Publicly Available

February 12, 2001

MEMORANDUM TO: Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

FROM. Thomas L. King, Director/RA/
Division of Risk Analysis and Applications
Office of Nuclear Regulatory Research

SUBJECT: MEETING WITH EXELON GENERATION COMPANY AND OTHER INTERESTED STAKEHOLDERS REGARDING THE PEBBLE BED MODULAR REACTOR

On January 31, 2001, from 1:30 PM to 4:00 PM, the NRC staff met with the representatives of Exelon (Exelon Nuclear and Exelon Generation Company) and other interested stakeholders to discuss the Pebble Bed Modular Reactor (PBMR) design, technology and the potential for Exelon's request for NRC review. The meeting was widely attended with representatives from industry, public interest groups, the national laboratories and ACRS. Attachment 1 contains a list of attendees. Presentation material distributed at this meeting is also included (Attachment 2).

In their opening remarks, W. Travers (NRC) and A. Thadani (NRC) stated the purpose of the meeting as an opportunity to learn more about the PBMR design with the principal focus on various technical and licensing issues. Referring to the NRC's advanced reactor policy, both emphasized the need for identifying key issues and their resolution, and soliciting Commission approval before proceeding with interactions on the PBMR design.

Exelon presenters were W. Sproat, V. Nilekani, K. Borton and J. Muntz. Following introductions, Exelon stated that their main objectives were to initiate a dialogue with the NRC on the PBMR licensing process by providing a summary of the PBMR design and identifying potential design and licensing issues and share initial thoughts on the licensing approach and projected schedule.

Phase 1 of the PBMR project has a multi-national ownership: Exelon Generation (12.5%), British Nuclear Fuel, Ltd. (22.5%), ESKOM (40%) and the Republic of South Africa (RSA) (25%). The design feasibility study for a 100 MWe PBMR is scheduled to be completed by June 2001 and the decision to build a prototype plant in South Africa is to be made by the end of 2001. If built, the prototype module will be a full-scale facility. This facility would allow testing under postulated accident conditions. Exelon's decision to proceed with the PBMR project will be based on several factors such as whether this design can be a source of low-cost electrical energy, with a high safety level, to be able to successfully compete in the U.S. energy markets.

The PBMR is ar high temperature gas-cooled reactor (HTGR) with helium as a coolant, typically, at 900°C and 1000 psi. The reactor pressure vessel is made of steel and is about 19.5-ft diameter and 65-ft high, which is housed in a 175-ft square and 108-ft high reactor building - about two-thirds of which is below grade. The accidents for which the plant is to be designed to accommodate without loss of fuel integrity include a range of internal and external events, including ATWS and station blackout. The fuel element is a multi-layer 60-mm diameter graphite sphere (about the size of a tennis ball), in which low-enriched UO_2 is embedded in the form of coated particles about 0.5 mm in diameter. This sphere is not only a carrier of the fuel, but it is also the neutron moderator. The core contains 330,000 spherical fuel elements and 110,000 graphite spheres The fuel spheres are similar to those used in Germany at AVR Julich - a small research reactor which operated for 22 Years, and at thorium high temperature reactor (THTR) Hamm- Uentrop - a 300 MWe demonstration reactor, which operated for 6 years.

According to Exelon, the safety of the PBMR design is inherent in (1) low ratio of power density to heat capacity leading to slow temperature change in the event of failure affecting heat production and heat removal; (2) high temperature resistence of the ceramic fuel elements and structural material of the core (the failure rate of fuel elements is negligible up to temperatures of 2000°C); (3) strong negative temperature and power coefficients of reactivity under all operating conditions (low excess reactivity is possible in continuously-fueled pebble bed); (4) plant design features mitigate air and water ingress; and (5) helium - a chemically inert, single-phase, neutron-transparent gas - is used as a coolant. In addition to the fuel design and use of passive systems, other features are: its modular design; continuous power generation augmented by on-line fuel loading and unloading (plant outage 30 days every 6 years for turbine and generator maintenance); direct cycle gas turbine; on-site spent-fuel storage for 40 years; and low excess reactivity.

Exelon identified various design issues as well as alternate potential licensing schemes under either 10 CFR Part 50 or Part 52 licensing processes including early site permit. Issues included: qualification of the fuel and its fabrication as an integral aspect of the facility certification; development of the source term; assessment of a leak-tight versus vented containment; materials qualification, such as, high temperature metal creep, and thermal fatigue; verification and validation of design and safety analysis codes; and PRA and classification of SSCs. Other concerns which also need to be addressed include: regulatory treatment of non-safety systems; and determination of appropriate tests, including NDE.

In closing remarks, A. Thadani stated the need (1) to identify key issues for the NRC to develop the needed technical expertise to work efficiently and effectively; (2) to seek Commission approval of a review plan before proceeding; and (3) for the industry to provide more information to help the staff develop the plan.

For further information, please contact Thomas L. King at (301)-415-5790 or tlk@nrc.gov.

cc: SECY
OGC
OCA
OPA
CFO
CIO

Attachments: As stated

Distribution: (w/attachments) DRAA Chron, MWeber (NMSS)

w/o attachments

WTravers	TKing	RTripathi	MMayfield	FEltawila	SRubin	ALevin
CEmeigh	TKress	MEl-zeftawy	SMorris	RCarusso	SArndt	GBagchi
KErwin	JGolder	BJazinski	RBarrett	DCarlson		

All attendees listed in Attachment 1

DOCUMENT NAME: G:\Tom's General\Meeting Notice EXELON-1-31-01.WPD
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Publicly Available? (Y or N) Y DATE OF RELEASE TO PUBLIC ___________ SENSITIVE? ____
*See previous concurrence

OFFICE	DRAA*		DRAA		
NAME	Tripathi		King		
DATE	02/8/1		02/12/1		

Attachment 1

January 31, 2001

Meeting Between NRC and Exelon on PBMR

List of Attendees

Name	Organization	Telephone #
Thomas King	NRC/RES/DRAA	301-415-5790
Stu Rubin	NRC/RES/DSRE	301-415-7480
David Brown	Exelon Corp.	202-347-7500
Neil McDermott	Exelon Corp.	610-765-6909
Joe Green	Stone & Webster	617-589-1400
Ed Wallace	Exelon Corp.	610-765-5721
John Hufnagel	Exelon Corp.	610-765-5507
Ron Smard	NEI	202-739-8128
Alan Levin	NRC/OCM/RAM	301-415-1750
Howard Brusch	Westinghouse	412-256-2800
Vijay M. Nilekani	Exelon	610-765-5950
Bill Travers	NRC/EDO	301-415-1700
Jim Muntz	Exelon Corp.	610-765-5660
Kevin Borton	Exelon Corp	610-765-5528
Ward Sproat	Exelon Corp.	610-765-5930
David W. Woods	Exelon Corp.	610-765-6900
Ed Cullen	Exelon Corp.	610-765-5700
Tom Clements	Nuclear Control Institute	202-822-8444
Kenneth Chuck Wade	DOE	301-903-1031
Chuck Emeigh	NRC/NMSS	301-415-7836
Tatsuya Taminami	Tokyo Electric	202-657-0790
Jenny Weil	McGraw-Hill	202-383-2161
Geary Mizuno	NRC/OGC/Rulemaking & Fuel Cycle	301-415-1639
Jim Kendall	IAEA/Nuclear Energy Dept.	431260022820
John Matthews	Morgan Lewis	202-467-7524
John Ireland	Los Alamos National Labs.	505-667-8777
Syd Ball	Oak Ridge National Lab.	865-574-0415
Robert E. Uhrig	University of Tenn and NRC/ACRS	352-367-0374
Thomas S. Kress	ACRS	865-483-7548
Herb Fontecilla	Dominion	703-838-2314
Med El-Zeftawy	NRC/ACRS	301-415-6889
Jan Freeman	Exelon Corp.	610-765-6906
Mike Callahan	GSI	202-544-4522
Jim Muckerheide	Mass	508-820-2039
Rod Adams	Adams Atomic Engines, Inc.	727-641-1081

Name	Organization	Telephone #
Cal Reid	Bechtel	301-228-6533
Dave Lockbaum	Union of Concerned Scientists	202-223-6133
Dean Raleigh	US, Scientech	301-258-2551
Scott Morris	NRC/EDO	301-415-1730
Rogon Huston	Licensing Support Services	703-671-9738
Madeline A. Feltus	US Department of Energy	301-903-2308
Mark Haynes	General Atomics	202-496-8209
Edwin Lynien	Nuclear Control Institute	202-822-6594
Patricia Bryant	Nuclear Energy institute	202-739-8020
Jackie Carney	Foundation for Nuclear Studies	202-544-1488
Ralph Caruso	NRR/DSSA	301-415-1813
Steven Arndt	NRC/RES	301-415-6502
Russ Bell	NEI	202-739-8089
Goutam Bagchi	NRC/NRR	301-415-3298
Ken Erwin	NRC/NMSSS/SFPO	301-415-2443
Jennifer Golder	NRC/OCFO/DPBA/PAB	301-415-0115
Bob Jasinski	NRC/OPA	301-415-8200
Theodore Rockwell	Radiation Science & Health, Inc.	301-652-9509
Richard Barrett	NRC/NRR	301-415-3183
Craig D. Sellers	Innovative Technology Solns .	410-394-1504
John M. Osborne	Constellation Nuclear	410-495-2252
Len Ward	ISL, Inc	301-255-2279
Robert A. Bari	BNL	631-344-2629
Mark R. Holbrook	INEEL	208-526-4362
Steve Frantz	Morgan, Lewis & Bochnius, LLP	202-467-7460
Peter Murray	Westinghouse	301-881-7040
Terry Rudek	Westinghouse	860-285-2170
Charles Brinkman	Westinghouse	301-881-7040
Stan Ritterbusch	Westinghouse	860-285-5206
Paul Gunter	NIRS	202-328-0002
Tom Miller	USDOE	301-903-4517
Ashok C. Thadani	NRC/RES	301-415-6641
Jerry Wilson	NRC/NRR/DRIP	301-415-3145
Jeff Benjamin	UP Licensing - Exelon	630-663-7969
Farouk Eltawila	NRC/RES	301-415-7499
Margaret Federline	NRC/RES	301-415-8003
Carl Paperiello	NRC/EDO	301-415-1705
Howard Faulkner	NRC/OIP	301-415-2762
Raji R. Tripath	NRC/RES	301-415-7472
Carey W. Fleming	Winslow & Strawn	202-371-5839
Gail H. Marcus	DOE/NE	202-586-2240
Joe Muscara	NRC/RES/DET/MEB	301-415-5844

see RE1709409

UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Notation Vote)

May 17, 1994

SECY-94-133

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: UPDATED COMMISSION POLICY STATEMENT ON ADVANCED REACTORS TO REFERENCE THE COMMISSION'S METRICATION POLICY

PURPOSE:

In accordance with the Commission's request in COMSECY-94-002, the NRC staff is requesting Commission approval to issue the revised policy statement on advanced reactors to reference the Commission's policy on metrication.

BACKGROUND:

On January 8, 1992, the Commission (with all Commissioners agreeing) approved the proposed policy statement in SECY-91-390, "Metrication Policy," for publication for public comment. The proposed policy statement was published in the Federal Register on February 10, 1992 (57 FR 4891). In October 1993, the NRC issued its final policy statement on metrication (57 FR 46202) after receiving comments from 12 responders from both the public and private sectors. All commenters supported the policy.

On April 2, 1993, Asea Brown Boveri-Combustion Engineering (ABB-CE) requested an exemption from the Commission's metrication policy. In its letter, ABB-CE stated that the Combustion Engineering Standard Safety Analysis Report for the System 80+ had been submitted for review several years before

Contact:
Stephen Sands
504-3154

NOTE: TO BE MADE PUBLICLY AVAILABLE WHEN THE FINAL SRM IS MADE AVAILABLE the NRC had issued its current metrication policy statement and to convert to the metric system would not only add significant schedular delays, but would generate unnecessary costs of millions of dollars. Therefore, in the memorandum of May 5, 1993, from Samuel Chilk, Office of the Secretary of the Commission (SECY), to James Taylor, Office of the Executive Director for Operations (EDO), the Commission (with all Commissioners agreeing) granted ABB-CE an exemption from NRC's metrication policy for the ABB-CE System 80+ design certification documents (DCD). The Commission also requested that the staff identify other vendors that might wish to be granted similar exemptions.

On July 7, 1993, in response to the Commission's request, the staff sent a letter to GE Nuclear Energy (GE) asking if GE wanted relief from the responsibility of performing unit conversions or of providing dual units in any documents that are part of the design certification process. In its August 27, 1993, response, GE requested that the NRC exempt the advanced boiling water reactor (ABWR) design certification application from compliance with the Commission's metrication policy. The staff then forwarded its recommendation along with GE's request for exemption on September 30, 1993, to the Commission. Specifically, the staff recommended that GE be required to prepare the ABWR DCDs in the International System of Units SI (SI units). It also recommended that GE provide a conversion table from SI to English units in the DCDs instead of giving dual units. However, the staff indicated it would prepare the ABWR final safety evaluation report (FSER) in dual units in accordance with Commission guidance.

In COMSECY-93-051, "Metrication Policy Exemption Request," the Commission (with all Commissioners agreeing) agreed with the staff recommendation that GE be required to prepare the ABWR DCDs in SI units and provide a conversion table from SI to English units instead of providing dual units.

On October 12, 1993, Westinghouse replied to the staff's letter of July 6, 1993, requesting an exemption from the metrication policy for the AP600 design similar to that requested for the System 80+ design. In COMSECY-94-002, the Commission (with all Commissioners agreeing) approved the staff recommendation in its January 4, 1994, memorandum from James Taylor, EDO, to the Commission to grant an exemption from the NRC's metrication policy to Westinghouse for the AP600 design.

In its response of November 5, 1993, to the staff's July 7, 1993, letter, GE informed the staff that the DCDs for the simplified boiling water reactor (SBWR) would be consistent with the requirements of the metrication policy. In its oral response of March 1994, Atomic Energy of Canada Ltd. (AECL) informed the staff that the CANDU DCDs would also be consistent with the requirements of the metrication policy.

DISCUSSION:

On July 8, 1986, the NRC issued a document entitled "Regulation of Advanced Nuclear Power Plants; Statement of Policy" (51 FR 24643). Additionally, in June 1988, it published NUREG-1226, "Development and Utilization of the Policy Statement on the Regulation of Advanced Nuclear Power Plants." Both the advanced reactor policy statement and NUREG-1226 were written to provide guidance for the development of new regulatory requirements to support the advanced designs. However, both documents preceded the Commission's metrication policy, which was published in the Federal Register on October 7, 1992 (57 FR 46202). Therefore, the documents do not reflect the current Commission position on metrication.

Since the mandate by Congress, together with economic pressure on U.S. companies to compete in global markets, has increased the motivation for metric conversion in the United States, it is important to provide adequate notice to any prospective applicant that the Commission takes its policy on metrication seriously. Some industry codes and standards developed in the United States use the metric system in the form of dual-unit reporting or conversion tables. Additionally, new codes and standards are increasingly being written in metric units to maintain their international presence and acceptance.

Therefore, in COMSECY-94-002, the Commission granted an exemption from NRC's metrication policy to Westinghouse for the AP600 design on the basis that conversion to the metric system would impose significant costs and serious schedular delays. The Commission also requested that the staff update its policy statement on advanced reactors to reference the metrication policy.

The Commission has also granted ABB-CE an exemption from the metrication policy for the System 80+ design on the basis that conversion to the metric system of measurement would generate unnecessary costs. While economic impracticability is not directly addressed in NRC's final statement of policy, published on October 7, 1992 (57 FR 46202), it is mentioned in the background section of the Federal Register notice. Therefore, the Commission has determined, that under certain circumstances, the costs associated with conversion to the metric system outweigh the benefits, such as improvements in administrative efficiency.

CONCLUSIONS:

The NRC believes that conversion to the metric system is important to the national interest. The Commission strongly encourages its licensees and license applicants to employ the metric system of measurement wherever and whenever its use is not potentially detrimental to the public health and safety or is not economically impracticable. Therefore, the NRC is issuing this revised policy statement on advanced reactors to incorporate the Commission's policy on metrication.

RECOMMENDATION:

The staff requests Commission approval to issue this revised policy statement on advanced reactors (enclosed).

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. Additionally, since the Advisory Committee on Reactor Safeguards had previously reviewed both policy statements, a copy of this paper is being sent for their information.

Hugh L. Thompson
James M. Taylor
Executive Director
for Operations

Enclosure:
Revised Policy Statement
on Advanced Reactors

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Friday, June 3, 1994.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Thursday, May 26, 1994, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DISTRIBUTION:
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NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

REGULATION OF ADVANCED NUCLEAR POWER PLANTS;
STATEMENT OF POLICY

AGENCY: Nuclear Regulatory Commission

ACTION: Final Policy Statement

SUMMARY: The Nuclear Regulatory Commission (NRC) intends to improve the licensing environment for advanced nuclear power reactors to minimize complexity and uncertainty in the regulatory process. This statement gives the Commission's policy regarding the review of, and desired characteristics associated with, advanced reactors. This policy statement is a revision of the final policy statement titled "Regulation of Advanced Nuclear Power Plants, Statement of Policy" (51 Federal Register 24643) that was published on July 8, 1986. The purpose of this revision is to update the Commission's policy statement on advanced reactors to reference the Commission's metrication policy.

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: Stephen P. Sands U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: 301-504-3154.

SUPPLEMENTARY INFORMATION:

BACKGROUND

On July 8, 1986, the Commission published its final policy statement on advanced reactors in the Federal Register (51 Federal Register 24643). The Commission's primary objectives in issuing the advanced reactor policy statement were threefold:

- First, to maintain the earliest possible interaction of applicants, vendors, and government agencies, with the NRC;
- Second, to provide all interested parties, including the public, with the Commission's views concerning the desired characteristics of advanced reactor designs; and
- Third, to express the Commission's intent to issue timely comment on the implications of such designs for safety and the regulatory process.

On August 10, 1988, Congress passed the Omnibus Trade and Competitiveness Act [the Act], (19 U.S.C. 2901 *et seq.*), which amended the Metric Conversion Act of 1975, (15 U.S.C. 205a *et seq.*). Section 5164 of the Act (15 U.S.C 205a) designates the metric system as the preferred system of weights and measures for U.S. trade and commerce.

In an effort to effect an orderly change to the metric system, the Act requires that all Federal agencies convert to the metric system of measurement in their procurement, grants, and other business-related activities by the end of fiscal year 1992, "except to the extent that such use is impractical or is likely to cause significant inefficiencies or loss of markets to U.S. firms, such as when foreign competitors are producing competing products in non-metric units," Section 5614(b)(2).

In response to the Act, the NRC published its metrication policy statement for comment in the Federal Register on February 10, 1992 (57 Federal Register 4891). The purpose of the metrication policy statement was to inform NRC licensees and the public how the Commission intended to meet its obligations under the Act. Comments on the draft statement were submitted by 12 responders, including 5 power reactor licensees, 3 standards organizations, a reactor vendor, a materials licensee, the Nuclear Management and Resources Council, and a joint letter submitted by three individuals. All commenters supported the Commission's position and the final policy statement was published on October 7, 1992 (57 Federal Register 46202).

The Commission supports and encourages the use of the metric system of measurement by NRC licensees and applicants. However, Commission experience to date in design certification reviews is that it is impracticable and uneconomical to convert a design to the metric system late in the design process and that applicants should consider metrication early in the design process. Therefore, the Commission is revising the advanced reactor policy statement to incorporate its policy on metrication to encourage licensees and license applicants to employ the metric system of measurement wherever and whenever its use is not potentially detrimental to the public health and safety or is not economically impracticable.

COMMISSION POLICY

Consistent with its legislative mandate, the Commission's policy with respect to regulating nuclear power reactors is to ensure adequate protection of the public health and safety and the environment. Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current-generation light water reactors. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The Commission also expects that advanced reactor designs will comply with the Commission's safety goal policy statement and the policy statement on conversion to the metric system.

Among the attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, and that therefore should be considered in advanced designs, are:

- Highly reliable and less complex shutdown and decay heat removal systems. The use of inherent or passive means to accomplish this objective is encouraged (negative temperature coefficient, natural circulation, etc.).

- Longer time constants and sufficient instrumentation to allow for more diagnosis and management before reaching safety systems challenge and/or exposure of vital equipment to adverse conditions.

- Simplified safety systems that, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe shutdown conditions. Such simplified systems should facilitate operator comprehension, reliable system function, and more straightforward engineering analysis.

- Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems.

- Designs that provide reliable equipment in the balance of plant (BOP) (or safety-system independence from BOP) to reduce the number of challenges to safety systems.

- Designs that provide easily maintainable equipment and components.

- Designs that reduce potential radiation exposures to plant personnel.

- Designs that incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for and consequences of severe accidents.

- Design features that can be proven by citation of existing technology or that can be satisfactorily established by commitment to a suitable technology development program.

If specific advanced reactor designs with some or all of the above foregoing attributes are brought to the NRC for comment and/or evaluation, the Commission can develop preliminary design safety evaluation and licensing criteria for their safety-related aspects. Combination of some or all of the above attributes may help obtain early licensing approval with minimum regulatory burden. Designs with some or all of these attributes are also likely to be more readily understood by the general public. Indeed, the number and nature of the regulatory requirements may depend on the extent to which an individual advanced reactor design incorporates general attributes such as those listed above. However, until such time as conceptual designs are submitted, the Commission believes that regulatory guidance must be sufficiently general to avoid placing unnecessary constraints on the development of new design concepts.

To provide for more timely and effective regulation of advanced reactors, the Commission encourages the earliest possible interaction of applicants, vendors, other government agencies, and the NRC to provide for early identification of regulatory requirements for advanced reactors, and to provide all interested parties, including the public, with a timely, independent assessment of the safety characteristics of advanced reactor designs. Such licensing interaction and guidance early in the design process will contribute toward minimizing complexity and adding stability and predictability in the licensing and regulation of advanced reactors.

While the NRC itself does not develop new designs, the Commission intends to develop the capability for timely assessment and response to innovative and advanced designs that might be presented for NRC review. Prior experience has shown that new reactor designs -- even variations of established designs -- may involve technical problems that must be solved in order to ensure adequate protection of the public health and safety. The earlier such design problems are identified, the earlier satisfactory resolution can be achieved. Prospective applicants are reminded that, while the NRC will undertake to review and comment on new design concepts, the applicants are responsible for documentation and research necessary to support a specific license application. (NRC research is conducted to provide the technical bases for rulemaking and regulatory decisions, to support licensing and inspection activities, and to increase NRC's understanding of phenomena for which analytical methods are needed in regulatory activities.)

During the initial phase of advanced reactor development, the Commission particularly encourages design innovations that enhance safety and reliability (such as those described above) and that generally depend on technology that is either proven or can be demonstrated by a straightforward technology development program. In the absence of a significant history of operating experience on an advanced concept reactor, plans for innovative use of proven technology and/or new technology development programs should be presented to the NRC for review as early as possible, so that the NRC can assess how the proposed program might influence regulatory requirements. To achieve these broad objectives, the Advanced Reactor Projects Directorate (PDAR) was established in the Office of Nuclear Reactor Regulation. This group is the focal point for NRC interaction with the Department of Energy, reactor designers, and potential applicants, and coordinates the development of regulatory criteria and guidance for proposed advanced reactors. In addition, the group maintains knowledge of advanced reactor designs, developments, and operating experience in other countries, and provides guidance on an NRC-funded advanced reactor safety research program to ensure that it supports, and is consistent with, the Commission's advanced reactor policy. The PDAR also provides guidance regarding the timing and format of submittals for review. The Advisory Committee on Reactor Safeguards plays a significant role in reviewing proposed advanced design concepts and supporting activities.

The NRC believes that conversion to the metric system is important to the national interest. The Commission strongly encourages its licensees and license applicants to employ the metric system of measurement wherever and whenever its use is not potentially detrimental to the public health and safety or is not economically infeasible. In order to facilitate use of the metric system by licensees and applicants, the NRC began publishing, as of January 7, 1993, the following documents in dual units: new regulations, major amendments to existing regulations, regulatory guides, NUREG-series documents, policy statements, information notices, generic letters, bulletins, and all written communications directed to the public. Licensees and applicants should follow the guidance outlined in the Commission's position and final policy statement on metrication published on October 7, 1992 (57 Federal Register 46202).

GAS TURBINE - MODULAR HELIUM REACTOR (GT-MHR)

COMMERCIALIZATION PROGRAM BRIEFING

March 2001

GENERAL ATOMICS

GAS TURBINE - MODULAR HELIUM REACTOR (GT-MHR)

COMMERCIALIZATION PROGRAM BRIEFING

- **PLANT DESCRIPTION**
- **PROGRAM DESCRIPTION**

GENERAL ATOMICS

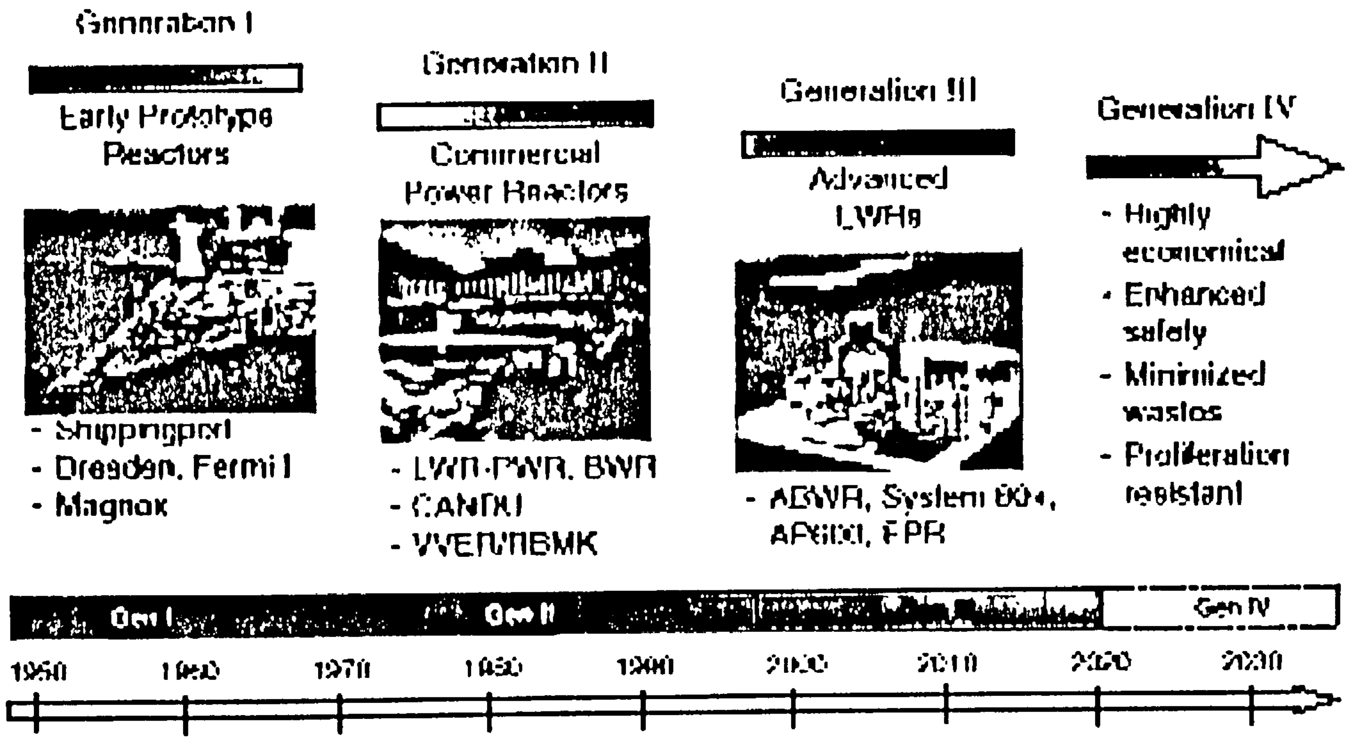
GT-MHR COMMERCIALIZATION PROGRAM

PLANT DESCRIPTION

GENERAL ATOMICS

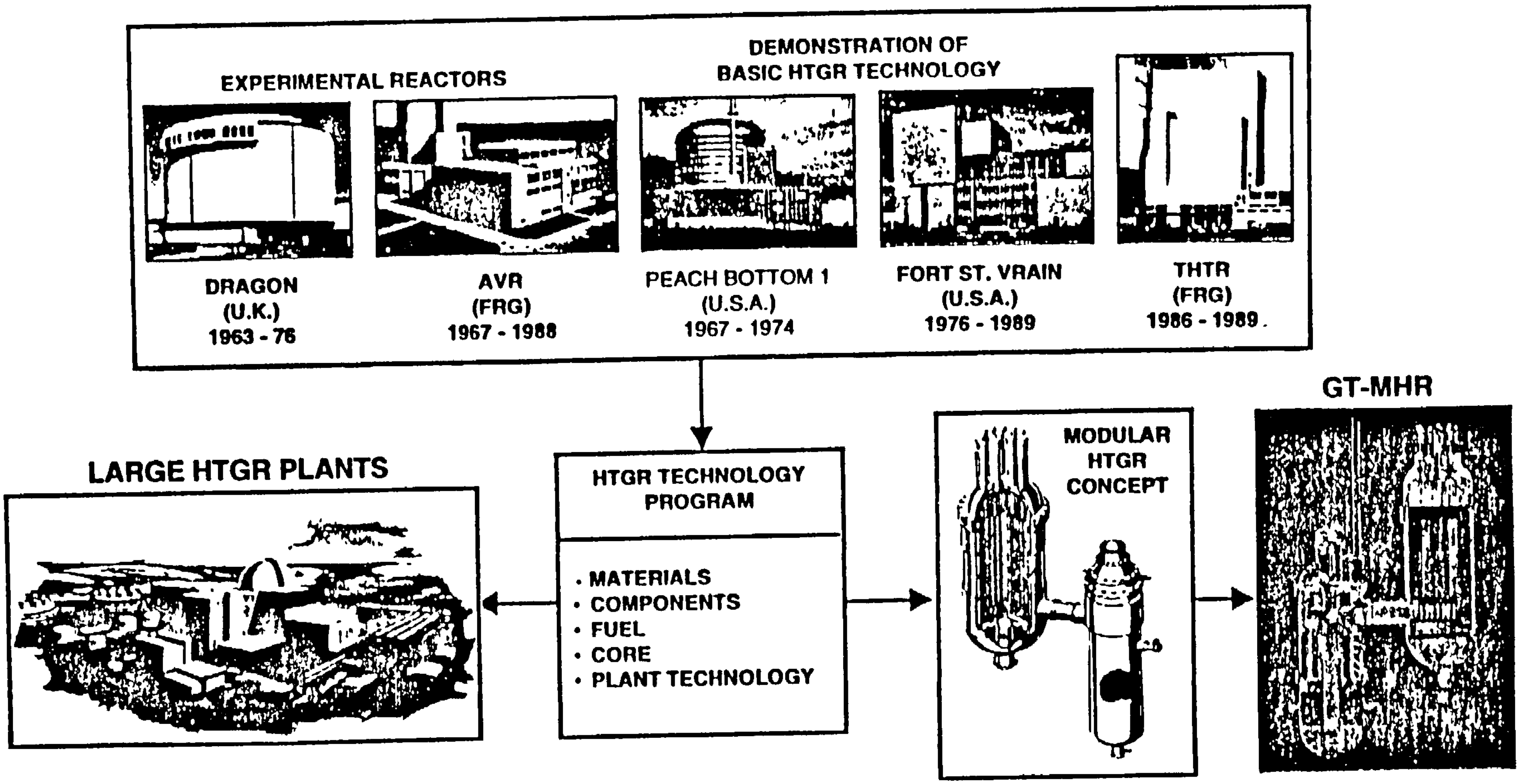
Nuclear Power Generation IV Initiative

The Evolution of Nuclear Power



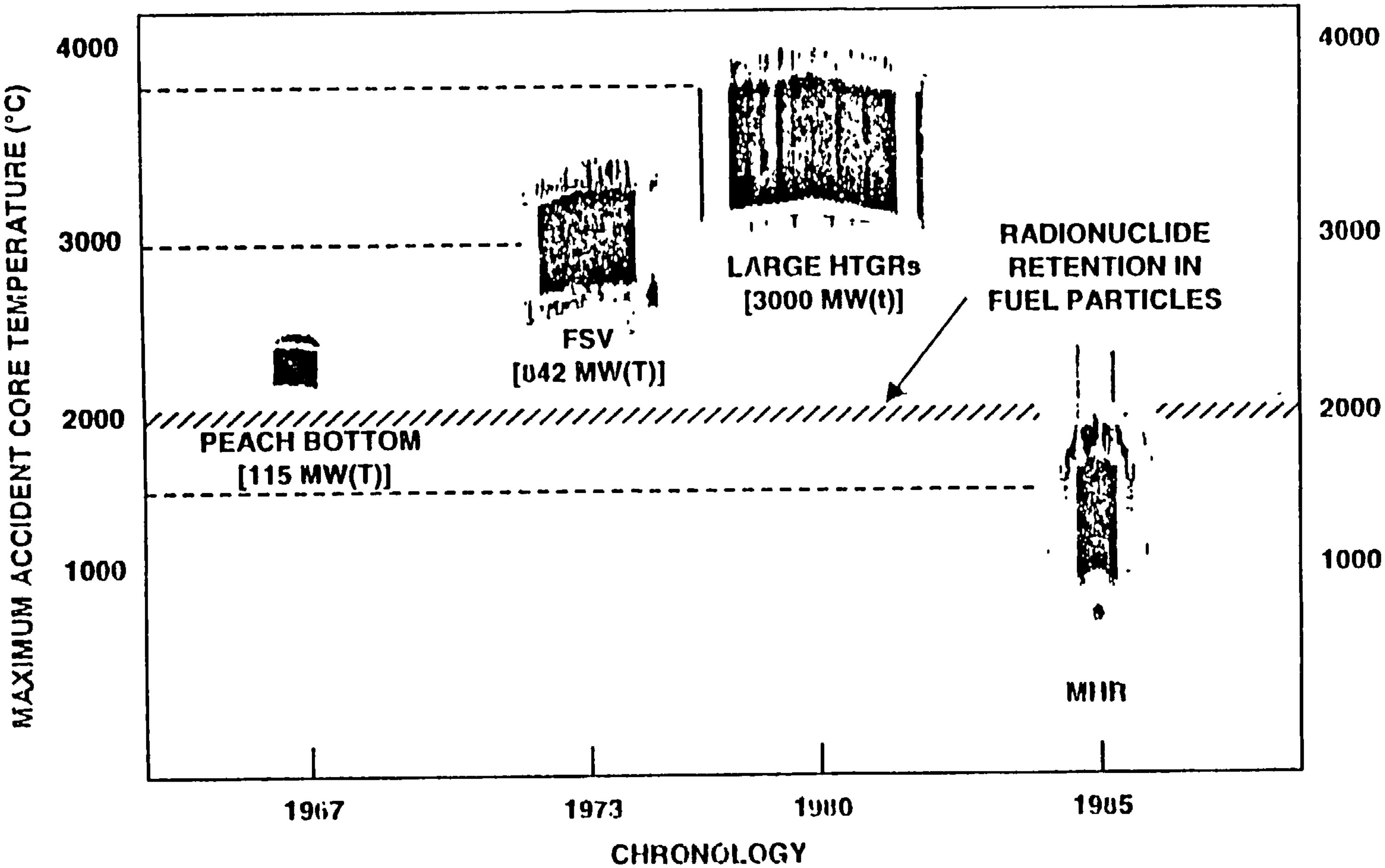
U.S. AND EUROPEAN TECHNOLOGY BASES FOR MODULAR HIGH TEMPERATURE REACTORS

BROAD FOUNDATION OF HELIUM REACTOR TECHNOLOGY



GENERAL ATOMICS

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

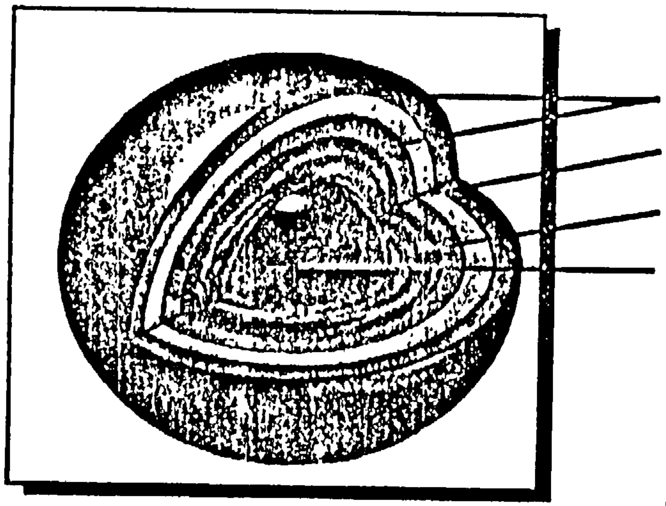


SIZED AND CONFIGURED TO TOLERATE EVEN A SEVERE ACCIDENT

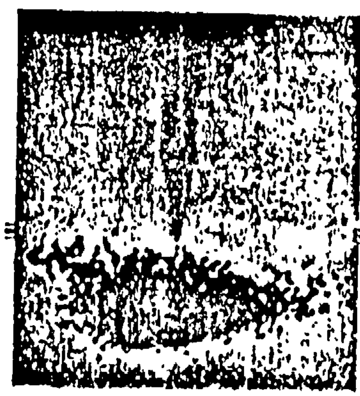
GENERAL ATOMICS

I-222(1)
1-12-96

CERAMIC FUEL RETAINS ITS INTEGRITY UNDER SEVERE ACCIDENT CONDITIONS



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



PARTICLES

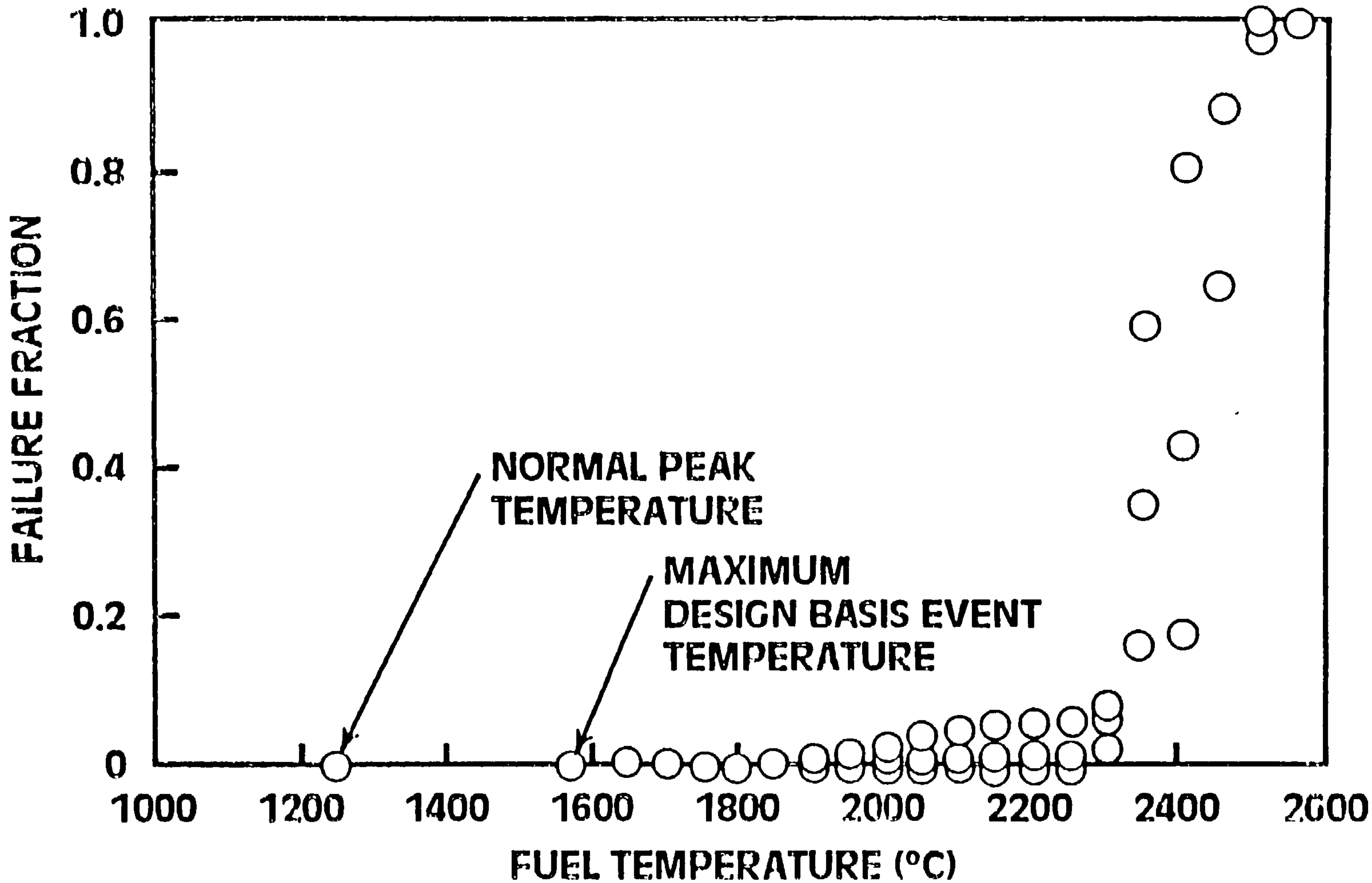
COMPACTS

FUEL ELEMENTS

GENERAL ATOMICS

L-029(5)
4-14-94

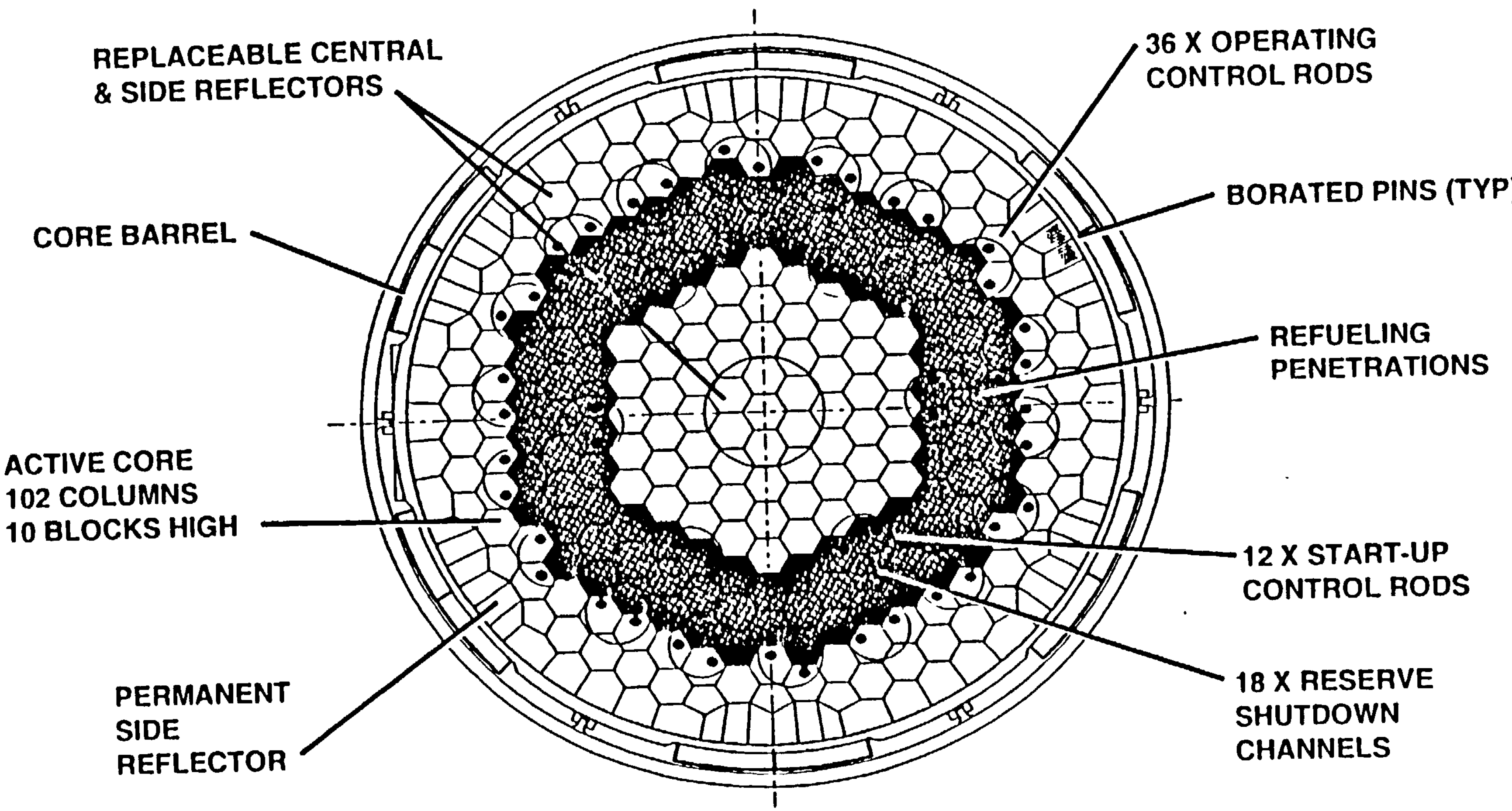
COATED PARTICLES STABLE TO BEYOND MAXIMUM ACCIDENT TEMPERATURES



GENERAL ATOMICS

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

ANNULAR REACTOR CORE LIMITS FUEL TEMPERATURE DURING ACCIDENTS

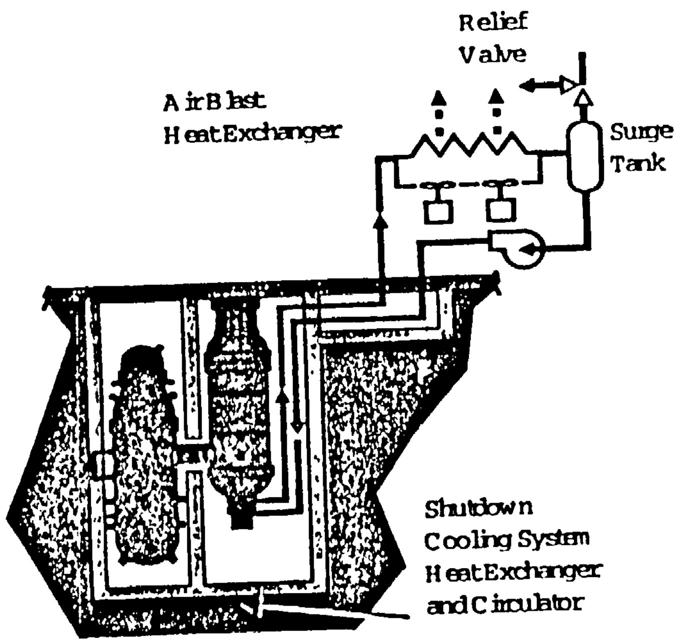


...ANNULAR CORE USES EXISTING TECHNOLOGY

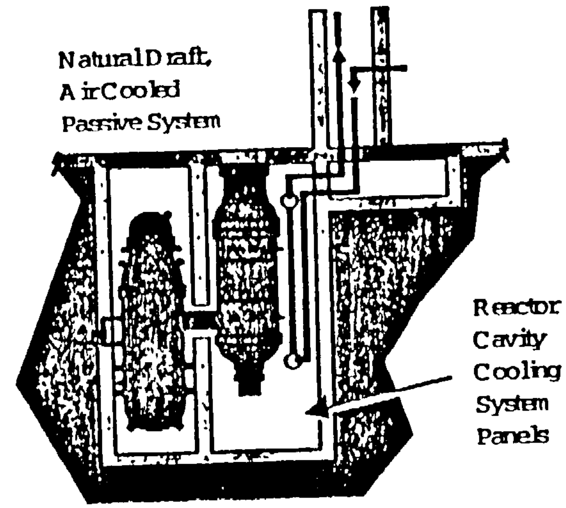
GENERAL ATOMICS

L-199(10)
6-9-95

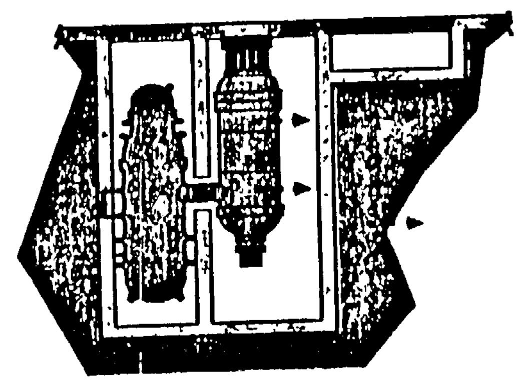
POSSIBLE DECAY HEAT REMOVAL PATHS WHEN NORMAL POWER CONVERSION SYSTEM IS UNAVAILABLE



A) Active Shutdown Cooling System



B) Passive Reactor Cavity Cooling System



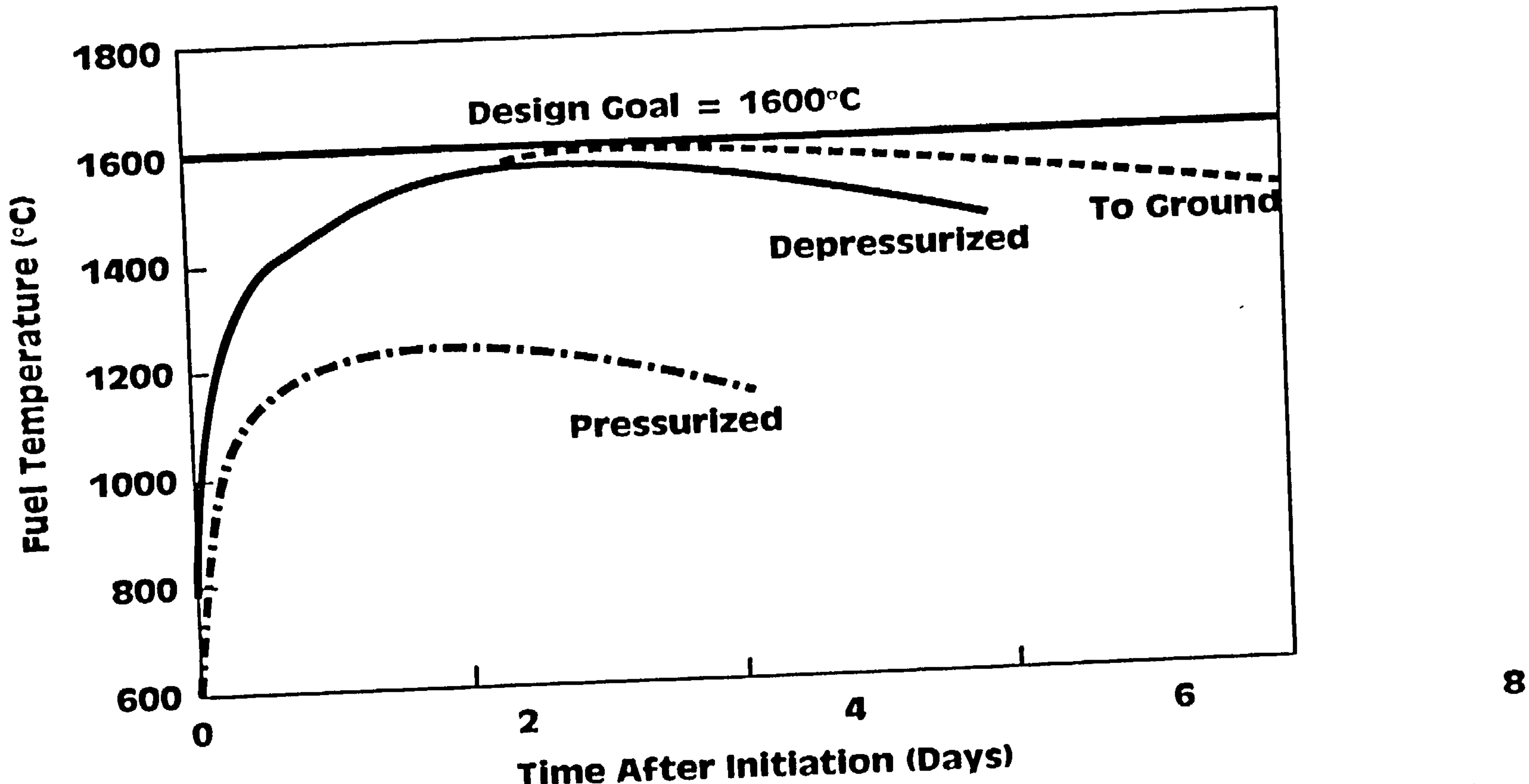
C) Passive Radiation and Conduction of Afterheat to Silo Containment (Beyond Design Basis Event)

. . . DEFENSE-IN-DEPTH BUTTRESSED BY INHERENT CHARACTERISTICS

GENERAL ATOMICS

L-266(2)
7-28-94

FUEL TEMPERATURES REMAIN BELOW DESIGN LIMITS DURING LOSS OF COOLING EVENTS



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

GENERAL ATOMICS

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*

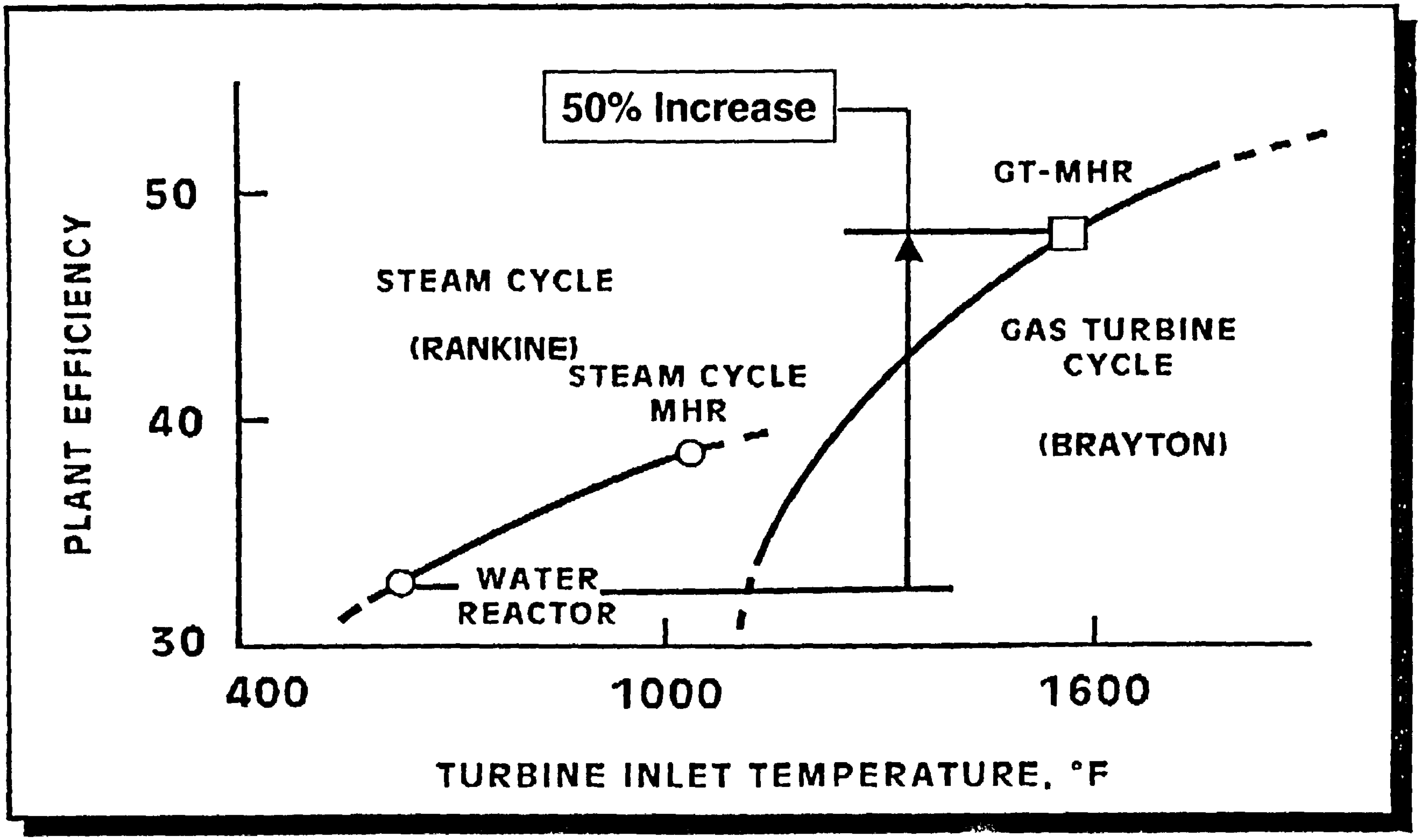
- **Worst case fuel temperature limited by design features**
  - *Low power density*
  - *Low thermal rating per module*
  - *Annular Core*
  - *Passive heat removal*

....CORE CAN'T MELT

- **Core Shuts Down Without Rod Motion**

GENERAL ATOMICS

HIGH TEMPERATURE GAS REACTORS HAVE UNIQUE ABILITY TO USE BRAYTON CYCLE



GENERAL ATOMICS

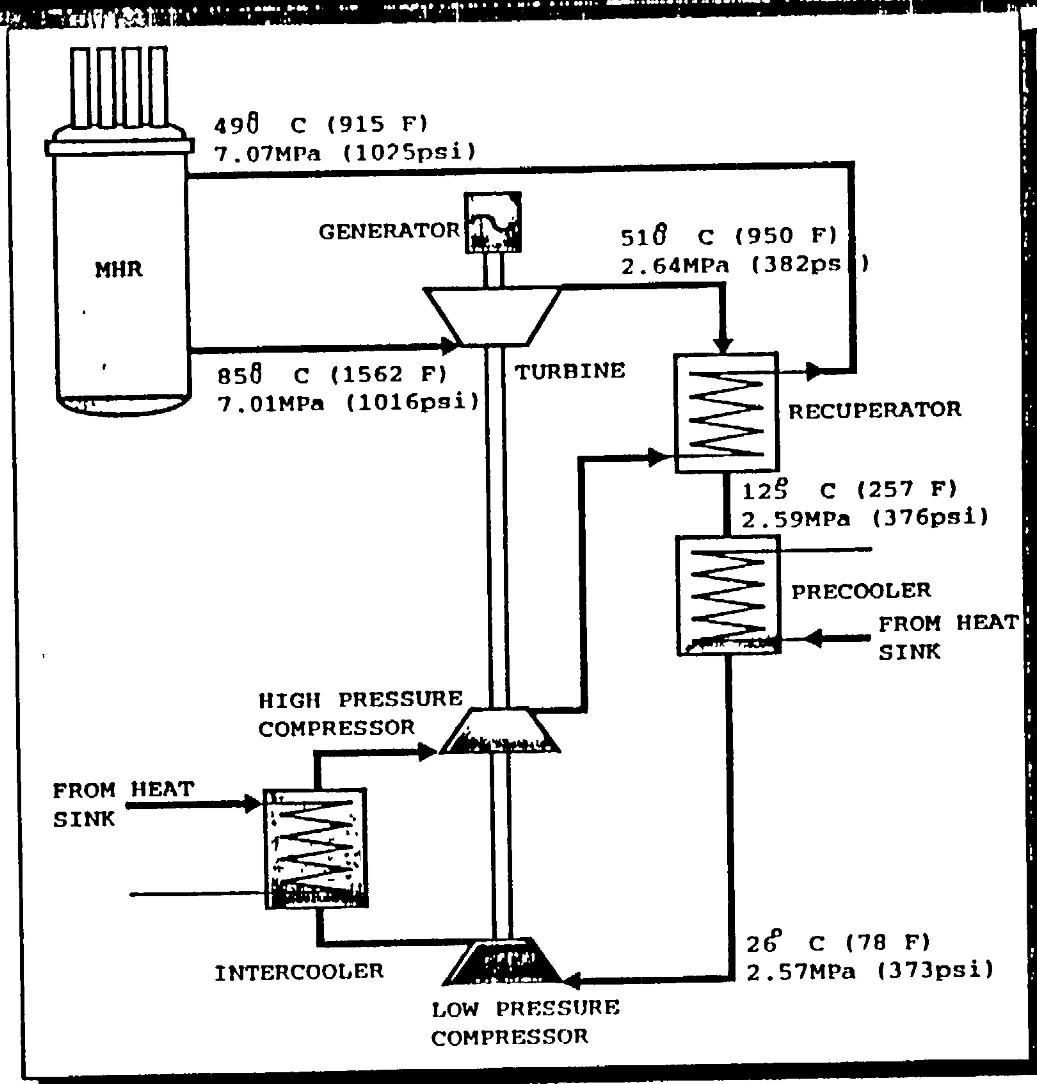
L-029(4)
6-9-94

TECHNOLOGY ADVANCEMENTS HAVE ENABLED THE GT-MHR

- Small Passively Safe Modular Helium Reactor
  - *turbine size requirements reduced*
  - *insensitive to turbine failure accidents*
- Large Gas Turbine Engines
  - *significant increase in industrial applications*
  - *size now match modular reactor size*
- Magnetic Bearings
  - *eliminates oil ingress concerns*
  - *improves performance and reliability*
  - *rapidly increasing industrial experience; larger sizes*
- Compact Heat Exchangers
  - *dramatically improves efficiency*
  - *size improves design integration*
  - *extensive fossil operating experience*

GENERAL ATOMICS

GT-MHR FLOW SCHEMATIC



GENERAL ATOMICS

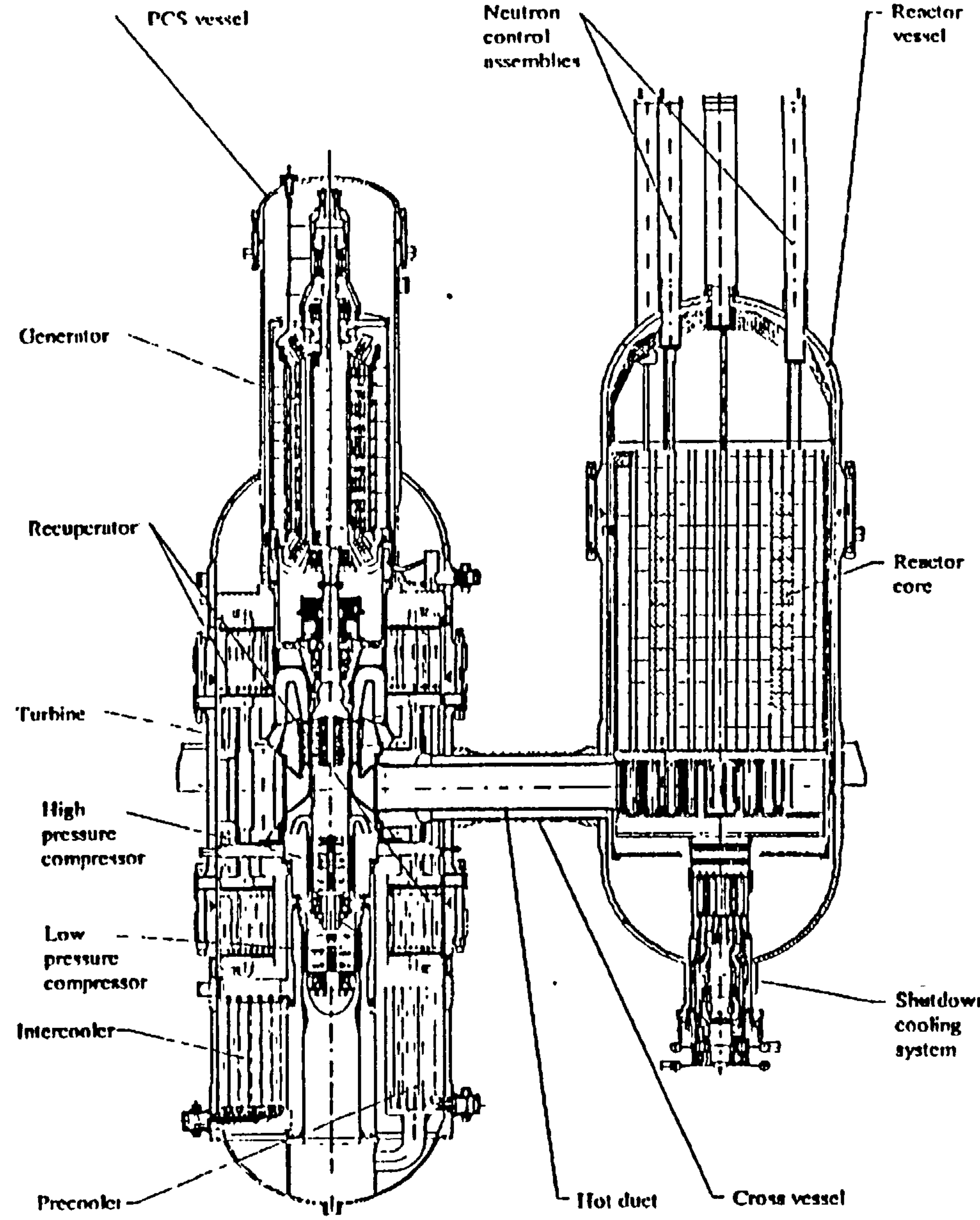
L-271(12a)
8-14-94
A-36

[redacted]

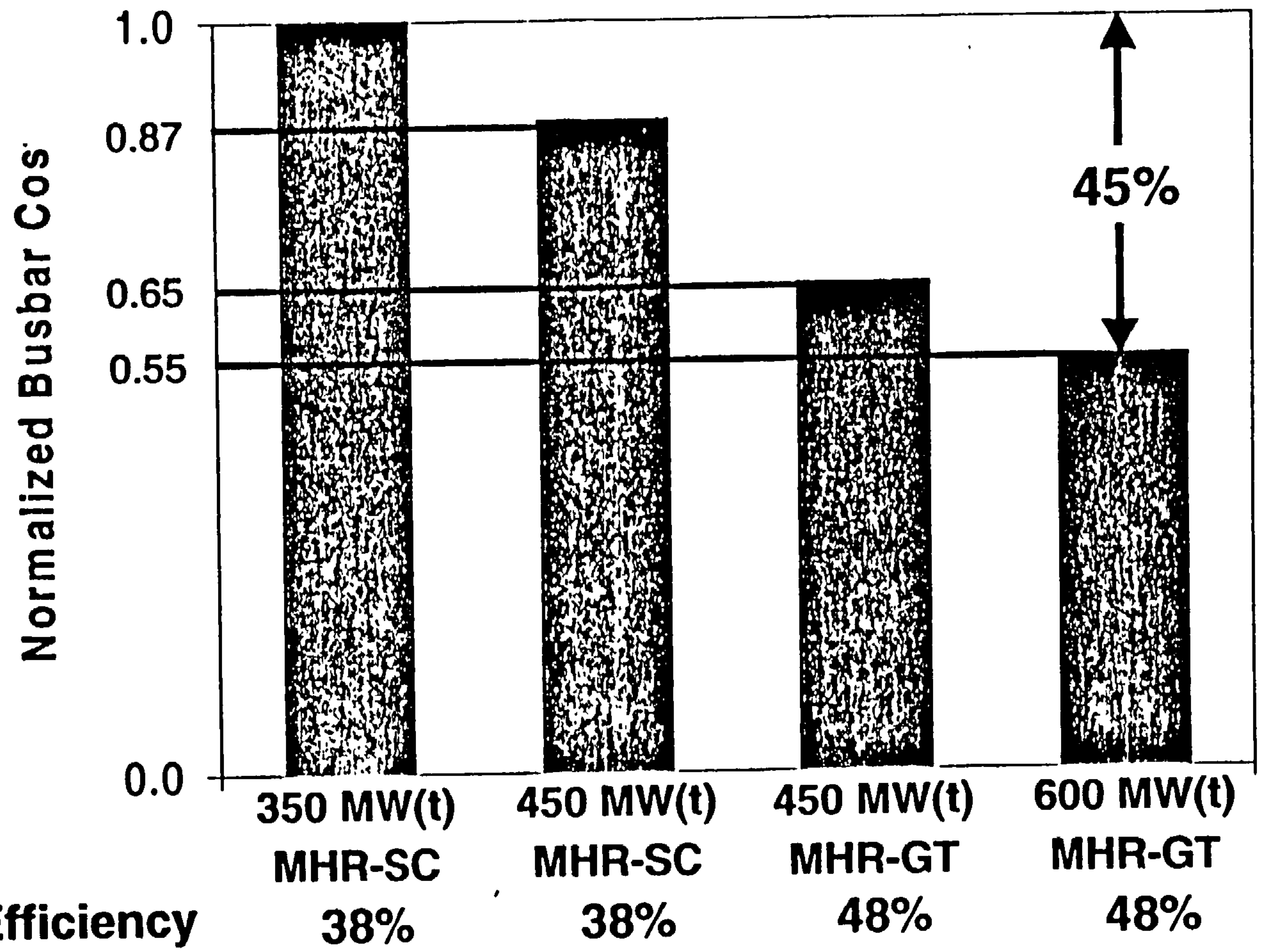
GT-MHR COMBINES MELTDOWN-PROOF ADVANCED REACTOR AND GAS TURBINE

POWER LEVEL 600 MWt

GENERAL ATOMICS



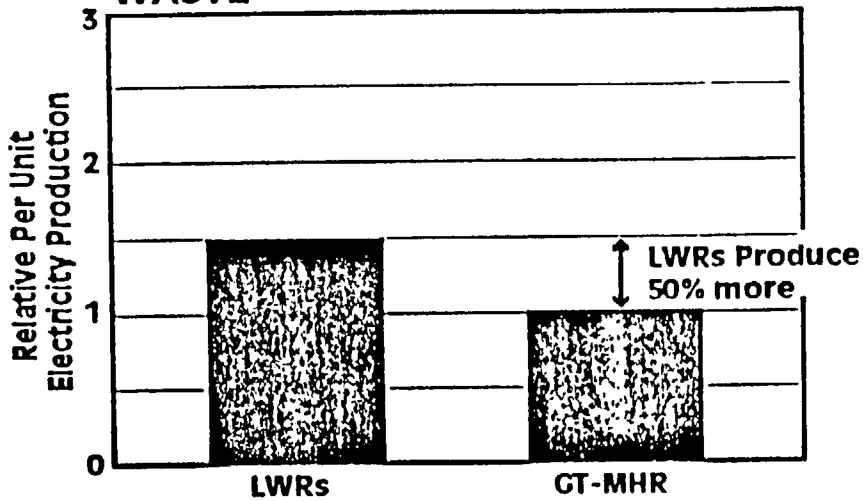
600 MW(t) GT-MHR REDUCES POWER COST BY 45% COMPARED TO 350 MW(t) STEAM CYCLE



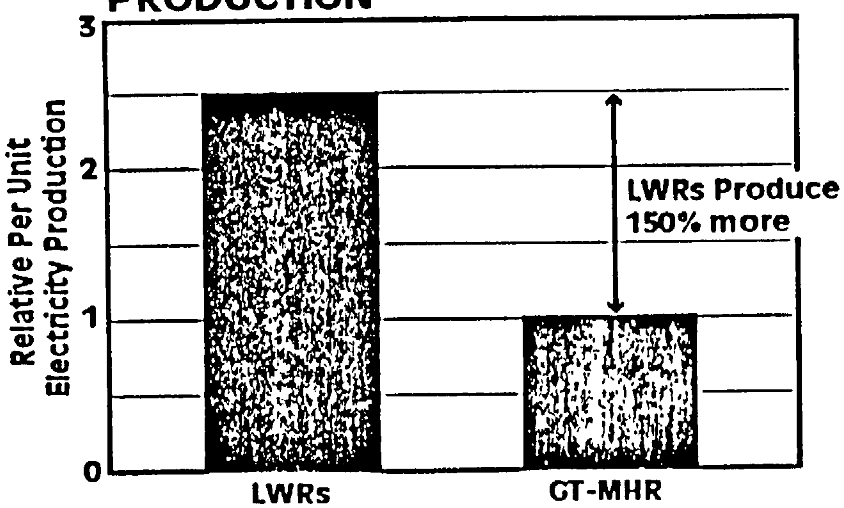
GENERAL ATOMICS

GT-MHR OFFERS MAJOR ENVIRONMENTAL BENEFITS

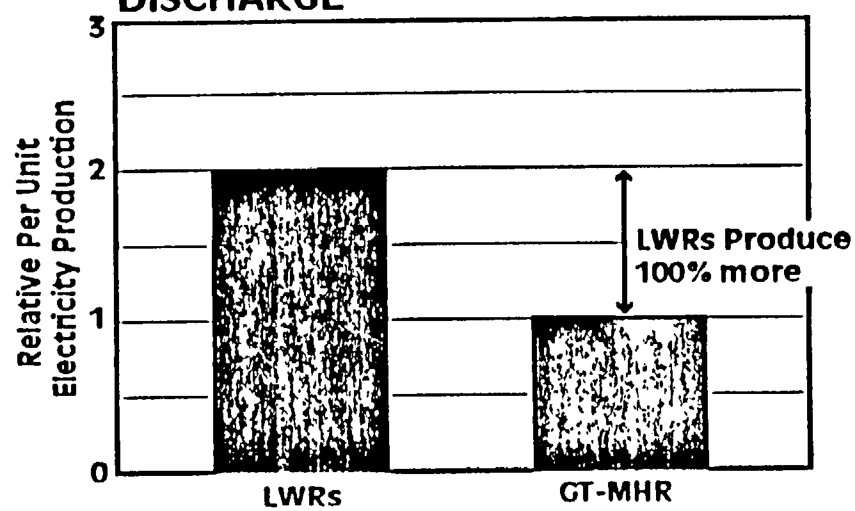
HIGH LEVEL WASTE



ACTINIDE PRODUCTION



THERMAL DISCHARGE



GENERAL ATOMICS

L-592(2)
11-14-95

IN SUMMARY, GT-MHR IS A GENERATION IV SYSTEM

- Inherent safety Features- No core melt
- High thermal efficiency resulting Lower Cost
- Significantly reduced environmental impact
- Superior radio-nuclide retention for long-term spent disposal

GENERAL ATOMICS

GT-MHR COMMERCIALIZATION PROGRAM

PROGRAM DESCRIPTION

GENERAL ATOMICS

GT-MHR NOW BEING DEVELOPED IN INTERNATIONAL PROGRAM

- In Russia under joint US/RF agreement for management of surplus weapons Pu
- Sponsored jointly by US (DOE) and RF (Minatom); supported by Japan and EU
- Conceptual design completed; preliminary design complete early 2002

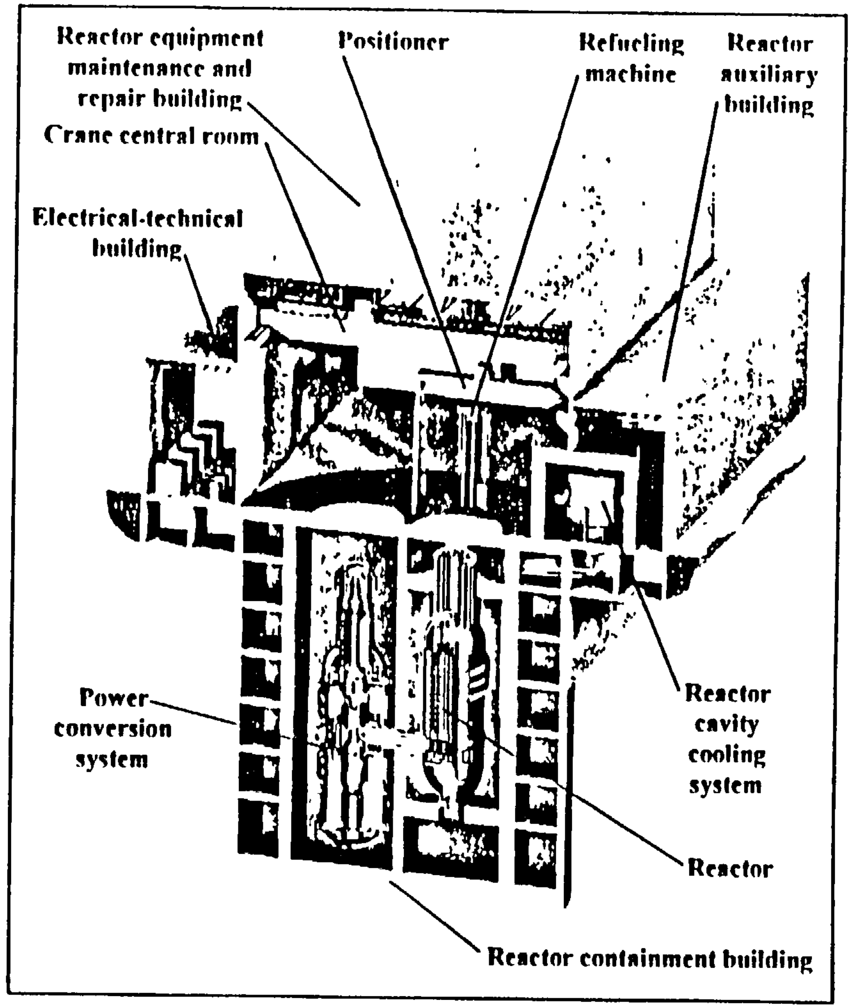
GENERAL ATOMICS

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

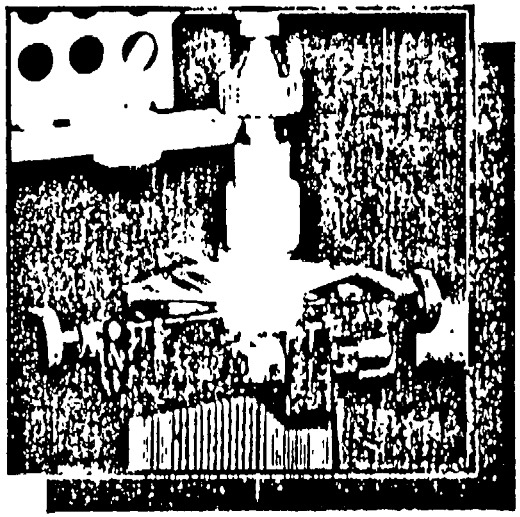
......No fertile component



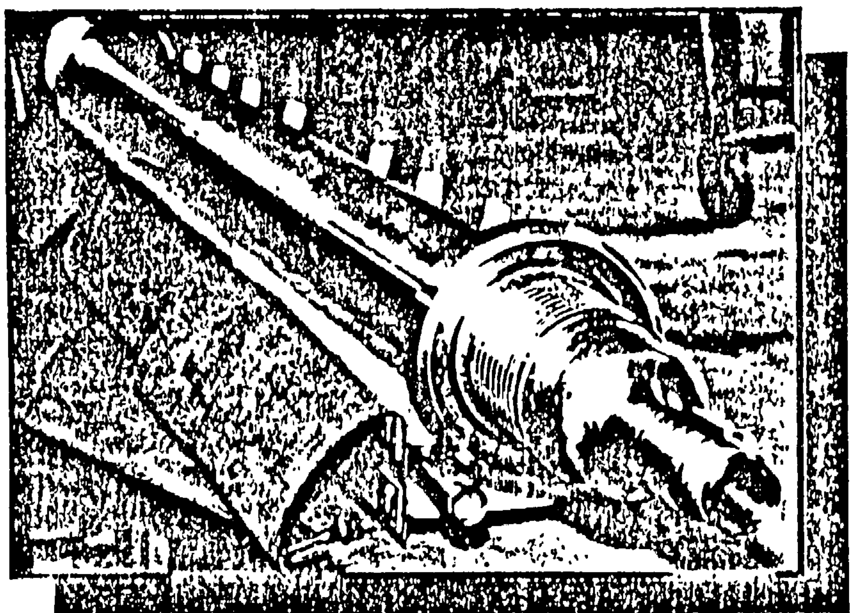
GENERAL ATOMICS

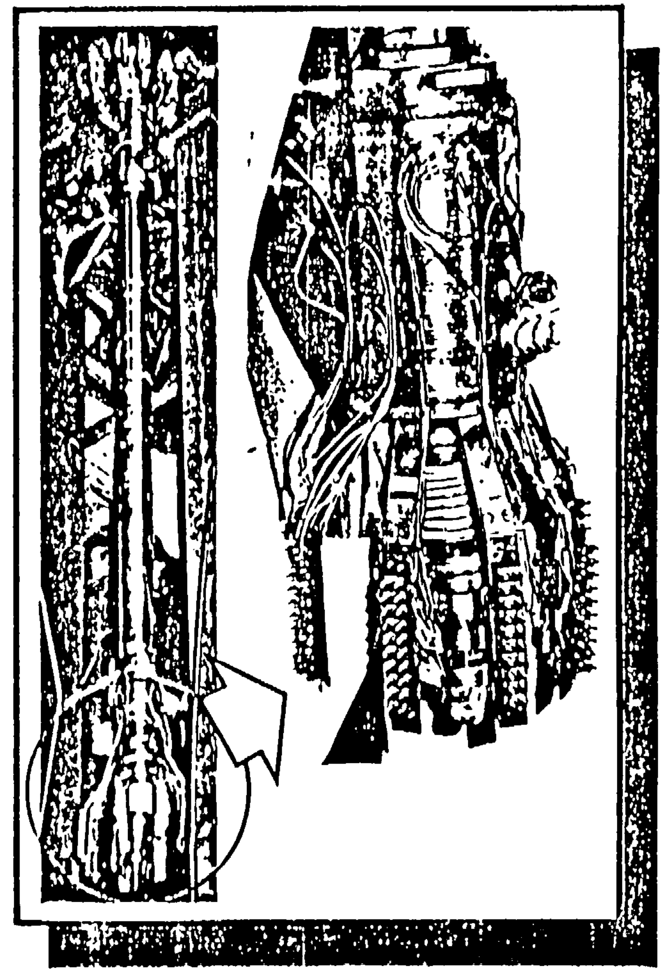
Russian Technological Developments. Recuperator

Heat Exchange Element Fabrication



Recuperator Heat Exchange Element





Tests of full scale heat exchange element in helium test facility

ОКБМ

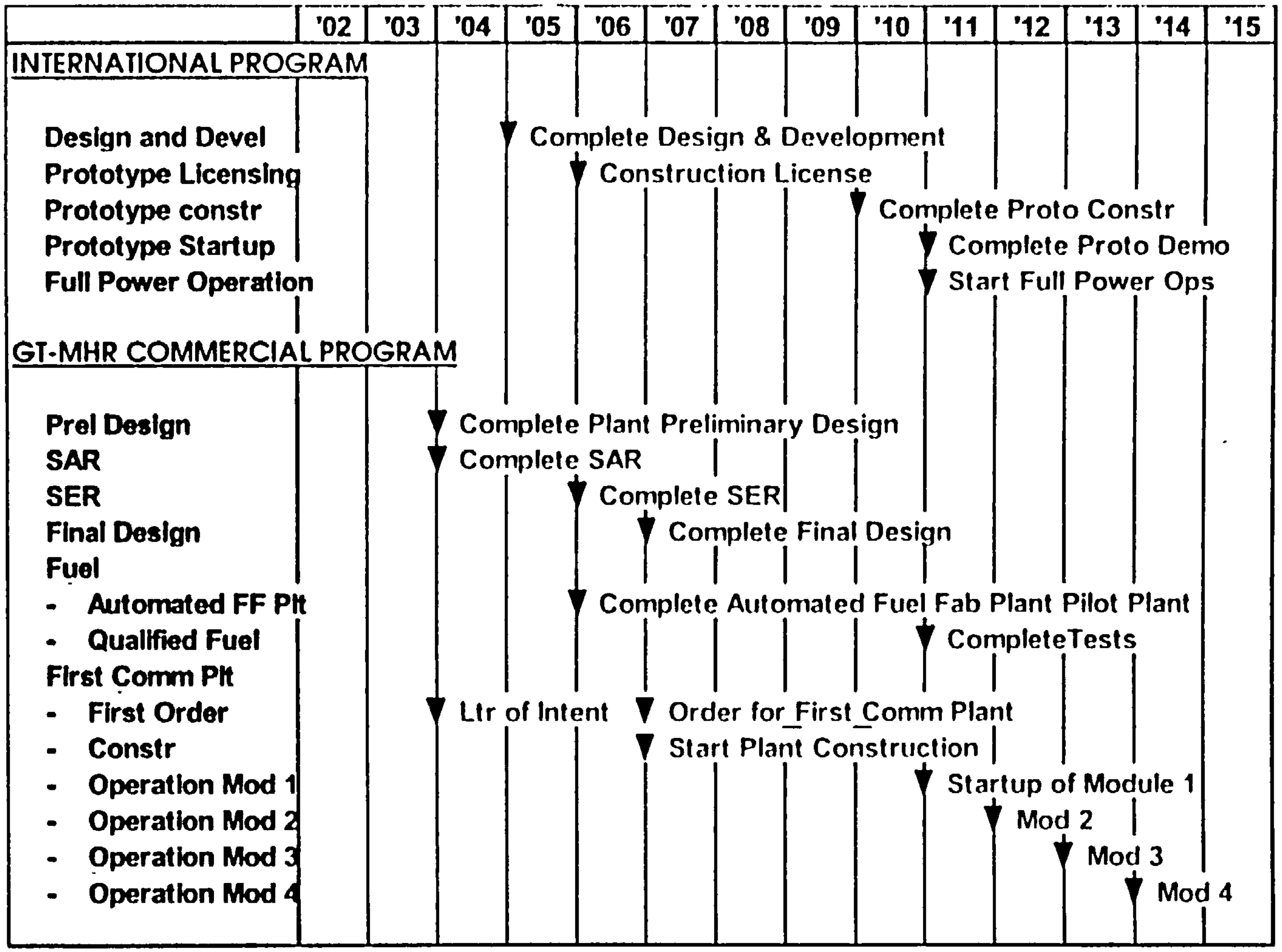
COMMERCIALIZATION PROGRAM

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

Plant construction can start in 5 years

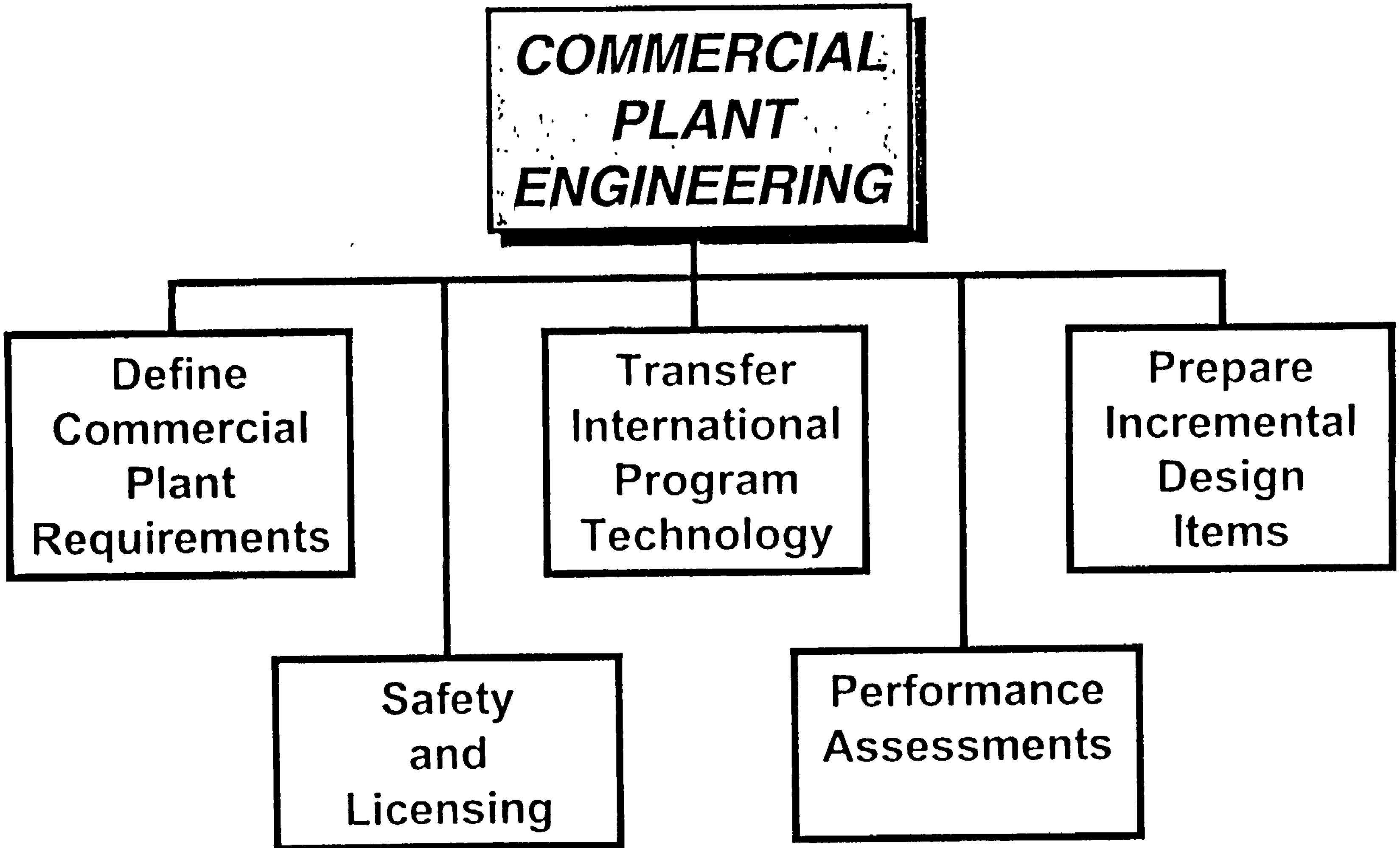
GENERAL ATOMICS

COMMERCIAL PROGRAM FOLLOWS INTERNATIONAL PROGRAM



GENERAL ATOMICS

LIMITED ENGINEER. JG WORK REQUIRED



GENERAL ATOMICS

PLANT REQUIREMENTS PLANNED FROM SEVERAL SOURCES

Externally Imposed Requirements
- US regulatory requirements
- US codes and standards

Utility/User Requirements
(safety, economics, etc)

Technology from International Program

COMMERCIAL PLANT REQUREMENTS

GENERAL ATOMICS

TECHNOLOGY TRANSFER ACTIVITIES

INTERNATIONAL PROGRAM TECHNOLOGY

- Preparation of SDDs to US standards
  - info from equivalent docs prepared to Russian stds
- Adaptation of design & tech dev reports
  - verify compliance to US requirements
- Adaptation of dwgs & specs
  - convert to US codes and stds

GENERAL ATOMICS

INCREMENTAL ENGINEERING WORK ACTIVITIES

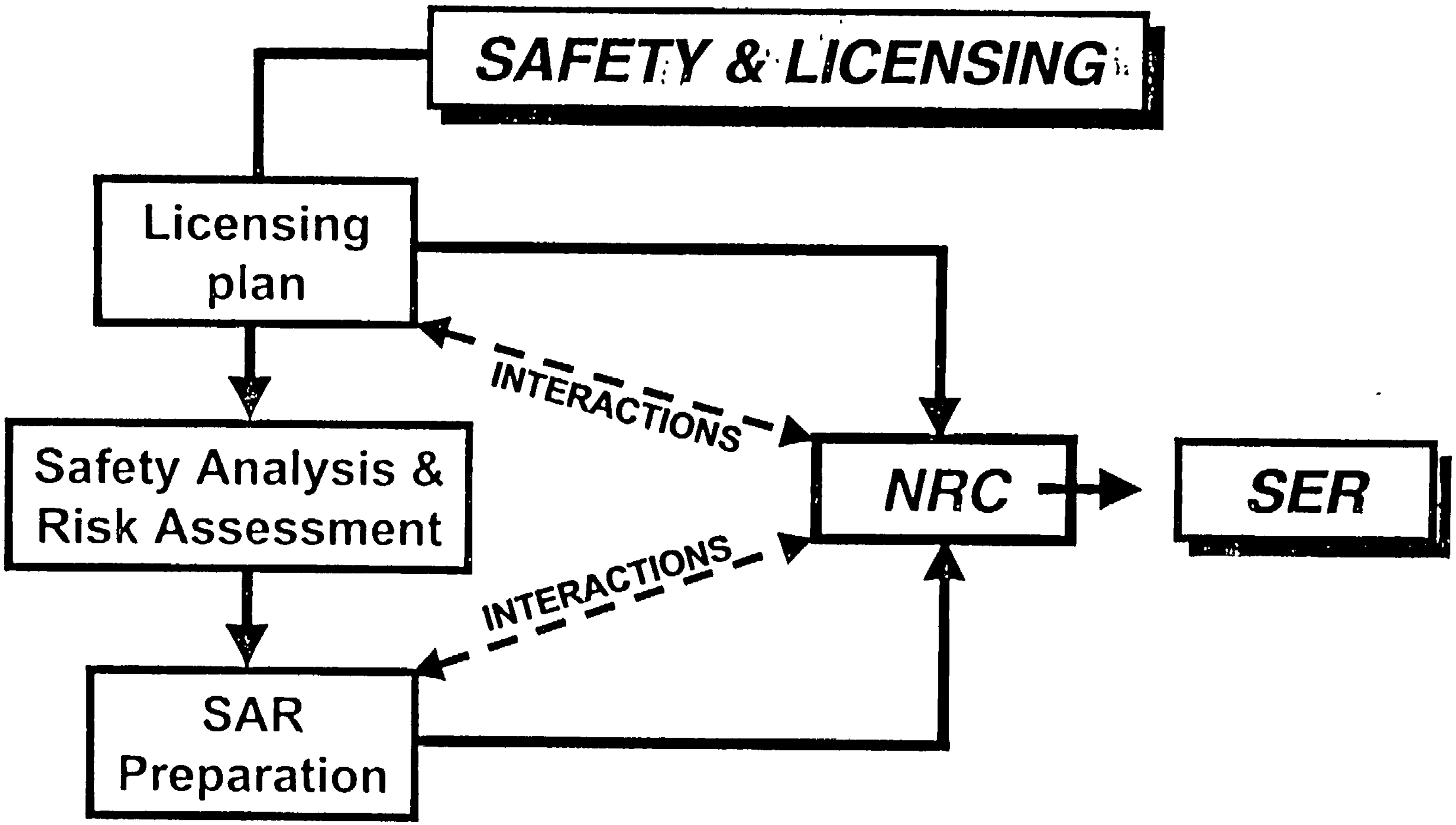
INCREMENTAL ENGINEERING

- Uranium Core Design
- RCCS for LPV
- Uranium Fuel Fab Plant Design & Qual
- 60 hz Power Generation
- Low Press, Vented (LPV) Reactor Bldg

..........No New R&D

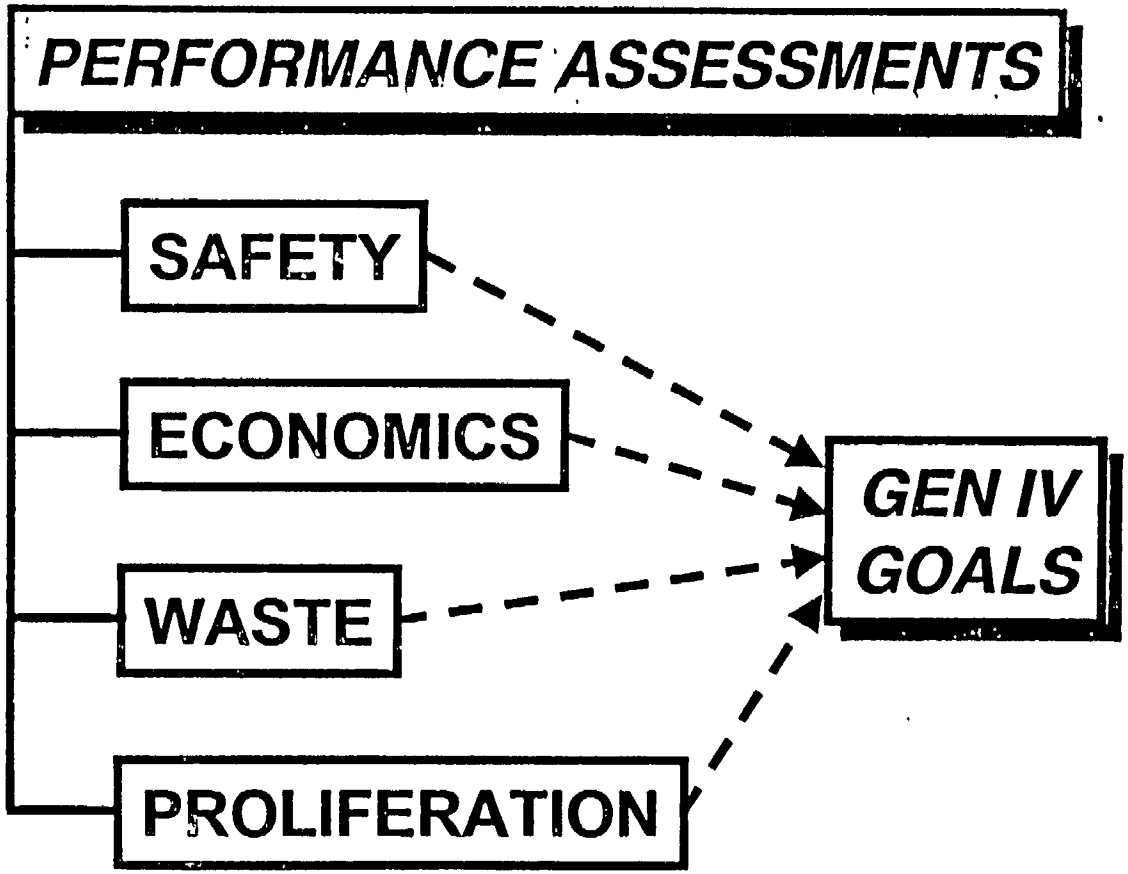
GENERAL ATOMICS

SAFETY & LICENSING ACTIVITIES



GENERAL ATOMICS

PERFORMANCE ASSESSMENT ACTIVITIES PLANNED



GENERAL ATOMICS

COMMERCIAL PROGRAM SUMMARY

- GEN IV PLANT
- COST EFFECTIVE
- NEAR TERM

GENERAL ATOMICS

JNW

IRIS

International Reactor Innovative and Secure

M. D. Carelli

Presentation to NRC

May 7, 2001

carellmd@westinghouse.com
Ph: 412-256-1042
Fax: 412-256-2444

Westinghouse Science & Technology

PURPOSE

- Introduce IRIS
- Feedback from NRC Staff needed to maintain progress
- Outline needed testing program

(W) - wants technical feed back

AGENDA

- Overview
  - Team Partnership
  - Funding
  - Schedular Objectives
- Neutronics and Fuel Selection
- Configuration (Integral vessel, internal shield, steam generators)
- Enhanced Safety Approach (Safety by Design)
- Maintenance Optimization
- Technology Gaps and Regulatory Issues
- Conclusions

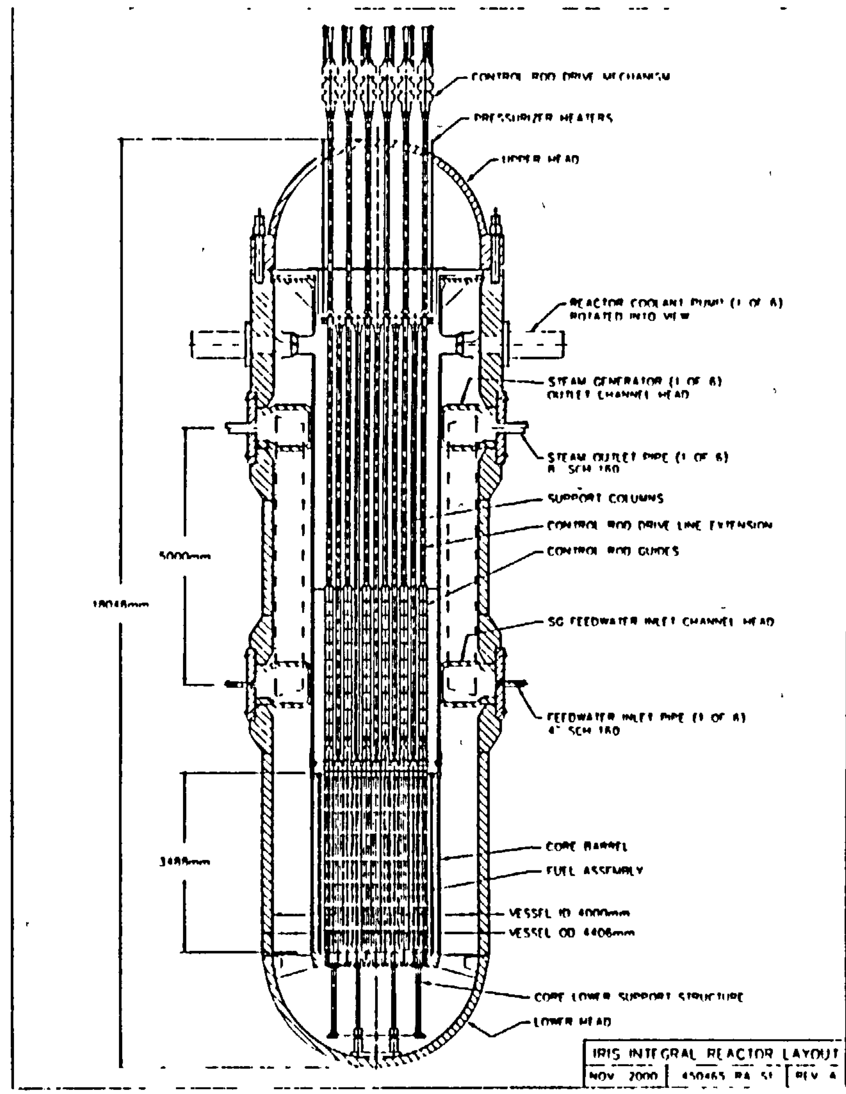
OVERVIEW

– emphasizing proliferation resistance

– 4 steam lines & 4 feed lines for 8 SGs

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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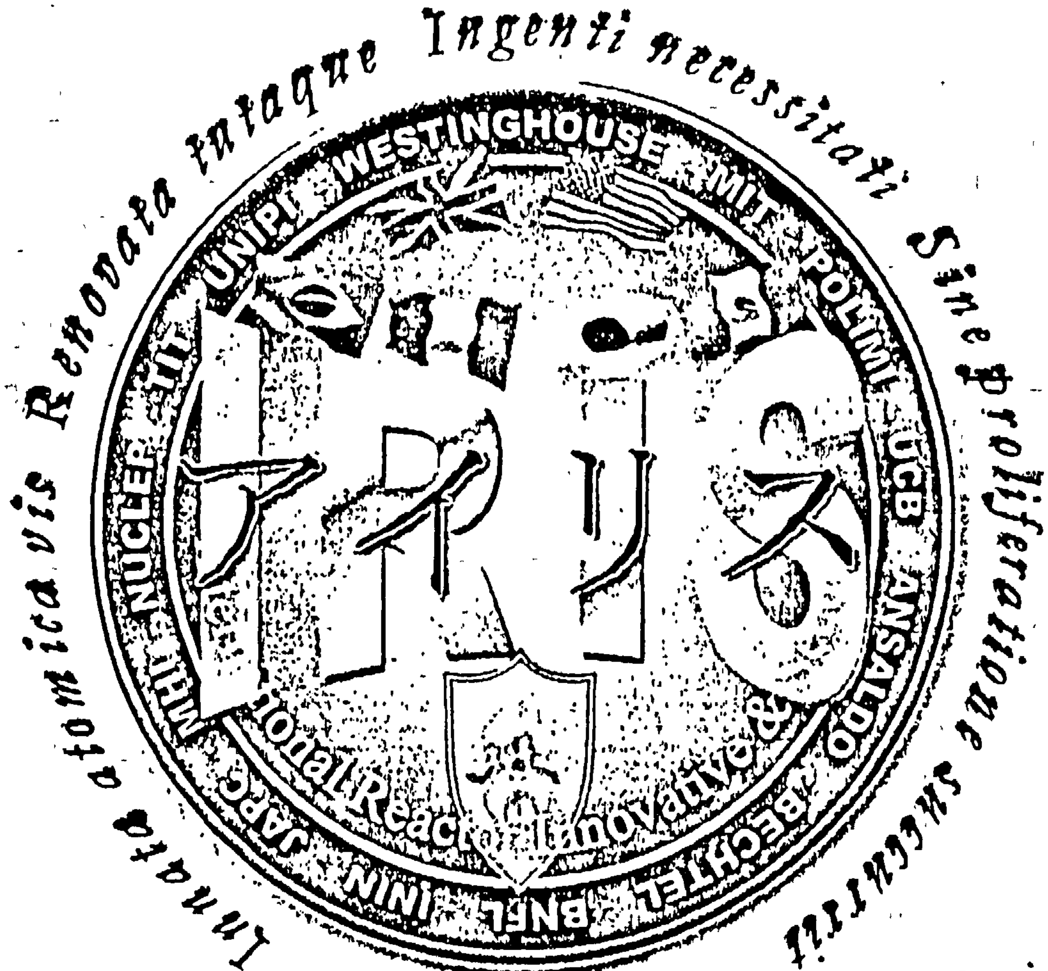
WHY IRIS ?

Originally: To respond to DOE Generation IV solicitation

Design feature	DOE's NERI Requirement			
	Proliferation resistance	Enhanced Safety	Economic competitiveness	Reduced waste
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	✓		✓	

Evolved into: Attractive commercial market entry



Spain joined in April 2001

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
Nuclear Energy Commission, Brazil (CNEN) (Pending)	•		•	Transient, structural analyses, testing
Oak Ridge National Laboratory, USA (ORNL) ~~(Pending)~~ joined in May 2001	•		•	Core analyses, safety, cost evaluation, testing
Associates				
University of Tennessee, USA			•	Modularization, transportability
Ohio State University, USA			•	Novel In-Core Power Monitor

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	? → marketing decision in 2003
• Complete SAR	2005
• Obtain design certification	2007
• First-of-a-kind deployment	2010-2015

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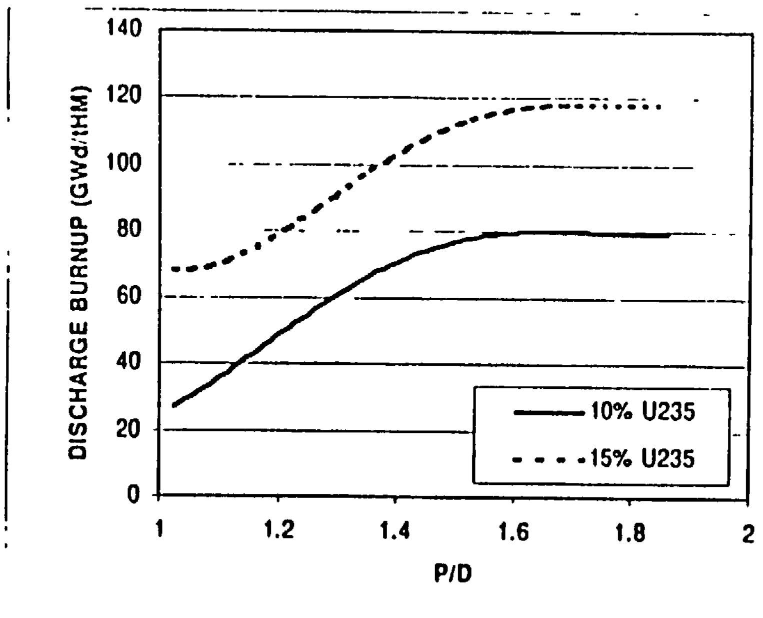
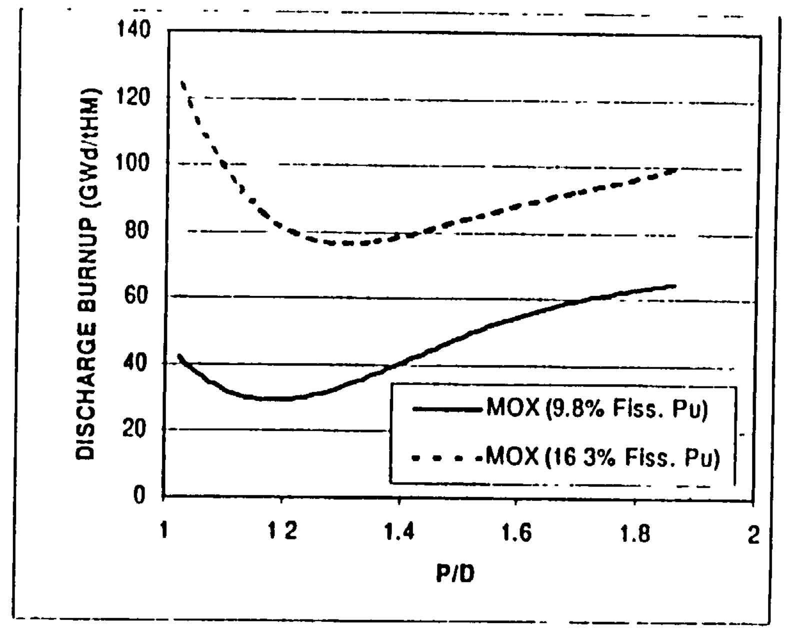
NEUTRONICS AND FUEL SELECTION

VARIOUS CONFIGURATIONS YIELD LONG LIFE





(Discharge burnup based on End-of-Life Kinf = 1.075)

- UO_2 fuel
  - open lattice
- MOX fuel
  - lower enrichment - open lattice
  - higher enrichment - tight lattice

- all negative power coefficients

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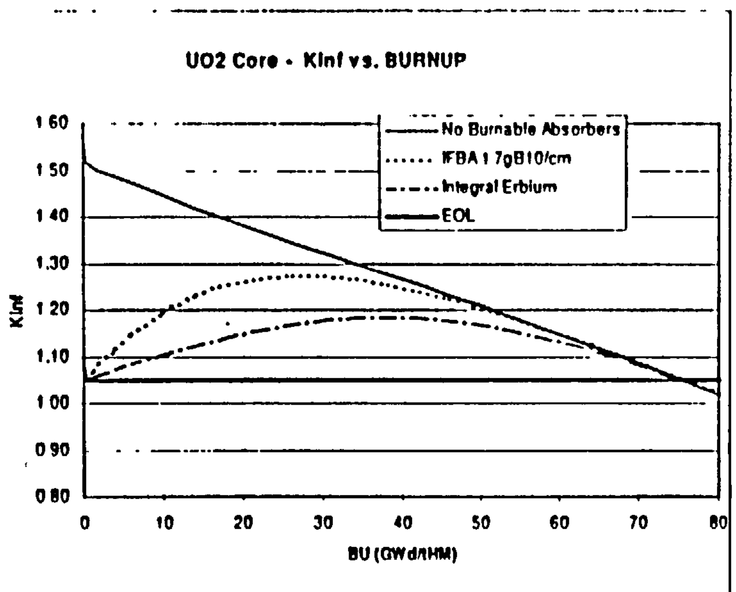
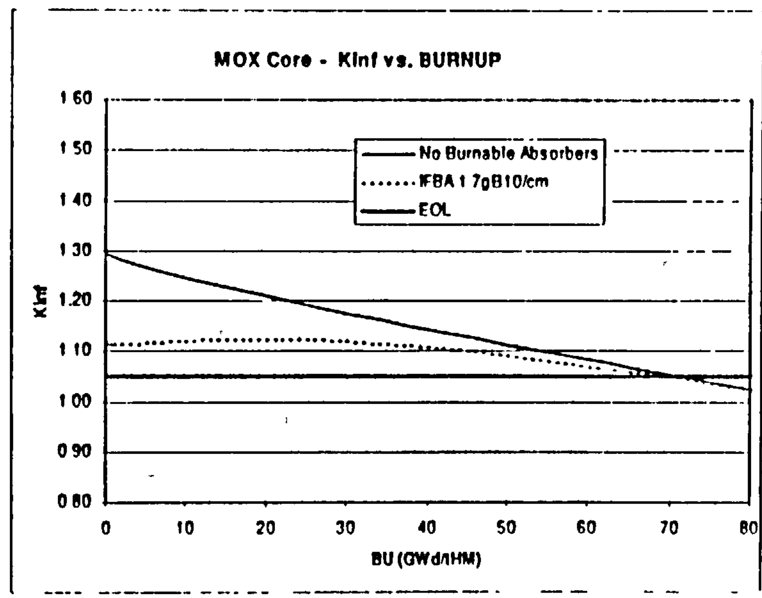
EXCESS REACTIVITY CONTROL BY BURNABLE ABSORBERS

Lattice cell analyses, EOL assumed at Kinf=1.05
IFBA = Integral Fuel Burnable Absorber (ZrB_2 coating)
Er = Erbium mixed with fuel





UO_2:

- IFBA reduces reactivity swing Δk from 50% to 22%
- Erbium reduces reactivity swing Δk from 50% to 14%

MOX:

- IFBA reduces reactivity swing Δk from 25% to 7%

5/2/01
Viewgraph 15

Goal = no Soluble Boron

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UO_2 VERSUS MOX

UO_2 FUEL
- commercial PWR experience
- U.S. policy

MOX FUEL
- lower initial excess reactivity
- fuel fabrication available (BNFL)
- disposal of available plutonium
- of interest to international IRIS partners

ENRICHMENT CONSIDERATIONS

- 8-year core requires higher enrichment than current practice
  - New fabrication facilities
  - Regulatory approval
- 8-year core will attain higher burnup than current state-of-the-art
  - Data and models needed
  - Licensing review
- Not consistent with early deployment objective

IRIS DESIGN OPTIONS

IRIS 5-YEAR DESIGN (~ 5% equivalent)

CURRENT FUEL TECHNOLOGY

PROVIDES MINIMUM-RISK PATH FORWARD

(DETAILED CORE DESIGN IN PROGRESS)

IRIS 8-YEAR DESIGN (goal 2015 – 2020 time frame) ~ 70,000 Md/T

BOTH UO_2 and MOX MAY BE USED

EMPHASIZES PROLIFERATION RESISTANCE

(SCOPED INTERCHANGEABLE CORE DESIGN)

TIGHT LATTICE CORE/HIGHER ENRICHMENT/NOVEL FUEL TYPES (~ 120,000 – 130,000 Md/T)

POTENTIALLY FURTHER EXTEND CORE LIFE

(RESEARCH EFFORTS CONTINUING)

Goal for Univ: Research

IRIS 335 MWe CORE DESIGN APPROACH

PROLIFERATION RESISTANCE

IMPROVED ECONOMICS

PATH FOR FUTURE ENHANCEMENTS

FUEL AVAILABLE NOW

EARLY DEPLOYMENT

DEMONSTRATES EXTENDED MAINTENANCE

PROVES INTEGRAL REACTOR FEATURES

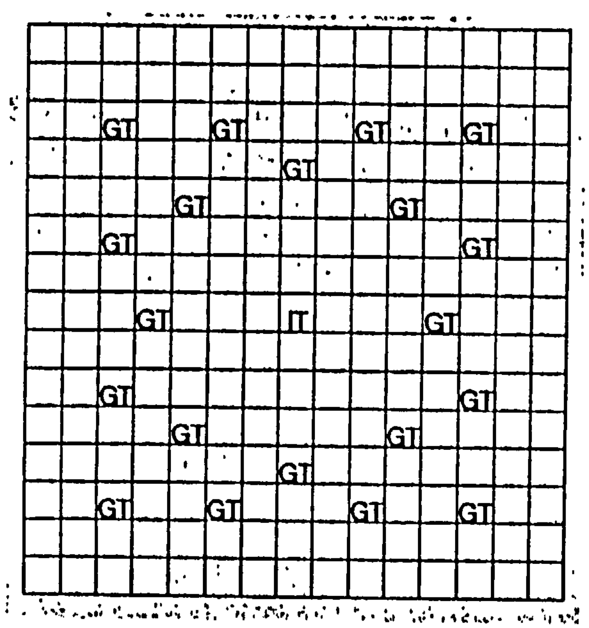
FIRST CORE FUEL ASSEMBLY DESIGN UTILIZES CURRENT TECHNOLOGY

- 1000 MWt
- 89 FA, square lattice
- 5-year core lifetime
- 4.95 w/o U235
- 15x15 square lattice
- 14 ft active core height
- extended gas plenum
- ZIRLO-type cladding
- Pitch = 0.592"
- p/d = 1.4
- 1 instrumentation tube
- 20-24 control rod "fingers"
- 4 kW/ft average power
- ~40-45,000 MWd/tHM average discharge burnup

IRIS 335 MWe FUEL ASSEMBLY AND CORE CONFIGURATION

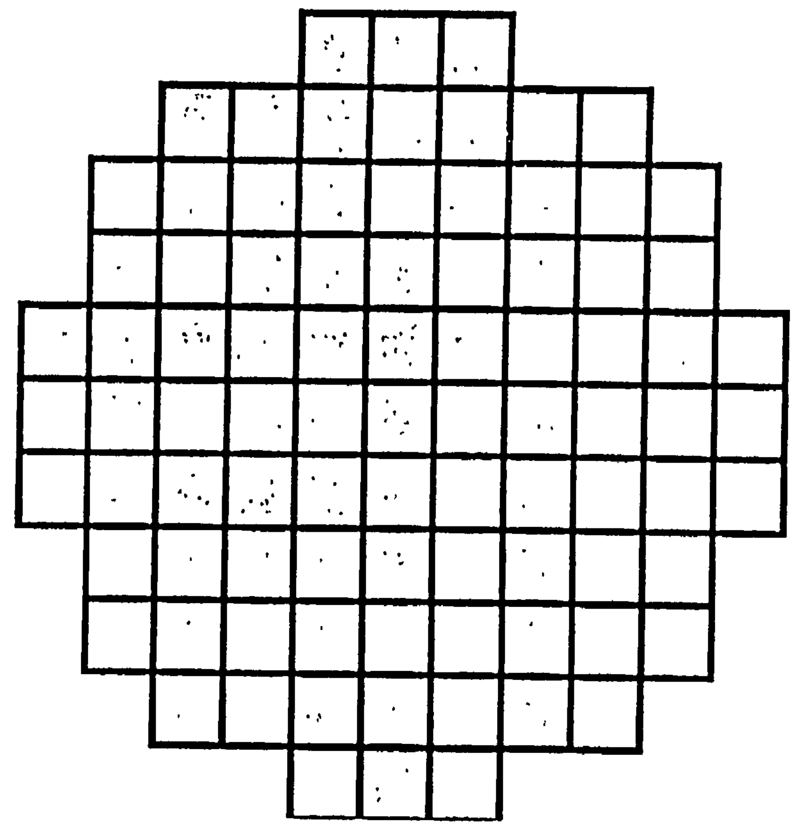
FUEL ASSEMBLY INCORPORATES EXISTING W DESIGN FEATURES:

- 15x15 fuel assembly
- XLA (14 ft active core)
- Robust design



— more benign than current PWR fuel assemblies

CORE CONFIGURATION (1000 MWt) INCLUDES 89 FUEL ASSEMBLIES



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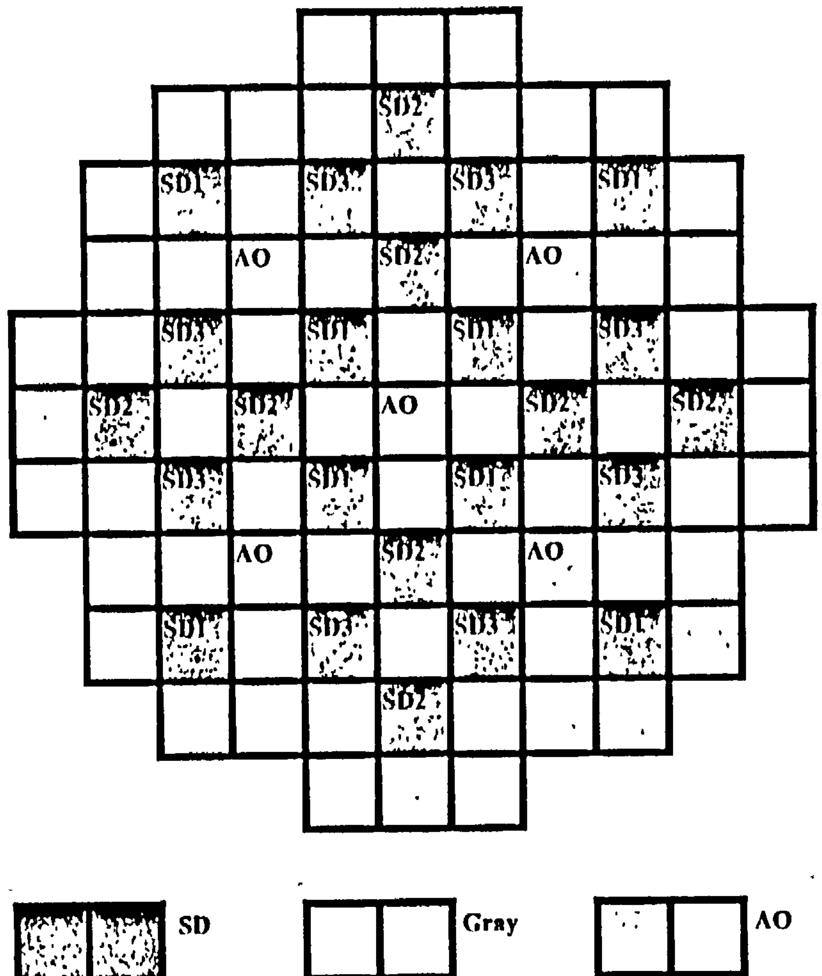
CONTROL RODS

Shutdown (black) - SDB1, SD2, SD3 (8 RCCA each bank)
Excess reactivity control (gray) - 4 banks, 8 RCCA each
Axial offset control



Red Green

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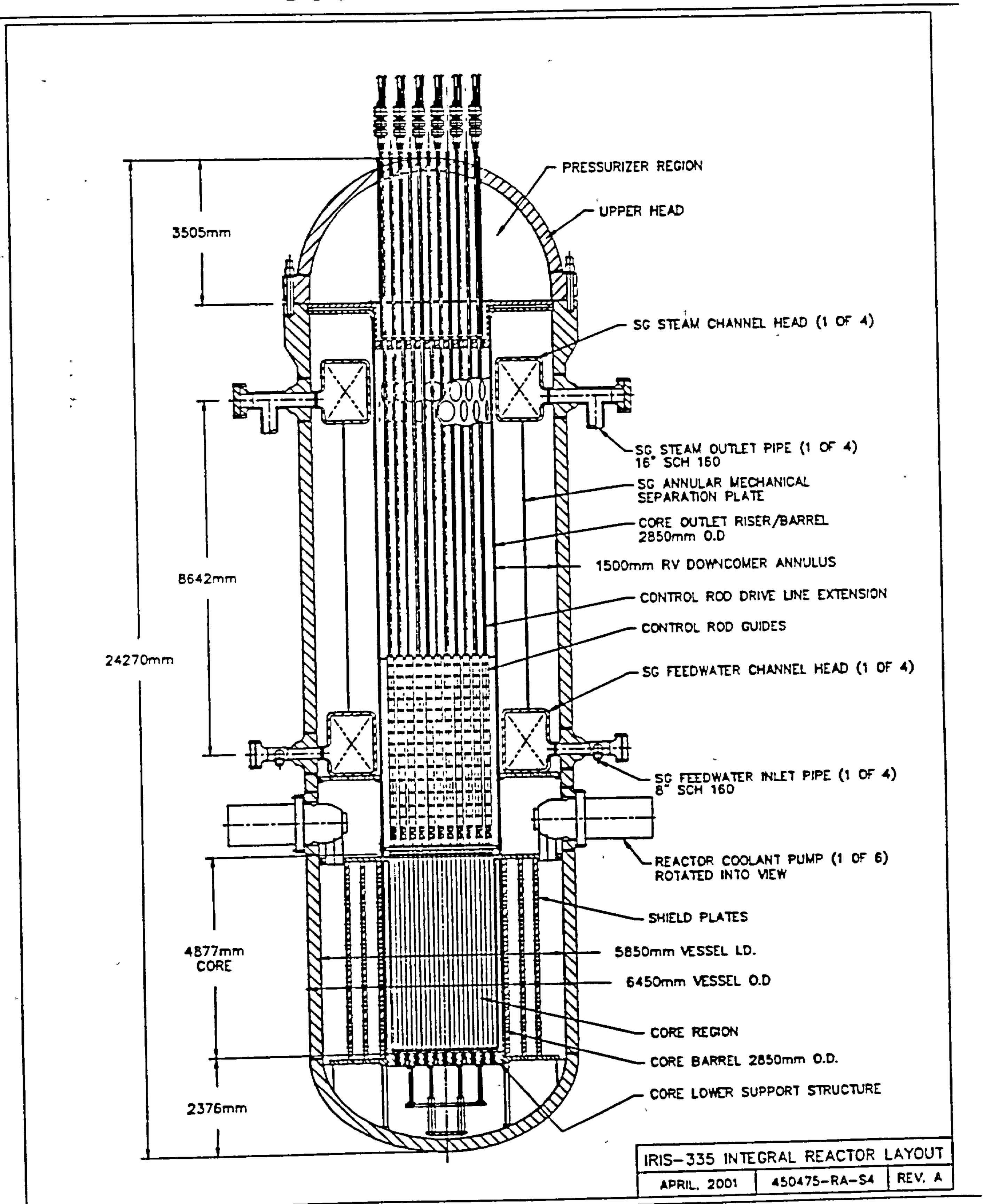
CONFIGURATION

- Goal for size of Rx vessel is to be able to ship by Barge.
this limits power to ~ 1000 MW thermal.

335 MWe Vessel



INTERNAL SHIELDS

→ Neutron shield between annular SGs and Rx core

Steel volume fraction (%)	Vessel Activation at shutdown (Bq/g)	Ratio	Dose rate(*) (μSv/h)	Ratio
20	310	1	3	1
↓ 20+B_4C (**)	10	1/31	0.006	1/500
30	30	1	0.14	1
↓ 30+ B_4C (**)	8	1/4	0.002	1/70

(*) Evaluated on the inner biological shield surface
(**) A boron carbide fraction of 10% is considered in the shield

Dept. of Nuclear Engineering - Politecnico di Milano - Italy - 2001 April

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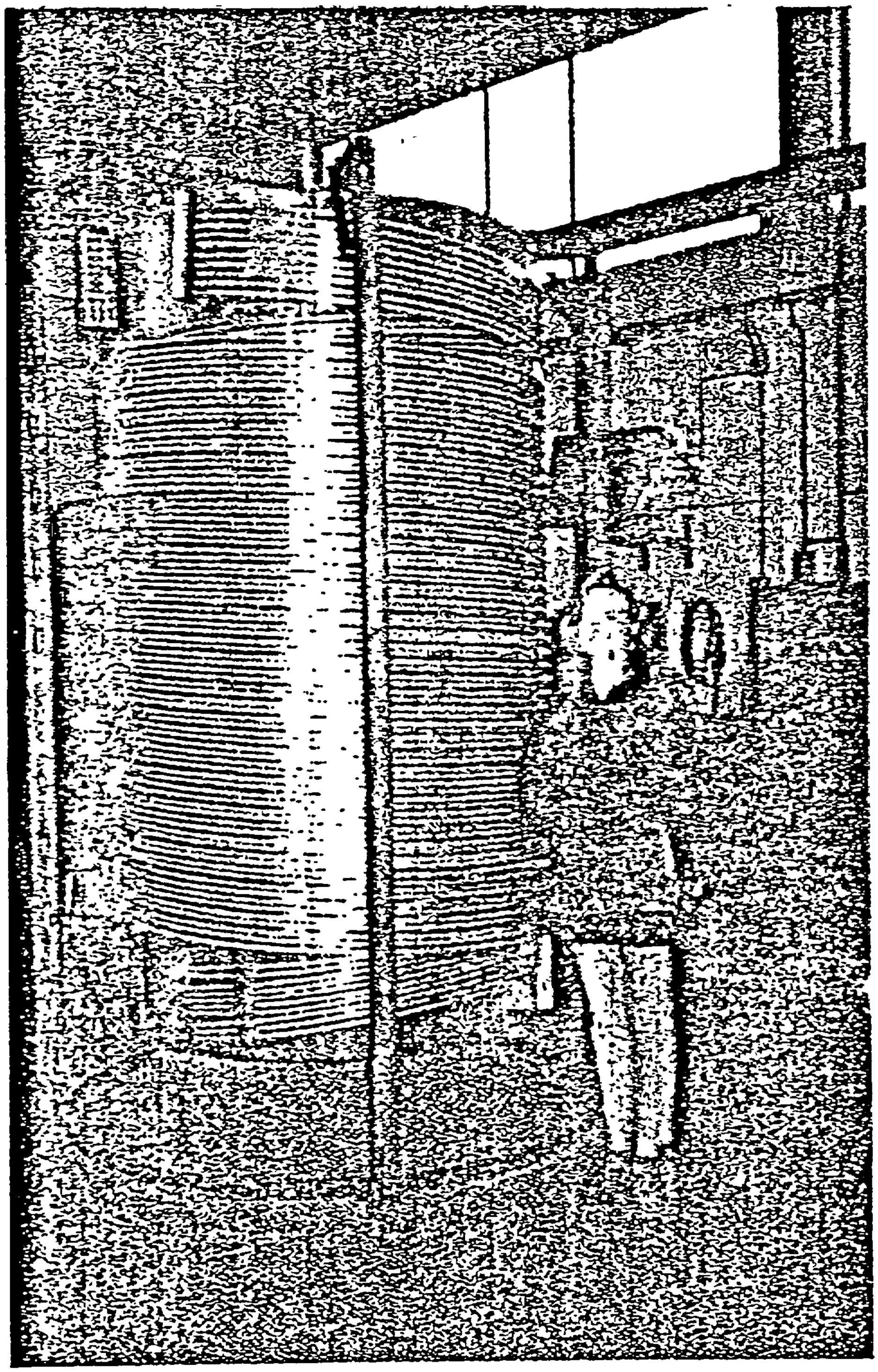
INTERNAL SHIELDS

– Results in much less activation of Rx vessel

- No restrictions to workers in containment
- Simplified decommissioning
- Vessel (minus fuel) acts as sarcophagus



20 MW mock-up of the helical-tube SGU
Test campaign at SIET

HELICAL STEAM GENERATOR

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

8 SGs provides redundancy

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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"

– Integral design eliminates Rx loop accidents (pipe breaks)

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	High degree of natural circulation	LOFAs (e.g., pump seizure)	Either eliminated (full natural circulation) or mitigated consequences (high partial natural circulation)
	Can accommodate internal control rod drives	Reactivity insertion due to control rod ejection	Can be eliminated
Low pressure drop flow path and multiple RCPs	N-1 pumps keep core flow above DNB limit, no core damage occurs	LOFAs (e.g., RCP shaft break or rotor seizure)	Condition IV accident eliminated
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 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> → for spherical containment

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)

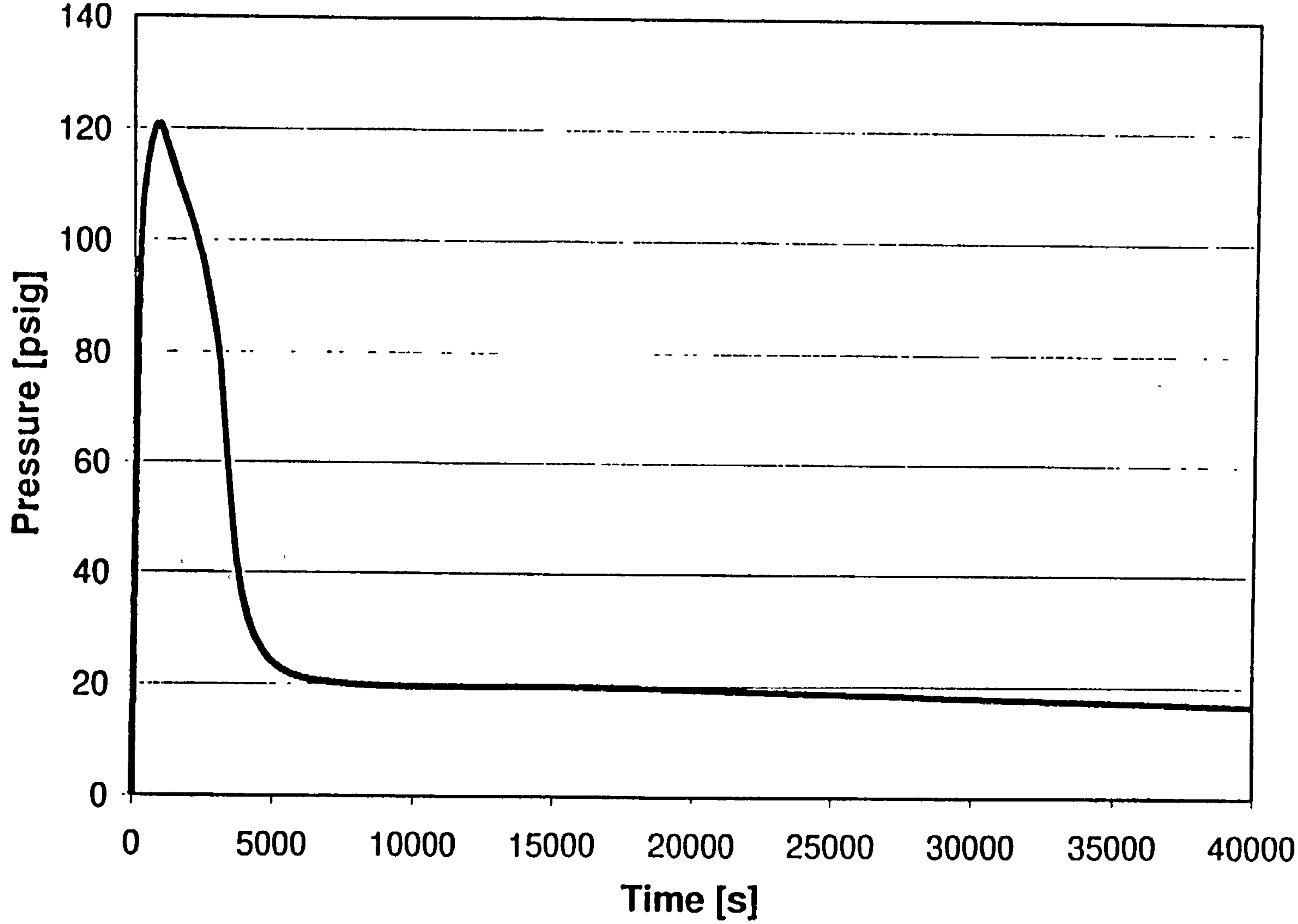
IRIS CONTAINMENT (100/335 MWe)

- Spherical, steel containment, 20/25 meter diameter
- ~15/12 bar_g design pressure (220/175 psig)
- Small, elevated suppression pool limits peak pressure to ~ 9 bar_g (130 psig) and can provide gravity driven core makeup if needed
  - 150/375 m^3 water
  - 300/750 m^3 air
- RV in cavity that floods to level above core
- External air/water cooling of steel shell
- Refueling performed through closure head directly into fuel building

ANALYSES PERFORMED

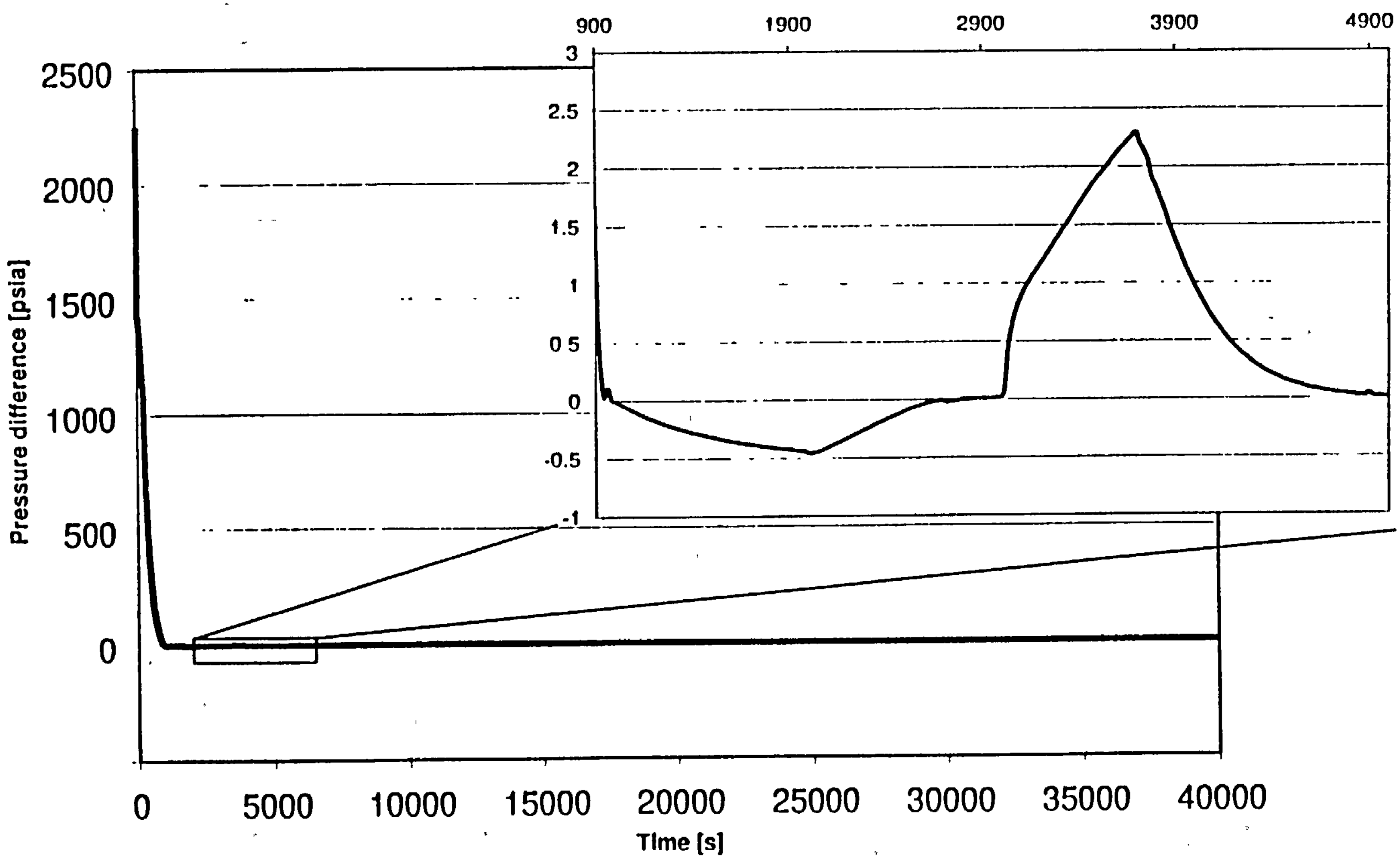
- Break size: 1, 2, 4” holes in Rx vessel
- Elevation: Bottom of vessel, above core (inside and outside cavity), 12.5 m above bottom
- No water makeup or safety injection
- Three codes provided consistent results
  - Proprietary (POLIMI)
  - GOTHIC (Westinghouse)
  - FUMO (Univ. Pisa)

HIGHER CONTAINMENT PRESSURE DECREASES QUICKLY



– steam condenses & water
goes back into vessel → core
heat removal through SGs. SGs must be reliable

REACTOR VESSEL/CONTAINMENT PRESSURE DIFFERENTIAL EQUALIZES QUICKLY



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Viewgraph 34

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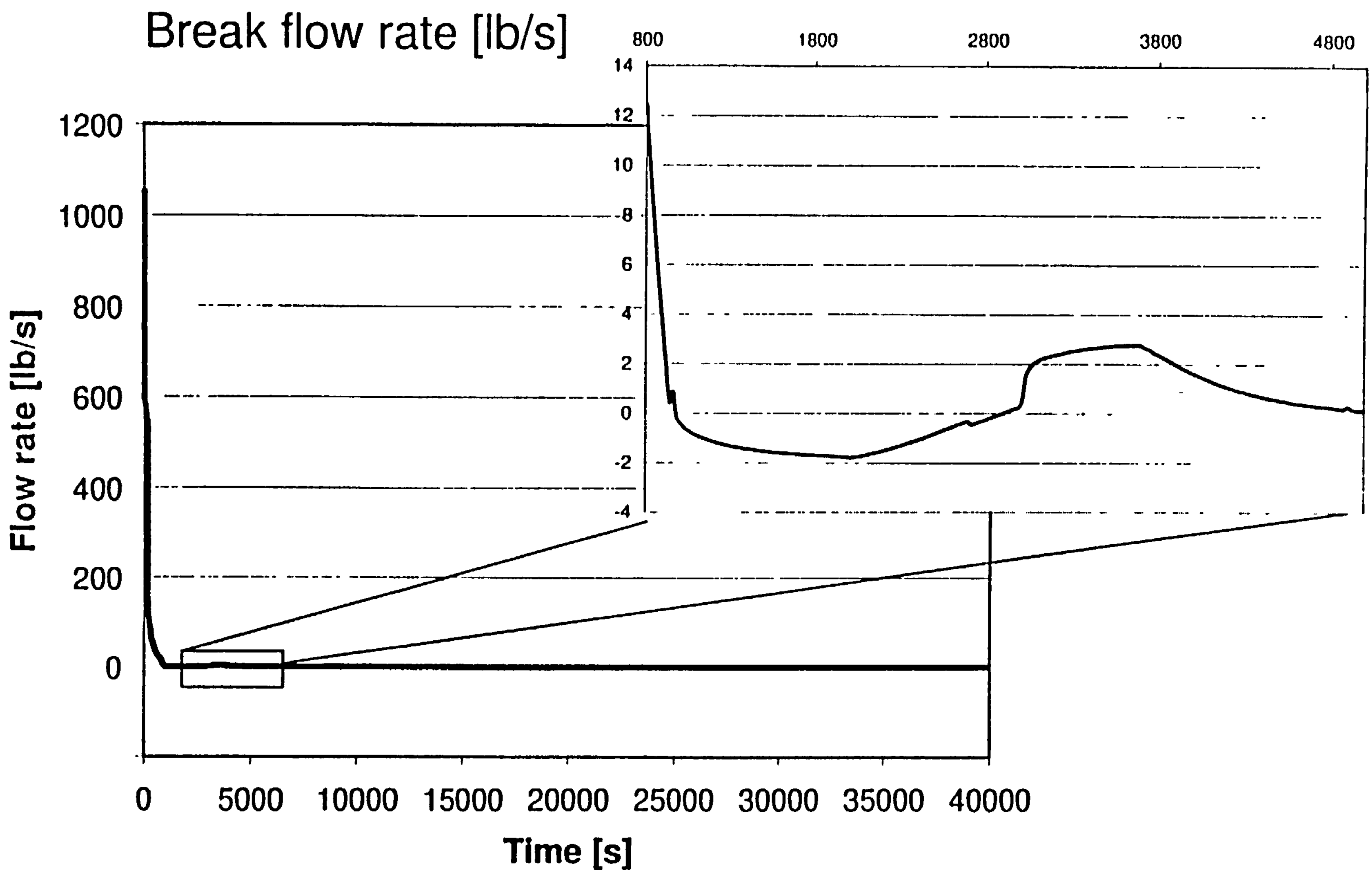
COOLANT FLOW THROUGH RUPTURE DROPS QUICKLY



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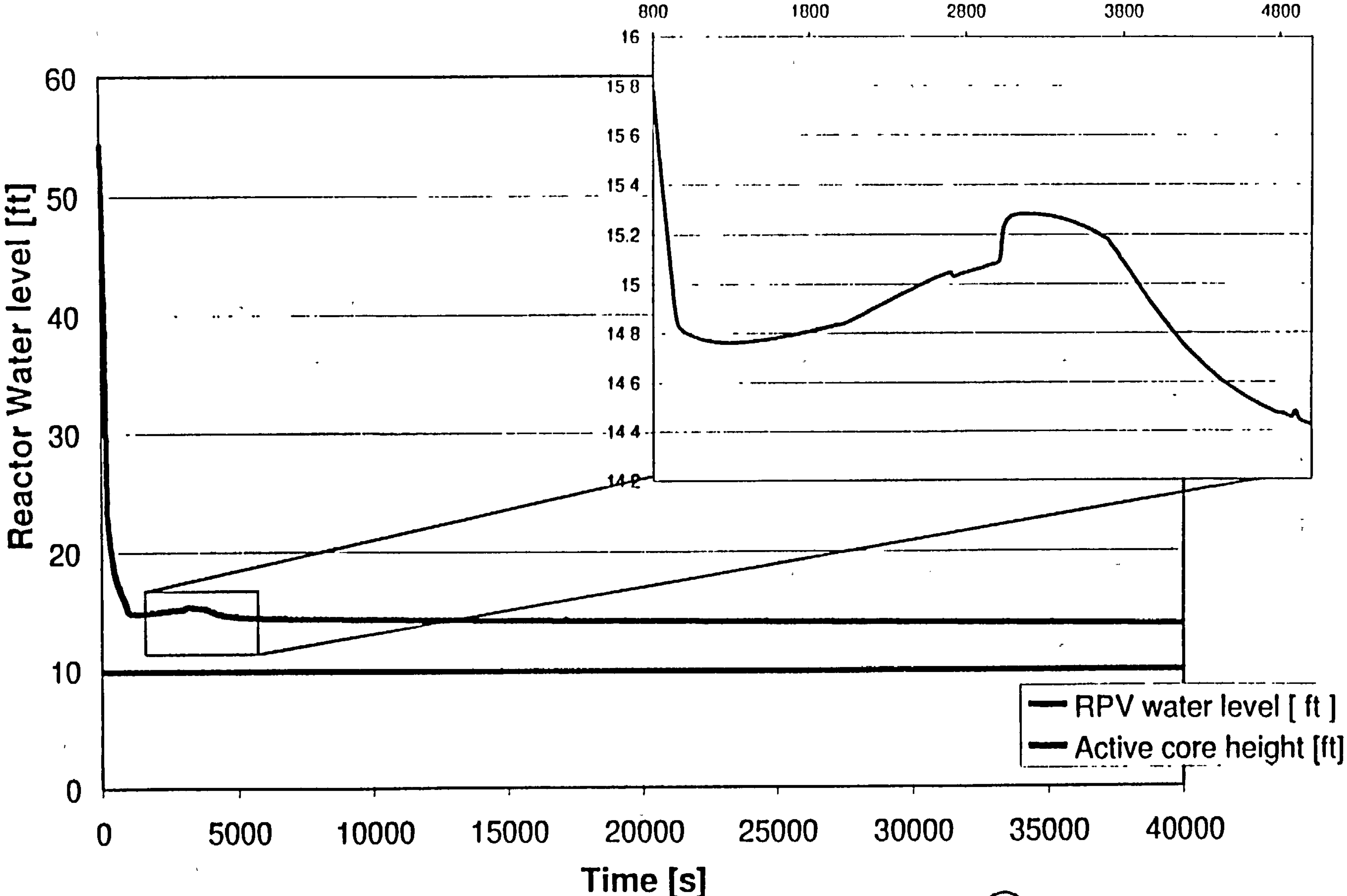
CORE REMAINS SAFELY COVERED FOR EXTENDED PERIOD OF TIME



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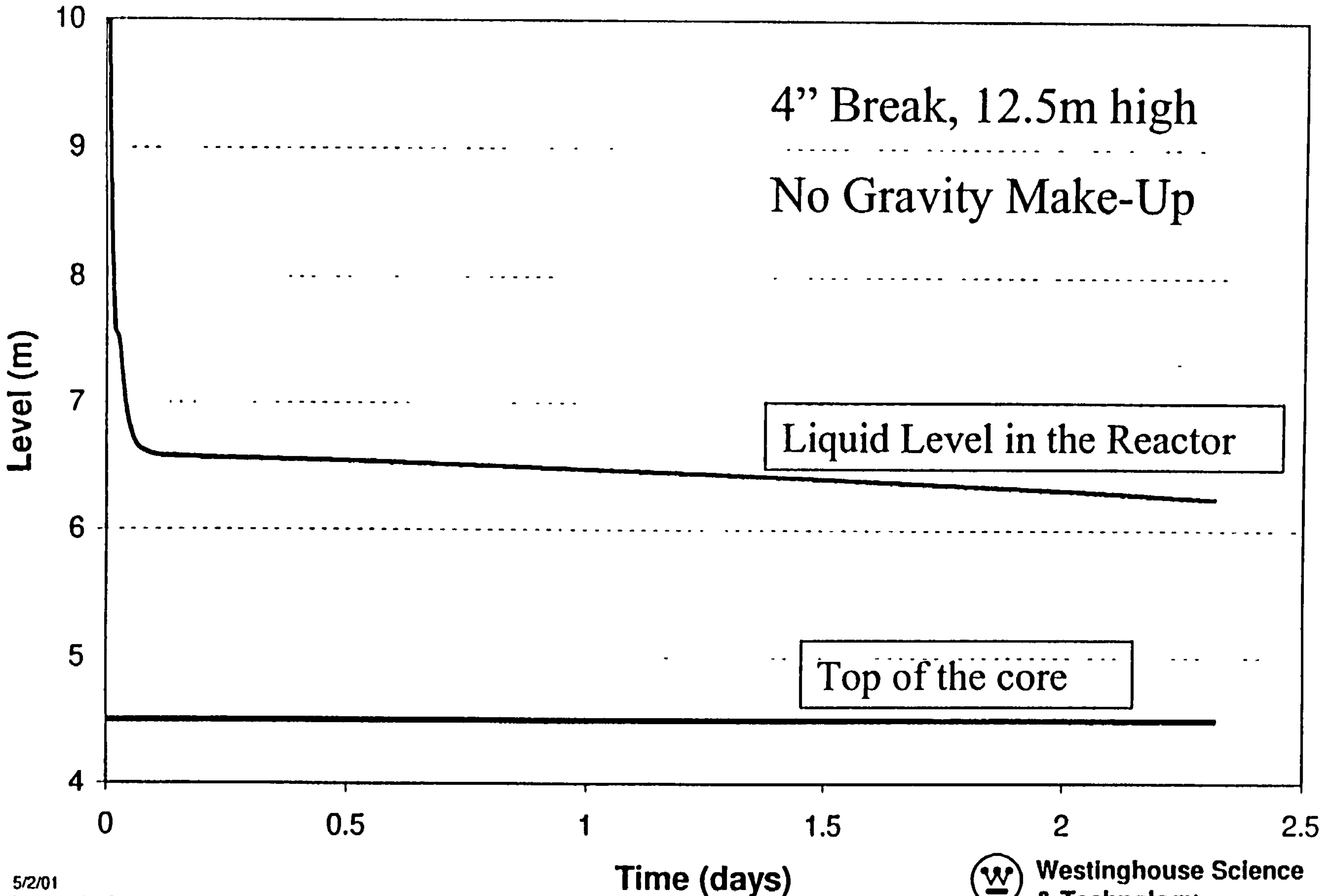
CORE STILL UNDER 2 METERS OF WATER AFTER 2 DAYS



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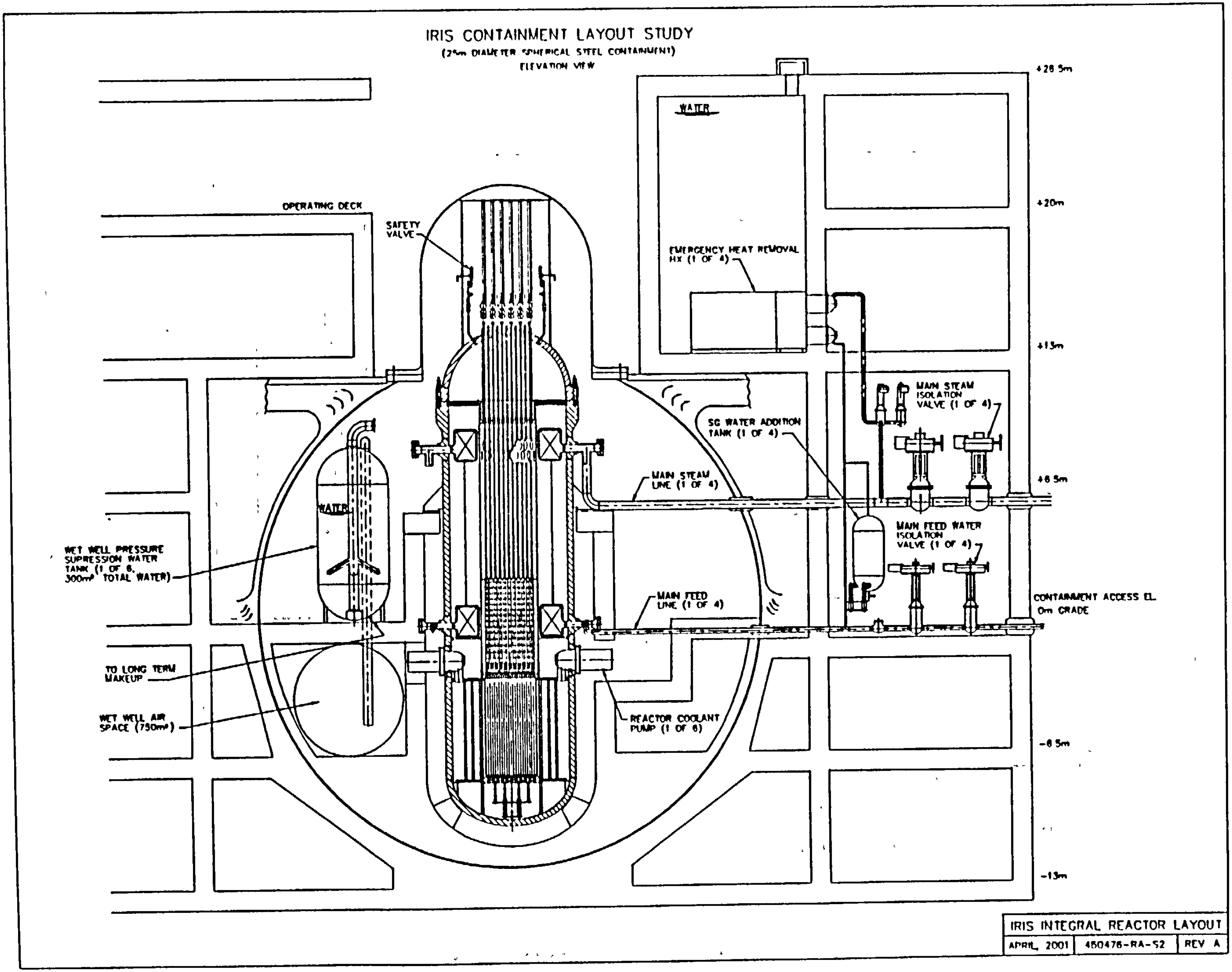
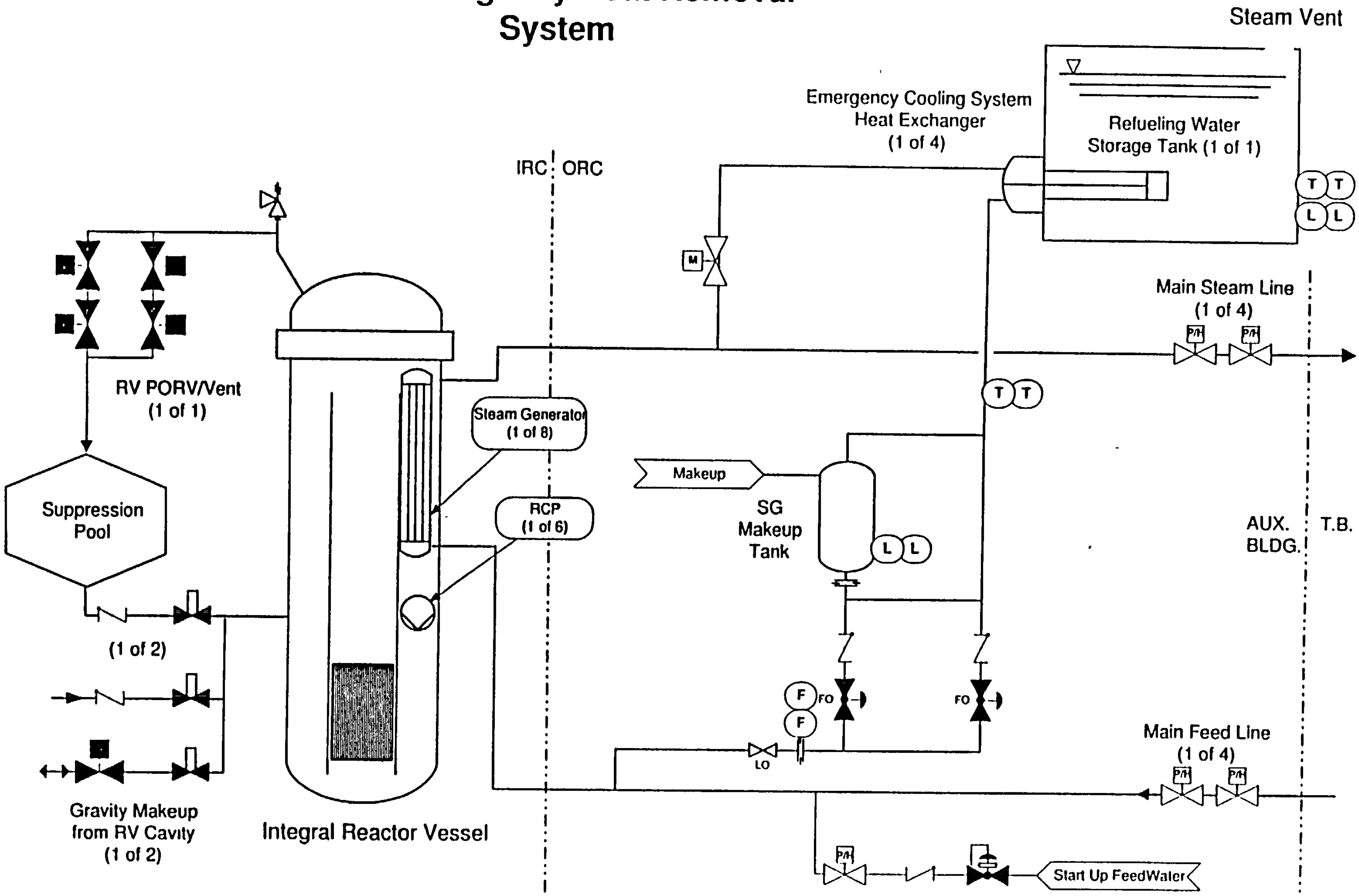

IRIS CONTAINMENT LAYOUT STUDY
(25m DIAMETER SPHERICAL STEEL CONTAINMENT)
ELEVATION VIEW
WATER
OPERATING DECK
SAFETY VALVE
EMERGENCY HEAT REMOVAL HX (1 OF 4)
MAIN STEAM ISOLATION VALVE (1 OF 4)
SG WATER ADDITION TANK (1 OF 4)
MAIN STEAM LINE (1 OF 4)
WATER
MAIN FEED WATER ISOLATION VALVE (1 OF 4)
WET WELL PRESSURE SUPRESSION WATER TANK (1 OF 6, 300m³ TOTAL WATER)
MAIN FEED LINE (1 OF 4)
CONTAINMENT ACCESS EL. 0m GRADE
TO LONG TERM MAKEUP
WET WELL AIR SPACE (750m³)
REACTOR COOLANT PUMP (1 OF 6)
+26.5m
+20m
+13m
+6.5m
-6.5m
-13m
IRIS INTEGRAL REACTOR LAYOUT
APRIL, 2001
460476-RA-52
REV A

IRIS Emergency Heat Removal System



IRIS - Safety by Design (LOFA)

- Condition IV loss-of-flow-accident (LOFA) is the sudden reduction in core flow caused by a pump shaft break or rotor seizure event resulting in DNB and fuel damage.
- These LOFA consequences are eliminated in IRIS
- Primary system flow path delta-P is very low
  - 60 ft. vs 250 to 350 ft. ΔP in loop type reactors
  - RCP's have flat head vs flow curve, and excess runout flow capability
- Multiple RCP's (6)
- Core flow maintained at 83% of full flow with n-1 RCPs
- With low power density core, no DNB, no core damage

IRIS - Safety by Design (SGTR)

- Current SGTR event causes radiation release, potential containment bypass, operator action to depressurize
- IRIS SG's, piping and isolation valves are designed for full RCS pressure - primary system cannot over-pressure
- SG tube rupture recovery greatly simplified
  - Steam and feed isolation valves of faulted SG automatically closed
  - SG fills, primary/secondary side pressure equalizes, terminating the leak
  - No operator actions required, other than normal shutdown and cooldown
- Adequate redundancy for continued heat removal assured by multiple SG's, steam/feed paths, normal and emergency heat removal

IRIS - Safety by Design (SLB & FLB)

- IRIS has high pressure SG's, piping, and isolation valves
  - No SG safety valve needed, thus no corresponding SLB
  - Increased margin to pipe rupture
- 8 modular, once-through SG's connected to 4 steam and feed piping connections
  - Once-through design contains very little water inventory, thus very little release to containment following SLB
  - 3 of 4 normal and emergency heat removal paths available
- IRIS design reduces both the probability and severity of credible and major steam and feed line breaks

IRIS - Safety by Design (Station Blackout)

- Relevant IRIS safety systems:
  - Reactor trip (same as other LWRs)
  - Decay heat removal (passive, following one time valve actuation)
  - Primary system water inventory and NC core cooling (safety by design)
  - Containment cooling (passive)
- Necessary actuations and monitoring are battery powered for extended time ($\geq$ 3 days)
- Canned RCP's have no seals (no consequential LOCA)
- Station blackout is not a core damage event for IRIS

Resolution of AP-600 Class IV Accidents in IRIS

	(Class 4) Accidents from AP600	IRIS Safety by Design	
1.	Steam system piping failure (major)	Reduced probability Reduced consequences	*
2.	Feedwater system pipe break		
3.	Reactor coolant pump shaft seizure or locked rotor	Reduced consequences No core damage occurs	*
4.	Reactor coolant pump shaft break		
5.	Spectrum of RCCA ejection accidents	Can be eliminated	
6.	Steam generator tube rupture	Reduced consequences	*
7.	Large LOCAs	Eliminated	
8.	Design basis fuel handling accidents	Reduced probability	

* Can be reclassified as Class III

INTERNAL CONTROL ROD SYSTEM

Conventional control rod drive mechanisms configuration is the current reference for IRIS first deployment. However a fully internal configuration has many advantages:

- Eliminates head penetrations
  - Simpler, more economical vessel design and fabrication
  - No stress corrosion cracking of penetration and seals (maintenance, replacement)
  - No Class IV rod ejection accident (safety by design)
- Eliminates long drivelines
  - Seismic concern alleviated
  - Cost reduction
  - Better utilization of internal space
- Simplifies containment

OPTIONS FOR INTERNAL CONTROL SYSTEM

A. Internal control rod drive mechanisms (CRDMs)

A1. Hydraulically driven motion

- In operation in NHR-5 Chinese reactor
- Design and tested for Argentina CAREM reactor
- Analyses and proof of principle tests performed by POLIMI

A2. Electromagnetically driven motion — don't know life & materials

- Design and extensive testing by MHI and JAERI for Japan MRX marine reactor

B. Liquid control rods

- Manometer type design patented by EdF
- In same positions as mechanical rods
- Fine sensitivity to power shaping

IRIS POSITION ON CONTROL SYSTEM

- Internal system is the logical solution for IRIS and integral configuration in general
- Hydraulically driven internal CRDMs are proven
- Materials behavior for electromagnetically driven internal CRDMs still a question for IRIS
- More investigation of liquid rods is necessary
- Development effort and schedule are critical for application to IRIS — Top priority for testing program
- Conventional system remains the reference until there is consensus that the internal system is mature and can replace it

MAINTENANCE OPTIMIZATION

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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Study based on MIT study of change from 18 month interval to 4 yr. interval between maintenance outages

Objective

Enable the target IRIS operating cycle length by eliminating maintenance-related barriers

Steps to achieving this objective:

- **Identify barriers in an existing PWR program**
- **Identify barriers due to IRIS design differences**
- **Focus the IRIS design effort to eliminate these identified barriers**
- **Develop techniques to eliminate emerging barriers**

The Bottom Line

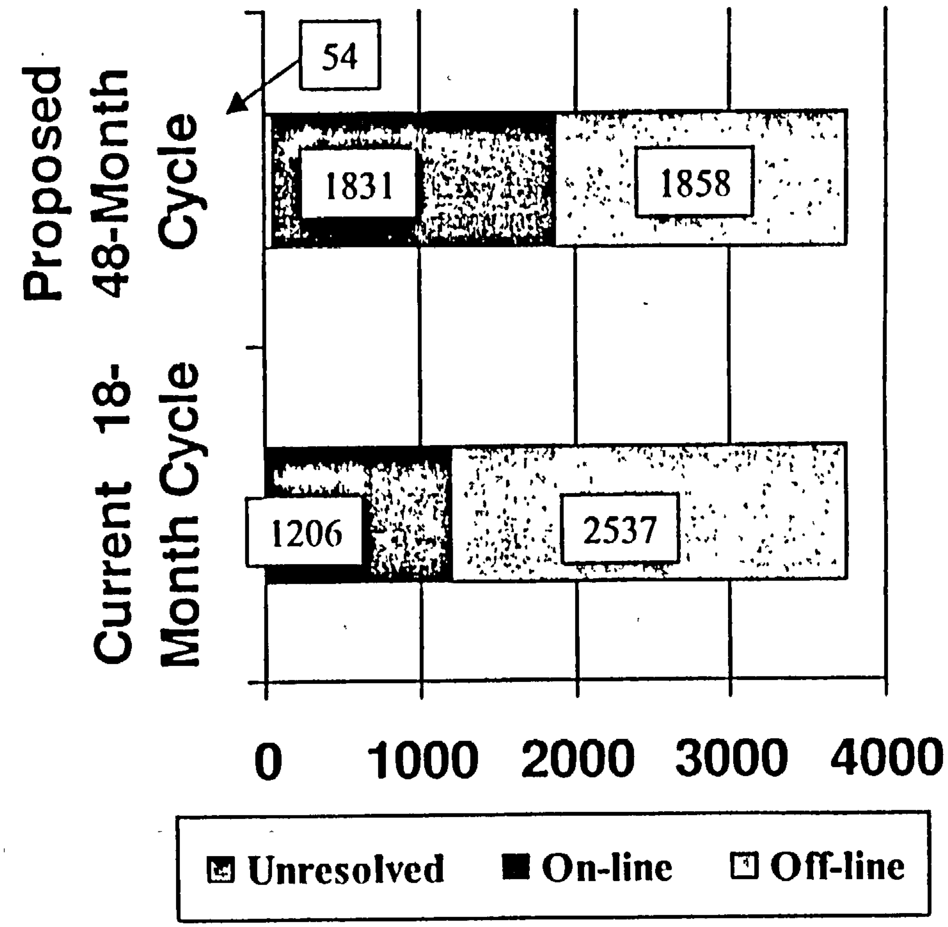
- IRIS must utilize components and systems which are either *accessible on-line* for maintenance or *do not require any off-line* maintenance for the duration of the operating cycle
- IRIS must utilize *high reliability* components and systems to minimize the probability of failure leading to unplanned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

IRIS design requirement = 4 yr. maintenance intervals

Extended Fuel Cycle Project

- MIT 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



– 54 items could not do anything with

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Designs in Service

Years of Relevant Data

- No impetus exists for currently operating plants to transition to very long cycles
  - Extensive backfit required
  - Plant cycles synchronized with demand
  - Finally good at outage management
- Possess significant amount of material history which can aid in justifying surveillance deferral

How can we unlock this data for evaluation?

Outline

- Strategy Overview
- Identifying Cycle Length Barriers
- Known Barriers
- The "Big Seven"
- Closing Comments

- Relief valve testing
- Steam generator inspection
- Main condenser cleaning
- Safety system testing
- Main turbine throttle control
- Rod control system testing
- Reduced power window items

7 items that are blocking IRIS for 4 yr intervals

MAINTENANCE ISSUES RESOLUTION

- Issues identified
- Tasks assigned
- Review progress in October 2001
- Major obstacles not expected

— Expect to achieve the 4 yrs maintenance cycle

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	✓	✓	✓

ISSUES

TECHNOLOGY GAPS

Identified technology gaps which need to be resolved.

For first-of-a-kind:

- Safety by design testing confirmation
  - Mockup of IRIS vessel/containment and associated safety systems. Possible facilities APEX (Oregon State Univ.), SPES (SIET, Italy), PANDA (PSI, Switzerland)
- Integral steam generator
  - Performance and reliability testing
  - Ansaldo has already tested a 20 MW helical steam generator
- Maintenance optimization
  - Address issues preventing four-year maintenance interval. Includes design, testing, instrumentation, procedures, regulatory
- Steam generator inspection procedure
  - Develop procedures, testing, regulatory

- safety by design
- Integral SGs
- Internal CRDs

} testing will be started very soon.

TECHNOLOGY GAPS (Cont'd.)

- System performance modeling
  - Select best analytical code capable of modeling IRIS, modify and run it

For first-of-a-kind/Nth-of-a-kind

- Internal control system
  - Assess alternatives, choose best and complete development

For Nth-of-a-kind

- High burnup fuel demonstration
  - Obtain data necessary for licensing
- Extended cycle operation
  - Qualify fuel and fuel assemblies for 8-10 years straight burn cycle
- Licensing of higher enrichment fuel
  - Assure regulatory approval

IMPORTANT REGULATORY ISSUES

- Establish review process to support goal of design certification by 2007
  - Periodic NRC/project interfacing
  - Initiate long lead testing – pacing item to licensing
- License a first-of-a-kind since IRIS is based on proven LWR technology (precedent: AP600)
- Successful resolution of technology gaps
- Assess IRIS design and operational characteristics versus current PWR regulations and requirements
  - Two major areas:
    - » Safety by design. Some accidents scenarios not applicable
    - » Extended maintenance. Evaluate compatibility with current regulations
- Licensing of higher enrichment fuel for subsequent IRIS modules
  - Higher enrichment fabrication facilities
  - Higher burnup fuels
- Multiple modules (shared control room) requesting DOE funding to determine optimum #
- How to translate into licensing IRIS improved safety "story". For example, can siting requirements (exclusion, low population zones) be relaxed?

SUMMARY AND CONCLUSIONS

- IRIS is part of BNFL/Westinghouse advanced reactors portfolio
- DOE and large international support
- 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
- Major design choices completed
- Continuing interaction with and feedback by NRC and ACRS will be extremely beneficial
- Testing program needs to start in 2002 on selected high priority tests

Testing and Research Needs

Testing and Research Needs: Safety by Design

- Safety by design is a key feature of the IRIS design which allows the elimination of traditional safety systems such as the ECCS. It is therefore necessary that correctly simulated tests be performed to corroborate analytical predictions of the IRIS response to a gamma of safety challenging events.
- A mockup of the IRIS vessel/containment and associated safety related systems will be built, utilizing existing facility such as APEX (Oregon State Univ., USA), SPES (SIET, Italy) or PANDA (PSI, Switzerland). Properly simulated tests will be performed to investigate the system response to specified transient and accident conditions.
- Testing specifications, directions and data evaluation will be performed by Westinghouse supported by most of the academic team members. Construction of the facility and performance of the testing will be performed by the selected facility operator which will be funded by DOE.
- Successful resolution of this need is critical to IRIS deployment. We judge this to be the most critical item

Testing and Research Needs: Integral Steam Generator

- The IRIS steam generator has several unique features: it is located inside the vessel; the primary flow is on the outside of the tubes; the currently preferred design is a helical tube bundle. It also has an expanded safety role in the vessel/containment thermal-hydraulic coupling
- An extensive testing program of a reasonably sized module would be required to demonstrate performance and reliability, and ability to perform required safety functions. It might be beneficial to perform an integral vessel-containment-steam generator-emergency heat removal system test. This will be performed in FY 05-07 in cooperation with the safety by design testing.
- Ansaldo, Italy, is currently performing the preliminary steam generator design. In the first three years the steam generator will be tested "per se". Later, interactive testing with the integral vessel/containment will be performed at the same facility as the safety by design and funding will be shared with DOE.
- The development of a steam generator suitable for integral vessel layout, exhibiting satisfactory performance and reliability and capable of performing required safety functions is one of few engineering development issues facing the IRIS design.

Testing and Research Needs: Maintenance Optimization

- A key distinguishing IRIS feature is the extended (at least four years) maintenance shutdown interval. The various issues which have been identified to prevent attainment of this goal need to be removed.
- Solutions might include some or all of the following: reassessment of maintenance needs in light of the IRIS design characteristic; adequate instrumentation and diagnostic; new designs of components to allow ease of inspection; regulatory rule abrogations or changes.
- Westinghouse has overall responsibility for its implementation, but primary responsibility for individual components will be of the partner with design responsibility for that component. All the other IRIS team members will provide support as appropriate. NRC will be requested to review the proposed regulation amendments. Laboratories will support enhanced instrumentation and diagnostic.
- Successful attainment of the 4 years' maintenance interval will significantly reduce O&M costs, allow attainment of high capacity factors and dramatically increase IRIS attractiveness to utilities.

Testing and Research Needs: Steam Generator Inspection

The IRIS SG tubes are in compression under external primary system pressure and function differently than the traditional SG tubes. The required inspections and procedures must be modified and tailored responding to the different functions and failure modes of the IRIS SG tubes to assure SG performance and to implement the IRIS extended maintenance approach.

<u>Sharing of Development Responsibilities:</u>

- Industry defines the design functions and failure modes of the SG tubes
- Industry proposes amendment to SG inspection requirements and procedures
- NRC amends the SRP requirements as necessary
- Laboratories support industry to develop and test inspection procedures for defined failure modes

Testing and Research Needs:System Performance Modeling

- The IRIS integral vessel/coupled small containment requires modeling of the system performance during normal and abnormal conditions as input to the design of the control system and the mitigation of transient initiators.
- Performance Modeling will determine the differences between the performance characteristics of a standard PWR and IRIS, and it will also provide a basis for licensing of the IRIS transient performance.
- Model the IRIS system response and the interaction of different subsystems based on first principles, system modeling and test data. Existing reactor performance simulation codes will be assessed and a reference code will be selected and modified as necessary to appropriately simulate IRIS conditions. The control system functional requirements will be established.
- IRIS consortium members will, with laboratories' assistance, evaluate existing codes and select the best candidate for IRIS simulation. They will subsequently model the system performance, predict the accident scenarios and verify predictions.

Testing and Research Needs: Internal Control Rod Drive Mechanisms

- The integral vessel configuration results in long drivelines which need to be engineered for seismic events. Also, the straight burn core requires more control rods than a conventional LWR. The integral configuration is ideal for locating the CRDMs inside the vessel. This is consistent with safety by design since the rod ejection accident is eliminated. Vessel head penetrations are eliminated resulting in simpler and cheaper design and elimination of stress corrosion cracking of seals and penetrations.
- Internal CRDMs can be electromagnetically or hydraulically actuated. Liquid control rods have also been proposed. A system will be selected, designed and tested. While it is advantageous to have this system ready for incorporation in the FOAK, it is possible that actual deployment will not occur until the subsequent IRIS modules.
- IRIS consortium will have primary responsibility for selection of CRDM system, design, and qualification. DOE supported laboratories and universities will provide testing and analyses. NRC evaluation and eventual approval of this novel system will be necessary.
- Successful demonstration of the internal CRDM system has many benefits in the areas of safety, economics, performance, and operation.

Testing and Research Needs: Extended Fuel Cycle

- IRIS fuel assemblies operating initially in a 4-5 year and subsequently in a 8-10 year fuel cycle must be qualified for operating for such a long time without interim inspection.
- No development is required for operating for 4 years under IRIS conditions. For reload core conditions the limiting performance parameters, primarily corrosion, must be predictable.
- Qualification testing is required of the fuel rod cladding, grids and assembly structures. Material testing and post-irradiation examination will confirm the adequacy of the materials, design and licensing data:
  - Westinghouse has the prime responsibility for data collecting, primarily Zirconium alloy corrosion, growth and hydriding data
  - Reactor operators (these could be the utilities who have joined the IRIS consortium) include low power fuel rods for extended low power operation in conventional PWRs
  - National Laboratories examine the data to support licensing
- Satisfaction of this need will provide the data necessary for designing and licensing fuel reloads up to 90 GWd/T-HM

Testing and Research Needs: Licensing of Higher Enrichment Fuel

- Reload cores of IRIS will use up to 9% enriched fuel. At this time no US facility is allowed to enrich the uranium at this level. Fuel vendors have no facilities licensed to package the fuel into assemblies and handle the fuel. To produce the higher enriched fuel and increase the core lifetime/burnup the present licensing barriers need to be raised:
  - Westinghouse and BNFL define fuel processing requirements, criticality limits and current licensing constraints
  - Amend the licensing requirements to produce and handle fuel with higher enrichment
  - Design or modify and license a production line to produce such fuel
- Requirements developed for handling and diluting weapons grade, high enriched uranium and MOX may provide a guide for changing the requirements.
- Providing the regulation and conditions for fabrication of fuel with higher enrichment will allow IRIS fuel reloads with a lifetime of 8 to 10 years. This results in reduced high level waste and improved economics.

Testing and Research Needs: High Burnup Fuel Demonstration

- The high burnup capability of IRIS type fuel must be demonstrated. Present fuel burnup is primarily limited by cladding corrosion, fuel rod internal pressure, enrichment in cores with a relatively high power density
- Some of the limitations of present fuel performance can be avoided in a low power density, highly moderated IRIS core. However, the high burnup achieved at a slow rate needs to be demonstrated in near prototypical tests to confirm the performance predictions and provide licensing data:
  - Industry will define and design the fuel to be tested
  - DOE and NRC will enable regulation to test 9% enriched fuel
  - Industry and DOE Laboratories will fabricate test rods for lead test assemblies
  - Reactor operations will enable testing of rods in reactors and prepare documentation for reactor licensing changes
  - IRIS consortium and DOE Laboratories will test rods/assemblies in reactors and evaluate results
- High burnup will allow the use of advanced reloads with reduced waste and economic advantages

INTERNATIONAL SYMPOSIUM ON THE ROLE OF NUCLEAR ENERGY IN A SUSTAINABLE ENVIRONMENT

The GenIV Nuclear Energy System Program: Expectations and Challenges

Professor Neil E. Todreas

The Center for Advanced Nuclear Energy Systems
Massachusetts Institute of Technology

April 20, 2001

MIT Concept Development Activity

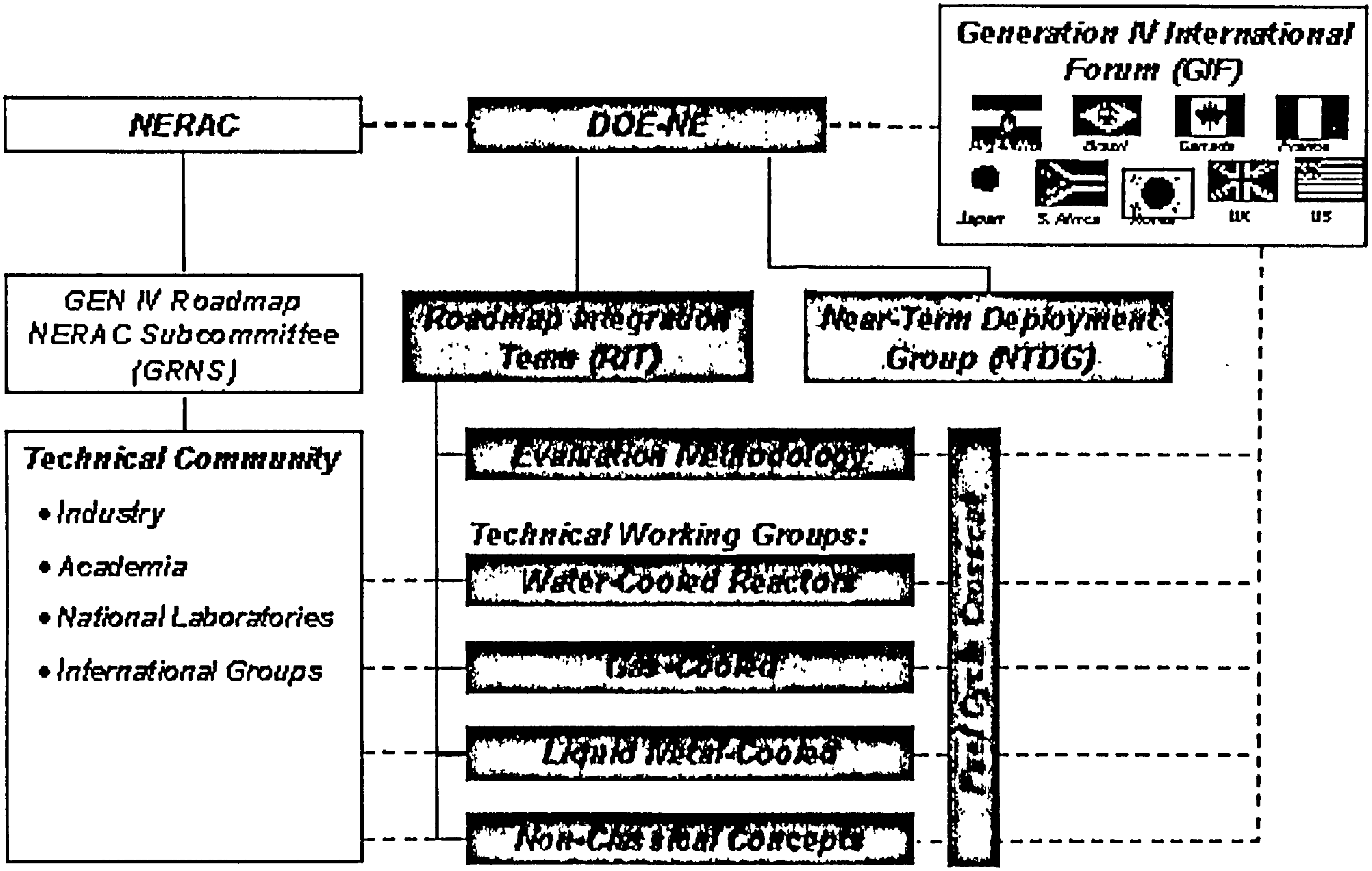
	EXTENSIVE ACTIVITY	PRELIMINARY ACTIVITY
WATER COOLANT	• CANDU LIGHT • IRIS - EPITHERMAL FUEL & CORE DESIGN, MAINTENANCE STRATEGY • THORIUM FUEL CYCLE - HOMOGENEOUS, SEED & BLANKET, MICRO HETEROGENEOUS CORES	• INTERNALLY COOLED ANNULAR FUEL PIN
GAS COOLANT	• MODULAR PEBBLE BED REACTOR (MPBR)	• MODULAR FAST GAS-COOLED REACTOR (MFGR-GT)
LIQUID METAL COOLANT	• LEAD-BISMUTH EUTECTIC COOLED REACTOR	• MINOR ACTINIDE BURNER REACTOR (MABR)

What is the GENIV Program?

Generation IV is a new generation of nuclear energy systems that can be made available to the market by 2030 or earlier, and that offer significant advances toward challenging Goals defined in the broad areas of sustainability, safety and reliability, and economics.

- Systems - **includes the entire fuel cycle.**
- 2030 or earlier - **how then linked to deployments in next 10 years?**
- Made available to market - **but not necessarily will be commercialized.**
- GOALS - **but first what is Roadmapping team structure?**

Overall GENIV Roadmap Organization



Center for Advanced Nuclear Systems

Principles from which Goals derive:

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

SUSTAINABILITY

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

Sustainability–1.

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

Sustainability–2.

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

Sustainability–3.

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

SAFETY AND RELIABILITY

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

Safety and Reliability –1.

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

Safety and Reliability–2.

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

Safety and Reliability–3.

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

ECONOMICS

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

Economics–1.

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

Economics–2.

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

What are the GENIV Challenges / Outcomes?

1) **Should Goals be Prioritized? No!**

- **Not all goals must be met by each system**

and
- **Goals must not be construed as regulatory requirements.**

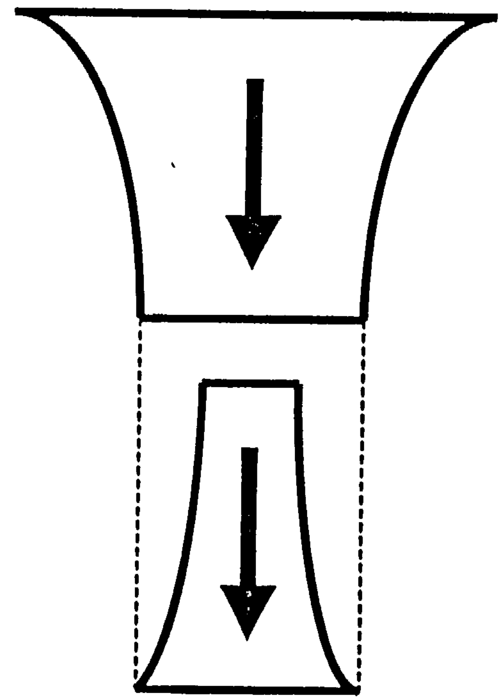
The desired outcome is:

A portfolio of systems each with likely different inherent characteristics best matched to the challenge of different goals.

What are the GENIV Challenges / Outcomes? (cont.)

2) How will Viable GENIV Systems Emerge?

Envisioned!



Reality?

What are the GENIV Challenges / Outcomes? (cont.)

3) How will the GENIV Process be Harmonized with Near-Term Deployment (NTD)?

Principle **- GENIV process should not impede commercial NTD aspirations.**

Fact of Life **- Systems at various stages of maturity will continually exist.**

Decision **- Systems unable to be deployed by 2010 will become GenIV systems and will be evaluated against GenIV Goals.**

What is the Near Term Deployment Process?

A Near Term Deployable Technology must satisfy the following criteria:

- Credible plan for gaining regulatory acceptance
- Existence of industrial infrastructure
- Credible plan for commercialization
- Cost-sharing between industry and government
- Demonstration of economic competitiveness
- Reliance on existing fuel cycle industrial structure

What is the Near Term Deployment Process? (Continued)

U.S. Government participation envisioned to resolve institutional and technical gaps impeding the deployment of nuclear technology in the near term (by 2010):

For example
- Regulatory process
- Early site permitting
- Technical demonstrations / uncertainty resolution

What are the GENIV Challenges / Outputs? (cont.)

4) What are the Likely Outputs of the GENIV Program?

Unlikely - Commercial Designs?
NO - RATHER

Possible - Designs for Potential Commercialization

Almost Assured - International Joint R+D on coolant family generic needs & possibly concept specific needs.

Desirable - Convergent Positions on policy directions (fuel cycle, waste, nonproliferation) and concept characteristic objectives.

What are the GENIV Challenges / Outputs? (cont.)

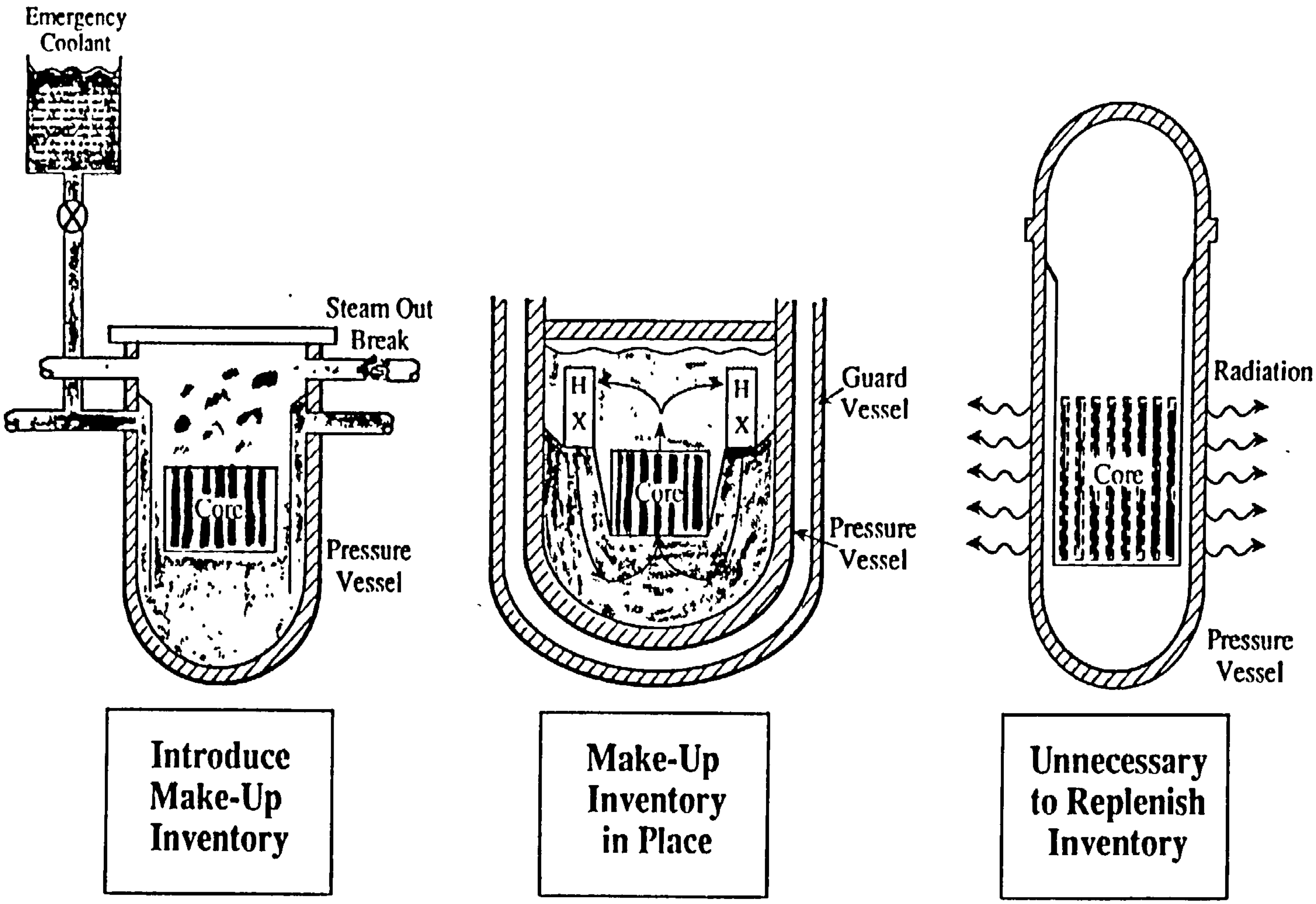
5) Is the GENIV Program Sustainable in the U.S.?

- Roadmap Completion - Sept, 2002
- Feasibility R + D - 5 Years
- Performance R + D - subsequent (with industry)

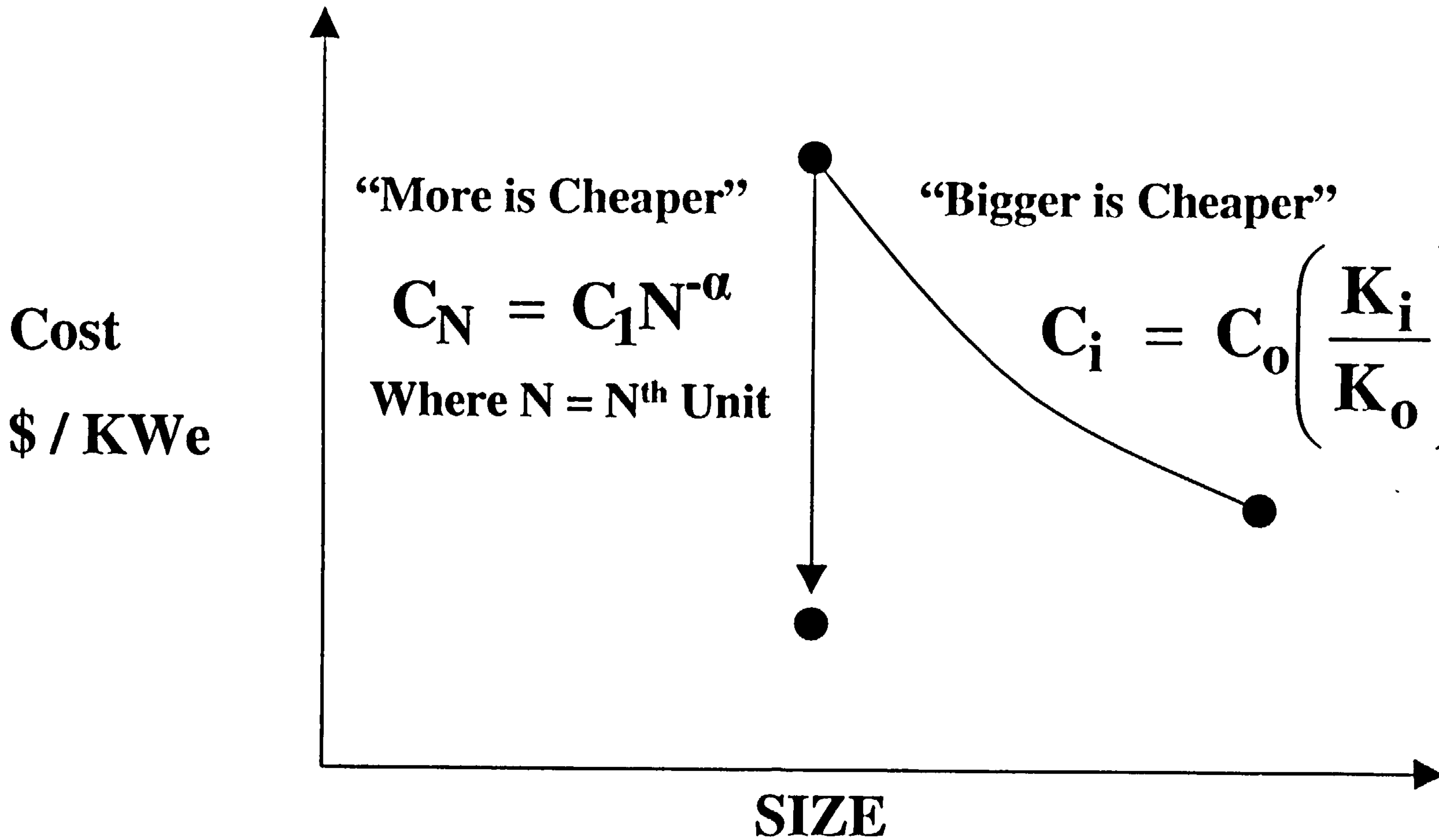
Selecting Among The Technologies

- Reactor Size
  - Safer
  - Cheaper
  - Simpler

CONCEPTS TO INSURE CORE COOLING



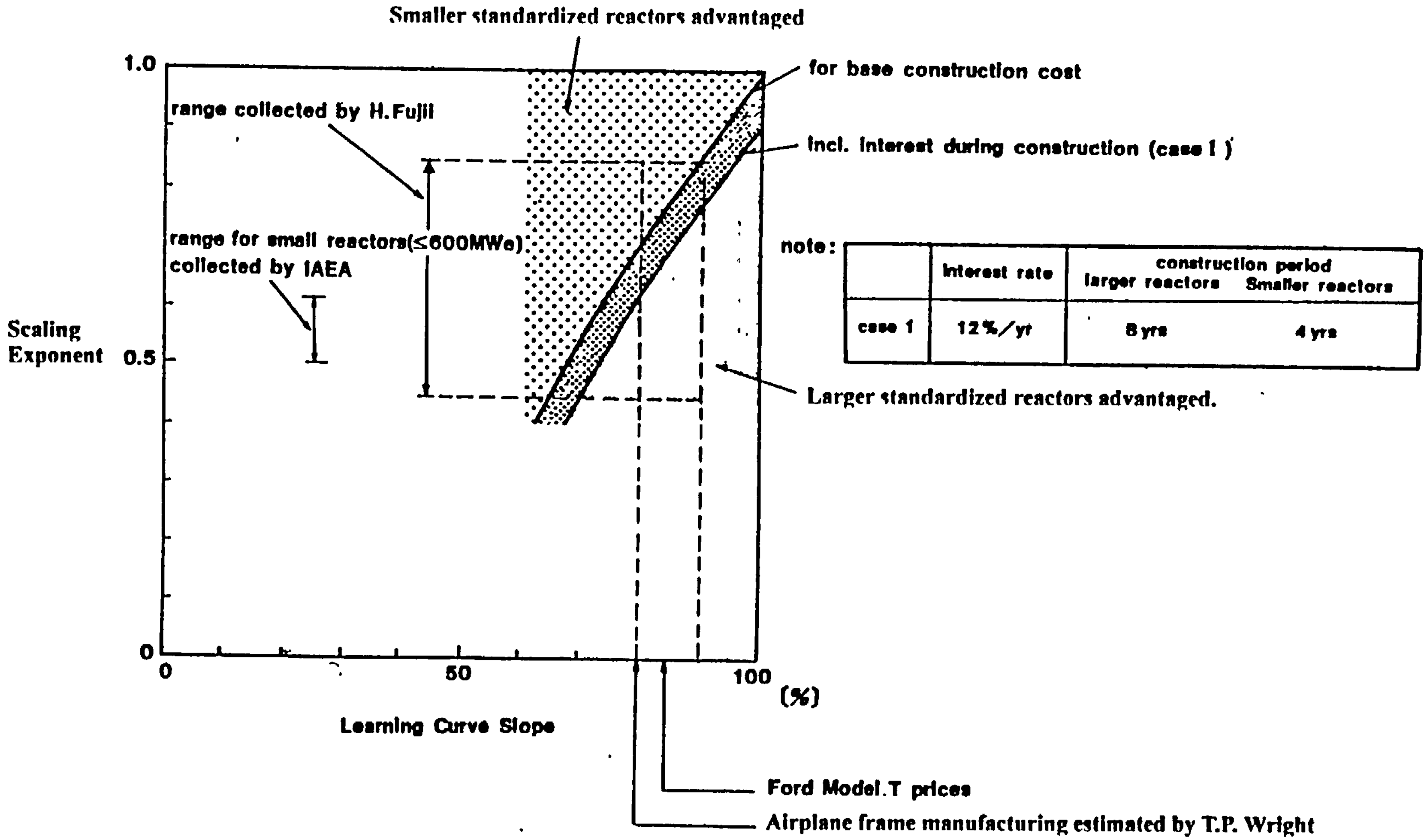
Learning Curve and Economy of Scale



Where n = scale exponent

and $\alpha = -\dfrac{\ln \frac{f}{100}}{\ln 2}$ (progress ratio)

Condition for Economic Advantage of Smaller Standardized Reactors Source (CRIEPI-E585001, 12/85)

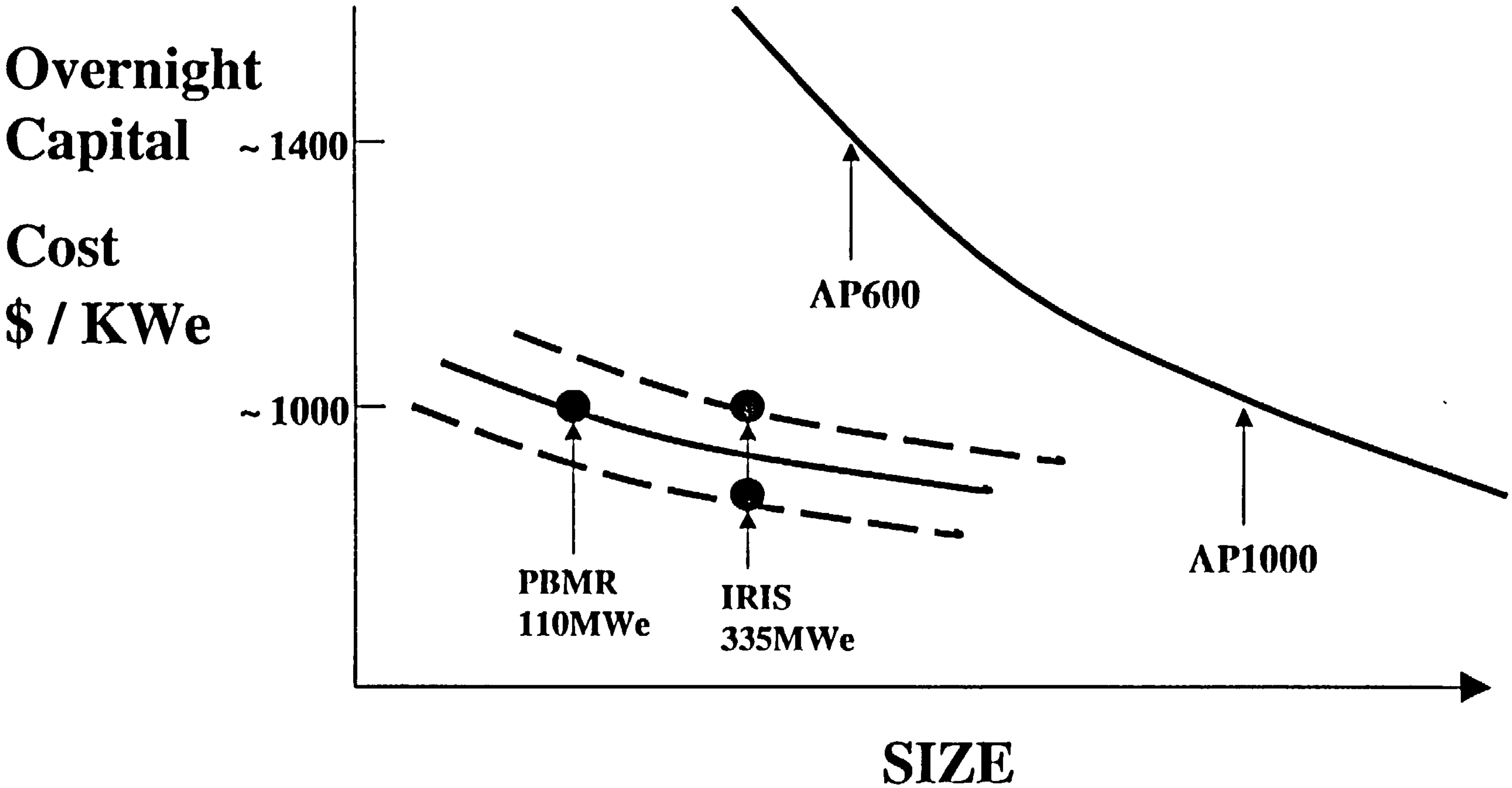


note:

	interest rate	construction period larger reactors	construction period Smaller reactors
case 1	12%/yr	8 yrs	4 yrs

BASE CASE: MARKET SIZE 20,480MWe
UNIT CAPACITY 80 to 1280MWe

Capital Cost Projections of Existing Concepts



Paper Reactors, Real Reactors

Characteristics of an Academic Plant

- It is simple.
- It is small.
- It is cheap.
- It is light.
- It can be built very quickly.
- It is very flexible in purpose.
- Very little development is required. It will use mostly off-the-shelf components.
- The reactor is in the study phase. It is not being built now.

Characteristics of a Practical Reactor Plant

- It is being built now.
- It is behind schedule.
- It is requiring an immense amount of development on apparently trivial items. Corrosion, in particular, is a problem.
- It is very expensive.
- It takes a long time to build because of the engineering development problems.
- It is large.
- It is heavy.
- It is complicated.

(H.G. Rickover, The Journal of Reactor Science & Engineering, June 1953)