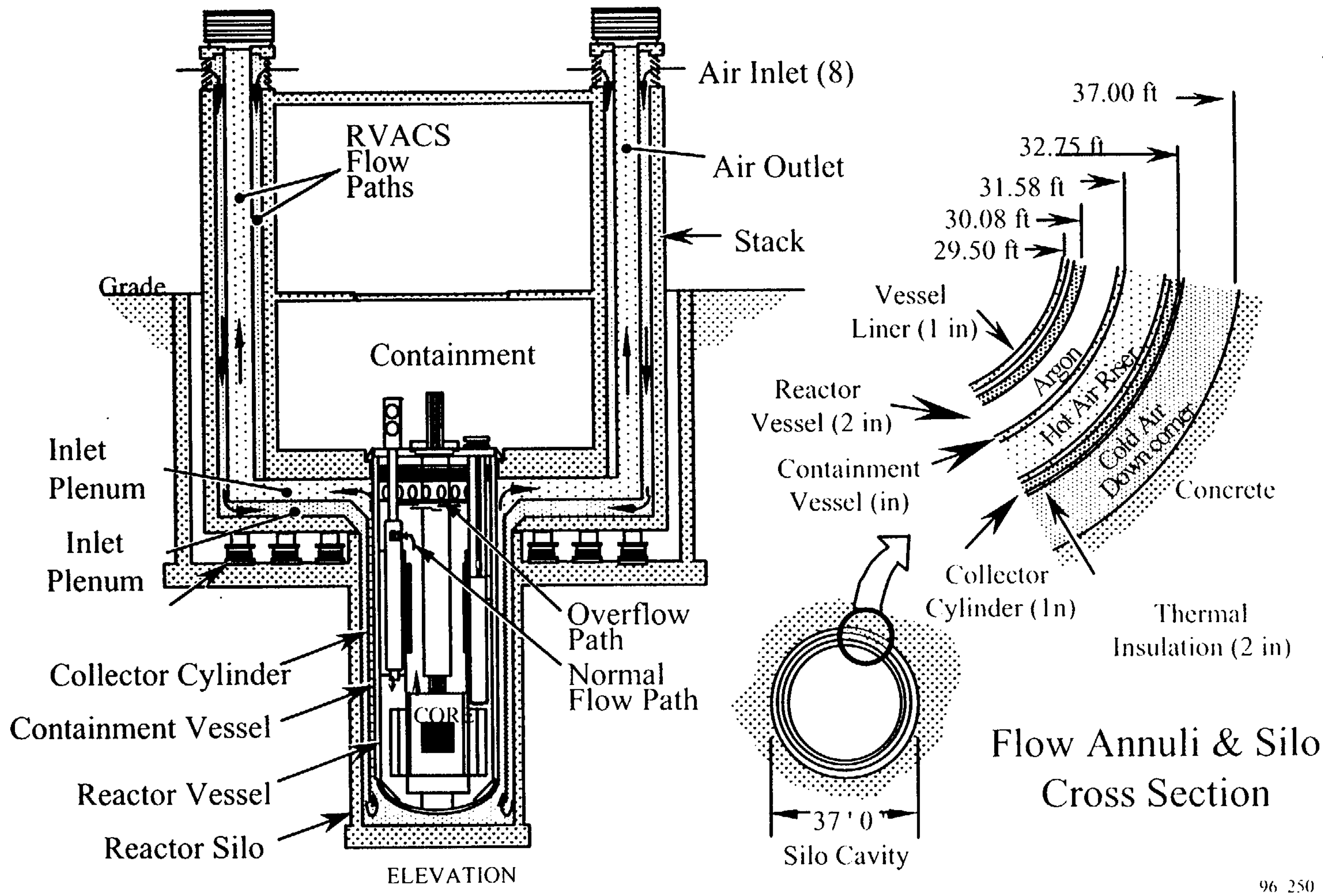
- ***Natural Frequency***
  - ***Horizontal*** ***0.70 Hz***
  - ***Vertical*** ***21 Hz***

- ***Lateral Load Reduction > 3***

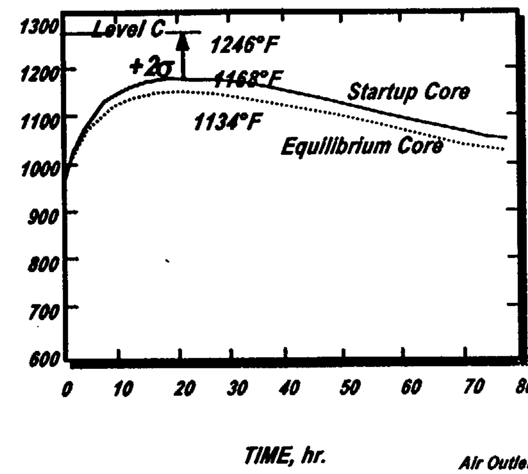


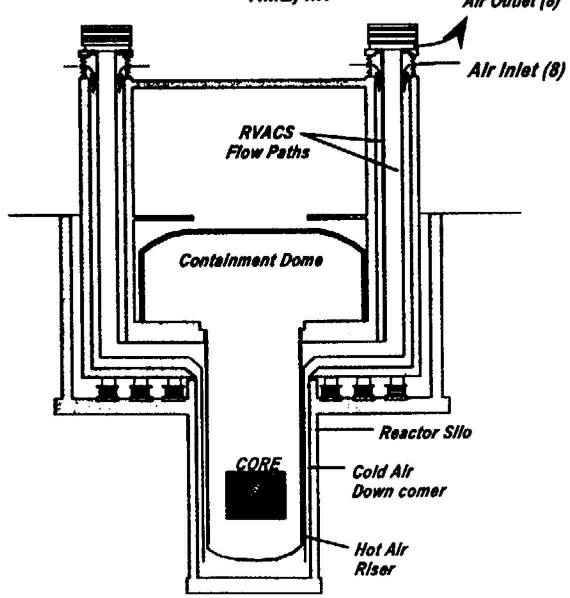
Seismic Isolators (66)

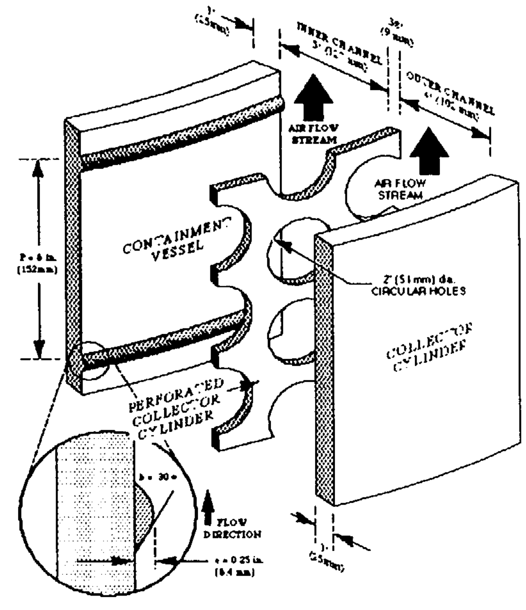
Reactor Vessel Auxiliary Cooling System (RVACS)



Passive Shutdown Heat Removal (RVACS)

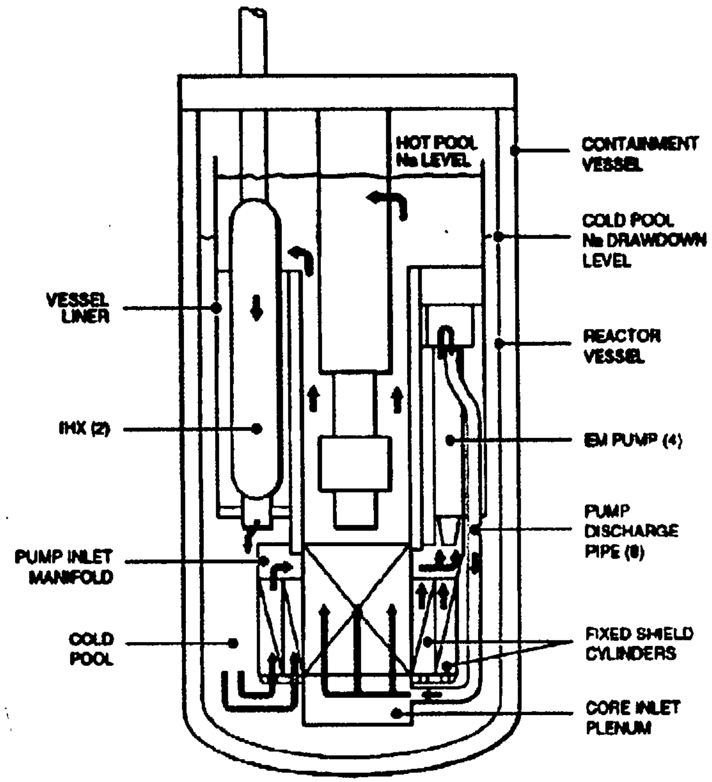




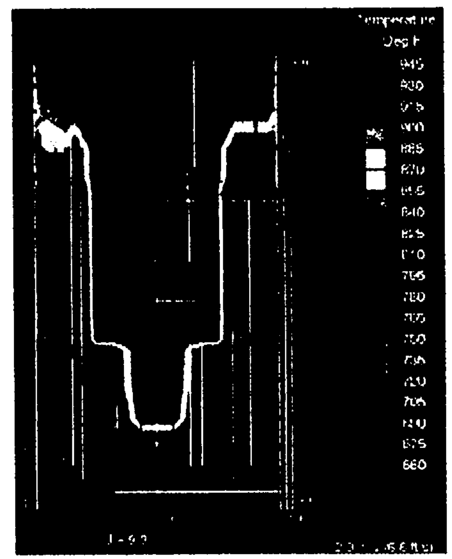


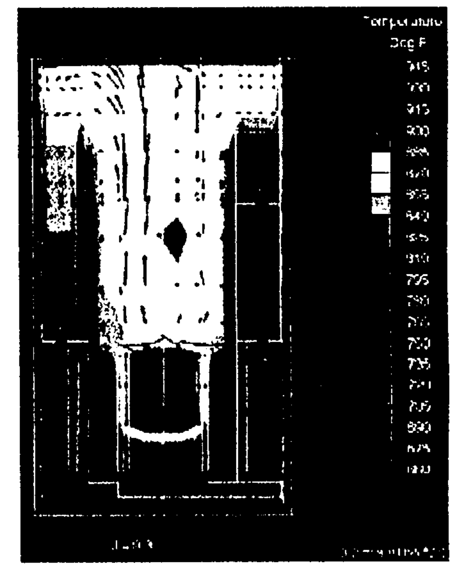
ENHANCED RVACS HOT AIR RISER WITH BOUNDARY LAYER TRIPS
AND PERFORATED COLLECTOR CYLINDER

Natural Circulation Confirmed by 3 Dimensional T/H Analysis



Normal Operation

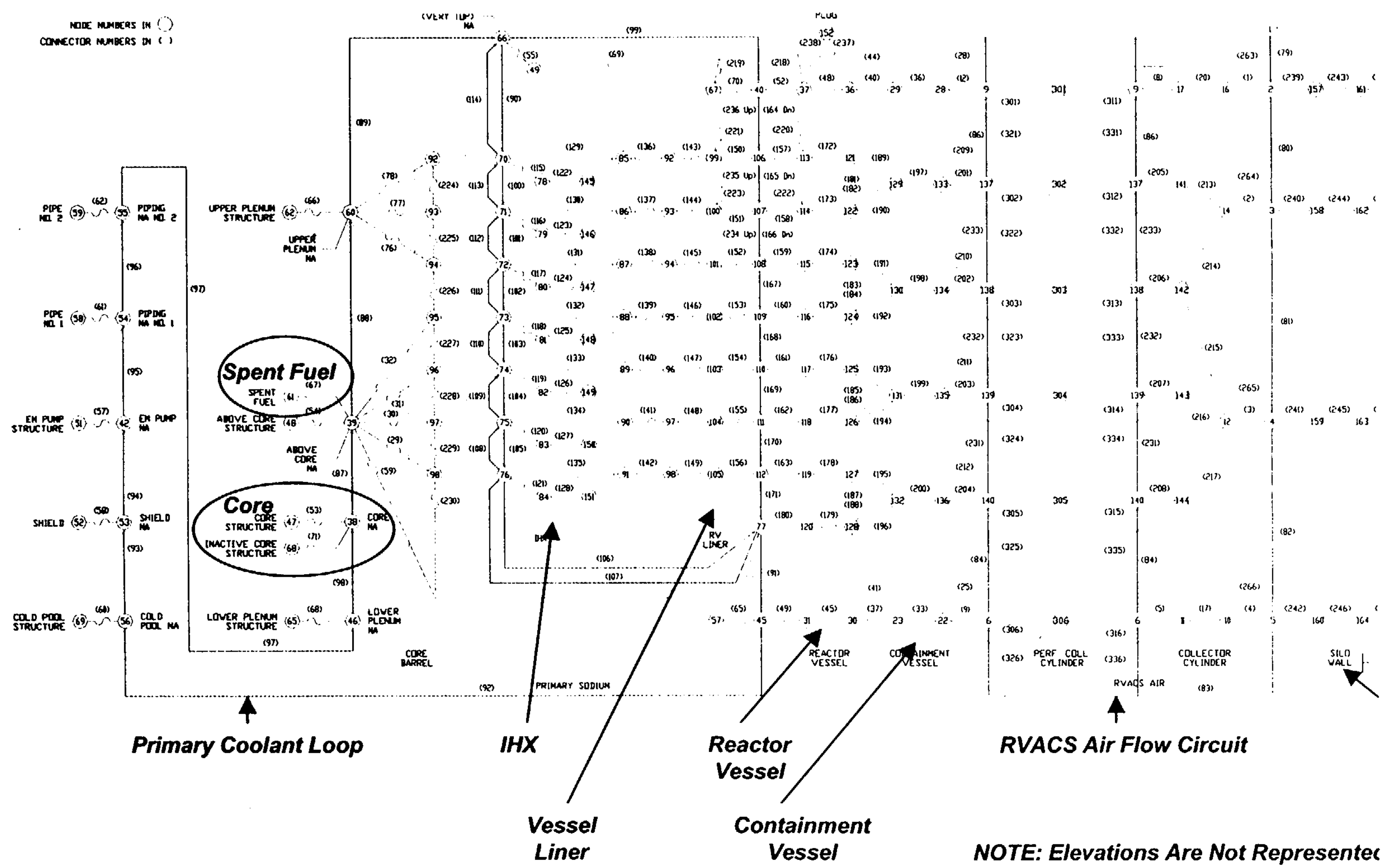




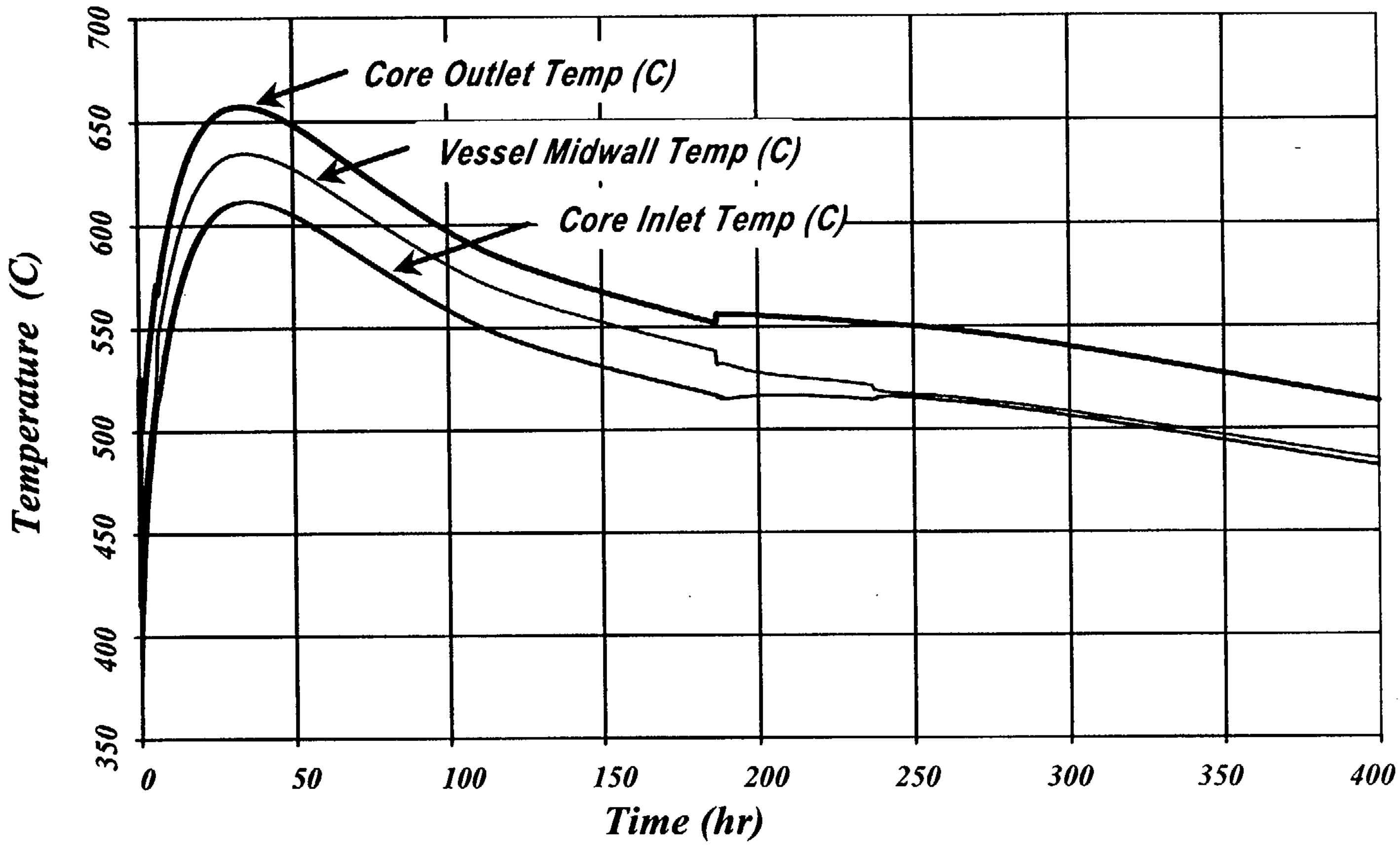
Examples

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

Decay Heat Removal Analysis Model

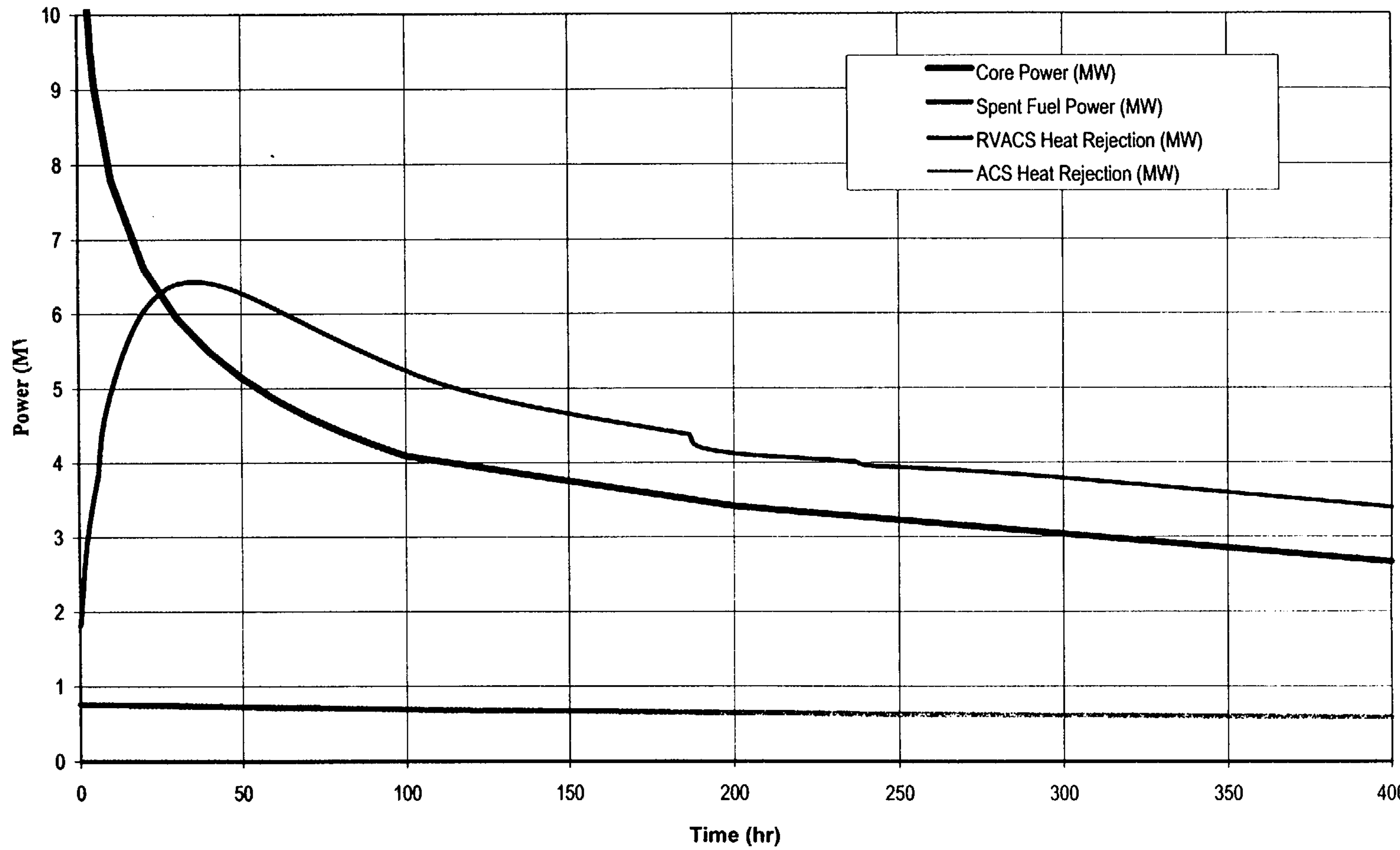


RVACS Cooling - Nominal System Temperatures

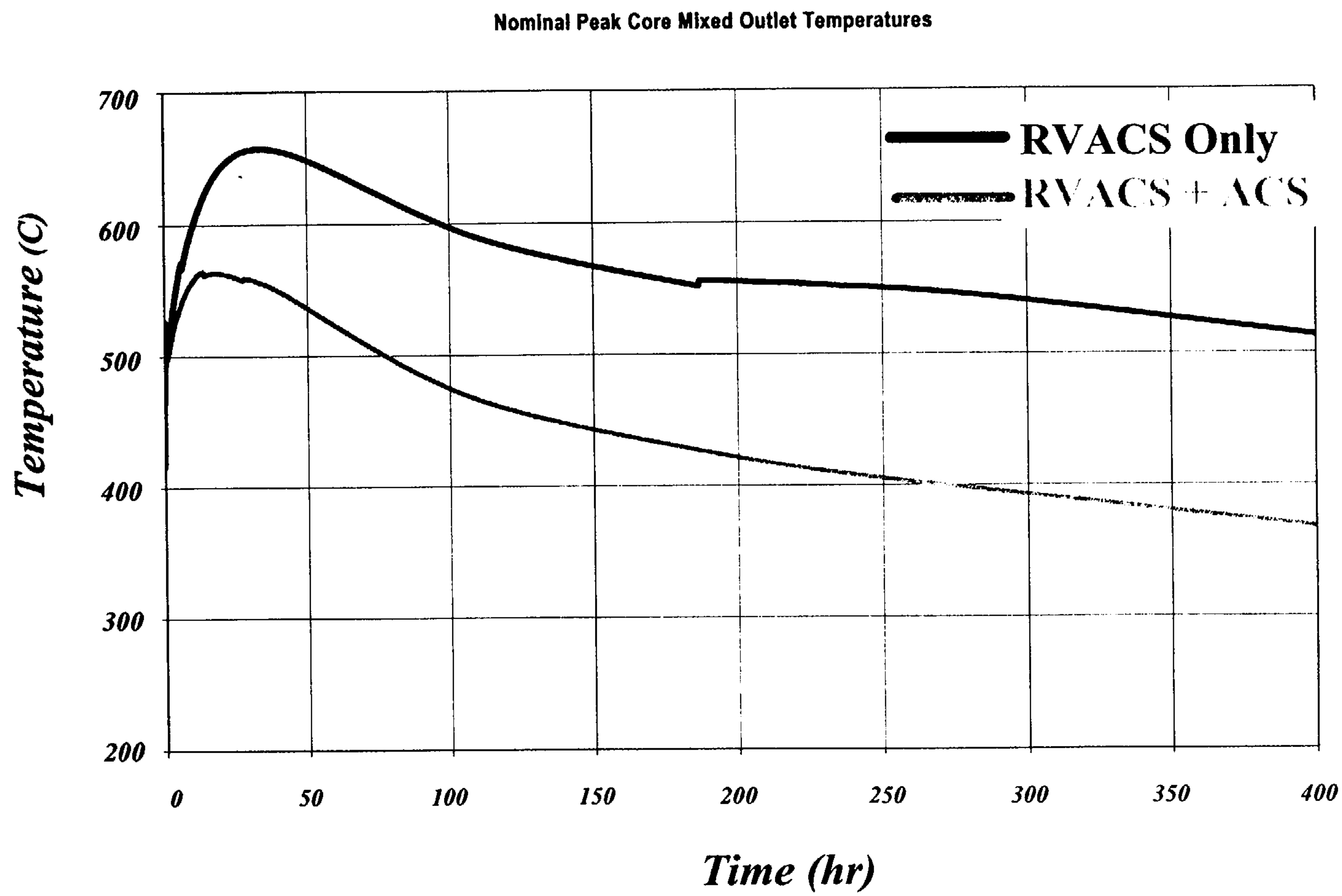


RVACS Transients Are Slow Quasi Steady State Events

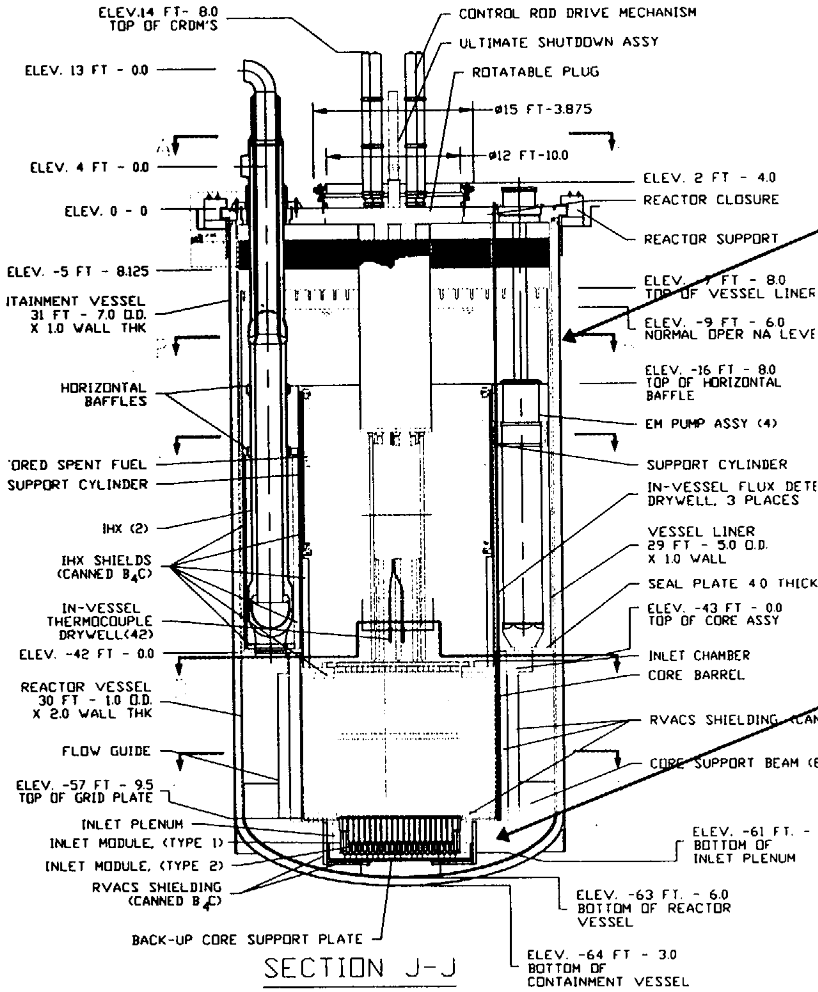
RVACS Heat Rejection and Heat Load versus Time



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.

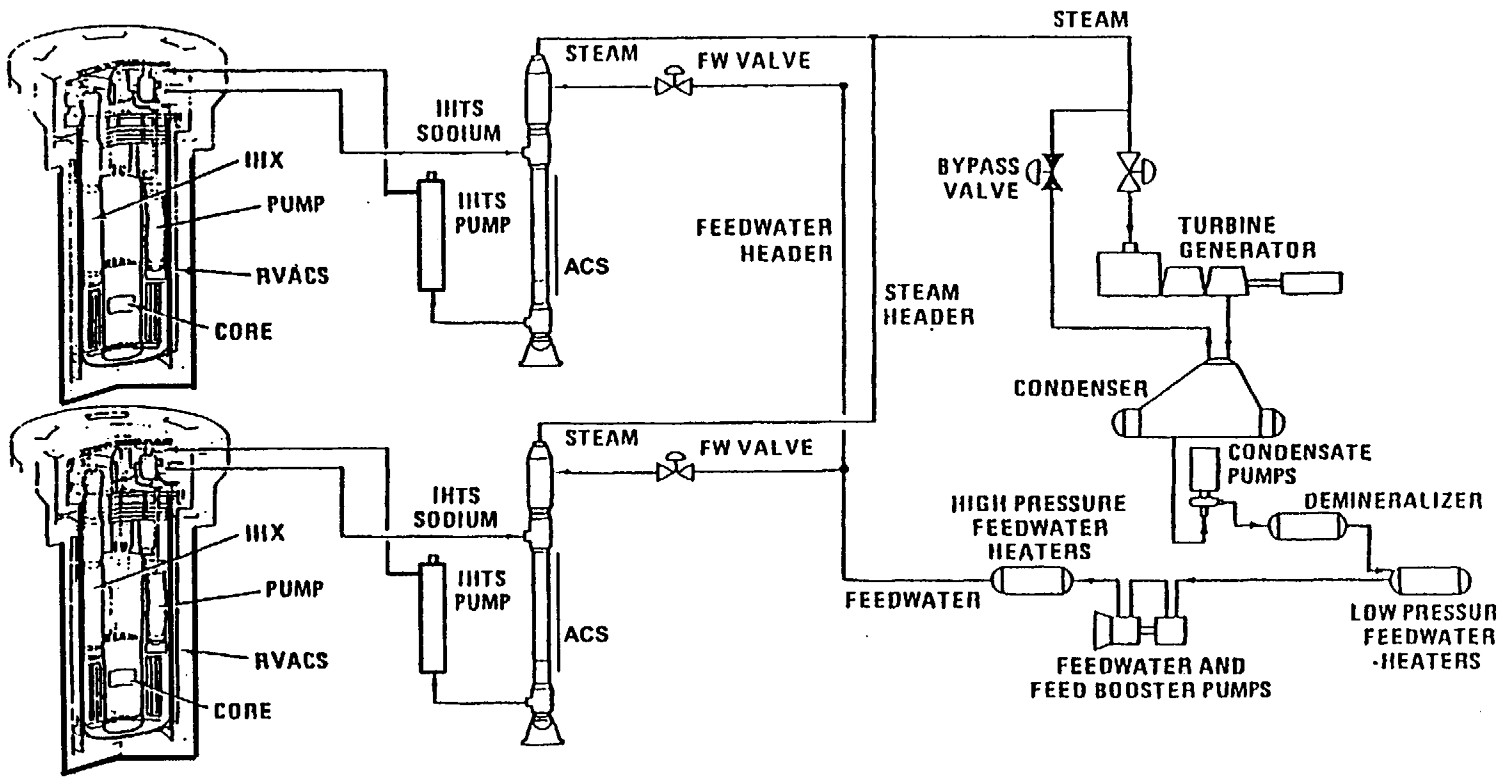
- *Loss of Normal Heat Sink*
- *Loss of Forced Flow*
- *Loss of Flow and Heat Sink*
- *Transient Overpower w/o Scram*

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

S-PRISM Requirement:

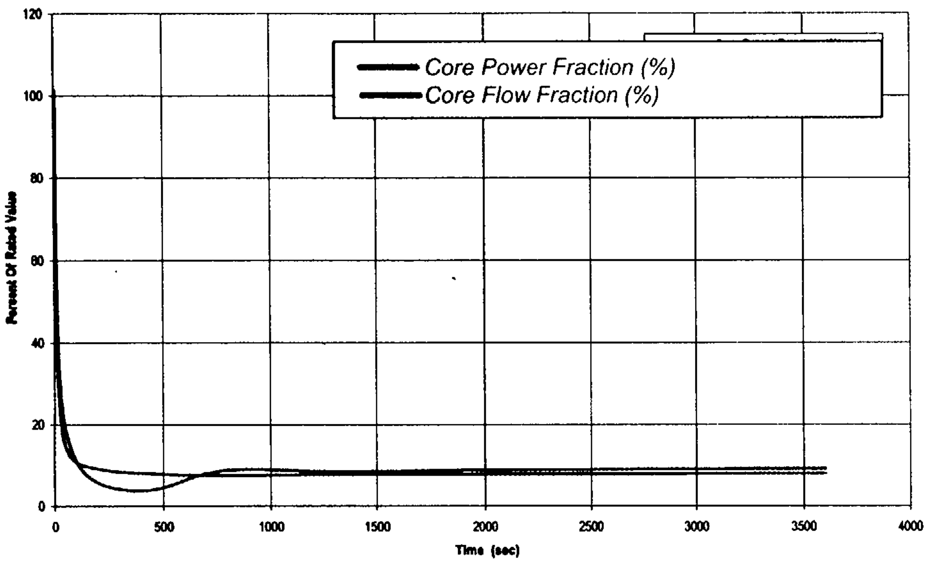
Accommodate the above subset of events w/o loss of reactor integrity or radiological release using passive or inherent natural processes. A loss of functionality or component life-termination is acceptable.

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***

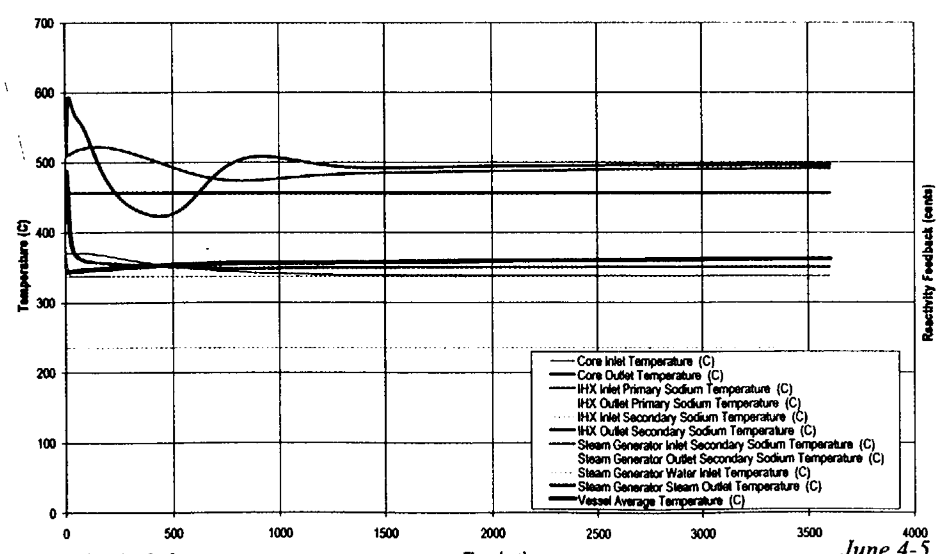
Example ATWS - Loss Of Flow Without Scram



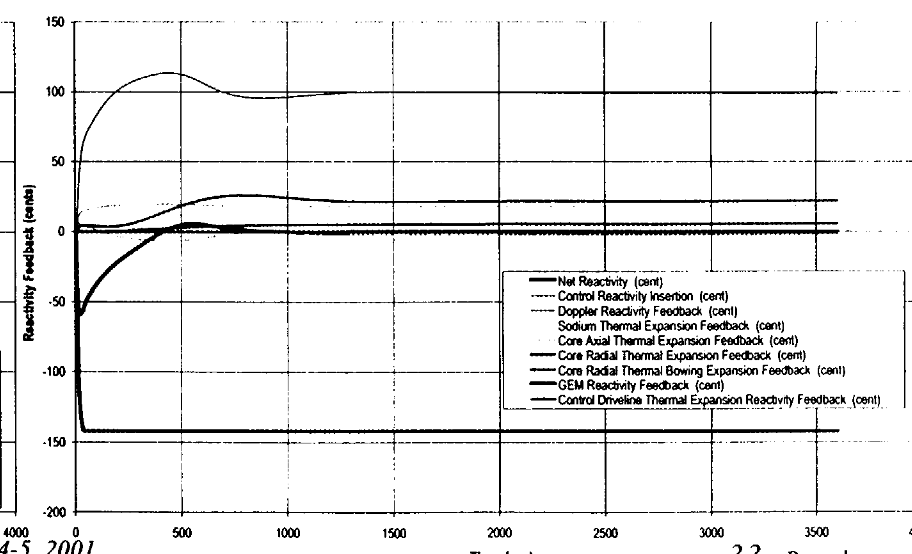
Loss of Primary Pump Power w/o Scram

- *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*

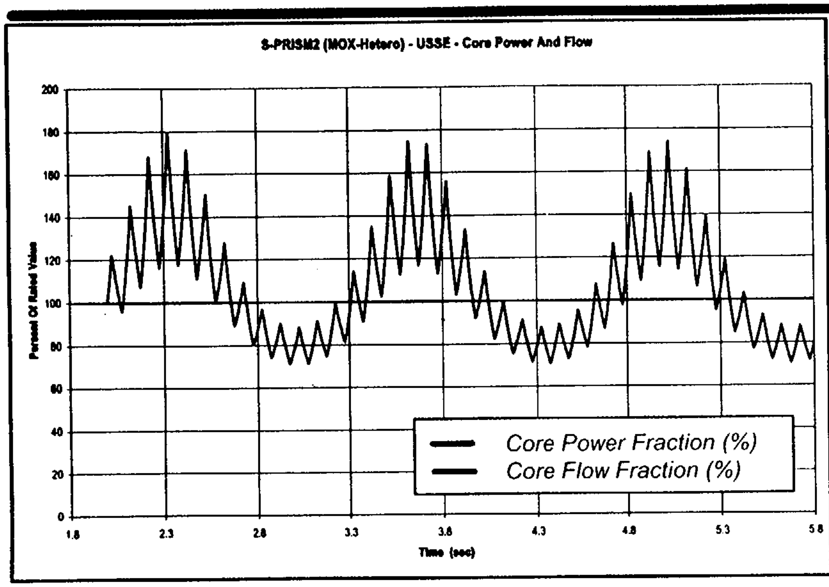
S-PRISM2 (MOX-Hetero) - ULOF - System Temperatures



S-PRISM2 (MOX-Hetero) - ULOF - Reactivity Feedback



Example - 0.5 g ZPA Seismic Event Without Scram



- ***Reactivity:***

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

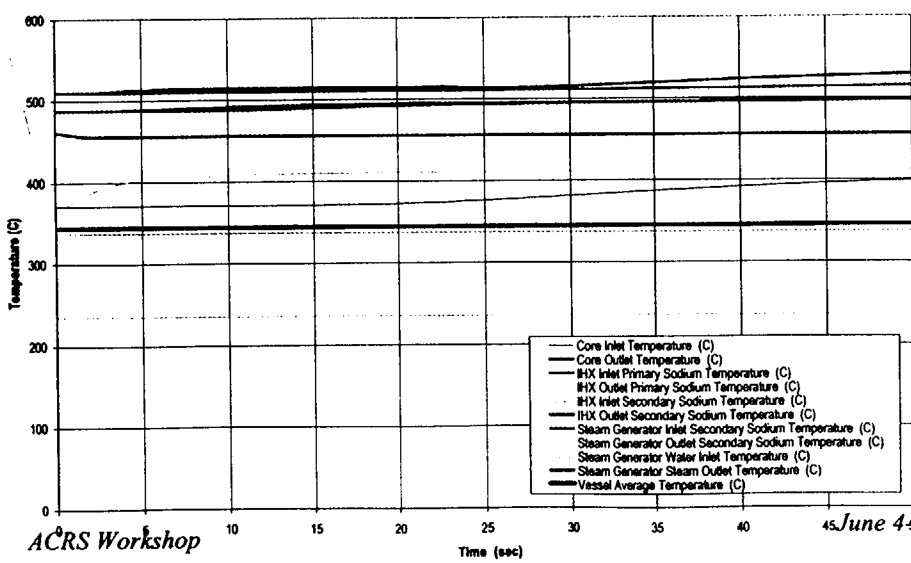
  + - 0.16$ at 10 Hz (vertical CR-core motion with opposite phases)

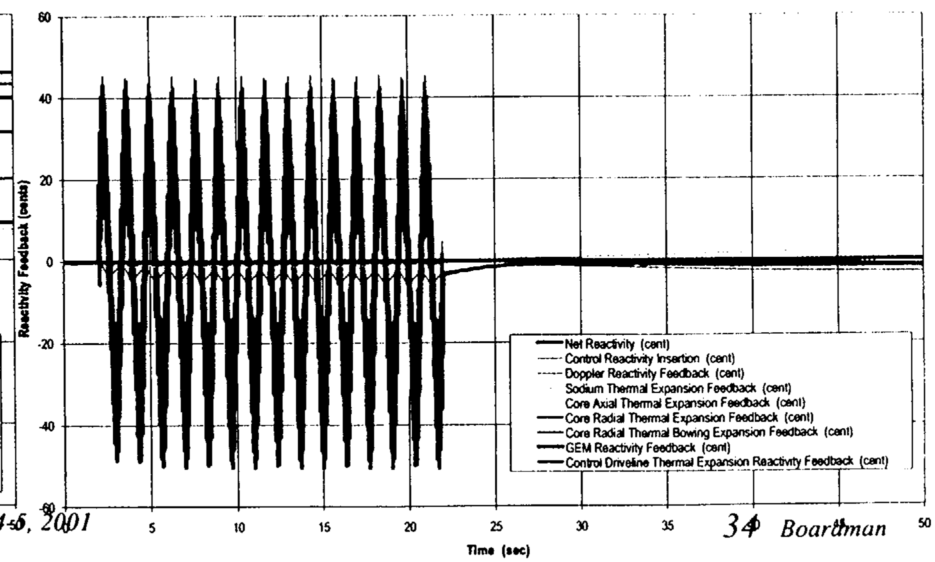
- ***Power oscillations to 180%, short duration, not supercritical***

- ***Fuel heat capacity absorbs power oscillation without melting***

- ***Fuel releases heat to structures slowly and gives small Doppler feedback to reduce power peaks***

S-PRISM2 (MOX-Hetero) - USSE - System Temperatures



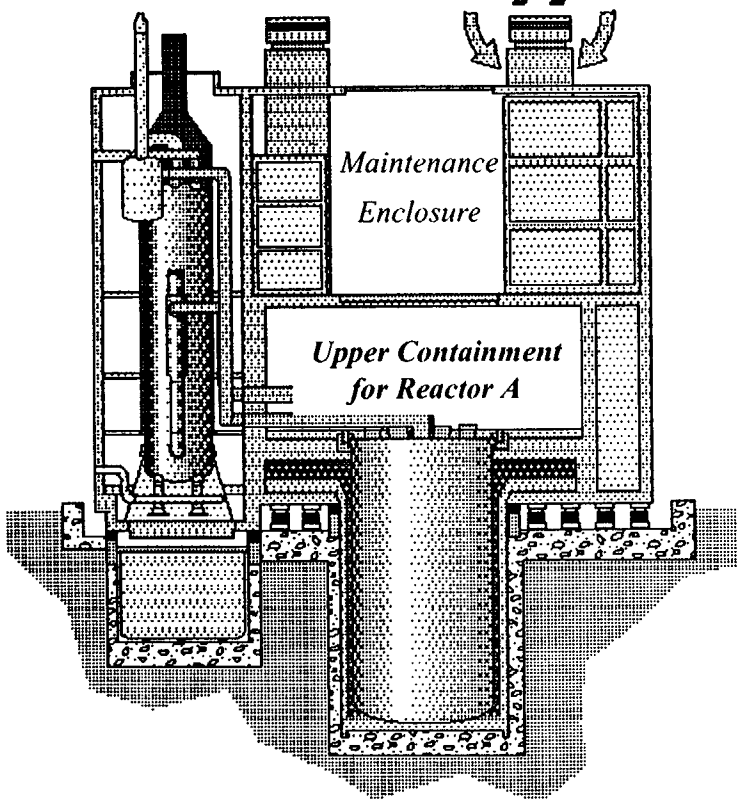


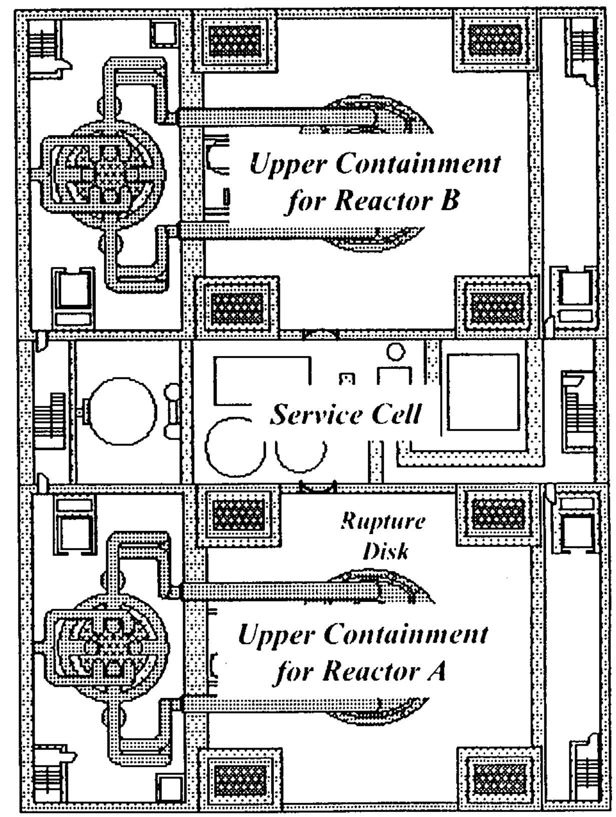
S-PRISM Transient Performance Conclusions

S-PRISM tolerates ATWS events within the safety performance limits

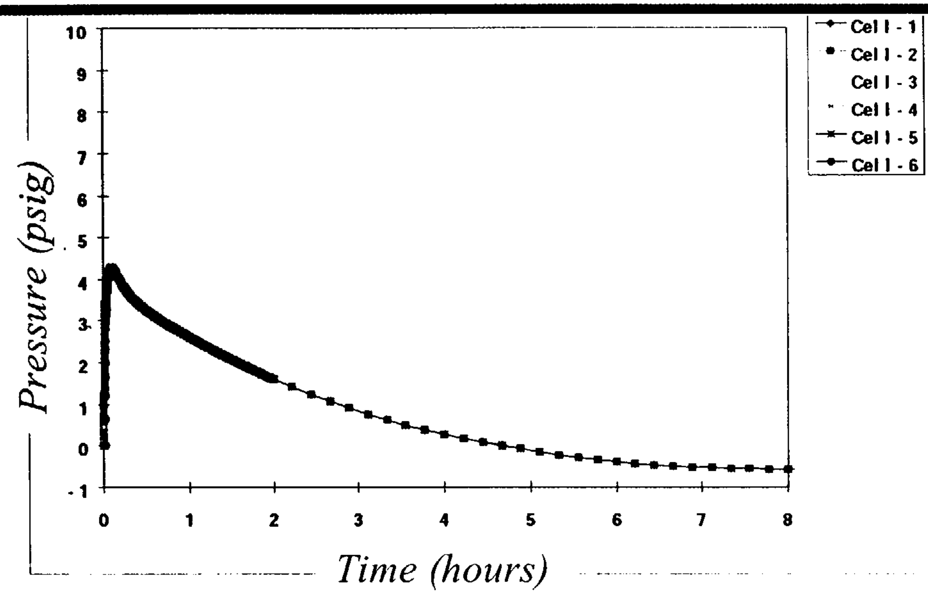
The passive safety performance of S-PRISM is consistent with the earlier ALMR program

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

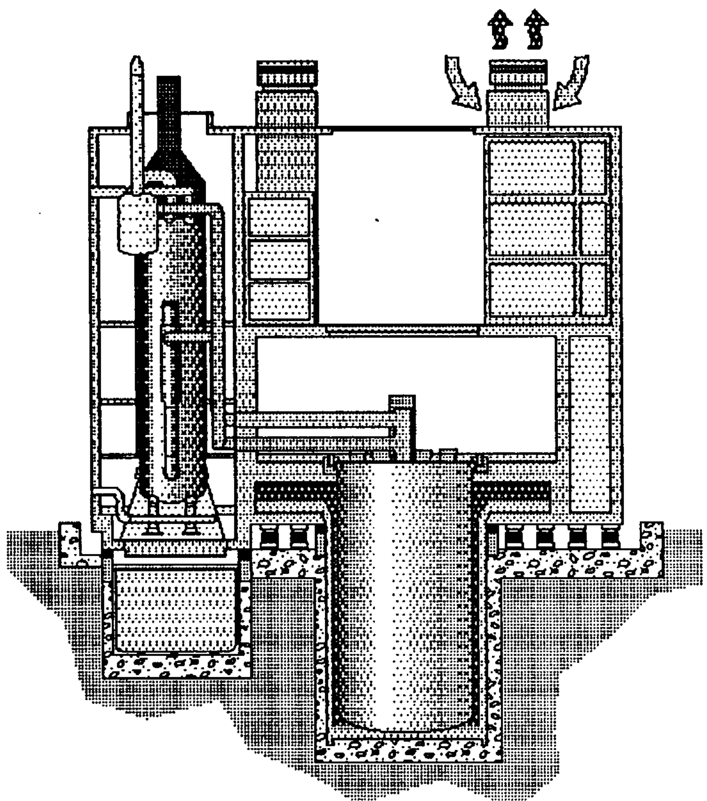
---

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	Most Rods Must Function
	Self Actuated Scram on Secondary Rods	Boron injection
	Passive Accommodation of ATWS Events	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

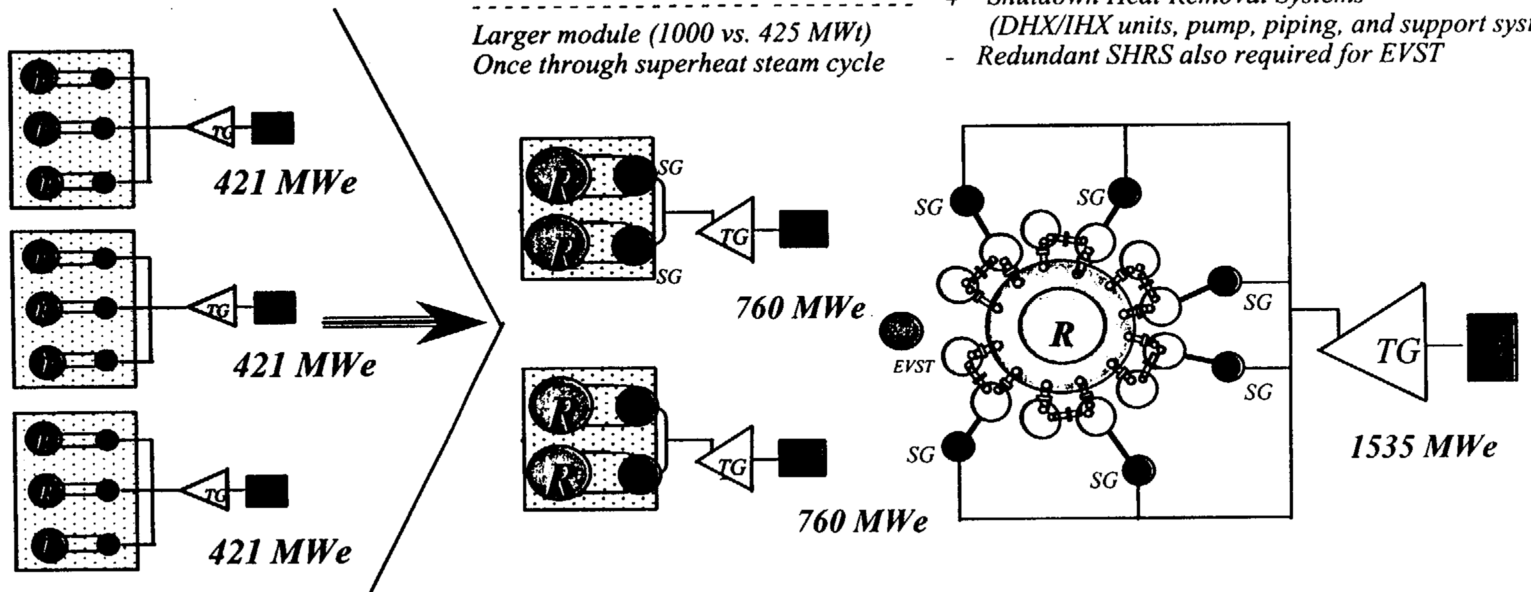
Parameter or Feature	1995 ALMR	S-PRISM
Core Power, MWt	840.	1000.
Core Outlet Temp, °C	499	510
Main Steam, °C / kg/cm^2	454/153	468/177
Net Electrical, MWe (two power blocks)	1243.	1520
Net Electrical, MWe (three power blocks)	1866	2280
Seismic Isolation	Yes. Each NSSS placed on a separate isolated platform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

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?*

Optimizing the Plant Size

1988 PRISM ⟹ S-PRISM		**Large Commercial Design**
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*	*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*	*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*

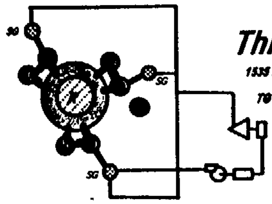


Simplicity allows Reduction in Commodities and Building Size

Scale Up -- LWR versus Fast Reactor

1600 MWt Sodium Cooled Fast Reactor

Three 533 MWt Loops

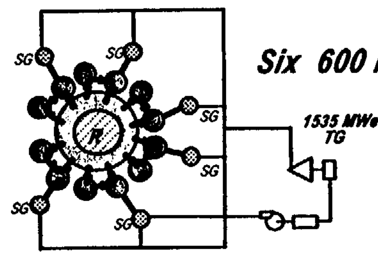


1600 MWt Light Water Cooled Reactor

Two 800 MWt Loops

3600 MWt FR

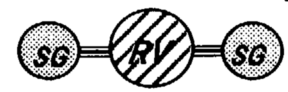
Six 600 MWt Loops



Rating Limited by:
IHTS Piping: < 1 m diameter

3600 MWt PWR

Two 1800 MWt Loops

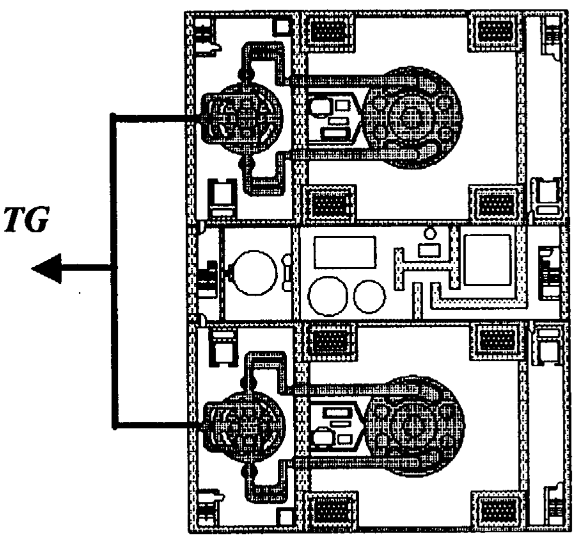


Two Loops Viable Because:
Specific heat of water 5 x sodium at operating temperatures

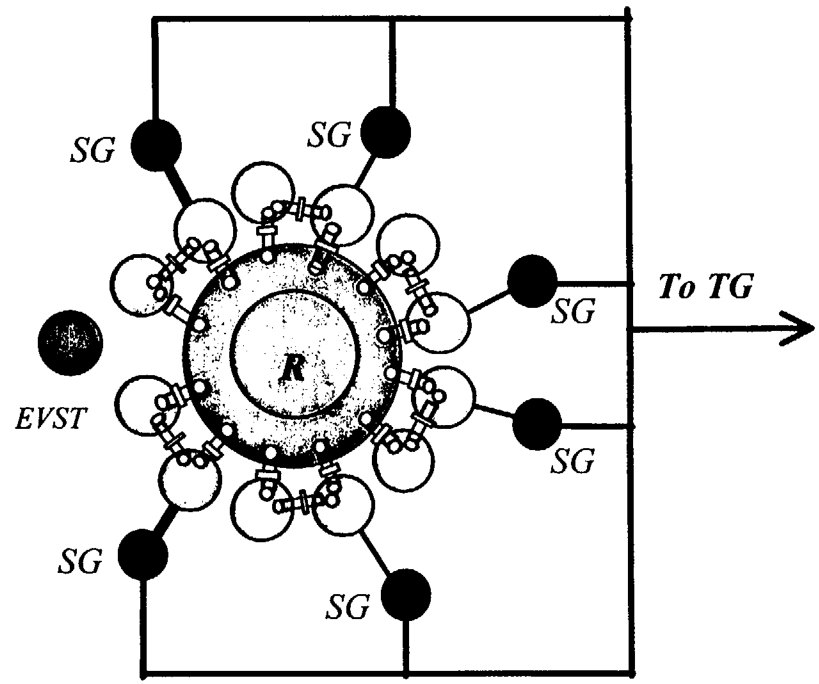
- ***The complexity and availability of a PWR is essentially constant with size***
- ***Due to the lower specific heat of sodium, six or more loops are required in a large FR.***

The Economy of Scale is Much Larger for LWRs then FBRs

Modular versus Monolithic (Fast Reactors)



Modular (S-PRISM)

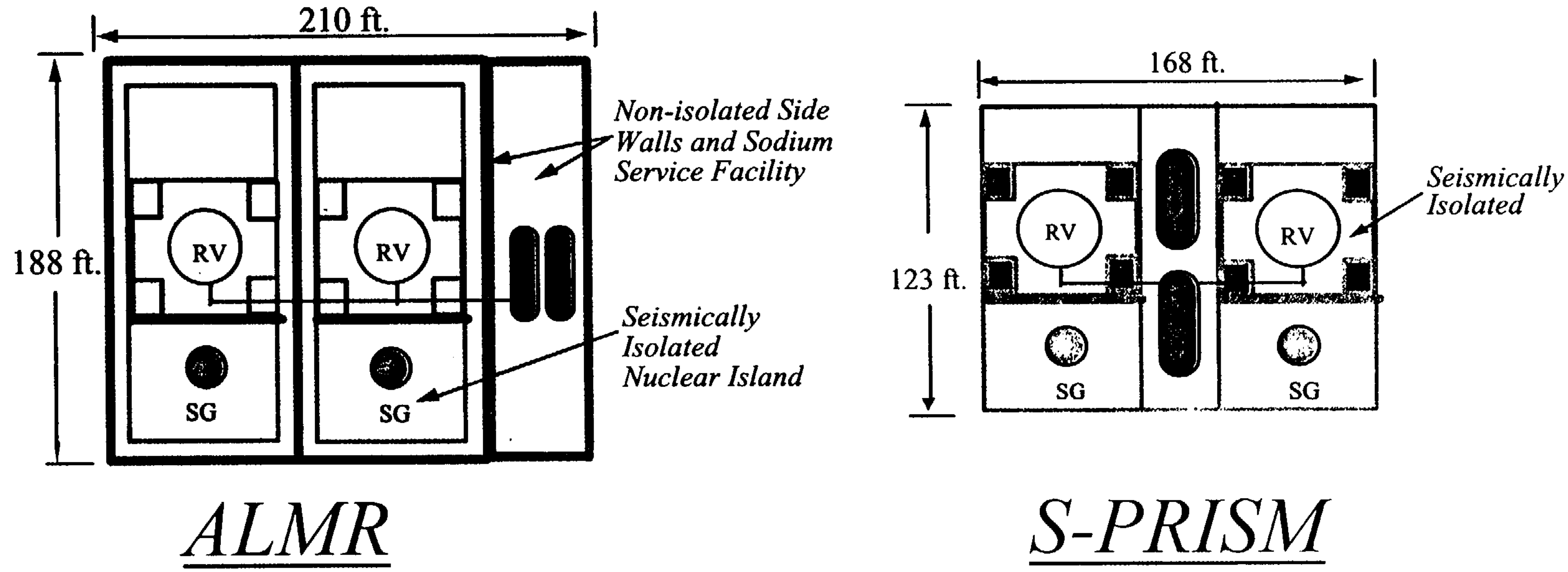


Monolithic Fast Reactor

The one-on-one arrangement:

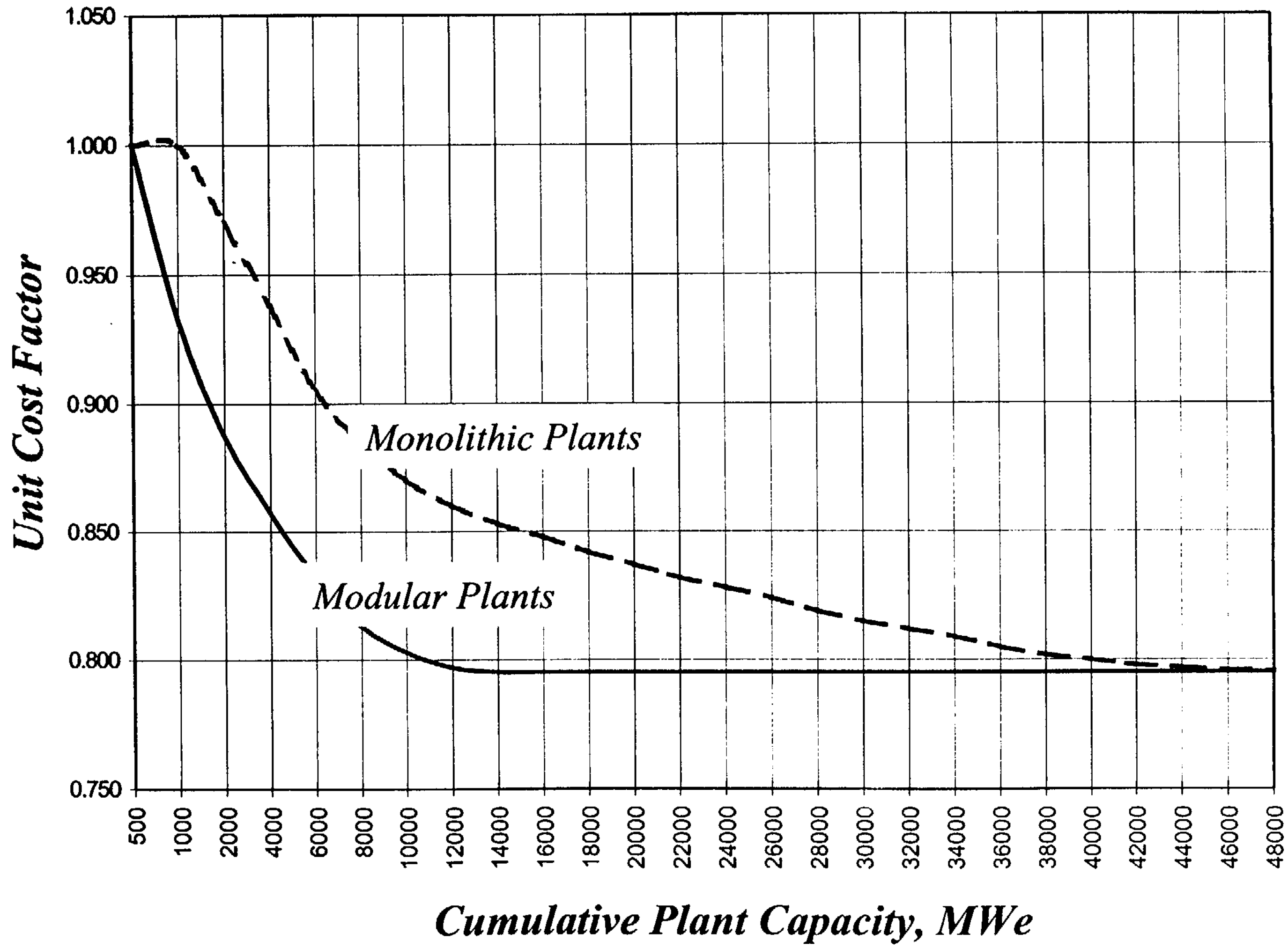
- *simplifies operation,*
- *minimizes the size of the reactor building*
- *improves the plant capacity factor*
- *reduced the need for backup spinning reserve*

NSSS Size, ALMR verses S-PRISM



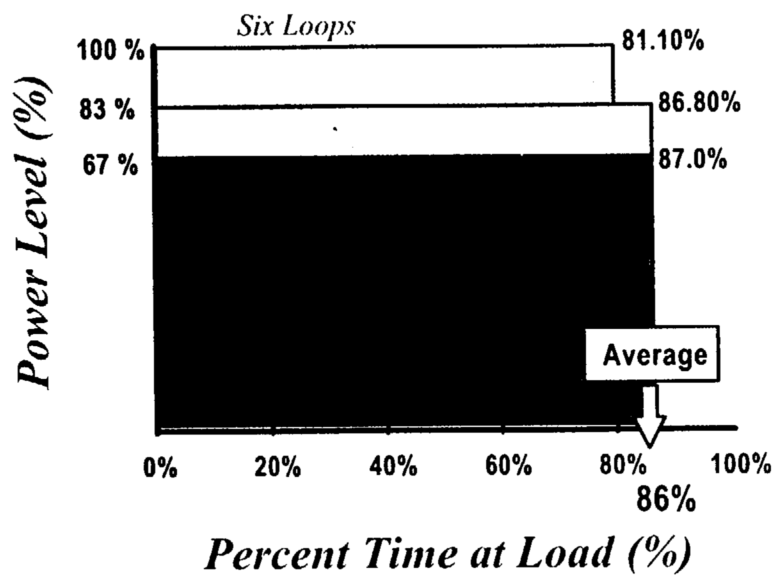
22 % More Power
from
Smaller NI

Learning Effect Favors Modular Plant Designs

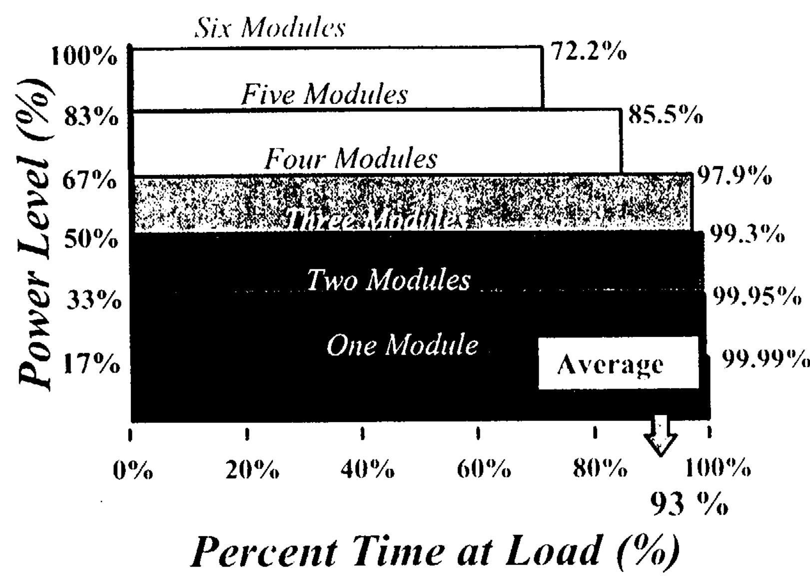


Modular vs. Monolithic Availability and Spinning Reserve

Monolithic Plant
6 Loops



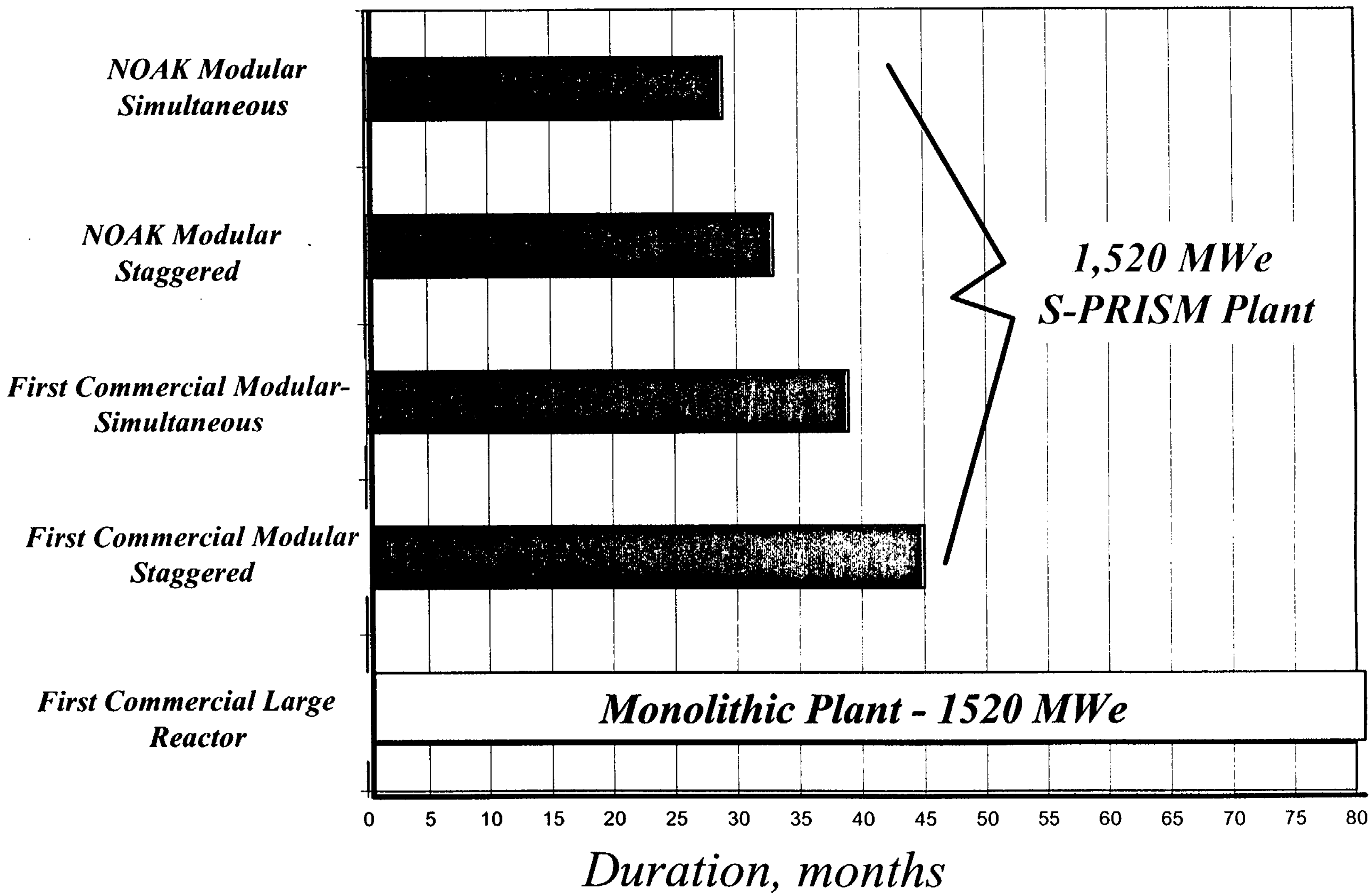
6 Module S-PRISM Plant



<u>Seven point advantage caused by:</u>

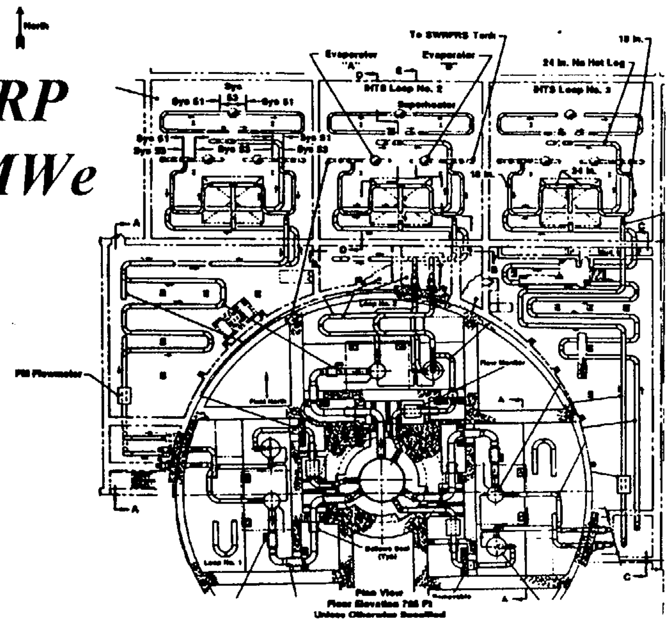
- *Relative simplicity of each NSSS (one SG System rather than 6)*
- *Ability to operate each NSSS independently of the others*

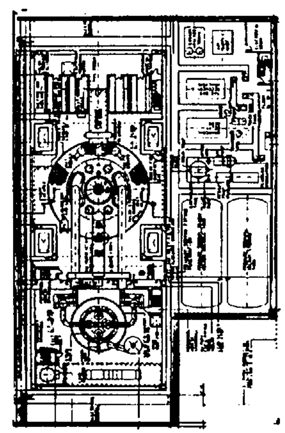
Comparison of Plant Construction Schedules



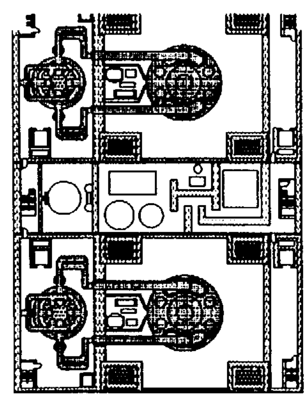
NSSS Size, CRBRP/ALMR /S-PRISM

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]95

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]87

[redacted]SM

- 30 $ M
Competition for
National Program

1981 - [redacted]84

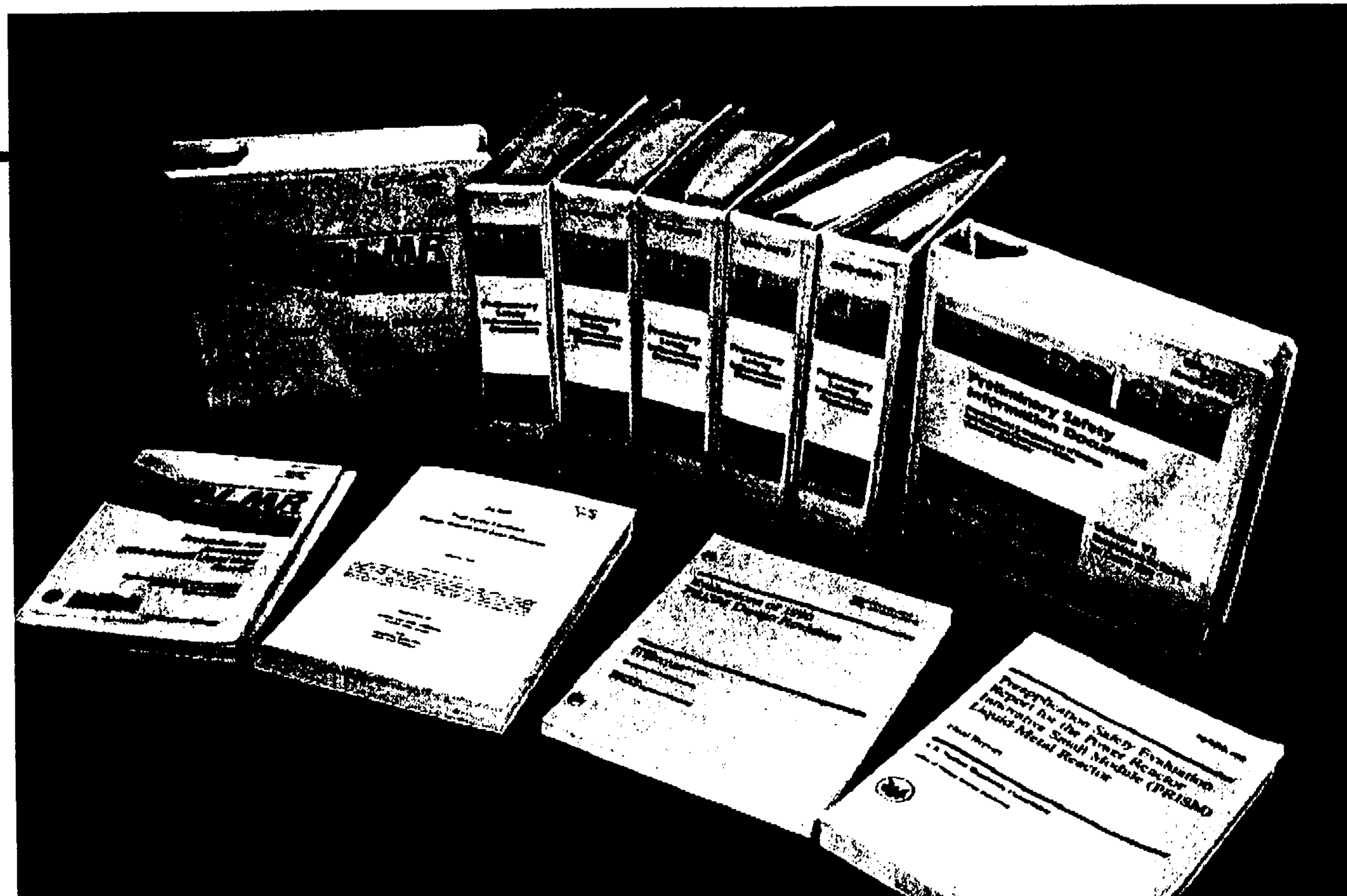
GE PRISM Program

GE Funded

Innovative Design Studies

$ 77 Million
DOE Program

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



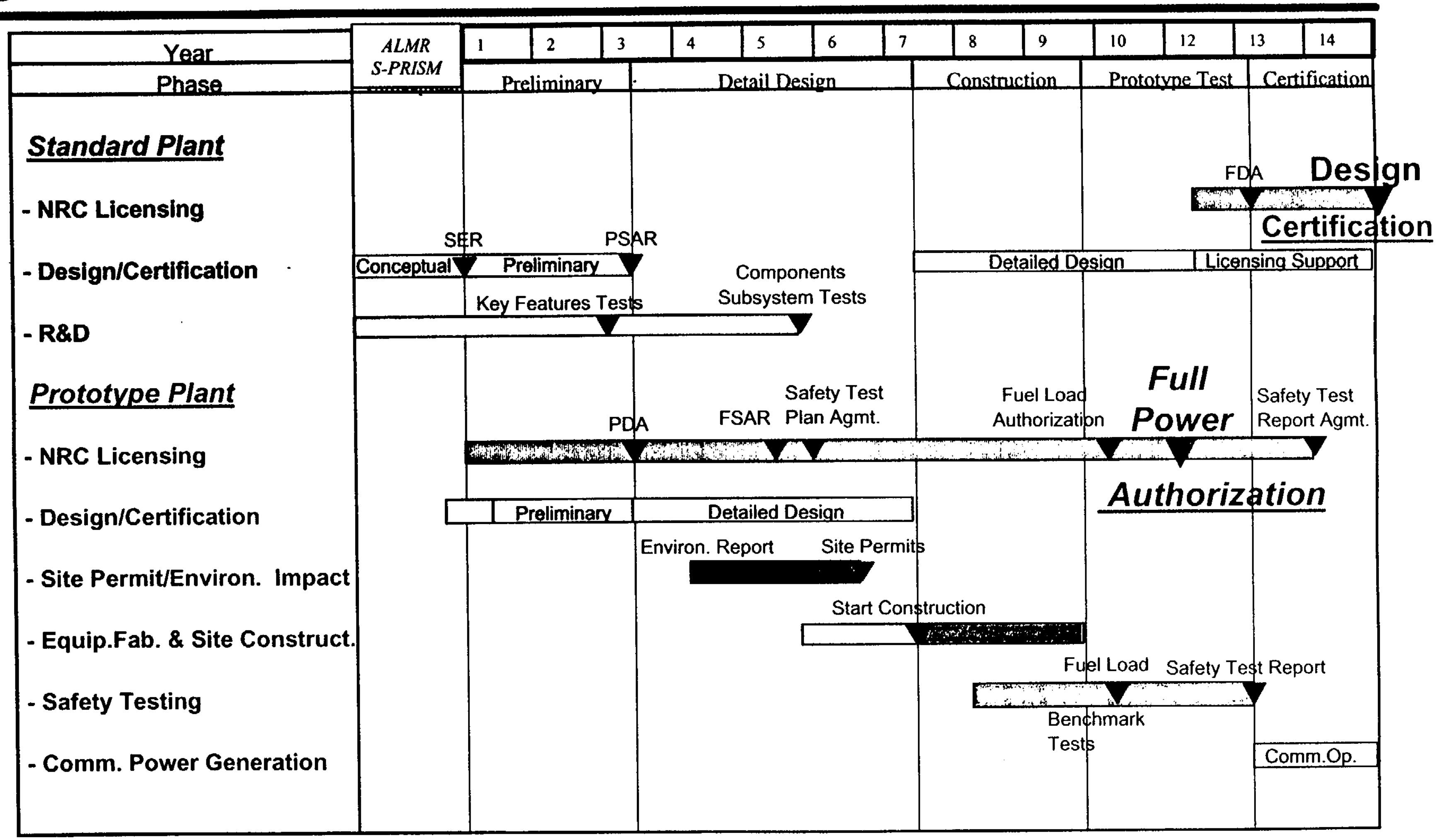
The NRC's Pre-application Safety Evaluation of the ALMR (NUREG-1368) concluded:

"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

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

NAME	LOCATION							
France								
Rapsodie	Cadarache							
Phenix	Marcoule							
SuperPhenix	Creys Malville							
INDIA								
FBTR	Kalpakkam							
ITALY								
PEC	Brasimone							
JAPAN								
Joyo	Oaral							
Monju	Ibarakl							
UK								
DFR	Dounreay							
PFR	Dounreay							
USA								
Clemetine	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

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

<u>Licensing</u> through the <u>testing of a prototypical reactor module</u> should be an efficient approach to obtaining the data needed for design certification.

<u>Defining the T/H and component tests</u> 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

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

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SES Manager

Marsha Gamberoni
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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

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

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

United States
Nuclear Regulatory Commission

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

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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