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ADVANCED NUCLEAR REACTOR TYPES AND TECHNOLOGIES

Part I Reactors for Power Production

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Studsvik EcoSafe

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Abstract

The document is a comprehensive world-wide catalogue of concepts and designs of advanced fission reactor types and fuel cycle technologies. Two parts have been prepared:

Part I Reactors for Power Production
Part II Heating and Other Reactor Applications

Part III, which will cover advanced waste management technology, reprocessing and disposal for different nuclear fission options is planned for compilation during 1995.

This catalogue was prepared according to a special format which briefly presents the project title, technical approach, development status, application of the technology, reactor type, power output, and organization which developed these designs.

Parts I and II cover water cooled reactors, liquid metal fast reactors, gas-cooled reactors and molten salt reactors. Subcritical accelerator-driven systems are also considered. Various reactor applications as power production, heat generation, ship propulsion, space power sources and transmutation of such waste are included.

Each project is described within a few pages with the main features of an actual design using a table with main technical data and figure as well as references for additional information. Each chapter starts with an introduction which briefly describes main trends and approaches in this field. Explanations of terms and abbreviations are provided in a glossary.

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Preface

For several years Studsvik AB has had the special task, disignated by the Swedish Ministry of Industry and Commerce of surveying, analysing and reporting on the development efforts and achievements abroad, concerning advanced nuclear fission technology. Program management and most of the task is performed within the subsidiary company Studsvik Eco & Safety AB. The results are presented in an annual summary report as well as in topical reports in Swedish. Contacts and collaboration with colleagues abroad and other institutes and research organisations are of importance in this work. In recent years, close contacts have been established between the Russian Research Centre "Kurchatov Institute", in Moscow and Studsvik. As a joint project and in close co-operation it was decided to prepare a document which briefly presented the plans, projects and achievements within the area of advanced nuclear fission technology and their present status in various countries.

The first and second parts of this work concern nuclear fission reactors. The last will examine front- and back-end of advanced nuclear fuel cycles. The intention is to extend and also update the material compiled in new editions when necessary. The first two parts were compiled by Victor Ignatiev (Editor), Olga Feinberg and Alexei Morozov who are leading scientists of the RRC "Kurchatov Institute" and I would like to thank them all for their efforts which made this report possible.

Acknowle ge ents are due to Acad. N Ponomarev-Stepnoi and Prof. A C e n ski from RRC "Kurchatov Institute" who supported the work and ozitz, V Krett, C Goetzman from IAEA, Stanislav Subbotine from RC " Kurchatov Institute", Tor Pedersen of ABB Atom and Bengt Pershager for providing necessary information and useful suggestions.

I would also like to thank other contributors, among them Elisabet Appelgr for the typing and Monica Bowen-Schrire for correcting the English

I hope that the report will be of help to those who would like to obtain an overview of recent advancements in this area

Lennart Devell
Program Manager
Advanced Nuclear Technology

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Part II Heating and Other Reactor Applications

Introduction

Since the beginning of the nuclear fission era, illustrated e.g. by the start-up of the Fermi pile in 1942, increasing efforts have been made to develop and use improved nuclear fission reactors. The civilian application of nuclear fission is mainly power plants for electricity generation but, to a small extent, also for heat generation. There are reactors for research, radio-nuclide production, medical treatment, water desalination, ship propulsion and space applications. Also, some different systems in critical or subcritical accelerator driven mode have been studied for waste transmutation, weapon plutonium incineration, tritium production etc.

Many overviews and proceedings from recent conferences are available which describe the evolution of reactor design, technologies and applications in various countries (some of them are listed below). The present document is a comprehensive collection of short descriptions of new and advanced nuclear reactor types and technologies on a global scale (see Table 1). The purpose of the document is to provide an overview of present plans, projects, designs and concepts of future reactors. The material has been collected from the open literature and from information from the vendors.

The description of each design or concept follows a standardized format (see Table 2) to facilitate comparison. The current status of the technology is described for each design as well as type of application, reactor type, power output etc.

Compilation of data began in January 1994 and the present document has been issued as the first edition which is intended to be updated. The authors would therefore appreciate any comments, corrections and additions.

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Nürnberg, March 1990.

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Lyon, September 1990.

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The Hague 25 - 28 April, 1993.

o Global '93 - International Conference and Technology Exhibition on Future Nuclear Systems: Emerging Fuel Cycles and Waste Disposal Options.
September 12 - 17, 1993,, Seattle, Washington.

- 7th International Conference on Emerging Nuclear Energy Systems (ICENES '93).
  September 20 - 24, 1993, Makuhari, Chiba, Japan.
- Advanced nuclear power systems - design technology and strategies for their deployment.
  18 - 22 October 1993, Seoul, Rubublic of Korea.
- ARS '94 International Topical Meeting on Advanced Reactors Safety.
  17 - 21 April 1994, Pittsburg, USA.
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Table 1 Advanced Reactors

Abbre-viation	Name	Type	Size/ MW(e)	Country	Vendor Organization	Development status
Part I	**Reactors for Power Production**					
A	**Advanced Water Cooled Reactors**					
AE	**Evolutionary Designs for Large Power Plants**					
AEP1	N4	PWR	1400	France	Framatome	Commercial
AEP2	Sizewell B	PWR	1250	US/UK	Westinghouse/ Nuclear Electric	Commercial
AEP3	KWU-Convoy	PWR	1287	Germany	Siemens-KWU	Commercial
AEP4	APWR	PWR	1350	Japan/US	Mitsubishi//KEPCO/ Westinghouse	Detailed design
AEP5	System 80 Plus	PWR	1345	US	ABB-CE	Detailed design
AEP6	VVER-1000 (V-392)	PWR	1000	RF	Hydropress	Detailed design
AEP7	EPR	PWR	1500	Germany/ France	Siemens-KWU/ Framatome	Basic design
AEB1	ABWR	BWR	1356	Japan/US	Hitachi/Toshiba/GE	Commercial
AEB2	BWR 90	BWR	1379	Sweden	ABB Atom	Commercial
AEH1	CANDU 9	HWR	1050	Canada	AECL	Commercial
AT	**Evolutionary Designs for Medium Power Plants**					
ATP1	VVER-500/600	PWR	600	RF	Hydropress	Detailed design
ATP2	AP-600	PWR	630	US	Westinghouse	Detailed design
ATP3	MS-600	PWR	630	Japan	Mitsubishi	Basic design
ATP4	AC-600	PWR	600	China	CNNC	Basic design
ATB1	SBWR	BWR	600	US	General Electric	Detailed design
ATB2	HSBWR	BWR	600	Japan	Hitachi	Basic design
ATB3	TOSBWR-900P	BWR	310	Japan	Toshiba	Basic design

Table 1 Advanced reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(e)	Country	Vendor Organization	Development status
ATB4	1000 Natural Circulation BWR	BWR	1000	US	General Electric	Studies
ATB5	SWR 1000	BWR	1000	Germany	Siemens-KWU	Basic design
ATG1	MKER-800	LWGR	800	RF	RDIPE	Studies
ATH1	CANDU 3	HWR	450	Canada	AECL	Detailed design
AI	**Innovative Designs**					
AIP1	PIUS-600	PWR	600	Sweden	ABB Atom	Basic design
AIP2	ISER	PWR	200	Japan	JAERI	Studies
AIP3	ISIS	PWR	200	Italy	Ansaldo Spa	Studies
AIP4	MAP	PWR	300	US	Combustion Engineering	Basic design
AIP5	VPBER-600	PWR	600	RF	OKBM	Basic design
AIP6	SIR	PWR	320	US/UK	ABB-CE /Rolls Royce	Basic design
AIP7	SPWR	PWR	600	Japan	JAERI	Studies
AIP8	B-500 SKDI	PWR	515	RF	RRC-KI/Hydropress	Studies
AIP9	SCLWR	PWR	1100	Japan	University of Tokyo	Studies
B	**Liquid Metal Fast Reactors**					
BT	**Evolutionary Designs**					
BTS1	EFR	LMR	1500	France/GB/ Germany/Italy	EdF/NE/ Bayernwerk/ENEL	Basic design
BTS2	BN-600M	LMR	600	RF	OKBM	Detailed design
BI	**Innovative Designs**					
BIS3	DEMO FBR	LMR	660	Japan	FEPS	Basic design
BIS4	SAFR	LMR	450	US	Rockwell Int/CE	Basic design
BIS5	ALMR (PRISM)	LMR	150	US	General Electric	Basic design

Table 1 Advanced reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(e)	Country	Vendor Organization	Development status
BIL1	BREST 300	LMR	300	RF	RDIPE	Studies
BIL2	LFBR	LMR	625	Japan	JAERI	Studies
C	**Gas Cooled Reactors/Innovative designs**					
CI1	HTR-Modul	HTGR	80	Germany	Siemens-KWU	Basic design
CI2	MHTGR	HTGR	190	US	General Atomics	Basic design
CI3	GT-MHR	HTGR	300	US	General Atomics	Studies
CI4	HTGR-MHD	HTGR	260	Japan	JAERI	Studies
D	**Molten Salt Reactors/Innovative designs**					
DI1	USR	MSR	625	US	ORNL	Studies
DI2	FUJI-Pu	MSR	100	Japan	Tokai University	Studies
DI3	MSR-NC	MSR	470	RF	RRC-KI	Studies

Table 1 Advanced Reactors (cont'd)

Part II Heating and Other Reactor Applications						
Abbre-viation	Name	Type	Size/ MW(t)	Country	Vendor Organization	Development status
E	District Heating Reactors					
E1	AST-500 M	PWR	500	RF	OKBM	Commercial
E2	MARS	PWR	600	Italy	Rome University	Studies
E3	SECURE-H	PWR	400	Sweden	ASEA	Studies
E4	NHP-200	PWR	200	Germany	Siemens-KWU	Detailed design
E5	Thermos-100	PWR	100	France	CEA	Basic design
E6	NHR-200	PWR	200	China	INET	Detailed design
E7	RUTA	LWR	20	RF	RDIPE	Studies
E8	KNDHR	LWR	10	Korea	KAERI	Studies
E9	Slowpoke (SES-10)	LWR	10	Canada	AECL	Commercial
E10	Geyser	PWR	10-50	Switzerland	PSI	Studies
E11	GHR-20	HTGR	20	Germany/Switz.	KWU	Basic design
F	Decentralized Nuclear Heating Power Reactors					
F1	CAREM-25	PWR	100	Argentina	INVAP	Studies
F2	PAES-100	PWR	160-170	RF	OKBM	Commercial
F3	TRIGA Power System	PWR	64	US	General Atomic	Commercial
F4	ABV	PWR	60	RF	OKBM	Detailed design
F5	ELENA	PWR	3	RF	RRC-KI	Basic design
F6	Compact HTGR Gas Turbine	HTGR	29	US	General Atomic	Studies
G	Ship Reactors					
G1	OK-900A	PWR	170	RF	OKBM	Commercial
G2	KLT-40	PWR	170	RF	OKBM	Commercial

Table 1 Advanced Reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(t)	Country	Vendor Organization	Development status
G3	MRX	PWR	100	Japan	JAERI	Basic design
G4	DRX	PWR	0.75	Japan	JAERI	Basic design
I	**Space Nuclear Reactors and Isotope Batteries**					
I1	GPHS-RTG	Radioisotope Thermo-electric	$3 \cdot 10^{-4}$ MWe	US	Westinghouse	Commercial
I2	ROMASHKA	Fast, Thermo-electric	$3 \cdot 10^{-4}$-10^{-2} MWe	RF	RRC-KI	Detailed design
I3	TOPAZ-II	LMR, Thermionic	$6 \cdot 10^{-3}$ MWe	RF	RRC-KI	Commerical
I4	STAR-C	Thermionic	$5 \cdot 10^{-3}$-$4 \cdot 10^{-2}$ MWe	US	Rockwell Institute	Basic design
I5	SP-100	Fast, Thermionic	0.1 MWe	US	General Electric	Basic design
I6	ERATO	LMR/HTR	$2 \cdot 10^{-2}$ MWe	France	CNES/CEA	Studies
I7	LMCPBR	LMR/HTR	10	Japan	JAERI, Toshiba	Studies
I8	Rover/NERVA	HTGR	367 - 1 566	US	LANL, Westinghouse	Basic design
I9	NPPS	HTGR	1 200	RF	RRC-KI	Studies
J	**Subcritical Accelerator Driven Waste Transmutation and Energy Generation Systems**					
J1	Energy Amplifier	LWR/HTGR	200	Switzerland	CERN	Studies
J2	PHOENIX	LMR	3 600	US	BNL	Studies
J3	JAERI-TPC	MSR/LMR	820	Japan	JAERI	Studies
J4	ATW/ABC	MSR/LMR	3 000	US	LANL	Studies
J5	AMSB	MSR	1 000-2 100	Japan	Tokay University	Studies
J6	BBR	MSR	5 000	France	CEA	Studies

Table 2 Advanced Nuclear Design Description Format

Format	Description	
Technical approach	E	Evolutionary design for large power plants
	T	Evolutionary design for medium power plants
	I	Innovative design
Title	Title of project	
Applicaton of the reactor	NPP	Nuclear power plant
	NHP	Nuclear heating plant
	NHPP	Nuclear heat and power plant
	SMS	Ship mobile system
	SNS	Space nuclear source
	TWS	Transmutation waste system
Reactor type	PWR, BWR, HWR, LWGR, FBR, HTGR, MSR, SADS (subcritical accelerator-driven system) e.a.	
Power output	L	Large power (more than 1 000 MWe)
	M	Medium power (more than 100 MWe, less than 1 000 MWe)
	S	Small power (more than 1 MWe, less than 100 MWe)
	VS	Very small power (less than 1 MWe)
Organization	Type of organization, which developed design	
Development status	C	Commercially available design
	D	Detailed engineering design
	B	Basic engineering design
	S	Studies of concept
Description	General description of the design. This will generally include a simple description of the reactor plant. This may be followed by a summary with the main features of an actual design using tables with main technical data or figures	
Schedule	Schedule for development	
References	References for additional information.	

Part I Reactors for power production

Introduction

In many countries, nuclear power has demonstrated its capacity to generate electricity on a large scale, cost-efficiently and with low environmental impact.

Of the current 430 operating NPP units, almost 350 are LWR ones. The operating experience of these plants shows that they represent an economic and reliable source of electricity production. They also represent a broadly developed and mature basis of technology for further nuclear power development. The objectives for improvements cover a broad range of interest such as risk minimization, lower energy cost production, improved reliability and enhanced safety.

Progress and schedule for the development and deployment of ALWR

After the severe accident at Three Mile Island Unit 2 authorities in several countries requested measures for enhanced safety at operating plants. Among these measures, filtered venting of the containment was installed in many plants in order to mitigate consequences in the event of some specific severe accident scenarios.

The development of the ALWR Program started in the USA in the beginning of the 80's on the initiative of EPRI supported by utilities, DOE, US reactor vendors and architect engineering firms.

This program provided a foundation for the Nuclear Power Oversight Committee (NPOC) comprehensive initiative for revitalizing nuclear power in the USA, as set forth in its "Strategic Plan for Building New Nuclear Power Plants", published in November 1990 and updated annually each November.

Work on the Utility Requirement Document, URD, started in 1985 and was completed in 1990. The cooperative efforts, First-of-a-Kind Engineering, (FOAKE) between the US utilities, DOE and the vendors started in 1992.

In the USA the licensing process was changed in 1994. An important new feature of this process is the issuing of a final design approval (FDA).

The time schedule for safety evaluation, FDA and FOAKE is given in the following table, reproduced from an IAEA document. In fact, the FDA for ABWR and System 80 Plus was issued in mid 1994 not far behind schedule.

	ABWR	System 80+	AP-600	SBWR
Final Safety Evaluation Report	5/94	6/94	~4/96	~3/96
Final Design Approval	6/94	8/94	~6/96	~5/96
First-of-a-Kind Engineering	9/96	-	9/96	-

After the severe accident at Chernobyl Unit 4, the IAEA requested its International Nuclear Safety Advisory Group INSAG to give recommendations for action to enhance reactor safety. In 1988 the IAEA issued the INSAG-3 report with the title of Basic Safety Principles for Nuclear Power Plants. This was followed by further reports:

o	INSAG-4	Safety Culture (1991)
o	INSAG-5	The Safety of Nuclear Power (1992)
o	IAEA Safety Fundamentals	The Safety of Nuclear Installations (1993)

Already in the INSAG-3 report from 1988, a distinction was made between operating plants and future plants. For future plants, the probability of severe core damage should be less than 1 for every 100 000 reactor years and the probability of large releases to the environment should be less than 1 for every 1 million reactor years.

A similar development effort for a European Pressurized Water Reactor (EPR) was launched in Europe between France and Germany in early 1992 followed by the start in 1992 of producing a European Utility Requirement Document, (EUR). The first two volumes of the EUR were published in February 1994 and the main part of the remaining document is scheduled for completion by the end of 1994.

The advanced nuclear power plant designs currently being developed represent a wide range of alternatives. Most of the reactor concepts are evolutionary and represent small or moderate extensions of current designs. Some of them use upgrades of operating plants. Few concepts based on well-known LWR, HWR, LMFBR technologies have significant departures from the current ones. There are also some new designs that use innovative principles, which are unique compared to current practice, e.g. the PIUS principle for LWR, modular high temperature gas-cooled reactors (HTR), liquid metal reactors (LMR) and molten salt reactors (MSR).

The typical designs belonging to water-cooled, LMR, HTR and MSR groups are presented in Table 1 (see p 4).

In this catalogue, reactor design improvement follows three directions (see Figure 1, p 14) described below.

Evolutionary designs for large power plants

A design approach involving "step by step" modifications of present-day designs, with an emphasis on design proveness. Here, more traditional designers find smaller ways to modernize design to meet more stringent new standards. These (mainly large power - more than 1 000 MWe) provide the improvement of NPP reliability and operating safety with the simultaneous improvement of economic features due to further improvement and simplification, to some extent of the design. These designs, which incorporated extra safety features, require only small additional testing.

Evolutionary designs for medium power plants

A design approach which allows for the improvement of the NPP reliability and safety with simultaneous improvement of the economy and competitiveness due to simplification of the NPP design and reduction of specific core power and the unit power of a reactor (mainly about 600 MWe), the construction period, the degree of financial risk and expected broadening of the market in the developing countries. This approach refers to the evolutionary plant with significant innovative features, which are, however, within the range of existing technical knowledge.

Innovative designs

Some of the new ALWR designs ensure the improvement of LWR-based NPP and NHP reliability and safe operation mainly due to changes in the design arrangement and performances or in plant responses to the accidents. The major development goals stated for some of these plant designs are simplification and an enhanced use of passive safety features, based on current plant design technology. Others, mainly non-ALWR reactor types, have more ambitious development goals that, in turn, require more radical modifications of design aiming at the maximum possible utilization of inherent safety features and self-protection capacity.

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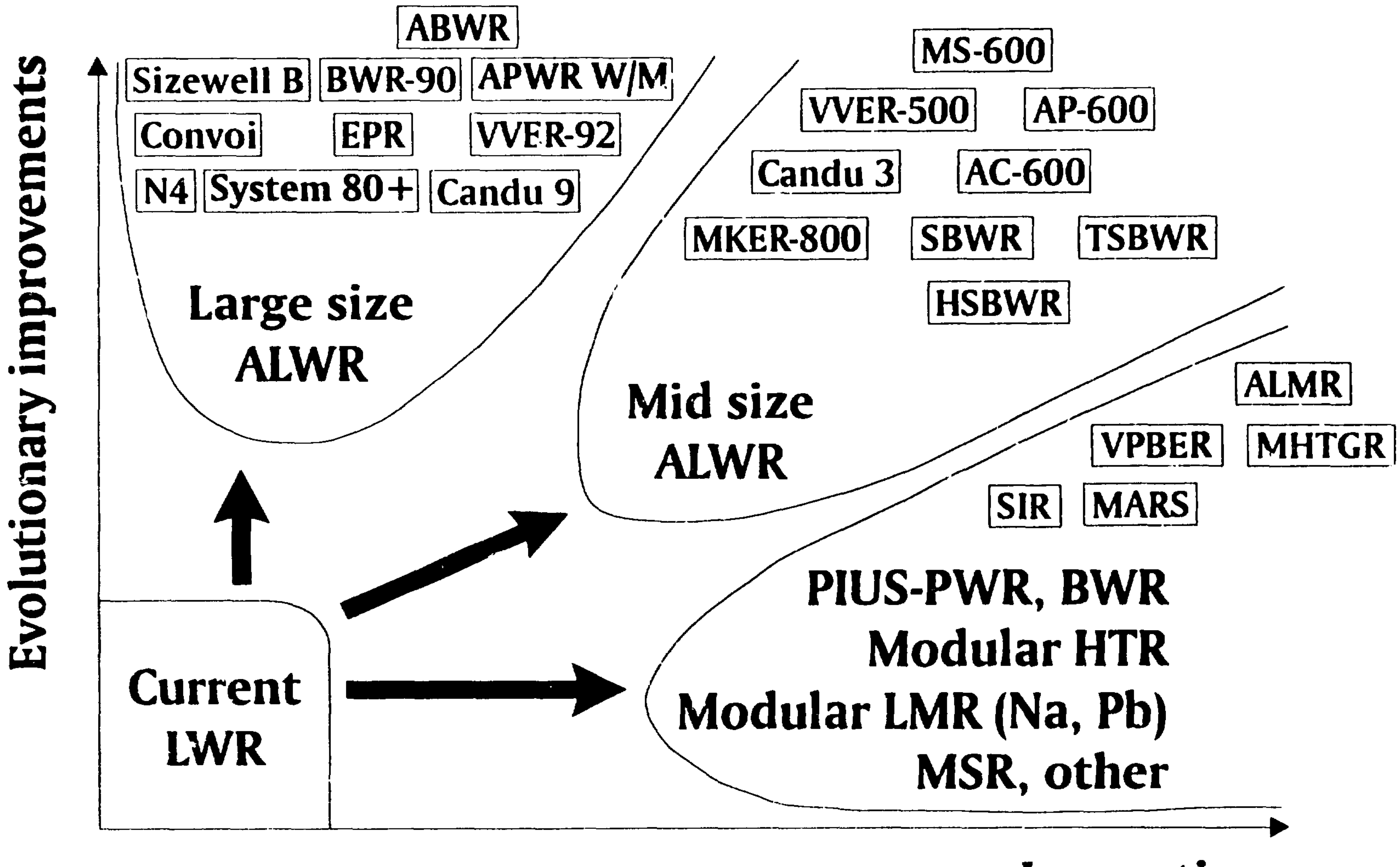

Figure 1
Different approaches to improvements of advanced fission reactor designs.

A Advanced Water-Cooled Reactors

AE Evolutionary designs for large power plants

AE0 Introduction

The designs of new light water reactors have to meet criteria concerning improved safety, economic competitiveness and user friendliness. Criteria have been issued by utility groups and authorities, which have been mentioned earlier.

The evolutionary approach has the benefit of providing the following:

- o Predictable performance based on experience
- o Enhanced safety, performance and economics
- o Incremental changes in technology
- o Availability of proven equipment and systems
- o Avoiding need to additonal experimental verification
- o Assured acceptance and licensibility

In comparison with the first stage LWR generation evolutionary ALWR designs underwent a considerable change. The change involved, basically, the enhancement of a power of units, thermodynamic parameters of a steam-turbine cycle, fuel utilization, operational reliability and safety.

The safety of evolutionary power plant designs has been improved on a realistic and efficient basis without jeopardizing economics.

In summary, the following is a list of the most significant safety improvements applicable to one or more reactor types:

- o Reduction of the core power density
- o Increase in the number of control drives
- o Application of spectral controls to compensate the variation of reactivity with burnup
- o Increase in the cooling water inventory capable of being stored within the containment and primary circuit space
- o Application of the emergency core cooling system (ECCS) to each loop
- o Improvement of NPP protection in case of a loss of off-site power

- Improvement of off- and in-reactor monitoring for predicting the reactor power density and temperature distributions on a real time basis

- Reduction of the neutron fluence on the reactor vessel

- Mitigation measures for severe core damage accidents with failure of the reactor pressure vessel (RPV)

- Filtered containment venting

- Application of a double containment protected from external effects by strong reinforced concrete structure

- Replacement of block-type control boards with multifunctional displays

- Division of the personnel's function among the operators responsible for maintaining rated operating conditions, engineers responsible for NPP control under emergency conditions, operators responsible for maintaining reactor availability during outage, refuelling, repair

- Increased use of periodical simulator training of operator, maintenance personnel etc

- Development of systems capable of checking the main equipment operability during reactor operation

This design approach obviously implies a step-wise evolution of the plant design on the basis of the utility requirements and stated preferences. Large power ALWR designs are now under consideration in France, Germany, UK, USA, Japan, Russia, Sweden and some other countries. Canadian HWR lines are also presented.

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AEP1

Format	**Description**
Title	N4
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1400 MWe
Organization (name)	Electricité de France EdF/Framatome, France
Development status	Commercial

Description

The French N4 plant is a 1400 MWe PWR with 4 train systems organization. The N4 is fuelled by enriched UO_2 pellets and uses Zircalloy-4 as a cladding material. The active core height is 4.2 m and it contains 205 fuel assemblies with a 17x17 array. As regards safety, it was designed in an evolutionary perspective, benefiting the experience gained from the previous standardized series of 900 MWe and 1300 MWe plants, and using technological progress and new developments after careful validation.

The safety improvements are the following:

- o More accurate and functionally adapted instrumentation
- o A better global methodological approach for fire hazards
- o Means to deal with total loss of frequently used systems, and with shut-down conditions
- o An increased role for PSA in the design and licensing process

Other important features of N4 plant are a computerized control room and mitigation of core melt in ultimate conditions, with filtered venting containment as last option.

Schedule

The N4 three units are under construction in France. The first N4 unit Chooz B-1 is to be commissioned in 1995.

N4 Unit Data

Thermal power		4 270 MW
Reactor Core		
-	Equivalent diameter	3.47 m
-	Number of control assemblies	73
-	Initial enrichment	1.8/2.4/3.1 % U-235
-	Enrichment at equilibrium	3.4 % U-235
-	Average fuel burn-up at equilibrium	39 MWd/kg
-	Total weight of UO_2	110.4 t
Reactor Coolant System		
-	Operating pressure	155 bars.
-	Reactor vessel inlet temperature	292.2 ^{o}C
-	Reactor vessel outlet temperature	329.6 ^{o}C
-	Number of reactor coolant pumps	4
Reactor Pressure Vessel		
-	Inside diameter	4.5 m
-	Total height	13.64 m
-	Design pressure	172 bar
-	Design temperature	343 ^{o}C
Containment		
-	Configuration	Double
-	Gross volume	72 000 m^3
-	Design pressure	5.3 bar
Steam Generators		
-	Number	4
-	Steam pressure at SG outlet	72.3 bar
-	Steam temperature at SG outlet	288 ^{o}C

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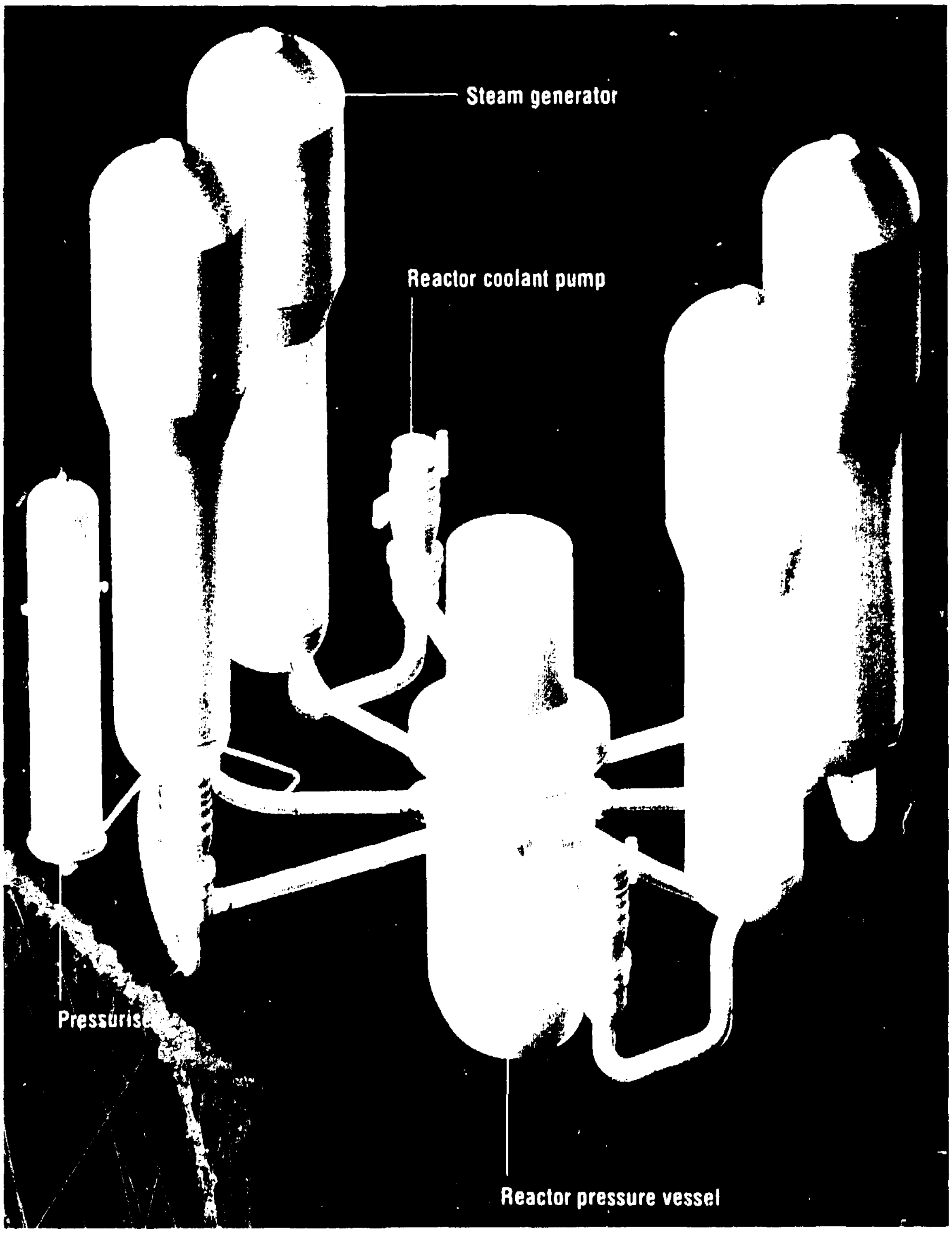


Figure AEP1
Reactor coolant system components

AEP2

Format	**Description**
Title	Sizewell B
Application of the reactor	NPP
Reactor type	PWR
Power output	Large Power/1250 MWe
Organization (name)	Westinghouse/Nuclear Electric, US/United Kingdom
Development status	Commercial

Description

The reactor design is developed on the basis of well-established PWR practices, derived from the Westinghouse SNUPPS (Standardised Nuclear Unit Power Plant System) design.

The fuel is low-enriched uranium oxide claded in Zircaloy 4. The core contains 193 fuel assemblies, consisting of 236 fuel rods each. Control rods forming rod cluster control assemblies can be inserted into the core in interstitial spaces within certain of the fuel assemblies and they are grouped into a number of "banks" for reactor control and for shutdown.

The core is cooled by water at a pressure of 15.5 MPa. The reactor coolant system comprises the reactor pressure vessel, four centrifugal reactor coolant pumps, four vertical steam generators with the primary cooling circuit passing through an inserted U-tube bundle, a pressurizer and linking pipework.

The important engineering safety features of the reactor plant are:

- 4-train system organization
- 30 minutes tolerance before operator action
- Fully computerized reactor protection system
- Station black-out addressed by two reliable grids and four emergency systems

The reactor design incorporates a considerable degree of redundancy and diversity in systems providing safety functions.

Schedule

The plant is owned and operated by Nuclear Electric, the electrical utility operator of nuclear power stations in England and Wales, with scheduled start-up in 1994. Nuclear Electric has formally applied for planning consent to build Sizewell C unit which would replicate the Sezewell B design.

Sizewell B Unit Data

Thermal capacity		3 411 MWt
Fuel		
-	Material	sintered UO_2
-	Enriched (feed fuel)	3.1 %
-	Fuel clad material	Zircaloy 4
Reactor core		
-	Active height	3.66 m
-	Equivalent diameter	3.37 m
-	Mass of UO_2 in the core	101 t
Reactor vessel		
-	Overall height	13.52 m
-	Inside diameter	4.394 m
-	Material	low alloy steel
-	Internal cladding	stainless steel
-	Dry weight	435 t
Steam generator		
-	Number	4
Turbo generator		
-	Number	2
Containment		
-	Configuration	Double
-	Gross volume	91 000 m^3
-	Designed pressure	3.5 bar

References

BOARD, L A and QUICK, M V
The Sizewell B PWR Design,
Review of Advanced Light Water Reactor Design Approaches, 10-13 May, 1994, RF, Moscow, IAEA-TC-879.

Nuclear Electric Ann^unces Sizewell C.
Atom, 1993, 431, November/December, p 10.

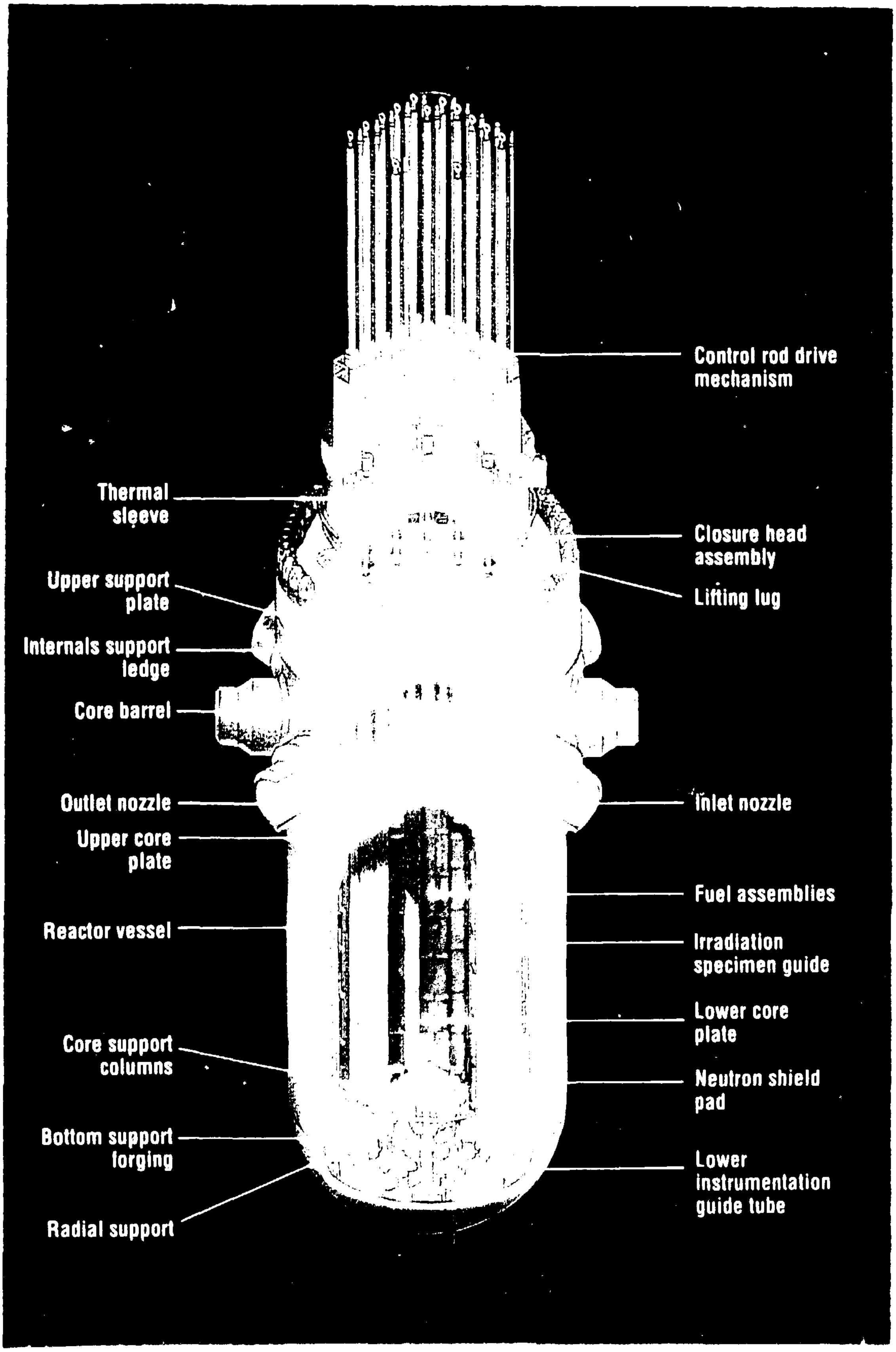


Figure AEP2
Sizewell B: Reactor Pressure Vessel

AEP3

Format	**Description**
Title	KWU - Convoy
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1287 MWe
Organization (name)	Siemens - KWU, Germany
Development status	Commercial

Description

The KWU-Convoy Nuclear Steam Supply System (NSSS) consists of a PWR with four primary coolant loops, a pressurizer connected to one of the loops, 4 steam generators, four reactor coolant pumps and the auxiliary and safety systems directly related to the NSSS. The NSSS generates approximately 3 782 MWt producing steam at 63.5 bar at the steam generator outlet. The turbine generator provides a net power of approximately 1 287 MWe. Full load rejection is accepted without a reactor or tubine trip. The turbine plant is completely automatic and supervised from the control room.

A short summary of main safety and operational features is given below.

The reactor core concept includes:

- An RPV with a wide gap between the core barrel and RPV shell which keeps the integral neutron dose to the RPV beltline at a low level (limitation of embrittlement).
- Low linear heat generation rate, resulting in high burnup capability and increased fuel management flexibility.
- The use of Zircaloy 4 guide thimbles and spacers increases neutron economy.
- Gadolinium poisoning of fuel rods improves fuel utilization.

The reactor building concept includes:

- The use of the spherical steel containment, designed for full pressure and temperature after a double-ended LOCA.
- A fuel pool inside containment, which facilitates handling and shortens refueling time.

o The use of a reinforced concrete containment protects the NPP against external events.

Schedule

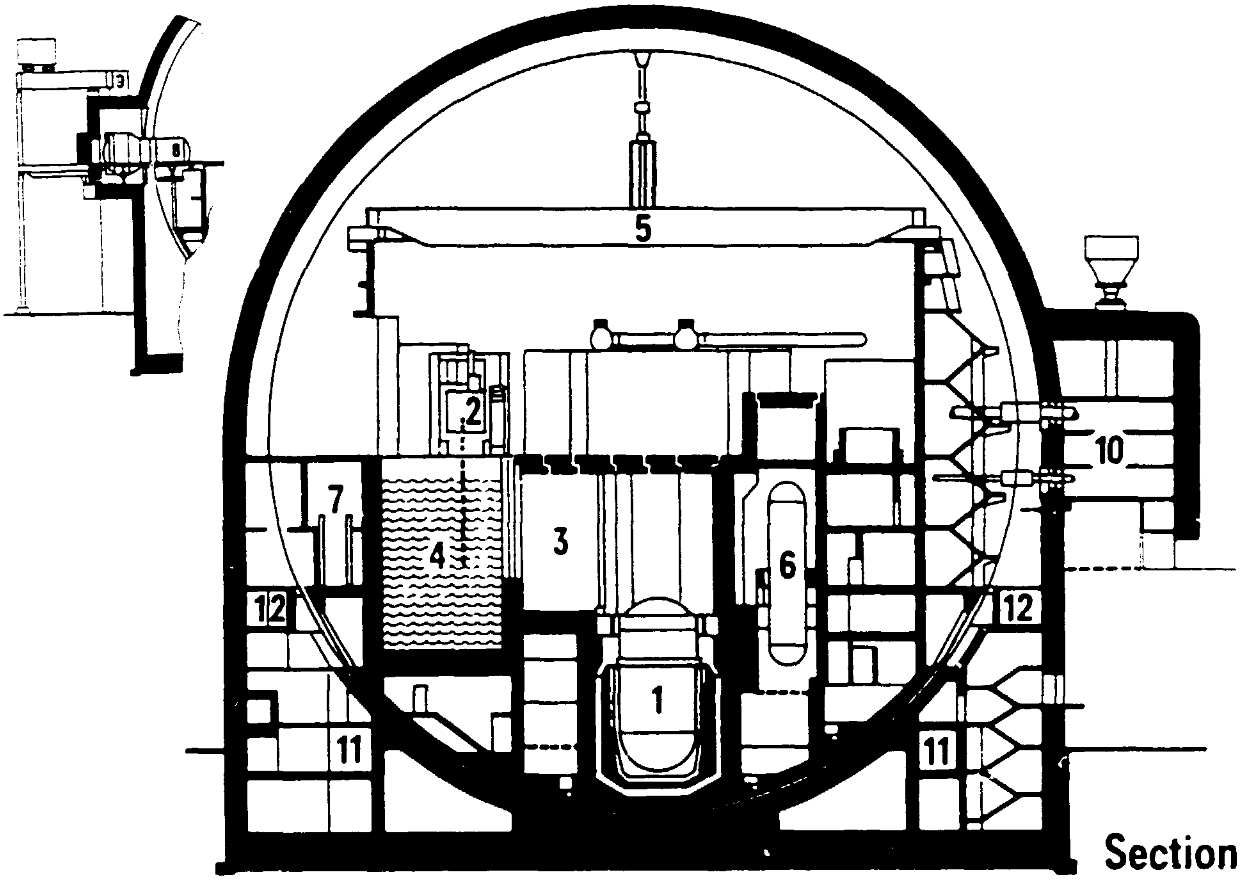
The last series of PWRs built in Germany is the standardized type of KWU-Convoy concept. However, there is always room for further improvements in the technical field.

KWU-Convoy Unit Data

Thermal power		3 782 MWt
Fuel assembly		
-	Array	18 x 18-24
-	Number of fuel rods	300
-	Number of guide tubes for absorber	24
Fuel Rod		
-	Length	4.4 m
-	Outside diameter	9.5 mm
-	Cladding material	Zircaloy 4
Fuel pellet		
-	Enrichment for the first core	1.9; 2.5; 3.2 %
-	For the reload core	3.2; 3.4 %
-	Average fuel burnup	31. 8 MWd/kg
-	Total weight of the UO_2	103 t
Reactor coolant system		
Design conditions		
-	Pressure	175 bar
-	Temperature	350 ^oC
Reactor coolant pump		
-	Type	Single-stage centrifugal
-	Number	4
Steam generator		
-	Type	U-tube, vertical
-	Number	4
Containment		
-	Configuration	Double
-	Diameter	56 m
-	Designed pressure	5.3 bar

References

CZECH, L and FE:GEL, A etc.
Technical Information on Design Features of Siemens Convoy PWR
Review of Advanced Light Water Reactor Design Approaches.
10-13 May, 1994, RF, Moscow, IAEA-TC-879.



1 Reactor pressure vessel
2 Refuelling machine
3 Lay down position for core internals
4 Fuel pool
5 Reactor building crane
6 Pressurizer
7 New fuel store
8 Equipment lock
9 Gantry
10 Main steam and feedwater valve room
11 Pipe duct
12 Cable duct

Figure AEP3
KWU-Convoy: containment

AEP4

Format	**Description**
Title	APWR (Advanced Pressurized Water Reactor)
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1350 MWe
Organization (name)	Mitsubishi/Kansai Electric Power Co (KEPCO), Japan; Westinghouse, US
Development status	Detailed design
Description	The APWR is based on the design, which incorporates the principles of simplification and additional operating margins, and operating experience of current PWRs. Basically, the APWR is a four-loop PWR. The thermal power of the reactor is 3823 MWt. The fuel rods are assembled in square arrays. Compared with Sizewell B and KWU-Convoy plants (see AEP2 and AEP3), the number of fuel assemblies is unchanged, but the core density is lowered by 20 % relative to conventional designs, by using larger size assemblies. Each fuel assembly consists of a 19 x 19 array of pins cladded by Zircaloy 4 (instead of a 17 x 17 traditional array). A fuel assembly contains 16 thimbles to house spectral control by introducing water-displacing rods. The displacers remain inserted into the fuel assembly during the first half of the fuel element lifetime. At the core level in the fuel assembly, a large number of Zircaloy 4 grids is used. To enhance the mechanical strength of the fuel assembly, the fuel element claddings and grids are thickened. To reduce neutron leakage, the core is enclosed by a steel neutron reflector. With 3 % fuel enrichment in U-235, the period of continuous power operation is 13.5 months. The 10 % cost reduction of the APWR fuel cycle results from above noted features. The major part of these improvements is due to a spectral control. With "grey" control rods replaced by conventional ones, the reactor can operate with a mixed UO_2-PuO_2 fuel.

Other typical features are:

- o The increased size of the RPV(all large nozzles are now well above the core top)
- o The containment with an emergency water storage tank at the bottom and the four-train arrangement of emergency cooling pumps (with "pairs" of diversified design)

The steam generator and turbine designs are improved. To reduce the radiation dose of the personnel, access is improved to the units which require to repair work. Low cobalt containing structural materials are then applied. The size of the hatches for steam generator examination and repair is increased.

Schedule

The designation APWR is not generally known in the US. Westinghouse has not used the US certification process. Some design activities have continued in Japan, but a decision on construction or siting has not been taken yet. Also, because utilities expressed a desire for an APWR at a nominal 1 000 MWe rating, Westinghouse developed the nominal 1 050 MWe three loop APWR.

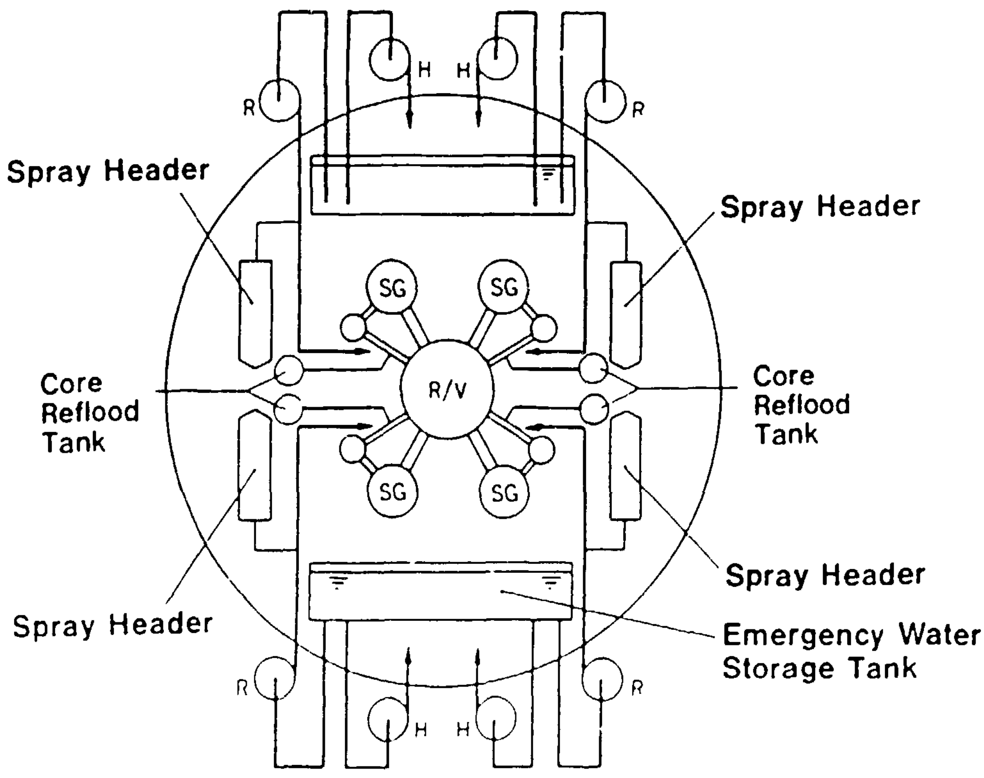
APWR Unit Data

Electrical output	1 350 MWe
Reactor core	
- Number of fuel assemblies	193
- Active height	3.9 m
- Equivalent diameter	3.98 m
- Total weight of UO_2	119.2 t
- Number of rod cluster control assemblies	69
- Average power density	80 kW/l
Reactor coolant system	
- Pressure at vessel outlet	157 bar
- Flow rate	88 000 m^3/h
Reactor vessel	
- Inside diameter	5 m
- Overall height	16 m
Reactor coolant pump	
- Speed	1 185 rpm
- Weight	92 t
Steam generator	
- Heat transfer surface	6 039 m^2
- Outlet steam pressure	69 bar
Containment	
- Configuration	Double
- Diameter	60 m

References

Technical Information on Design Features of the APWR.
Review of Advanced Light Water Reactor Design Approaches.
10-13 May 1994, Moscow, RF, IAEA-TC-879.

The Westinghouse APWR
Nuclear News, 1992, September, p 74-75.



H : High Head SI Pump (4 sets)
R : RHR/CV Spray Pump (4 sets)

Figure AEP4
APWR: Integrated safeguards system layout around containment vessel.

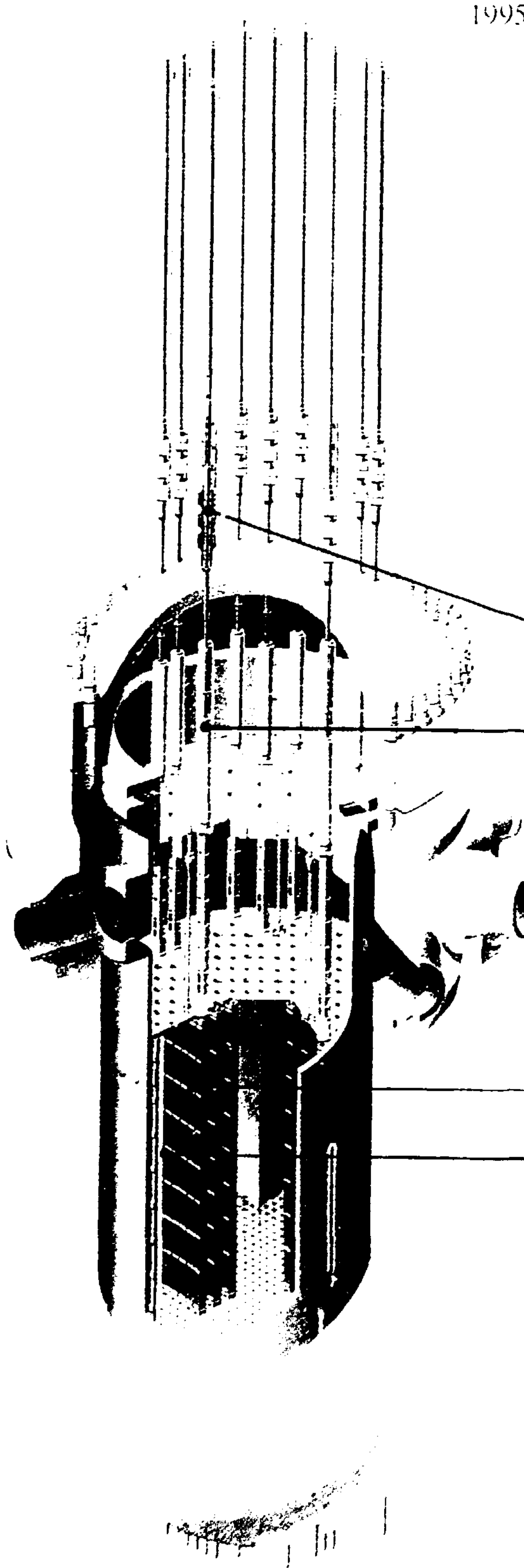


Figure AEP4
APWR: Reactor pressure vessel internals.

AEP5

Format	Description
Title	System 80 Plus
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1345 MWe
Organization (name)	ABB-CE Nuclear Power, US
Development status	Detailed design
Description	System 80 Plus is an advanced, evolutionary PWR plant and it represents enhancements of the System 80 standardized design. The System 80 plant design is used for Palo Verde units in Arizona, the USA. The enhancements, as integrated design process, address new licensing issues, safety improvements, improved operability, availability and maintainability, plant simplification and cost reductions.

The thermal power of the reactor is 3817 MWt. The reactor vessel is designed to contain and support the core and the fuel. The design of the core is based on that one of System 80. The reactor vessel is a vertically mounted cylindrical vessel with a hemispherical bottom attached to the vessel and a removable hemispherical upper closure head. The reactor vessel is fabricated from low alloy steel and the internal surfaces that are in contact with the reactor coolant are cladded with austenitic stainless steel. The reactor core consists of 241 fuel assemblies and 94 or more control elements. An advanced burnable absorber strategy is implemented to simplify reactivity control and provide for extended fuel burnup. Full-strength control element assemblies consist of a Inconel clad with boron carbide or silver-indium-cadmium absorber rods. Reduced strength control rods composed of solid Inconel provide the capability to change operating power level using control rods only. This simplifies reactivity control during plant load changes and reduces liquid waste processing requirements that normally accompany changes in soluble boron concentration.

Like previous ABB-CE reactors, the Reactor Coolant System (RCS) for System 80 Plus has two loop configurations. Each loop consists of one hot leg, one steam generator, two cold legs and two pumps.

A pressurizer with an increased volume (in specific volume, the increase amounts to 33 %) to reduce the pressure changes during transients such as reactor trip and load rejection. The increased secondary side water inventory (by 25 %) and heat transfer surface of the steam generator are also aimed to improved operating performance.

The most significant changes are in the engineering safety systems, in particular, in the Safety Depressurization System (SDS) and in the Safety Injection System (SIS). The SIS has been made simplier, more reliable and the performance has been improved. It incorporates four-train safety injection, an in-containment refuelling water storage tank, and direct vessel injection. The containment structure is a steel sphere enclosed in a cylindrical concrete building. Gas turbines as a diverse AC source are used.

A PRA indicated a reduction in the risk of severe accidents by two orders of magnitude compared to that with previous systems.

Schedule

No System 80 Plus plant has been ordered yet. The final design approval for System 80 Plus was issued in 1994. Also, construction of System 80 NPPs with some System 80 Plus design features is now underway in the Republic of Korea.

System 80 Plus Unit Design

Electrical output		1 345 MWe
Fuel assembly		
-	Array	16 x 16
-	Number of fuel rods	236
-	Clad material	Zircaloy 4
-	Enrichment levels	3.2, 2.8, 1.9 %
.	Fuel outside diameter	9.7 mm
Reactor core		
-	Number of fuel assemblies	241
-	Core height (active fuel)	3.81 m
-	Core diameter	3.65 m
Reactor coolant system		
-	Design pressure	170 bar
-	Design temperature	343 °C
-	Reactor inlet temperature	291 °C
-	Reactor outlet temperature	323 °C
-	Number of coolant pumps	4
-	Number of steam generators	2

Containment

-	Configuration	Double
-	Cross volume	95 930 m^3
-	Designed pressure	3.7 bar

References

Technical information on design features of System 80 Plus.
Review of Advanced Light Water Reactor Design Approaches.
10-13 May, 1994, RF, Moscow, IAEA-TC-879.

TURK, R S and MATIZIE
System 80 +TM: PWR technology takes a major step up the evolutionary ladder.
Nuclear Engineering International, 1992, November, p 15 - 24.

The ABB Combustion Engineering System 80 +
Nuclear News, 1992, September, p 68-69.

SECTION 1

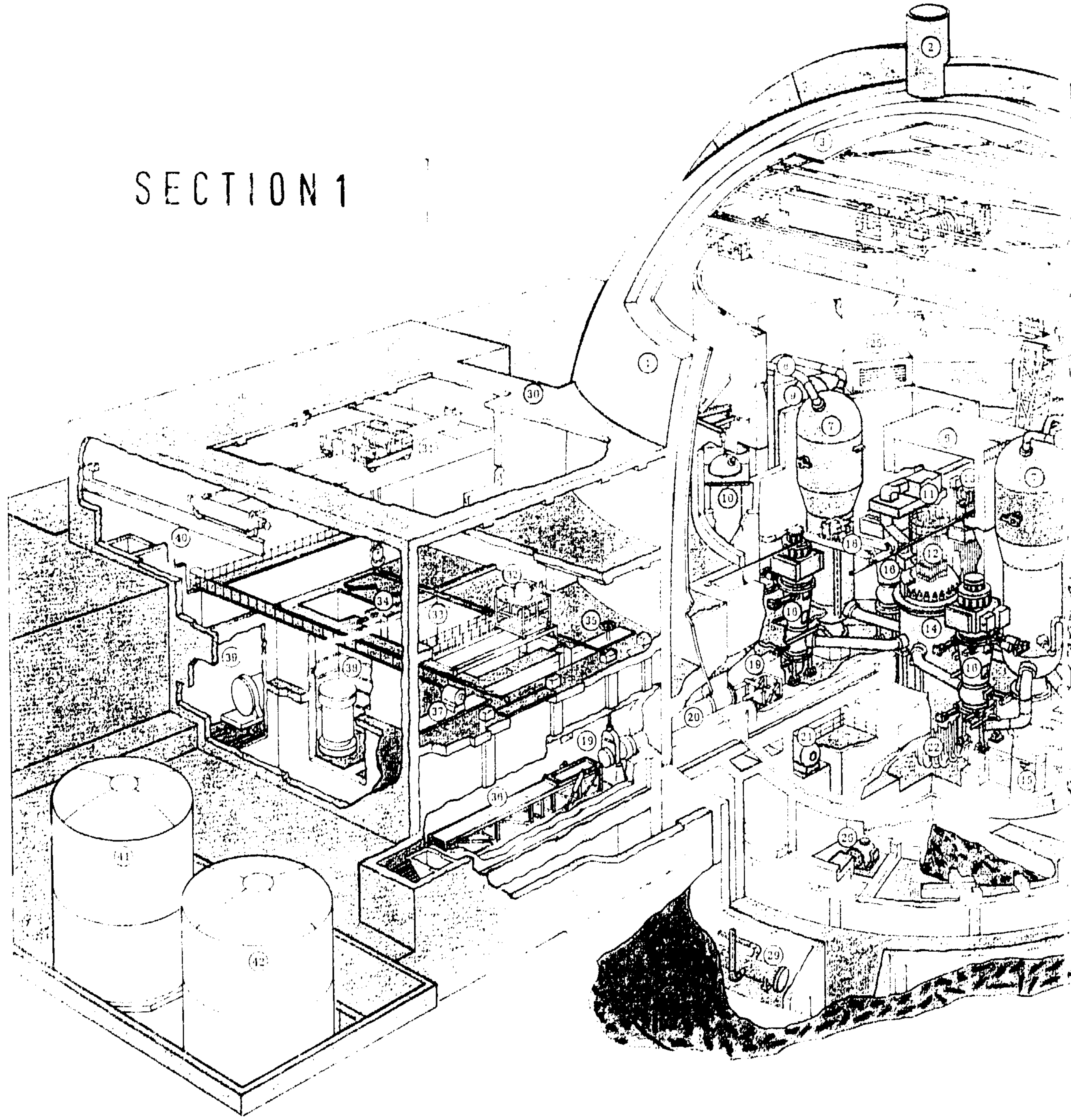
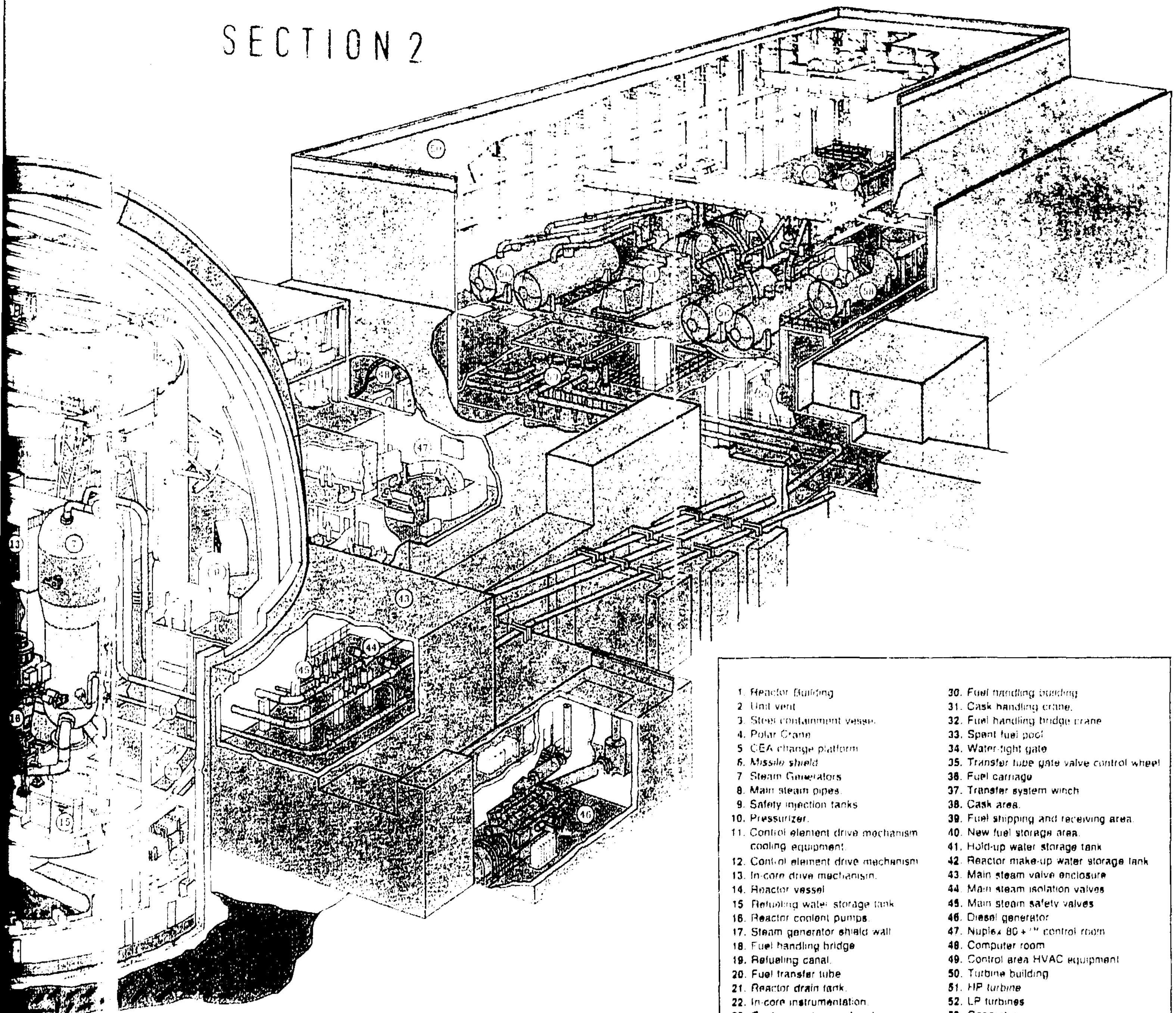


Figure AEP5
System 80 Plus: Layout

SECTION 2



1. Reactor Building
2. Unit vent
3. Steel containment vessel
4. Polar Crane
5. CEA change platform
6. *Missile shield*
7. Steam Generators
8. Main steam pipes.
9. Safety injection tanks
10. Pressurizer.
11. Control element drive mechanism cooling equipment.
12. Control element drive mechanism
13. In-core drive mechanism.
14. Reactor vessel
15. Refueling water storage tank
16. Reactor coolant pumps.
17. Steam generator shield wall
18. Fuel handling bridge
19. Refueling canal.
20. Fuel transfer tube
21. Reactor drain tank.
22. In-core instrumentation.
23. Containment spray headers
24. Feedwater pipe.
25. Containment ventilation cooling units
26. Charging pump.
27. Pipe chase.
28. Cable chase and HVAC duct
29. Shut-down cooling heat exchanger
30. Fuel handling building
31. Cask handling crane.
32. Fuel handling bridge crane
33. Spent fuel pool
34. Water-tight gate
35. Transfer tube gate valve control wheel
36. Fuel carriage
37. Transfer system winch
38. Cask area.
39. Fuel shipping and receiving area.
40. New fuel storage area.
41. Hold-up water storage tank
42. Reactor make-up water storage tank
43. Main steam valve enclosure
44. Main steam isolation valves
45. Main steam safety valves
46. Diesel generator
47. Nuplex 80+™ control room
48. Computer room
49. Control area HVAC equipment
50. Turbine building
51. HP *turbine*
52. LP turbines
53. Generator
54. Alterrex (exciter).
55. Condensers
56. Moisture separators
57. FW heaters
58. Upper surge tanks.
59. Mainsteam stop and control valves

AEP6

Format	Description
Title	VVER-1000 (V-392)
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1000 MWe
Organization (name)	EDO Hydropress, RF
Development status	Detailed design
Description	A number of advanced VVER-1000 versions (V-392, V-410 and V-413) with an electric output 1 000 - 1 100 MWe are now under development in EDO Hydropress.

The following measures aimed at safety improvement in comparison with the commercial reactor plant V-320 have been realized:

- o Advanced steam generators with a large water inventory
- o Four safety train design principle
- o The reactor coolant pump with the design of sealings excluding their leaks in the event of long-term blackout and the absence of sealing water supply
- o Realization of a diagnostic system set and leak-before-break approach
- o A filtered double containment with reinforcement of the floor
- o Containment filtered venting system
- o System of returning the primary circuit coolant
- o The design lifetime of the plants has been increased up to 40 years.

The V-392 and V-413 as well as the V-320 commercial reactor plants have 4 horizontal steam generators. The V-410 version introduces 4 vertical steam generators.

All advanced VVER-1000 reactors use the same fuel rods (diameter 9.1 mm and height 3.54 m) and practically the same fuel assembly design.

Compared with V-392 and V-413, the latest V-410 version has a decreased volumetric core power density (from about 110 to about 85 KW/l), increased RPV diameter and as a result reduced RPV neutron fluence.

The V-413 version, developed with IVO Engineering International (Finland) is based only on proven V-320 safety system technology. To improve the safety the V-392 and V-410 versions incorporate additional water accumulators for emergency core cooling and passive residual heat removal systems via steam generator with air coolers outside the containment.

Schedule

No advanced VVER-1000 plant versions have been ordered yet. The V-413 version has been one of contenders for NPPs in Finland. Especially for the V-410 version significant verification testing and detailed design work remains to be done, prior to commercial utilization.

VVER-1000 Unit Data (V-392, V-413)

Primary pressure	157 bar
Primary temperature	320 °C
Secondary pressure	63 bar
Steam temperature	278 °C
Nominal coolant flow rate	88 000 m^3/h
Average linear heat rating at fuel elements	166.7 W/cm
Humidity at the outlet at full power	0.2 %
Number of control members	97
Number of assemblies	163
Average fuel burnup fraction, (at stationary fuel cycle)	44 MW d/kg
Time of operation at rated power for a year (effective)	7 000 h

References

FEDOROV, V G et al
Designs of advanced VVER-1000 reactors.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, v 4, p 4.4.3.

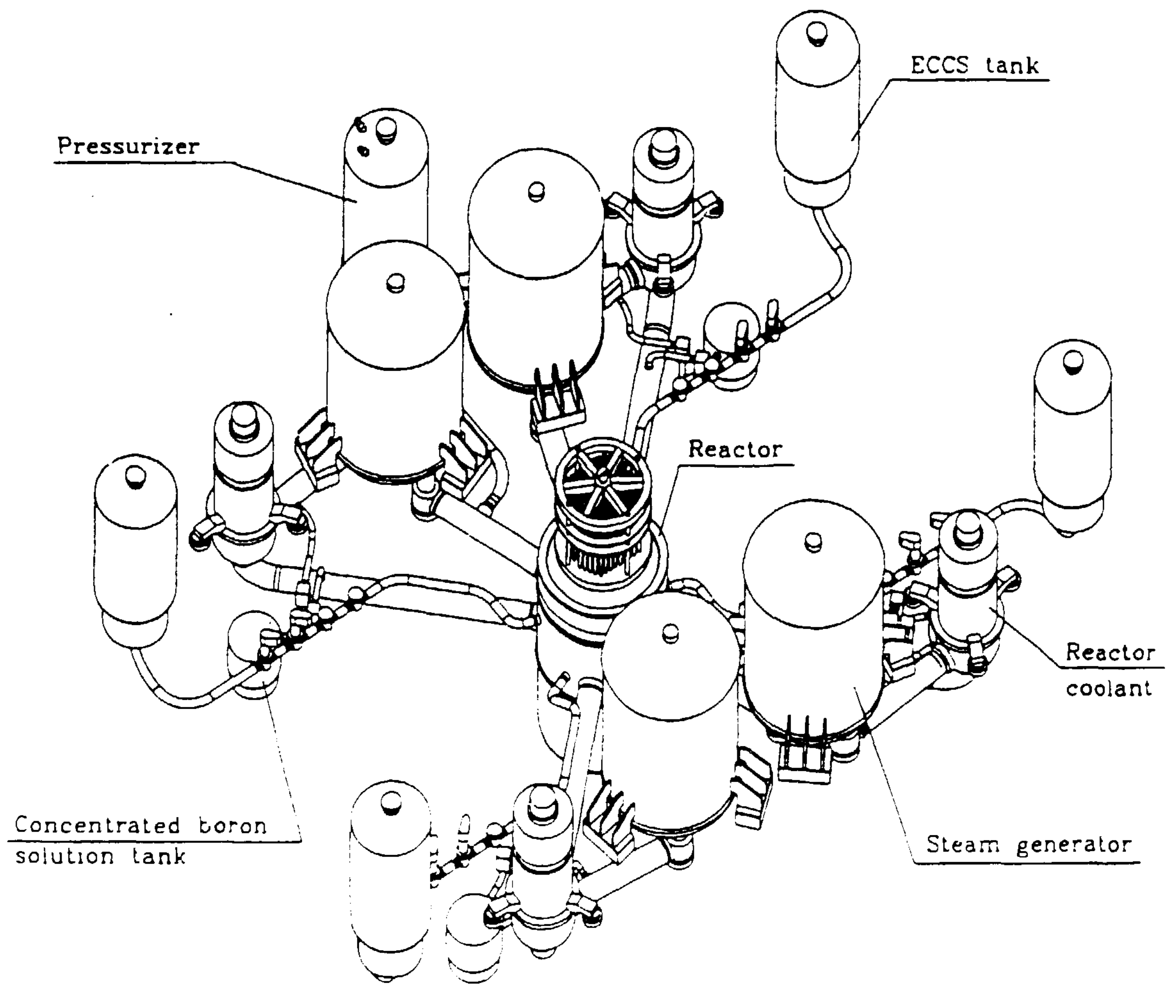


Figure AEP6
V-410: Reactor coolant system.

AEP7

Format	Description
Title	EPR (European Pressurized Water Reactor)
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1500 MWe
Organization (name)	Siemens - KWU, Germany; Framatome, France
Development status	Basic design
Description	The EPR represents an advanced, evolutionary PWR-type plant design which has been developed by Nuclear Power International (a company formed by Framatome and Siemens-KWU), based on N4 plant and KWU-Convoy plant designs. The design goals for EPR include improving functional safety by system design simplifications, strengthening of redundancy by strict physical separation, extending the grace period (up to 24 h) by additional and enlarged water storage capacity. Special attention is given to severe accidents involving core melt corium spreading; strong full pressure double containment for margins and no need for early heat removal from containment. The thermal power of the reactor is 4 250 MWt. The core is based on N4 fuel with 17 x 17 - 25 rods per assembly. The core control principles are based on a combination of the Siemens-KWU and Framatome experience. The containment consists of a prestressed concrete cylinder which can be fitted with a steel liner. To provide a dual wall containment function, the containment is surrounded by a second non-prestressed concrete cylinder, forming an annular secondary containment. Safety systems are installed in an annular building surrounding the outer containment. Both the reactor building and the annular building will be built on the same foundation mat (for seismic protection reasons). Separation between the four safety divisions and one operational division is accomplished in a sector arrangement. The lower area of the containment houses the water supply for the safety injection systems, the in-containment refuelling water storage tank.
Schedule	The development of the EPR has not yet been completed. French and German authorities have agreed upon the harmonization of licensing requirements. According to the overall time schedule, the Basic Design shall be completed during 1995 with an application for construction license scheduled for the end of 1995. The target construction data for FOAKE is 1998.

EPR Unit Data

Electrical output		1 500 MWe
Reactor core		
-	Number of fuel assemblies	264
-	Active length	420 cm
-	Total fuel assembly length	480 cm
-	Number of control rods	81
-	Vessel inlet/outlet temperature	291 ^{0}C/326 ^{0}C
-	Enrichment (max)	4.9 % U235
-	Average fuel burnup	60 MWd/kg
Primary and secondary systems		
-	RCS operating pressure	155 bar
-	RCS design pressure	176 bar
-	Main steam pressure at hot standby	84 bar
-	Secondary side design pressure	91-94 bar
Reactor pressure vessel		
-	Fluence (design-target-60 years)	$1 \cdot 10^{19}$ n/cm^2
Steam generator		
-	Heat transfer surface (with economizer)	7 300 m^2
-	Tube material	Incalloy 800
-	Water amount of 2-nd side	75 t
Containment		
-	Configuration	Double
-	Gross volume	75 000 m^3
-	Designed pressure	7.5 bar

References

IVON, M and KRUGMANN, U et al
Basic Information on the Design Features of the EPR.
Review of Advanced Light Water Reactor Design Approaches
10-13 May, 1994, Moscow, RF, IAEA-TC-879.

WATTENAU, M and SEIDELBERGER, H
The European PWR - a progress report
Nuclear Engineering International, 1994, Oct, p 70-74.

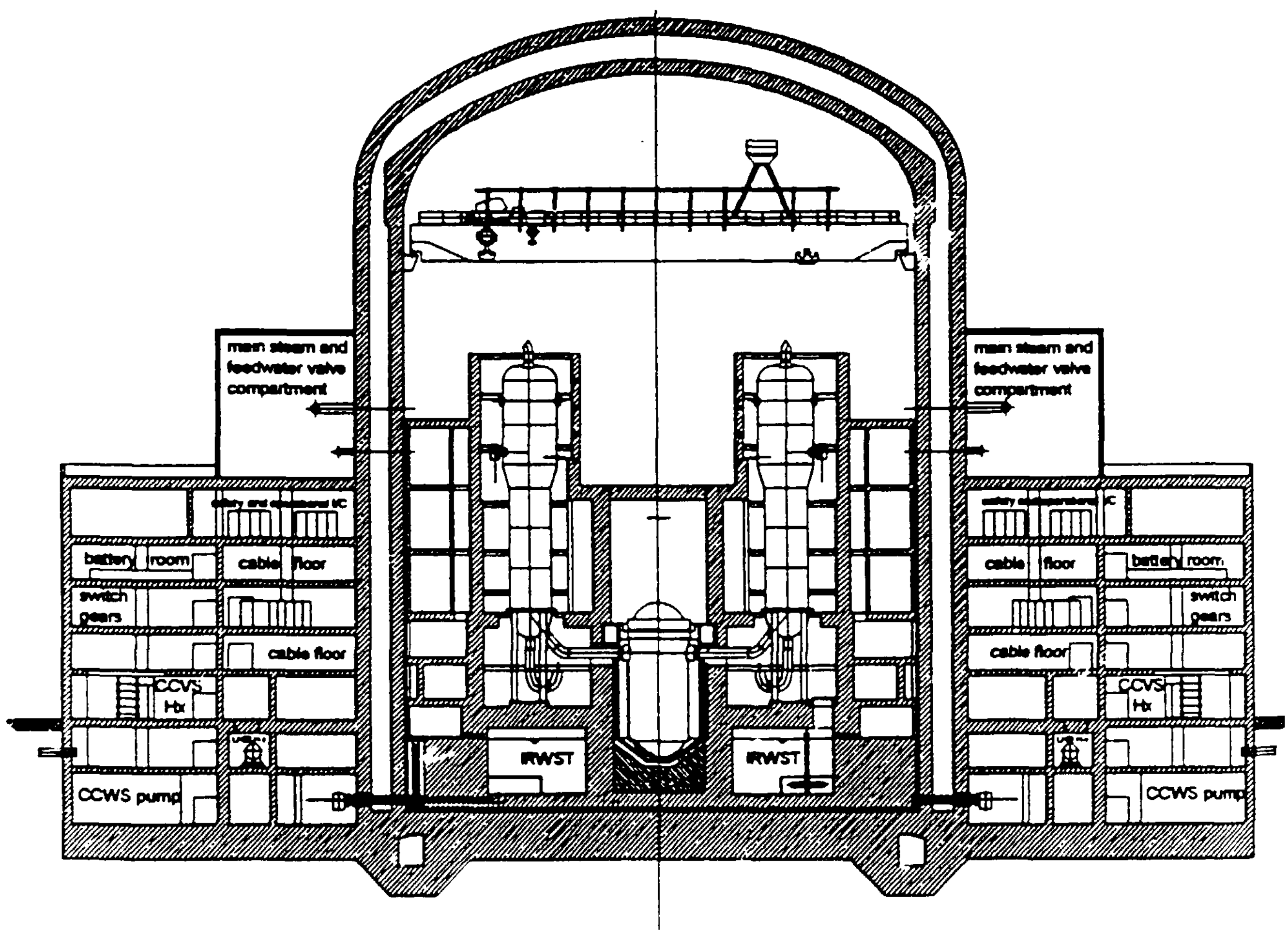


Figure AEP7
EPR: Containment building.

AEB1

Format	**Description**
Title	ABWR
Application of the reactor	NPP
Reactor type	BWR
Power output	Large power/1356 MWe
Organization (name)	General Electric, US; Hitachi/Toshiba Tokyo Electric Power Co (Tepco), Japan
Development status	Commercial
Description	The ABWR with the thermal power output of 3 926 MWt is a combined application of the technical solutions already proven individually in different BWRs. The main features of the ABWR are:

- o Advanced fuel and core design
- o Internal recirculation pumps, at the bottom of the core
- o Decreased hydraulic resistance of RCS. Lower recirculation flow pumping power
- o Fine motion control rod drives with diversified scram actuation (hydraulic piston drive + electric motor drive backup)
- o Digital and solid state control
- o Multiplexing
- o Improved reinforced concrete primary reactor containment and reactor building
- o Three emergency core cooling systems (ECCS) divisions.

The core and fuel designs aim at improving operating efficiency, operability and fuel economy by means of PCI (Pellet Clad Interaction)-resistant fuel, axially- zoned enrichment, control cell core design and increased core flow capability. The latter allows for hydraulic spectral shift operation to provide additional burnup towards the end of the operating cycle. The reactor pressure vessel has been designed to reduce the number of welds and to permit maximum in-service inspections of welds by automatic equipment. All large pipe nozzles to the vessel below the top of the active core have been eliminated, improving safety performance and reducing ECCS capacity.

Schedule

Two ABWR units are under construction in Japan. These units are scheduled to be taken into operation in 1996 and 1997. The ABWR is also undergoing a design certification process in the US. It has been selected by US utilities as one of two designs in a FOAKE program of 1996 that is supported by the utilities and the US DOE.

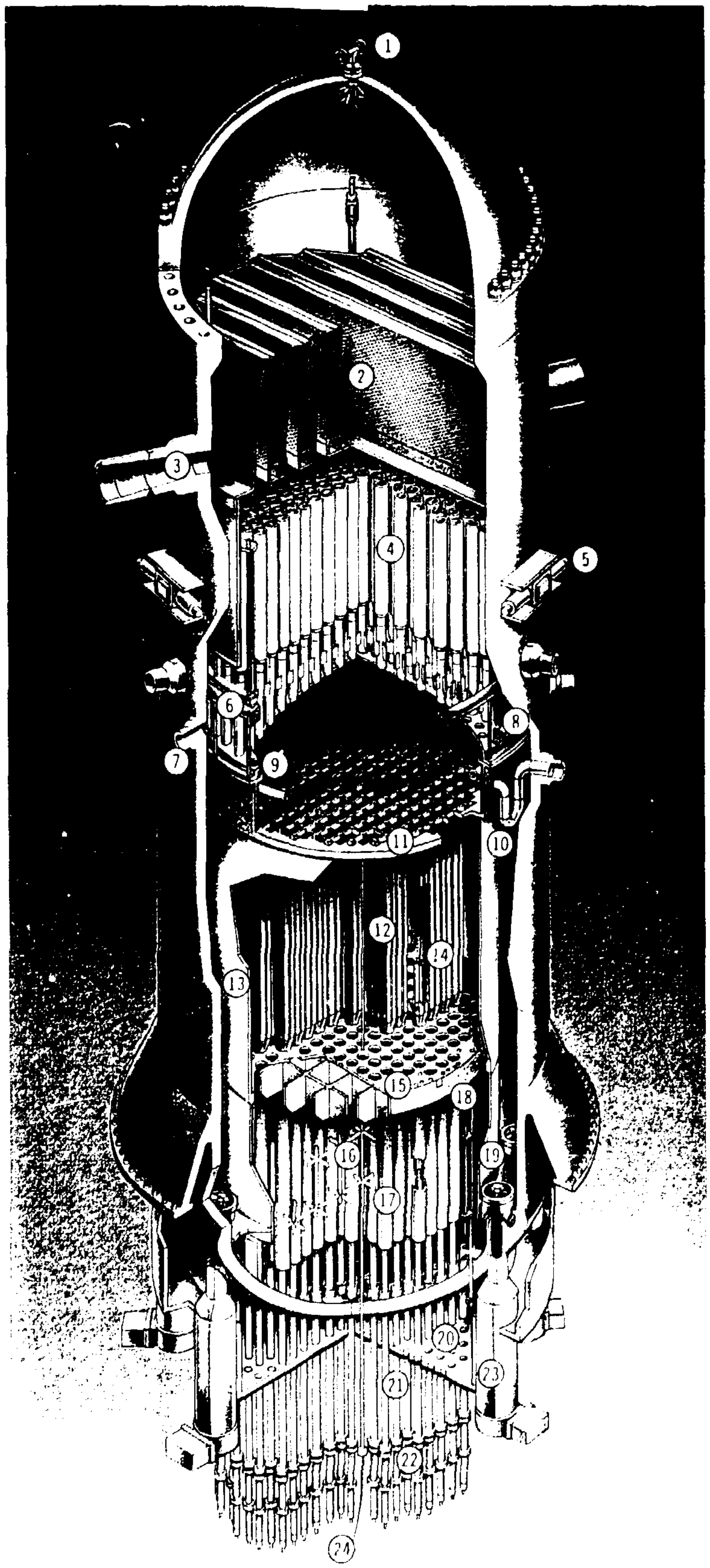
ABWR Unit Data

Electrical output		1 356 MWe
Reactor vessel pressure		71.6 bar
Core power density		50.6 MW/m^3
Number of fuel assemblies		872
Number of control rods		205
Core height		3.71 m
Core diameter		5.16 m
Fuel		
-	Average UO_2 enrichment	3.2 %
-	Number of fuel rods	62
-	Fuel rod array	8 x 8
-	Cladding material	Zircaloy 2
-	Fuel burnup	32 MWd/kg
Reactor coolant system		
-	Inlet feedwater temperature	215.5 °C
	Outlet feedwater temperature	287.4 °C
	Number of pumps	10
Turbine		
-	Number	1
-	Peak electric power	1381 MWe
-	Inlet steam pressure	67.5 bar
-	Inlet steam temperature	283.7 °C

References

Safety Aspects of Designs for Future Light Water Reactors (Evolutionary Reactors)
IAEA, Vienna, 1993, IAEA-TECDOC-7123.

GE's ABWR and SBWR
Nuclear News, September, 1992, p 70-74.



1 Vent and Head Spray
2 Steam Dryer
3 Steam Outlet Flow Restrictor
4 Steam Separators
5 RPV Stabilizer
6 *Feedwater Sparger*
7 Shutdown Cooling Outlet
8 Low Pressure Flooder (LPFL) and Shutdown Cooling Sparger
9 High Pressure Core Flooder (HPCF) Sparger
10 HPCF Coupling
11 Top Guide
12 Fuel Assemblies
13 Core Shroud
14 Control Rod
15 Core Plate
16 In Core Instrument Guide Tubes
17 Control Rod Guide Tubes
18 Core Differential Pressure Line
19 Reactor Internal Pumps (RIP)
20 Thermal Insulation
21 Control Rod Drive Housings
22 Fine Motion Control Rod Drives
23 RIP Motor Casing
24 *Local Power Range Monitor*

Figure AEB1
ABWR: Reactor pressure vessel and internals.

AEB2

Format	Description
Title	BWR 90
Application of the reactor	NPP
Reactor type	BWR
Power output	Large power/1379 MWe
Organization (name)	ABB Atom, Sweden
Development status	Commercial
Description	The BWR 90 is an advanced BWR plant design, based on the reliable operation of BWR 75 design Forsmark 3 and Oskarshamn 3 units in Sweden. Modifications are mainly moderate and have been made to adapt to updating technologies, modern safety requirements and to achieve cost savings measures for simplified operation, testing and maintenance. The core design is based on the advanced fuel assembly design, SVEA-100, which uses thinner rods (9.62 mm in diameter instead of 12.25 mm for SVEA-64 fuel assembly) and provides improved moderation and increased heat transfer efficiency. The design margins for the fuel have increased, and some of these margins have been utilized to raise the power levels, compared with predecessors. The advanced burnable absorber strategy, with axial and radial grading is maintained. In the BWR 90 plant core cooling is provided by 8 internal recirculation pumps, as in BWR 75 plants, although somewhat larger in size. Also the prestressed concrete reactor containment is equipped with filtered venting and an extensive redundancy and separation of safety-related systems has been installed.
Schedule	The BWR 90 plant is currently ready for commercial use. No BWR 90 plant has been ordered yet. The BWR 90 was offered to Finland as a contender for the fifth Finnish nuclear power plant that was cancelled last year.

BWR 90 Unit Data

Overall plant

-	Thermal power	3 300 MW
-	Net electric power	1 170 MW

Reactor plant

-	Assembly geometry	4 x (5 x 5)
-	Number of fuel elements per assembly	100
-	Total uranium weight in first core	126 t
-	Specific power of reactor core	52 KW/l
-	Total coolant flow rate	13 100 kg/s
-	Feedwater temperature	215 ^{0}C
-	Coolant temperature at core outlet	286 ^{0}C
-	Reactor operating pressure	70 bar
-	Average fuel burnup	47 MWd/kg

Turbine plant

-	Total main steam throughput at full load	1 775 kg/s
-	Turbo-generator power	1 215 MW
-	Number of turbine casings	1 HP + 3 LP

References

LÖNNERBERG, B and PEDERSEN, T J
BWR 90: An evolutionary ABWR plant for the next decades
ASME/ISME Nuclear Engineering Conference, Vol. 2, ASME, 1993, p 633-638.

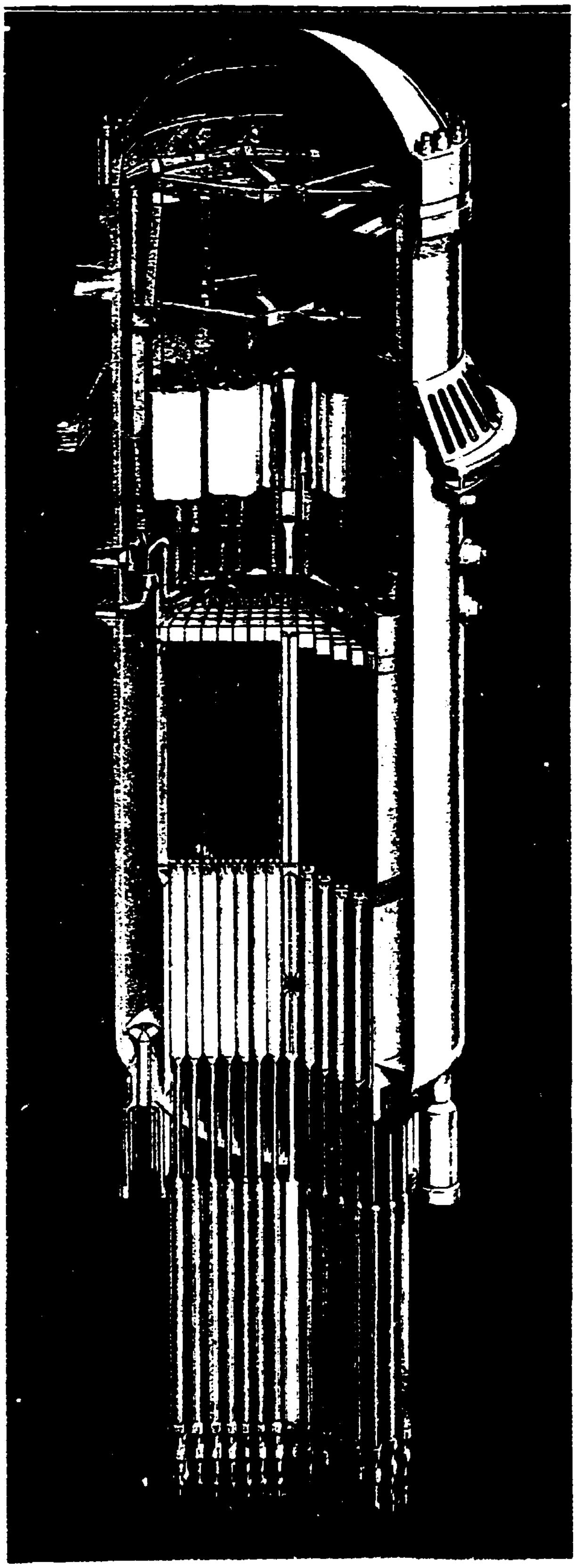


Figure AEB2
BWR 90: Reactor pressure vessel and internals.

AEH1

Format	Description
Title	CANDU 9
Application of the reactor	NPP
Reactor type	HWR
Power output	Large power/1050 MWe
Organization (name)	AECL (Atomic Energy of Canada, Ltd), Canada
Development status	Commercial

Description

The first CANDU NPP was taken into operation in 1962. Currently, there are about 30 CANDU reactor units operating, or under construction around the world. Four CANDU 6 units are in operation and five are under construction in three different countries. The design rating of the Mk I units is of the order of 650 MWe, while the Mk II has a net rated output of about 800 MWe; increased ratings mean a lower special capital cost. The power increase was achieved with only minor changes in design and equipment.

The CANDU 9 design follows the same evolutionary path. This reactor plant also incorporates advanced features: reduced capital and generating costs, and a minimized construction schedule, achieved by a high level of standardization and modularization. The more extensive use of computers for safety systems, the use of multiplexing and data highway technology reduce instrumentation and control wiring. Self-checking is also used.

The use of slightly enriched uranium (1.2 % U-235) in CANDU 9 in place of natural uranium opens the opportunity to increase electrical output to the range of 1 050 MWe within the same operating limits as CANDU plants now in service.

The four safety systems comprise the two diverse, dedicated reactor shutdown systems (gravity-drop absorbing rods and liquid neutron absorber). The emergency core cooling system uses ordinary water for injection. Improvements incorporated in the CANDU 9 emergency core cooling system (ECCS) include:

o Placement of all ECCS equipment, except the gas tanks, within the reactor building, will eliminate the need for a number of isolation valves, thereby simplifying the ECCS system.

o The use of one-way rupture discs to separate the heat transport system from the ECCS, thereby greatly simplifying the ECCS and reducing both the capital and maintainance costs.

Schedule

No CANDU 9 plant has been ordered yet. The detailed design is under way now.

CANDU 9 Unit Data

Electrical output	1 050 MWe
Primary pressure	80 bar
UO_2 enrichment	1.2 %
Assembly	37-element bundle
Number of fuel channels	480
Primary circuit configuration	Loop
Reactor pressure tube	
- Material	Zr-2.5 Nb
- Diameter	0.1 m
Height	6.0 m
Containment construction	Steel lined reinforced concrete

References

FEHRENBLACK, P J and DULTON, R
Evolutionary of CANDU technology: a status report.
Advanced technologies for water cooled reactors.
IAEA, Vienna, Austria, 1993, IAEA-TC-633.20.

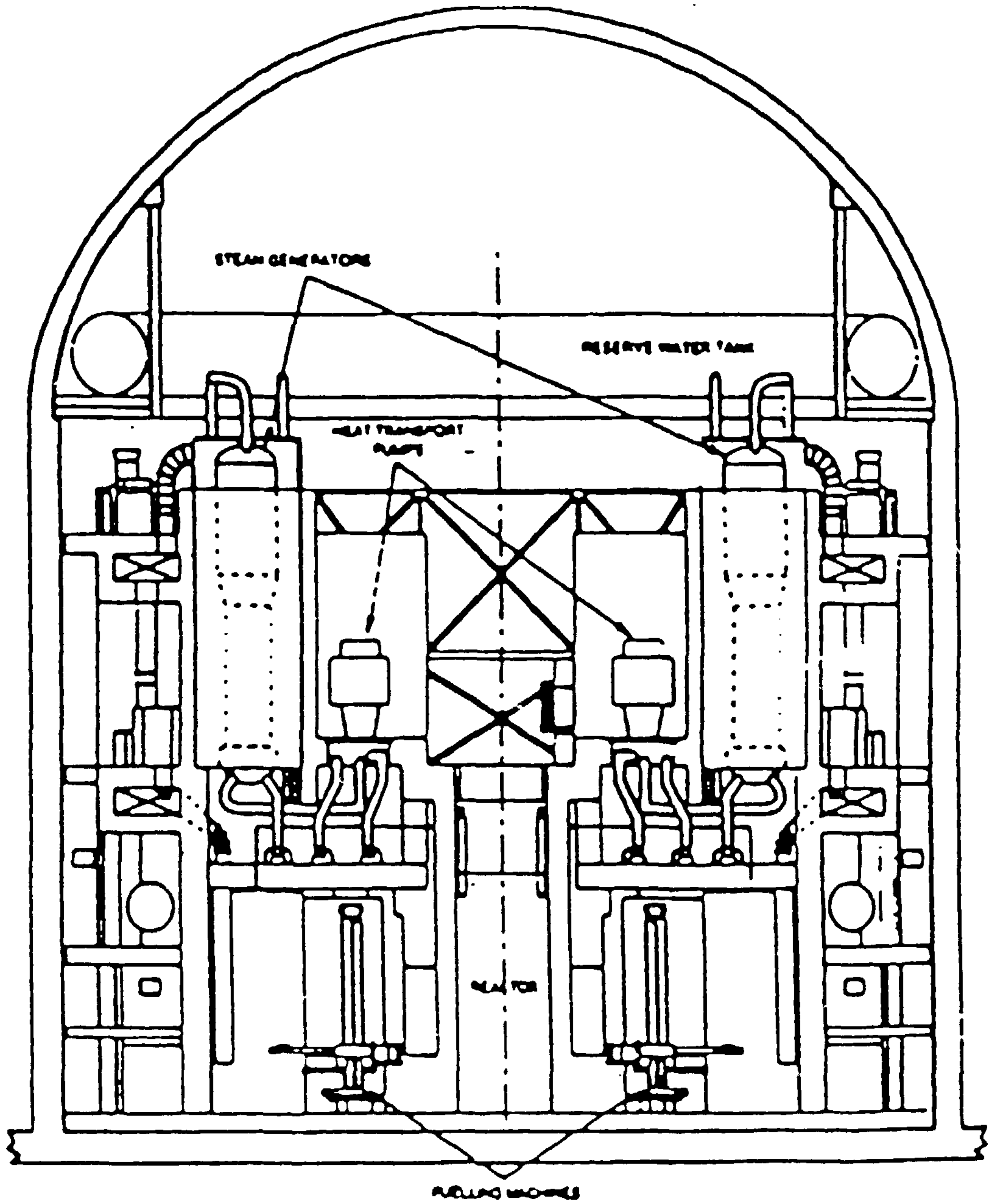


Figure AEH1
CANDU 9: Reactor building.

AT Evolutionary Designs for Medium Power Plants

AT0 Introduction

The medium power (600 MWe or less) range for most of these designs is compatible with the present trend of reduced growth rates in electricity production in the developed countries and will more easily connect with local and limited electrical networks of developing ones.

Also, to meet the economic challenges posed by cheap oil and coal, some design teams have looked at two power levels: 600 MWe and 1 000 MWe ranges (e.g. SBWR, AP-600 and SWR-1000).

Common to most of the evolutionary designs for medium power plants is a significant departure from present-day designs. In this case, the design safety principles addressed to different functions and systems are integrated into comprehensive and balanced sets of measures at the NPP level.

The main features and objectives of the considered approach are the following:

o Plant simplification, both at system and at component level, including construction and operational characteristics. This aims to save plant costs, along with standardization

o Additional safety margins on the "traditional" design basis accidents (basically increased thermal margins for the fuel and coolant inventory)

o Extensive application of inherent safety characteristics (e.g. temperature, density and size feedbacks) and passive safety features (e.g. gravity, natural circulation, evaporation, condensation e.a.)

o Slow accident progression rates (i.e. the reaching of critical limits of response parameters is delayed under perturbations)

o The capability to cope with any considered event for a pre-determined "grace period" without necessarily relying on human actions for the management of the accident

o Improved reliability of safety functions, with particular reference to reducing human error importance and system failure occurrence rates

o Protection against a large set of accidents including severe core damage accidents with failure of the reactor vessel

- o Reduction of the rate of initiating faults (e.g. deviations from normal operation requiring reactor scram)
- o Simplification of emergency preparedness requirements
- o Adoption, as extensively as possible, of fully proven, state-of-the-art technologies, without the exclusion of promising technological advances

Considerable experience accumulated in construction and operation has led power-producing companies to renew their interest in medium power plants.

As noted above, medium power range is more open (due to the lower unit power and decreased specific core power) for the implementation of the natural processes in the safety systems. It is expected that reliance on passive* safety features will lead to a better understanding by the general public and recognition that a major improvement in public safety has been achieved.

The use of passive systems and components offers a number of obvious advantages, but may introduce new problems, such as:

- o Different design methods that have not yet been definitely assessed, including the tools to evaluate performance, capabilities and margins
- o More difficult thermohydraulic circuit-balancing, including the characterization of natural circulation loops (with small pressure drops across the circuit) when operating together with active (forced convection) systems
- o Augmented importance of phenomena such as thermal stratification and the presence of incondensable gases
- o Increased importance of heat losses and circuit layout to avoid conditions adverse to the initiation of natural circulation
- o Low water heads (especially in the long term), requiring large diameter pipes for water/steam flows and to avoid steam binding and small differential pressure heads to open check valves that normally respond to large reverse pressures
- o Difficulties in testing and maintenance during normal operation and in assessing availability for emergency operations

---

* The word "passive" is perceived as negative by the general public but it is hard to find a relevant short expression: why not "natural"?

Medium power evolutionary technology LWR designs are now under consideration in the USA, China, Great Britain, Japan, Russia and some other countries. Only one mid-design represents HWR. It makes full use of proven technology with relevant features resulting from the Canadian research and development program. Another one represents the continuation of the Russian RBMK line.

References

PEDERSEN, T J
Current trends in safety philosophy and design goals for advanced nuclear power plant designs.
Proceedings ASME/ISME Nuclear Engineering Conference, Vol 2, 1993 ASME, p 539-544.

Safety aspects of Designs for Future Light Water Reactors (Evolutionary Reactors).
IAEA, Vienna, 1993, July, IAEA-TECDOC-712.

International Nuclear Congress - Atom for Energy
Proceedings ENS-94, Vol 1, 1994, October 2-6, Lyon, France.

ATP1

Format	Description
Title	VVER-500/600
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/600 MWe
Organization (name)	EDO Hydropress, RF
Development status	Detailed design
Description	The VVER-500/600 reactor plant with a thermal power of 1 800 MW is developed on the basis of well-established VVER practice. Approaches are close to the world tendency for medium-power loop-type advanced PWR plant designs:

- o Reduction in the unit power
- o Reduction in specific core power density
- o Simplification of the structural scheme of the reactor plant
- o Use of proven technical solutions
- o Four 60 bar accumulators and four 36 bar accumulators
- o Coolant natural convection for residual heat removal
- o Primary circuit depressurization system opening to pressure decrease down to 3 - 5 bar
- o Compensation for water losses from the steam generator by passive systems
- o External water cooling for the steel containment
- o 4 diesel generators (2 large and 2 small)

In the design the four-loop reactor with horizontal steam generators is used. The prototype reactor is the VVER-1000 reactor design. The core power density is about 65.4 kW/l and the reactor pressure vessel has been made rather large (the same as the VVER-1000) in order to guarantee large water inventory.

The core with absorber rods inserted is subcritical at the initial moment of the cycle for a coolant temperature less than 100°C even in the event of borated water substitution for pure condensate. The efficiency of fuel is increased when compared to operating NPPs equipped with VVER-1000 by 20 - 25 %. The service life of the major equipment is extended up to 50 years.

The containment has a cylindrical arrangement: the inner shell is leaktight made of steel, the outer shell of reinforced concrete. Filter vents from the leaktight shell and corium trap installation are envisaged.

The NPP is designed to survive blackout under normal operating conditions, operating events and design basis accidents for not less than 24 hours.

The design envisages NPP construction in different climatic conditions and in seismically dangerous areas with an earthquake intensity of 8 points as per MSK-64.

For regions with an earthquake intensity of 8-10 points the application of low-frequency seismic isolating devices reducing the seismic loads by 10-30 times is envisaged.

Schedule

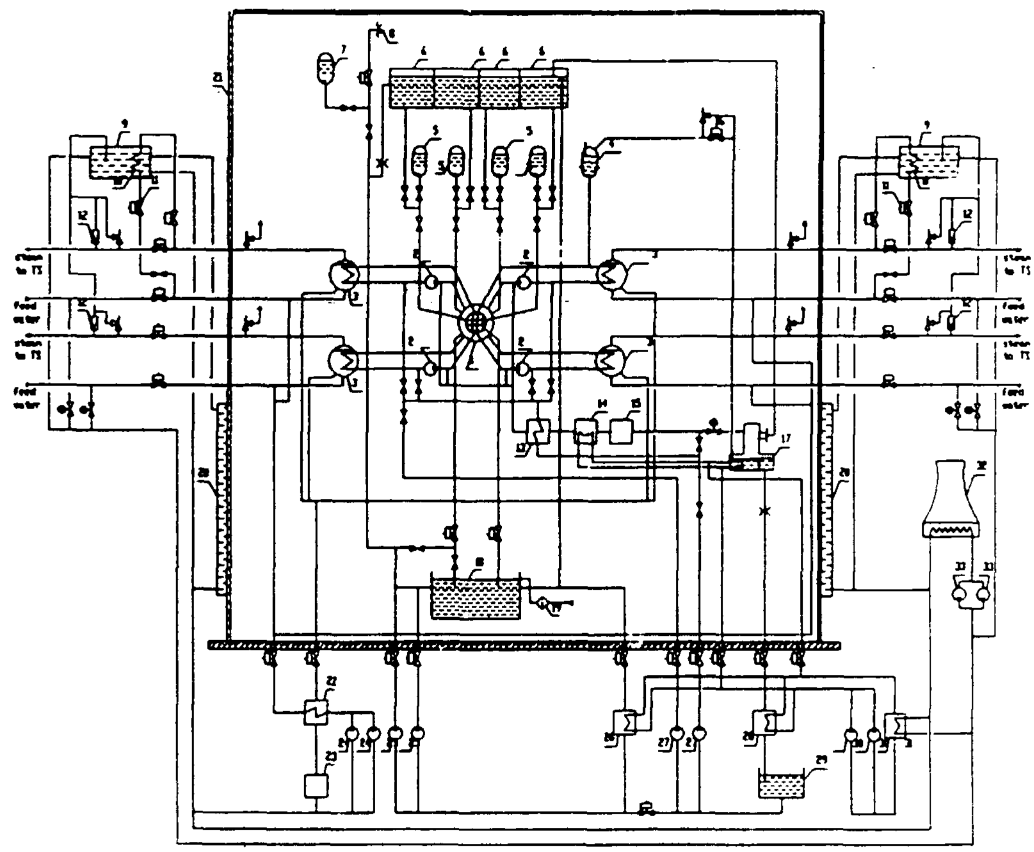
No VVER-500/600 plant has been ordered yet. Some testing programs are in progress to support the final design and licensing process.

VVER-500/600 Unit Data

Rated thermal power	1 800 MW
Primary coolant pressure	157 bar
Primary coolant average temperaturs	320 °C
Pressure of produced saturated steam	70.6 bar
Average core power density	65.4 kW/l
Average burnup	39.6 MWday/kg
Average makeup fuel enrichment	3.45 % (mass)
Reactor vessel neutron fluence (for 50 years)	$2.5 \cdot 10^{19}$ n/cm^2
Containment gross volume	60 000 m^3

References

VOZNESENSKY, V, KUHTEVITSH, I and ROGOV, M
Advanced NPP VVER-500 with passive safety systems.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, v 1, p 44.



1. Reactor
2. Main coolant pump
3. Steam generator
4. Pressuriser
5. Hydroaccumulator
6. Low pressure tank
7. Tank of chemicals for iodine precipitation
8. Spray header
9. Special heat-accumulating tank
10. Heat exchanger of passive heat removal system (PHRS)
11. Connecting valves
12. Safety bypass valves
13. Regenerative heat exchanger
14. Letdown heat exchanger
15. Ion exchanger
16. Pressure relief system
17. Barboter-degasator
18. Spent fuel storage pool (SFSP)
19. Pipeline between SFSP and accidental heat removal basin
20. Coolers of containment heat removal system
21. Hermetic steel shell
22. Regenerative heat exchanger
23. SG water cleanup system
24. SG water cleanup system pump
25. SFSP cooling system pump
26. SFSP cooling system heat exchanger
27. Primary circuit charging pump
28. Primary water discharge cooler
29. Makeup surge tank
30. Intermediate cooling water pump
31. Intermediate water cooler
32. Dry cooling tower
33. Essential service water pump

Figure ATP1
VVER-500/600: Safety and normal operation systems.

ATP2

Format	**Description**
Title	AP-600 (Advanced Passive)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/630 MWe
Organization (name)	Westinghouse, US
Development status	Detailed design

Description

The AP-600 is a plant design developed by Westinghouse together with Bechtel, Avondale Industries, Burns & Roe and others. The reactor coolant system comprises the RPV, two vertical SGs, four RCPs, pressurizer and linking pipework. A two loop Westinghouse PWR arrangement was modified to have conservative safety margins and permit simplification of many supporting subsystems. The AP-600 features incorporate an improved RCS configuration utilizing canned motor reactor coolant pumps directly coupled to the steam generator bottom. This eliminates the "cross-over leg" from the RCS, lowering the overall system flow resistance, and improving the performance with respect to small break loss-of-coolant accidents. This greatly simplifies the RCS loop support configuration.

The thermal power rating of the reactor core is 1 812 MW. The fuel assembly design is basically the same as in other Westinghouse PWRs, but the power density has been reduced by approximately 30 %, the linear heat rating has been reduced correspondingly. Combined with the introduction of a radial neutron reflector, this has reduced the enrichment needs for the fuel. The low power density core, the radial reflector and a somewhat larger distance between the core and the RPV wall have resulted in a significant reduction in neutron fluence. The reactor internals are essentially of a conventional design.

The large reactor vessel and the increased core coolant flow cross section reduced the hydraulic resistance of the RCS to the level at which 17 - 25 % power can be removed by natural convection at nominal coolant heating. The increased water inventory resulting from increased RPV and pressurizer volumes improves the reactor safety ability.

The key safety features of the AP-600 reactor plant are:

o Small leaks up to 6" diameter can be handled without RCS depressurization

- For large pipe breaks the peak clad temperature does not exceed 980 °C

- Containment cooling is accomplished by a combination of wetting and the natural circulation of air through gaps between the shield building and the steel containment

- Passive systems are expected to perform their safety functions for up to 72 hours after the initiation event, independent of operator action and off-side power

This approach provides an opportunity for higher availability, NPP simplification, reduction of costs and a shorter construction period. Also Westinghouse began working with Japanese utilities and EdF (France) to investigate the feasibility of applying the AP-600 passive concepts to a three loop 1 000 MWe reactor plant, known now as AP-1000.

Schedule

The AP-600 development programme to obtain a design certification from the US NRC by 1996 is well underway. Some testing programs are in progress to support final design and licensing processes. No AP-600 plant has been ordered yet, but Westinghouse, and its associates in the design team have been awarded a FOAKE contract with US utilities and US DOE (1996 Sept).

AP-600 Unit Data

Gross electric power	630 MWe
Number of cold pipelegs	4
Number of hot pipelegs	2
Number of fuel assemblies	145
Type of fuel assembly	17 x 17
Number of fuel elements	38 280
Maximum coolant temperature	324.4 °C
Average linear heat rating of fuel elements	12.6 kW/m
Average core power density	73.9 MW/m^3
Reactor fuel loading	61.02 t
Reactor vessel inner diameter	399 cm
Neutron fluence for reactor vessel (for 60 years)	$2 \cdot 10^{19}$ n/cm^2

References

Safety aspects of designs for future LWRs (Evolutionary reactors) IAEA-TECDOC-712, July 1993, Vienna, Austria.

The Westinghouse AP-600
Nuclear News, 1992, September, p 76-77.

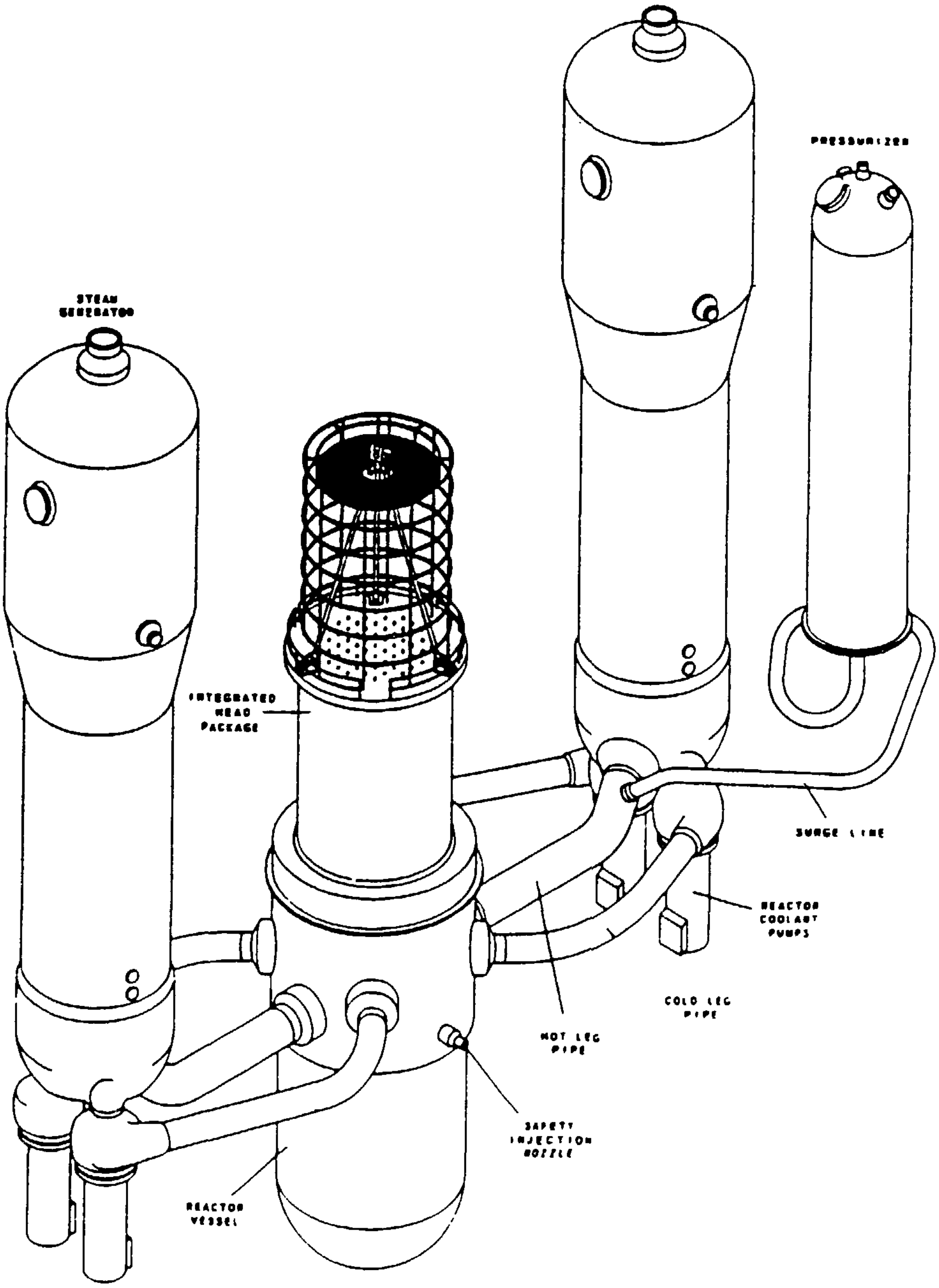
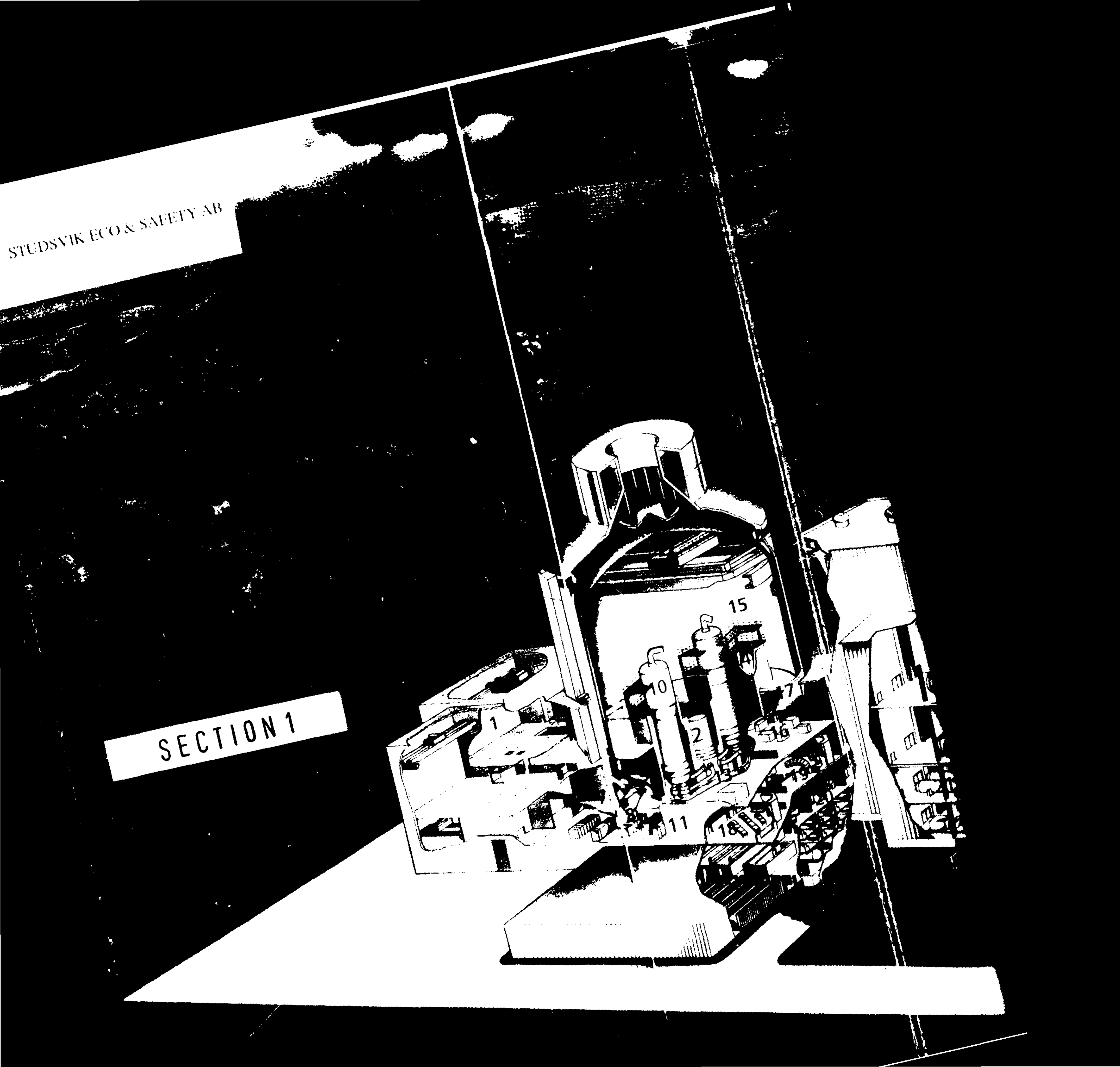


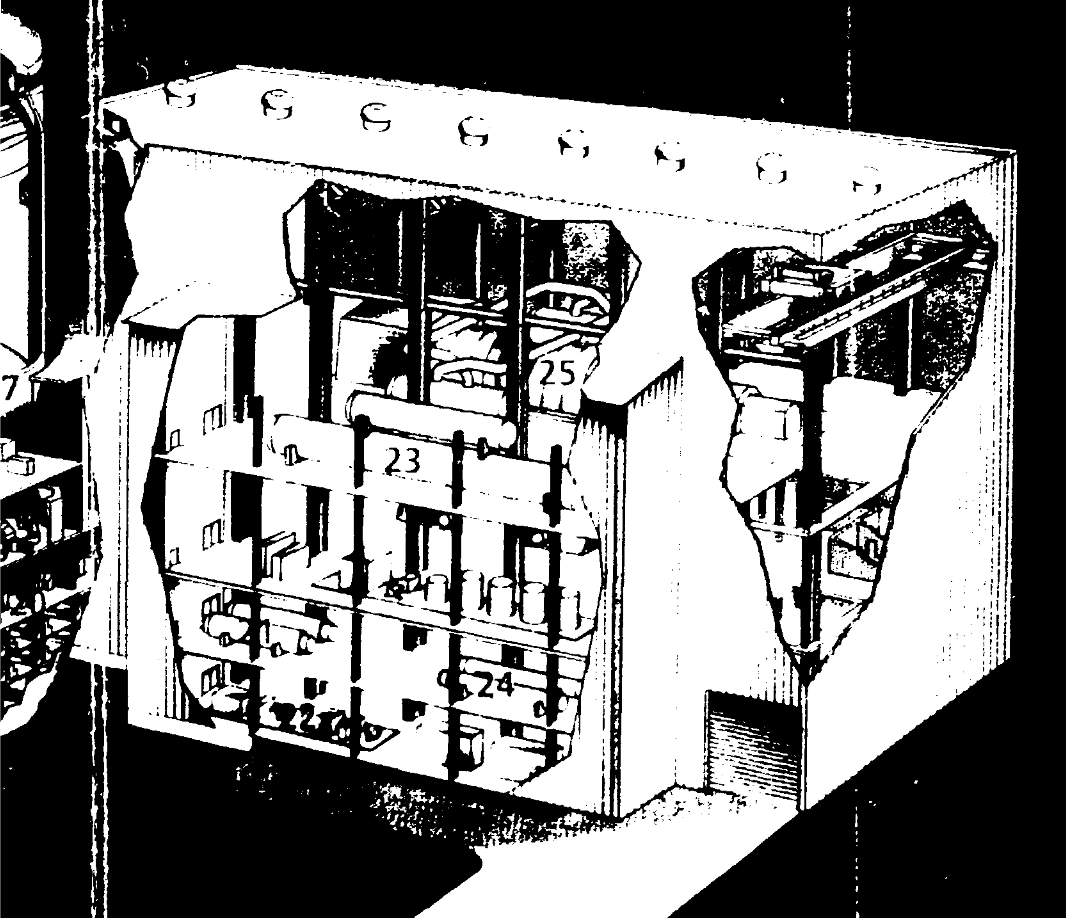
Figure ATP2
AP-600: Reactor plant layout.


STUDSVIK ECO & SAFETY AB
SECTION 1
1
10
15
2
3
7
16
19
11
18

SECTION 2

The Westinghouse AP600

1 Fuel handling area
2 Concrete shield building
3 Steel containment
4 Passive containment cooling water tank
5 Passive containment cooling air baffles
6 Passive containment cooling air inlets
7 Equipment hatches (2)*
8 Personnel hatches (2)
9 Core makeup tanks (2)
10 Steam generators (2)
11 Reactor coolant pumps (4)
12 Integrated head package
13 Reactor vessel
14 Pressurizer
15 Depressurization valve module location
16 Passive residual heat removal heat exchangers
17 Refuelling water storage tank
18 Technical support centre
19 Main control room
20 Integrated protection cabinets
21 High pressure feedwater heaters
22 Feedwater pumps
23 Deaerator
24 Low pressure feedwater heaters
25 Turbine generator



ATP3

Format	Description
Title	MS-600 (Mitsubishi Simplified)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/630 MWe
Organization (name)	Mitsubishi, Japan
Development status	Basic design
Description	The MS-600 reactor plant has a two-loop PWR arrangement, aiming at enhanced safety, reliability and cost reduction.

The thermal power of the reactor is 1 825 MWt. The core consists of 17 x 17 array assemblies and the volume power density has been lowered by 30 % of that of present-day large power PWR plants. Neutron economy and fuel cycle costs are improved by means of a steel radial neutron reflector. Core instrumentation is top-mounted, in turn, this simplifies the containment design and facilitates inspection and maintenance.

Two horizontal steam generators with increased water inventory are used in MS-600. Previously, it was a special feature of Russian VVER designs. This would eliminate a lot of traditional vertical SG problems due to crud depositions and subsequent stress corrosion cracking. The reactor coolant pumps are of a more efficient type, and with improved shaft sealings.The primary system, the spent fuel storage and the water tanks for safety injection are located inside a spherical steel containment A secondary containment was made up by a concrete filled steel. The space between the primary and secondary containments is vented to the environment via a passive filtering system.

The safety systems are based upon current technology with application of new materials and have been simplified to improve reliability and operation. The MS-600 plant design uses active systems for the minor accidents (small LOCAs, transients without flooding the cointainment) and passive systems for the serious accidents.

The active sytems includes pumps for safety injection, auxiliary feedwater supply and small diesel generators. Passive safety functions are achieved by an automatic depressurization system, pressure and gravity-driven hydroaccumulators.

Schedule

A detailed design and design artification test program is scheduled for completion by 1997. An expanded development effort, involving cooperation with utilities in Japan, is planned for the near future, aiming at the design of a 1 200 MWe NPP based on the MS-600 principles. This programme has been given a high priority with respect to development of the next PWR generation in Japan.

MS-600 Unit Data

Electrical output	630 MWe
Average linear heat rating of fuel elements	15.1 kW/m
Fuel assemblies	
- Type	17 x 17
- Number	145
Turbine	TC4F40
Reactor coolant system	
- Number of loops	2
- Operating pressure	154 bar
Temperature	
- Reactor outlet	325.0 ^{o}C
- Reactor inlet	290.6 ^{o}C
Steam generators	
- Number	2
- Type	Horizontal, U-Tube type
Steam pressure	57 bar
Reactor coolant pumps	
- Number	2

References

Technical Information on Design Features of MS-600
Nuclear Power Engineering Corporation, Tokyo.
Review of Advanced Light Water Reactor Design Approaches.
10-13 May, 1994, Moscow, RF, IAEA-TC-879 .

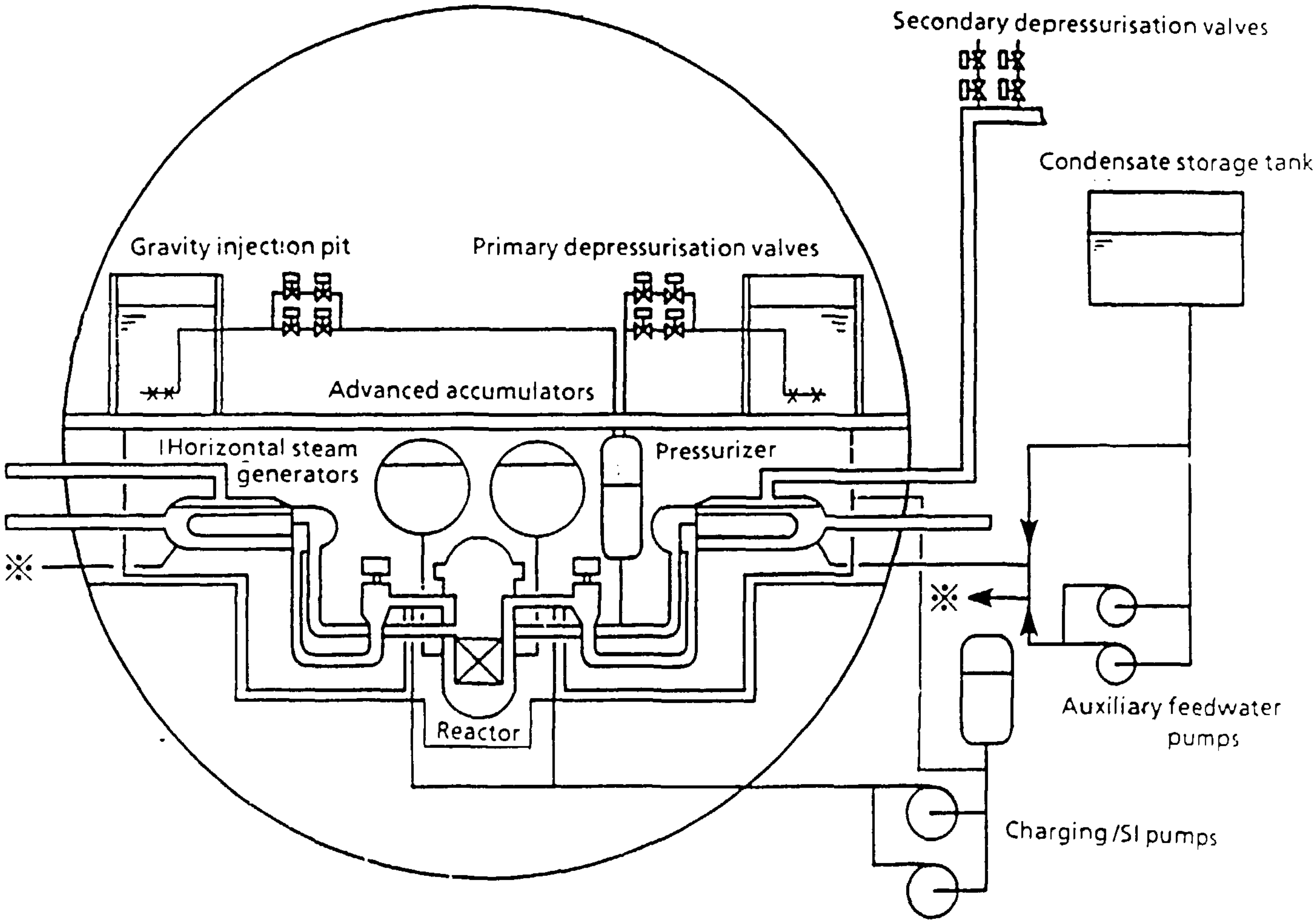


Figure ATP3
MS-600: Configuration of hybrid safety systems.

ATP4

Format	Description
Title	AC-600
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/600 MWe
Organization (name)	China National Nuclear Corporation, China
Development status	Basic design
Description	The AC-600 design is expected to improve the economy and safety of current standard PWR designs by using system simplification, passive means and modular construction. Many features of the AC-600 reactor plant are similar to the well-known Westinghouse AP-600 design (see ATP2). AC-600 has an increased margin of operation because of a decreased core power density. Increased natural circulation flow rate in RCS is an important factor of the AC-600 concept. The increased volume of reactor pressure vessel and pressurizer improves the reactor ability for self-control and makes the black-out emergency situation milder. The AC-600 concept eliminates the high pressure safety injection pumps, utilizes the full pressure core make-up tanks and larger hydroaccumulators.
Schedule	AC-600 will become a major type of reactor for the next generation 600 MWe nuclear power plants in China. The conceptual design is completed and is ready for detailed design development.

AC-600 Unit Data

Rated thermal power		1 936 MW
Fuel assembly		
-	Array	17 x 17
-	Number of fuel rods	38 280
-	Outside diameter	9.5 mm
Reactor core		
-	Number of fuel assemblies	145
-	Total weight of UO_2	66.8 t
Reactor coolant system		
-	Nominal pressure	15.8 bar
-	Vessel outlet/inlet temperature	327/293 °C
-	Average linear rating	134.2 W/cm
-	Average core power density	77 kW/l
-	Neutron fluence for 60 years	$2 \cdot 10^{19}$ n/cm^2
Steam generator		
-	Type	Vertical, U-tube
-	Number	2
-	Heat transfer surface	5 430 m^2
-	Water volume of 2-nd side	169 m^3
Containment		
-	Configuration	Double (steel/concrete)
-	Gross volume	50 000 m^3

References

MIN YUAN-YOU e.a.
Basic Information of Design features of AC-600 Advanced Reactor Plant.
Review of ALWR Design Approaches.
10-13 May, 1994, Moscow, RF, IAEA-TC-879.

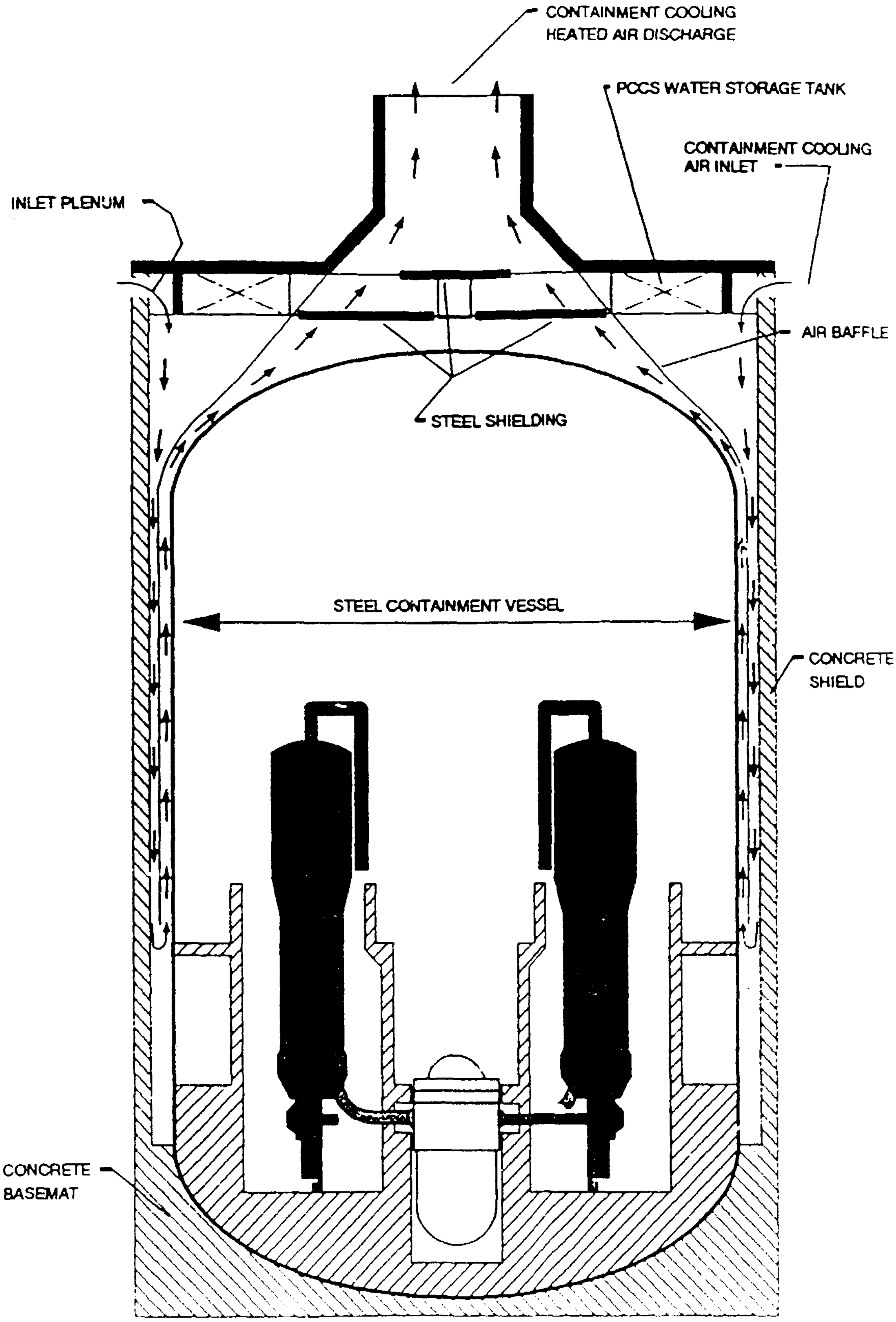


Figure ATP4
AC-600: Containment cooling.

ATB1

Format	**Description**
Title	SBWR (Simplified Boiling Water Reactor)
Application of the reactor	NPP
Reactor type	BWR
Power output	Medium power/600 MWe
Organization (name)	General Electric, US
Development status	Detailed design
Description	The design of the SBWR is the common result of the design team efforts involving organizations in Europe, Asia and North America. This BWR is designed to meet passive ALWR requirements, including no dependence on operator action for three days after a core damaging event.

The SBWR key features are the following:

- o An integral dryer-separator simplifies refuelling
- o A large reactor vessel water inventory increases margins and enhances safety
- o Standard failure-resistant fuel
- o A low power density core
- o Recirculation system eliminated (natural circulation)
- o Fine-motion control rod driver (CRD) for easy operation and maintenance

The power rating of the reactor is 1 800 MWt. The nuclear fuel is similar to that used in General Electric BWR designs, but due to the use of the natural circulation, the core height is reduced by about 30 %. The volumetric core power density decreased up to about 36.6 kW/l. Also in the SBWR the RPV is 2.6 m higher than that required for the 1 356 MWe ABWR (see AEB1). The natural circulation, compared with the forced circulation, means lowered fuel cycle costs, a reduced number of transients and increased thermal margins.

The safety features of the SBWR are a gravity-driven core cooling system that will keep the core covered and cooled in the event of a LOCA, an isolation condenser located in an elevated water tank, which allows for residual heat removal by natural circulation, and a containment of the pressure suppression type with a passive containment cooling system (PCCS). The isolation condensers and the PCCS passively transfer decay heat to the atmosphere. In a LOCA, the reactor vessel is depressurized by means of depressurization valves, the gravity driven cooling system floods the reactor and the isolation condenser removes the decay heat. There is no containment flooding for most LOCA events. The flooding of the lower drywell occurs mainly in response to severe accidents. There is a sufficient inventory in PCCS pools to handle at least 72 hours of decay heat removal.

The design team has also looked at a 1 000 MWe power level for SBWR. In the area of standardization and overall plant economics, the SBWR 1000 is likely to satisfy the utility needs for a simplified and economic plant in the coming decade (see ATB4).

Schedule

No SBWR plant has been ordered yet. An SBWR development programme FOAKE to obtain FDA by mid 1994 and FOAKE by the end of 1996 is well underway. Some large scale and full scale tests are still in progress.

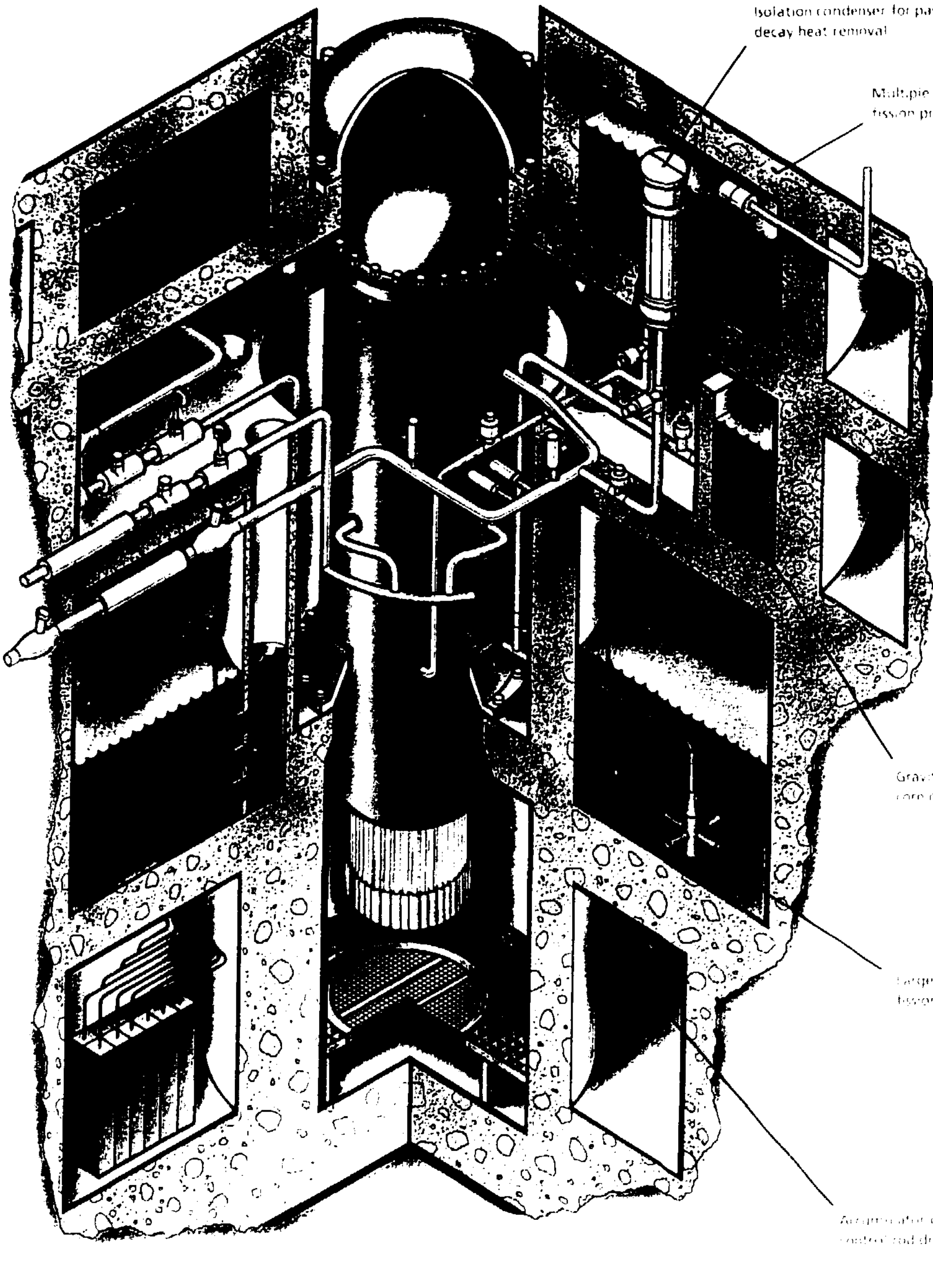
SBWR Unit Data

Electric power	600 MWe
Primary pressure	72 bar
Power density	36.6 kW/l
Reactor pressure vessel	
- Diameter	6 m
- Height	24.6 m
- Material	Steel
Refuelling time	~ 15 days
Reference fuel cycle length	24 months
Plant design life	60 years

References

Safety Aspects of Designs for Future Light Water Reactors (Evolutionary reactors)
Vienna, Austria, 1993, IAEA-TECDOC-712

Simplifying the BWR
Atom No. 430, 1993, Sept/Oct.



Figur ATB1
SBWR: Reactor vessel and internals.

ATB2

Format	Description
Title	HSBWR (Hitachi Small Boiling Water Reactor)
Application of the reactor	NPP
Reactor type	BWR
Power output	Medium power/600 MWe
Organization (name)	Hitachi, Japan
Development status	Basic design

Description

The HSBWR is a proposed natural circulation BWR plant, with a thermal power of 1 800 MW. The core power density is about 70 % of that of current BWRs because of natural circulation.

The main features of the HSBWR design are as follows:

- o Short fuel assembly (an active fuel length of 3.1 m) is selected to improve seismic resistance of the core.
- o A riser, 9 m in height, above the core and elimination of steam separators will increase the rate of natural circulation.
- o A low power density of 34.2 kW/l allows for the extension of the operating cycle by up to 2 years.
- o Simple reactor components and systems without forced recirculation systems and active emergency core cooling systems.
- o No core uncovery during any LOCA events in the reactor pressure vessel; coolant injected by the actuation of the steam-driven reactor core isolation cooling system, automatic depressurization system and accumulators.
- o A pool of water surrounds the metal primary containment and heat rejection from the containment to the pool by conduction through the containment wall removes decay heat for 72 hours without intervention.
- o Reactor pressure vessel bottom flooding by spillover of emergency core cooling system water.

- o Depressurization by an automatic depressurization system and borated water injection by the accumulator makeup plant shutdown during the anticipated transient without scram (ATWS).

- o Decreased investment costs by reducing the construction period up to 32 months.

Schedule

Conceptual design complete. The HSBWR is apparently still in the development stage. No HSBWR plant has been ordered yet.

HSBWR Unit Data

Electrical output	600 MWe
Rated pressure	70 bar
Core flow	178 000 t/h
Core length (total/active)	3.7/3.1 m
Number of fuel assemblies	708 (8 x 8 array)
Number of control rods	169
Core equivalent diameter	4.7 m
Power density	34.2 kW/l
Uranium enrichment	3.6 %
Average burnup exposure	39 MWd/kg
Recirculation system	Natural circulation
RPV diameter	6.3 m
RPV height	25 m
Steam separator	None free surface separator
Turbine type	TC 2F 40
Size of reactor building	47 m x 47 m x 47 m

References

Small and Medium Reactors, Technical Supplement
Nuclear Energy Agency, OECD, Paris, France, 1991

Safety aspects of designs for future light water reactors (evolutionary reactors)
IAEA-TECDOC-712, IAEA, Vienna, 1993

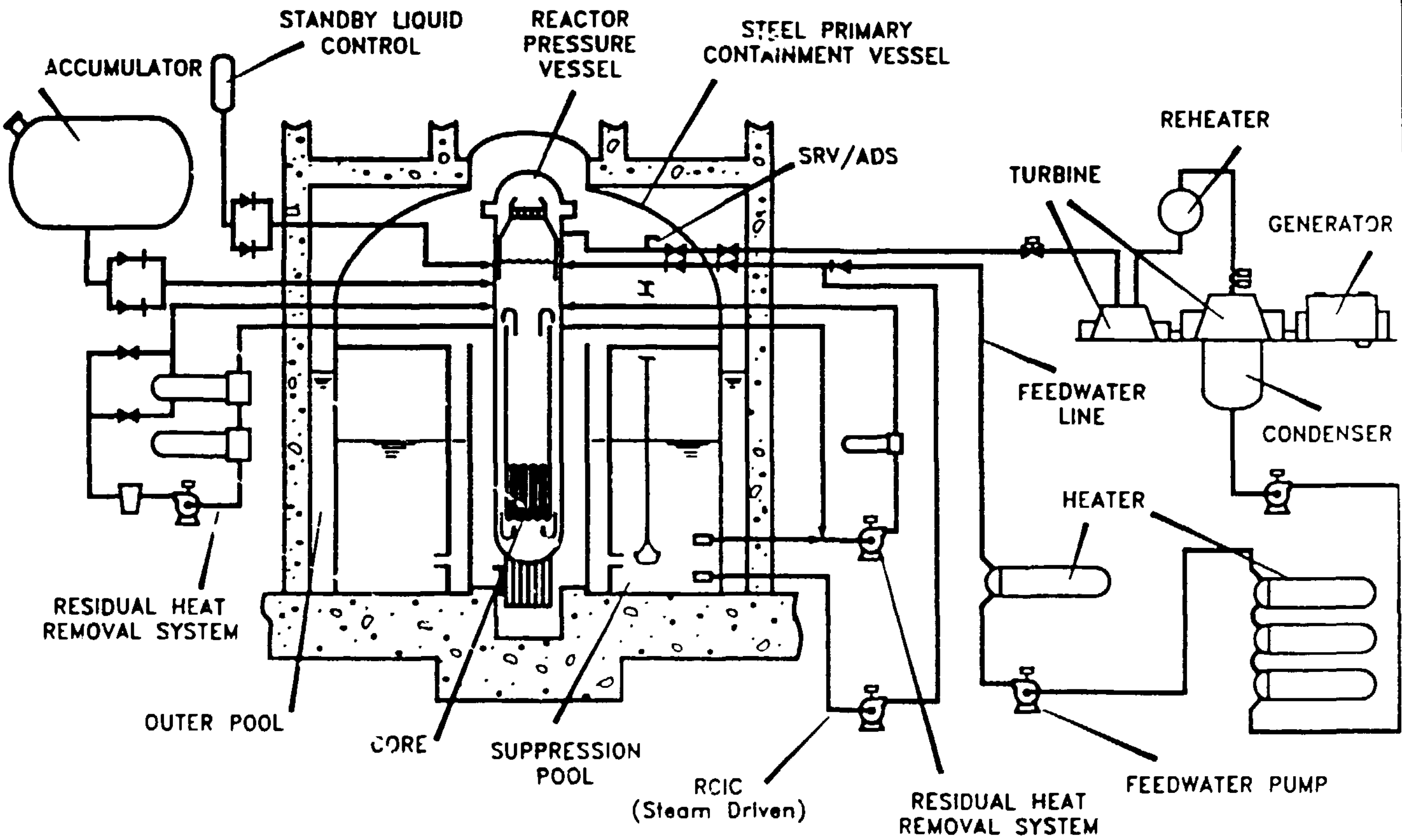


Figure ATB2
HSBWR: Schematic.

ATB3

Format	**Description**
Title	TOSBWR-900P (Toshiba Simplified BWR)
Application of the reactor	NPP
Reactor type	BWR
Power output	Medium power/310 MWe
Organization (name)	Toshiba, Japan
Development status	Basic design
Description	The TOSBWR-900P reactor plant is a proposed 900 MWt natural circulation BWR. The main number of features for the TOSBWR-900P design are similar to those of the GE SBWR. The primary design objectives are: to simplify the plant systems, to decrease investment costs, to reduce the construction period and use passive safety measures to improve the safety.

The main features of the TOSBWR-900P include:

- The reduced size of the reactor vessel due to shorter fuel assemblies and the elimination of steam separators and driers (steam drums)
- The use of hydro accumulators for water injections and an automatic depressurization system
- Decay heat removal provided by a gravity-driven cooling system and passive containment spray
- Top entry gravity-driven control rods made possible by the elimination of components above the core. This design provided the elimination of all penetrations in the RPV bottom
- Steel containment wall which can transfer heat from the flooded containment to an external pool which uses sea water as a heat sink
- Isolation condensers located at an elevated position in the suppression pool in the wetwell region in the primary containment.

Schedule

Conceptual design complete, ready for detailed design development. No TOSBWR-900P plant has been ordered yet.

TOSBWR-900P Unit Data

Thermal power	900 MWt
Electrical power	310 MWe
Type of fuel assemblies	8 x 8 array
Number of fuel assemblies	388
Effective core size (diameter/height)	3.4 m/2.5 m
Power density	40 kW/l
Size of pressure vessel (diameter/height)	4.7 m/17 m
Operating pressure	72.1 bar

References

Safety aspects of designs for future light water reactors (Evolutionary reactors)
Vienna, Austria, 1993, IAEA-TECDOC-712.

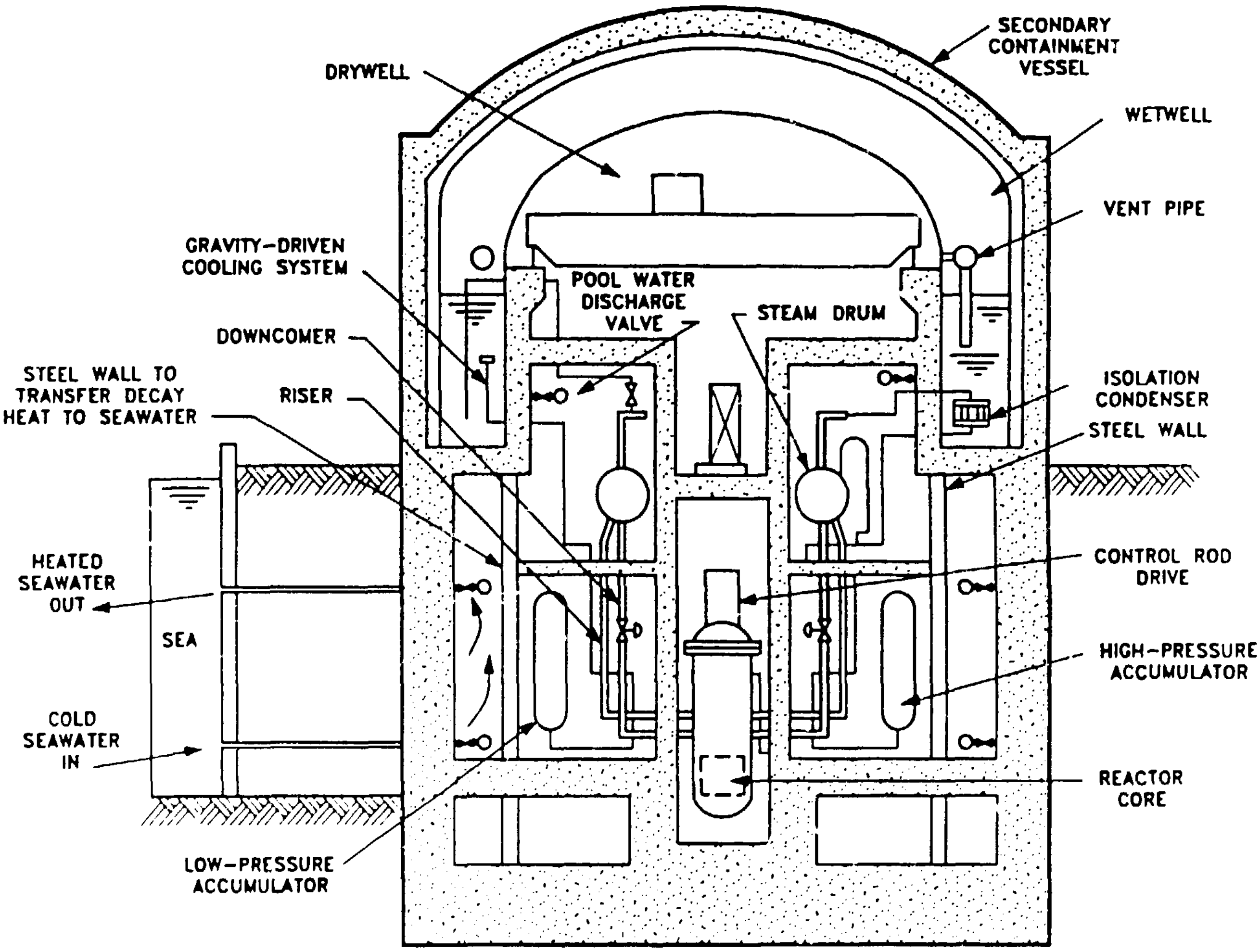


Figure ATB3
TOSBWR-900P: Reactor building.

ATB4

Format	Description
Title	1000 Natural Circulation BWR
Application of the Reactor	NPP
Reactor type	BWR
Power output	Medium power/1000 MWe
Organization (name)	General Electric, US
Development status	Studies
Description	The 1000 Natural Circulation BWR is based on the GE SBWR design with the 600 MWe power. The key differences are the power output and associated adjustment of equipment sizing. Both plants use similar passive safety features and a common approach for the reactor building arrangement and containment. The riser height and separator standpipe lengths were chosen to minimize RPV height and refueling time impacts when scaling up to 1 000 MWe output. Compared to SBWR, the 1 000 MWe reactor has a shorter riser above the core and longer separator standpipes. These differences permit a shorter span from normal water level to the second level isolation setpoint for the 1 000 MWe reactor, since the longer standpipes increase the in-vessel water inventory. The effective stack height which determines the natural circulation flow driving force is 0.5 m greater for the 1 000 MWe reactor. By providing a greater in-vessel water inventory while maintaining the effective stack height, the overall height of the two reactors differs by less than a meter. A higher steam flow is accomodated by increasing the number of steam lines from two in SBWR to four in the 1 000 MWe design. However, the steam line diameter is reduced from 1 000 mm to 600 mm. The number of safety relief valves is increased from 8 to 14 and the additional ones are placed on the additional two main steam lines. The total number of depressurization valves is increased from 6 to 8 (the additional ones are installed on each of the two additional steam lines). It was shown by calculations that from the viewpoints of natural circulation application and passive safety features, the SBWR design can be scaled up to a 1 000 MWe capacity.
Schedule	No 1 000 MWe Natural Circulation BWR has yet been ordered.

1000 Natural Circulation BWR Unit Data

Power		1 000 MWe
Reactor pressure vessel		
-	Diameter	7.1 m
-	Height	25.2 m
-	Material	Steel
Cycle length (effective full power days)		357
Maximum linear heat generation rate		39.3 kW/m

References

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1000 MWe Natural Circulation BWR Technology
Proceedings International Conference on Desigh and Safety of Advanced Nuclear Power Plants. Vol. II, October 25-29, 1992, Tokyo

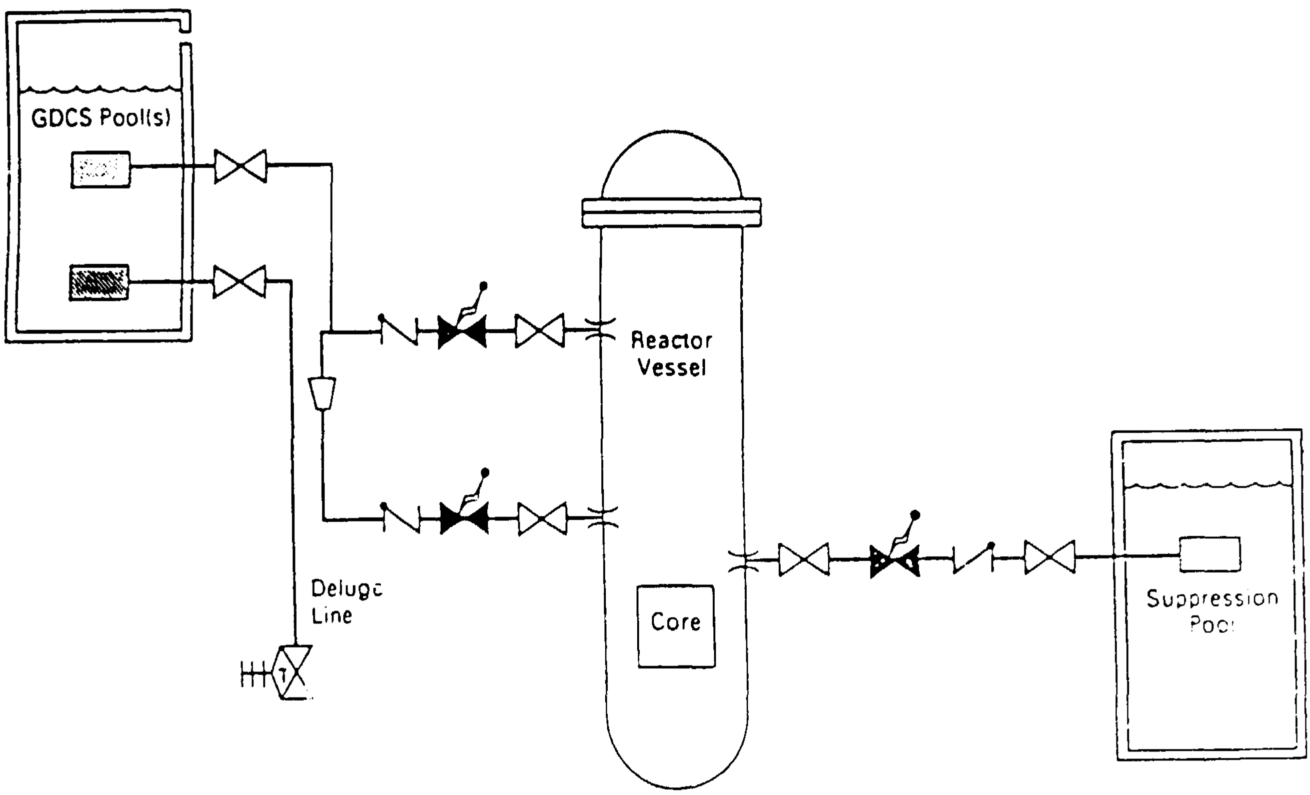

Figure ATB4
1000 Natural Circulation BWR: Gravity driven cooling system (GDCS).

ATB5

Format	**Description**
Title	SWR 1000
Application of the reactor	NPP
Reactor type	BWR
Power output	Medium power/1000 MWe
Organization (name)	Siemens-KWU, Germany
Development status	Basic design
Description	The SWR-1000 is an advanced, evolutionary-type BWR plant with the rated electrical output of 1 000 MWe. The main features of SWR-1000 safety concepts are:

o A higher degree of safety is achieved through the introduction of passive systems for accident prevention and control

o In the event of an accident only after several days of intervention by operating personnel is necessary

o A large RPV with a low core power density, large reactor coolant inventory and an increase in the water inventories stored inside and outside containment

The core design is based on an enlarged Type 12 x 12-16Q fuel assembly, which was designed by adding two additional fuel rods to each axis of a 10 x 10 array fuel assembly. This assembly is characterized by a lower average rod linear power density of 108 W/cm. The number of control assemblies in the core has been reduced by approximately 40 % to 157, and a number of detector assemblies by some 60 % to 18.

The passive safety systems of SWR-1000 are the following:

o Emergency condenser, removes decay heat from the reactor at the RPV drops, requiring neither electric power nor activation by the I&S system

o Containment cooling condensers remove heat due to steam formation inside the containment

- A gravity-driven core flooding system floods the reactor by gravity flow during RPV and pressure drops

The containment is a pressure suppression-type design, provided with a grace period for the initial 7 days (hypothetical loss of coolant core melt for which a 100 % Zr-H_2O reduction is postulated) and an inert atmosphere for prevention.

Schedule

No SWR 1000 plant has been ordered yet.

SWR 1000 Unit Data

Overall plant		
-	Thermal output	2 778 MWt
-	Net electric output	977 MWe
Reactor core		
-	Number of fuel assemblies	648
-	Total uranium weight	127 t
-	Active core height	2.8 m
-	Average power density	48 kW/l
-	Discharge burnup	65 GWd/t
-	Average enrichment	4.95 wt %
-	Reload fuel	13 t/a
-	Coolant flow rate	12 000 kg/s
Reactor pressure vessel		
-	Inside height	22.8 m
-	Inside diameter	7 m
-	Design pressure	88 bar
-	Number of recirculation pumps	6
Turbine		
-	Inlet flow	1 475 kg
-	Inlet pressure	67 bar
-	Number	1
-	Speed	3 000 rpm
-	Number of HP casings	1
-	Number of LP casings	3
Plant operation		
-	Spent fuel storage capacity	40 years
-	Plant design life	60 years
-	Plant construction period	48 months

References

SWR 1000 Technology of the future. Medium capacity boiling Water Reactor With Passive Safety Features.
Siemens AG - Bereich Energieerzeugung (KWU), Germany Aug 1994.

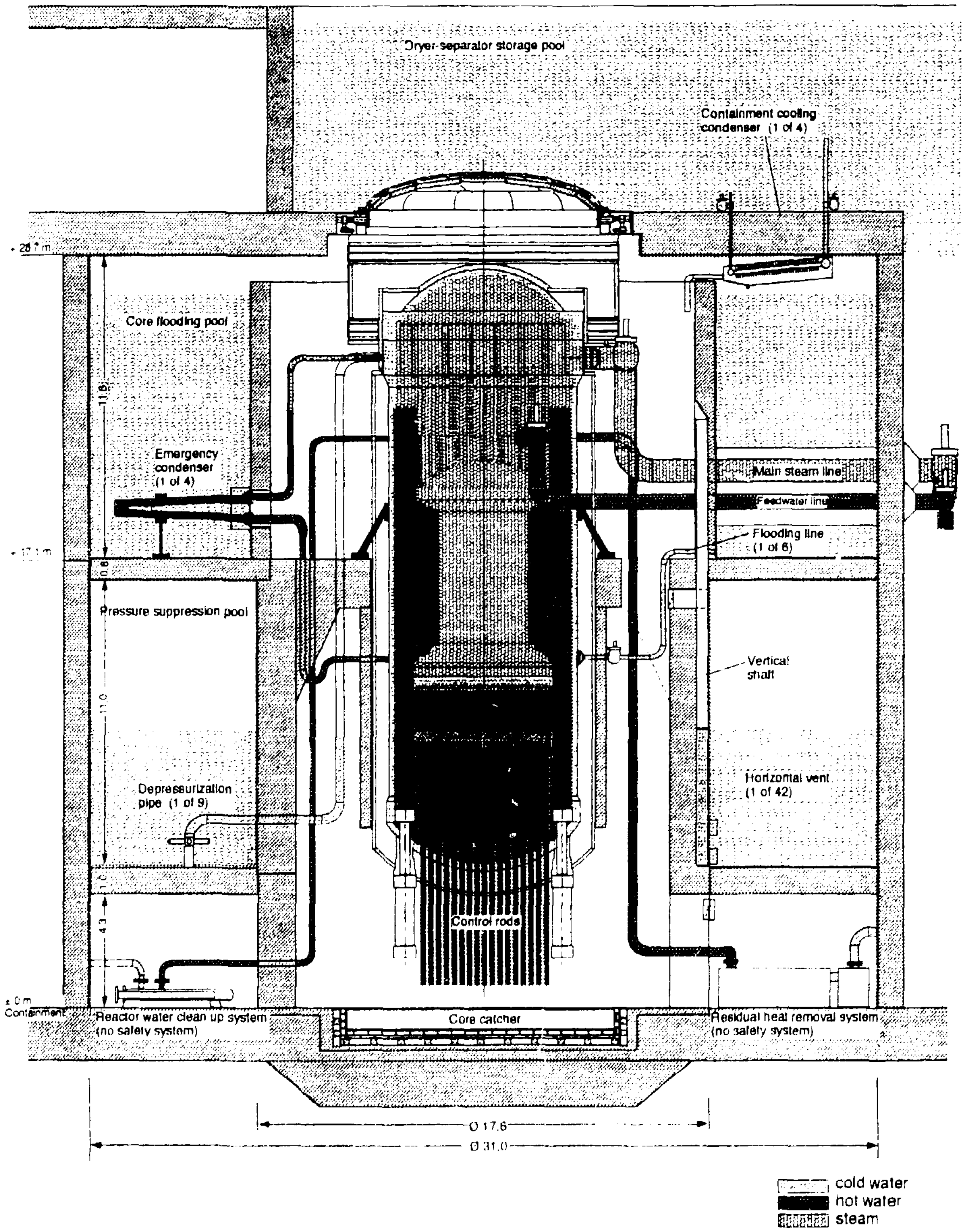

Figure ATB5
SWR 1000: Containment building.

ATG1

Format	Description
Title	MKER-800
Application of the Reactor	NPP
Reactor type	LWGR
Power output	Medium power/860 MWe
Organization (name)	RDIPE, RF
Development status	Studies
Description	With a net electrical power output of 860 MWe, the MKER-800 plant design represents the latest version of the post Chernobyl RBMK line. It makes use of the proven technology of current RBMK plants. In this graphite channel reactor with water cooling, the lessons learnt from operating RBMKs and from the Chernobyl disaster have been incorporated. MKER-800 reactor plant safety and reliability improvements include: o Reasonable reactivity characteristics of the core o Natural circulation providing heat removal up to 600 MWe with an additional application of the jet pumps at nominal power o Absence of any valves in the primary circuit o Increased efficiency of the emergency shut-down system o Decreased linear specific power to 28 kW/m o Containment with improved seismic stability o Passive cool-down system located inside the containment
Schedule	No MKER-800 plant has been ordered yet. Conceptual MKER-800 design complete; ready for detailed design development.

MKER-800 Unit Data

Reactor thermal power	2 450 MWt
Electric power	860 MWe
Core linear power	28 kW/m
Steam rate	4 600 t/h
Average burnup fuel	28 MWd/kg
Steam pressure	70 bar
Fuel enrichment	2.4 %
Temperature of the feeding water	200 °C
Thermal load	250 Gcal/h
Electric power in the regime without heating	860 MWe
Electricity consumption for needs of the plant	4.5 %
Power unit efficiency/net	33.5 %

References

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Book of Abstracts. Nuclear Power & Industry International Conference, 27 June - 1 July, 1994, Obninsk, RF.

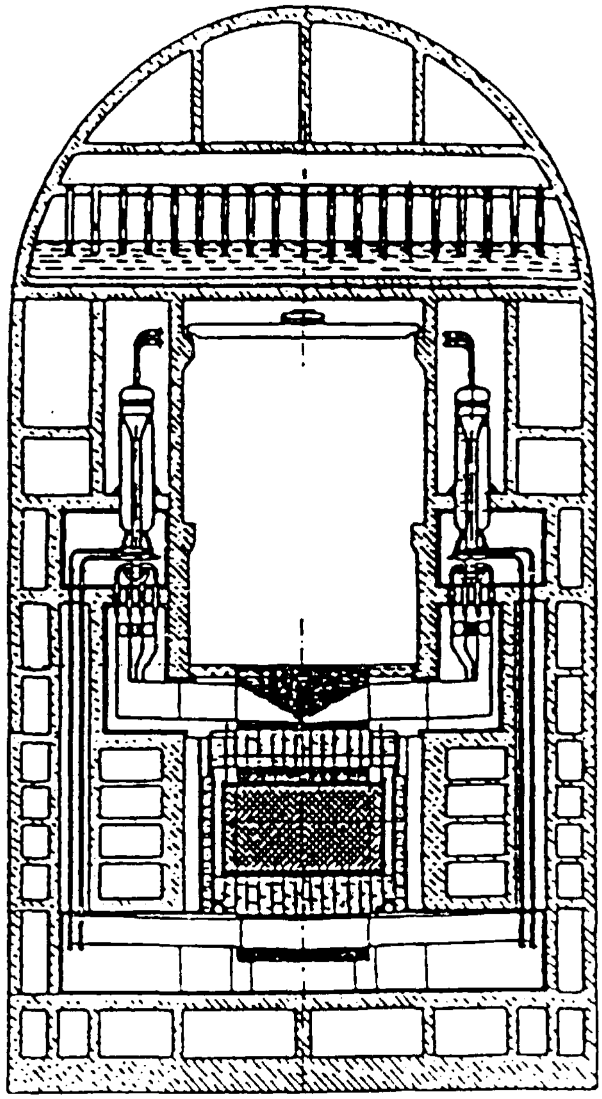

Figure ATG1
MKER-800: Reactor building.

ATH1

Format	Description
Title	CANDU 3
Application of the reactor	NPP
Reactor type	HWR
Power output	Medium power/450 MWe
Organization (name)	AECL (Atomic Energy of Canada LTD), Canada
Development status	Detailed design

Description

With a net electrical power output of 450 MW the CANDU 3 represents the smallest version of the CANDU-reactor line. It makes full use of the proven technology of the established 665 MW mid-size CANDU 6 being updated with relevant features resulting from the ongoing Canadian research and development programme.

The CANDU 3 is fuelled by natural uranium and permits on-power refuelling as well as low excess reactivity. The fuel bundles are situated on Zr-2.5Nb horizontal pressure tubes being surrounded by a Zircaloy calandria tube with CO_2 as the thermal isolator in the annulus. Except for the pressure tubes all other components of the reactor assembly operate under low temperature and low stress. The fuel bundles are cooled by pressurized heavy water and the secondary heat transport system consists of one loop with two steam generators and two pumps. The arrangement allows for natural coolant circulation in the event of loss of power in the heat transport pumps.

The additional cooling circuit for the heavy water moderator can also be used for decay heat removal in the unlikely event of LOCA with failure of the light water emergency core cooling system.

The main design goals aim at:

- o Enhanced safety and a high lifetime capacity factor
- o Low occupational exposure (50 per cent reduction from CANDU 6 to the reference design)
- o Maximization of the components/plant life (up to about 100 years)
- o Easy replacement of any component

- Short and manageable construction schedule (30 months)

CANDU 3 incorporates many safety enhancements, including:

- Simplified and increased containment design pressure eliminating the need for dedicated pressure-suppression systems
- Increased pressurizer size and reduc... ...eed condenser size to maintain the heat transport syste... ...ntory over a wider range of upset conditions
- Improved moderator cir... ...aon - to increase cooling capability following a LOCA ar... ...oss of emergency coolant injection
- Improved man-... ...ine communication, simplifying central operations and ...ger operator action times (new control room)
- Severe accident mitigation

Schedule

The CANDU 3 plant detailed design is under way and is scheduled for completion in 1996. No CANDU 3 plant has been ordered yet.

CANDU 3 Unit Data

Power capacity		1 439 MWt
Number of fuel channels		232
Fuel		
-	Fuel	Natural UO_2
-	Form	Fuel bundle assembly of 37 elements
-	Length of bundle	495 mm
-	Outside diameter	102.4 mm
-	Bundle weight	23.5 kg (18.4 kg U)
-	Bundles per fuel channel	12
Heat transport system		
-	Number of steam generators	2
-	Steam generator type	Vertical U-tube
-	Number of heat transport pumps	2
-	Heat transport pump type	Vertical, centrifugal, single section
-	Reactor outlet header pressure	99 bar
-	Reactor outlet temperature	310 °C
-	Reactor coolant flow	5.3 Mg/s
-	Steam temperature (nominal)	260 °C
-	Steam pressure	46 bar

References

Small and medium reactors, v 2 Technical supplement, Nuclear energy agency, OECD, Paris, France, 1991.

The AECL Candu 3 reactor. Nuclear News, September, 1992, p 80-81.

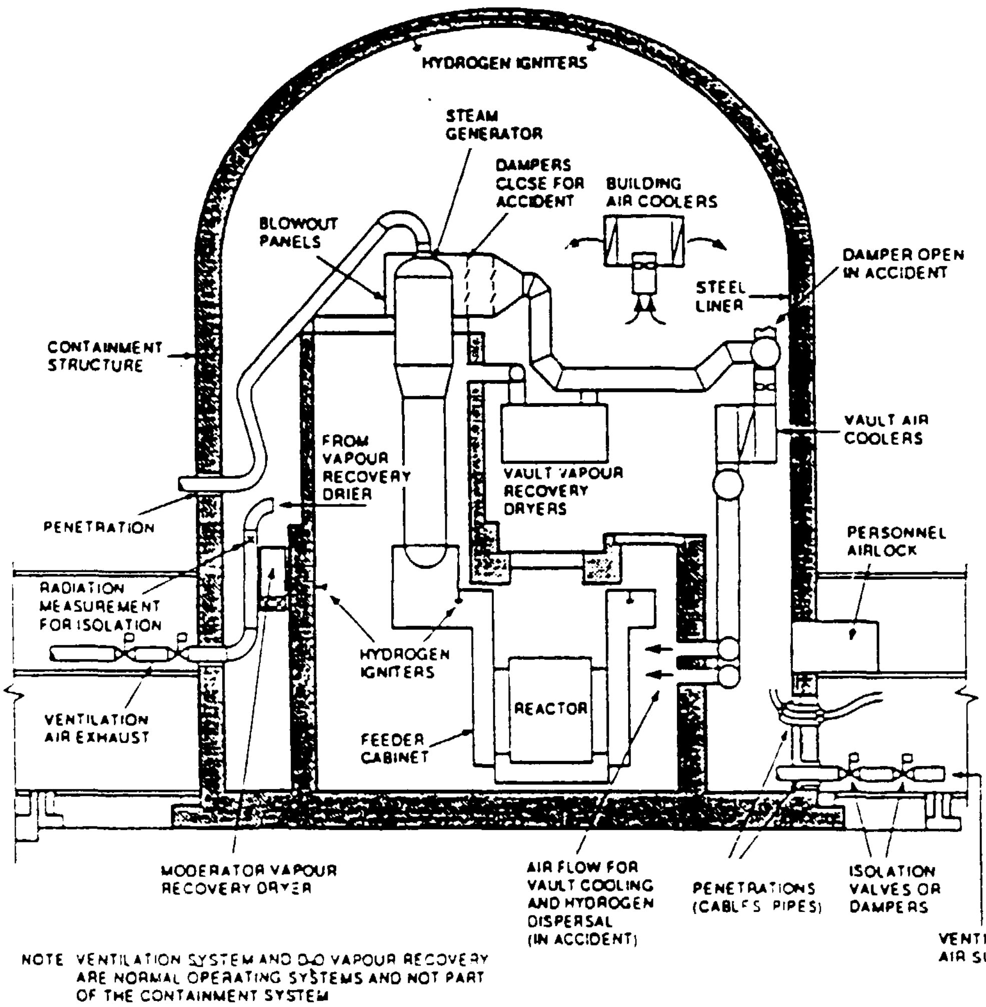

Figure ATH1
CANDU 3: Containment system.

ww victor/ATH1 ea

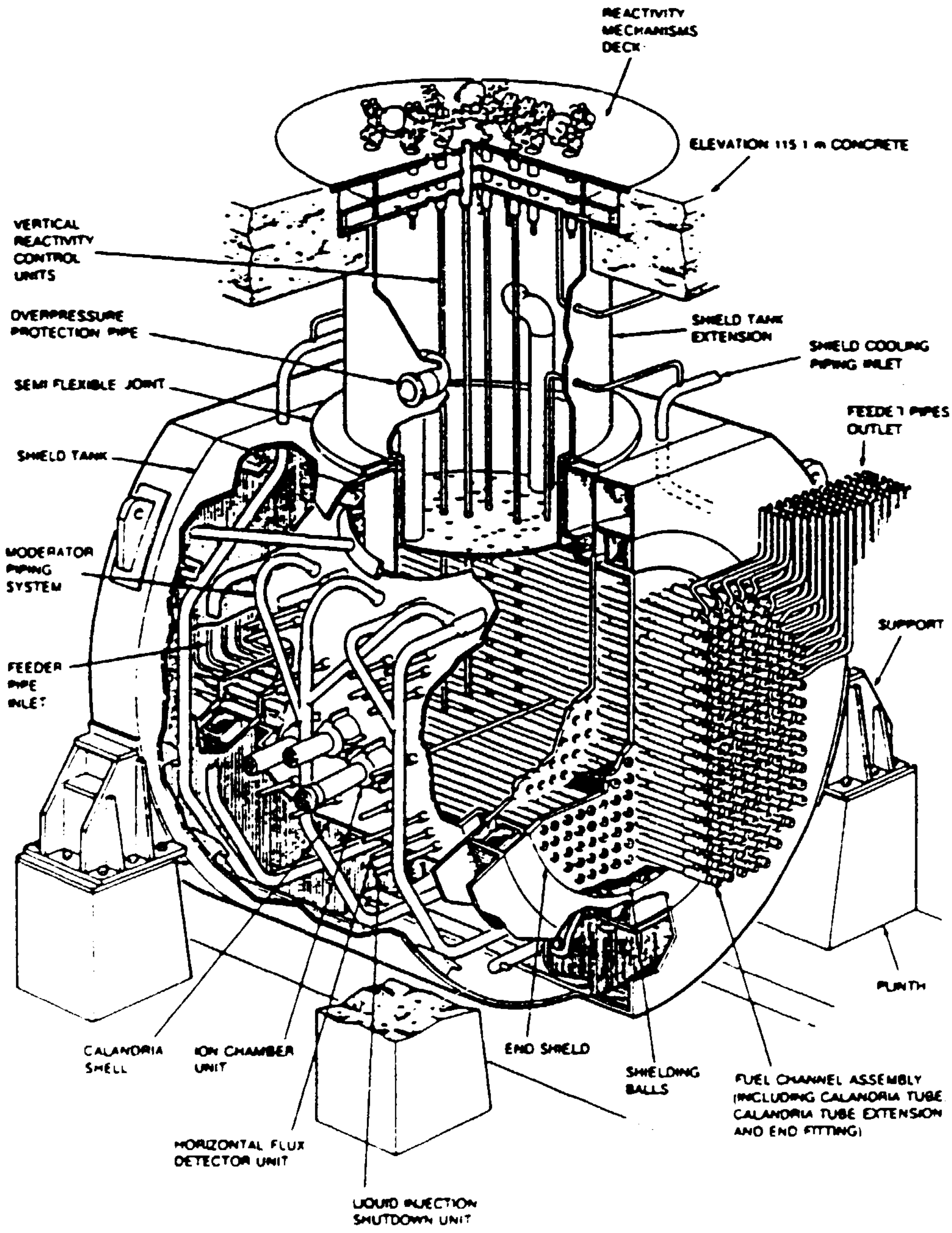

Figure ATH1
CANDU 3: Calandria/Reactor Assembly.

AI Innovative Designs

AI0 Introduction

The designers of these LWRs (600 MWe or less) concentrate their efforts primarily towards achieving a reliable termination of the fission reaction in accident situations using passive means and inherent reactor properties and towards assurance of reliable passive cooling in the shut down reactor. Some innovative designs incorporate significant departures from present-day designs in reactivity control, reactor coolant system configuration and other systems.

If the reactor vessel integrity is not lost, a reliable way of terminating the fission reaction is core flooding with high concentrated boron water. For a more reliable core flooding under accidental conditions different methods are proposed in Sweden, Japan, Russia, Italy and the USA:

- Integrated arrangement of the primary circuit in a reactor vessel of prestressed concrete;
- Integrated arrangement of the primary circuit with a reactor vessel provided with a guard vessel;
- Accommodation of the whole primary circuit equipment in a system of boxes of prestressed concrete, filled with "cold" water.

For reliable and rapid core flooding with "cold" water with a high boron concentration, the PIUS-principle was developed in Sweden. This concept is used as a basis for some later designs in Japan both for innovative and current PWR (PSSS - passive safety shutdown system) and in the USA for BWRs and steam cooled fast reactors.

The grace period is practically inversely proportional to the nominal thermal power and directly proportional to the water inventory in the passive cool-down systems. To increase the self-protection time (grace period) in the PIUS-principle concepts RPVs with a water inventory of 0.3 - 1.5 m^3/MWt are used, which permits a grace period from 1 up to 7 days, respectively. The decay heat is removed by natural circulation.

Since the innovative designs deviate more than evolutionary ones from present-day technology, a prototype or demonstration plant is necessary for testing and validation.

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Advanced LWRs: analysis of new approaches and ideas.
Nuclear Society International, Moscow, RF, 1993.

o FORSBERG, C W et al
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Report ORNL-6554, 1989.

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Current Trends in Safety Philosophy and Design Goals for Advanced Nuclear Power Plants Designs.
Proceedings ASME/ISME Nuclear Engineering Conference, Vol 2, 1993, ASME, p 539-433.

AIP1

Format	Description
Title	PIUS-600 (Process Inherent Ultimate Safety)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/600 MWe
Organization (name)	ABB Atom, Sweden
Development status	Basic design
Description	The thermal power of the PIUS-600 PWR type reactor is 2 000 MWt. The core consists of 18 x 18 array fuel assemblies, with a reduced height to limit the hydraulic pressure drop across the core during operation. The core is housed at the bottom of a double-walled flow guide structure which is provided with a wet metallic thermal insulation of the outer surface. This structure is installed in a pool of highly borated water. The reactor pool, enclosed in a prestressed concrete reactor vessel (PCRV). The main design parameters (the core power density, reactor pressure and temperature) are red·r ·d in comparison to current PWRs.

" ·o ur SGs and the four reactor coolant pumps are located outside the · ·V. The PCRV is installed inside a primary containment of the pressure-r· ·o' ession type which, in turn, is largely enclosed in a reactor building.

[r ing normal operation, and during minor disturbances, normal control ·stems keep the plant in operation. With drastic deviations from nominal onditions safety systems will try to restore acceptable conditions or initiate ·actor shutdowns. PIUS is provided with a scram valve system which ·eparates the borated water pool from the primary loop. If the scram system vere to fail, a self-protective mechanism would take over, the pressure balance in the density locks would be disrupted and pool water would enter ·ie primary loop.

The main differences compared with traditional PWR designs are:

- o The PCRV
- o The use of lower and upper density locks towards the PCR
- o The siphon breakers and wet thermal insulation

- The implementation of a passive residual heat removal system with a protected period of 7 days

- Reactivity control without control rods.

Schedule

No PIUS plant has been ordered. Significant verification testing and design work remain to be done, before a commercial application. This design will require a prototype demonstration before it can be artified.

PIUS-600 Unit Data

Electric power		600 MWe
Fuel assembly		
-	Number of fuel rods	312
-	Cladding material	Zircaloy 4
-	Mean power density in fuel	24.8 kW/kg
Reactor core		
-	Number of fuel assemblies	213
-	Active height	2.50 m
-	Equivalent diameter	3.75 m
-	Enrichment (first core/reload)	2.0 %/3.5 %
-	Average fuel burnup (equilibrium core)	45.5 MWd/kg
-	Total weight of U	80 t
-	Volume core power density	72 kW/l
Reactor coolant system (operation conditions)		
-	Pressure at vessel inlet	95 bar
-	Pressure at vessel outlet	93 bar
-	Temperature at pressure inlet	260 ^{O}C
-	Temperature at vessel outlet	289.3 ^{O}C
Prestressed Concrete Reactor Vessel		
-	Overall height (without upper part)	44 m
-	Overall width	26.8 m
-	Inside diameter	12.2 m
-	Total weight	63 000 t
-	Design pressure	105 bar
Reactor coolant pump type		Variable speed, wet
Steam generator type		Vertical once through

References

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Basic Information on Design Features of PIUS.
Review of Advanced Light Water Reactor Design Approaches
10-13 May, 1994, Moscow, RF, IAEA-TC-879.

PEDERSEN, T
A New Generation Nuclear Power Plants.
Proceedings ASME/JSME Nuclear Engineering Conference (ICONE-2), San Francisco, CA, 1993, March, Vol 2, p 627-631.

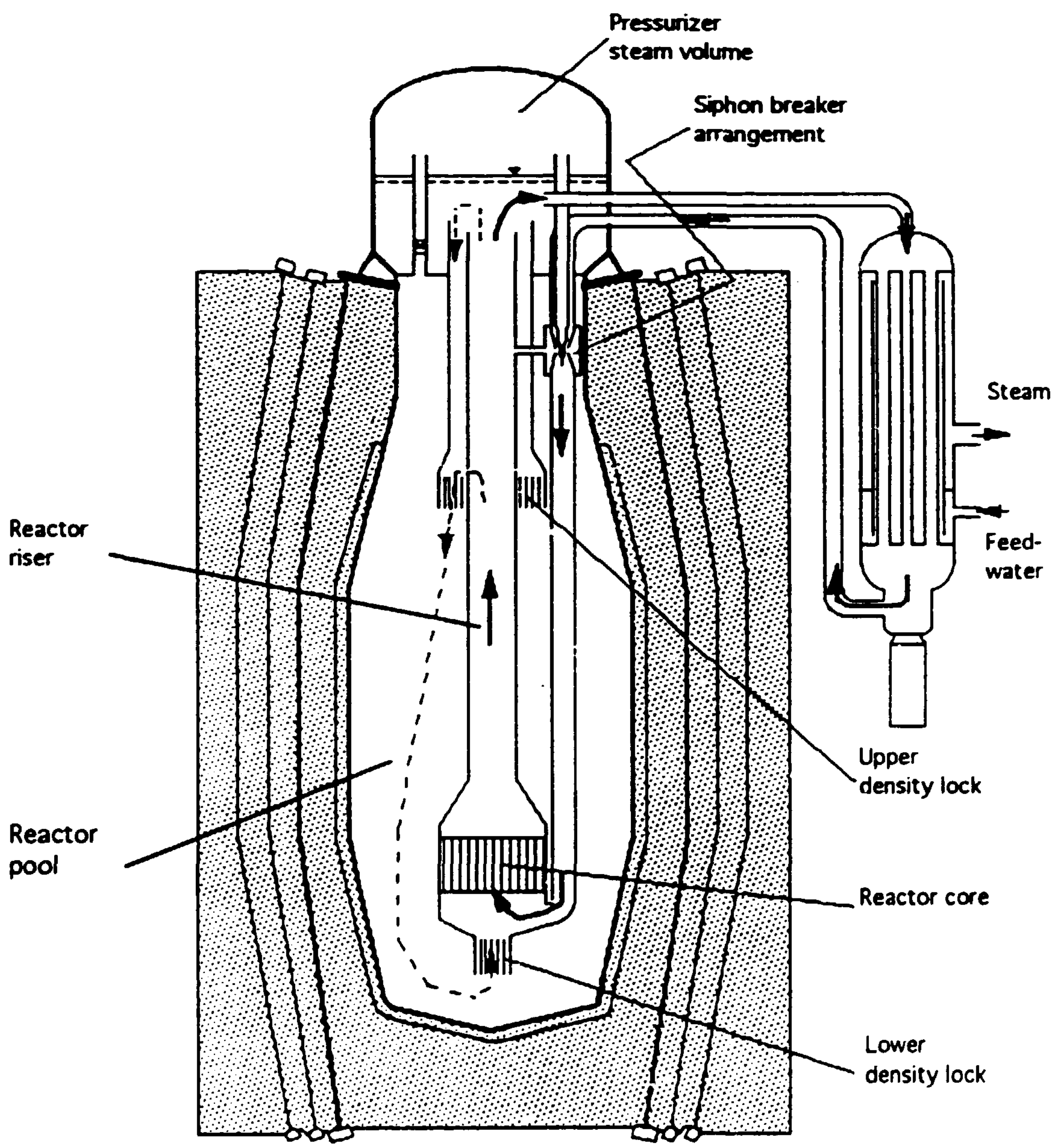

Figure AIP1
PIUS-600: Principle arrangement.

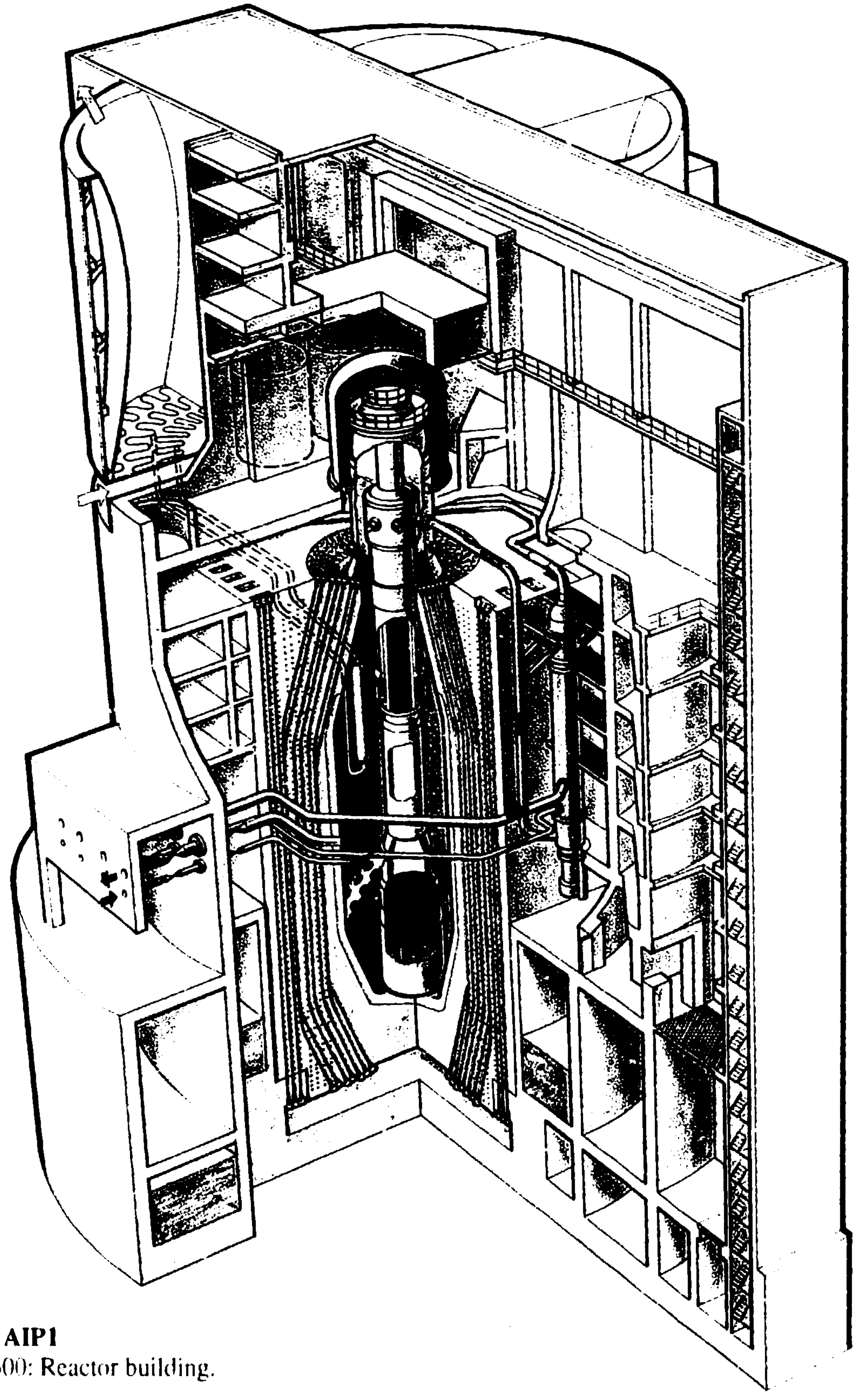

Figure AIP1
PIUS-600: Reactor building.

AIP2

Format	Description
Title	ISER (Inherently Safe and Economical Reactor)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/200 MWe
Organization (name)	JAERI, Japan
Development	Studies
Description	The ISER with a reactor thermal power of 645 MWt is a modified PIUS PWR type version enclosed in a steel vessel. The use of a steel vessel yields certain advantages, since the operating pressure of the primary system can be raised to 16 MPa compared with 9.5 MPa for PIUS-600. The borated water inventory sufficient for a grace period of 24 hours. The core utilizes 16 x 16 array fuel assemblies 2 m in height, and the power density is 61.3 kW/1. Four RCPs pump coolant flows down to the core inlet through the steam generator. The RCP motors are mounted on the outside of the steel vessel head and long pump shafts extend to the impeller on top of the SG. In the ISER, the temperature of the inner pool with borated water is kept at about 100 ^{o}C. The ISER has an innovative scram system, similar to the scram density lock valves system in PIUS. No correspondence to the long-term passive RHR system with heat dissipation was proposed. The normal operation performance and safety responses are rather similar to those of PIUS, with a few differences due to the integration of the RCS and the SG into the steel vessel that is submerged in a borated water pool. Obviously, this eliminates some accident situations and affects the decay heat removal in others.
Schedule	No ISER plant has been ordered yet. A conceptual design has been prepared, ready for detailed design development.

ISER Unit Data

Electrical power	200 MWe
Coolant pressure	155 bar
Coolant temperature	
- at the core inlet	289 ^{o}C
- at the core outlet	323 ^{o}C
Boric water temperature	100 ^{o}C
Steam pressure in SG	57 bar
Average core power density	61.3 kW/l
Reactor vessel dimension	
- height	26.4 m
- inner diameter	6 m
Fuel assembly array	16 x 16

References

Safety Aspects of Designs for future Light Water Reactors (Evolutionary reactors)
IAEA, Vienna, Austria, 1993, IAEA-TECDOC-712.

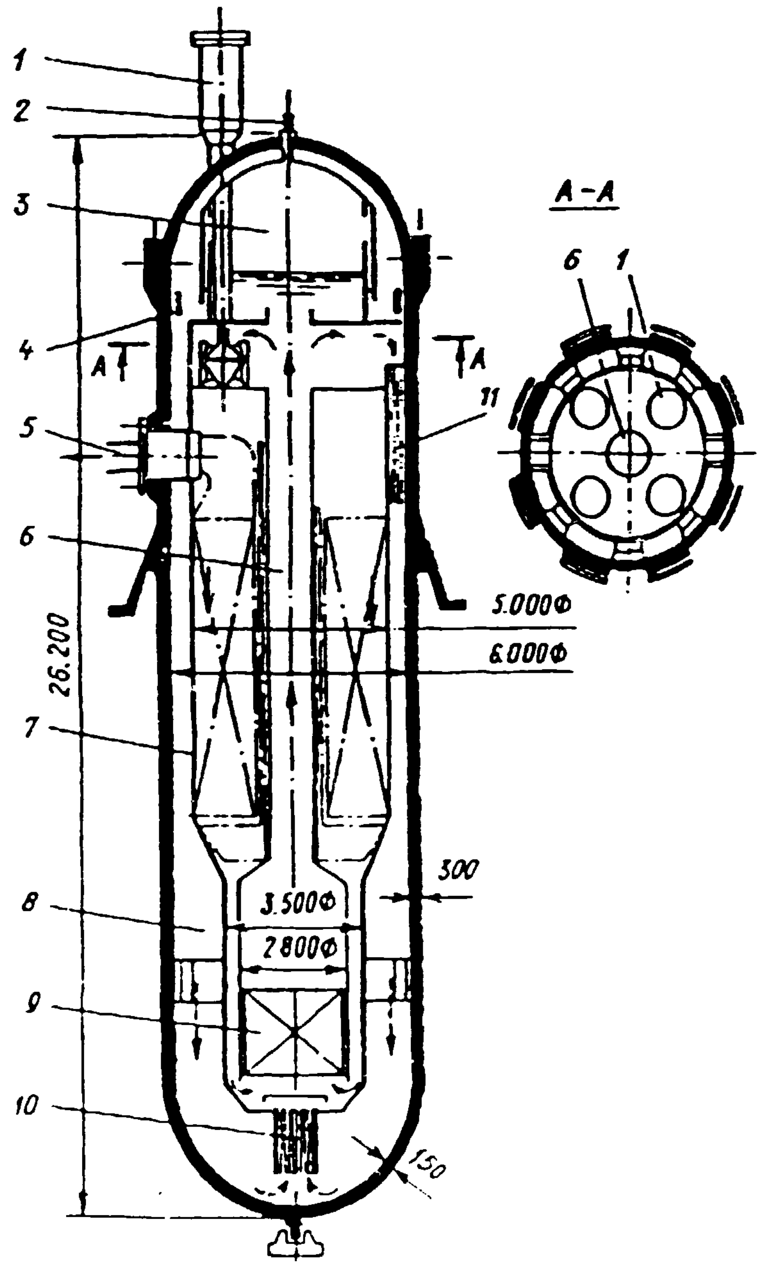

Figure AIP2
ISER: Reactor plant.

1 RCP. 2 Pressure relief valve. 3 Pressurizer. 4 Heat exchanger of borated water pool cooling system. 5 Steam and feed water pipes. 6 Riser section. 7 Steam generator. 8 Pool of borated water at 100 °C. 9 Core. 10, 11 Upper and lower density locks.

AIP3

Format	Description
Title	ISIS (Inherently Safe Immersed System)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/200 MWe
Organization (name)	Ansaldo Spa, Italy
Development status	Studies
Description	The ISIS is a reactor concept of the integrated type PWR using the PIUS principle. Reactor coolant system with the reactor coolant pumps and the steam generator is housed in a steel reactor pressure vessel to which steam and feedwater pipes are connected. The RPV contains the intermediate pool with the highly borated water. A reactor tank, provided with wet metallic insulation, separates the circulating low-boron content primary water coolant from the surrounding highly borated water in the intermediate pool. The primary loop and the intermediate pool are hydraulically connected at the bottom and top of the tank by means of open-ended tube bundles, the lower and upper density locks (see AIP1: PIUS-principle).

The ISIS reactor plant consists of 3 modules of 630 MWt each. The core consists of 17 x 17 arrays of reduced length fuel assemblies. The SG is made of Inconel pipes arranged in two helicoidal bundles, outside the riser annulus. Steam is generated inside the pipes, while primary water flows transversally on the outside. The two reactor coolant pumps are located at the top of the SG. Motors of these pumps are cooled by highly borated water.

The pressurizer which is installed outside the RPV consists of a hot upper part with thermal insulation, containing saturated water and steam, and a cold lower part containing water at the same conditions as in the intermediate pool. The cold and hot parts are hydraulically connected by a number of vertical pipes to provide mixing in case of surge flows. One pipe connects directly to the steam dome, acting as a spray in certain situations. The pressurizer and the RPV are hydraulically connected by means of two pipes, one to the bottom and one to the top of the intermediate pool. This arrangement improves the natural circulation.

The module is located in a cavity in a strong concrete structure, enclosed in a reactor building. The flange of the RPV head is sealed off against an opening in the shaft of this structure, and the volume above it is filled with water, creating a pretty large reactor or plant pool, that will serve as an emergency heat sink. The upper, hot part of the pressurizer is also located in this pool. Two passive pool cooling loops are connected to air coolers.

The ISIS concept features deal with the absence of:

- o The control rods and their driving system
- o The safety grade diesel generator set
- o The safety grade active decay heat removal systems
- o The safety injection systems

The ISIS NPP power plant application can be the following: Electricity generation and district heating or water desalination.

Schedule

No ISIS plant has been ordered yet. Significant amounts of design work and verification testing remain to be done.

ISIS Unit Data

Electrical output	205 MWe
Core and reactivity control	
- Fuel materiai	UO_2
- Fuel inventory	24.3 t
- Average core power density	26.7 kW/kg U
- Average discharge burnup	38 MWd/kg
- Initial enrichment	2.0 - 3.0 %
- Reload enrichment at equilibrium	3.5 %
- Refuelling frequency	18 months
- Active core height/diameter	2.92 m/2.0 m
- Number of fuel assemblies	69
- Number of fuel rods per assembly	264
- Clad material	Zircaloy 4
Reactor coolant system	
- Coolant pressure	140 bar
- Core temperature inlet/outlet	271 ^{o}C/310 ^{o}C
Reactor pressure vessel	
- Overall length/inside diameter	26.5 m/4.9 m
- Vessel material (lining)	Carbon steel (SS)
- Gross weight	782 t

Steam generator

-	Type	Helical tube
-	Configuration	Vertical
-	Feed water pressure/temperature	52.6 bar/120 °C
-	Steam pressure/temperature	46 bar/290 °C

References

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Obninsk, Russia, 16-20 May, 1994, IAEA-622-I3-TC-633.17.

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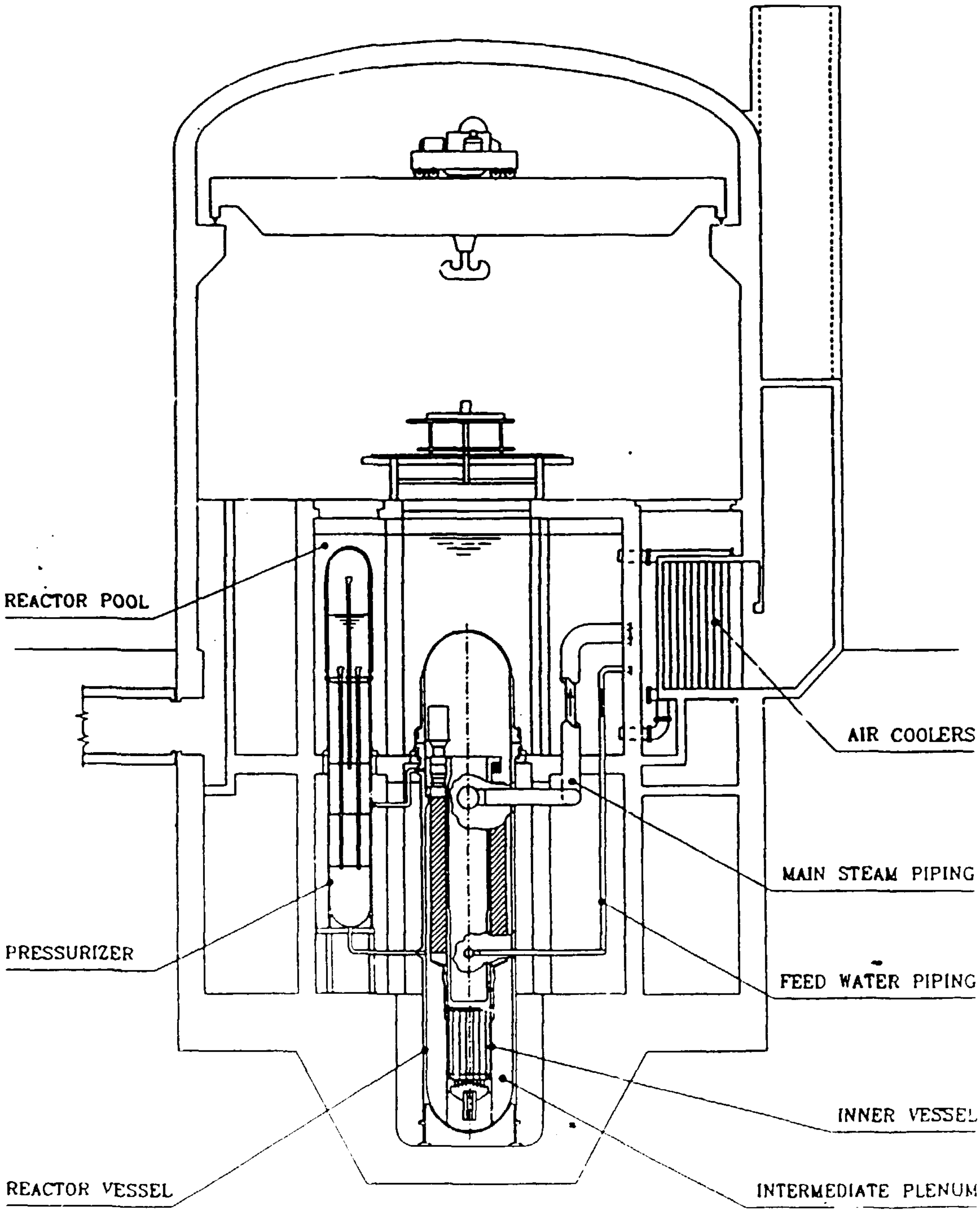


Figure AIP3
ISIS: Reactor module.

AIP-4

Format	**Description**
Title	MAP (Minimum Attention Plant)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/300 MWe
Organization (name)	Combustion Engineering, US
Development status	Basic design
Description	The MAP design is a PWR with an integrated arrangement of the primary circuit, which is rated at 300 MWe (the 900 MWe concept was also developed). The safe MAP operation is ensured by a nearly triple reduction of the average core power density compared with that of the existing PWRs. The heat is removed from the core to the once-through SG by natural circulation. The fuel assembly consists of four bundles of standard PWR fuel rods (arranged within a square 9 x 9 array) divided by a water channel, where absorbing elements can be inserted. The fresh fuel assembly contains 32 burnable poison rods made of Gd_2O_3 in natural UO_2. These rods are permanently installed in the fuel assembly. After the first irradiation cycle sixteen burnable poisons of Gd_2O_3 in the low-neutron-absorbing diluent (Al_2O_3 or ZrO_2) are inserted into the fuel assembly, which also stay in the fuel assembly until the end of the fuel residence time. The reactivity margin ensured by burnable poisons is 0.08 $\Delta K/K$. The reactivity effects resulting from power variations and the fuel burnup are compensated for by changes in the coolant density. For this purpose, the outlet coolant temperature is kept equal to the saturation temperature under all operating conditions. The coolant density is controlled by changing the saturation pressure in the primary circuit, with the upper plenum of the reactor vessel serving as a pressurizer. The MAP design envisages that the following events would occur without operator action or the reactor shutdown: absorbing rod drop, a feed water pump trip, switch-off of a steam generator module, switch-off of the condenser.

The residual heat is removed through the secondary circuit. The reactor vessel size is such that at the chosen nominal thermal power of 300 MWe, in the event of the small leak accident accompanied by a loss of heat sink and the emergency water supply, the core uncovery begins in 2.6 - 6 hours depending on the initial coolant volume.

The major departures of the MAP from the traditional PWR design are:

- o Incorporation of all primary system components, including the steam generators, within a single pressure vessel
- o Small vessel penetrations which are limited to the feedwater inlet, steam outlet and control element drive mechanisms (on the top of the vessel)
- o Intrinsic reactor control to compensate for both the long-term effects of fuel depletion and short-term power manoeuvring requirements, without any requirement for soluble boron or regulation control rods. The strongly negative moderator density coefficient of reactivity allows for power self-regulation; burnable absorber rods provide for long-term reactivity control
- o Control rods insertion is required for few hypothetical accidents

Schedule

No MAP plant has been ordered yet. No current development work.

MAP Unit Data

Thermal power	900 MWth
RPV diameter	5.5 m
RPV height	24.7 m
RPV wall thickness	292 mm
Fuel assembly array	Square 9 x 9
Fuel material	UO_2
Maximum fuel heating rating	170 W/cm

References

Safety aspects of designs for future light water reactors (evolutionary reactors).
IAEA-TECDOC-712, 1993, Vienna, Austria.

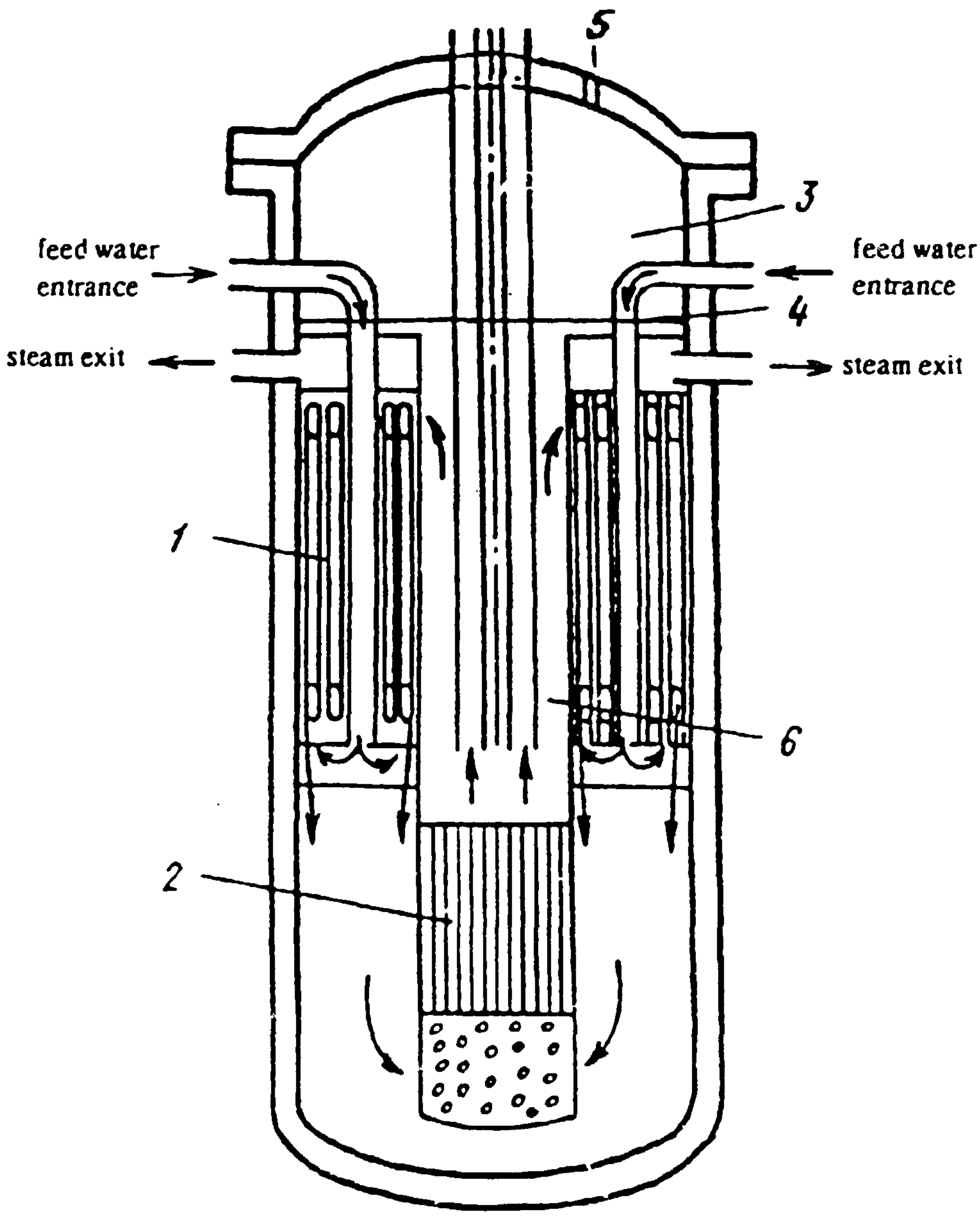


Figure AIP4
MAP: Reactor block diagram.

1 Steam generator module, 2 Core, 3 Pressurizer, 4 Water level in the primary circuit, 5 Line of pressure relief and gas blow, 6 CSS drive guides

AIP5

Format	Description
Title	VPBER-600
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/600 MWe
Organization (name)	OKBM (Experimental machinery design bureau), RF
Development status	Basic design

Description

The VPBER-600 reactor plant with a net electric output 600 MWe is a reactor with an integral arrangement of the primary circuit. Steam generators, pressurizer, main circulation pumps, heat exchangers for the ECCS are located in one vessel of about 6 m diameter. An additional "passive" barrier is provided, in the form of the stainless steel guard vessel, enclosing the reactor and primary circuit systems. The reactor plant development is based on the creation and successful operation of ice-breaker plants as well as AST-500 reactor plant experience.

The VPBER-600 arrangement has the following advantages:

- o Lower probability of primary circuit depressurization
- o No "large leak" accidents
- o Elimination of rapid dryout of the core
- o Reduced effect of the failed steam generator on the primary circuit
- o Decrease of neutron fluence
- o Reactor vessel is not subjected to cold water flooding
- o Hydroaccumulators, coolant recirculation could be excluded

The core heat removal is achieved with forced coolant circulation by six canned circulating pumps which are installed in the reactor vessel bottom.

In the circular gap between the reactor vessel and in-vessel well at the level above the core, the steam generator is arranged. The steam generator is a once-through device consisting of 12 independent sections, each consisting of 18 subsections.

Cooling the reactor down in the emergency is achieved by built-in heat exchanger-condensers through an intermediate circuit and by the evaporation of water from two tanks with natural coolant circulation in all heat transfer loops for not less than 3 days. The reactor depressurization system is intended in the event of the RPV integrity loss.

Schedule

No VPBER-600 plant has been ordered yet. There is significant design work to be done, before commercial application. No current development work is being carried out. Today the design team is looking at a large power level for loop type APWR.

VPBER-600 Unit Data

Reactor thermal power	1 800 MWt
Primary circuit circulation	Forced
Primary circuit coolant parameters	
- Pressure	157 bar
- Core inlet temperature	294.5 °C
- Core outlet temperature	325 °C
Core power density	69 kW/l
Secondary circuit parameters	
- Steam output	3 350 t/h
- Superheated steam pressure	65 bar
- Superheated steam temperature	305 °C
Power variation range	30-100 % Nnom
Lifetime	60 years
Maximum design earthquake, on MSK-64 scale	8

References

MITENKOV, F M
VPBER-600 enhanced safety water cooled medium power reactor.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, v 1, p 4.6.

MITENKOV, F M
VPBER-600 Conceptual features on safety analysis results.
Nuclear Safety, 1993, Vol 39, No. 2, p 237-242.

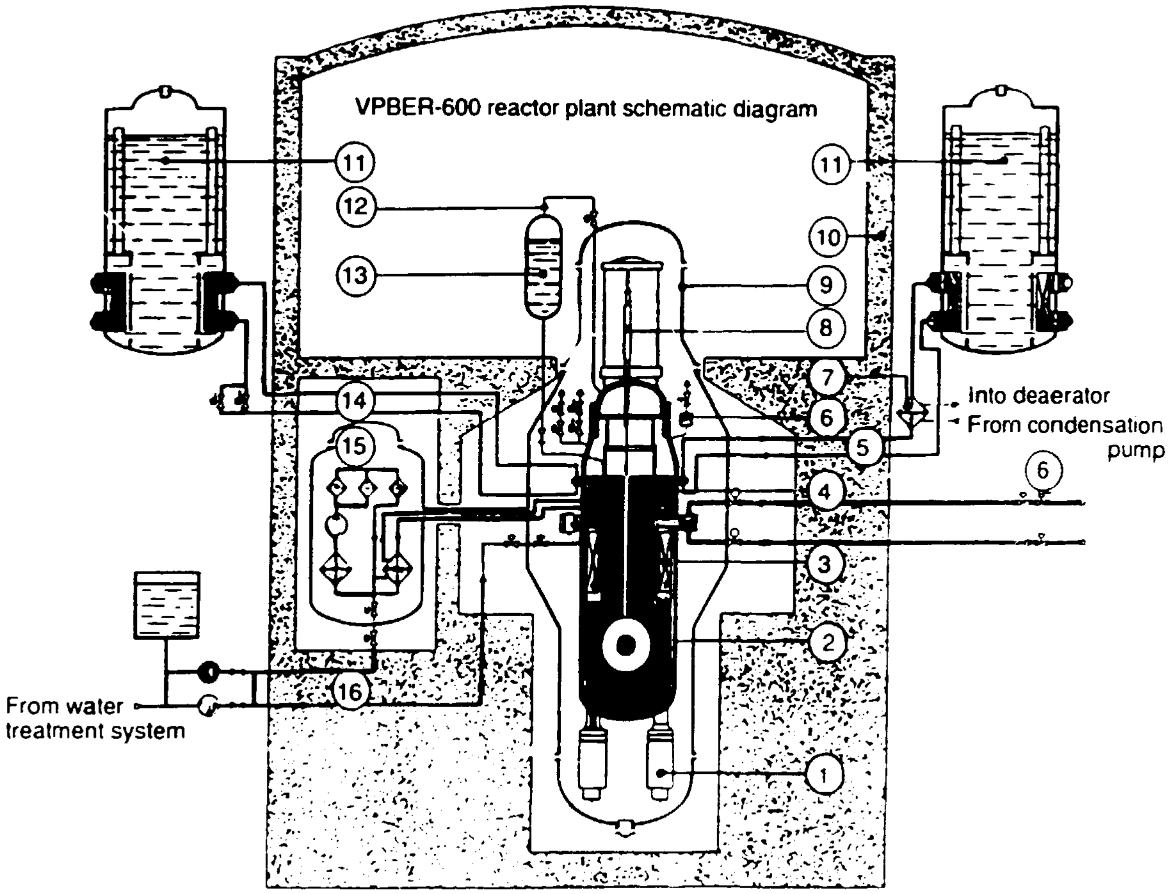


Figure AIP5
VPBER-600: Reactor plant flow diagram.

1	Main circulating pump	9	Guard vessel
2	Reactor	10	Containment
3	Steam generator	11	Heat exchangers unit
4	Heat exchanger - condenser	12	Emergency boron injection system
5	Continuous heat removal system	13	Tank with boron solution
6	Self-actuating devices (direct action)	14	Passive heat removal system
7	Intermediate heat exchanger	15	Coolant clean-up and boron reactivity control system
8	CRDM	16	Primary circuit makeup system

AIP6

Format	Description
Title	SIR (Safe Integral Reactor)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/320 MWe
Organization (name)	ABB-CE, USA; Rolls Royce, UK
Development status	Basic design
Description	In the SIR power plant with a thermal power of 1 000 MWt the primary system components (including the steam generator, pressurizer and main circulation pump) are enclosed in a single steel pressure vessel. The safety systems make extensive use of stored water energy and natural circulation. The size and modular design provide the opportunity for utilities to make incremental power additions with a low capital investment and a short construction period. A 1 200 MWe station could be provided by having four reactor modules feeding into a common steam supply system.

The reactor size was limited by the current practice of RPV diameters which could be manufactured. This resulted in a power output of 320 MWe keeping the conservatism of the design margins. There is a prospect of increasing the power up to 400 MWe.

The integral reactor design incorporates the core, twelve once-through steam generators, six canned rotor pumps at a high level in the vessel. The core consists of 65 fuel assemblies of size 28.5 x 28.5 cm. The top part of the vessel forms the pressurizer with its electric heaters. Its specific capacity is five times larger than that of the PWRs. There is a passive spray system in the pressurizer which takes water from the riser region and sprays it into the steam space in the pressurizer. The specific inventory of the reactor coolant is three times as large as that of standard large power PWRs. The containment has a novel form of pressure suppression factory where the water for pressure control is contained in 8 tanks connected to the reactor cavity by large diameter pipes.

Schedule

The SIR reactor was designed to meet safety criteria of the US and UK and was submitted to US DOE for consideration for funding in the ALWR programme. It was not successful. No current development work is being carried out.

SIR Unit Data

Net electrical output		320 MWe
Core and reactivity control		
-	Fuel inventory	46.1 t
.	Average core power density	54.6 kW/l
.	Average discharge burnup	38 MWd/kg
-	Initial enrichment	3.3 - 4.0 %
-	Active core height	3.47 m
-	Core diameter	2.59 m
-	Number of fuel assemblies	65
-	Number of fuel rods in assembly	432
-	Rod array in assembly	22 x 22
-	Clad material	Zircaloy 4
-	Number of control rods or assemblies	65
-	Control rod neutron absorber material	B_4C
Reactor coolant system		
-	Operation cooling pressure	155 bar
-	Core inlet temperature	294 ^oC
-	Core outlet temperature	318 ^oC
Reactor pressure vessel		
-	Length	23.8 m
-	Diameter	5.8 m
-	Material	Carbon steel
-	Lining material	Stainless steel
Steam generator		
-	Number of SG	12
-	Configuration	Vertical
-	Type	Once through

References

GIBSON, I
The SIR Reactor
Integral Design Concepts on Advanced Water-Cooled Reactors
Obninsk, May 16-20, 1994, IAEA-TC-633.17.

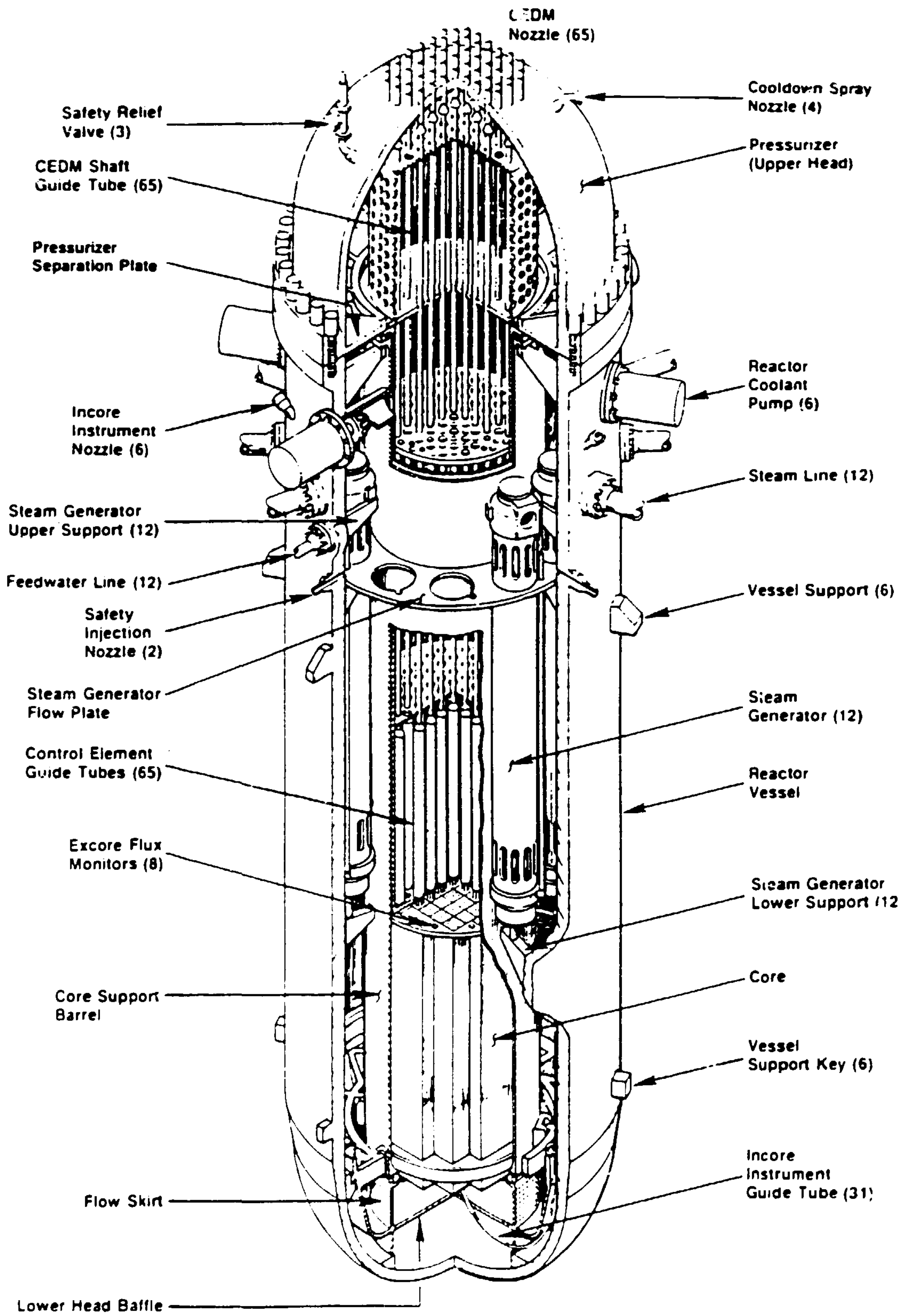
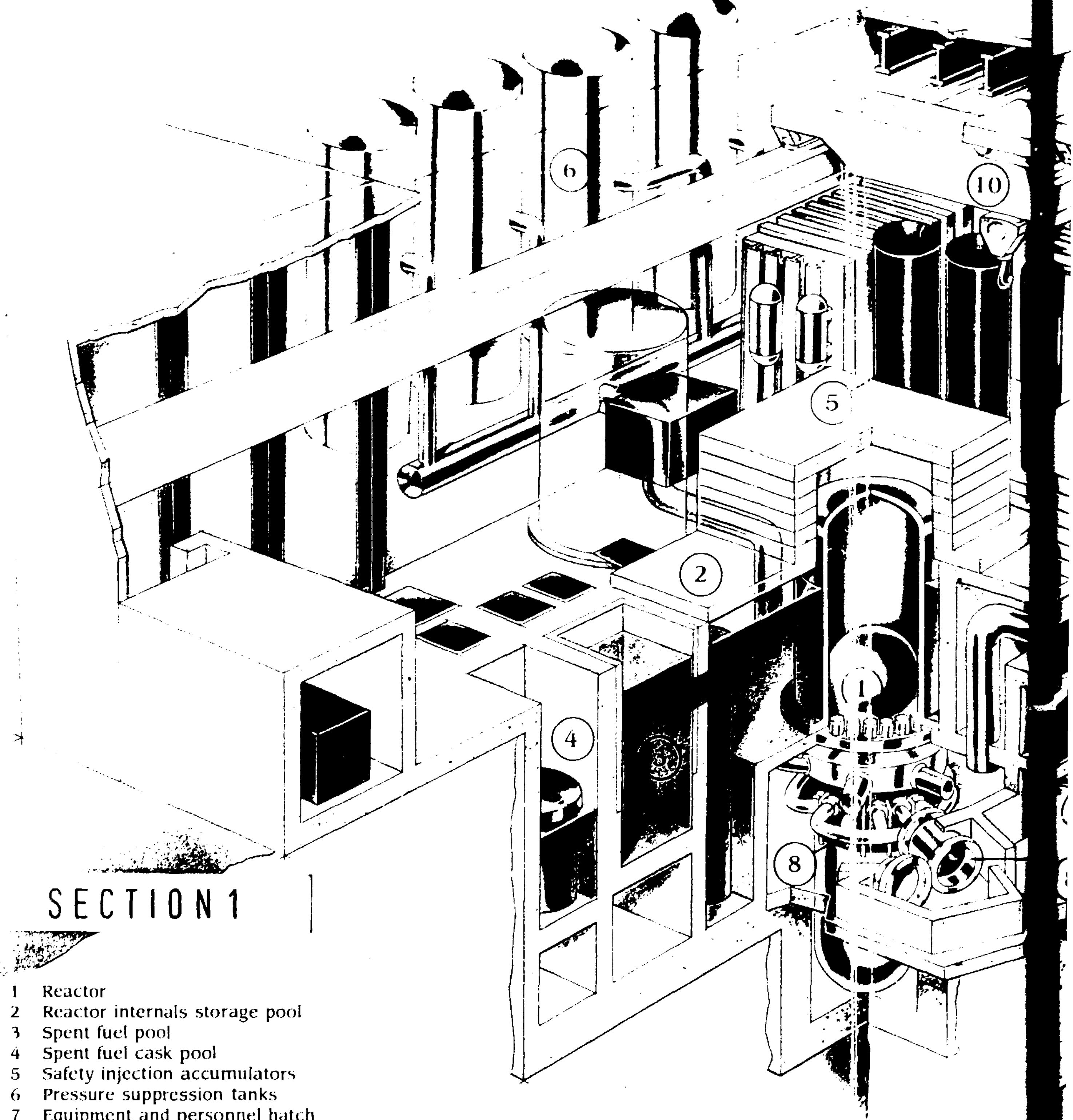


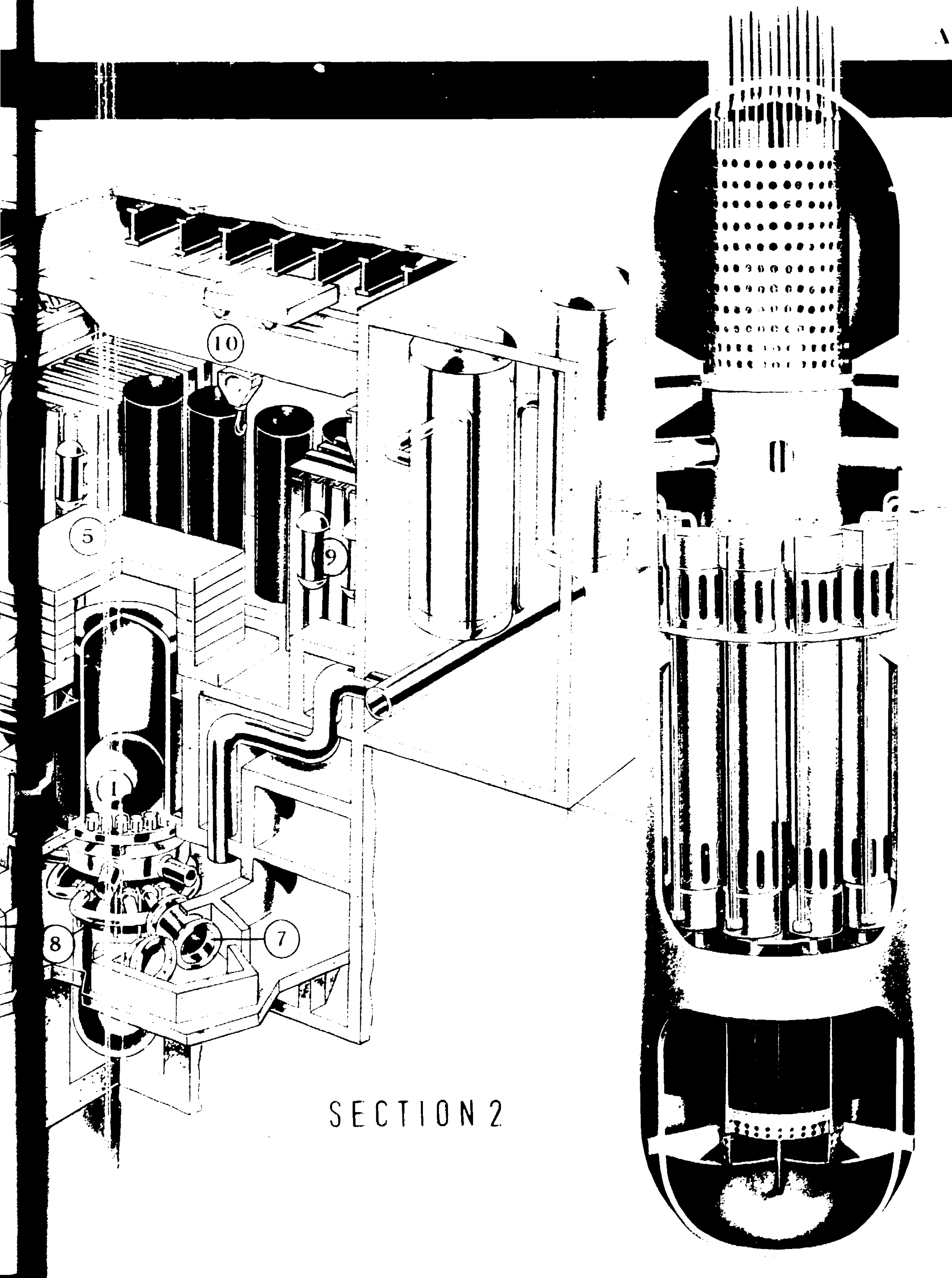
Figure AIP6
SIR: Reactor pressure vessel and internals.

Cut-away View of Reactor and Building



SECTION 1

1 Reactor
2 Reactor internals storage pool
3 Spent fuel pool
4 Spent fuel cask pool
5 Safety injection accumulators
6 Pressure suppression tanks
7 Equipment and personnel hatch
8 Steam headers
9 Natural CRC cooling tank
10 Crane



SECTION 2

AIP7

Format	**Description**
Title	SPWR (System-integrated PWR)
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/600 MWe
Organization (name)	JAERI, Japan
Development status	Studies
Description	The SPWR is an integral type reactor with the thermal power of 1 800 MWt. The steam generator, the reactor coolant pump (RCP), and pressurizer, are incorporated in a reactor pressure vessel (RPV). The tank with borated water is located outside the RPV. This yielded more space inside the RPV and enabled an SG capacity increase and a simplified internals arrangement. The RPV is covered with a water-tight shell with mirror insulation on its inside. The RPV with the water-tight shell is installed in a steel containment vessel filled with borated water. The SPWR uses some technical solutions of the PIUS plant design, but the implementation of the SPWR may take a much shorter time because of its similarity in many respects to existing LWRs.

The core utilizes hexagonal fuel assemblies, some with burnable absorbers mixed with the fuel. Control rods have been eliminated. The canned motor is located at the top of the riser. The two-loop, helical coil type SG has a once-through design. The pressure head of the RCP is utilised to keep the line to the poison injection tank closed; if the RCP head is lost, an hydraulic pressure valve will be opened by spring force (borated water enters the primary loop and shuts the reactor down). An active shutdown system also uses the hydraulic pressure valve - by closing a valve between the RCP and the valve. Scheduled shutdowns are performed using traditional active boron injection.

The emergency injection system is designed to keep the core submerged in water during all LOCA events. The large water inventory and the elimination of large breaks reduce the necessary injection capacity. The engineered safety systems comprise: a two-train Pressure Balanced Injection System (PBIS) which passively injects containment water into the RPV during LOCA events; an Automatic Depressurization System (ADS), four trains with eight valves, which serves to prevent overpressure in the RPV, and to depressurize the RPV to assist the PBIS during small LOCA events; a four train Containment Water Cooling System (CWCS) which passively transfers heat from the containment water to the environment by a heat pipe system and a two train Residual Heat Removal System (RHRS) which removes decay heat under normal shutdown and long-term accident conditions.

Schedule

The SPWR is still in the development by JAERI. Experimental and calculational studies are being done in support of design activities. No SPWR plant has been ordered yet.

SPWR Unit Data

Net electrical output		600 MWe
Core and reactivity control		
-	Fuel material	UO_2
-	Fuel inventory	74.8 t
-	Average core power density	65.1 kW/l
-	Average discharge burnup	48 MWd/kg
-	Initial enrichment/reload enrichment at equilibrium	4.0/4.5 %
-	Moderator material	H_2O
-	Active core height	2.4 m
-	Core diameter	3.83 m
-	Number of fuel assemblies	199
-	Number of fuel rods per assembly	325
-	Rod array in assembly	Triangle
-	Clad material	Zircaloy 4
Coolant		
-	Operating coolant pressure	138 bar
-	Core inlet temperature	288 °C
-	Core outlet temperature	314 °C
Reactor pressure vessel		
-	Overall length of assembled vessel	29 m
-	Inside vessel diameter	6.6 m
-	Gross weight	1 534 t
Steam generator		
-	Number	1
-	Configuration	Vertical
-	Feed water pressure	66 bar
-	Feed water temperature	210 °C
-	Steam pressure	56 bar
-	Steam temperature	295 °C

References

SAKO,K, OIKAWA, T and ODA, L
SPWR (System-integrated PWR)
Integral Design Concepts on Advanced Water-cooled Reactors
16-20 May, 1994, Obninsk, RF, IAEA-622-I3-TC-633.17

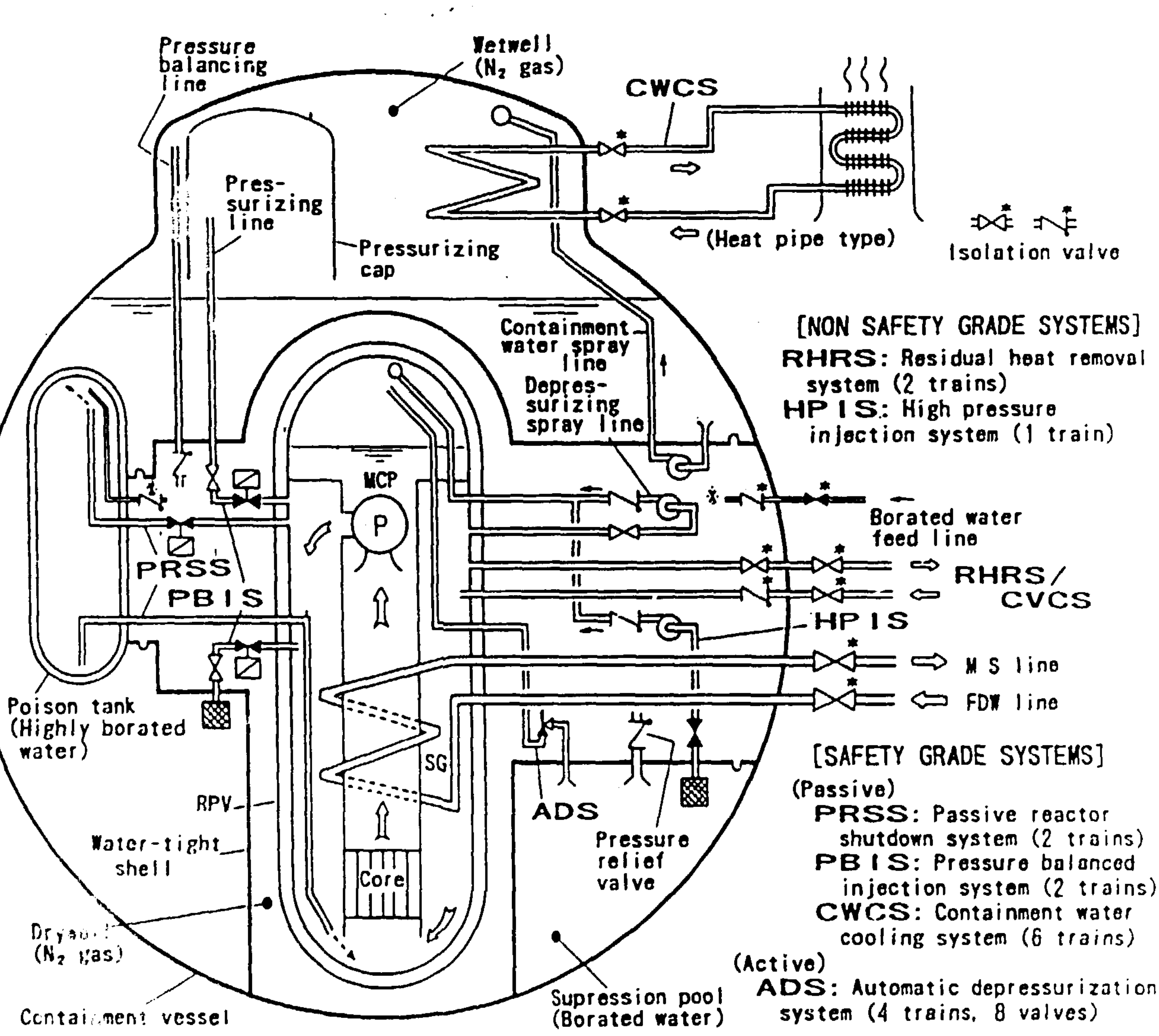


Figure AIP7
SPWR: Safety systems.

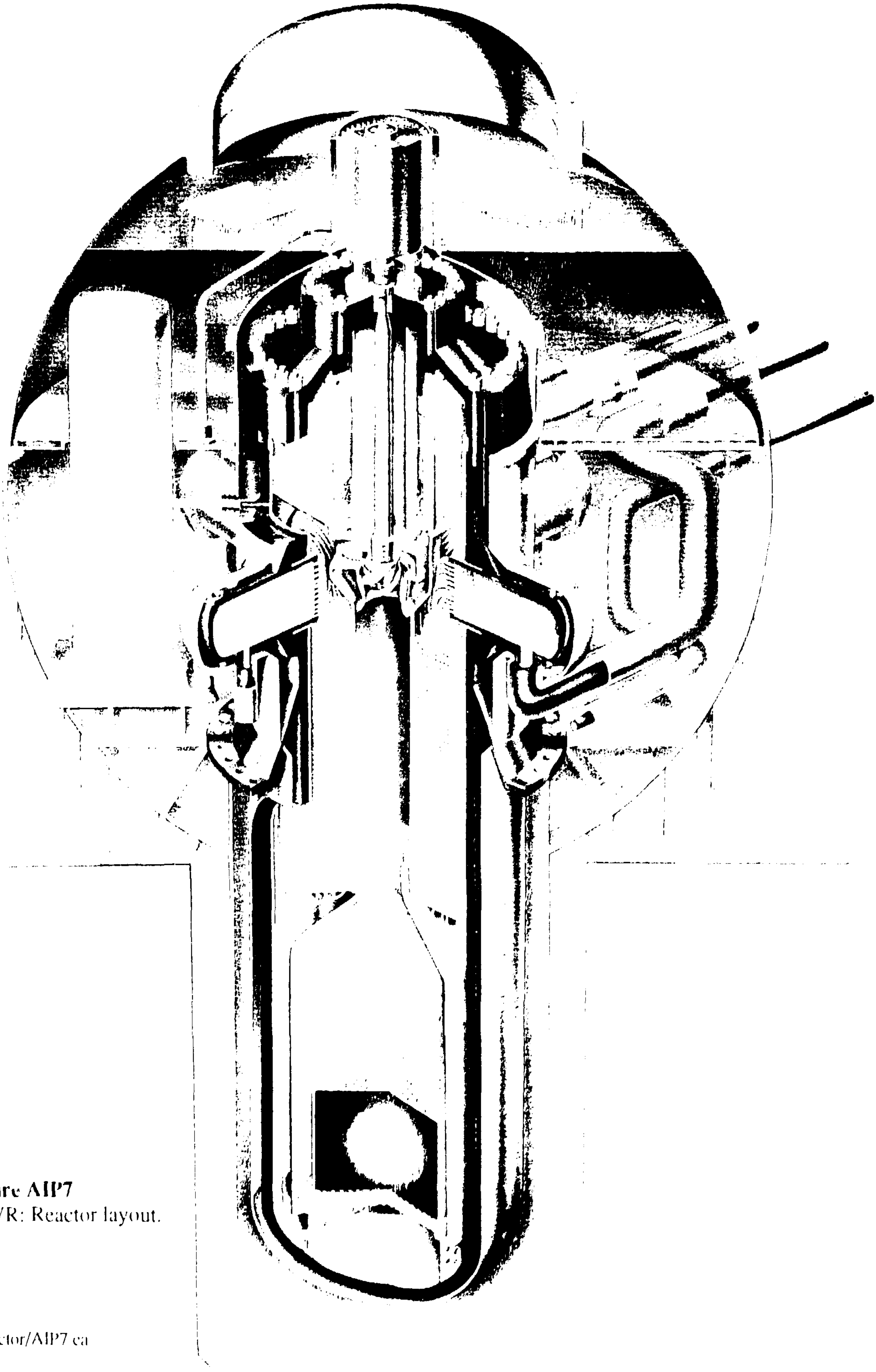


Figure AIP7
SPWR: Reactor layout.

AIP8

Format	**Description**
Title	B-500 SKDI
Application of the reactor	NPP
Reactor type	PWR
Power output	Medium power/515 MWe
Organization (name)	RRC "Kurchatov Institute", EDO Hydropress, RF
Development status	Studies
Description	For the B-500 SKDI reactor plant pressure increase in the primary circuit over the critical value makes it possible to construct the 515 MWe LWR with an integral arrangement and with the natural circulation of the coolant in the vessel with a diameter of less than 5 m. The B-500 SKDI core and steam generator are contained within the steel pressure vessel. The pressurizer is located apart from the pressure vessel. The tube block shroud is 2.8 m in diameter and it separates the riser and downcomer parts of the coolant. The core design is based on the contemporary VVER plant technology.

The steam generator is a once-through vertical heat exchanger arranged in an annular space between the vessel and tube block shroud. It consists of 18 modules with the titanium alloy tubes of 10.8 m in length, 12 mm in outer diameter. The total pressurizer volume is 70 m^3.

In addition to integral arrangement advantages the supercritical coolant pressure results in the following inherent safety features:

- Absence of departure from nucleate boiling on the fuel cladding
- Highly reliable coolant natural circulation
- Affect the reduction of the secondary circuit pressure perturbation on the primary circuit parameters due to a large temperature difference in the steam generator
- Reduced reactor-specific thermal power due to an efficiency increase.

The core cooling passive safety system consists of the systems which remove heat through steam generators and the secondary circuit system PHRS1 and the primary circuit system PHRS2. They include hydroaccumulators and hydrotanks for core flooding.

Schedule

No B-500 SKDI plant has been ordered yet. No development work is being carried out.

B-500 SKDI Unit Data

Thermal power	1 350 MWt
Electric power	515 MWe
Operation pressure at the core outlet	235 bar
Coolant temperature, °C	
Core inlet	365/345 °C*)
Core outlet	381.1/378.8 °C*)
Core coolant flow	2470/2880 kg/s*)
Time period between refuelling (rated power)	2 years
Fuel lifetime	6 years
SG steam pressure	100 bar
Feedwater temperature	252/240 °C*)
Generated steam temperature	379/375 °C*)
Number of steam generator modules	18
Heat exchange tube material	Ti alloy
Tube diameter/thickness	12/1.3 mm
Number of tubes per module	698
Pitch of SG tube bundle	21 mm
Calculated effective length of heat exchange tube	10.8 m
Full heat transfer area	5 120 m^2

*) (Beginning of fuel lifetime/end of fuel lifetime.)

References/contacts

SILIN, V A et al
The integral LWR with natural circulation of the coolant at supercritical pressure B-500 SKDI.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, Proceedings, Vol 1.

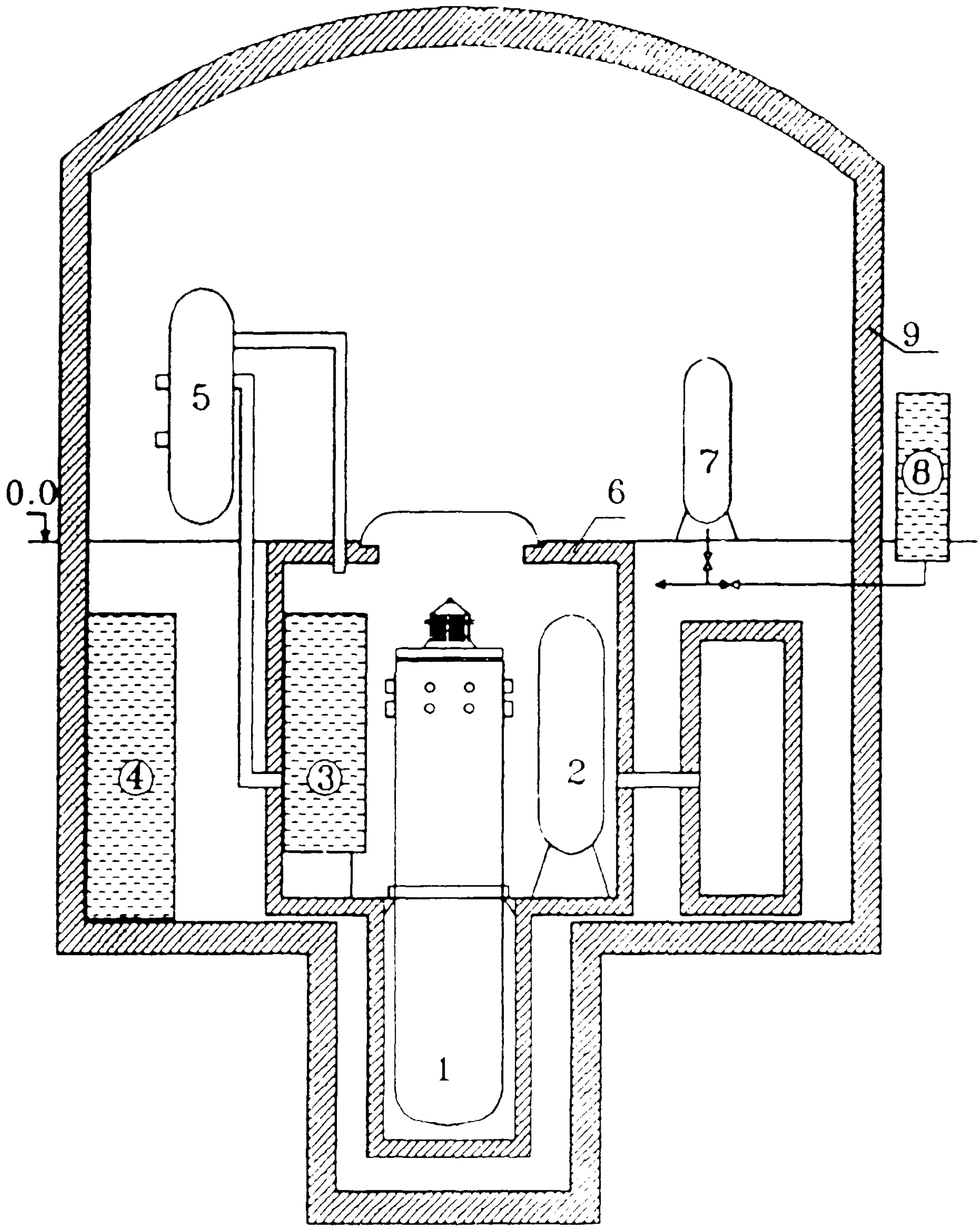


Figure AIP8
B-500 SKDI: General layout of the containment arrangements.

1	Reactor	6	Guard vessel
2	Pressurizer	7	Hydroaccumulator
3	Hydrotank	8	Water storage tank
4	Spent fuel pool	9	Containment
5	Bubble-condenser		

AIP9

Format	Description
Title	SCLWR
Application of the reactor	NPP
Reactor type	PWR
Power output	Large power/1 100 MWe
Organization (name)	University of Tokyo, Japan
Development status	Studies

Description

The supercritical water is considered to be coolant for direct-cycle SCLWR plant design with a thermal power of 2 703 MW. Compared to the BWR plants, a steam separator, a dryer and recirculation system are not necessary. The thermal efficiency is high due to the high coolant enthalpy (24 % higher than current BWR plants).

The main features of the SCLWR concept are the following:

- o The maximum surface temperature of the stainless steel cladding for the fuel is conservatively limited to below 450 °C in order to avoid corrosion
- o The coolant pressure is 250 bar, well above the critical pressure of 221 bar
- o All RPV walls are cooled by the inlet coolant as a PWR vessel
- o The control rods are inserted from the top, since there are no steam separators and dryers
- o There are only two coolant loops because of the low mass flow rate
- o The diameter of main coolant pipes is smaller than those of PWRs
- o The main feedwater pumps which are similar to those of a fossil-fuelled plant pump with the coolant above the supercritical pressure
- o The turbines are smaller than those of an LWR plant

- The emergency core cooling system is similar to that of an ABWR
- The containment is the same as for a BWR

Schedule

No SCLWR plant has been ordered yet. Conceptual design work is under way.

SCLWR Unit Data

Neutron spectrum	Thermal
Fuel/cladding	UO_2/SS
Fissile enrichment	4.2/4.4/4.5 %
Discharge burnup	42 MWd/kg
Thermal/electric power	2 703 MW/1 100 MW
Thermal efficiency	40.7 %
Inlet/outlet temperature	310 °C/407 °C
Coolant flow rate	2 082 kg/s
Pressure	250 bar
Core height	3.7 m
Average power density	74 kW/l
RPV: diameter/thickness	4.14 m/0.34 m

References

OKA, Y et al
Systems design of direct-cycle Super-Critical-Water-Cooled Reactors.
Proceedings ENS '94, Vol 2, Paris, 1994, p 473-476.

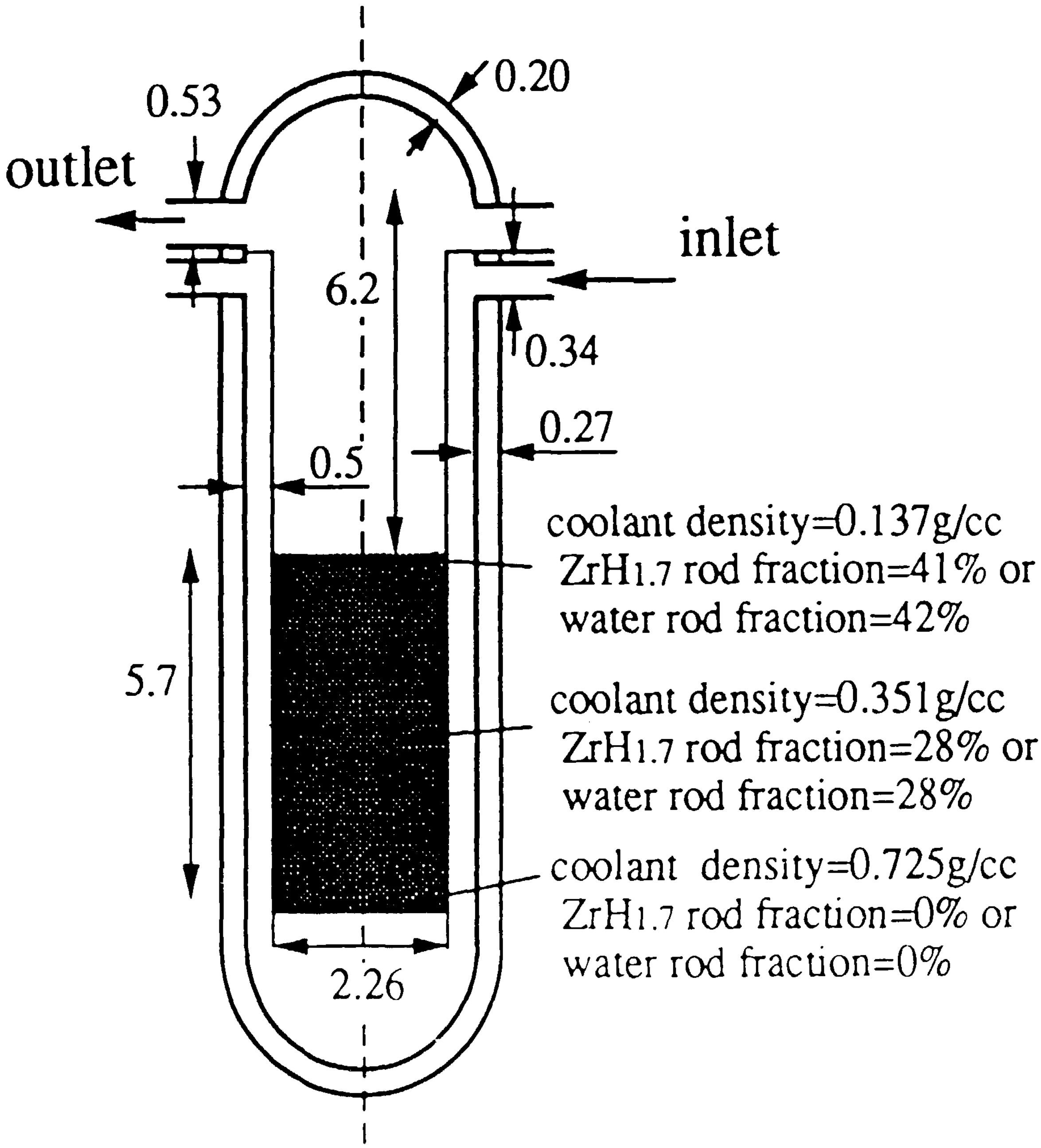


Figure AIP9
SCLWR: Reactor pressure vessel.

B Liquid Metal Fast Reactor

B0 Introduction

The development of the fast reactors in the world has a history of more than 40 years. Based on this the following conclusions may be drawn:

- The volume of theoretical, technological, designing and experimental work accomplished in the LMR field is more than that accomplished for the substantiation of LWRs and HWRs at the stage preceding their large-scale commercialization.

- Energy produced by LMR-based NPPs is still more expensive than that from LWR reactor plants for quite understandable reasons: high enriched fuel, additional intermediate coolant circuits, additional auxiliary systems (preheating, fire protection, prevention of water-sodium interaction etc).

- A considerably higher neutron excess for fission in LMRs compared with thermal reactors permits the core design to be changed using various fuel compositions (metal, oxide, carbide, nitride, cermet) and various structural materials (austenitic and ferritic steel) permit LMR to be used, if necessary, for artificial fuel production (Pu, U-233), for incineration of minor actinides (Np, Am, Cm), for the transmutation of harmful long-lived fission products (Tc-99, I-129).

- In the perspective of nuclear power development, the LMR should not be considered as an alternative to the LWRs but rather as a necessary addition for improving the neutron balance and efficient use of uranium in the world nuclear power system.

- In the decade to come, the LMRs are potential reactors for the effective utilization of weapon grade plutonium and high radioactive plutonium of LWRs.

During the last years, new radiation-resistant structural materials and fuel compositions have been developed for LMRs. Some design solutions for the optimization of reactivity effects and for the incorporation of passive safety measures have become known. However, there is still no common opinion on the optimal capacity (large or medium), reactor design, loop or pool arrangement etc.

An LMR base is available today for the design and construction of evolutionary plants such as large power reactor plants (e.g. Joint European Fast Breeder Reactor and Russian BN-1600), as well as medium power

BN-600M in Russia. A more innovative way is in taking use of simplification and modulization in the small and medium power range (e.g. Advanced Liquid Metal Reactor and (ALMR), and Sodium Advanced Fast Reactor (SAFR) in USA; Top Entry Loop Type DEMOFBR in Japan). For both directions, the designers use passive safety features mainly through high heat capacity, thermal conductivity and the natural convection capability of the sodium coolant for decay heat removal.

Also, there are some innovative LMR concepts which use Pb or Pb-Bi eutectic as coolant. Instead of sodium, the utilization of heavy metal coolant will allow to increase the fuel breeding ratio (by more than 0.05), to decrease the void and density reactivity effects, to widen the coolant margins. All of these, together with a high natural circulation level and wide possibilities of reactivity effects means that there is reason to hope for the creation of LMR with the high inherent safety level.

References

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IAEA Publication IWGFR/83, 1991.

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The Electricité de France Power Programme.
International Conference "Nuclear Energy in 21st century - an environmental bonus?", Proceedings, 14/15 April, 1994, Bath, UK.

o International meeting on sodium cooled fast reactor safety, Proceedings, 3-7 October, 1994, Obninsk, RF.

o Shunsuke Kondo
Design trends and major technical issues of advanced non-water nuclear reactors.
International symposium on advanced nuclear power systems - design, technology, safety and strategies for their deployment, Proceedings 18-22 October, 1993, Seoul, Republic of Korea.

BTS1

Format	**Description**
Title	EFR (European Fast Reactor)
Application of the reactor	NPP
Reactor type	LMR
Power output	Large power/1 500 MWe
Organization (name)	EDF, France; Bayernwerk/Preussen Elektra/RWE, Germany; Nuclear Electric, UK; ENEL, Italy
Development status	Basic design

Description

The EFR project started in 1984, when Belgium, France, Germany, Italy and UK joined efforts to harmonise their fast reactor development programs. The design companies have rapidly made significant progress in common understanding, common methods and common standards.

The main objectives of the EFR project have been:

o EFRs should be regarded as the lead plant of a commercial series that could be ordered in the early part of the next century.

o EFR design should retain the basic characteristics of European prototypes (in particular Super Phenix) and develop innovative design features for further improvement of safety and economics.

o EFRs should achieve parity in economic performance between fast reactors and contemporary PWRs.

o EFRs should be licensable in all participating countries with an overall safety level equivalent to that of future PWRs.

The core has oxide fuel and two configurations are possible in a homogeneous core 1 m high or a heterogeneous core with a fertile slice just below the core mid plane. Both cores have 3 zones with different plutonium contents; 207 sub-assemblies in the inner zone and 72 in each of the intermediate and outer zones.

Economic competition is achieved through a compact reactor block, significantly reducing the size and number of structures and components. A secure design requires a demonstration of robustness against a demanding duty cycle and a high level of confidence in the performance through R&D in the sodium pool thermo-hydraulics, structural behaviour and material properties.

The compactness achieved is demonstrated by the main primary vessel diameter of 17.2 m (compared to 21 m for Superphenix) largely due to the following:

Simplified internal structures:

- o Single redan
- o Redan welded on the diagrid
- o Above core structure integrated to the small rotating plugs
- o Compact design of main vessel cooling

Reduced number of main components

- o 6 intermediate heat exchangers
- o 3 primary pumps
- o External purification

Compact fuel handling

- o Intermediate put-down for the central zone of fuel sub-assemblies using 2 charge machines (direct lift and fixed arm machines)
- o Small diameter, solid plate design of rotating plugs
- o In-vessel fuel storage on the diagrid

A number of structural features have also been improved responding to the demands for robustness and improved possibility of inspection, as follows:

- o Core support with a second load line within the main vessel improving fabrication quality, integrity of support and in-service monitoring

- o Diagrid main duct nozzle and vessel main cooling pipe connections are improved with respect to fatigue and ratchetting

- o Single redan design has been validated for the demanding load following requirement of the duty cycle

Schedule

The validation phase of the EFR project is now complete. The next step should have been the utilities decision to invest in a FOAK plant but all decisions have been postponed until the end of 1994.

EFR Unit Data

Thermal power	3 600 MW
Temperature of coolant at core inlet	395 °C
Average temperature of coolant at core outlet	545 °C
Feedwater temperature	240 °C
Steam temperature	490 °C
Steam pressure	185 bar
Primary pump flow rate	20 172 kg/s
Primary circuit pressure drop	6 bar
Secondary pump flow rate	15 330 kg/s
Secondary circuit pressure drop	4 bar
Nominal max fuel linear rating	520 W/cm/410 W/cm*)
Nominal max fuel clad temperature	645 °C
Peak burn-up (target value)	20 at %
Maximum fuel burnup	15 %
Fuel life time	6 years

*) Start of life/End of life

References

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An Example of International Co-operation:
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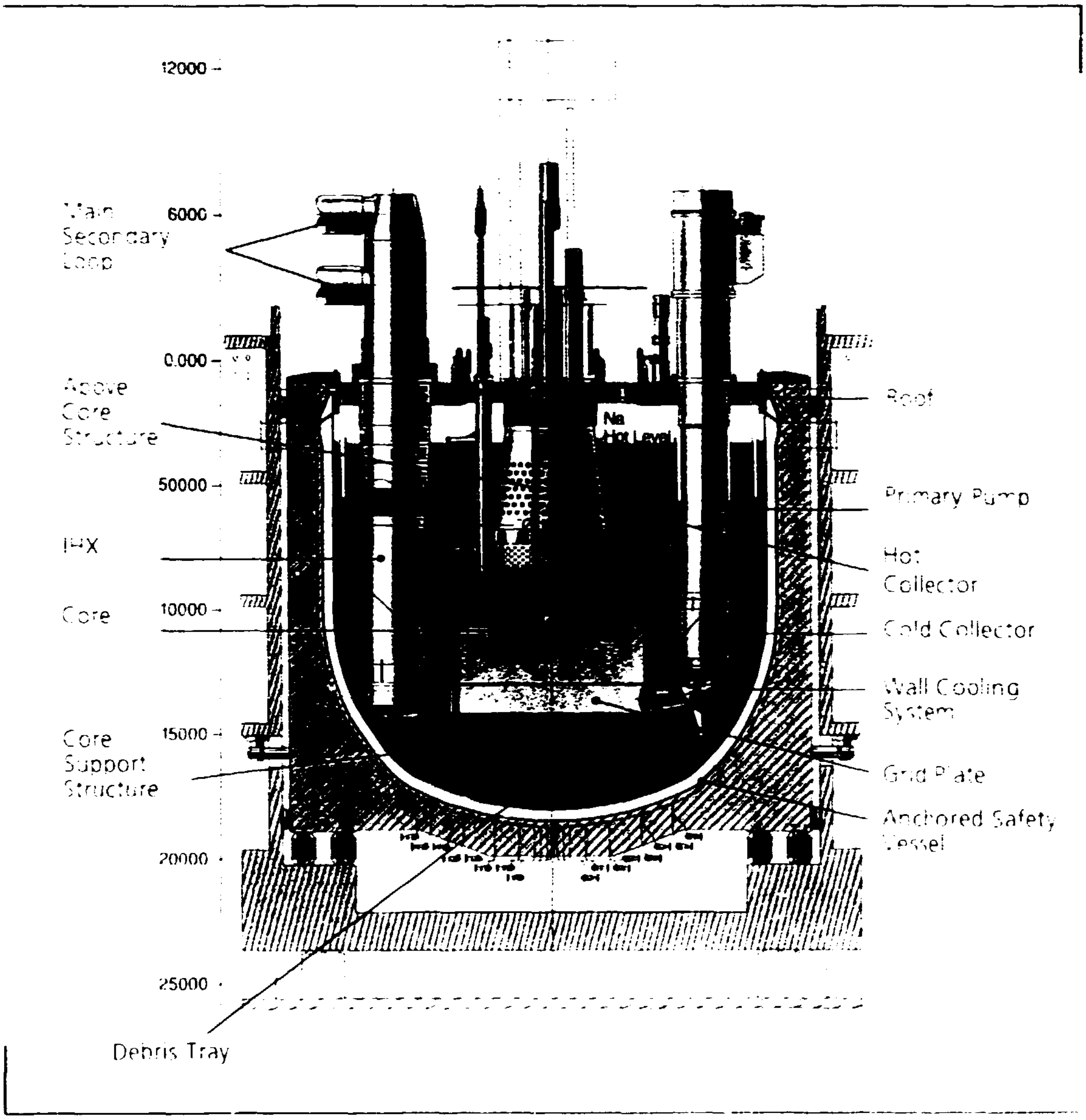


Figure BTS1
EFR: Reactor structures.

BTS2

Format	Description
Title	BN-600 M
Application of the reactor	NPP
Reactor type	LMR
Power output	Medium power/600 MWe
Organization (name)	OKBM, RF
Development status	Detailed design

Description

R&D works on BN-800 and BN-1600 carried out in Russia confirm the possibility of commercial power unit creation on the basis of LMFBR, which will be competitive in efficiency and safety with advanced PWRs designed at present.

The main principles of the advanced BN-600 M concept based on maximum usage of components and systems the reliability of which have been demonstrated by BN-600 operating experience, use of inherent passive safety systems, the prevention and mitigation of severe accident consequences by design measures. The BN-600 reactor has been operated at Beloyarskaya NPP since 1980, could be considered as a prototype of an advanced commercial fast reactor. The development showed that the advanced BN-600 M design as compared with the operating BN-600 reactor allows the next advantages to be achieved:

- Increased safety level
- Improved technical-economical characteristics by 30 - 40 %
- Improved fuel utilization efficiency by 30 - 35 %
- To reduce components and specific weight
- To reduce plant lifetime by up to 50 - 60 years

Schedule

The possible dates of realization of advanced BN-600 M commercial reactor design are:

The component manufacturing and delivery	1998
Commercial NPP with BN-600 M reactor commissioning	2002

BN-600 M Unit Data

Reactor capacity		1 440 MWt
Power output		600 MWe
Primary circuit sodium parameters		
-	Core inlet temperature	395 °C
-	Core outlet temperature	550 °C
Secondary circuit sodium parameters		
-	Steam generator inlet temperature	530 °C
-	Steam generator outlet temperature	365 °C
Steam parameters		
-	Pressure	137 bar
-	Temperature	490 °C
Life time		50 - 60 years
Seismic stability, on MSK scale point		8
NPP specific weight		8.9 t/MWe

References

MITENKOV, F M and KIRUSHIN, A I
The Advanced Commercial Fast Reactor BN-600 M
Proceedings International Conference on Advanced Nuclear Power Plants, October 25-29, 1992, Tokyo, Japan

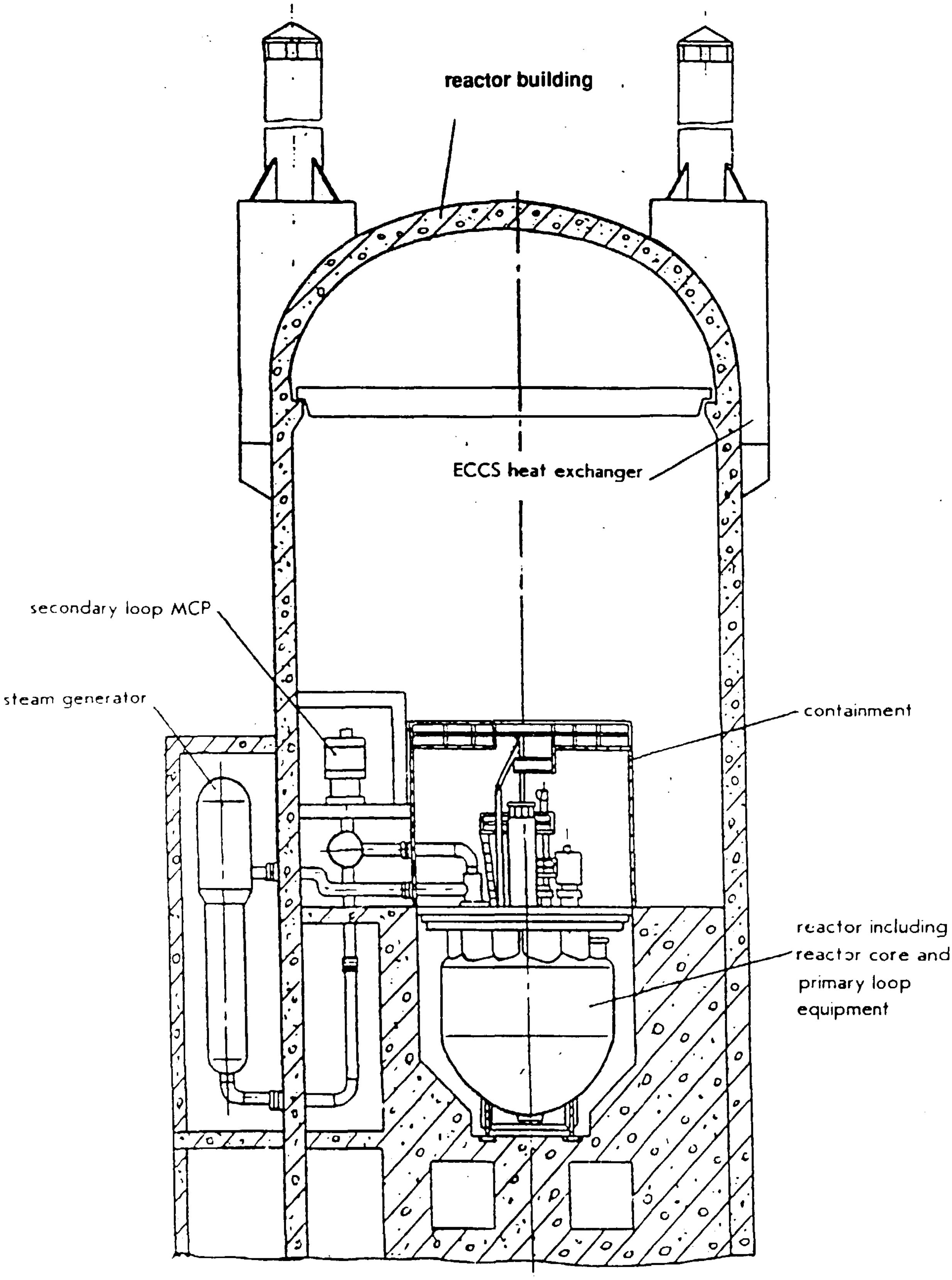


Figure BTS2
BN-600 M: reactor building.

BIS3

Format	**Description**
Title	DEMO FBR
Application of the reactor	NPP
Reactor type	LMR
Power output	Medium power/660 MWe
Organization (name)	FEPC (Federation of Electric Power Companies), Japan
Development status	Basic design

Description

Japan's electricity utility industry will construct a 660 MWe, top-entry, loop-type, demonstration FBR early in the 2000s.

The main targets of these designs are:

- o Realization of safety grades comparable with LWRs
- o Achievement of a reactor outlet temperature of 550 ^{o}C for a higher plant thermal efficiency
- o 1.5 times the construction cost of LWRs
- o A High burn-up and long operational cycle aiming at a reduction of generating cost

The top-entry loop-type reactor plant is designed so that the primary cooling system is composed of the reactor vessel, IHX vessels, and the primary circulation pump vessels, in which each vessel has a free liquid surface and is connected with the top-entry reversed U-shape tube pipings to each other, and secondary system is composed of a steam generator and the secondary circulation pumps.

As for a heat balance, the main steam temperature is set at 495 ^{o}C and the pressure at 169 bar based on the evaluation results of reduction in generating cost, steam generator water-flow stability, etc, at the reactor outlet temperature of 550 ^{o}C, resulting in the gross output of approximately 670 MWe and a thermal efficiency of 42 %.

A homogeneous 2-region core concept that is adopted in the currently operating or designed reactors, is selected for the initial core stage. Modified austenitic stainless steel, which has excellent properties to resist creep strength and swelling, is used for the cladding tube. 8.5 mm-diameter fuel pins are adopted to guarantee a 15-month operation and a good breeding performance. Based on burn-up reactivity at the high burn-up core stage, 30 control rods are provided to compensate for shift excess reactivity.

The top-entry design of DEMO FBR enables a significant shortening of the primary piping system and solves the expansion and construction problems of pipe systems caused by high temperatures. According to estimates the total R&D cost for the DEMO FBR unit 1 will consist of almost Yen 200 billion.

Schedule

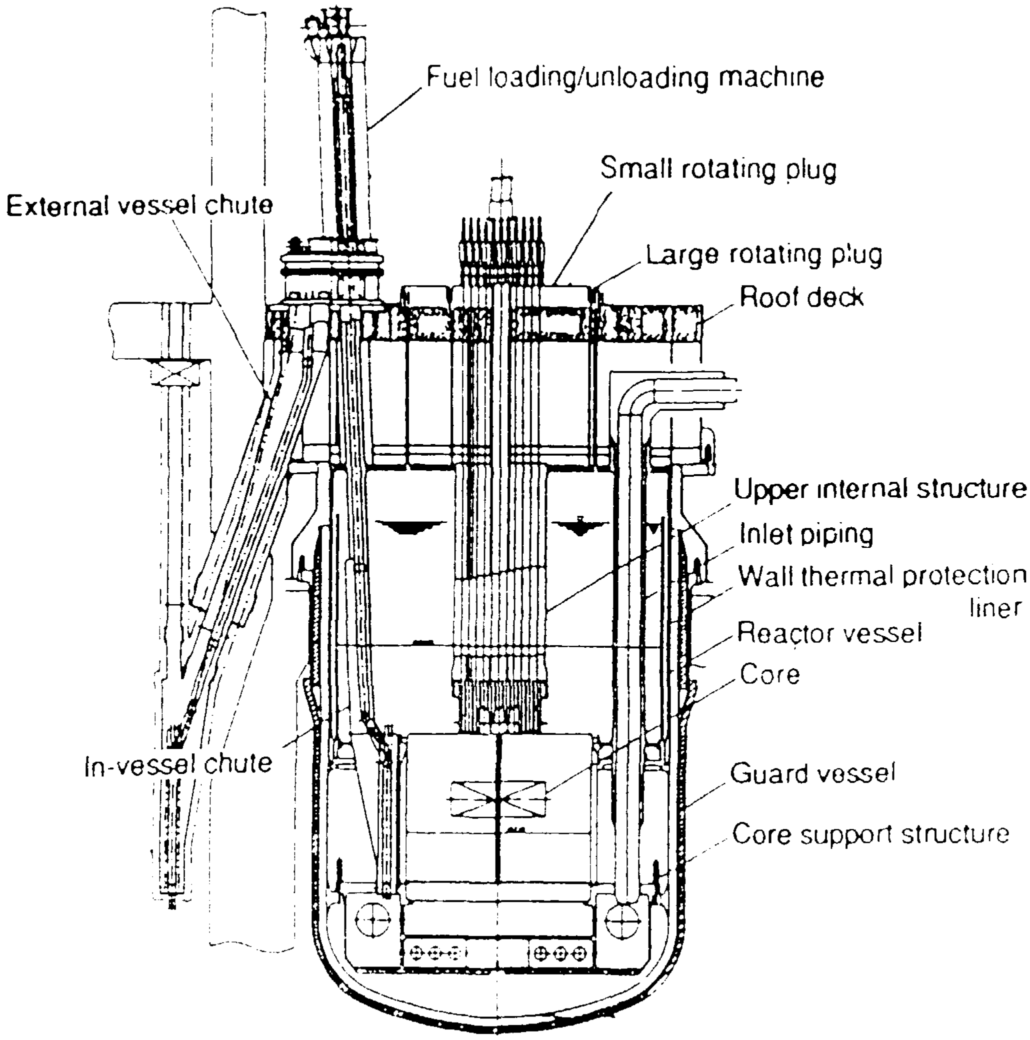
In January 1994, the top managementof electrical power companies at their presidental meeting of FEPC formally decided to launch the construction of DEMO FBR in 2000s. The construction work on unit 1 will depend on the development schedule for innovative technologies and operating results of the prototype reactor Monju.

DEMO FBR Unit Data

Thermal power		1 600 MWt
Number of loops		3
Reactor cooling temperature		550 °C
Steam temperature/pressure		495 °C/169 bar
Reactor core		
-	Core type	Homogeneous
-	Fuel	Pu-U mixed oxide
-	Burn-up	90 GWD/t (initial core) 150 GWD/t (future high burn-up core)
Breeding ratio		
-	With blanket	1.2
-	Without blanket	1.05

References

Utility Industry Decides to Launch Construction of 660 MW Top-Entry Type DEMO FBR Early in 2000s
Atoms in Japan, January, 1994



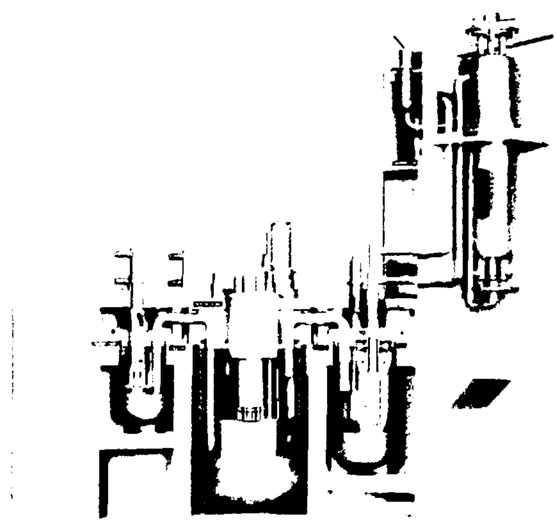


Figure BIS3
DEMO FBR: Reactor structure.

BIS4

Format	**Description**
Title	SAFR (Sodium Advanced Fast Reactor)
Application of the reactor	NPP, WTS
Reactor type	LMR
Power output	Medium power/450 MWe
Organization (name)	Rockwell International, Combustion Engineering and others, US
Development status	Basic design
Description	The SAFR design is based on the large existing LMFBR technology experience in the USA and abroad. It is based on results of supporting development test activities by ANL's Integral Fast Reactor (IFR) The SAFR design and economics have evolved on the basis of the lessons learned from LWR experience. One of the most significant SAFR features of maintaining low capital costs is design standardization and certification licensing approach. SAFR is a modular system with a power output of1 800 MWe. Each 450 MWe module consists of one Nuclear Island (reactor assembly and associated heat transfer equipment) coupled to a single turbine-generator system. The SAFR plant design has a pool arrangement with all its radioactive primary sodium coolant contained within the reactor vessel. The reactor system, SAFR, has a heterogeneous core with fuel and blanket assemblies. The mixed uranium-plutonium metal fuel is considered as the fuel load. The SAFR reactor plant has been designed to provide an advanced level of safety assurance. For this reason, linear-specific fuel power decreased significantly in comparison with traditional values for oxide fuel. The ultimate safety protection mode relies on the inherent responses of the SAFR plant to ensure a safe response to all credible events in addition to postulated accidents without scram. This inherence is made economically possible by such distinct SAFR design characteristics as a pool-type configuration and a natural convention decay heat removal systems. Reactor decay heat can be stored for 1 day following reactor shutdown without discharging any heat outside the containment.

The SAFR reactor design can also be converted for the burning of long-lived minor waste actinides (neptunium, americium and curium) produced in LWRs. As a result, the long-lived nuclides of concern can be efficiently separated from spent LWR fuel and destroyed. Under equilibrium conditions, the spent fuel is reprocessed and recycled back to the reactor. For transmutation, the minor actinides are also recycled back to the reactor along with the recycled U and Pu. The reactor operates with a breeding ratio slightly greater than unity, so that the Pu concentration remains relatively constant. Depleted uranium is added to make up for the heavy metal converted to fission products.

Schedule

No SAFR reactor plant has been ordered yet. No current development work is being carried out.

SAFR Unit Data

Thermal power	1 160 MW
Core inlet temperature	350 °C
Core outlet temperature	510 °C
Core equivalent diameter	3.4 m
Core height	1.52 m
Fuel (core/blanket)	UO_2-PuO_2/UO_2; U-Pu-Zr Metal
Fuel cladding	HT-9
Number of fuel rods (core/blanket)	271/127
Fuel rod diameter (core/blanket)	8.4/12.5 mm
Maximum linear specific power	180 W/cm
Full fuel rod length	2.2 m
Reactor vessel	
- height (oxide/metal fuel)	17.9/14.2 m
- diameter	11.7 m
Steam outlet temperature	475 °C
Feed water temperature	257 °C

References

MEDIUM-size liquid metal reactor, sodium advanced fast reactor (SAFR), Rockwell international info, 1993.

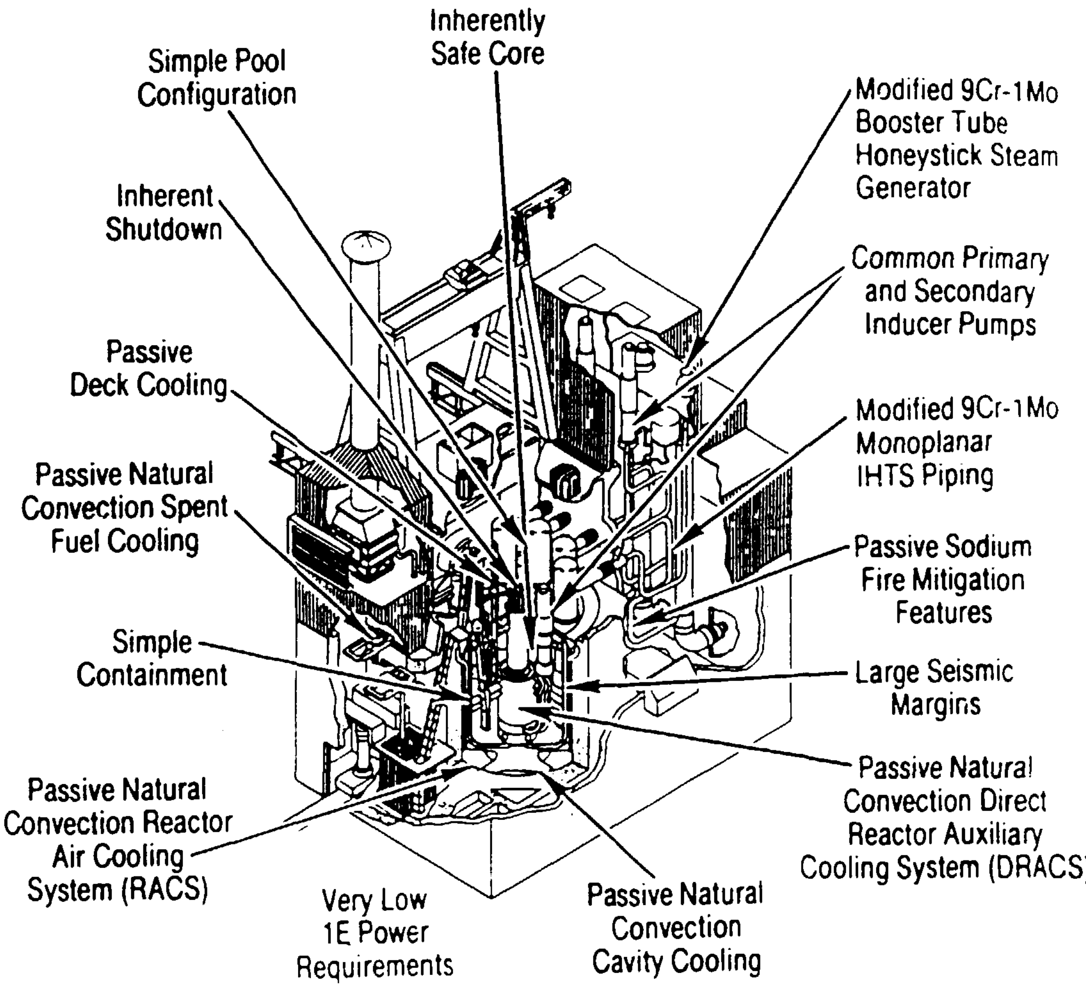


Figure BIS4
SAFR: Nuclear Island Design Features.

BIS5

Format	**Description**
Title	ALMR (Advanced Liquid Metal Reactor)
Application of the reactor	NPP, WTS
Reactor type	LMR
Power output	Medium power/155 MWe
Organization (name)	General Electric and others, US
Development status	Basic design
Description	The Advanced Liquid Metal Reactor (ALMR) design is based on the PRISM (Power Reactor, Innovative Small Module) concept developed by General Electric (GE), while the fuel system is based on the IFR (Integral Fast Reactor) concept developed by ANL. Major emphasis is placed on simplicity, inherent safety features and economic operation.

The design approach included the following main elements:

- o Compact reactor modules
- o Seismically isolated reactor module
- o Safety-related equipment limited to nuclear island systems
- o Passive shut down heat removal for loss of coolant events
- o Passive reactivity control for undercooling and overpower events with failure to scram
- o Maximum radiation release levels during severe accidents are sufficiently low enough that a formal public evacuation ias not needed

The reactor module consists of a containment vessel, reactor vessel and its internals.

The reactor module, the intermediate heat transport system, and most of the steam generator system are underground, an approach that has an estimated cost-benefit in meeting the requirement for radioactivity containment, seismic design, sodium fire mitigation, and protection from external threats such as sabotage and missiles.

The reactor module is about six meters in diameter and 19 meters high. The core of about 1 m in height is a radially heterogeneous arrangement of fuel and blanket assemblies. The refuelling interval is eighteen months. Metal U-Pu-Zr fuel provided negative reactivity feedback for loss of coolant and transient overpower events as demonstrated in EBR-II, and the potentially competitive fuel cycle cost with pyroprocessing. The six control rod assemblies are controlled by the Plant Control System.

Primary sodium is circulated in the core by four electromagnetic pumps. The heat is transferred to the intermediate sodium. The single-wall helical coil steam generator (HCSG) has been selected for ALMR.

In the unlikely event, when that the intermediate systems become unusable during power operation, the entirely passive reactor vessel auxiliary cooling system (RVACS), which operates continuously, will automatically come into full operation without the need for operation action. The ALMR concepts offer a high level of safety against severe accidents, involving core damage.

The ALMR design can also be converted for the burning of long-lived minor actinides produced in current LWRs.

Schedule

It must be concluded that the design and licensing activities are far from completed. Conceptual design phase to be completed.

ALMR Unit Data

Reactors per power block	3
Net electrical output	465 MWe
Turbine throttle conditions (saturated)	66 bar
Reactor thermal power	471 MWt
Primary sodium temperature	
- Inlet/outlet	338 °C/485 °C
Secondary sodium temperature	
- Inlet/outlet	282 °C/443 °C
Fuel type	
- Reference	U-Pu-Zr Metal
- Alternative	U-Pu Oxide
Refuelling interval	18 months
Time for breeding	50 years, capability

References

FIPPETS, F E et al
Design of PRISM of Inherently Safe, Economic and Testable Liquid Metal Breeder Reactor Plant.
Proceedings International Conference Fast Breeder Systems, USA, 1993, p 13-17.

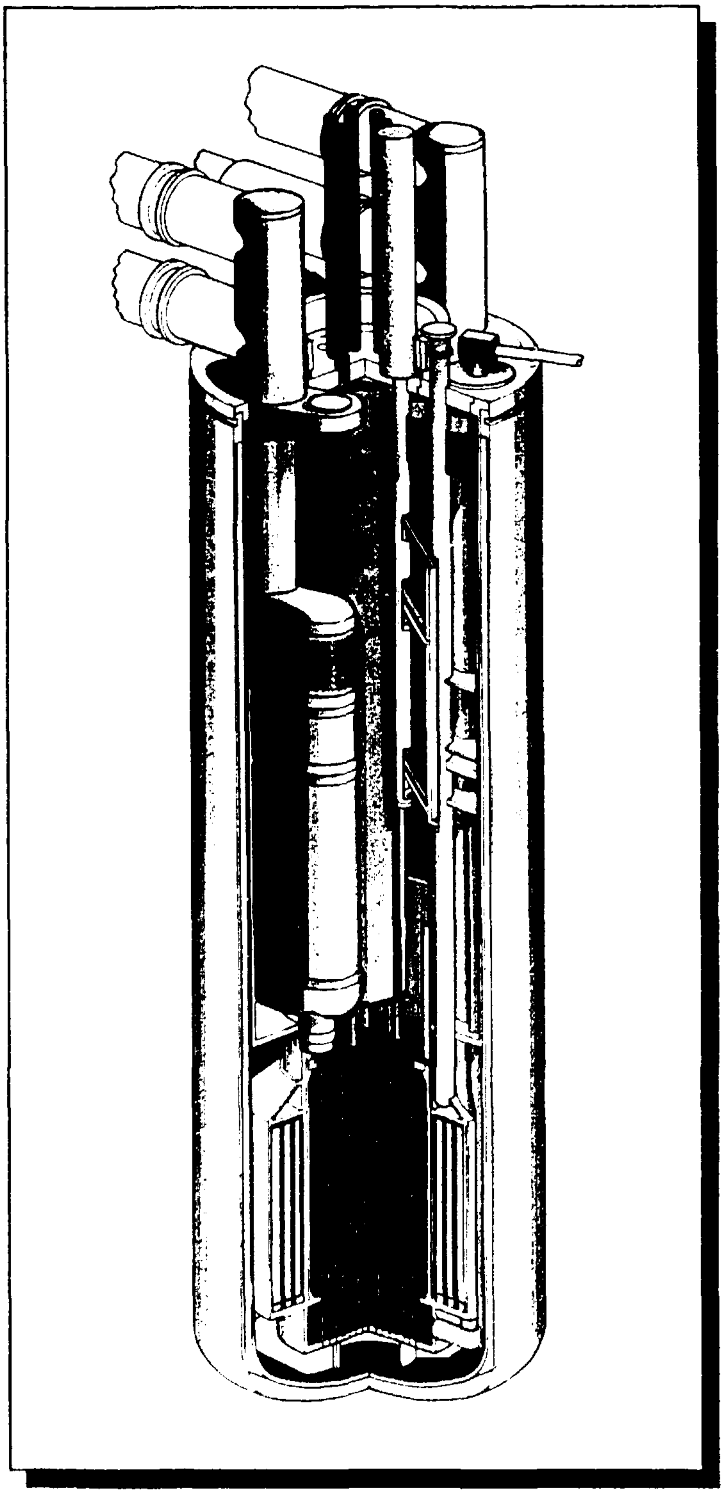


Figure BIS5
ALMR: Reactor module.

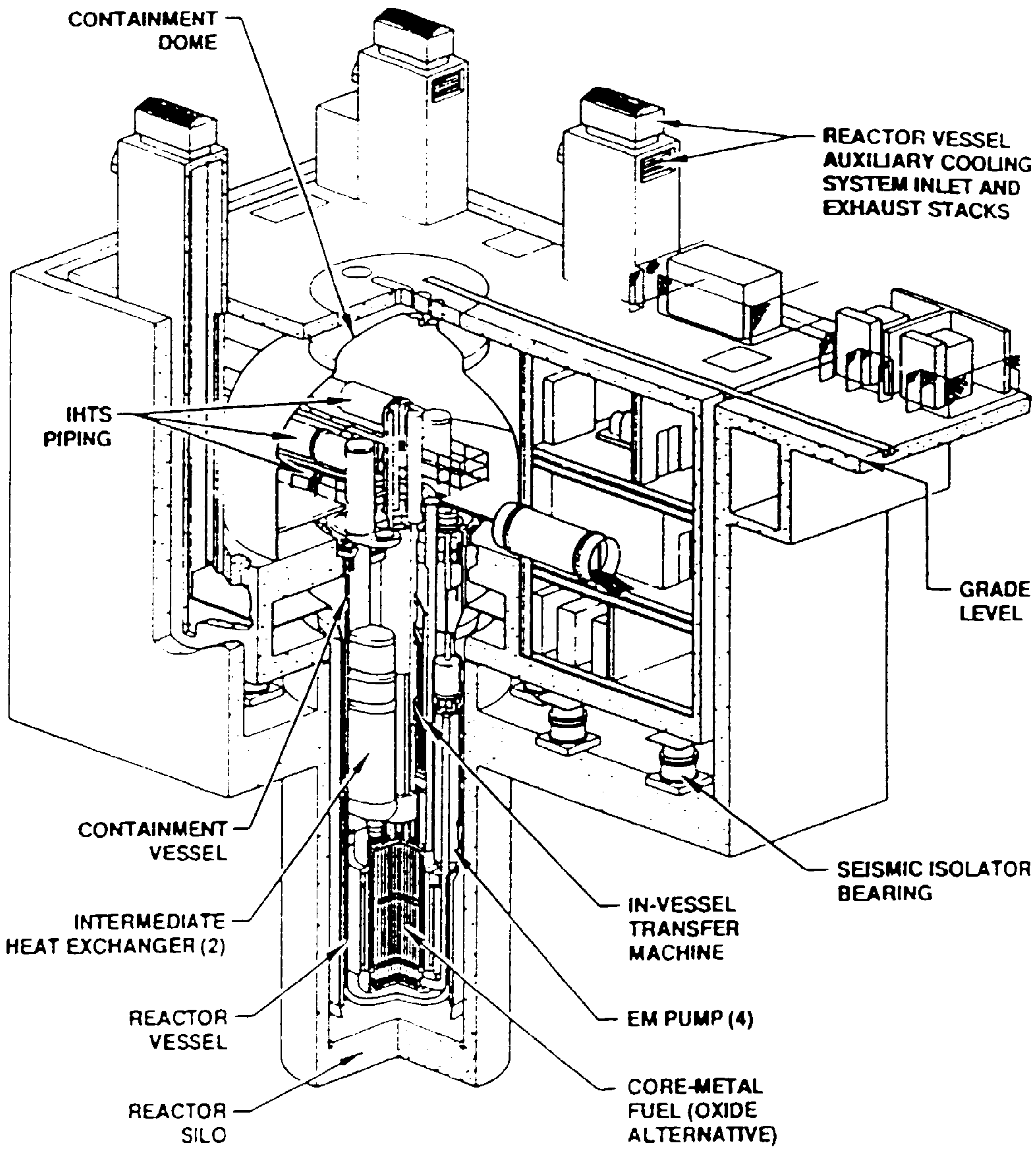


Figure BIS5
ALMR: Reactor building.

BIL1

Format	Description
Title	BREST 300
Application of the reactor	NPP
Reactor type	LMR
Power output	Medium power/300 MWe
Organization (name)	RDIPE, RF
Development status	Studies
Description	With a 300 MWe electrical output, the BREST 300 reactor plant represents progress in the implementation of the innovative safety features specific for the reactor fuel, lead coolant and other reactor components, as well as some of its design peculiarities.

The BREST 300 reactor core is made up of fuel assemblies of canless design with high-density nitride fuel. The latter has a high thermal conductivity and agrees favourably with the steel cladding and lead coolant.

The canless design of the fuel assemblies and a wide relative spacing of the fuel element lattice provide for the removal of a high level of power by natural convection (up to 15 % Nnom), as well as the fuel assembly failure by closing one of the flow sections at the core inlet is eliminated due to radial coolant overflowing in the core.

In order to enhance safety during accidents with a steam-generator piping rupture, increase of maintainability and the availability factor, reduce vessel dimensions and the volume of the lead circuit - the BREST 300 design considers that such a type of reactor layout permits the steam-generators and pumps to be located beyond the boundaries of the main vessel.

With respect to the design, lead circulation is organized in such a way that its flow rate through the core is determined not by pump pressure but the difference of the level of hot and cold coolant.

The system of emergency and operating air cool-down is completely passive, using air tube heat exchangers surrounding the reactor vessel and steam generators. Heat is transferred through vessel surfaces and discharged by emission into atmosphere due to natural air thrust.

Investigations of transients considering extra feedbacks showed that the BREST 300 reactor plant had high self-protection during the following severe accidents without the scram:

- o Termination of forced circulation in the primary or secondary circuits
- o Self-motion of all the control rods at any power
- o Coolant cooldown at the core inlet during the secondary circuit rupture

Schedule

It must be concluded that the design and licensing activities are far from completed. Conceptual design work is under way.

BREST 300 Unit Data

Thermal power	700 MW
Net electric power	300 MW
Number of fuel assemblies	185 (57+72+56)
Core diameter	2.3 m
Core height	1.1 m
Fuel assembly lattice spacing	150 mm
Core fuel element diameter	9.1;9.6;10.4 mm
Fuel element spacing	13.6 mm
Core fuel	UN+PuN
Core fuel load	16 t
Load of Pu/(^{239}Pu+^{241}Pu)	2.2/1.6 t
Maximum fuel burn-up	up to 12 %
Fuel lifetime	5 - 6 years
Refuelling intervals	1 - 2 years
Core breeding ratio	~ 1
Total breeding ratio	~ 1
Inlet/outlet lead temperature	420/540 °C
Inlet (water)/outlet (steam) temperature	340/515 °C
Maximum cladding temperature	650 °C
Maximum lead velocity	1.8 m/s

References

ADAMOV, E O
Progress in lead-cooled fast reactor design.
Proceedings International Conference on Advanced Nuclear Power Plants, October 25-29, 1992, Tokyo, Japan, Vol 2.

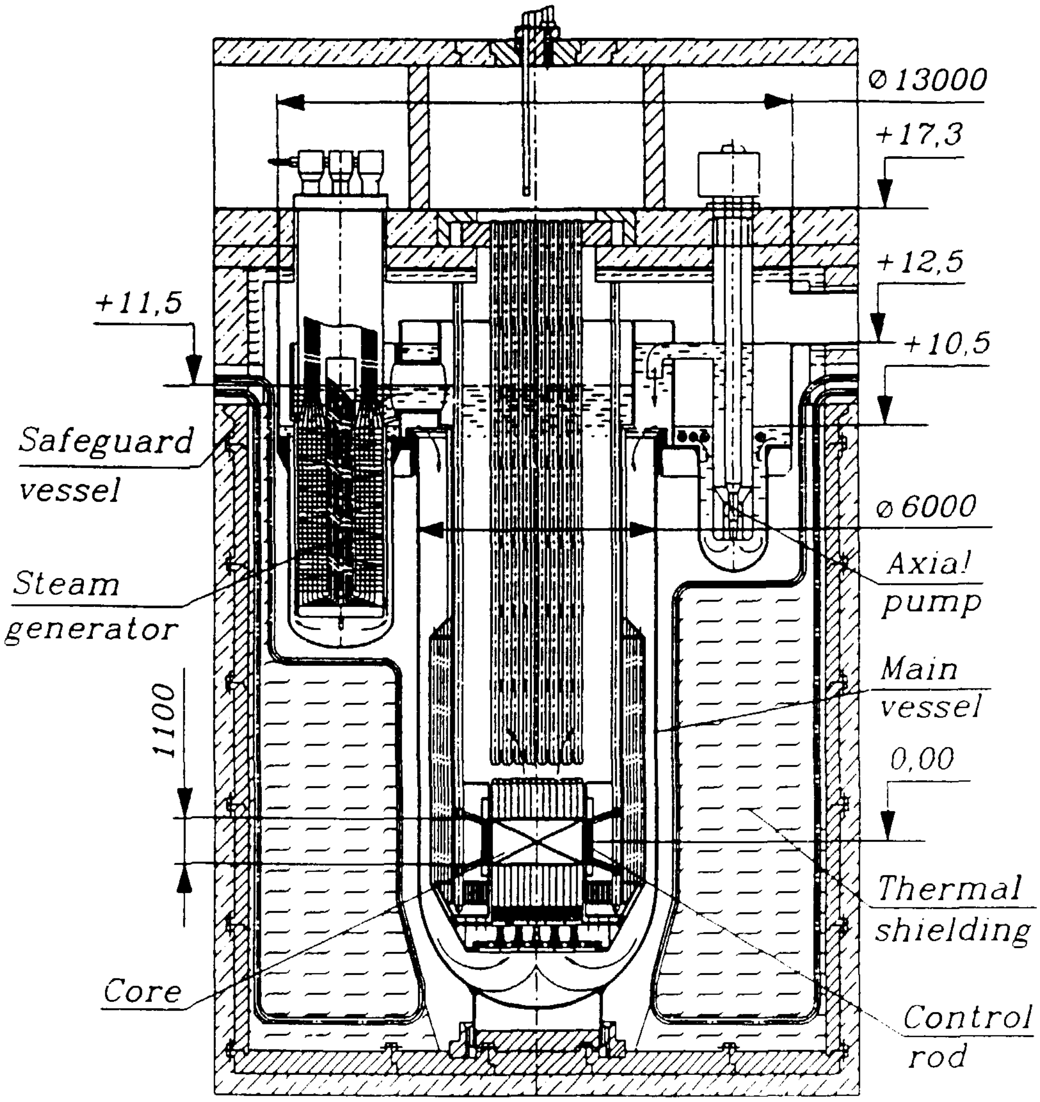


Figure BIL1
BREST 300: Reactor arrangement.

BIL2

Format	Description
Title	LFBR
Application of the reactor	NPP, TWS
Reactor type	LMR
Power output	Medium power/625 MWe
Organization (name)	JAERI, Japan
Development status	Studies
Description	The LFBR with an electric output of 625 MWe represents an innovative type of LMR plant which is being developed by JAERI and others. The design goals for the LFBR include higher safety, economics and minor actinide (MA) transmutation efficiency than those of MOX-fuel fast reactors.

The core, the core support structure and the primary heat transport system components are contained in a reactor vessel. The primary heat exchanger is the steam generator and its helical coil tubes encircle the center part of the reactor which includes the core and the hot coolant plenum. The primary pumps are located at the cold region of the primary coolant flow path. The coolant flows along the arrows in the figure.

The nitride thorium fuels are used to burn and transmutate Pu and MA, and to achieve a negative void reactivity. The U-233 of about 1.8 tonnes are produced and almost the same amount of Pu is transmutated in the burn-up times of 3 000 days. MA of 0.8 - 1.4 tonnes generated from 33 - 58 LWRs with 33 MWd/kg are transmutated.

The main safety and design features of the LFBR are the following:

o Reactor coolant paths are designed to prevent Pb-H_2O reaction products entering into the core in the event of an accident

o The reactor structure can withstand 2G seismic acceleration conditions

o The SG tube is designed as a helical coil type, running along the reactor vessel wall to reduce the reactor vessel diameter

- o The lead-cooled SG size is as large as the sodium-cooled one
- o The pressure drops at the fuel assemblies and the SG are designed as low as possible
- o The decay heat removal system is designed as a water/steam cooled Primary Reactor Auxiliary Coolant System (PRACS) for eliminating lead piping
- o The electrical pre-heating system is selected
- o An upper internal structure plug removal is selected for the refuelling system

Schedule

It must be concluded that the design and licensing activities are far from completed. Conceptual design work is under way.

LFBR Unit Data

Type of plant		Pool type
Thermal/electrical output		1 500 MWt/625 MWe
Reactor outlet/inlet temperature		570/470 °C
-	Fuel	12 wt % Pu $(U,Pu)^{15}N$
-	Loaded Pu	4 tonnes
-	Burn up	92 MWd/kg
Reactor vessel		
-	Diameter	9 m
-	Height	17 m
-	System weight	9 300 tonnes excluding cover
Thickness		100 mm
Primary system		
-	Heat exchanger	Integrated in the reactor
-	Number of pumps	3
-	Lead flow	3.70×10^8 kg/h
Intermediate system		None
Feed water system		
-	Feed water/steam temperatire	330/530 °C
-	Feed water flow	290×10^6 kg/h

References

TAKANO, H et al
A concept of self completed fuel cycle based on lead-cooled nitride-fuel fast reactor.
Proceedings ICENES '93, Makuhari, Chiba, Japan, 20-24 Sept, 1993, p 309-315.

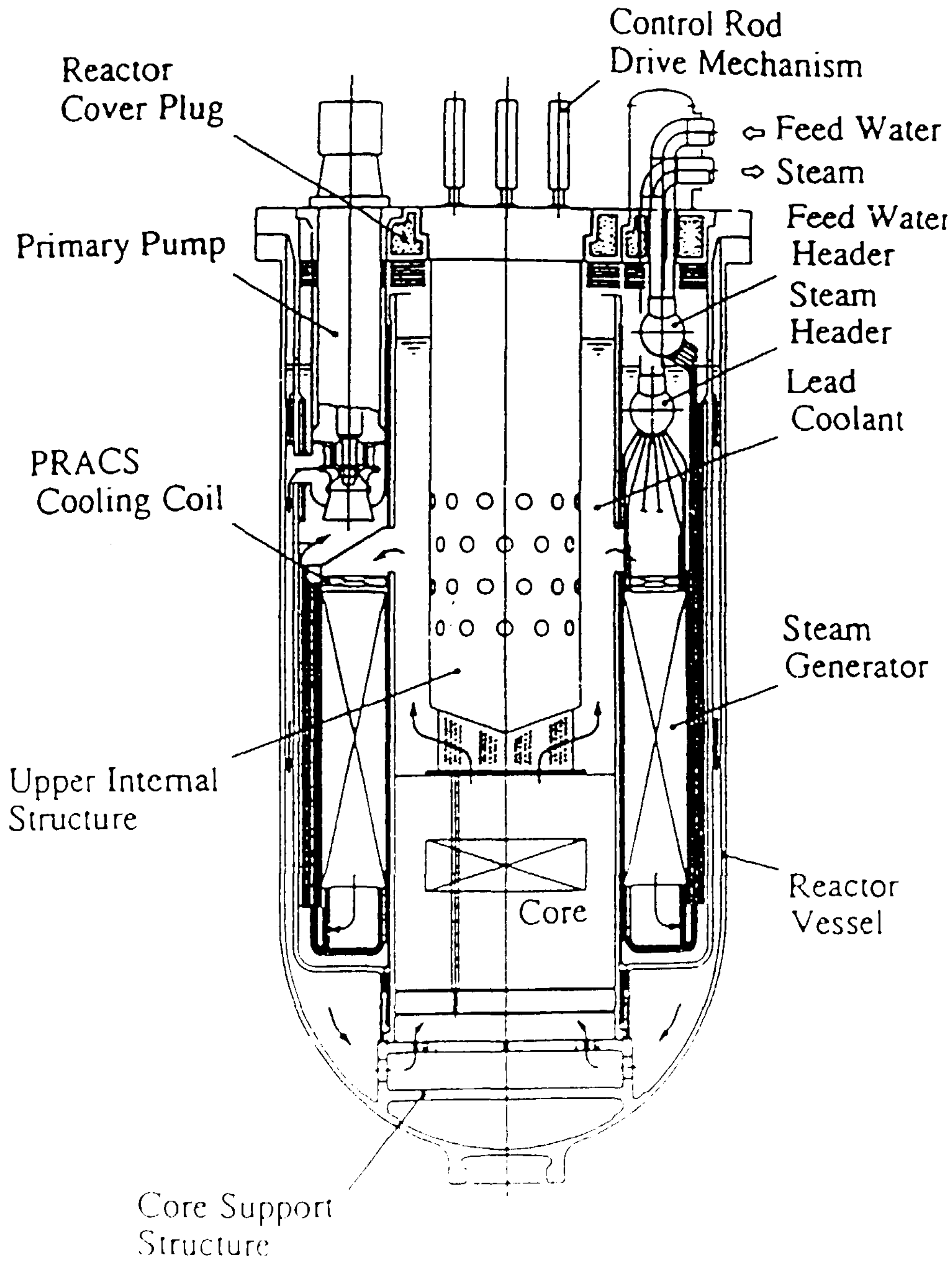


Figure BIL2
LFBR: Reactor plant arrangement.

C Gas-Cooled Reactors

C0 Introduction

Graphite-moderated gas-cooled reactors have been operated since 1956 for commercial nuclear power. MAGNOX and AGR designs are based on metallic cladding of the fuel with CO_2 as coolant. The further development of gas-cooled reactors in the USA, Germany, Switzerland, Russia and Japan has concentrated on the High Temperature Gas Cooled Reactor (HTGR) type using helium as coolant and ceramic cladding of the fuel. The operating temperatures of HTGR can be adapted not only to the effective electricity generation, but also for industrial cogeneration applications.

The specific safety features of HTGRs are mainly based on:

- The high temperature resistance of the ceramic core structure
- The large margin between operating temperatures and the failure limits of the coated particles
- A negative reactivity temperature coefficient and large margin for allowable temperature rise
- The low power density of the core
- The large heat capacity of the graphite moderator and structures
- The inert helium coolant.

The modular HTGR concept has been developed since the 1980s in Germany, the USA, Japan and Russia. These innovative designs incorporate significant departures from the large power HTGRs in the reactor coolant system arrangement and safety system organization. The design of modular HTGRs is governed by the aim to keep, solely by passive heat transfer, the fuel temperatures so low under all accident conditions that no significant fission product release can take place even including the depressurization accident combined with a loss of active cooling. This leads to limitations in power and power density as well as to a larger surface/volume ratio because the decay heat has to be transformed to coolers or structures outside the reactor vessel only by radiation and conduction of the heat via the surface of the core and pressure vessel.

The Modular HTR reactor plants are included in the nuclear research programs in the USA, Japan, Germany and some other countries.

References

Gas cooled reactor design and safety
Vienna, Austria, IAEA, Technical Reports Series N313, 1990.

Shunsuke Kendo
Design trends and major technical issues of advanced non-water nuclear reactors.
International symposium on advanced nuclear power systems - design, technology, safety and strategies for their deployment, Proceedings 18-22 October, 1993, Seoul, Republic of Korea.

CI1

Format	Description
Title	HTR-Modul
Application of the reactor	NPP, NHPP
Reactor type	HTGR
Power output	Small power/80 MWe
Organization (name)	Siemens-KWU, Germany
Development status	Basic design
Description	The HTR-Modul reactor concept has been developed since 1980 in contrast to the time tendency to enhance the unit sizes for further improvements in economy as an economic nuclear heat source with high safety margins. This concept is based on the modular NPP approach. One to eight modular units with a rating of 200 MWt each can be connected to the one plant (up to 640 MWe). The design of modular HTRs is governed by the aim to keep, solely by passive heat transfer (heat conduction, natural convection) the fuel temperatures so low (< 1600 °C) under all accident conditions that no significant fission product release can take place even including the depressurization accident combined with a loss of all active cooling systems. This leads to limitations in power, power density and radial size of the core as well as to a larger surface/volume ratio because the decay heat transported to coolers or structures outside the reactor vessel only by heat radiation and heat conductions. This leads to a relatively large height of the core for a small diameter, to reduce the temperature gradients in the fuel pebble bed. The reactor is fueled with spherical TRISO fuel elements which pass through the core several times until the design burn-up is reached. The possible safety problems of the HTR-Modul deal with the accidental ingress of the water into the core. The system and fuel behaviour have been successfully demonstrated also in the 50 MWt AVR reactor plant. Today a HTR-Modul power plant is ready to be constructed and operate commercially.
Schedule	No HTR-Modul reactor plant has been ordered yet. No current development work is being carried out.

HTR-Modul Unit Data

Overall Plant

-	Number of modular units	8
-	Reactor thermal power	8x200 MW
-	General output at terminals	688 MWe
-	Net power output	640 MWe

Reactor Core (one modular unit)

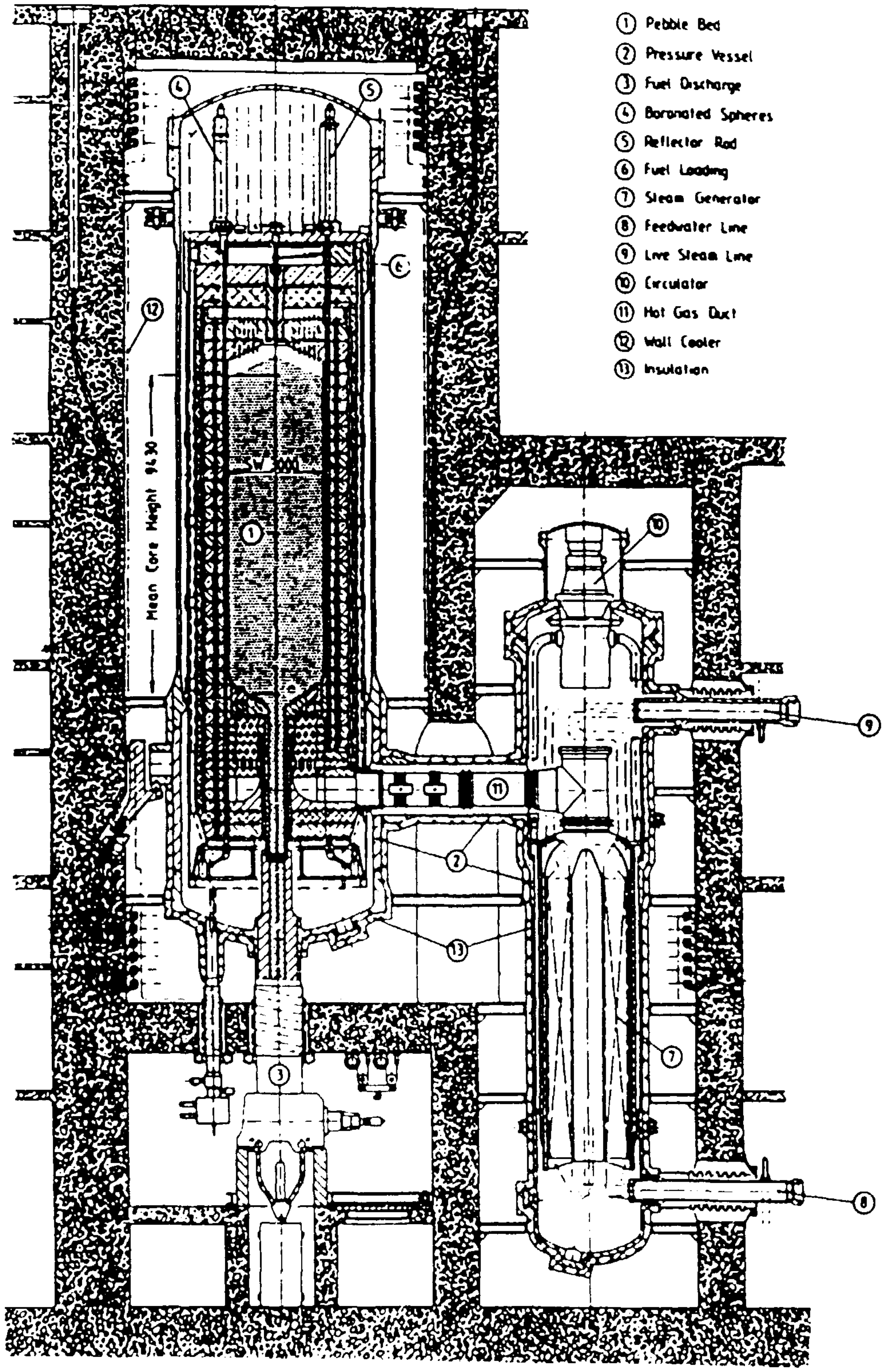
-	Core height	9.6 m
-	Core diameter, m	3.0 m
-	Mean power density	3.0 MW/m^3
-	Number of fuel elements	360 000
-	Fuel cycle	U/Pu
-	Heavy metal loading per fuel element	7 g
-	Enrichment	7.8 %
-	Fuel in-core period	1 000 days
-	Fuel burn up	80 MWd/kg
-	Coolant temperature (inlet/outlet)	250/700 °C
-	Mean helium pressure	60 bar
-	Helium mass flow	85 kg/s

Steam generator (one modular unit)

-	Live steam pressure/temperature	190 bar/525 °C
-	Live steam mass flow	82 kg/s

References

Small and medium reactors. II Technical Supplement, Nuclear Energy Agency, OECD, Paris, France, 1991.



Figur CI1
HTR-Modul: Nuclear steam supply system.

CI2

Format	Description
Title	MHTGR
Application of the reactor	NPP, NHPP
Reactor type	HTGR
Power output	Medium power/190 MWe
Organization (name)	General Atomics and others, US
Development status	Basic design

Description

The MHTGR concept has a unit thermal power size of 450 MWt in comparison with 200 MWt for HTR-Modul (See CI1). The block type fuel is arranged as an annular cylinder to keep the core temperatures under the 1 600 °C limit. The power density can be raised to 6 MW/m^3 (for HTR-Modul only 3 MW/m^3) due to this geometrical form.

The side by side arrangement is identical to the German HTR-Modul to prevent the ingress of water as well as overheating of metallic components and to permit hot stand-by operation with subsequent smooth restarting. Decay heat removal of MHTGR is normally achieved by an additional cooling circuit.

The MHTGR development strategy is the following:

- o No reactor safety systems
- o No operator action required
- o Insensitive to incorrect operator action

The MHTGR can withstand a loss of helium coolant in combination with the loss of all forced circulation from full power without fuel temperatures reaching the limit level.

The reactor is embedded underground, thus reducing the need for further sheltering against air plane crash or sabotage. The reactor cavity cooling is done by natural air convection. In hypothetical accidents the decay heat can also be absorbed by the surrounding earth.

Schedule

The design requires a prototype demonstration plant. No MHTGR reactor plant has been ordered yet. No current development work.

MHTGR Unit Data

Configuration (electricity generation)	4 reactor modules and 4 turbine/generator sets
Thermal power	1 800 [4x450] MWt
Net electrical output	692 MWe
Thermal power per module	450 MWt
Helium pressure	64 bar
Helium temperature (cold/hot	260/690 °C
Design lifetime	40 years
Design basis operation, capacity factor	84 %
Other applications	Co-generation, process heat
Core height	9.45 m
Core diameter (outer/inner)	3.2/11.5 m
Fuel element	Prismatic hex-block
Core configuration	84-column annual cylinder
Fissile material	Uranium oxycarbide
Power density	5.99 W/cm^3
Average enrichment	19.9 % U-235
Fertile material	UO_2
Control rods	30 (6 inner, 12 in-core, 18 outer reflector rods)
Reserve shutdown channels	12 (in-core)
Reactor vessel (diameter/height)	6.8/22 m
Steam generator	1, once-through helical, upflow boiling
Main circulator	1, electric motor-driven, magnetic bearings
Shutdown cooling heat exchanger	1, once-through helical
Shutdown circulator	1, electric motor-driven, magnetic bearings

References

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MHTGR design status and perspective in the USA.
Proceedings Ineternational Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 24-29, 1992, Tokyo, Japan, vol 1, p 7.2.1.

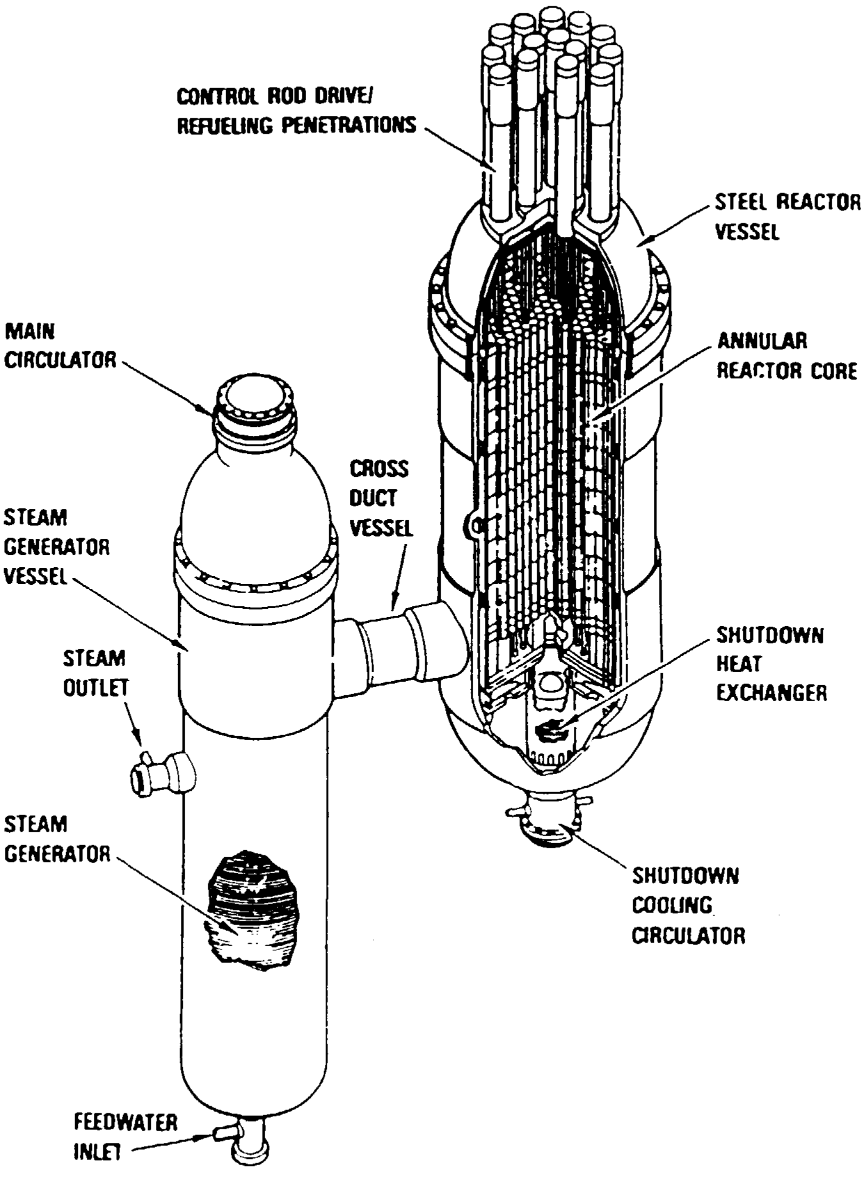

Figure CI2
MHTGR: Reactor Plant Modular.

CI3

Format	**Description**
Title	GT-MHR (Gas Turbine-Modular Helium Reactor)
Application of the reactor	NPP, NHPP
Reactor type	HTGR
Power output	Medium power/300 MWe
Organization (name)	General Atomics, US
Development status	Studies

Description

The GT-MHR reactor plant with a 600 MW output represents an innovative approach developed by General Atomics. This concept is the result of coupling the evolution of a small passively safe modular helium cooled reactor with key technology development in the US during the last decade: large industrial gas turbines, large active magnetic bearings, and compact, highly effective plate-fin heat exchangers.

The main stages of the GT-MHR plant evolution are the following:

o 350 MWt MHTGR Steam Cycle (1985)
- provides passive safety and standardized modular design

o 450 MWt MHTGR Steam Cycle (1990)
- retains passive safety and modular design
- reduces power costs comparable to coal and ALWR

o 550 - 600 MWt GT-MHR (1994)
- retains passive safety and modular design (see CI1)
- increases efficiency (48 % vs 32 % for LWR)
- reduces thermal and radioactive wastes
- reduces power costs (as plant output increases)

GT-MHR is showing a promising potential for a significant reduction in power generating costs. The reactor system is based on the steam cycle MHTGR developed over the last 10 years. It retains many of the key design features including the ceramic coated TRISO fuel, with its capability to retain fission products up to very high temperatures, low power density annular core, factory fabricated steel vessels, and entirely passive decay heat removal. The key difference from the steam cycle design is that the GT-MHR uses the Brayton cycle to produce electricity directly with the primary helium coolant driving the turbine generator. The precooler and intercooler are helium-to-water heat exchangers which operate in a very benign environment (metal temperatures are less than 121 °C.

Schedule

No GT-MHR reactor plant has been ordered yet. Conceptual design work is under way.

GT-MHR Unit Data

Plant parameters	
Module power level	550 - 600 MWt
Core inlet/outlet temperature	490 °C/850 °C
Core inlet pressure	70 atm
Net efficiency	47.7 %
Reactor parameters	
Active core columns	102
Control	
- Start-up rods	12
- Operating rods	36
- RSCE channels	28
Vessel parameters	
Inner diameter	7.2 m
Outer diameter at flange	8.4 m
Material	9 Cr-1 Mo-V
Fuel	
Average/maximum temperature	
- Normal operation	855 °C/1 200 °C
- Accident	1 270 °C/1 554 °C

References

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Design status of the Gas Turbine Modular Helium Reactor (GT-MHR)
Proceedings Nuclear Power & Industry International Conference,
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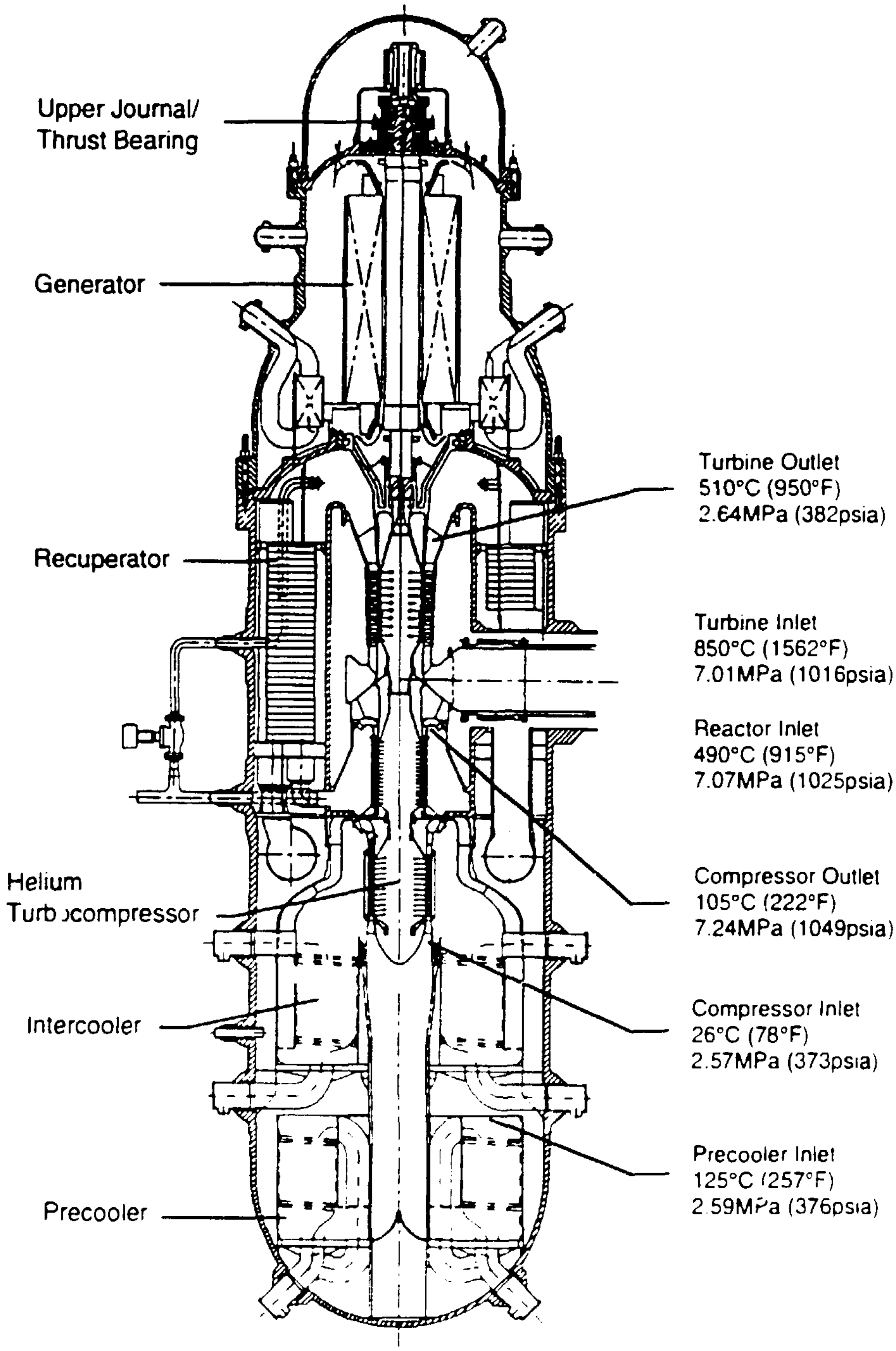

Figure CI3
GT-MHR: Power conversion vessel assembly.

CI4

Format	Description
Title	HTGR-MHD
Application of the reactor	NPP
Reactor type	HTGR
Power output	Medium power/860 MWe
Organization (name)	JAERI, Japan
Development status	Studies

Description

A conceptual study on a combination of high temperature gas-cooled reactor and magneto-hydrodynamic (MHD) power generators has been considered by JAERI and others aiming at achieving a high efficiency of around 60 % for a core exit gas temperature at 2 300 K. An MHD direct power conversion system needs no moving or rotating component exposed to a working fluid at a very high temperature.

The proposed HTGR-MHD plant consists of a 1 500 MWt HTGR and four identical helium loops. The helium gas serves as both the coolant for the HTGR core and the working fluid of the MHD power conversion. Each loop has an MHD generator, a regenerative heat exchanger (RHX), a pre-cooler, an inter-cooler, and a turbine-compressor set. The helium gas exits the HTGR at 2 300 K, is seeded with cesium, and then enters the MHD generator to produce electricity. The RHX recovers the heat remained in the generator exit gas to drive the turbine-compressor for gas circulation. An auxiliary cooling system is provided as an independent means of removing the core decay heat.

The HTGR-MGD reactor plant generates electricity with an efficiency of about 57 %.

The major R&D issues identified for the HTGR-MHD concepts are:

- Further improvement in high-temperature capability and in the life-time of the coated fuel particle (up to ~ 2600 K)
- Thorough understanding of behaviour of non-equilibrium plasma and electrode phenomena
- Optimization of generator configuration and electrode design

- Development of a helium compressor with a high pressure ratio and a high-temperature (~ 1 500 K) helium turbine
- Development of a large capacity (~ 270 MWt), compact RHX
- Development of a large diameter (~ 1.2 m), high-temperature piping and valves
- Development of a Cs feed and recovery system

Schedule

Significant design and experimental work remains to be carried out before application. The conceptual design study is under way.

HTGR-MHD Unit Data

Rated reactor power	1 500 MWt
Fuel type	Spherical pebble
Pebble diameter	50 mm
Core diameter/height	5.0 m/2.0 m
Core power density	38.2 MW/m^3
Core inlet pressure	7.9 bar
Core pressure drop	1.5 bar
Coolant core velocity	19.5 m/s
Coolant flow rate	204.8 kg/s
Coolant temperature in/out	919/2 300 K
Maximum fuel temperature	2 600 K

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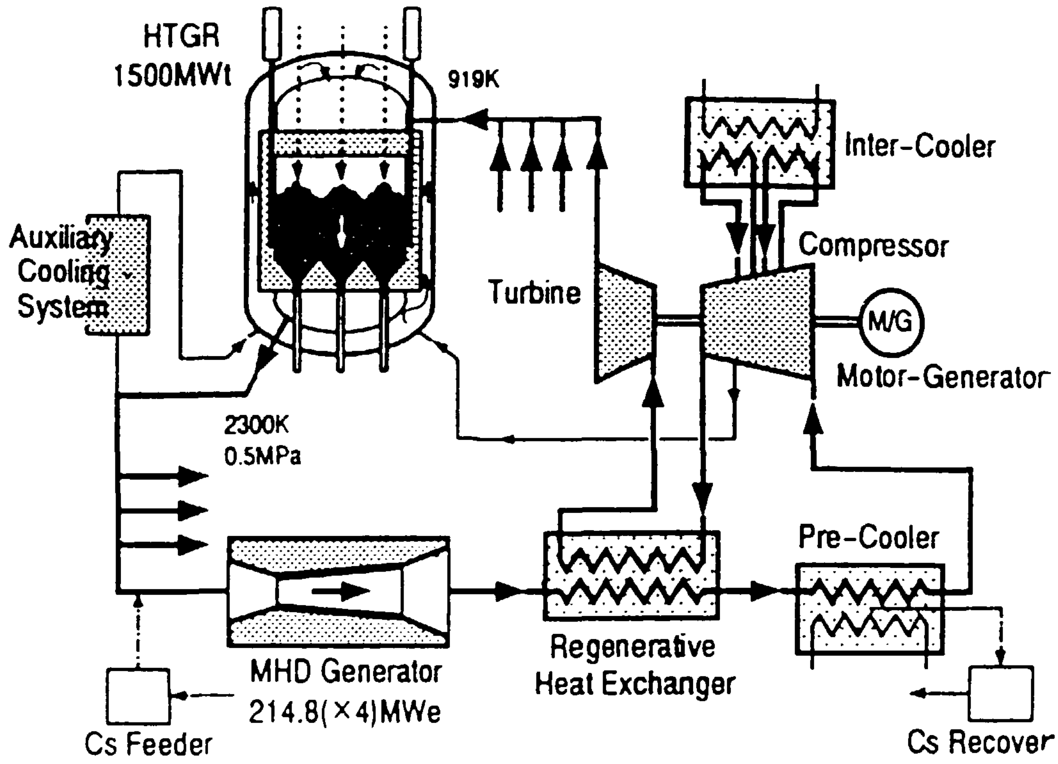

Figure CI4
HTGR-MHD: Plant flow diagram.

D Molten Salt Reactors

D0 Introduction

Molten salt reactors (MSR) are fluid fuel reactors. One of the advantages of fluid fuel reactors over solid fuel reactors is the ability to carry out on-line fueling and fuel processing. Specifically, the MSRs that are fueled with UF_4 have the fluoride volatility process available to them, which enables simple, proven, full processing.

Molten salt reactors can be designed with or without on-line processing, or with various degrees of processing. For fluid fuel reactors with processing, some of the common nuclear reactor terminology is not applicable. Except for some start-up periods, the reactors are operated at an equilibrium steady state. The fuel concentration and content do not vary with time. Fissioned fuel is replenished by feeding or by breeding. The term fuel burnup thus has no meaning as there is no specific amount of energy generation related to a particular identifiable original amount of fissile material. For the same reason, there is no excess reactivity to compensate for burnup or for the continuous poisoning caused by fission products accumulation in solid fuel reactors. Also, fluid fuel does not suffer from fuel damage, nor is a meltdown possible for a fuel that is already molten. Furthermore, when the fluid fuel itself is circulated out of the core to an external heat exchanger, fluid fuel reactors cannot have a loss-of-coolant accident, they do not have a separate coolant that will leave the fuel to generate heat with no cooling. Also decay heat problems for the MSR are not as strong in comparison with traditional solid fuel ones because of small, 1 - 2 % mole, UF_4 concentrations in the fuel salt.

MSRs were investigated intensively at Oak Ridge National Laboratory (ORNL, USA) and summarized in a design study of molten salt-breeder reactor (MSBR) 25 years ago. Construction material problems and reactor chemical aspects including fission product behaviour have been basically solved by using beatifully operated experimental reactor MSRE (ORNL) and many test loops in ORNL and RRC "Kurchatov Institute" (Russia). Today the worldwide research on MSR development is quite small.

This innovative reactor type is under conceptual consideration in the USA, Russia and Japan not only for NPPs, but also for high temperature industrial heat supply and for the utilization of weapon-grade Pu and transmutation of the long-lived fission products.

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DI1

Format	Description
Title	USR (Ultimate Safe Reactor)
Application of the reactor	NPP
Reactor type	MSR
Power output	Medium power/625 MWe
Organization (name)	ORNL, US
Development status	Studies

Description

The USR utilizes an innovative form of asymptotic safety that eliminates hazards (excess reactivity, decay heat and chemical energy) rather than preventing them or mitigating the consequences.

The USR reactor plant with a thermal power of 1 420 MWt is a reactor that eliminates the customary safety concerns of nuclear fission reactors. The USR concept has an insignificant source term and no reasonable criticality accident. Furthermore, the negligible residual after-heat in the reactor renders its shutdown capability comparable or superior to conventional power sources in that no actions or precautions are required following a shutdown of power.

The USR plant utilizes two principles to achieve ultimate safety. Fission products are continuously removed at the rate that they are produced, thus retaining the inherent source term at an insignificant level. The reactor is operated with no excess criticality, hence no criticality accident is reasonably possible. The reactor is safely controlled by its negative temperature coefficient. The reactor maintains criticality by an internal breeding ratio that is trimmed to be exactly one.

To facilitate the continuous fission product removal, the USR reactor plant needs an extremely short time (1 - 6 hours) for processing of fuel loading.

The USR plant has a traditional loop arrangement of the primary circuit. Centrifugal RCPs pump LiF-BeF_2-ThF_4 fuel through the cylindrical homogeneous core.

The passive inherent safety features of the USR reactor plant make it a potentially economical competitor with high reliability, thereby making it an attractive option for future energy generation.

Schedule

The USR concept is very far from commercial application. Significant research and design work need to be done before commercial application. No current development work is being carried out.

USR Unit Data

Fuel salt	71.6 LiF-6 BeF_2-12 ThF_4-0.4 UF_4 mol %
Secondary salt	8NaF-92$NaBF_4$ mol %
Thermal power	1 420 MWt
Mass flowrate	5.6 tonne/s
Volume flowrate	1.7 m^3/s
Inlet temperature	510 °C
Outlet temperature	700 °C
Power density	100 MW/m^3
Core volume	4.2 m^3
Core inner diameter	2.6 m
Core height	2.7 m
Time for salt to pass through cores	8.3 s
Salt velocity in core	0.33 m/s
Conversion/Breeding ratio	1

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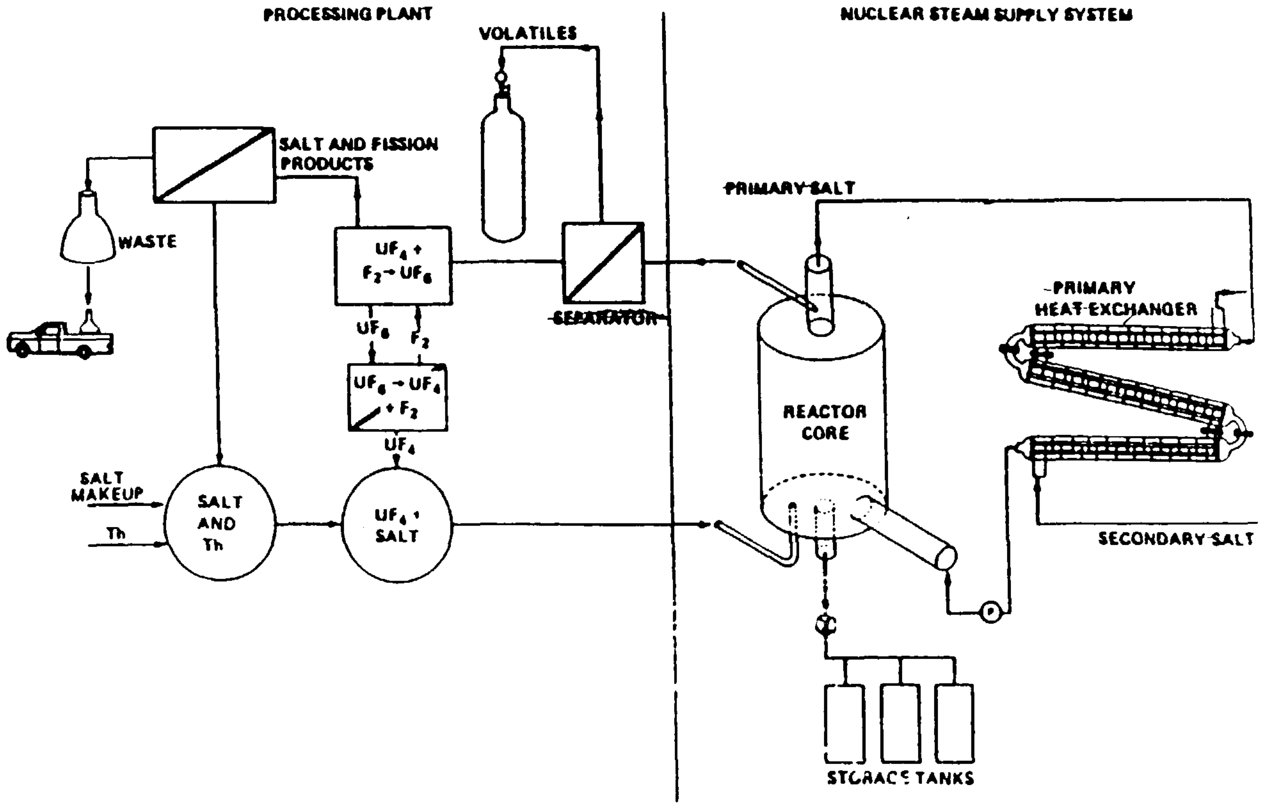

Figure DI1
USR: Schematic diagram.

DI2

Format	**Description**
Title	FUJI-Pu
Application of the reactor	NPP, WTS
Reactor type	MSR
Power output	Small power/100 MWe
Organization (name)	Tokia University, Japan
Development status	Studies

Description

The FUJI-Pu reactor plant with a power output of 250 MWt fuelled with the plutonium recovered from the spent fuel of LWRs has the following performance:

- o Fuel self-sustaining characteristics, which means, in practice, a conversion ratio of 0.9. The amount of initial inventory and additional inventory in the transient stage is also small. It results in a few fuel transportation requirements.

- o Simple structure, easy operation and maintenance work enhance its safety and economy, because of no opening of the reactor vessel, no fuel assembly fabrication, no fuel-processing in full life except the removal of Kr, Xe and T, achieving a burn-up of 500 %, nearly no control rods required due to the very low excess reactivity.

For FUJI-Pu the fuel salt composition in the initial load is (in mol %) 71.8 LiF-16 BeF_2-12 ThF_4-0.2 PuF_3. The reactor core is of the graphite-moderated type in which the single fuel salt-containing fissile and fertile materials serve as the core fluid and the blanket fluid.

The temperature of the fuel salt at the outlet of the reactor is 665 °C capable of generating the super heated steam at a pressure of 25 MPa and temperature of 565 °C. Hastelloy NM is used as the construction material for the loop type primary circuit.

This system could also be effective for TRU-incineration and anti-terrorism.

Schedule

Significant design work remains to be carried out before application.

FUJI-Pu Unit Data

Thermal capacity		250 MWt
Thermal efficiency		40 %
Reactor vessel		
-	Diameter	5.95 m
-	Height	4.99 m
Core-zone I		
-	Maximum radius	127 m
-	Fuel ratio	7 vol %
Core-zone II		
-	Maximum radius	1.97 m
-	Fuel ratio	10 vol %
Blanket zone		
-	Thickness (radial direction)	0.33 m
-	Thickness (axial direction)	0.40 m
-	Fuel ratio	20 vol %
Reflector		
-	Thickness	0.50 m
-	Fuel ratio	1 vol %
Power density		
-	Average	4 kW/l
.	Peak - initial	11 kW/l
Volume of fuel salt		
-	In reactor vessel	11.2 m^3
-	In primary system (sum)	16.7 m^3
-	Flow rate	0.55 m^3/s

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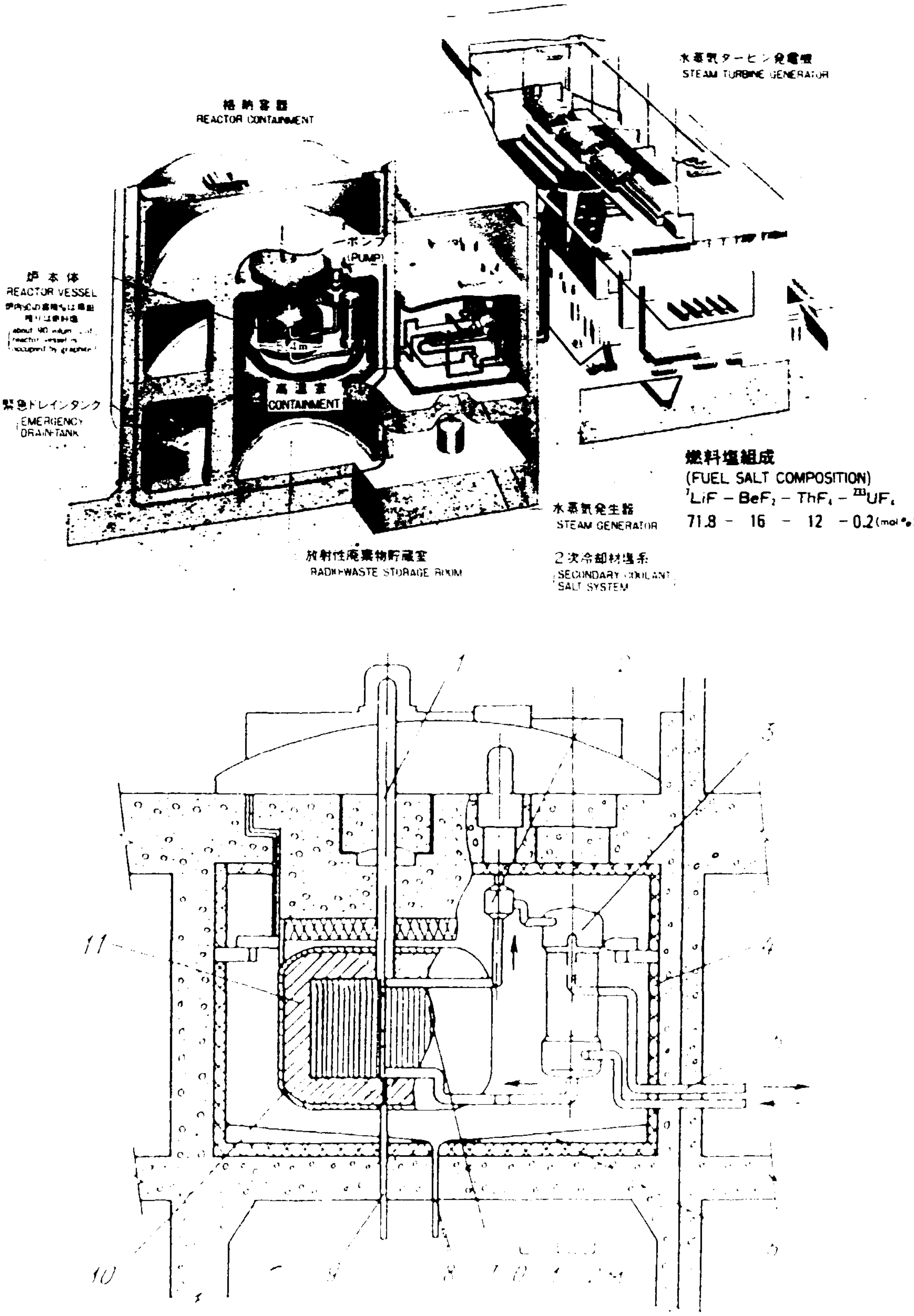

Figure DI2

FUJI-Pu: Reactor core section and building.

1 Control rods, 2 Fuel pump, 3 Heat exchanger, 4 Reactor box, 5 Intermediate NaF-NaBF$_4$ circuit, 6 Core catcher, 7 Core, 8,9 Drain pipes, 10 Reactor vessel, 11 Reflector

DI3

Format	Description
Title	MSR-NC
Application of the reactor	NPP
Reactor type	MSR
Power output	Medium power/470 MWe
Organization (name)	RRC "Kurchatov Institute", RF
Development status	Studies

Description

The MSR-NC reactor plant with a 1 070 MWt output represents an innovative approach being developed by RRC Kurchatov Institute.

After the MSBR concept was published, new designs of MSR concepts appeared aimed at reaching a complete self-protection of the reactor. They use the same structural materials as in the MSBR design.

The MSR-NC design differs from MSBR in the complete absence of continuous fuel regeneration, fuel circuit due to the natural convection as well as to the rearrangement of the core and the circuit. In particular, the MSR permits the graphite as a moderator to be completely excluded and only to be used as a reflector. In addition, the integrated arrangement of the fuel circuit is proposed, which facilitates solving the problem of radioactivity release from the primary circuit.

As applied to the MSR-NC with the homogeneous core, the self-protection is reached due to:

- o Negative effects of reactivity resulting from the increase in the fuel-coolant temperature (practically the only component of the core), $\alpha = 6.42 \cdot 10^{-5}/^{o}C$
- o Use of the free convection of the LiF-BeF_2-ThF_4-UF_4 fuel in the nominal power
- o High heat-accumulating properties of the primary circuit
- o Use of a passive emergency heat exchanger permitting the decay heat to be removed from the outer surface of the reactor vessel to the environment

Schedule

No current development work is being carried out.

MSR-NC Unit Data

Power output	1 070 MWt
Fuel composition	71LiF-27.9BeF_2-1ThF_4-0.075UF_4 mol %
Coolant salt composition	92$NaBF_4$-8NaF mol %
Fuel salt inlet/outlet temperatures	901 K/1 023 K
Coolant salt inlet/outlet temperatures	699 K/883 K
Reactor vessel outside diameter and height	5.1/18 m
Core diameter and height	4/4 m
Graphite reflector thickness	0.4 m
Fuel flow rate	3.55 t/s
Average fuel circuit power density	7.2 W/cm^3

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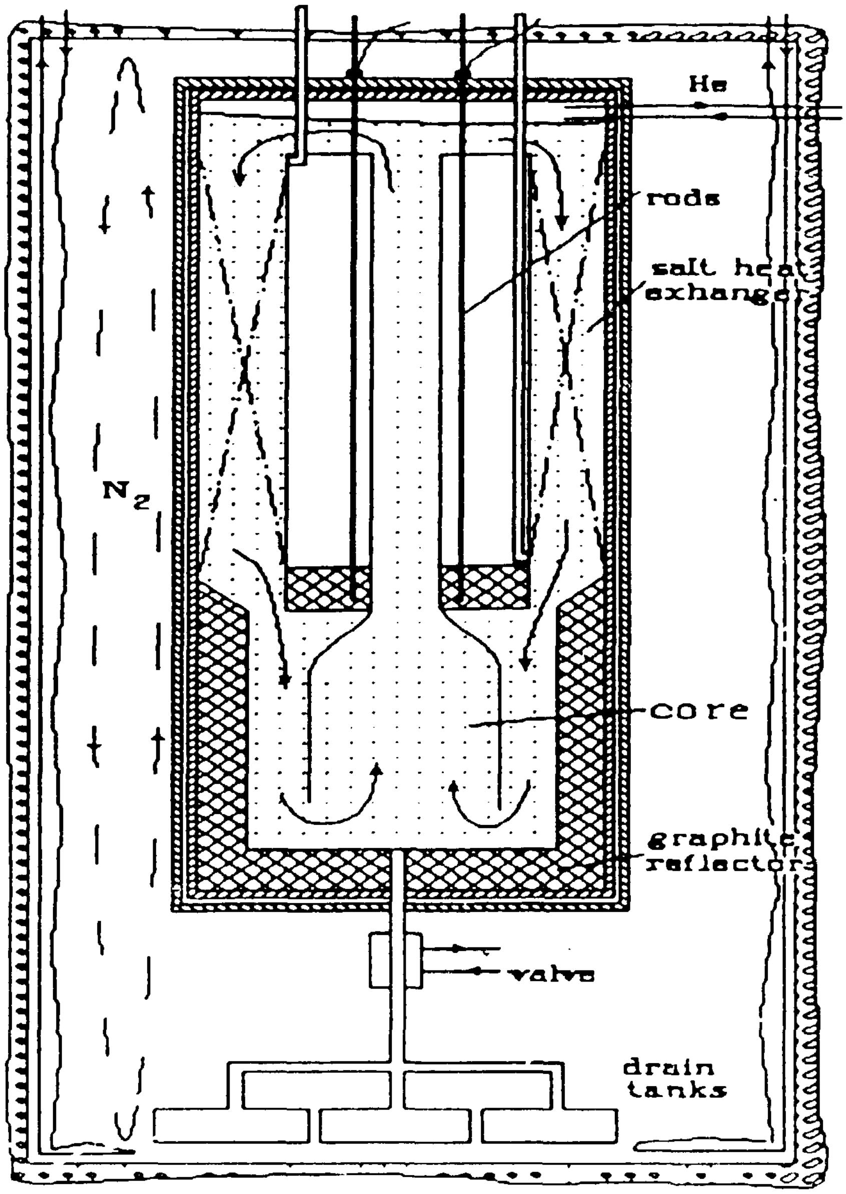

Figure DI3
MSR-NC: Reactor plant arrangement.

List of Abbreviations and Glossary of Terms

A

ABB	Asea Brown Boveri
ABWR	Advanced Boiling Water Reactor, see BWR
Accelerator	A device that increases the velocity and energy of charged particles such as electrons and protons; also referred to as a particle accelerator. In a "linear" accelerator, particles are accelerated in a straight path
Actinides	The elements with atomic numbers above 88 (actinium, element 89). The actinides series includes uranium, atomic number 92, and all the man-made transuranic elements
Active component	Any component that is not passive is active
Active system	Any system that is not passive is active
ADP	Advanced Double Pool reactor
ADS	Automatic Depressurization System
AEC	(U.S) Atomic Energy Commission
AECL	Atomic Energy of Canada Limited. A designer/supplier of nuclear reactors
AGR	Advanced Gas-Cooled Reactor
ALARA	As low as reasonable achievable
ALMR	Advanced Liquid Metal Reactor
ALSEP	Apollo Lunar Scientific Experiment Package
ALWR	Advanced Light Water Reactor, see LWR
AMSB	Accelerator Molten Salt Breeder
amu	Atomic mass unit
ANL	Argonne National Laboratory, US
AP	Advanced Passive
APWR	Advanced Pressurized Water Reactor, see PWR
ATR	Advanced Thermal Reactor: A heavy water moderated, light water-cooled reactor
ATWS	Anticipated Transients Without Scram
Availability Factor	The availability factor of a nuclear unit or station is the ratio of time when energy can be produced to the total time
AWTS	Accelerator-driven Waste Transmutation System

B

Base Load	The minimum load produced over a given period. A station used for base load is a station that is normally operated when available to provide power to meet the minimum load demands
BDA	Beyond Design Accident
BMDO	Ballistic Missile Defence Organization, US
BN	Russian version of sodium cooled Fast Breeder Reactor
BNL	Brookhaven National Laboratory
BOL	Beginning Of life
BOM	Beginning Of mission
BOP	Balance Of Plant
Breeder Reactor	A nuclear reactor that produces more fissile material than it consumes. In fast breeder reactors, high-energy (fast) neutrons produce most of the fissions, while in thermal breeder reactors, fissions are principally caused by low-energy (thermal) neutrons. See Fast reactor
Breeding Ratio	The conversion ratio when it is greater than unity. A high breeding ratio results in a short doubling time
BWR	Boiling-Water Reactor. A light-water reactor that employs a direct cycle; the water coolant that passes through the reactor is converted to high pressure steam that flows directly through the turbines

C

CANDU	CANadian Deuterium Uranium reactor: A type of heavy water reactor
Capacity Factor	See Load Factor
CAREM	Conjunto Argentina de Reactores Modulares
CCV	Cooldown Control Valve
CDA	Control Drum Actuator
CE	Combustion Engineering
CEA	Commissariat à l'Energie Atomique
CEC	Commission of the European Communities
Charge of a reactor	The fuel placed in a reactor
CHP	Combined Heat and Power

Conversion as used in reactor technology	The ratio between the number of fissile nuclei produced by conversion to the number of fissile nuclei destroyed. If the ratio for a given reactor is greater than one, it is a breeder reactor; if it is less than one, it is a converter reactor
Conversion, chemical	The operation of altering the chemical form of a nuclear material to a form suitable for its end use
Converter reactor	A reactor that produces some fissile material, but less than it consumes
Coolant	The medium in a nuclear reactor which removes heat from the reactor core where fission occurs and heat is produced, and transfers it to systems which convert the heat into steam
COPUOS	(UN) Committee On Peaceful Uses of Outer Space
Critical	Capable of sustaining a nuclear chain reaction
CRU	Combined Rotating Unit
CWCS	Containment Water Cooling System

D

DARPA	Defense Advanced Research Project Agency
DBA	Design Basis Accident
DEC	Direct Energy Conversion
Decay Heat	The heat produced by the decay of radioactive nuclides
DIPS	Dynamic (Radio)isotope Power System
DNHPP	Decentralized Nuclear Heating Power Plant
DOE	US Department of Energy

E

E-MAD	Engine Maintenance, Assembly and Disassembly
EASEP	Early Apollo Scientific Experiment Package
ECCS	Emergency Core Cooling System
EFR	European Fast Reactor
EHRS	Emergency Heat Removal System
EM	Electromagnetic
EMP	Electromagnetic Primary Pump

EMP	Electromagnetic Pump
Enriched fuel	See Fuel, enriched
Enrichment	i) the fraction of atoms of a specified isotope in a mixture of isotopes of the same element when this fraction exceeds that in the naturally occurring mixture; ii) any process by which the content of a specified isotope in an element is increased
EOL	End Of Life
EOM	End Of Mission
EOT	Earth Orbital Terminal
EPA	(U.S.) Environmental Protection Agency
EPR	European Pressurized Reactor
EST-1	Engine Test Stand Number One
eV	Electron volt
EVA	Extravehicular Activity
EWST	Emergency Water Storage Tank

F

Fail-safe	The term describes the behaviour of a component or system, following a failure (either internal or external). If a given failure leads directly to a safe condition, the component or system is fail-safe with respect to that failure
Fast neutrons	See **Neutron, fast**
Fast reactor	A nuclear reactor in which no moderator is present in the reactor core or reflector. So the majority of fissions are produced by fast neutrons. If a fertile species is present in the fast reactor core or in the blanket surrounding the core, it will be converted into fissile material by neutron capture. When more fissile material is produced than is used to maintain the fission chain, the reactor is called a breeder
Fault-/error-tolerant	(also called forgiving). The term fault-/error-tolerant, also called forgiving, describes the degree to which equipment faults/human inaction (or erroneous action) can be tolerated
FBR	Fast Breeder Reactor (see Fast reactor)
FDA	Final Design Approval
FEPS	Federation of Electric Power Companies

FFTF	Fast Flux Test Facility
Fissile	i) of a nuclide: capable of undergoing fission by interaction with slow neutrons;
	ii) of a material: containing one or more fissile nuclides
Fission products	Nuclides produced either by fission or by the subsequent radioactive decay of the nuclides thus formed
Fissionable	i) of a nuclide: capable of undergoing fission by any process;
	ii) of a material: containing one or more fissionable nuclides
FOAKE	First-Of-A-Kind-Engineering
Foolproof	Safe against human error or misguided human action
Fossil fuel	A term applied to coal, oil and natural gas
FP	Fission Product(s)
FPSE	Free Piston Stirling Engine
FSAR	Final Safety Analysis Report
Fuel cycle	The sequence of processing, manufacturing, and transportation steps involved in producing fuel for a nuclear reactor, and in processing fuel discharged from the reactor
Fuel, enriched	Nuclear fuel containing uranium which has been enriched in one or more of its fissile isotopes or to which chemically different fissile nuclides have been added
Fuel, nuclear	Material containing fissile nuclides which when placed in a reactor enables a self-sustaining nuclear chain to be achieved
Fuel reprocessing	The chemical or metallurgical treatment of spent (used) reactor fuel to recover the unused fissionable material, separating it from radioactive waste. The fuel elements are chopped up and chemically dissolved. Plutonium and uranium and possibly other fissionable elements are then separated for further use

G

GA	General Atomic
GCR	Gas-Cooled Reactor
GE	General Electric
GEO	Geostationary Earth Orbit, Geosynchronous Orbit
GHR	Gas-Cooled Heating Reactor

GIS	Graphite Impact Shell
GLFC	Graphite Lunar Module Fuel Cask
GmbH	German term signifying "Limited"
Grace period	The period of time which a safety function is ensured without the necessity of personnel action in the event of an incident/accident
GSOC	Geosynchronous Orbit Complex
GWe	Gigawatt (10^9 watts) electric

H

Half-life	The period of time required for the radioactivity of a substance to drop to half its original value; the time that it takes for half of the atoms of a radioactive substance to decay. Measured half-lives vary from millionths of a second to billions of years
Heavy water	Deuterium oxide (D_2O): water containing significantly more than the natural proportion (1 in 6500) of heavy hydrogen (deuterium) atoms to ordinary hydrogen atoms
HEO	High Earth Orbit
HEU	Highly-Enriched Uranium
HR	Heat Rejection
HSBWR	Hitachi Small Boiling Water Reactor
HTR or **HTGR**	High-Temperature Gas-Cooled Reactor. A graphite-moderated, helium-cooled advanced reactor that utilizes low enriched uranium
HTTR	High Temperature Test Reactor
HWR	Heavy-Water Reactor. Heavy water is used as a moderator in certain reactors because it slows down neutrons effectively and also has a low cross section for absorption of neutrons
HX	Heat Exchange
Hydropress EDO	Reactor Plant Design Bureau, Podolsk, Russia

I

I&C	Instrumentation & Control system
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEA	International Energy Agency
IFR	Integral Fast Reactor
IHS	Intermediate Heat Exchange
IIASA	International Institute for Applied System Analysis
INET	Institute Nuclear Energy Technology, China
Inherent safety characteristics	Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept
INSRP	Interagency Nuclear Safety Review Panel
Isotopes	Nuclides having the same atomic number (i.e. identical chemical element) but different mass numbers; e.g. 92-Uranium-235 and 92-Uranium-238. Isotopes have the same chemical properties but slightly different physical properties
IUS	Inertial Upper Stage

J

JAERI	Japan Atomic Energy Research Institute

K

KEPCO	Kansai Electric Power Company
keV	Thousands electron volts
kT	Kiloton
kW	Kilowatt
kWh	Kilowatt hour (10^3 watt hour)

L

LANL	Los Alamos National Laboratory
LCP	Large Communications Platform
LEO	Low Earth Orbit
LET	Linear Energy Transfer
LEU	Low Enrichea Uranium
LH_2	Liquid hydrogen
LiH	Lithium hydrogen
LINAC	Linear Accelerator
Linear accelerator	A long straight tube (or series of tubes) in which charged particles (ordinarily electrons or protons) gain in energy by action of oscillating electromagnetic fields
LLFP	Long Lived Fission Products
LMCPBR	Liquid Metal Cooled Pebble Bed Reactor
LMFBR	Liquid Metal Fast Breeder Reactor. A fast reactor that employs liquid metal (sodium) as a coolant The sodium in the primary loop passes through the reactor and transfers its heat to sodium in a secondary loop. This sodium then heats water in a tertiary loop which produces steam and drives a turbine. See also Fast Reactor
Load Factor	The load factor of a nuclear unit or station for a given period of time is the ratio of the energy that is produced during the period considered to the energy that it could have produced at maximum capacity under continuous operation during the whole of that period. Also called Capacity Factor
LOCA	Loss Of Coolant Accident
LOF	Loss Of Flow
LOHS	Loss Of Heat Sink
LOT	Lunar Orbital Terminal
LRC	Lunar Resources Complex
LWGR	Light Water Graphite Reactor. A reactor that uses ordinary water as a coolant and graphite as a modertor and utilizes slightly enriched Uranium-235 (e.g. Russian multi-channel RBMK reactor plant)
LWR	Light-Water Reactor. A nuclear reactor that uses ordinary water as both a moderator and a coolant and utilizes slightly enriched Uranium-235 fuel. There are two commercial LWR types: the boiling-water reactor (BWR) and the pressurized water-reactor (PWR)

M

MAP	Minimum Attention Plant
MARS	Multipurpose Advanced Reactor Inherently Safe
MCP	Main Circulation Pump
MeV	Million electron volts
MHTGR	Modular HTGR, see HTR
MHW	Multihundred watt
Minor actinides	The transuranic elemer ... ninus plutonium. Usually this term is used to refer to neptunium ... icium, and curium. Some also refer to these as the "minor" tran ... ics. Plutonium is the dominant transuranic, but these minor tra ... anics contribute comparable radioactivity in spent fuel
Moderator	A material, such as ordinary water, heavy water, beryllium, graphite and some others used in a nuclear reactor to slow down fast neutrons so fissile nuclei can more easily and efficiently capture them, thus increasing the likelihood of further fission
MOX fuel element	Mixed Oxide fuel element. Fuel element in which fuel is an intimate mixture of uranium and plutonium oxides
MPD	Maximum Permissible Dose
MSBR	Molten Salt Breeder Reactor, see MSR
MSPWR	Mitsubishi Simplified PWR
MSR	Molten Salt Reactor. A nuclear reactor that uses fluid fuel, e.g. molten salt fluorides of Li, Be, Th, U.
MW	Megawatt
MWe	Megawatt (10^6 watts) electric
MWh	Megawatt hour (10^6 watts hour)
MWt	Megawatt (10^6 watts) thermal

N

NaK	Sodium-potassium eutectic mixture
NASA	(U.S.) National Aeronautics and Space Administration
NC	Natural Circulation
NCV	Nozzle Control Valve

NEA	OECD Nuclear Energy Agency
NEP	Nuclear Electric Propulsion
NEPSTP	Nuclear Electric Propulsion Space Test Program
NERVA	Nuclear Engine for Rocket Vehicle Application
Neutrons, fast	Neutrons of kinetic energy greater than some specified value. This value may vary over a wide range and will be dependent upon the application, such as reactor physics, shielding, or dosimetry. In reactor physics the value is frequently chosen to be 100 000 eV (electron-Volt)
Neutrons, slow	Neutrons of kinetic energy less than some specified value (see neutrons, fast). In reactor physics the value is frequently chosen to be 1 eV
Neutrons, thermal	Neutrons in thermal equilibrium with the medium in which they exist
NF	Nuclear Furnace
NHP	Nuclear Heating Plant
NHPD	Nuclear Heat and Power Plant
NPP	Nuclear Power Plant. A reactor or reactors together with all structures, systems and components necessary for the production of power (i.e. heat or electricity)
NPPS	Nuclear Power Propulsion System
NRC	(Nuclear Regulatory Commission): US body regulating the use of nuclear energy
NRDS	Nuclear Rocket Development Station
NSSS	Nuclear Steam Supply System
NTS	Nevada Test Site
Nuclear energy	Energy released in nuclear reactions or transitions
Nuclear fuel	See Fuel, nuclear
Nuclear reaction	A reaction involving a change in an atomic nucleus, such as fission, fusion, neutron capture, or radioactive decay, as distinct from a chemical reaction, which is limited to changes in the electron structure surrounding the nucleus
Nuclear reactor	A device in which a fission chain reaction can be initiated, maintained, and controlled. Its essential component is a core containing fissionable fuel. It is sometimes called an atomic "furnace"; it is the basic machine of nuclear energy
Nuclide	Any species of atom that exists for a measurable length of time. The term is used synonymously with isotope. A radionuclide is a radioactive nuclide.

O

OECD	Organisation for Economic Co-Operation and Development
OKBM	Reactor Plant Design Bureau, Nizni-Novgorod, Russia
OMEGA	Options for Making Extra Gains from Actinides and Fission Products
OTV	Orbital Transfer Vehicle

P

Passive component	A component which does not need any external input to operate
Passive system	Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation
PBIS	Pressure Balanced Injection System
PbTe	Lead telluride
PC	Power Convertor
PCCS	Passive Containment Cooling System
PCI	Pellet Clad Interaction
PCIV	Prestressed Cast Iron Vessel
PCRV	Prestressed Concrete Reactor Vessel
PCS	Power Conversion System
PCV	Pressure Containment Vessel
Peak load	The maximum load produced by a unit or group of units in a stated period of time. A station used for peak load generation is a station that is normally operated to provide power during maximum load periods only
Pebble bed reactor	A reactor, which utilizes spherical fuel elements
Per capita	Per unit of a population
PHRS	Passive Heat Removal System
PIUS	Process Inherent Ultimate Safety Reactor
Plutonium	A heavy, radioactive, man-made metallic element with the atomic number 94, created by absorption of neutrons in uranium-238. Its most important isotope is plutonium-239, which is fissile
PPO	Pressed Plutonium (-238) Oxide

Primary energy	The energy content of fuels before they are processed and converted into forms used by the consumer. Primary energy refers to energy in the form of natural resources: water flowing over a dam, freshly mined coal, crude oil, natural gas, natural uranium. Only rarely can primary energy be used to supply final energy; one of the few forms of primary energy that can be used as final energy is natural gas
Proton	A particle with a single positive unit of electrical charge and a mass that is approximately 1.840 times that of the electron. It is the nucleus of the hydrogen atom and a constituent of all atomic nuclei.
PSA	Probabilistic Safety Analysis
PSI	Paul Scherrer Institute, Switzerland
PSM	Power System Module
PSOV	Propellant Shutoff Valve
PUC	Public Utility Commission
PUREX process	The plutonium and uranium extraction (PUREX) process is an aqueous process used in several foreign commercial and U.S. defense programs for separating out elements in spent nuclear fuel
Pyroprocessing	Nonaqueous processing carried out at high temperatures. An example of this is the relatively new technology being developed for reprocessing
PWR	Pressurized-Water Reactor. A light-water moderator and cooled reactor that employs an indirect cycle; the cooling water that passes through the reactor is kept under high pressure to keep it from boiling, but it heats water in a secondary loop that produces steam that drives the turbine

R

R-MAD	Reactor Maintenance, Assembly and Disassembly
rad	Radiation absorbed dose
Radioactive	Referring to the spontaneous transformation of one atomic nucleus into a different nucleus or into different energy states of the same nucleus
Radioactive decay	The spontaneous transformation of one atom into a different atom or into a different energy state of the same atom. The process results in a decrease, with time, of the original number of radioactive atoms in a sample

Radioactive waste	The unwanted radioactive materials formed by fission and other nuclear processes in a reactor or obtained in the processing or handling of radioactive materials. Most nuclear waste is initially in the form of spent fuel. If this material is reprocessed, new categories or waste result: high-level, transuranic, and low-level wastes (as well as others)
Radioisotope	A radioactive isotope. An unstable isotope of an element that decays spontaneously, emitting radiation. Radioisotopes contained in the spent fuel resulting from the production of nuclear power generally fall into two categories: fission products and transuranic elements (known as transuranics, actinides, or TRU). and activation products produced by neutron absorption in structural material in the spent fuel
RBE	Relative Biological Effectiveness
RBMK	See LWGR
RBR	Rotating Bed Reactor
RCC	Reinforced Concrete Containment
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDIPE	Research and Development Institute of Power Engineering, Moscow, Russia
Reactor core	The central portion of a nuclear reactor containing the fuel elements and the control rods (and part of the coolant and moderator), where most of the energy is produced
Recycling	The reuse of fissionable material, after it has been recovered by chemical processing from spent reactor fuel
rem	Roentgen equivalent man
Reprocessing, fuel	A generic term for the chemical and mechanical processes applied to fuel elements discharged from a nuclear reactor; the purpose is to remove fission products and recover fissile (uranium-233, uranium-235, plutonium-239), fertile (thorium-232, uranium-238) and other valuable material
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RRA	Rolls-Royce and Associates
RRC-KI	Russian Research Centre "Kurchatov Institute", Moscow, Russia
RTG	Radioisotope Thermoelectric Generator
RVACS	Reactor Vessel Auxiliary Cooling System

S

SADS	Subcritical Accelerator-Driven System
Safeguards	Term used to refer to the total set of international verifications, observations, etc, which together constitute a determination that nuclear materials (or, in some international agreements, facilities or other materials) have not been diverted from nuclear power programmes to the production of nuclear weapons
SAFR	Sodium Advanced Fast Reactor
SBWR	Simplified or Small BWR. See BWR
SDR	Slowpoke Demonstration Reactor, see SES
SECURE	Safe Environmentally Clean Urban Reactor
SES	Slowpoke Energy System
SCLWR	Steam Cooled Light Water Reactor
SG	Steam Generator
SiGi	Silicon germanium
Simplified design	A system designed with a minimum number of components to achieve the related safety function and relying as little as possible on support systems
SIR	S[illegible] Integral Reactor
SIS	[illegible] Injection System
SMF	[illegible]e Manufacturing Facility
SMR	[illegible]all and Medium-sized Reactors
SMS	[illegible]p Mobile System
SNAP	Space Nuclear Auxiliary Power program, US
SNPS	Space Nuclear Power Source
SNR	Space Nuclear Reactor
SPAR	Space Power Advanced Reactor
Spent fuel	Nuclear reactor fuel that has been irradiated (used) to the extent that it can no longer effectively sustain a chain reaction and therefore has been removed from the reactor for disposal. This radiation fuel contains fission products, uranium, and transuranic isotopes
SPR	Advanced Space Nuclear Power Program
SPS	Satellite Power System
STA	Science and Technology Agency, Japan

STAR-C	Space Thermionic Advanced Reactor Compact, US
STS	Space Transportation System (Space Shuttle)
Subcritical	Not capable of sustaining a nuclear chain reaction, but involving some degree of multiplication of neutrons

T

T-111	Tantalum alloy (Ta-8W-2Hf)
TAGS	Silver antimony germanium telluride
Target	Material subjected to particle bombardment (as in an accelerator) in order to induce a nuclear reaction
TBCV	Turbine Bypass Control Valve
TCT	Thermionic Critical Technology program, US, see SNP
TE	Thermoelectric
TFE	Thermionic Fuel Element
TI	Thermionic
TITR	Thermionic Test Reactor
TOP	Transient Over Power
TOSBWR	Toshiba Simplified Boiling Water Reactor
TPC	Transmutation Plant System
TPS	TRIGA Power Safety System, see DNHPP
Transmutation	The transformation (change) of one element into another by a nuclear reaction or series of reactions
Transuranic	An element above uranium in the Periodic Table of elements - that is, one that has an atomic number greater than 92. All transuranics are produced artificially (during a man-made nuclear reaction) and are radioactive. They are neptunium, plutonium, americium, curium, berkelium, californium, einsteinium, fermium, mendelevium, nobelium, and lawrencium
TRIGA	Training of personnel, nuclear Research, Isotope production, General Atomic
TRU	Transuranium elements
TRUEX	A chemical solvent process under development to extract transuranics from high-level waste
TSCV	Turbine Shutoff Control Valve

TVA	Turbine Valve Actuator
TWh	Terrawatt (10^{12} watt) hour
TWS	Transmutation Waste System
TZM	Molybdenum alloy (Mo-0.6, Ti-0.1, Zr-0.035)

U

U-ZrH	Uranium zirconium hydride
UO_2	Uranium dioxide
Uranium	A radioactive element of atomic number 92. Naturally occurring uranium is a mixture of 99.28 per cent uranium-238, 0.715 per cent uranium-235, and 0.0058 per cent uranium-234. Uranium-235 is a fissile material and is the primary fuel of nuclear reactors. When bombarded with slow or fast neutrons, it will undergo fission. Uranium-238 is a fertile material that is transmuted to plutonium-239 upon the absorption of a neutron.
USR	Ultimate Safety Reactor, see MSR

V

VVER	Russian version of PWR plant

W

W	Westinghouse
WANO	World Association of Nuclear Operations
Waste separation	The dividing of waste into constituents by type (for example, high-level, medium-level, low-level) and/or by isotope (for example, separating out plutonium and uranium). The waste may be separated by a chemical solvent process such as PURES or by any of a number of other chemical or physical processes
WEC	World Energy Council
WOCA	World Outside CPE Areas (MEM ou Monde à Economie de Marché)
WRI	World Resource Institute

X

XE Experimental Engine

Z

ZrH Zirconium hydride

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Part I Reactors for Power Production

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ADVANCED NUCLEAR REACTOR TYPES AND TECHNOLOGIES

Part II Heating and Other Reactor Applications

Abstract

The document is a comprehensive world-wide catalogue of concepts and designs of advanced fission reactor types and fuel cycle technologies. Two parts have been prepared:

Part I	Reactors for Power Production
Part II	Heating and Other Reactor Applications

Part III, which will cover advanced waste management technology, reprocessing and disposal for different nuclear fission options is planned for compilation during 1995.

This catalogue was prepared according to a special format which briefly presents the project title, technical approach, development status, application of the technology, reactor type, power output, and organization which developed these designs.

Parts I and II cover water cooled reactors, liquid metal fast reactors, gas-cooled reactors and molten salt reactors. Subcritical accelerator-driven systems are also considered. Various reactor applications as power production, heat generation, ship propulsion, space power sources and transmutation of such waste are included.

Each project is described within a few pages with the main features of an actual design using a table with main technical data and figure as well as references for additional information. Each chapter starts with an introduction which briefly describes main trends and approaches in this field. Explanations of terms and abbreviations are provided in a glossary.

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Approved by

Preface

For several years Studsvik AB has had the special task, disignated by the Swedish Ministry of Industry and Commerce of surveying, analysing and reporting on the development efforts and achievements abroad, concerning advanced nuclear fission technology. Program management and most of the task is performed within the subsidiary company Studsvik Eco & Safety AB. The results are presented in an annual summary report as well as in topical reports in Swedish. Contacts and collaboration with colleagues abroad and other institutes and research organisations are of importance in this work. In recent years, close contacts have been established between the Russian Research Centre "Kurchatov Institute", in Moscow and Studsvik. As a joint project and in close co-operation it was decided to prepare a document which briefly presented the plans, projects and achievements within the area of advanced nuclear fission technology and their present status in various countries.

The first and second parts of this work concern nuclear fission reactors. The last will examine front- and back-end of advanced nuclear fuel cycles. The intention is to extend and also update the material compiled in new editions when necessary. The first two parts were compiled by Victor Ignatiev (Editor), Olga Feinberg and Alexei Morozov who are leading scientists of the RRC "Kurchatov Institute" and I would like to thank them all for their efforts which made this report possible.

Acknowledgements are due to Acad. N Ponomarev-Stepnoi and Prof. A Gagarinski from RRC "Kurchatov Institute" who supported the work and J Kupitz, V Krett, C Goetzman from IAEA, Stanislav Subbotine from RCC "Kurchatov Institute", Tor Pedersen of ABB Atom and Bengt Pershagen for providing necessary information and useful suggestions.

I would also like to thank other contributors, among them Elisabet Appelgren for the typing and Monica Bowen-Schrire for correcting the English.

I hope that the report will be of help to those who would like to obtain an overview of recent advancements in this area

Lennart Devell
Program Manager
Advanced Nuclear Technology

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G Ship Reactors

Introduction

Since the beginning of the nuclear fission era, illustrated e.g. by the start-up of the Fermi pile in 1942, increasing efforts have been made to develop and use improved nuclear fission reactors. The civilian application of nuclear fission is mainly power plants for electricity generation but, to a small extent, also for heat generation. There are reactors for research, radio-nuclide production, medical treatment, water desalination, ship propulsion and space applications. Also, some different systems in critical or subcritical accelerator driven mode have been studied for waste transmutation, weapon plutonium incineration, tritium production etc.

Many overviews and proceedings from recent conferences are available which describe the evolution of reactor design, technologies and applications in various countries (some of them are listed below). The present document is a comprehensive collection of short descriptions of new and advanced nuclear reactor types and technologies on a global scale (see Table 1). The purpose of the document is to provide an overview of present plans, projects, designs and concepts of future reactors. The material has been collected from the open literature and from information from the vendors.

The description of each design or concept follows a standardized format (see Table 2) to facilitate comparison. The current status of the technology is described for each design as well as type of application, reactor type, power output etc.

Compilation of data began in January 1994 and the present document has been issued as the first edition which is intended to be updated. The authors would therefore appreciate any comments, corrections and additions.

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Table 1 Advanced Reactors

Part I	Reactors for Power Production					
Abbreviation	**Name**	**Type**	**Size/ MW(e)**	**Country**	**Vendor Organization**	**Development status**
A	**Advanced Water Cooled Reactors**					
AE	**Evolutionary Designs for Large Power Plants**					
AEP1	N4	PWR	1400	France	Framatome	Commercial
AEP2	Sizewell B	PWR	1250	US/UK	Westinghouse/ Nuclear Electric	Commercial
AEP3	KWU-Convoy	PWR	1287	Germany	Siemens-KWU	Commercial
AEP4	APWR	PWR	1350	Japan/US	Mitsubishi//KEPCO/ Westinghouse	Detailed design
AEP5	System 80 Plus	PWR	1345	US	ABB-CE	Detailed design
AEP6	VVER-1000 (V-392)	PWR	1000	RF	Hydropress	Detailed design
AEP7	EPR	PWR	1500	Germany/ France	Siemens-KWU/ Framatome	Basic design
AEB1	ABWR	BWR	1356	Japan/US	Hitachi/Toshiba/GE	Commercial
AEB2	BWR 90	BWR	1379	Sweden	ABB Atom	Commercial
AEH1	CANDU 9	HWR	1050	Canada	AECL	Commercial
AT	**Evolutionary Designs for Medium Power Plants**					
ATP1	VVER-500/600	PWR	600	RF	Hydropress	Detailed design
ATP2	AP-600	PWR	630	US	Westinghouse	Detailed design
ATP3	MS-600	PWR	630	Japan	Mitsubishi	Basic design
ATP4	AC-600	PWR	600	China	CNNC	Basic design
ATB1	SBWR	BWR	600	US	General Electric	Detailed design
ATB2	HSBWR	BWR	600	Japan	Hitachi	Basic design
ATB3	TOSBWR-900P	BWR	310	Japan	Toshiba	Basic design

Table 1 Advanced reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(e)	Country	Vendor Organization	Development status
ATB4	1000 Natural Circulation BWR	BWR	1000	US	General Electric	Studies
ATB5	SWR 1000	BWR	1000	Germany	Siemens-KWU	Basic design
ATG1	MKER-800	LWGR	800	RF	RDIPE	Studies
ATH1	CANDU 3	HWR	450	Canada	AECL	Detailed design
AI	**Innovative Designs**					
AIP1	PIUS-600	PWR	600	Sweden	ABB Atom	Basic design
AIP2	ISER	PWR	200	Japan	JAERI	Studies
AIP3	ISIS	PWR	200	Italy	Ansaldo Spa	Studies
AIP4	MAP	PWR	300	US	Combustion Engineering	Basic design
AIP5	VPBER-600	PWR	600	RF	OKBM	Basic design
AIP6	SIR	PWR	320	US/UK	ABB-CE /Rolls Royce	Basic design
AIP7	SPWR	PWR	600	Japan	JAERI	Studies
AIP8	B-500 SKDI	PWR	515	RF	RRC-KI/Hydropress	Studies
AIP9	SCLWR	PWR	1100	Japan	University of Tokyo	Studies
B	**Liquid Metal Fast Reactors**					
BT	**Evolutionary Designs**					
BTS1	EFR	LMR	1500	France/GB/ Germany/Italy	EdF/NE/ Bayernwerk/ENEL	Basic design
BTS2	BN-600M	LMR	600	RF	OKBM	Detailed design
BI	**Innovative Designs**					
BIS3	DEMO FBR	LMR	660	Japan	FEPS	Basic design
BIS4	SAFR	LMR	450	US	Rockwell Int/CE	Basic design
BIS5	ALMR (PRISM)	LMR	150	US	General Electric	Basic design

Table 1 Advanced reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(e)	Country	Vendor Organization	Development status
BIL1	BREST 300	LMR	300	RF	RDIPE	Studies
BIL2	LFBR	LMR	625	Japan	JAERI	Studies
C	**Gas Cooled Reactors/Innovative designs**					
CI1	HTR-Modul	HTGR	80	Germany	Siemens-KWU	Basic design
CI2	MHTGR	HTGR	190	US	General Atomics	Basic design
CI3	GT-MHR	HTGR	300	US	General Atomics	Studies
CI4	HTGR-MHD	HTGR	860	Japan	JAERI	Studies
D	**Molten Salt Reactors/Innovative designs**					
DI1	USR	MSR	625	US	ORNL	Studies
DI2	FUJI-Pu	MSR	100	Japan	Tokai University	Studies
DI3	MSR-NC	MSR	470	RF	RRC-KI	Studies

Table 1 Advanced Reactors (cont'd)

Part II Heating and Other Reactor Applications						
Abbreviation	**Name**	**Type**	**Size/ MW(t)**	**Country**	**Vendor Organization**	**Development status**
E	**District Heating Reactors**					
E1	AST-500 M	PWR	500	RF	OKBM	Commercial
E2	MARS	PWR	600	Italy	Rome University	Studies
E3	SECURE-H	PWR	400	Sweden	ASEA	Studies
E4	NHP-200	PWR	200	Germany	Siemens-KWU	Detailed design
E5	Thermos-100	PWR	100	France	CEA	Basic design
E6	NHR-200	PWR	200	China	INET	Detailed design
E7	RUTA	LWR	20	RF	RDIPE	Studies
E8	KNDHR	LWR	10	Korea	KAERI	Studies
E9	Slowpoke (SES-10)	LWR	10	Canada	AECL	Commercial
E10	Geyser	PWR	10-50	Switzerland	PSI	Studies
E11	GHR-20	HTGR	20	Germany/Switz.	KWU	Basic design
F	**Decentralized Nuclear Heating Power Reactors**					
F1	CAREM-25	PWR	100	Argentina	INVAP	Studies
F2	PAES-100	PWR	160-170	RF	OKBM	Commercial
F3	TRIGA Power System	PWR	64	US	General Atomic	Commercial
F4	ABV	PWR	60	RF	OKBM	Detailed design
F5	ELENA	PWR	3	RF	RRC-KI	Basic design
F6	Compact HTGR Gas Turbine	HTGR	29	US	General Atomic	Studies
G	**Ship Reactors**					
G1	OK-900A	PWR	170	RF	OKBM	Commercial
G2	KLT-40	PWR	170	RF	OKBM	Commercial

Table 1 Advanced Reactors (cont'd)

Abbre-viation	Name	Type	Size/ MW(t)	Country	Vendor Organization	Development status
G3	MRX	PWR	100	Japan	JAERI	Basic design
G4	DRX	PWR	0.75	Japan	JAERI	Basic design
I	**Space Nuclear Reactors and Isotope Batteries**					
I1	GPHS-RTG	Radioisotope Thermo-electric	$3 \cdot 10^{-4}$ MWe	US	Westinghouse	Commercial
I2	ROMASHKA	Fast, Thermo-electric	$3 \cdot 10^{-4}$-10^{-2} MWe	RF	RRC-KI	Detailed design
I3	TOPAZ-II	LMR, Thermionic	$6 \cdot 10^{-3}$ MWe	RF	RRC-KI	Commerical
I4	STAR-C	Thermionic	$5 \cdot 10^{-3}$-$4 \cdot 10^{-2}$ MWe	US	Rockwell Institute	Basic design
I5	SP-100	Fast, Thermionic	0.1 MWe	US	General Electric	Basic design
I6	ERATO	LMR/HTR	$2 \cdot 10^{-2}$ MWe	France	CNES/CEA	Studies
I7	LMCPBR	LMR/HTR	10	Japan	JAERI, Toshiba	Studies
I8	Rover/NERVA	HTGR	367 - 1 566	US	LANL, Westinghouse	Basic design
I9	NPPS	HTGR	1 200	RF	RRC-KI	Studies
J	**Subcritical Accelerator Driven Waste Transmutation and Energy Generation Systems**					
J1	Energy Amplifier	LWR/HTGR	200	Switzerland	CERN	Studies
J2	PHOENIX	LMR	3 600	US	BNL	Studies
J3	JAERI-TPC	MSR/LMR	820	Japan	JAERI	Studies
J4	ATW/ABC	MSR/LMR	3 000	US	LANL	Studies
J5	AMSB	MSR	1 000-2 100	Japan	Tokay University	Studies
J6	BBR	MSR	5 000	France	CEA	Studies

Table 2 Advanced Nuclear Design Description Format

Format	Description	
Technical approach	E	Evolutionary design for large power plants
	T	Evolutionary design for medium power plants
	I	Innovative design
Title	Title of project	
Applicaton of the reactor	NPP	Nuclear power plant
	NHP	Nuclear heating plant
	NHPP	Nuclear heat and power plant
	SMS	Ship mobile system
	SNS	Space nuclear source
	TWS	Transmutation waste system
Reactor type	PWR, BWR, HWR, LWGR, FBR, HTGR, MSR, SADS (subcritical accelerator-driven system) e.a.	
Power output	L	Large power (more than 1 000 MWe)
	M	Medium power (more than 100 MWe, less than 1 000 MWe)
	S	Small power (more than 1 MWe, less than 100 MWe)
	VS	Very small power (less than 1 MWe)
Organization	Type of organization, which developed design	
Development status	C	Commercially available design
	D	Detailed engineering design
	B	Basic engineering design
	S	Studies of concept
Description	General description of the design. This will generally include a simple description of the reactor plant. This may be followed by a summary with the main features of an actual design using tables with main technical data or figures	
Schedule	Schedule for development	
References	References for additional information.	

Part II Heating and Other Reactor Applications

E District Heating Reactors

E0 Introduction

The potential for the nuclear heating application depends on the specific temperature requirements of different technologies provided below:

Networks		Nuclear source	
Application	Required temperature, °C	Reactor type	Supply temperature, °C
District heating and sea desalination	80 - 200	LWR	300
Petroleum refining	250 - 550	LMR	540
Petrochemical industry	600 - 880	MSR	700
Hydrogen production	900 - 1 000	HTGR	950

The current nuclear application is mainly considered for district heating distribution, but to a lesser extent also for industrial one (e.g. MARS reactor plant (see E2) and HTR-Modul (see CI1)).

District heating markets are currently restricted to specific northern countries, such as Finland, Germany, Sweden, Canada and Russia where heat distribution networks are already in existence in certain regions. More than 10 different NHP concepts are known today throughout the world. Most of them are based on LWR technology and designed to satisfy different energy boundary conditions and have following different technological lines.

The specific features of NHP are the following:

- The power ranges of NHP (from 2 to 500 MWt) are considerably lower than those of current and advanced LWRs

- The supply temperature of the district-heating networks fed by these reactors does not exceed 150 °C; the coolant temperature in the primary circuit (e.g. about 200 °C for water-cooled NHP) is therefore considerably lower than for NPPs (the typical temperature for PWR is 320 °C and for BWR 300 °C)

- The power density of most NHP with PWR reactor plant lies in the range of not more than 55 kW/l compared with about 100 kW/l for large power and 60 - 70 kW/l for medium power APWR

- The low temperature required allows for lower operating pressures, which in turn, along with the smaller size, lead to less massive RPVs and allows the components of the primary circuit to be integrated within the RPV

- All the NHPs considered make as much use as possible of components and systems proven in the operation of large NPPs

- The operation schemes proposed for all NHPs are considerably simplified as compared to large NPPs. Some of them can operate without operation staff

- The time between fuelling in NHPs is in general longer than in NPPs. Almost all concepts are based on existing fuel cycle technologies

- Most of the NHP concepts considered make extensive use of passive systems and components and rely more heavily on natural processes rather than engineered safeguards

However, only some reactors for district heating: AST-500 in Russia, SLOWPOKE in Canada and NHP in China have actually been realized so far but no operation experience of advanced designs has been gained as yet.

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E1

Format	Description
Title	AST-500M
Application of the reactor	NHP
Reactor type	PWR
Power output	Medium power/500 MWt
Organization (name)	OKBM, RF
Development status	Commercial
Description	The AST-500M district heating plant is a PWR with an integral arrangement of the primary circuit (core and heat-exchangers are located in one vessel) and natural circulation of coolant in the first circuits. The second circuit has a pressurizer. About 20 % of the nominal power can be removed under natural circulation conditions in the second circuit. The capacity of the nominal residual sink is equal to 2.5% of the nominal reactor power. The reactor has three loops for nominal heat removal and three independent emergency heat removal systems (EHRS). Each EHRS has a water storage tank connected to the loops of the second circuit, and can remove 2 % of nominal power. The quantity of water is enough for three days of cooling. For the emergency make up of the reactor during the AST-500M reactor plant black-out in the depressurized state (in particular during refuelling operations the EHRS is not capable of ensuring cool-down because of elimination of circulation over the second circuit) three independent pipes for the reactor make up with the water flowing by gravity from EHRS tanks are used. The containment (Voronej AST-500 plant) and shield building (Nijegorodskaj AST-500 plant) is used for protection from external events. The guard vessel is used for the reduction of loss of coolant accident probability. The calculated core damage probability is less than 10^{-7} per reactor-year. There is also a smaller unit, AST-200 district heating plant, of 200 MWt output.
Schedule	Two AST-500M units intended for district heating are under construction near Voronej, Russia.

AST-500M Unit Data

Reactor power		500 MWt
Parameters of the first circuit		
-	Pressure	20 bar
-	Inlet coolant temperature	131 °C
-	Outlet coolant temperature	208 °C
Parameters of the intermediate circuit		
-	Pressure	12 bar
-	Temperature in the heat-exchanger, inlet/outlet	87/160 °C
Parameters of district-heating water (inside plant boundary)		
-	Pressure	20 bar
-	Temperature, inlet/outlet	64/144 °C
Core parameters:		
-	Diameter	2.8 m
-	Height	3.0 m
-	Power density	30 MW/m^3
-	Fuel loading	50.0 t
-	Fuel residence time	6 years
-	Refuellings	3
-	Fuel burnup	14 MW d/kg
Reactor vessel		
-	Diameter/height	5.3/25.3 m
-	Water volume	190 m^3
-	Neutron fluence	10^{16} n/cm^2
Guard vessel		
-	Diameter/height	8.15/27.95 m

References

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Advanced reactor plant of passive safety for district heating plants AST-500M.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, Vol 1, p 4.1.

1 Reactor
2 Guard vessel
3 Boron solution storage tank
4 ERHR condenser
5 Water storage tank
6 Bubbler
7 Containment
8 Secondary circuit steam pressurizer
9 EHRH heat exchanger with water storage tank
10 Heating grid HX
11 Secondary coolant clean-up system
12 Primary coolant clean-up system
13 Secondary circuit make-up
14 Make-up by water, boron solution feed
15 Make-up by water, gas

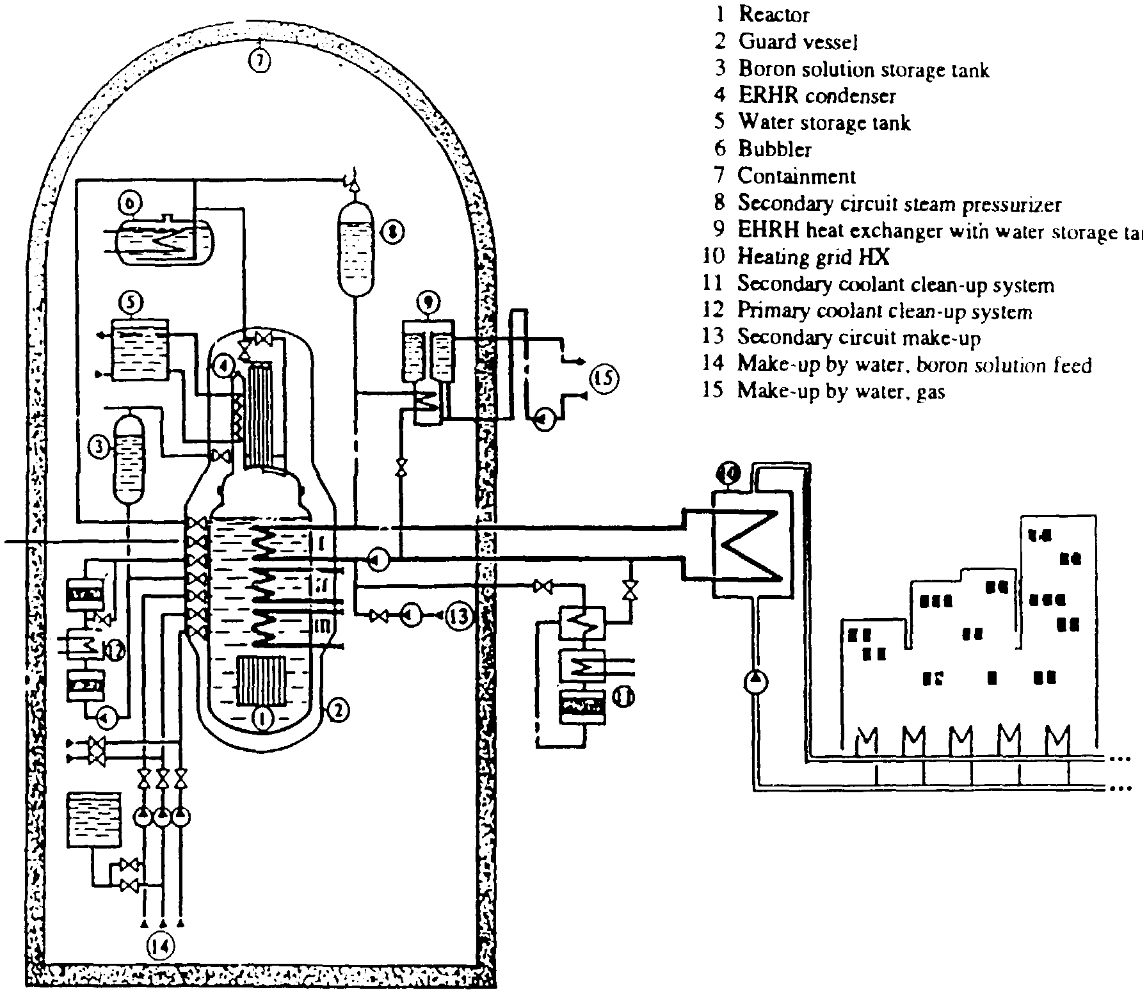

Figure E1
AST-500M: Flow diagram.

E2

Format	Description
Title	MARS (Multipurpose Advanced Reactor Inherently Safe)
Application of the reactor	NHP, NHPP
Reactor type	PWR
Power output	Medium power/600 MWt
Organization (name)	Rome University, Italy
Development status	Studies

Description

The MARS design concept combines the advantages of several reactor types: PWR (separation of the reactor coolant from the consumer working medium), BWR (lower coolant pressure and pools with a large cold water inventory within the containment), AST district heating plant (systems of emergency residual heat removal through the intermediate circuit with natural circulation), and HTGR (prestressed reinforced concrete guard vessel).

The MARS reactor safety is improved primarily due to reduced stresses in the elements of the primary circuit structures as all systems of the primary circuit are accommodated within a cavity of prestressed reinforced concrete, filled with cold water at a pressure equal to that of the primary circuit. This practically eliminates the accidents associated with depressurization and ruptures of the pipes in the primary circuit and reactor vessel.

The reactor is supposed to be used for industrial heating. For increasing the reliability and safety of the main technological process it was considered economically favourable to use the following technical solutions:

- Low average core power density (63 MW/m^3) and medium unit power (600 MWt) permit removal of the decay heat by employing natural circulation of the coolant in all circuits
- Low pressure reactor coolant (7 MPa), 40 ^{o}C coolant temperature margin to the saturation temperature at the reactor outlet (246 ^{o}C) and moderate maximum heat fluxes ensure a considerable critical heat flux ratio in transient and accident situations

o Stainless steel fuel cladding is resistant to the chemical reactions with the coolant, reduces the probability of fission product escaping into the coolant and, thus, reduces the occupational exposure and simplifies reactor maintenance.

The reactivity control is accomplished using the standard PWR clusters of absorbing rods. For the reactor scram the mechanism of passive insertion of the control rods at the core temperature exceeding the specified limit has been developed. In accordance with industrial requirements the time of continuous operation between the refuellings has been established as 17 months, which requires the use of liquid boron control of reactivity (the maximum boron content in the beginning of the fuel cycle is 1200 p.p.m.).

The MARS concept uses flange joints, instead of weld ones, connecting the pipelines with the reactor vessel and steam generators.

The inner containment sections are designed as a system of six inter-connected cylinders of prestressed reinforced concrete. In the upper part of the outer containment, the pools of two independent emergency cool-down systems are located. These pools are also used for spent fuel storage.

Innovative, passive technologies for check valve design and the control rod scram system have been used.

Schedule

No MARS plant has been ordered yet. The conceptual study has been completed. No current development work is being carried out.

MARS Unit Data

Net plant capability	600 MWt
Coolant temperature	
- Inlet	216 °C
- Outlet	246 °C
Coolant pressure	70 bar
Outer diameter of the fuel rod	9.78 mm
Fuel lattice pitch	13.0 mm
Core height	260 cm
Fuel assembly array	15 x 15
Number of fuel assemblies	96
Core equivalent diameter	216 cm
Average power density of the core	63 MW/m^3
Average burnup	30 MWd/kg
Fuel enrichment (first loading)	3.6 %
Fuel residence time	51 months

References

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IAEA-TECDOC-712, IAEA, Vienna, 1993.

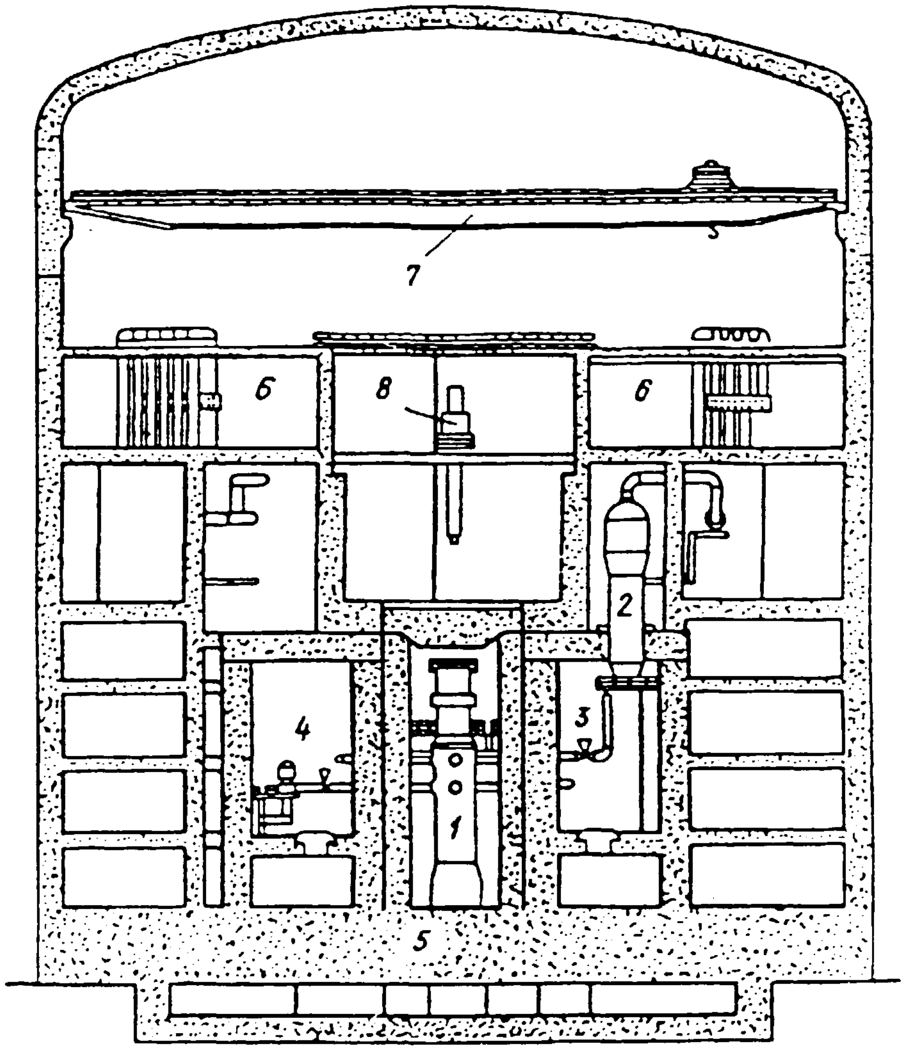

Figure E2
MARS: Reactor building.

1	Reactor	5	Inner containment
2	Steam generator	6	Pool
3	Check valve	7	Bride crane
4	MCP	8	Refuelling machine

E3

Format	Description
Title	SECURE-H
Application of the reactor	NHP
Reactor type	PWR
Power output	Medium power/400 MWt
Organization (name)	ASEA, Sweden
Development status	Studies
Description	SECURE-H reactor plants with a power range of 200 - 400 MWt are based on well-known PIUS principle (see AIP1). The reactor core is located at the bottom of a prestressed concrete vessel, which contains some 1 500 m^3 of cold heavily-borated water. The pressure in this vessel is 7 bar. The reactor tank separates the heavily-borated water of the pool from the less heavily borated water of the primary circuit. The SECURE-H plant has a two-loop arrangement of the primary circuit. The primary pumps and heat-exchangers are located outside the prestressed concrete vessel. The cooling water is heated up by the core from 90 °C to 120 °C. Reactivity control is achieved by the adjustment of the boron content of the primary circuit. The core power density is low, which means good margins against fuel failure. The particular safety feature of the SECURE-H reactor plant is its primary circuit, which is open towards the borated water of the pool. During normal operation, a pressure equilibrium is established between primary water and pool water. The pressure drop over the core is only 0.11 bar. In case of disturbance of the equilibrium, after a pump failure or due to a temperature rise in the primary circuit, the borated water of the pool enters the primary circuit and shuts down the reactor. The primary cooling system and the blowdown chamber are housed in a containment building below ground. The primary cooling system delivers heat from the core through the primary heat-exchanger to an intermediate cooling system connected to the district-heating grid through a secondary heat-exchanger. The intermediate cooling system operates at a higher pressure than the primary system, and its water has a high content of boron.
Schedule	No SECURE-H reactor plant has been ordered yet. No current development workis being carried out.

SECURE-H Unit Data

Thermal output	400 MWt
Fuel power density	15.0 kW/kg U
Number of fuel assemblies	308
Number of fuel rod positions per assembly	8x8
Active core height	1.845 m
Equivalent core diameter	2.51 m
Fuel enrichment	2 %
Core coolant flow	2 300 kg/s
Primary system operating pressure	20 bar
Coolant inlet temperature	150 °C
Coolant outlet temperature	190 °C
Power density	45 kW/l
Inner diameter of concrete RPV	9.5 m
Fuel residence time	2.5 years

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Nuclear Technology, Vol 79, Nov, 1987.

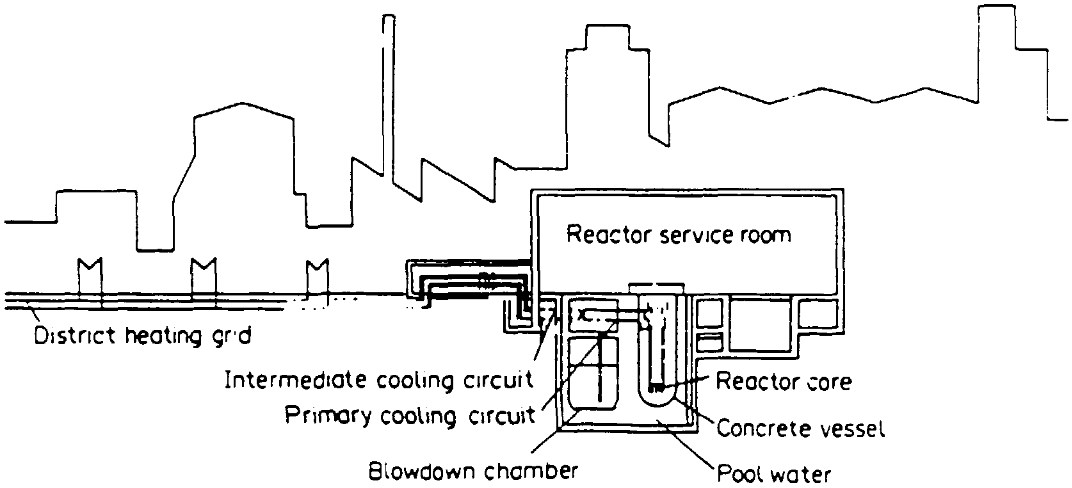

Figure E3
SECURE-H: Main cooling system.

E4

Format	**Description**
Title	NHP-200
Application of the reactor	NHP
Reactor type	PWR
Power output	Small power/200 MWt
Organization (name)	Siemens-KWU, Germany
Development status	Detailed design
Description	The NHP-200 reactor concept relies on the long experience of PWR and BWR design for nuclear power plants. The plant size has been fixed at 200 MWt. The primary circuit cooling water enters the core at a pressure of 15 bar and a temperature of 160 °C. The core configuration is similar to a BWR plant design. The temperature of the water/steam mixture at the core outlet is 200 °C. The rather low pressure of the primary coolant and low pressure capacity leads quite naturally to an integration of the primary heat-exchangers, the hydraulic drives of the control rods and the storage racks for spent fuel within the pressure vessel. The core heat is transferred to twelve heat-exchangers by natural circulation. The reactor pressure vessel is surrounded by a tight-fitting containment. A particular safety feature is that the core cannot be uncovered in case of a leakage in the primary pressure barrier, due to the small volume available between the RPV and the containment. Reactor shutdown is achieved by switching off the pumps of the hydraulic control rod drives. Residual heat removal is achieved by opening the valves to cooling circuits with natural circulation cooling towers. The intermediate circuit has a higher pressure compared to the primary system. The entire heating plant is housed in a single building complex with three parts: a reactor building, which is embedded in the ground, a reactor building operating hall and an auxiliary building.
Schedule	No NHP-200 plant has been ordered yet. The main design principles are already demonstrated in a prototype heating reactor in China.

NHP-200 Unit Data

-	Thermal power	200 MWt
-	Core coolant flow	1 030 kg/s
-	Number of fuel assemblies	180
-	Fuel assembly type	8x8, BWR type
-	Active core height	2.350 m
-	Primary pressure	15 bar
-	Coolant temperature	158/198 °C
-	Average volumetric steam content at core outlet	26 %
-	Maximum heat flux	670 kW/m^2
-	Average power density	20 kW/l
-	Average specific power	10 kW/kg
-	Number of control rods	45
-	Network temperature (forward/return)	120/80 °C

References

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Nuclear Energy Agency. OECD, Paris, France, 1991, Vol 2, p 193-198.

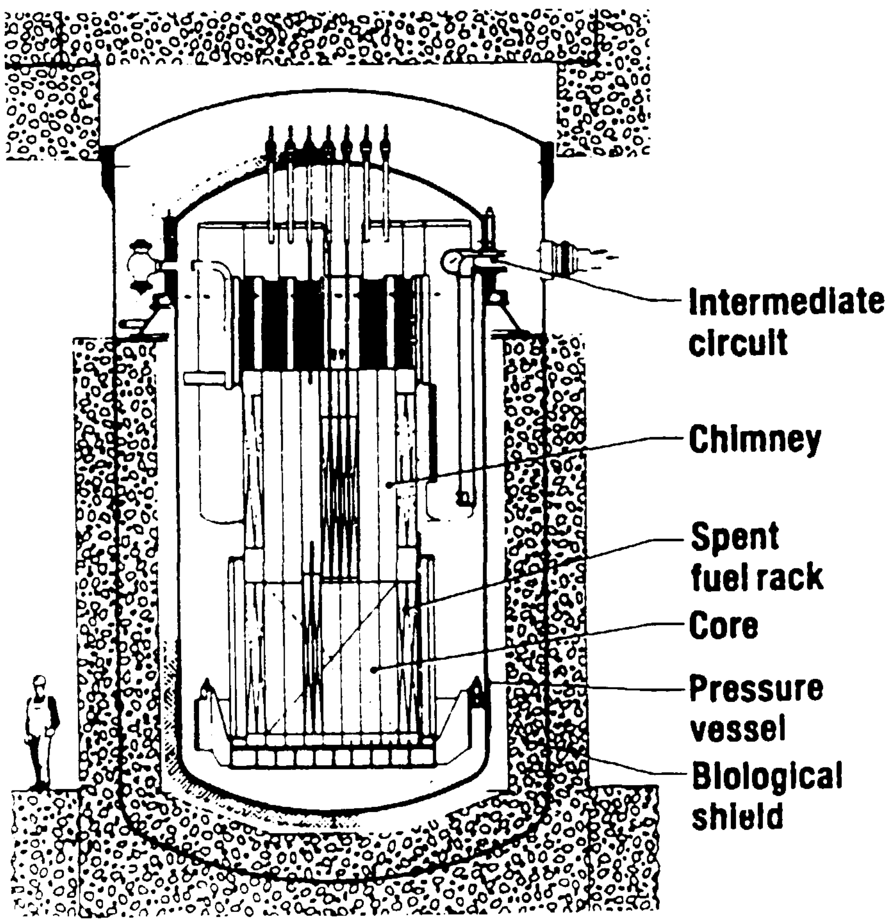

Figure E4
NHP-200: Primary system arrangement.

E5

Format	**Description**
Title	THERMOS-100
Application of the reactor	NHP
Reactor type	PWR
Power output	Small power/100 MWt
Organization (name)	CEA, France
Development status	Basic design
Description	The French Commissariat à l'Energie Atomique (CEA) has been developing the Thermos reactor plant since the beginning of the 1970s. It is a PWR, designed for a thermal output of 100 MW. The reactor core is located in a stainless-steel vessel filled with water. This vessel is, in turn, located in a reinforced concrete pool full of borated water. The reactor vessel contains three coolant pumps. The fuel elements are of the slab-type, with Zircaloy cladding. This fuel type has been developed for the power reactors of submarines.
Schedule	No THERMOS reactor plant has been ordered yet. No current development work is being carried out.

THERMOS-100 Unit Data

Power range	100 MWt
Power density	45 kW/l
Primary circuit type	integrated
Primary coolant	H_2O
Primary coolant pressure	11 bar
Primary coolant temperature, inlet/outlet	131/144 °C
Network forward temperature	130 °C
Heat removal by	pump
Residual heat removal by	natural circulation
Ultimate heat sink	pool
Fuel type	UO_2 slabs
Fuel residence time	2 years
Fuel enrichment	3.7
Reactivity control by	rods
Shutdown by	rods

References

Small and medium reactors. Technical supplement, Nuclear Energy Agency. OECD, Paris, France, 1991, Vol 1,2.

E6

Format	Description
Title	NHR-200
Application of the reactor	NHP
Reactor type	PWR
Power output	Small power/200 MWt
Organization (name)	INET (Institute Nuclear Energy Technology), China
Development status	Detailed design
Description	In 1989 the 5MW Test Heating Reactor was put into operation in China. In 1989 it was decided to construct a commercial NHR-200 with an output of 200 MWt. The main design principles of the NHR-200 are similar to the NHP-200 Siemens design. The main technical features of the NHR-200 are as follows: o Integral arrangement o Natural circulation o Double pressure vessel o Low temperature, low pressure, low power density o Hydraulic facility of control rods o Storage of spent fuel in the pressure vessel around the reactor core o No emergency cooling system o Residual heat removal system also uses natural circulation of coolant The core is located at the bottom of the RPV. The primary heat-exchangers are arranged on the periphery in the upper part of the RPV. The system pressure is maintained by inert gas and steam. A containment fits tightly around the RPV so that the core will not become uncovered under any postulated coolant leaks.
Schedule	It was decided to build a demonstration NHR-200 in the North of China in 1995, which was classified as one of the key national nuclear projects. The detailed design study for the NHR-200 plant is finished. The plant is planned to be completed in 1998.

NHR-200 Unit Data

Total thermal power	200 MWt
Reactor pressure	22 bar
Core outlet temperature	200 °C
Core inlet temperature	138 °C
Height of the core	2.1 m
Number of fuel bundles in core	120
Number of rods per bundle	141
Load of UO_2	~21 t
Enrichment of the fuel	3.0, 2.4, 1.8 %
Diameter of the fuel rod	1.0 cm
Average power density	~38 kW/l
Pressure of the secondary loop water	26 bar
Temperature of the secondary loop water	130/85 °C
Temperature of the heat grid water	110/70 °C

References

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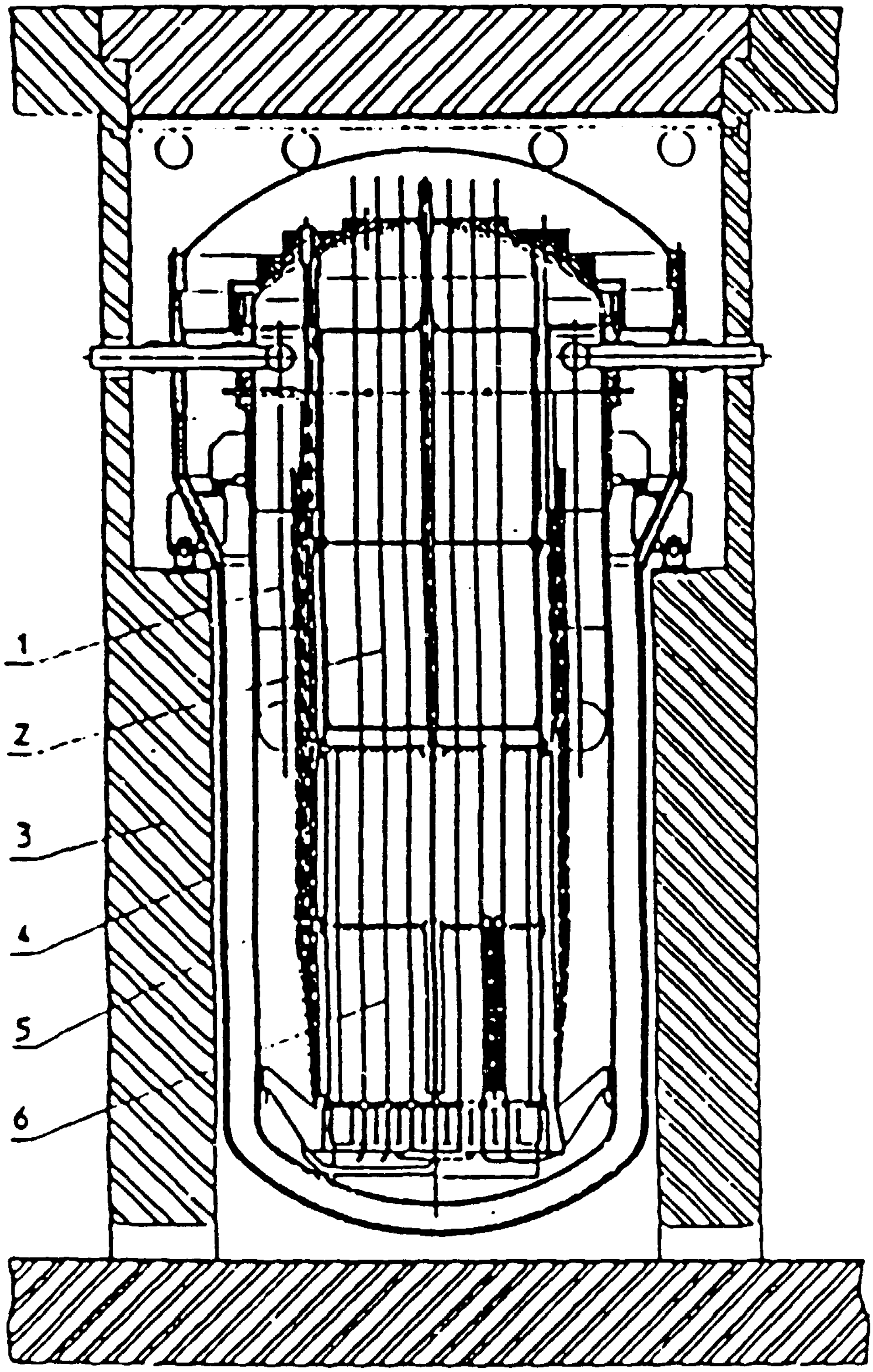

Figure E6
NHR-200: Reactor building.

1	Primary heat exchanger	4	C--ntainment
2	Riser	5	Pressure vessel
3	Biological shield	6	Core

E7

Format	**Description**
Title	RUTA
Application of the reactor	NHP
Reactor type	LWR
Power output	Small power/20 MWt
Organization (name)	RDIPE (Research and Development Institute of Power Engineering), RF
Development status	Studies
Description	The RUTA heating plant is a pool-type reactor with a 20 MWt output. All the primary equipment is integrated within the cylindrical reactor tank. The pool arrangement constitutes a circuit of natural circulation of the primary coolant, whereby heat is transferred from the core to the primary heat-exchangers and further on to the heating network via an intermediate circuit and the network heat-exchangers. The coolant circulation in the intermediate circuit is a natural one under all reactor conditions. The steel secondary circuit surfaces are used for decay heat removal from the reactor to the environment without reactor water boiling during any accidents in network circuit. One loop provides heat removal from the reactor to the environment without water boiling in a reactor when two others have failed. In the event of severe external impact, when all intermediate circuits fail, the provision is made for the a transfer from the reactor tank to the concrete well, then to the ground and for reactor water evaporation into the leak-tight reactor room (if it is necessary). This design forms a completely passive decay heat removal system. Underground siting of the reactor plant including secondary heat-exchangers has been proposed.
Schedule	No RUTA reactor plant has been ordered yet. The conceptual study has been completed.

RUTA Unit Data

Power	20 MWt
Coolant pressure	
- Primary circuit	atmosphere
- Intermediate circuit	4 bar
- Network	6 - 1 bar
Inlet/outlet coolant temperature	
- Primary circuit	70/100 °C
- Intermediate circuit	65/95 °C
- Network	60/90 °C
Outside tank diameter	4.8 m
Tank height	15 m
Number of loops in secondary circuit	3
Core dimensions	
- Active height	1.0 m
- Active diameter	1.23 m
Number of fuel assemblies	61
Fuel cycle	up to 1900 eff. day(8 yr)
Average burnup	23 MWd/kgU
Enrichment	4.0 %
Reactor lifetime (with account of special process for tank manufacturing)	80 - 100 years

References

ADAMOV, E O et al
Low-power nuclear plants for district heating with pool-type reactor (RUTA).
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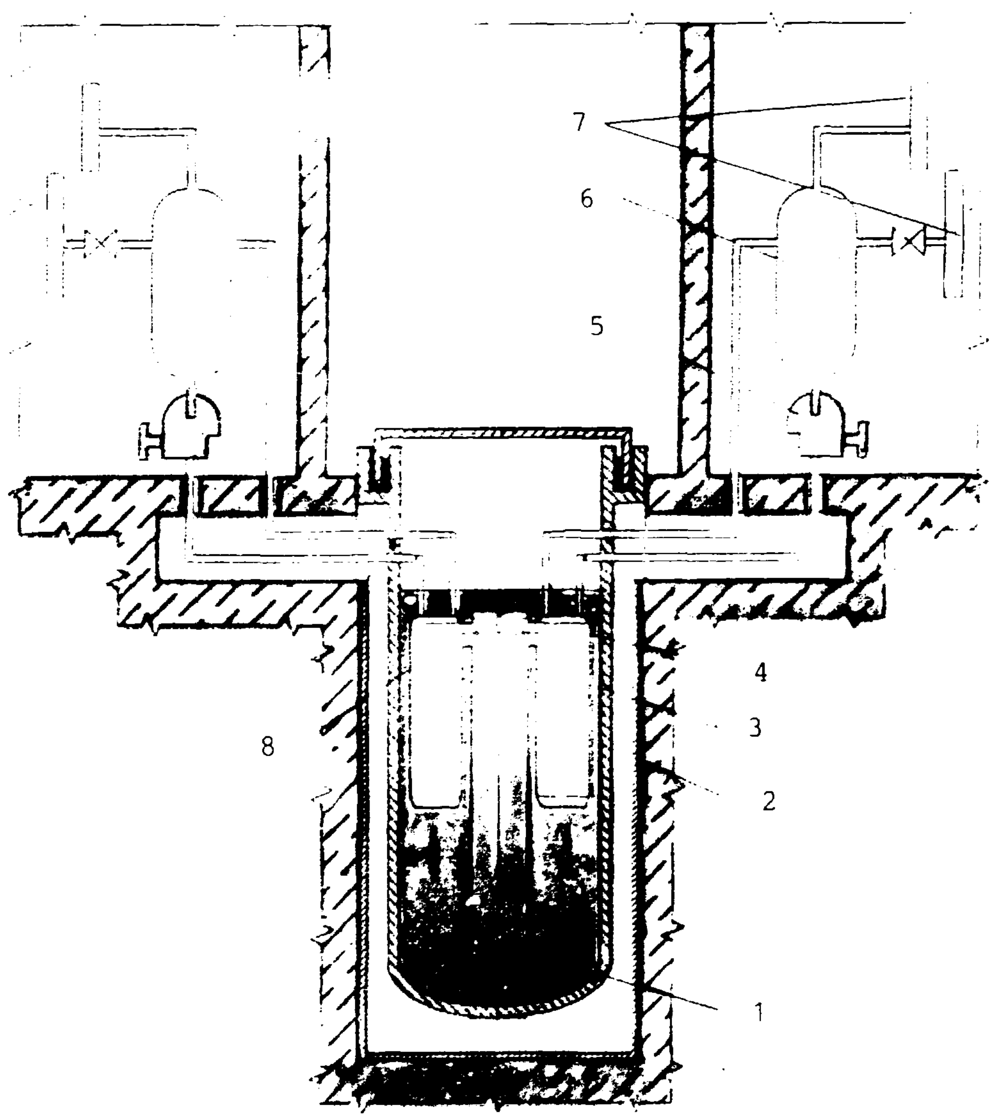

Figure E7
RUTA: Schematic.

1 Core, 2 Safeguard vessel, 3 Reactor tank, 4 Control rod, 5 RCP,
6 Secondary heat exchanger, 7 Pipelines of secondary circuit,
8 Primary heat exchanger

E8

Format	Description
Title	KNDHR
Application of the reactor	NHP
Reactor type	LWR
Power output	Small power/10 MWt
Organization (name)	KAERI (Korean Atomic Energy Research Institute), Korea
Development status	Studies
Description	KNDHR is a pool-type reactor with a maximum thermal power of 10 MW similar to SES-10 (Canada) and RUTA (Russia). It consists of three closed loops with natural circulation of the coolant. The heated water from the core transfers to two plate-type heat-exchangers, situated in the upper part of the pool and the outcoming water from the heat exchangers returns to the core. The KNDHR reactor plant operates at atmospheric pressure. The secondary circuit delivers the heat from the primary circuit to the building water supply system (user grid circuit) via the secondary heat-exchanger.
Schedule	In KAERI, a feasibility study for KNDHR development in Korea has been performed in 1991. No KNDHR reactor plant has been ordered yet.

KNDHR Unit Data

Maximum thermal power	10 MWt
Reactor size	∅5.3x13 m
Core size	∅1.1x1.2 m
Secondary pump head	10 m
Secondary circuit size	∅0.5x37 m
Secondary circuit pressure	3 bar
User-grid circuit pressure	1 bar
Core inlet/outlet temperature	76.1/94.0 °C
Secondary circuit inlet/outlet temperature	69.7/89.7 °C
User-grid circuit inlet/outlet temperature	65.3/85.3 °C
Core flow rate	122.3 kg/s
Secondary circuit flow rate	119.2 kg/s
User-grid circuit flow-rate	120.6 kg/s
Reactor free surface pressure	1.1 bar

References

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Thermal hydraulic analysis on the Korean nuclear district heating reactor. Proceedings International Conference on Design and Safety of Advance Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Japan, Vol 1, p 4.3.

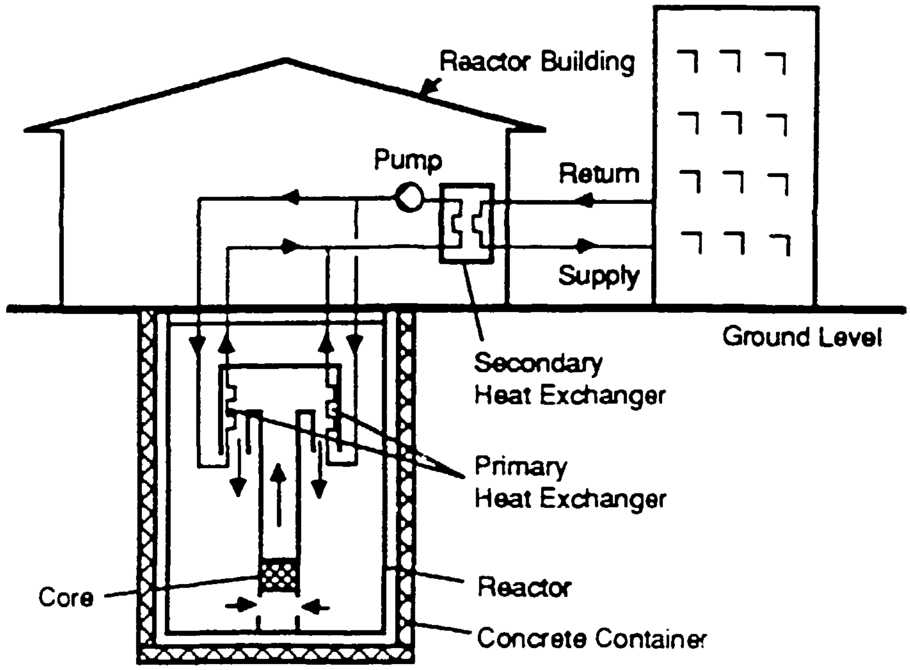

Figure E8
KNDHR: Schematic.

E9

Format	Description
Title	SES-10 (Slowpoke Energy System)
Application of the reactor	NHP
Reactor type	LWR
Power output	Small power/10 MWt
Organization (name)	AECL (Atomic Energy of Canada Limited), Canada
Development status	Commercial
Description	Based on the experience with its proven small 20 kWt Slowpoke swimming-pool reactors in operation in half of the Canadian research centres, the AECL has developed a swimming-pool reactor (SES-10) for heating purposes. One design goal of the SES-10 reactor plant is to fully automate all essential systems. Thus the unit can be operated for extended periods without the operator in the reactor building. The SES-10 plant has been designed for a thermal output of 5 - 20 MW. Its core is situated in a metallic cylinder near the bottom of a water pool and consists of CANDU-proven fuel elements with 3 per cent enriched uranium in a Zircaloy cladding. Natural circulation transfers the heat from the core to the heat-exchangers located at the top. Atmospheric pressure prevails at the water surface. Core reactivity is regulated by means of a single central control rod. The long-term reactivity compensation is achieved by means of a movable beryllium reflector. The reactor is designed to be operated unmanned. A particular safety feature is the lack of need for any pressure vessel, with the disadvantage that the supply temperature in the heating network is limited to 85 °C. The SES-10 reactor plant can heat 150 000 m^2 of floor area or approximately 1 500 individual apartments with an inflation-resistant fuel source. The reactor core and safety systems are being designed to remain fully functional during and after seismic events. Use of factory-fabricated modules contributes to the short construction time of one year proposed for the commercial use.
Schedule	A 2 MWt SDR demonstration plant of this type has been in successful operation in Whiteshell/Manitoba (Canada) since July 1987. No SES-10 reactor plant has been ordered yet.

SES-10 Unit Data

Power range	2 - 10 MWt
Power density	5.5 kW/l
Primary circuit type	tank
Primary coolant	H_2O
Primary coolant pressure	1.7 bar
Primary coolant temperature inlet/outlet	60/90 °C
Network temperature forward/return	85/70 °C
Heat removal by	natural circulation
Residual heat removal by	natural circulation
Ultimate heat sink	pool
Fuel type	UO_2 rods
Fuel residence time	3 years
Fuel enrichment	5 %
Reactivity control by	Be-reflector
Shutdown by	boric acid

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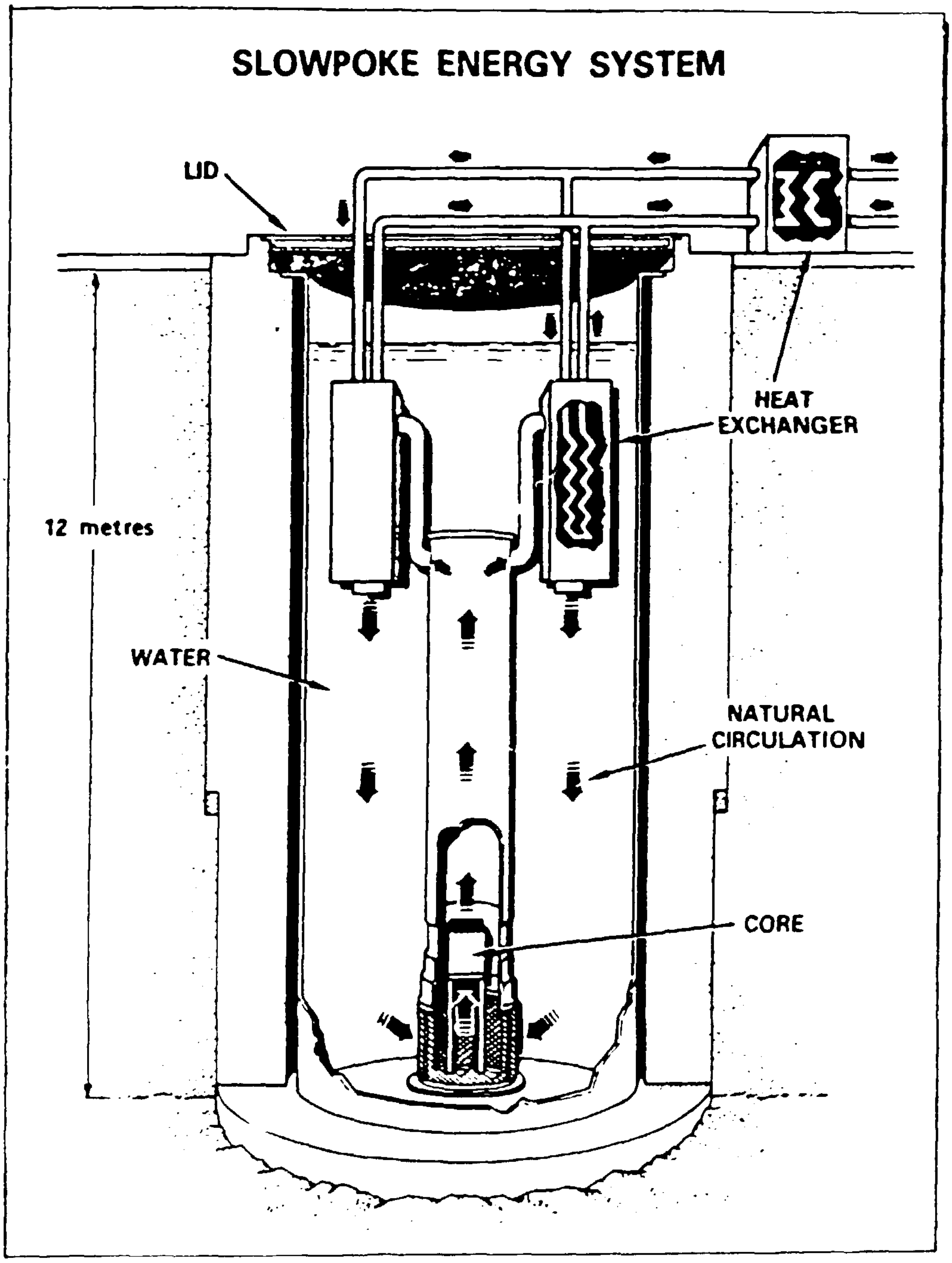

Figure E9
SES-10: Schematic diagram.

E10

Format	Description
Title	Geyser
Application of the reactor	NHP
Reactor type	PWR
Power output	Small power/10 - 50 MWt
Organization (name)	PSI (Paul Scherrer Institute), Switzerland
Development status	Studies

Description

The design requirements set for the Geyser reactor concept were:

- o Low power density and size
- o Completely autonomous operation
- o Emphasis on simplicity
- o Underground location

The Geyser reactor concept developed at the Paul Scherrer Institute (PSI) makes use of the static pressure of a high water column and does not need a proper pressure vessel. The reactor is located at the bottom of a concrete well some 50 m in depth and 5 m in diameter. With this arrangement the coolant reaches a saturated state at 150 °C at the core outlet. During its rise in a so-called diffuser, the coolant goes partially into the steam phase and transfers its heat to the primary heat-exchangers, which are also located in the well. These operate on their primary side as condensers and coolers and on the secondary side as evaporators. The secondary steam thus produced, transfers its heat through condensation to the heating network. Primary and secondary circuits operate by natural circulation.

A particular safety feature is the large amount of borated water in the well, which, depending on the equilibrium between the heat produced in the core and the heat demand by the network, can enter the primary circuit thus leading to a load dependent self-control and when necessary to a shutdown (see PIUS principle).

As another potential application sea water desalination is identified.

Schedule

No Geyser plant has been ordered yet. The conceptual design study has been completed.

Geyser Unit Data

Power range	10 - 50 MWt
Power density	5 kW/l
Primary circuit type	integrated
Primary coolant	H_2O
Primary coolant pressure	4.7 bar
Primary coolant temperature inlet/outlet	135/149 °C
Network temperature forward/return	118/60 °C
Heat removal by	natural circulation
Residual heat removal by	natural circulation
Ultimate heat sink	pool
Fuel type	UO_2 or UZrH rods
Fuel residence time	15 years
Fuel enrichment	8 - 20
Reactivity control by	boric acid
Shutdown by	boric acid

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Small and medium reactors. Status and prospects,
Nuclear Energy Agency. OECD, Paris, France, 1991, Vol 1.

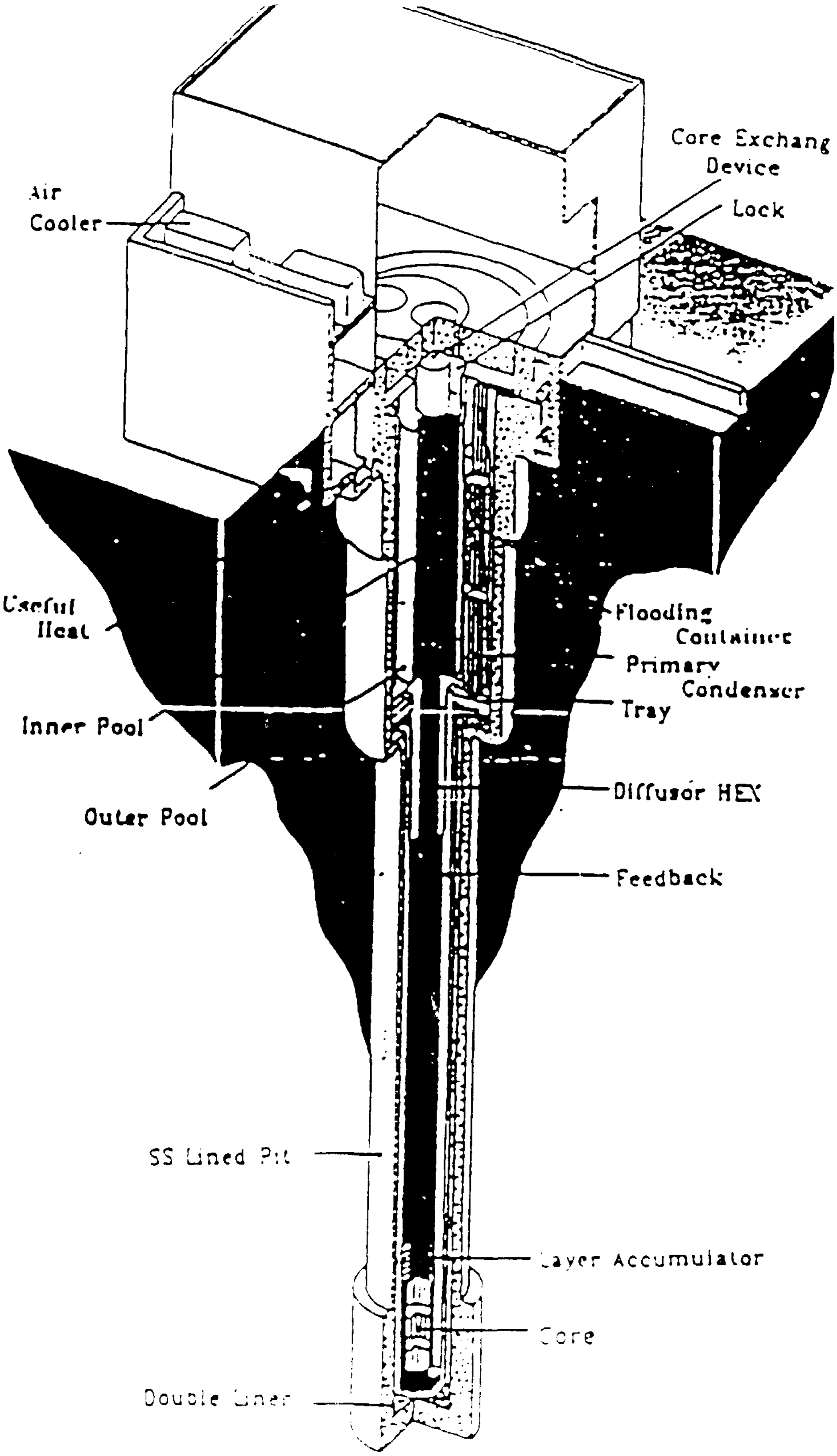

Figure E10
Geyser: Layout.

E11

Format	Description
Title	GHR-20
Application of the reactor	NHP
Reactor type	HTGR
Power output	Small power/20 MWt
Organization (name)	KWU, Germany/Switzerland
Development status	Basic design
Description	The GHR-20 district-heating plant is designed with a nuclear heat generating system of 20 MWt for generating 80 kg/s pressurized hot water at 15 bar and 120 °C with return temperature about 60 °C. The GHR-20 reactor and the secondary system are designed for a service life of 40 years with a capacity factor of 0.5. The core structure and fuel elements are based on the spherical fuel element containing low enriched fuel and on periodical refuelling. Reactivity control and shutdown are performed by absorber rods which are inserted into axial boreholes of the radial reflector. The primary circuit is designed with helium coolant inlet and outlet temperatures of 250 °C and 450 °C, with pressures of about 40 bar. The fuel elements have the standardized 6 cm diameter and contains TRISO-coated particles. Core meltdown is excluded due to the small power density of the fuel element and the small size of the reactor. The GHR 20 plant is designed so that during operation the reactor can be operated unmanned safely and with a high availability in a fully automatic way except for periodic inspections.
Schedule	Ready for detailed design development. No current development work.

GHR-20 Unit Data

Power range	10 - 50 MWt
Power density	2 kW/l
Primary circuit type	integrated
Primary coolant	helium
Primary coolant pressure	40 bar
Primary coolant temperature inlet/outlet	250/450 °C
Network temperature forward/return	120/160 °C
Heat removal by	pump
Residual heat removal by	natural circulation
Ultimate heat sink	graphite
Fuel type	LEU-graphite pebbles
Fuel residence time	16 years
Fuel enrichment	8
Reactivity control by	rods
Shutdown by	rods

References

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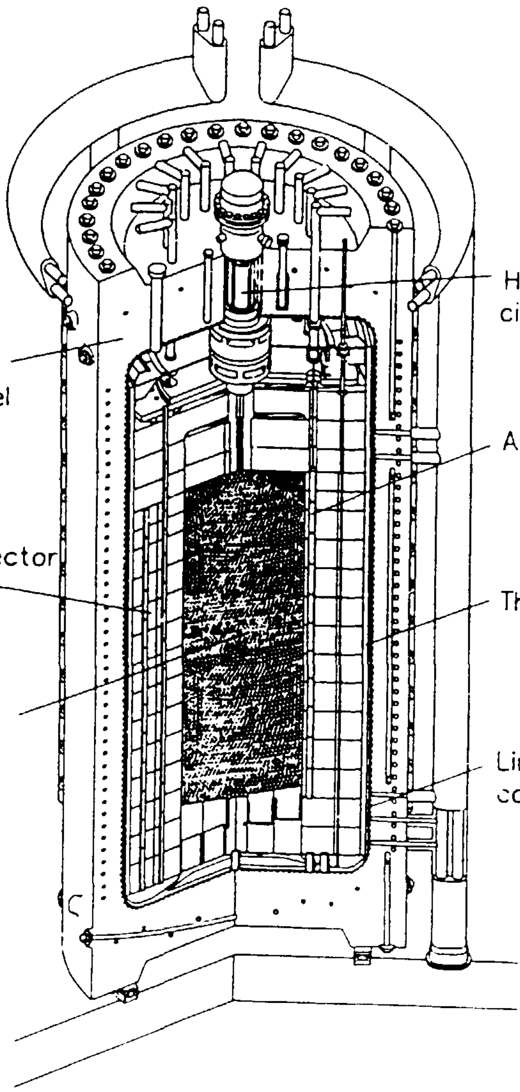

Figure E11
GHR-20: Reactor plant.

F Decentralized Nuclear Heating Power Reactors

F0 Introduction

The economic and geographical factors give a rise to a specific structure of power installations for electricity, drinking water production and heat supply for users located in hard-to-access isolated regions (e.g. far North and North East in Russia and others). The creation of multi-purpose, small power nuclear plants for remote regions is one of the possible solutions.

Every one of these applications has been technically demonstrated. However, no commercial market has been established for any of these applications. This can be attributed to two reasons:

- The global problems encountered by any nuclear application are world wide
- The availability of fossil fuel alternatives in these markets. Generally, these alternatives are considered cheaper and easier to manage in a general sense: technically, financially and from the project, regulatory and environmental points of views

The most essential conditions and requirements for decentralized nuclear power heating plants (DNHPP) include the following:

- Energy sources virtually run under autonomous conditions, they have to control varying loads, and therefore need high manoeuvrability
- Great distance away from the industrially developed regions and centres of the country, the seasonal character and unreliability of transport communications result in rigid requirements for energy source reliability, simplification of their operation and maintenance
- Severe climatic conditions mean rigid requirements for the security of the consumers' power supply. In particular, it involves heat supply

The main part of DNHPP specific features is the following:

- Traditionally verified and proved technical solutions
- Natural circulation of primary coolant under all operating conditions

- Natural coolant circulation in all circuits with a total loss of power or some other measures ensuring the passive principle of decay heat removal
- Low power density of fuel
- Large heat-accumulating capacity of reactor
- Integral arrangement of the primary circuit

More than 10 different DNHPP concepts for remote earth regions in Russia, the USA, Japan and Argentina are known today throughout the world. Most of them are based on LWR technology. Some of them represent a floating based design line. One HTGR concept is also under consideration. The unit capacity is from 3 to 340 MWt.

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F1

Format	Description
Title	CAREM-25 (Conjunto Argentina de Ractores Modulares)
Application of the reactor	NHPP
Reactor type	PWR
Power output	Small power/100 MWt
Organization (name)	INVAP, Argentina
Development status	Studies

Description

The CAREM-25 reactor plant applications considered include power generation, industrial steam production, water desalination, and urban heating for small grids or isolated location. The CAREM-25 reactor plant may generate up to 25 MWe. To overcome the increased specific cost due to the very small scale, standard design and modulized factory construction are envisaged.

The main design features of the CAREM-25 plant are:

- o Integrated arrangements of the RCS with a once-through, helical tube steam generator which is located at the top of the downcomer
- o Natural circulation of the coolant in the primary circuit
- o Steam-water pressurizer on the top of the RPV
- o Shut-down condenser, in the upper part of the RPV, provides for the removal of decay heat by the evaporator system in the environment for 72 hours after shut-down
- o Emergency core cooling injection envelops only the upper part of the RPV with the penetration

During the LOCA event the steam generated from the core will pass into the steel primary containment vessel and condense. Passive air cooling will dissipate the heat and condense the steam.

Schedule

The basic conceptual study of the CAREM-25 plant is already completed. In 1996, the CAREM-25 reactor plant will be ready for construction. The construction of the 100 MWt CAREM-25 plant may start as early as 1996.

CAREM-25 Unit Data

Net electrical output		25 MWe
Core		
-	Average core power density	55 kW/l
-	Average burn-up	22 MWd/kg
-	Initial enrichment	3.4 %
-	Active core height	1.4 m
-	Core diameter	1.29 m
-	Number of fuel assemblies	61
-	Number of fuel rods per assembly	108
-	Number of control assemblies	25
Reactor cooling system		
-	Pressure	122.5 bar
-	Core temperature, outlet/inlet	326/284 °C
Reactor pressure vessel		
-	Inside diameter	2.84 m
-	Height	11 m
Steam generator		
-	Number	12
-	Type	once-through
-	Feed water pressure	50 bar
-	Feed water temperature	200 °C
-	Steam pressure	47 bar
-	Steam temperature	290 °C
Containment		
-	Type	Pressure suppression
-	Dimension (height/diameter)	15.8/11.5 m

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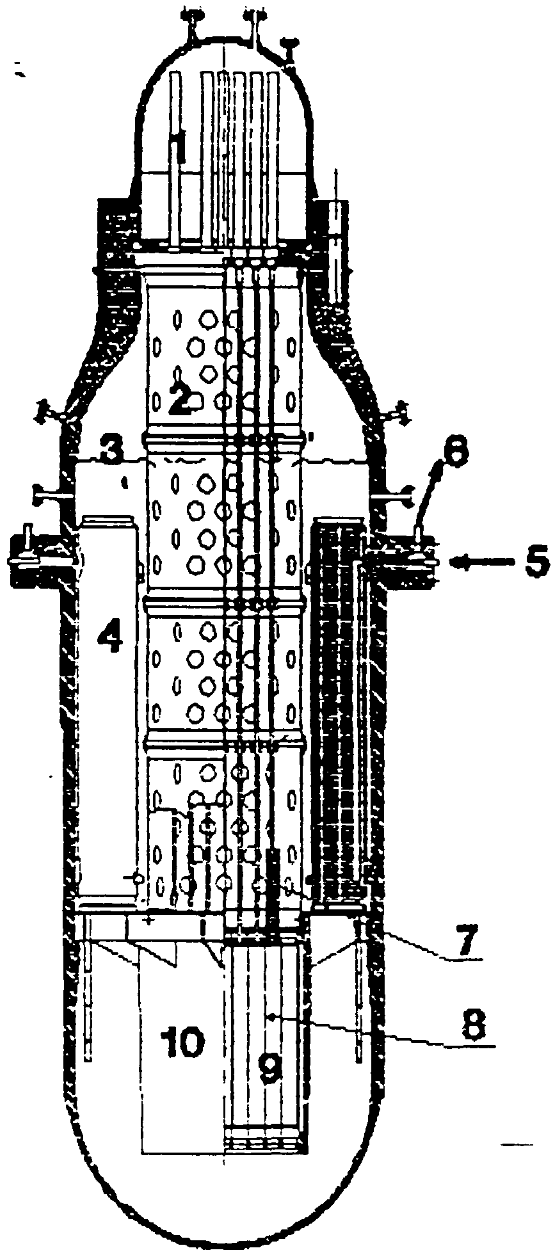

Figure F1
CAREM-25: Reactor coolant system.

1 Hydraulic control drive
2 Control drive rod structure
3 Water level
4 Steam generator
5 Water inlet
6 Steam outlet
7 Absorbing element
8 Fuel element
9 Core
10 Core support structure

F2

Format	Description
Title	PAES-100
Application of the reactor	NHPP
Reactor type	PWR
Power output	Medium power/160 - 170 MWt
Organization (name)	OKBM, RF
Development status	Commercial
Description	The PAES-100 floating nuclear plants can supply with power the costal industrial and remote areas and regions. Single unit concept: reactor - turbine - generator - power line are adopted for the PAES-100 power system. The plant design allows the customer to co-generate electric power, heat and drinking water according to the desire in any proportion. In the PAES-100 power system it is supposed to use two KLT-40 nuclear steam supply systems. KLT-40 (see G2) has development and operation experience of equipments and systems in Russian ice-breakers ("Arktika", "Sibir", "Russia"). Main components of the PAES-100 reactor plant are the following: reactor, four vertical steam generators and four primary circulating pumps are connected by short load-bearing-nozzles and form a steam generating unit. Core fuel assemblies contain pin-fuel elements. The steam generator is a one-through heat-exchanger, the tube system of which is made of the cylindrical helical corrosion resistant coil. The main circulation pump is a centrifugal, single stage, glandless unit with a canned two-speed motor.
Schedule	Two PAES-100 floating nuclear plants are intended for the remote north and far-east regions of Russia. Taking into account the availabilities of KLT-40 NSSS with equipment and systems already developed, the work on building of a floating PAES-100 reactor plant may be started immediately.

PAES-100 Unit Data

Main ship dimensions

-	Length	120 m
-	Width	30 m
-	Draught	6 m
-	Displacement	16 000 - 20 000 t

Reactor plant

-	Electric power (brutto)	100 MWe
-	Reactor heat rating	160 - 170 MWt
-	Reactor number	2
-	Steam temperature	290 °C
-	Steam pressure	40 kgs/cm^2
-	Lifetime of reactor vessel, main component vessels and pipelines	40 years
-	Staff	60
-	Primary coolant pressure	128 bar
-	Primary coolant temperature inlet/outlet	278/312 °C

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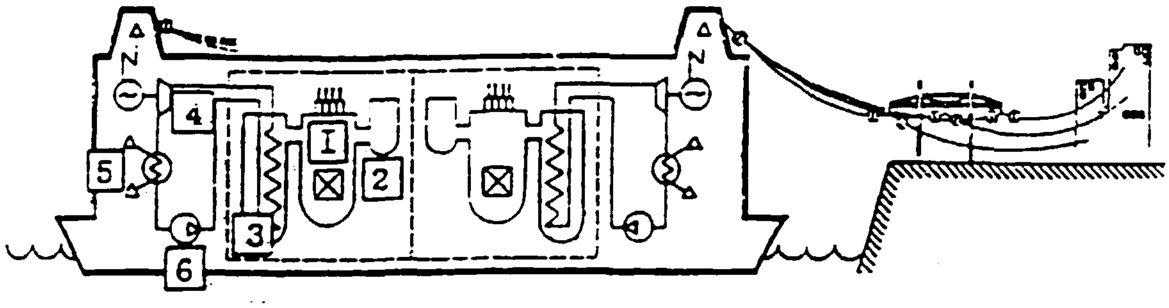

Figure F2
PAES-100: Layout.

1 Reactor
2 Reactor coolant pump
3 Steam generator
4 Turbo generator
5 Condenser
6 Secondary coolant pump

F3

Format	Description
Title	TRIGA Power System (TPS)
Application of the reactor	NHPP
Reactor type	PWR
Power outpu.	Small power/64 MWt
Organization (name)	General Atomic, US
Development status	Commercial
Description	The TRIGA Power System (TPS) is based on the proven TRIGA research reactor developed by GA technologies, which has been successfully operated during more than 800 reactor years world-wide. The concept uses a reactor pressure vessel to achieve temperatures adequate for a district-heating network. The primary system consists of a reactor unit and a heat-exchanger unit. A primary coolant pump is located between these two systems. The RPV is located at the bottom of a large tank filled with water, which provides an emergency heat sink. Particular safety features of the TRIGA concept are, on the one hand, the uranium-zirconium hydride fuel with its prompt negative temperature coefficient, and on the other hand the transition from normal operation to decay heat removal after failure of the forced convection without any active intervention by means of a pressure-balanced venturi nozzle.

An important feature of the TPS module design, which is motivated by safety, is the segregation of hot water from cold water. The intent is to minimize the amount of primary coolant that is available, the flash to steam in the event of a reactor pressure boundary rupture and thus ensure an adequate coolant inventory for decay heat removal for accidents. The amount of hot water involved in normal operation heat transfer is only about 9 % of the total reactor-free volume. The remaining water is maintained at about 71 °C by the auxiliary cooling system, which operates continuously by natural convection.

Optimum TPS electric power output is achieved with a sealed supercritical organic Rankine cycle with a single turbine generator.

Although several arrangements for extracting heat are possible, the preferred system arrangement is a cogeneration heat-exchanger located in series with a primary heat-exchanger.

Schedule

No TPS reactor plant for decentralized co-generation has been ordered yet.

Triga Power System Unit Data

Nuclear heat source		
-	Reactor power	64 MWt
-	Primary pressure	31.2 bar
-	Core inlet temperature	182 °C
-	Core outlet temperature	216 °C
-	Pool temperature	71 °C
-	Primary coolant flow	418 kg/s
Power conversion system		
-	Working fluid	R-114
-	Peak net power output	11.8 MWe
-	Turbine inlet temperature	204 °C
-	Turbine inlet pressure	41.4 bar
-	Flow rate	402 kg/s
-	Minimum load	1.0 MWe
-	Cold start time	$\pm$1.5 hours
-	Step load change	$\pm$15 %
-	Rate of continuous load change	$\pm$10 %
Desalination system		
-	Type	Reverse Osmosis
-	Peak water production	53 000 m^3/day
-	Specific power consumption	5.3 Wh/m^3
Heat system		
-	Peak thermal power	62 MWt
-	Peak temperature	188 °C

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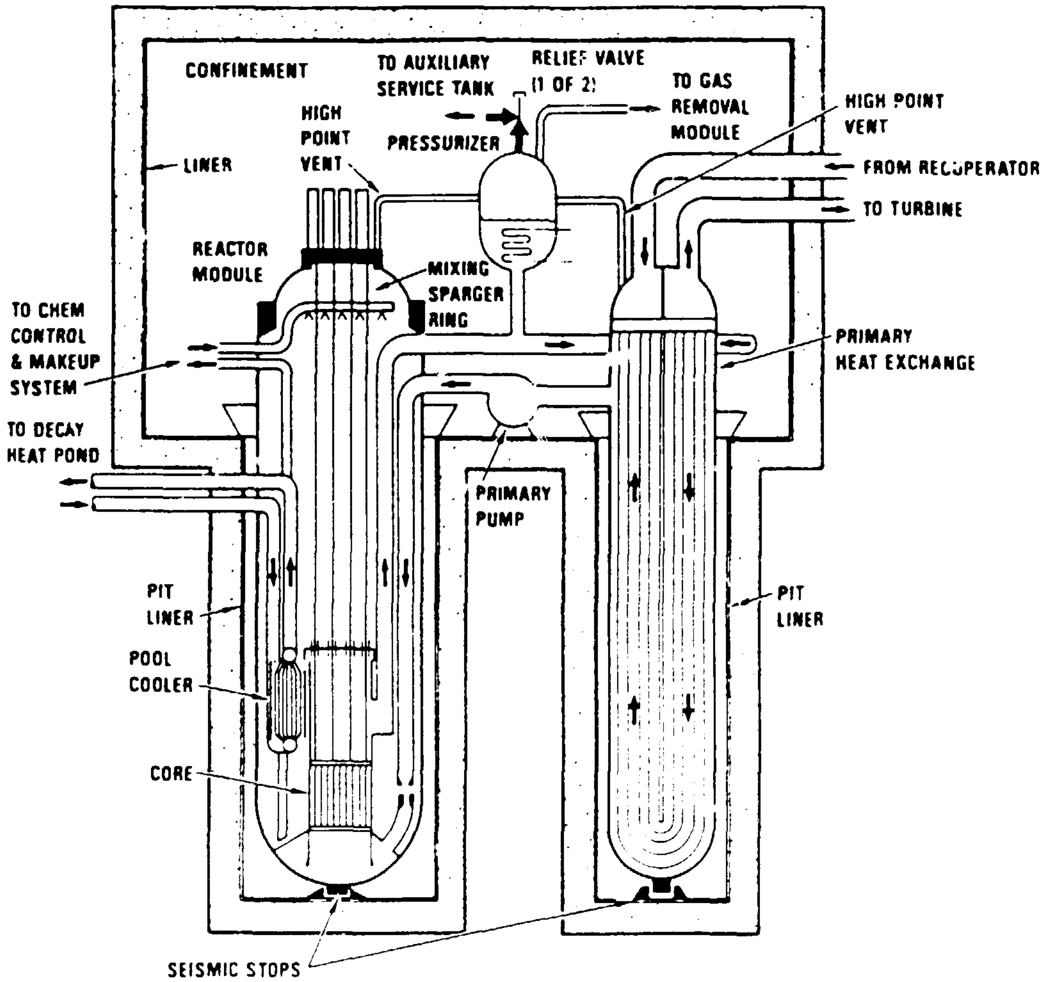

Figure F3
TPS: Nuclear heat source arrangement.

F4

Format	**Description**
Title	ABV
Application of the reactor	NHPP
Reactor type	PWR
Power output	Small power/60 MWt
Organization (name)	OKBM, RF
Development status	Detailed design
Description	A 60 MWt floating NPP with two ABV integral PWR-type reactors with primary coolant natural convection and passive safety systems is being developed in Russia for electric power and heat production in Northern and North-Eastern regions. The ABV reactor is provided with a gas pressurizing system. The steam generator tube system is located in the upper part of the annulus between the replaceable unit and reactor vessel. The tube system comprises some sections, each of them comprising some assemblies. The possibility of disconnection is provided in the event of leaks. Heat for prolonged emergency cooling is transferred, due to natural convection, to cooling system loops with pure water store evaporation (in steam generators) into the environment. Cooling system efficiency and water storage is such that residual heat removal may be carried out during several weeks without power and water supply from outside. Complete sluggishness of the ABV safety system operating without power and water consumption, is achieved due to the application of self-actuated equipment without conventional automatic protection circuits. The possibility of simultaneous electric power and heat supply to users in different proportions along with preservation of reactor thermal output provides for acceptable economic efficiency for energy sources, in spite of the low unit power of ABV plants.
Schedule	Ready for construction. One ABV reactor plant is intended for the decentralized north region of Russia.

ABV Unit Data

Reactor thermal output	60 MWt
Vessel height	4.8 m
Vessel diameter	2.6 m
Reactor volume	not less than 8 m^3
Primary pressure	15.7 bar
Temperature at core inlet/outlet	233/340 °C
Seismic stability by	
MSK-64 scale	9 magnitude
Lifetime	50 years
Number of fuel assemblies	55
Number of fuel elements in assembly	102
Number of control rods	6
Maximum linear heat rating	180 W/cm
Refuelling period	3 years

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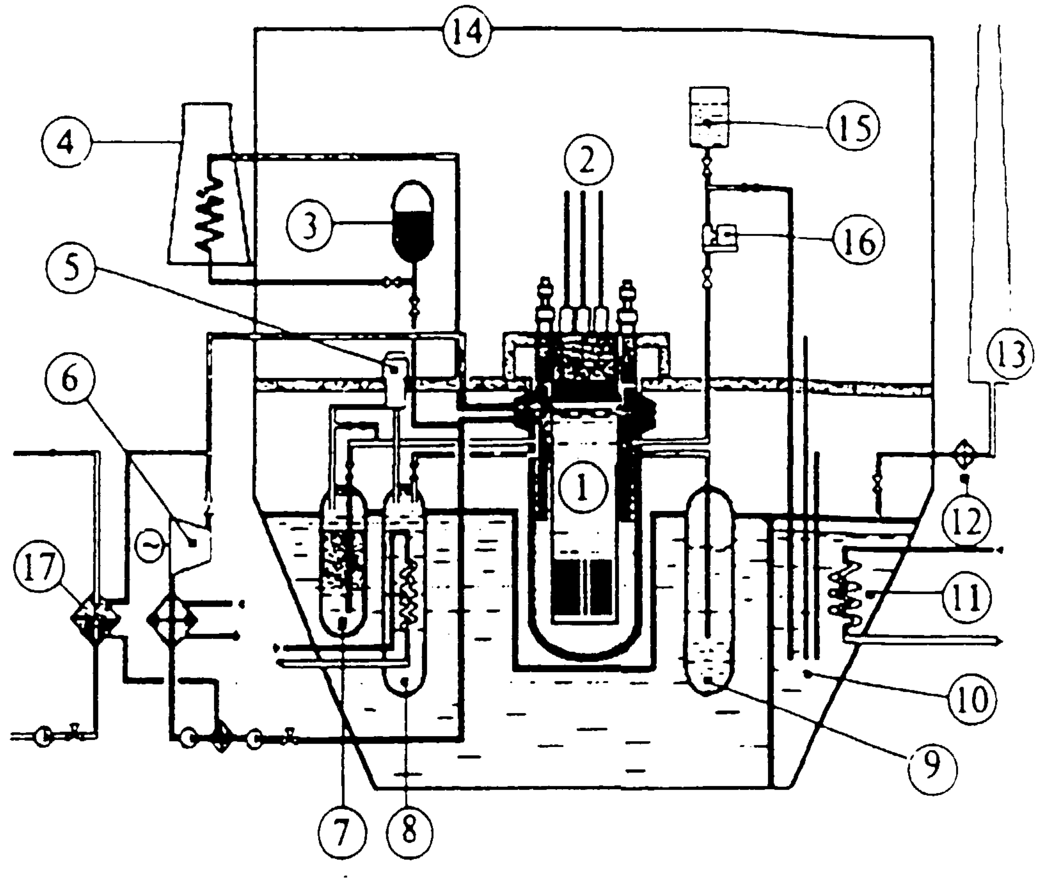

Figure F4
ABV: Power unit schematic.

1	Reactor	10	Bubbler
2	Control rod drive	11	Bubbler cooler
3	Accumulator of emergency cooling down systems	12	Filter
		13	Ventilation tube
4	Heat exchanger	14	Protective shell
5	Purification system pump	15	Water store tank
6	Turbine	16	Flushing-through system pump
7	Ion-exchange filter	17	Loop heat exchanger
8	Purification system cooler		
9	Pressurizer		

F5

Format	**Description**
Title	ELENA
Application of the reactor	NHPP
Reactor type	PWR
Power output	Very small power/3 MWt
Organization (name)	RRC-KI, RF
Development status	Basic design
Description	The unattended self-controlled nuclear thermoelectric DNHPP "Elena" is intended for the supply of a small settlement with heat (up to 3 MWt) and electricity (up to 0.1 MWe). This reactor can also be supplied with a desalination facility. The DNHPP "Elena" does not require human intervention during its total service lifetime (about 25 years). Its technological scheme does not include valves, mechanisms and devices requiring any operations and maintenance during lifetime. During its lifetime the plant operates on the initial core loading. The control and safety system including control rods, drive mechanisms, sensors etc only serves for the start-up and the shutdown of the reactor as well as for the trips under emergency conditions. The main features of the plant are the following: low power, small size, low power density, underground location. A reactor core consisting of the standard VVER-type fuel elements. A level of the negative temperature reactivity feedback makes it possible to compensate for the burnup effects by an insignificant reduction of the coolant temperature. Therefore the reactor does not need the excess reactivity for the burnup. There is natural convection of the primary and secondary coolant. Due to a low core power density (~3kW/l) the reactor does not require emergency cooling. Use of the thermoelectric energy conversion permits the reactor plant to be substantially or significantly simplified. The reactor and thermoelectric generators are accommodated inside the protection vessel filled with water of the secondary circuit. Due to the use of three strong steel vessels, containment and air-tight concrete box, the possibility that the nuclides will penetrate into the environment is ruled out. Also, the "Elena" type power source for deep sea application is under consideration in RRC-KI.

Schedule

Ready for construction. No "Elena" reactor plant has been ordered yet.

ELENA Unit Data

Power supplied to a district heating system	up to 3 MWt
Electric power	up to 100 kWe
Temperature of water supplied to a district heating system	up to 90 °C
Power unit size, diameter/height	4.5/12 - 14 m
Weight (without primary and secondary coolants)	160 t
Distillate output	up to 60 m^3/h
Distillate salt content	less than 25 mg/l
Operational life	up to 25 years

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Nuclear Engineering International, 1992, November, p 11-12.

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Moscow, RF, 1994, Report RRC "Kurchatov Institute".

1 Core
2 Protection screens
3 Strong vessel
4 Additional ensurance shell guard vessel
5 Protection vessel
6 Containment
8 II-III circuit heat exchanger
9 Heat exchanger of district heating system
10 Heat exchanger for residual heat removal
11 Pressurizer (or residual hat removal system)
12 Valves
13 Hot water supply pipes
14 Prime circuit pressurizer
15 Second circuit pressurizer
16 Concrete box wall
17 Concrete box top
18 Auxiliary units room
19 Control system driver
20 Electric power output

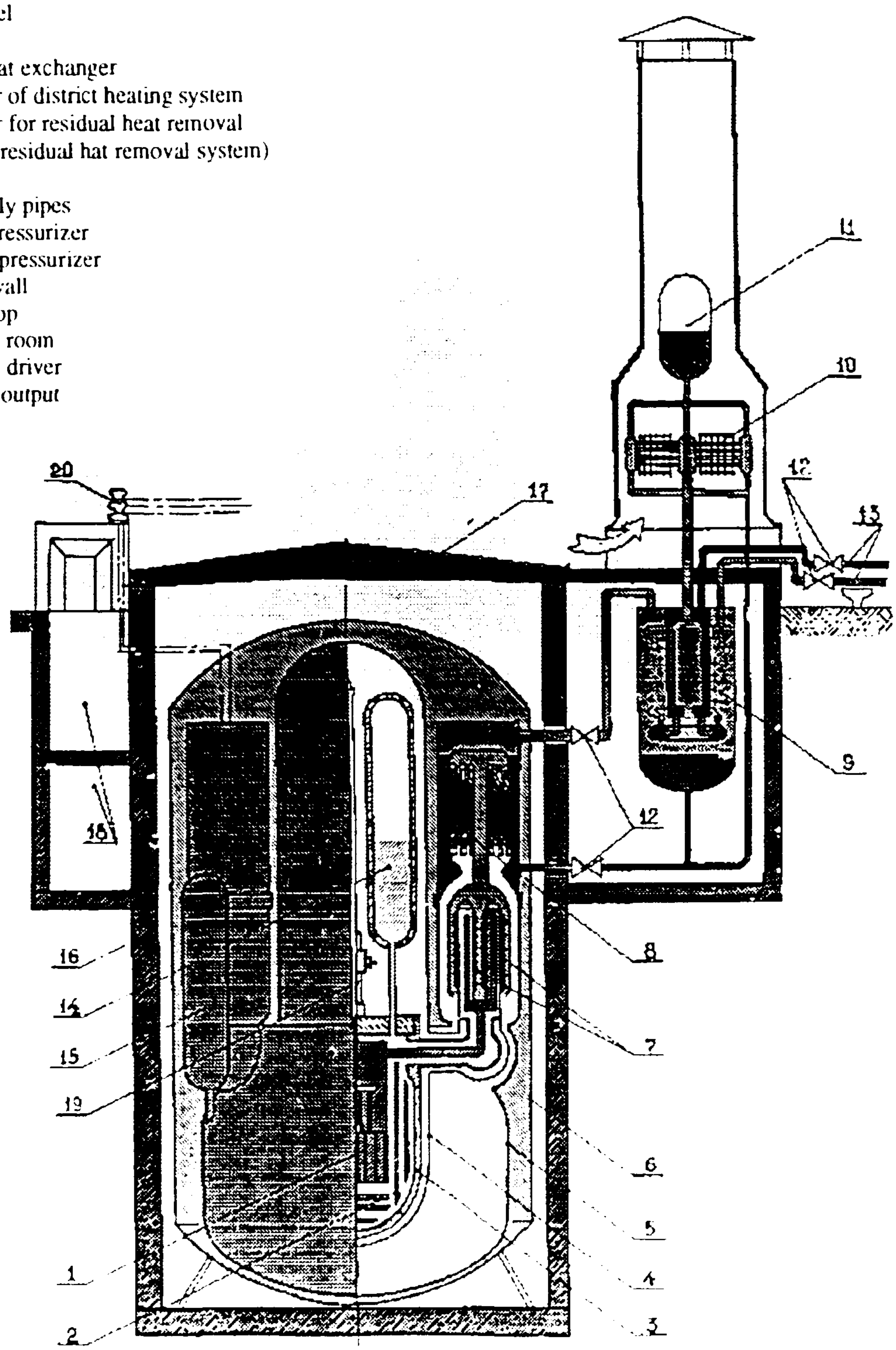

Figure F5
ELENA: Schematic.

F6

Format	Description
Title	Compact HTGR Gas Turbine
Application of the reactor	NHPP
Reactor type	PWR
Power output	Small power/29 MWt
Organization (name)	General Atomic, US
Development status	Studies
Description	The gas turbine has the highest output per unit weight of any currently available technology, but it requires gas inlet temperatures that can only be delivered by an HTGR core with TRISO-type particle fuel. The main elements of the Compact HTGR Gas Turbine system for remote sites are the reactor core, turbo compressor, power turbine, recuperator, precooler and generator. This design has an integrated arrangement. The core is surrounded by a combined graphite BeO reflector material. Control rods are located in the reflector. In the 10 MWe study the turbo machinery was conveniently divided into two horizontal sections, a free-running turbo compressor, the function of which is to provide the energy to transport the helium through the primary system and a separate power turbine driving the generator. A very compact 1 MWe unit is best suited to a single shaft vertical unit.
Schedule	No Compact HTGR Gas Turbine plant has been ordered yet. This program has a low funding level. As any gas turbine, the HTGR concept will have unique safety problems resulting form turbine overspeed or shaft failure and requires a decision on whether or not an intermediate heat-exchanger is needed for an early demonstration project.

Compact HTGR Gas Turbine Unit Data

Electric output	10 MWe	1 MWe
Core thermal power	29 MWt	3.3 MWt
Core outlet temperature	850 °C	850 °C
Core diameter	1.4 m	0.67 m
Core height	1.4 m	0.67 m
Core power density	13.5 w/cm^3	14.0 w/cm^3
Fuel element type	Cylindrical	Cylindrical
Number of elements	175	42
Approximate core life	7 years	7 years
Turbine pressure ratio	2.0	2.0
Turbine speed	8 000 rpm	12 000 rpm
Compressor speed	36 000 rpm	12 000 rpm
Vessel diameter	3.0 m	1.5 m
Vessel height	7.0 m	6.1 m

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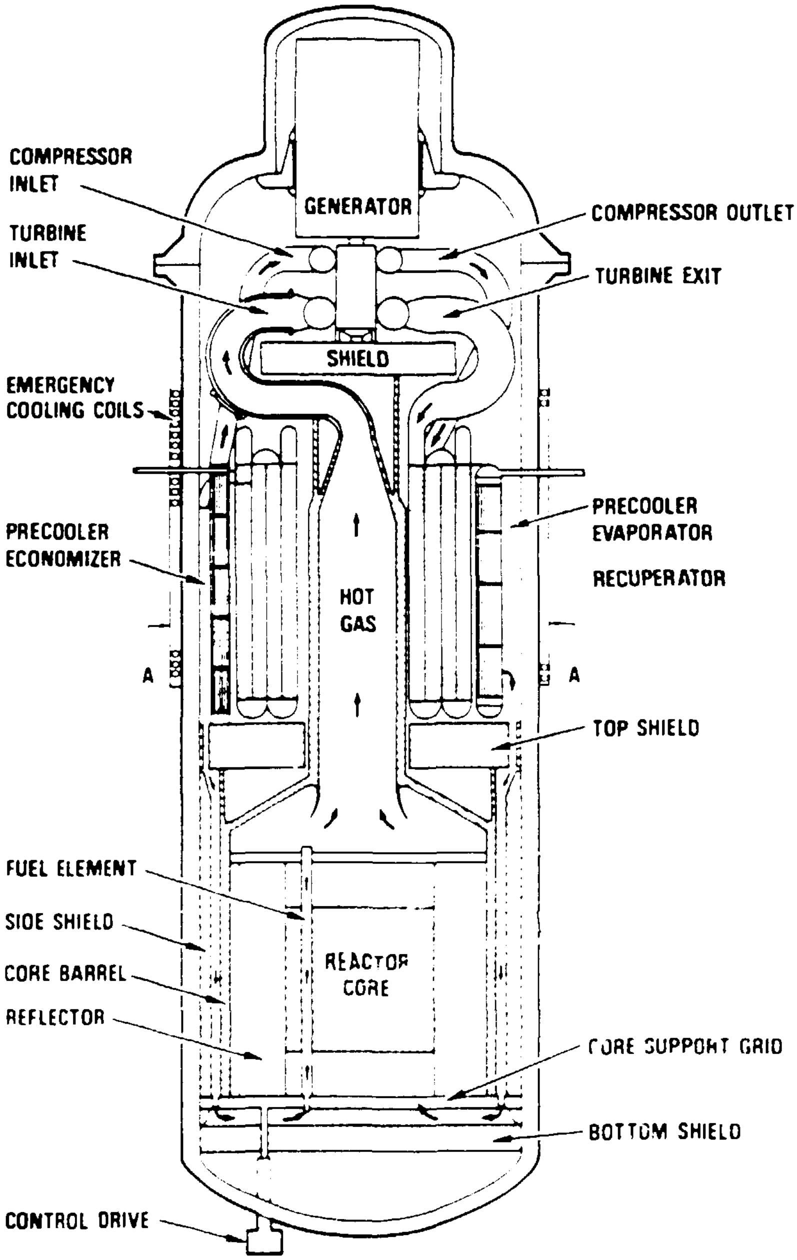

Figure F6
Compact HTGR Gas Turbine: Reactor vessel and internals.

G Ship Reactors

G0 Introduction

Civil nuclear ships have various excellent characteristics compared to conventional ships. The use of nuclear energy opened the possibility of designing ships of a high ice-cutting capability due to the absence of power restrictions and practically unlimited operation time (typically 400 days with a power utilization factor of about 0.6). Experience has shown that these ships are economically competitive with diesel ice-breakers. It gives a strong motivation to develop marine reactors and nuclear ships. At present, two main applications are relevant for ship mobile systems (SMS):

- o For ships navigating on sea surface
- o For power source used in deep sea

These concepts are now under development in Russia and Japan.

Russia has had excellent experience for about 30 years in the field of ice-breakers. At present, Russia operates ice-breakers of the "Arctica"-type. Their capacity is from 34 to 49 MW. The Russian concept of Arctic navigation development includes the necessity of the following classes of ice-breakers:

- o 30 - 35 MW ice-breakers for shipping in shallow waters of the Arctic coast and Siberian rivers
- o 60 MW ice-breakers for piloting transport ships
- o 110 MW ice-breakers designed for reliable all year round ship piloting

In Japan, after successful demonstrations of the operation of the nuclear ship "MUTSU", new research and development programs are now under way concerning nuclear ships and mobile systems. In JAERI the basic concepts of two types of marine reactors have been developed: the first one is for an ice breaking scientific observation ship and the latter is for deep-sea research.

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G1

Format	Description
Title	OK-900A
Application of the reactor	SMS
Reactor type	PWR
Power output	Small power/170 MWt
Organization (name)	OKBM, RF
Development status	Commercial

Description

The first generation of naval reactor plants was represented by an OK-150 loop type PWR that operated on the "Lenin" ice breaker in 1959-1966 and was subsequently replaced by a more modern one of OK-900 type NSSS.

Modular reactor plants of the OK-900 type have been developed for the next generation of ice breakers of the "Arctica" type. These systems are radically different from those of the first generation in their technical operational and economic parameters. Main components of the OK-900A NSSS including steam generators and primary circulating pumps are connected by short load bearing-nozzles and form a steam generating unit.

The reactor has inherent safety properties due to:

- o Negative reactivity coefficients of the core
- o Passive principle of reactor shutdown by control clusters
- o High heat accumulating capacity
- o Elimination of large diameter pipelines in the primary circuit and insertion of devices limiting coolant discharge at pipeline ruptures.

Long term operation of OK-900A NSSS in ice breakers has demonstrated that they completely satisfy the necessary requirements. The service time of NSSS equipment on the "Arctica" and "Siberia" ice breakers that have been in operation since 1974 and 1978, respectively, is close to 100,000 hr without any replacements of the system equipment. There have been no reactor plant accidents that have resulted in even a temporary interruption of the ice breaker operation.

Schedule

No new OK-900A reactor plant has been ordered yet.

OK-900A Unit Data

Type	OK-900	OK-900A
Number or reactors	2	2
Independence with one fuel loading	1 050 days	1 050 days
Total shaft power	44 000 hp	75 000 hp
Rated thermal reactor power	2x159 MWt	2x171 MWt
Primary pressure	128 bar	128 bar
Primary coolant temperature, inlet/outlet	277/320 °C	277/320 °C
Steam production	2x220 t/h	2x240 t/h
Secondary system		
- temperature	305 °C	290 °C
- steam pressure	30.9 bar	32.4 bar
Ice breaker	"Lenin"	"Arctica", "Siberia", "Russia", "Sov.Union", "Oct.Revolution"

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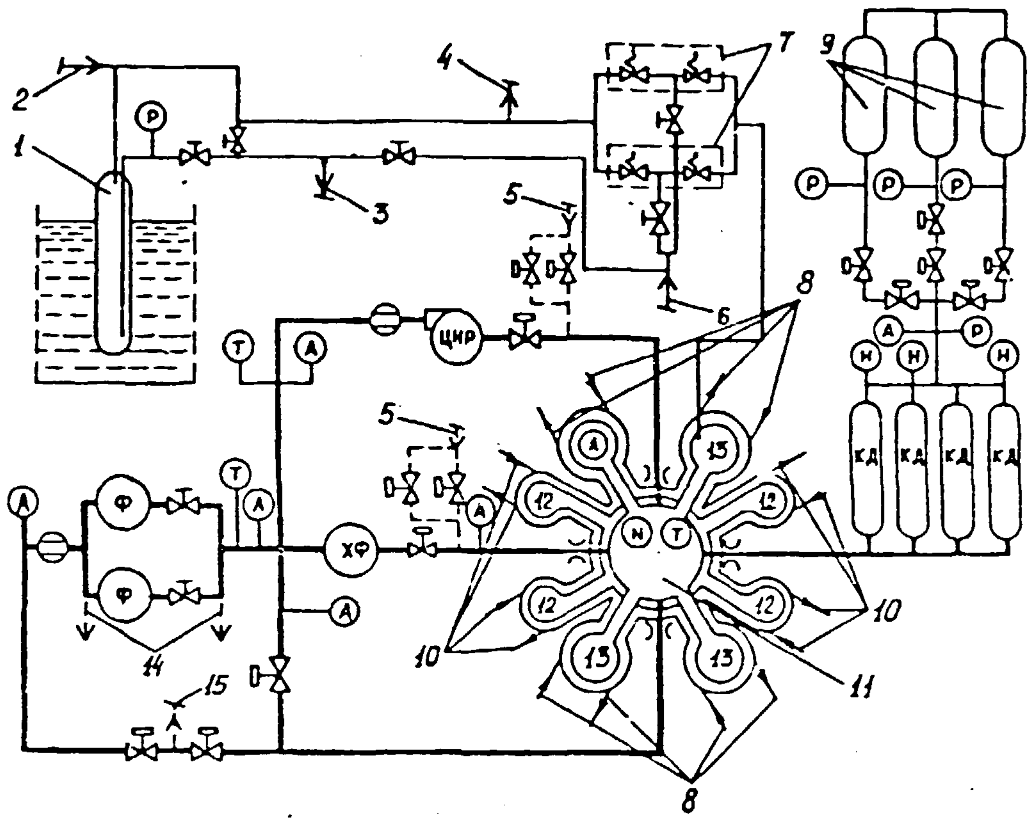

Figure G1
OK-900A: Reactor coolant system.

1	Expansion tank	8	Water inlet, outlet
2	Low pressure air system	9	Pressurizer system
3	Drainage	10	Water inlet, outlet
4	Air removal system	11	Water inlet, outlet
5	Makeup and emergency spray system	12	MCP
6	Pressure system	13	IHX
7	Safety device		

G2

Format	Description
Title	KLT-40
Application of the reactor	SMS
Reactor type	PWR
Power output	Small power/170 MWt
Organization (name)	OKBM, RF
Development status	Commercial

Description

The new generation of naval reactor plants represented by KLT-40 and KLT-40M reactor plants, designed for a new series of ice breakers of the "Timyr" type and "Sevmorput" large carrier corresponding to modern international requirements of nuclear safety KLT-40 reactor plants, was developed with the maximum use of the proven components of OK-900 NSSS (see G1).

The main differences between the KLT-40 and OK-900 reactor plants deal with the improvements related to the higher safety of nuclear facilities, their maintainability and better operating lifetime. In particular, the reactor design is simplified, measures are taken to improve the monitoring of the reactor vessel, its cover, drives of the control and safety system. The main modernization measures were implemented in the reactor plant structure covering cleaning system ducts, cooling down system and the pressurizer, and measures to reduce emergency coolant leakages.

Main safety systems of KLT-40 are:

o Control and protection system (three redundant supply lines)

o Liquid absorber injection system

o Residual heat removal system (through steam generator or through the heat-exchanger into the sea or air - two independent channels)

o Emergency core cooling (two independent channels: hydroaccumulators and electric pumps for water injection)

o System of emergency pressure reduction in the containment

o Containment system.

A more powerful reactor of the KLT type - KLT-3 reactor plant with thermal output 300 MWt is also developed by OKBM.

Schedule

No new KLT-3 or KLT-40 plants have been ordered yet.

KLT-40 Unit Data

Type	KLT-40	KLT-3
Thermal power	170 MWt	300 MWt
Electric power	50 MWe	90 MWe
Primary pressure	128 bar	146 bar
Primary coolant temperature		
- inlet	277 °C	274 °C
- outlet	320 °C	322 °C
Steam production	240 t/h	400 t/h
Steam temperature	305 °C	295 °C
Steam pressure	31 bar	40 bar
Feed water temperature	104 - 108 °C	194 °C

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Nuclear Steam Supply System KLT-40 enhanced safety as Independent Power Supply Source. Employment prospects.
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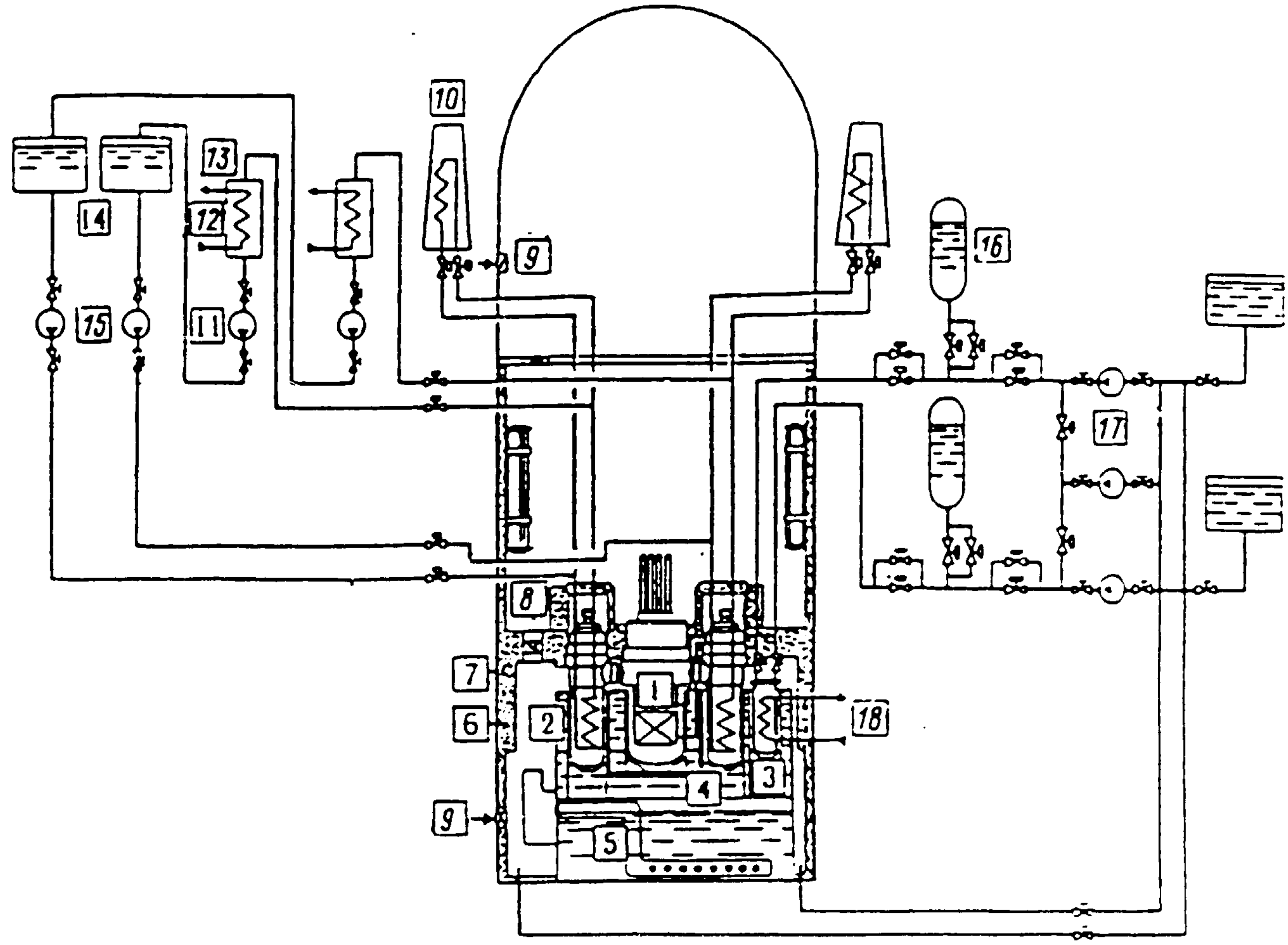

Figure G2
KLT-40: Reactor plant.

1	Reactor	10	Air cooling heat exchangers
2	Steam generator	11	Condensate pumps
3	Heat exchanger	12	Sea water
4	Metal/water shielding system tank	13	Condensers
5	Pressure suppression pool	14	Water storage tanks
6	Biological shielding	15	Emergency feed water pumps
7	Protective enclosure walls	16	Hydroaccumulators
8,9	Valves	17	ECCS pumps
		18	Third circuit

G3

Format

Description

Title	MRX
Application of the reactor	SMS
Reactor type	PWR
Power output	Small power /100 MWt
Organization (name)	JAERI, Japan
Development status	Basic design

Description

JAERI has conducted a design study of the Marine Reactor X (MRX) for ice-breakers. It has a 60 000 horse power propulsion system with two 100 MWt MRX reactor plants.

MRX is an integral type PWR and employs in-vessel type control rod drive mechanism and water-filled containment vessel including their reactor vessel surrounded by thermal insulation. At the upper part of the containment vessel, heat pipe type coolers are placed. Dip plates are also placed in order to enhance steam condensation in the event of loss of coolant accidents for mitigating containment water slashing.

The core consists of 19 fuel assemblies. Thirteen assemblies contain control rod clusters. Six are for reactivity control during operation and the other seven are for back-up reactor shutdown. Standard fuel pins with Zircaloy cladding (9.5 mm outer diameter), and burnable poison for reducing the peak factor, are also applied. Hexagonal arrangements are preferred rather than rectangular ones.

There are only eight pipes (50 mm diameter) carrying primary coolant and penetrating the reactor vessel. They include four for RCS/RHRS, two for pressurizer safety valves, one for aressurizer relief valve and one for pressurizer spray.

Steam generator tube inspection can be carried out from the secondary side after opening blind flanges on steam and feedwater headers. Main coolant pumps are inspected in an inspection pit. After the pumps are withdrawn horizontally from the vessel they are transferred to the pit.

Only small LOCAs can be possible in the MRX. In such a LOCA, the engineered safety system keeps the core flooded and removes decay heat without emergency water injection.

Schedule

Maintenance procedures for main components are now being studied. No MRX reactor plant has been ordered yet.

MRX Unit Data

Reactor power	100 MWt x 2 units
Reactor type	Integral type PWR
Reactor coolant	
- Operating pressure	120 bar
- Inlet/outlet temperature	282.5/297.5 °C
- Flow rate	4 500 t/h
Core/Fuel	
- Equivalent diameter	1.45 m
- Effective height	1.40 m
- Average linear heat rate	74 kW/m
- Average power density	43 kW/l
- Fuel type	UO_2 fuel rod
- U-235 enrichment	4 %
- Fuel inventory	6.5 t
- Fuel life time	8 years
- Fuel average burnup	25 GWd/t
- Number of fuel assemblies	19
- Fuel rod outer diameter	9.5 mm
Control rod drive mechanism	
- Type	In-vessel type
- Number of CRDMs	13
Main coolant pump	
- Type	Horizontal axial flow canned motor type
Number of pumps	2
Steam generator	
- Type	Once-through helical coil type
- Tube material	Incoloy 800
- Tube outer diameter	19 mm
- Tube inner diameter	14.8 mm
- Steam temperature/pressure	289 °C/40 bar

- Steam flow rate	168 t/h
- Heat transfer area	970 m^2
Reactor pressure vessel	
- Inner diameter/height	3.7/9.3 m
Containment vessel	
- Inner diameter/height	6.8/13.0 m
- Design pressure	40 bar
Overall length	150 m
Water line length	141 m
Maximum breadth	30 m
Water line breadth	28 m
Moulded depth	15 m
Full load draught	9 m
Full load disp	22 000 t
Continuous ice breaking ability	2.1 m thickness for level ice
Ship crew	60 persons
Observation personnel etc	40 persons
Propulsion shaft horsepower	20 000 ps x 3 shafts

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Proceedings International Conference Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Vol 1, p 6.5.

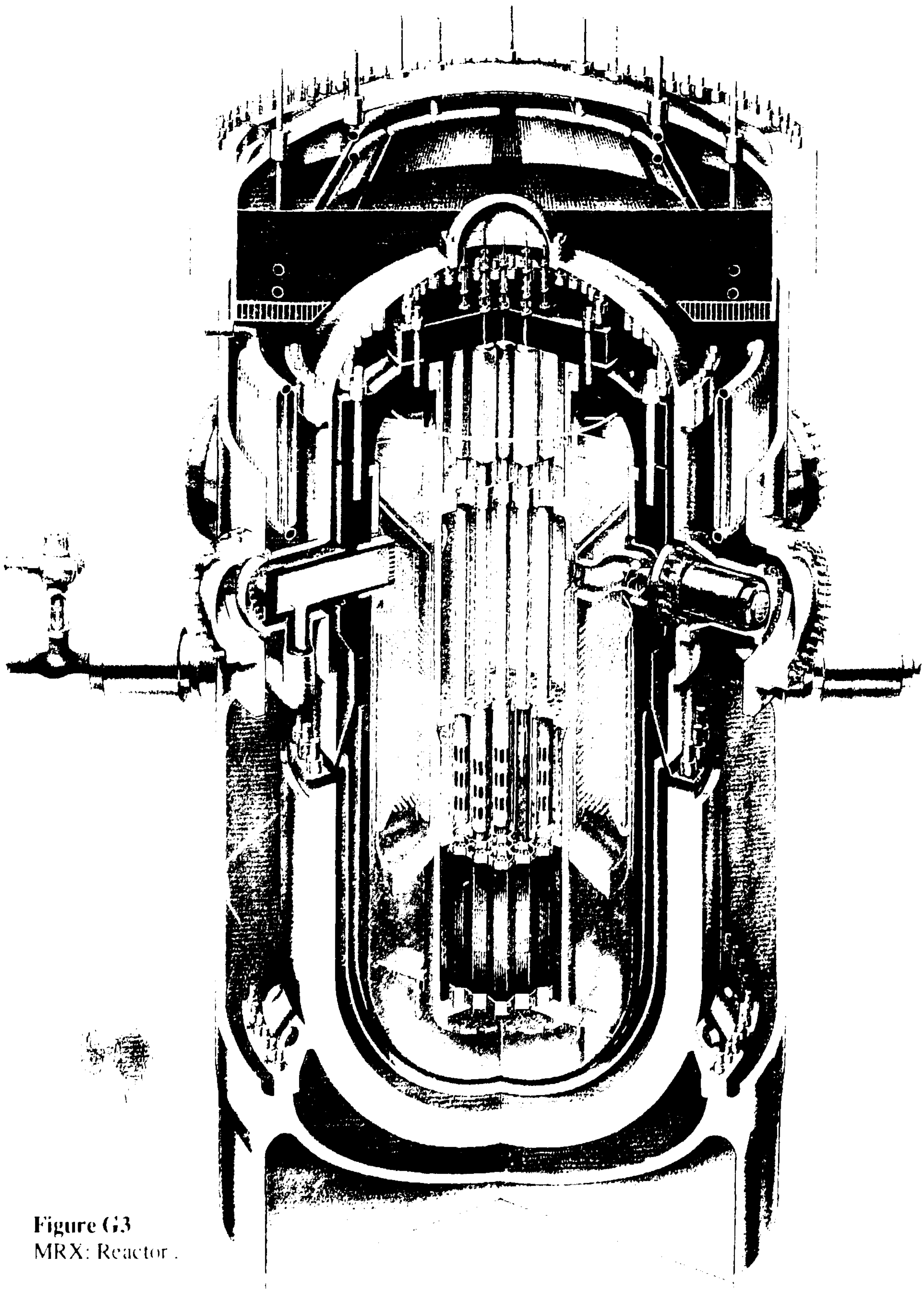
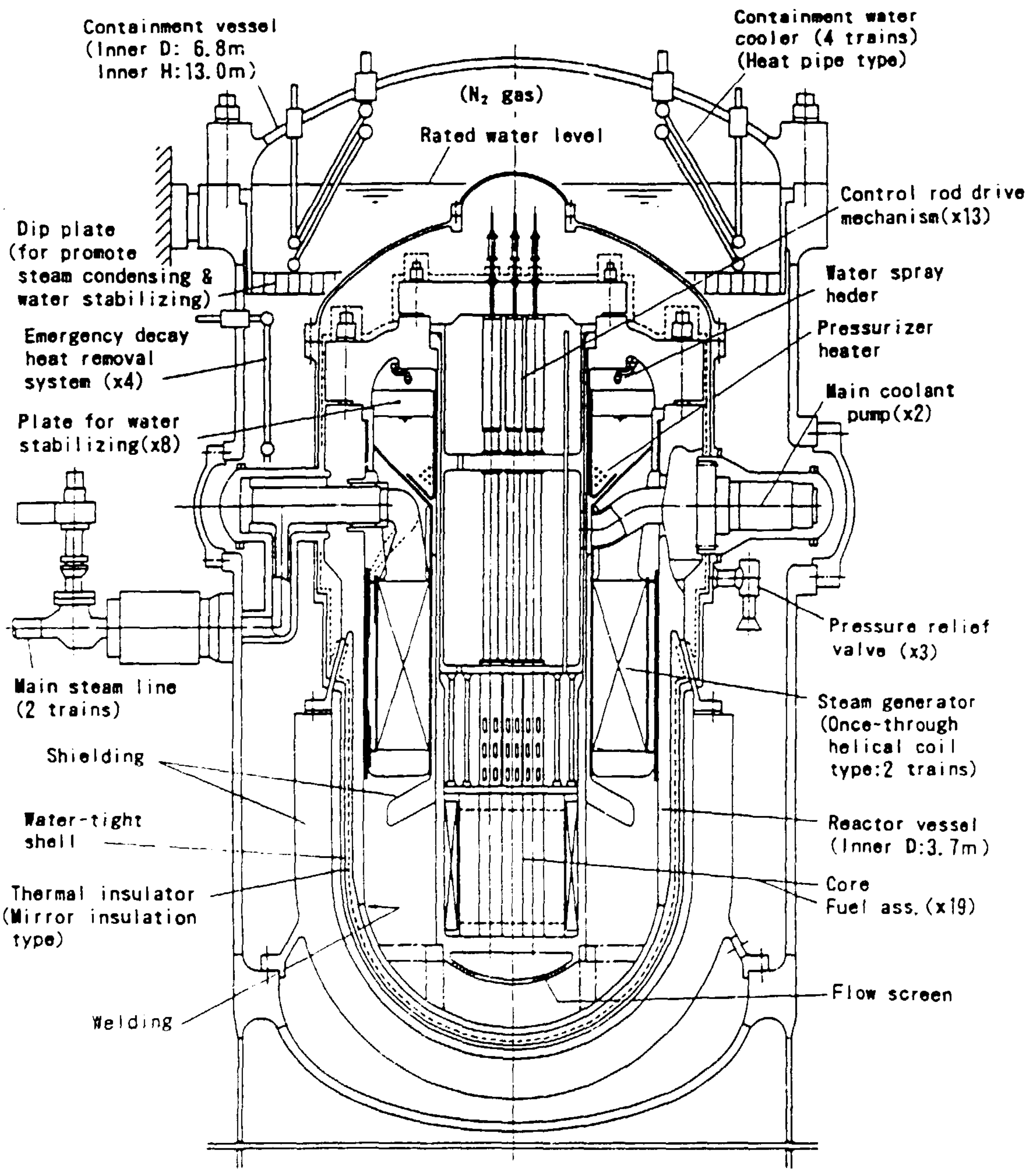


Figure G3
MRX: Reactor .


Containment vessel
(Inner D: 6.8m
Inner H:13.0m)
Containment water
cooler (4 trains)
(Heat pipe type)
(N_2 gas)
Rated water level
Control rod drive
mechanism(x13)
Dip plate
(for promote
steam condensing &
water stabilizing)
Water spray
heder
Emergency decay
heat removal
system (x4)
Pressurizer
heater
Plate for water
stabilizing(x8)
Main coolant
pump(x2)
Pressure relief
valve (x3)
Main steam line
(2 trains)
Steam generator
(Once-through
helical coil
type:2 trains)
Shielding
Water-tight
shell
Reactor vessel
(Inner D:3.7m)
Thermal insulator
(Mirror insulation
type)
Core
Fuel ass.(x19)
Flow screen
Welding

G4

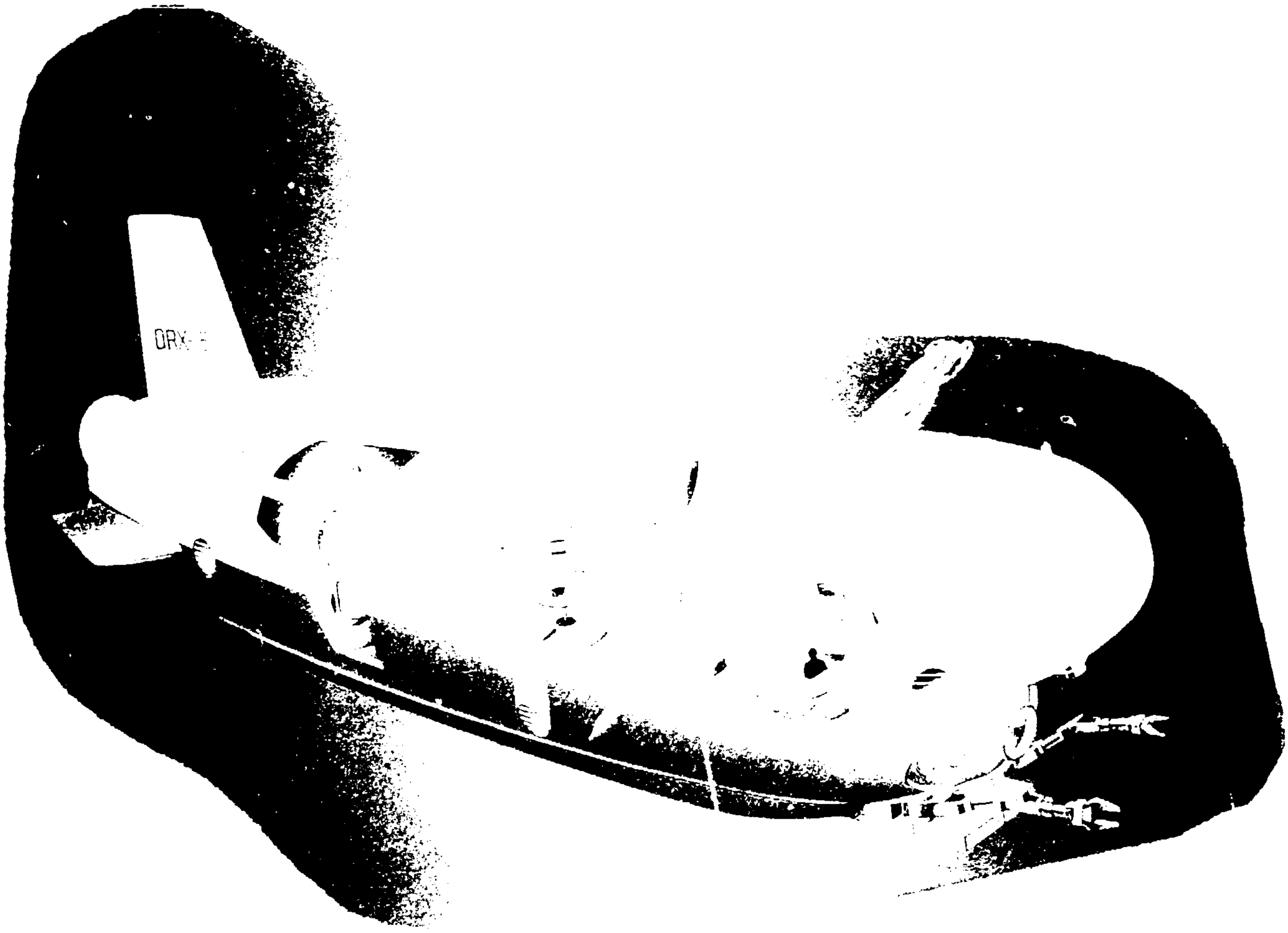
Format	Description
Title	DRX
Application of the reactor	SMS
Reactor type	PWR
Power output	Very small power/0.75 MWt
Organization (name)	JAERI, Japan
Development status	Basic design
Description	The Deep-sea Reactor X (DRX) is a small nuclear plant with electric output 150 kWe for undersea power sources, which can provide enough power capacity for a bathyscaph. It is a PWR with an integrated arrangement with a steam generator and turbine inside the reactor vessel. The combination of two spherical shells 2.2 m in diameter, made of titanium alloy, basically composes a pressure shell. The core can be operated for more than 400 days at 750 kWt, which corresponds to about 4 years operation without changing of the fuel assuming 30 % of effective load factor. The core consists of a single fuel assembly with a 36.8 cm equivalent diameter and a 34.4 cm effective height. The U-235 concentration of about 11 % is necessary to maintain criticality during 5 500 MWd/t burn-up. The BWR type DRX is also under consideration in JAERI.
Schedule	No MRX reactor plant has been ordered yet. JAERI has been conducting design studies of the DRX reactor plant.

DRX Unit Data

Maximum operating depth	6 500 m
Speed	
- Advance	3.5 kn
- Vertical, transverse	1.5 kn
Overall length	24.5 m
Breadth	4.5 m
Moulded depth	6.0 m
Continuous operating period	30 days
Reactor output	750 kWt
Generator output	150 kWe
Pressure shell inner diameter	
- Reactor container	2.2
- Living room	2.8 m
Number of crews (incl. scientists)	8

References

IIDA, H et al
Design study of deep sea reactor DRX.
Proceedings International Conference on Design and Safety of Advanced Nuclear Power Plants, Oct 25-29, 1992, Tokyo, Vol 1, p 6.6.


DRX

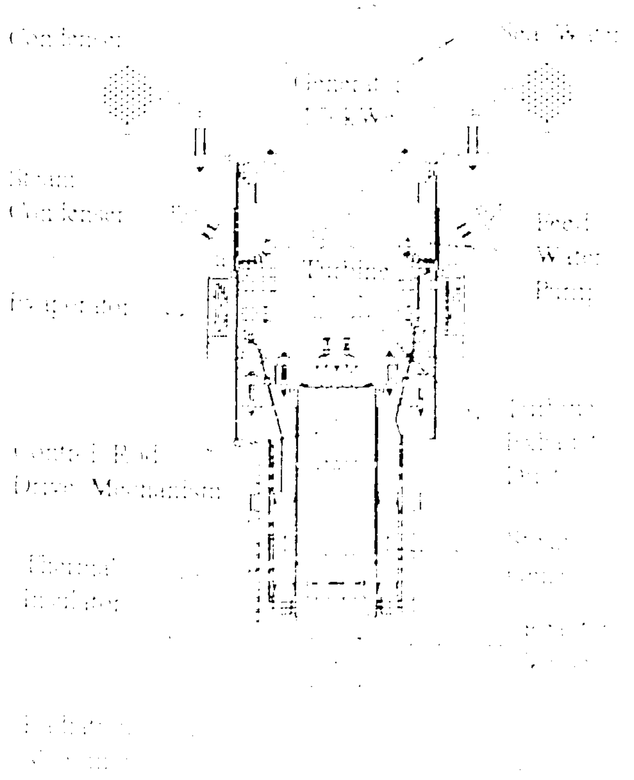

Condenser
Steam
Condenser
Turbine
Control Rod
Drive Mechanism

Figure G4

I Space Nuclear Reactors and Isotope Batteries

I0 Introduction

The field of space reactor designs is one of the most innovative areas in nuclear engineering and technology.

Since the start of the space age in the 50's, a set of nuclear power supply options has been developed by the US and Russia to support the military and civilian space programs.

The investigation and development of space nuclear reactors has mainly followed two lines:

- o Power sources for electricity generation
- o Nuclear propulsion

Space nuclear plants, which produce complex propulsion and electrical power are also discussed.

A generic classification of space nuclear power source types is depicted below:

Nuclear power system type	Electric power range (module size)	Power conversion
Radioisotope Thermoelectric Generator (RTG)	up to 0.5 kWe	Static: Thermoelectric
Radioisotope dynamic conversion generator	0.5 kWe - 10 kWe	Dynamic: Brayton or organic Rankine cycles
Reactor systems Heat pipe Solid core Thermionics	10 kWe - 1 MWe	Static: Thermoelectric, thermionics Dynamics: Brayton, Rankine or Stirling
Reactor system Heat pipe Solid core	1 - 10 MWe	Brayton cycle Rankine cycle Stirling cycle
Reactor Solid core Pellet bed Fluidized bed Gaseous core	10 - 100 MWe	Brayton cycle (open loop) Stirling MHD

Here, the primary system output is electrical energy, which is produced by converting radioisotope decay heat or the thermal energy released in the nuclear fission reactor. For conversion to electric power the static (thermoelectric or thermionics) or dynamic (e.g. Rankine or Brayton cycles) principles are used. For space power applications in the MWe and beyond range there is a number of interesting innovative nuclear reactor technology options.

From the late 1950's substantial efforts in different countries (primarily the US Rover-NERVA program) have also devoted to nuclear propulsion.

Nuclear rocket propulsion is attractive because nuclear heat generated in a high power density reactor core can be used to heat a low molecular weight propellant to high temperature, which enables a much higher propellant expulsion velocity than can be obtained in a chemical engine.

Three different categories of innovative reactor concepts have been considered for nuclear propulsion (solid core, liquid core and gas core). Each category contains several design concepts. In theory the gas core would produce the highest temperature, the liquid core intermediate temperature and the solid core lower temperature.

In 1973, the Rover-NERVA programs (based on solid core thermal engine) were terminated as a consequence of an indefinite delay of plans for space missions which would call on the capability of nuclear propulsion.

Since 1961, the US has launched another twenty civil and military nuclear space power systems. Russia had flying nuclear power systems in space since about 1965. These nuclear power sources have very small power nuclear reactors using thermoelectric and thermionic conversion principles (e.g. US SNAP-type and Russian Romashka and TOPAZ-I,II-types).

Russia has during the period 1970 to 1988 launched 31 space-crafts with thermo-electric nuclear reactors. In the late 80's Russia launched two space-crafts with the thermo-emission nuclear reactor Topaz 1. Later they developed the Topaz II thermo-emission nuclear reactor which has not yet found its application, but is currently tested in cooperation with the USA.

The USA have mainly used plutonium-238 batteries and have only launched the SNAP 2 nuclear reactor. The development of the US SP-100 nuclear reactor program, which started ten years ago, has recently been terminated and the technology achievements are nor transferred to the industry.

Some other countries (e.g. France and Japan) are also investigating the possible development of space nuclear reactors..

Further space activity will require even larger quantities of safe, reliable and autonomous power sources.

The following key issues can be addressed in the research field:

- o The physics of thermionic conversion (surface and near-surface phenomena on electrodes, the effect of modification of the surface on the operation of the output, plasma processes in the interelectrode gap, the effects on the thermionic energy conversion with fission products in the interelectrode gap
- o The research and development of fuel composites (including nitride and carbonitride fields, processes of fuel-cladding interaction, swelling processes, wastage, recondensation, and fission product behaviour)
- o Improvement of the structure (including geometrical optimization, selection of electrode materials, fission products release, methods of supplying Cs to the interelectrode gap, providing electrical insulation, switching, necessary thermal modes, and so on)
- o Reactor and structural materials (moderator, reflector, absorbers, thermal and electro-insulation, and cement compounds)
- o Coolant technology (corrosive processes and mass transfer in loops, methods and means of controlling the action of admixtures, and diagnostics and the prevention of leaks)
- o Improvement of methods for releasing waste heat (heat pipes, possible designs of radiators, liquid droplet, and other new types of radiators)
- o Measures to ensure lifetime
- o Diagnostics, identification, and information during ground and in-flight testing (sensors, instruments, methodology, and algorithms)
- o Guaranteeing nuclear and radiation safety

For the US and Russia 1994 - 1995 governmental proposals cut the research and development funding support and related facility funding for nuclear reactors that have no commercial or other identified application. At present the USA will perform the Cassini program for investigation of Saturnus and will use plutonium batteries as the internal power source.

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Fusion Technology, 1991, December, v 20, N4, part 2, p 698-710.

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Proceedings Nuclear Power Engineering in Space, Obninsk, RF, 15-19 May, 1990, p 136.

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1995, January 8-12, USA, NM, Albuquerque.

o Thermionic Energy Conversion Conference Proceedings, 1993, May 5-7, Sweden, Göteborg.

I1

Format	**Description**
Title	GPHS-RTG (General Purpose Heat Source RTG)
Application of the reactor	SNR, power source
Reactor type	Radioisotope thermoelectric generator
Power output	Very small power/0.3 kWe
Organization (name)	Westinghouse, US
Development status	Commerical
Description	The world's first "atomic battery" was developed in the US in the 1950s for Space Nuclear Auxiliary Power (SNAP) program. The RTG utilized the natural decay heat of a radioactive isotope, converting the heat directly into electricity via thermoelectrics. The advanced GPHS-RTG design is a general-purpose heat source and consists of a graphite block with Pu-238 pellets. Each source has a power output of 0.3 kWe. These RTGs were flown on the last two missions, Galileo and Ulysses. For lunar outpost applications interchangeable radioisotope heat source assemblies, the thermal source that drives the energy converter, may be developed. This concept allows heat source assemblies to be transferred between various generator units, thereby maximizing utilization of the Pu-238 supply. Advanced concepts, utilizing dynamic systems such as Sterling, or advanced static conversion are required to provide the lower powered mission projected by NASA. These advanced converters will also minimize the amount of Pu-238 required by their increased efficiency.
Schedule	As long as the US have programs for which "nuclear batteries" are power source, the DOE will be a key player in building and supplying power sources.

GPHS-RTG Unit Data

Power output	290 We (BOM) 250 We (minimum at EOM)
Operation life	40 000 hours
Weight	54.1 kg
Output voltage	28(±0.5) V
Envelope, diameter/longitude	46/< 115 cm
Hot junction temperature	1 275 K
Fuel	83.5±1 % $^{238}PuO_2$
Thermoelectric material	SiGe
Magnetic field	<10 x 10^{-9} T/m
On pad	30 days unattended capability
Storage life	1 year ground storage
Auxiliary cooling	375 K average outer shell temperature

BOM = Beginning Of Mission and EOM = End Of Mission

References

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Technical Issues Surrounding continued Supply of Radioisotope Thermoelectric Generator for NASA Programs.
Transaction American Nuclear Society, Space Nuclear Power, 1992, p 22-23.

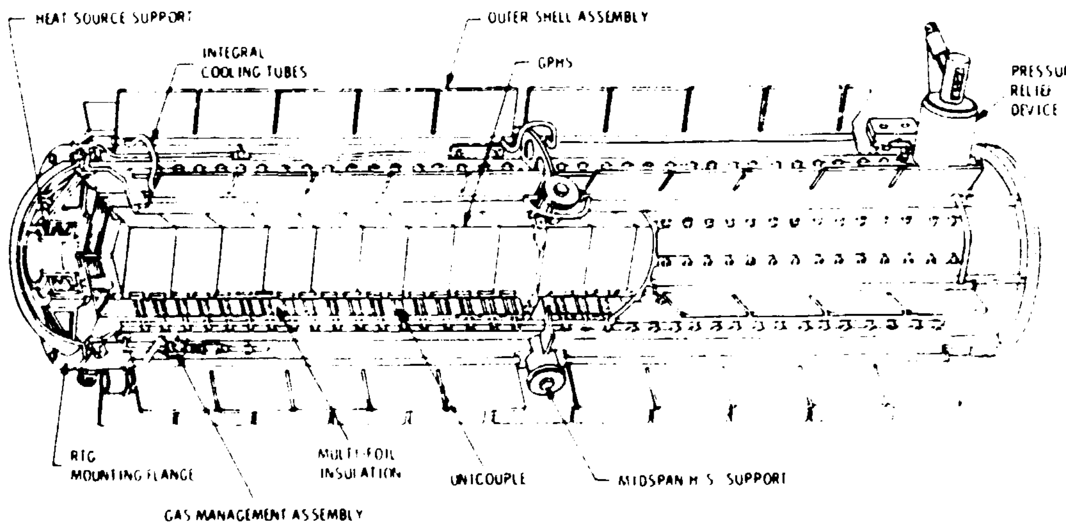

Figure I1
GPHS-RTG: Layout.

I2

Format	Description
Title	ROMASHKA
Application of the reactor	SNR, power source
Reactor type	Fast-neutron uranium carbide reactor
Power output	Very small power/0.3 - 10 kWe
Organization (name)	RRC-KI, RF
Development status	Detailed design
Description	For experimental verification of the possibility of creating very small reactors with the direct conversion of thermal power into electric power, the experimental reactor ROMASHKA (Camomile) was created in 1964 in RRC-KI (Russia). For ROMASHKA high temperature design the heat generated in the core is transferred via heat conducting materials to the thermoelectric converter with high temperature silicon-germanium elements. The reactor core consists of UC_2 fuel discs. The ROMASHKA reactor has been working successfully for 15 000 hours (~6100 kWh of electricity). The following study of the state of the ROMASHKA reactor plant elements showed that the achieved parameters and lifetime were not maximum and that they could be improved by some changes of the structure and particularly by using flat module thermionic elements between the core and radial reflector instead of thermoelectric converter. ROMASHKA reactor plant key features are the following: o Maximum compactness, low mass, high efficiency and reliability o Complete statics of operation and absence of the coolant loops o Self-correction due to the negative temperature reactivity coefficient o Possibility of preliminary testing.
Schedule	Today the ROMASHKA reactor plant is in the focus of the Thermionic Critical Technology (TCT) Program initiated by the USA Air Force for the bimodal concept with the electric output of about 25 kW in power mode, and electric power of up to 15 kW when propuslive thrust is produced in the range from 80 to 200 N.

ROMASHKA Unit Data

Converter-type		Thermoelectrical/thermionic
Power output		0.3 - 10 kWe
Fuel type		UC_2
Enrichment		90 % U-235
Reflector		Beryllium
Total temperature reactivity effect		-2.7 %
Life time		15 000 h
Core loading		49 kg
Mass		435 kg
Temperatures		
-	Core, maximum	2 173 K
-	Hot junction	1 253 K
-	Base of radiator, average	823 K
-	Differential across converter	~315 K
Neutron spectrum		Fast
Average flux		$\sim 9 \times 10^{12}$ n/cm^2/s
Converter characteristics		
-	Material	SiGe
-	Figure of merit	N/A
-	Material efficiency	N/A
-	Overall efficiency	~1.5 %
-	Working voltage	1.6 V/section

References

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A Thermionic Converters Success Story.
Proceedings 10th Symposium on Space Nuclear Power and Propulsion.
American Institute of Physics, New York, USA, 1993, p 1369.

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Conceptual design of the bimodal nuclear power system based on the "Romashka" type reactor with thermionic energy conversion system.
AIP Conference Proceedings No. 324.
12th Symposium on Space Nuclear Power and Propulsion, 1995, January 8-12, USA, NM, Albuquerque.

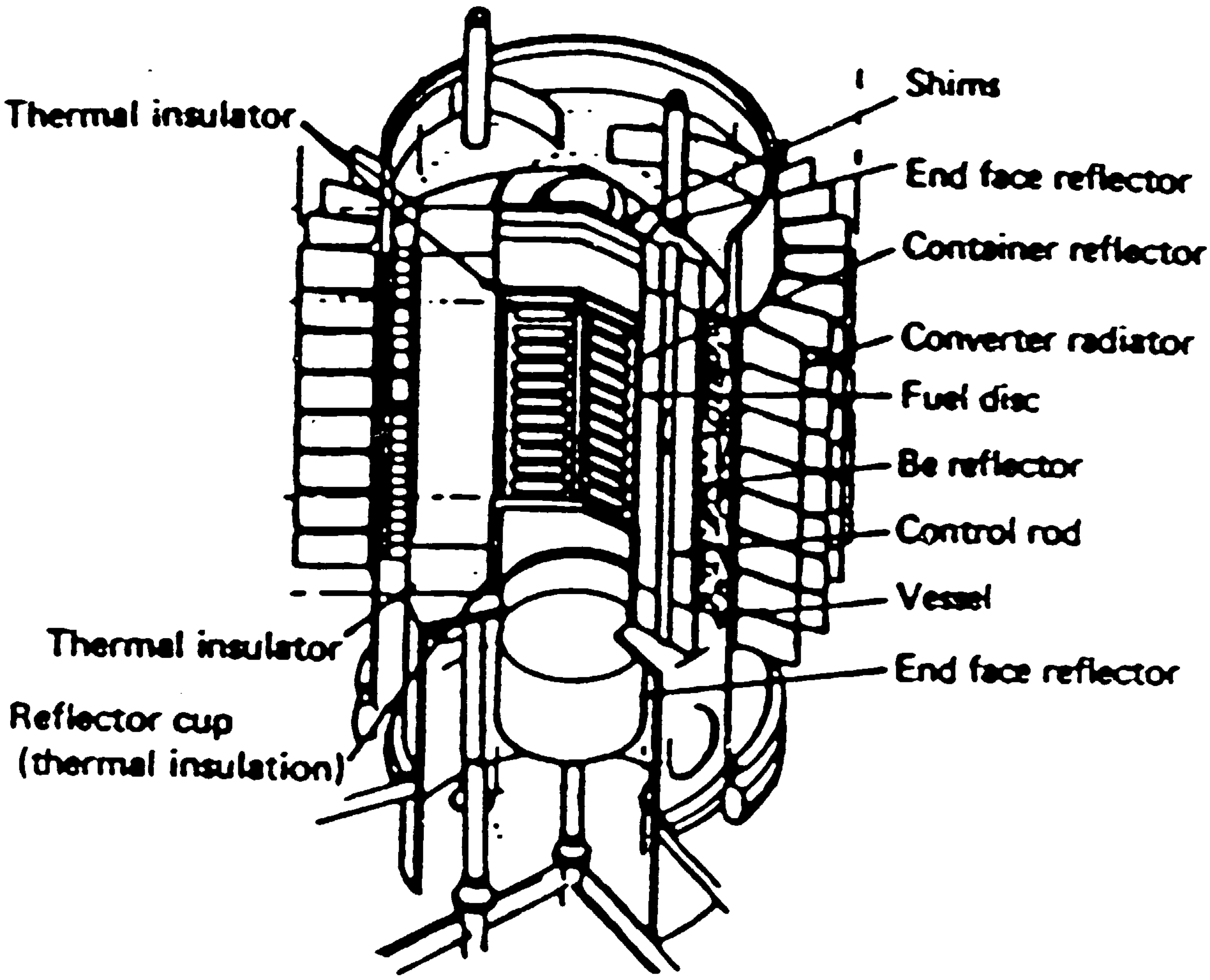

Figure I2
ROMASHKA: Reactor plant.

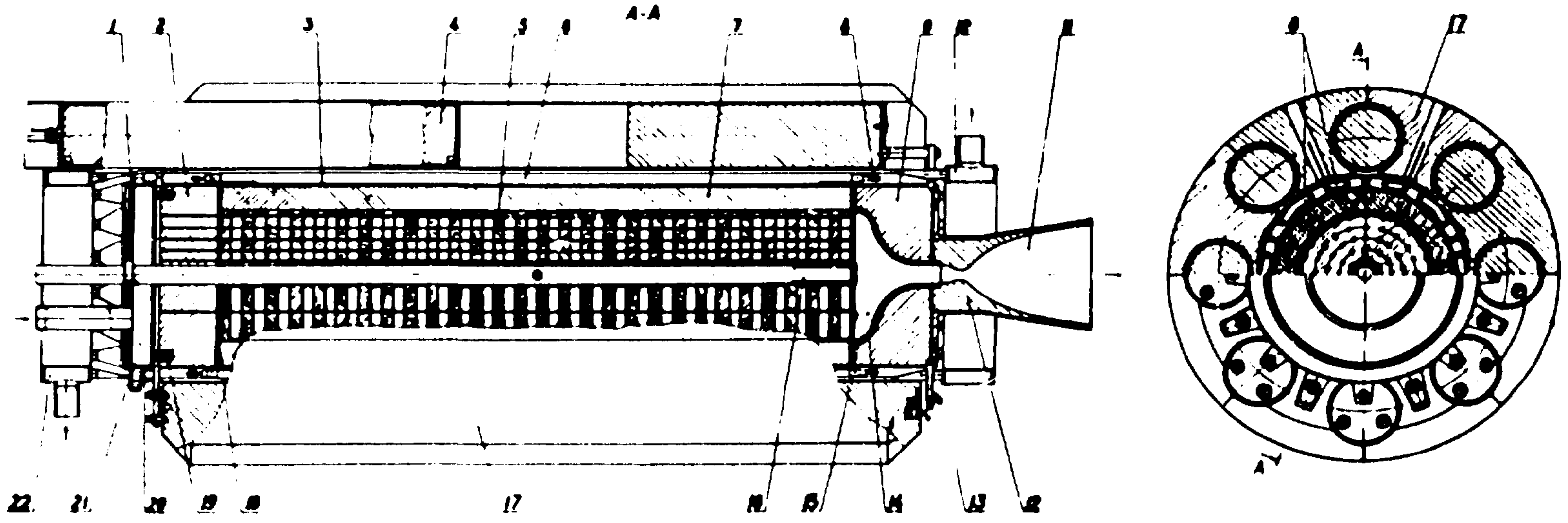

Figure 12
ROMASHKA: Type biomodal reactor.

1, 13	- Lid	7	- Bushing	15, 21	- Hydrogen header
2, 9	- End reflectors	8	- Electric bus	16	- Safety rod poison
3	- Vessel	10, 22	- Coolant headers	17	- Side reflector
4	- Rod	11	- Nozzle	18	- Disc
5	- Fuel element	12	- Insert	19	- Spring
6	- TEC	13	- Thermal insulation	20	- Ring

I3

Format	Description
Title	TOPAZ-II
Application of the reactor	SNR, power source
Reactor type	Zirconium hydride moderated reactor with in-core single-cell thermionic converter.
Power output	Very small power/5 - 40 kWe
Organization (name)	RRC-KI, RF
Development status	Commercial
Description	The TOPAZ-II reactor design with 4.5 - 6 kWe output is a small zirconium hydride moderated reactor with single-cell in-core thermionic converters developed by Russia. The major subsystems that comprise the power system are the nuclear reactor, which contains the thermionic converters, the radiation shield, the coolant system, the cesium system and the instrumentation and control (I&C) system. The TOPAZ-II core consists of 37 thermionic fuel elements, loaded with uranium dioxide with enrichment 96% U-23[illegible]. Fuel elements are placed into vertical holes within the cylindrical moderator blocks. The core is surrounded by a beryllium reflector. The TOPAZ-II reactor plant is controlled by the rotation of external beryllium drums containing segments of boron carbide. The basic components of the cooling system include: electro-magnetic pump, volume accumulator, gas absorber, bellows and radiator. The coolant is eutectic NaK. The coolant flows through the radiator to the lower collector. Leaving one, it is split into two opposite flows, which enter the shield. Then, each of the two coolant paths branches into three coolant pipes, enters the electro-magnetic pump through six coolant pipes before entering the lower plenum of the reactor. The coolant enters six inlets to the lower coolant plenum of the reactor before passing through the thermionic fuel element flow channels.

The power output and total thermal efficiency increase are realized due to the growth of the emitter temperature. In the range of 6 - 40 kWe the current design can be conserved without any essential modification because its mass and specific power characteristics lay in acceptable diapason.

The main features of the TOPAZ-II reactor are:

- o Possibility of introducing special electrical heaters instead of fuel load for testing
- o Simplicity of gas fission product release

o Possibility of refuelling at finally constructed system

Schedule

The US Ballistic Missile Defence Organization (BMDO) is investigating the possibility of launching a Russian TOPAZ-II space nuclear power system. The intended application for the TOPAZ-II reactor is the Nuclear Electric Propulsion (NEP) Space Test Mission. The program is referred to as the Nuclear Electric Propulsion Space Test Program (NEPSTP). The primary mission goal is to demonstrate and evaluate nuclear electric propulsion technology to establish a capability for future civilian and military missions. Also, in Russia, a conceptual development of a new space system, Topaz-Star, with the electric power 40 kW, has started.

TOPAZ-II Unit Data

Power output	6 kWe
Conversion type	Thermionic
TFE design	Single-cell
Coolant	NaK
Outlet temperature	600 °C
Inlet temperature	500 °C
Voltage of working section	28 - 30 V
Life time	>3 years
Dimension	
- length	3 900 mm
- max diameter	1 400 mm
Mass of reactor block	1 tonne

References

PONOMAREV-STEPNOI, N N et al
Nuclear Energy in Space.
Proceedings 6th Symposium on the Space Nuclear Power Systems, Albuquerque, NM, USA, 1989.

MARSHALL, A C
Status of the nuclear safety assessment for the NEPSTP (TOPAZ-II) space reactor program.
Proceedings ECENES '93, Makuhari, Chiba, Japan, 20-24 Sept, 1993, p 351-355.

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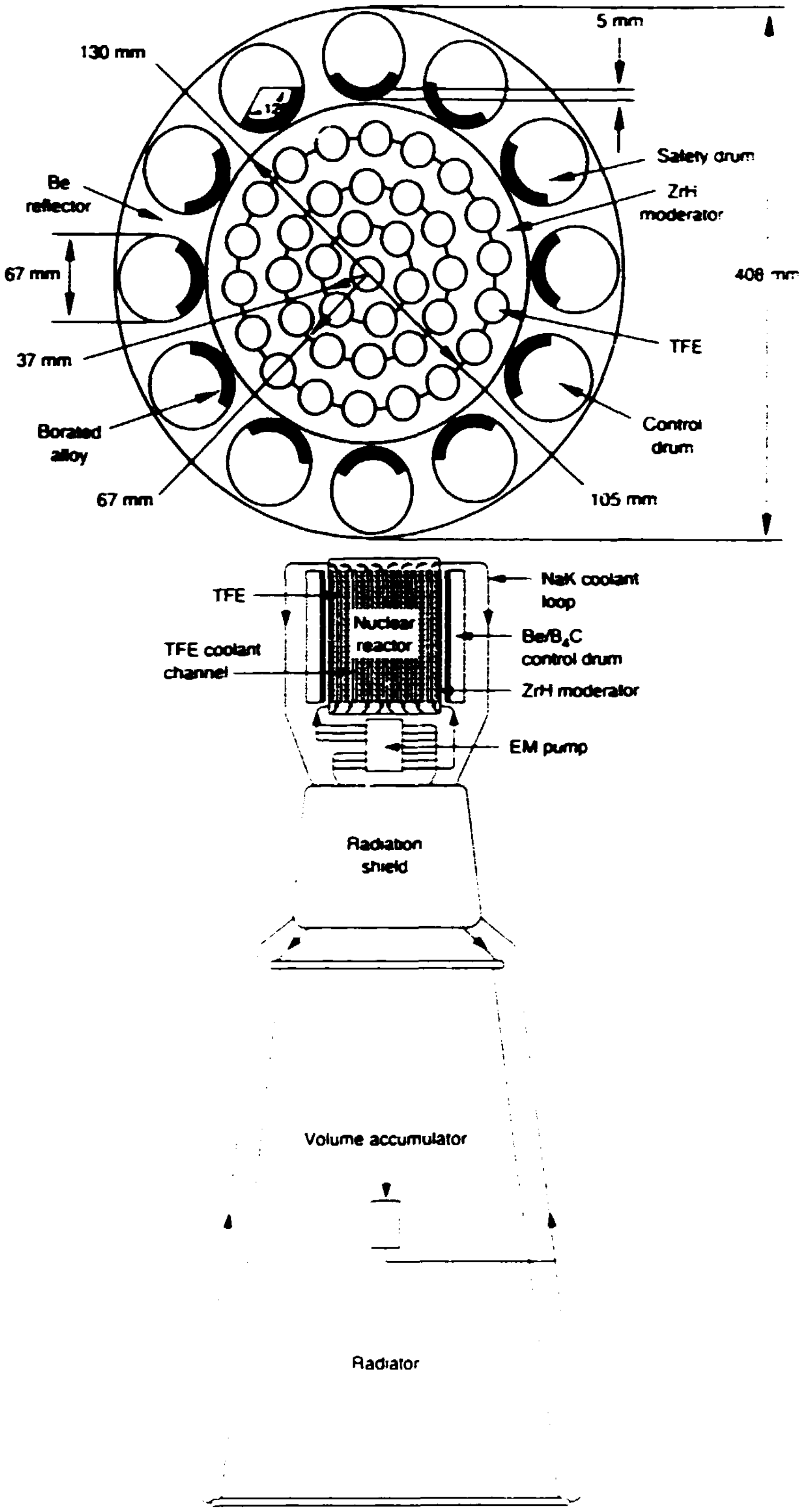

Figure I3
TOPAZ-II: Reactor layout.

I4

Format	Description
Title	STAR-C
Application of the reactor	SNR, power source
Reactor type	Graphite moderated uranium carbide reactor with out-or-core thermionic conversion process
Power output	Very small power/5 - 40kWe
Organization (name)	Rockwell International, US
Development status	Basic design
Description	A 40 kWe Space Thermionic Advanced Reactor Compact (STAR-C) design with electric output 5 - 40 kWe has been developed in response to evolving Air Force space power requirements in the US. The STAR-C reactor plant is based on out-of-core thermionic power conversion elements. The STAR-C reactor core consists of hot pressed UC_2 fuel plates held in graphite fuel trays. There are two core/converter configuration options. The solid core has converters placed adjacent to its outside surface. In the annual core configuration, the centre of solid graphite core is removed, providing an area for converters to be also placed within the interior core surface.

The out-of-core static power conversion subsystem consists of a modular array of series and parallel-connected thermionic converters.

The separation of the nuclear heat source and the conversion system allows for a very flexible, rigorous, non-nuclear systems test in which the entire power conversion subsystem can be tested for efficiency and functionality under operating, transient and hostile conditions. This reduces the cost and the risk schedule of the development program.

The moving reflectors are used for reactivity control in the STAR-C power source. Radial reflectors exist in the area between converters. The resulting neutron leakage is sufficient to produce the reactivity swing necessary for control. In addition to the moving reflectors, the STAR-C design utilizes an additional, independent shutdown system in the form of B_4C rods, located in the core.

Converter components operate in regions of low neutron fluence, which allowed to decrease swelling below the existing ROMASHKA database.

Schedule

The STAR-C reactor design is in focus of the Thermionic Critical Technology (TCT) program initiated by US Air Force.

STAR-C Unit Data

Thermal power	340 kWt
Reactor output	42.8 kWe
Net electric power	40.9 kWe
Net system effect	12.0 %
Peak fuel temperature	2 150 K
Core surface temperature	2 000 K
Emitter temperature	1 854 K
Collector temperature	1 031 K
Main radiator area	5.9 m^2
Mass	2 502 kg
Life time	10 years

References

BEGG, L L, WUCHTE, T H and ODDING, W D
STAR-C Thermionic Space Nuclear Power System.
CONF 920104, 1992, American Institute of Physics.

I5

Format	Description
Title	SP-100 (Space Power)
Application of the reactor	SNR, power source
Reactor type	Fast spectrum reactor
Power output	Very small power/105 kWe
Organization (name)	General Electric, US
Development status	Basic design

Description

The SP-100 Space Reactor Power System is being developed by General Electric under contract to the DOE to provide electric power in the range of tens to hundreds of kilowatts. The SP-100 represents an enabling technology for a wide variety of earth orbital and interplanetary science missions, nuclear electric propulsion stages, and lunar/Mars surface power for the Space Exploration Initiative.

The major elements of an SP-100 system are:

- A fast-spectrum reactor made of a niobium base alloy with a ceramic form of uranium nitride fuel
- A set of actuators and controls that position the neutron reflectors and the safety rods automatically, or by command
- A shadow shield (lithium-hydride/tungsten) that attenuates both neutrons and gammas and protects the spacecraft electronics from the radiation emitted by the reactor
- A heat transport loop that transfers heat from the reactor to a set of power conversion devices by means of a liquid metal (lithium) pumped by a self-actuated electromagnetic pump, and a heat rejection loop that transports waste heat from the power conversion devices to radiator panels
- A solid-state power conversion package (SiGe/GaP) that transforms the reactor thermal power into electricity

- The largest and most visible component of the system, a waste heat radiator that dissipates the heat to space (beryllium or carbon-carbon matrix structure with titanium/potassium heat pipes).

In the fully deployed configuration, the 100 kWe power generation unit measures about 12 meters from the tip of the reactor to the tip of the radiator.

Schedule

The SP-100 reactor development program started tenfold years ago, but the program has recentely been terminated and 20 types of advanced technologies are now under transfer to the industry.

SP-100 Unit Data

Thermal power	2.3 MWt
Gross power generated	105.3 kWe
Peak reactor outlet temperature EOL	1 375 K
HT loop ΔT	92 K
HT loop mass flow	5.9 kg/s
Peak radiator inlet temperature	840 K
HR loop ΔT	48 K
HR loop mass flow	10.4 kg/s
Average radiator surface temperature	790 K
Radiator block body area	94 m^2
Radiator physical area	104 m^2
PC thermopile area	5.5 m^2
T/E leg length	0.55 cm

References

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SP-100 space nuclear Power Technology Requirements for kW- to MW systems.
Transaction American Nuclear Society, 1992, Space nuclear power, p 21-22.
AIP Conference Proceedings No. 324, 1995, January 8-12, USA, NM, Albuquerque.

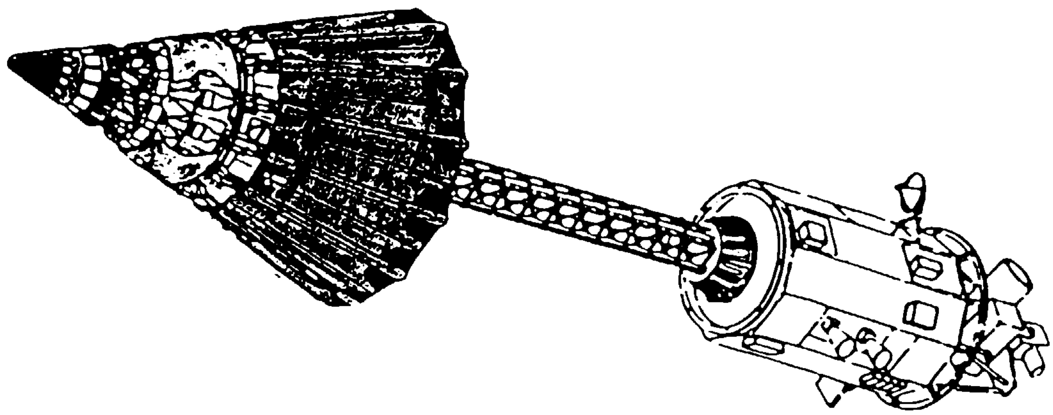

Figure I5
SP-100: Layout.

I6

Format	Description
Title	ERATO
Application of the reactor	SNR, power source
Reactor type	LMR/HTGR
Power output	Very small power/20 kWe
Organization (name)	CNES/CEA, France
Development status	Studies

Description

Since 1982, France has been investigating the possible development of 20 to 200 kWe space nuclear power systems adapted to the Ariane V launcher for two reference missions: space-based radar and electric propulsion.

An ERATO space nuclear power system concept consists of:

- o A nuclear reactor cooled by gas or liquid metal which is the heat source of the reactor
- o An energy conversion system to produce electric power from nuclear heat carried by the coolant
- o A radiator which is the heat sink of the system and radiates the nuclear heat not transformed into electricity in space
- o A shield to protect payload and sensitive equipment from neutron and gamma photon irradiation

For the 20 kWe ERATO plant the following concepts were considered:

- o A gas-cooled epithermal reactor with a UC_2 fuel particle bed and superalloys (HRA) as structural materials (Hastelloy X for instance) with a core exit gas temperature of 820 ^oC
- o A sodium or sodium potassium (NaK) cooled UO_2 fuel fast neutron reactor with 316 stainless steel material operating at a maximum temperature of 700 ^oC

The 20 kWe ERATO reactor plant considered generator employs a Brayton cycle energy conversion system with He-Xe working fluid. The gas is heated through a liquid metal-gas heat-exchanger in liquid metal-cooled reactor systems and directly in the reactor core in the gas-cooled reactor system.

One or two Brayton turboelectric converters (gas turbine, compressor, alternator) are intended in the 20 kWe unit to permit a back-up operation at quasi nominal power in the event of partial unavailability conversion unit and to improve the global system reliability.

Schedule

Considering CNES and CEA experience in the space and nuclear fields, it would be possible to develop in France a space nuclear power system if it was decided. But up to now no mission requiring such electric power has been envisaged in France, or in Europe. Nevertheless the development of high electric power system which represents a new milestone in the conquest and utilization of space constitutes a very long process.

ERATO Unit Data

Reactor		
- Thermal power	110 kWt	125 kWt
- Uranium mass	70 kg	137 kg
- Fuel	UO_2	UC_2
- Structural material	S steel	Superalloy
- Primary coolant	NaK	He Xe
- Pressure	2.5 bar	80.0 bar
- Core inlet/outlet temperature	577/682 °C	511/820 °C
- Control drums (Be + B_4C)	7	19
- Safety rods	(B_4C)	(B_4C)
Shield	B_4C+LiH	B_4C+LiH
Intermediate heat exchanger		
- Gas inlet/outlet temperature	430/670 °C	430/670 °C
Brayton turboelectric converter		
- Turbine inlet temperature	670 °C	819 °C
- Turbine inlet pressure	9.0 bar	78.0 bar
- Compressor inlet temperature	59 °C	119 °C
- Compressor pressure ratio	2.06	2.16
Radiator		
- Power	80 kWt	104 kWt
- Inlet/outlet temperature	236/53 °C	368/119 °C
- Area	86 m^2	54 m^2
Global system efficiency	0.19	0.16

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Space nuclear power system studies in France and envisaged missions.
Proceedings Nuclear Power Engineering in space, Obninsk, RF, 15-19 May, 1990.

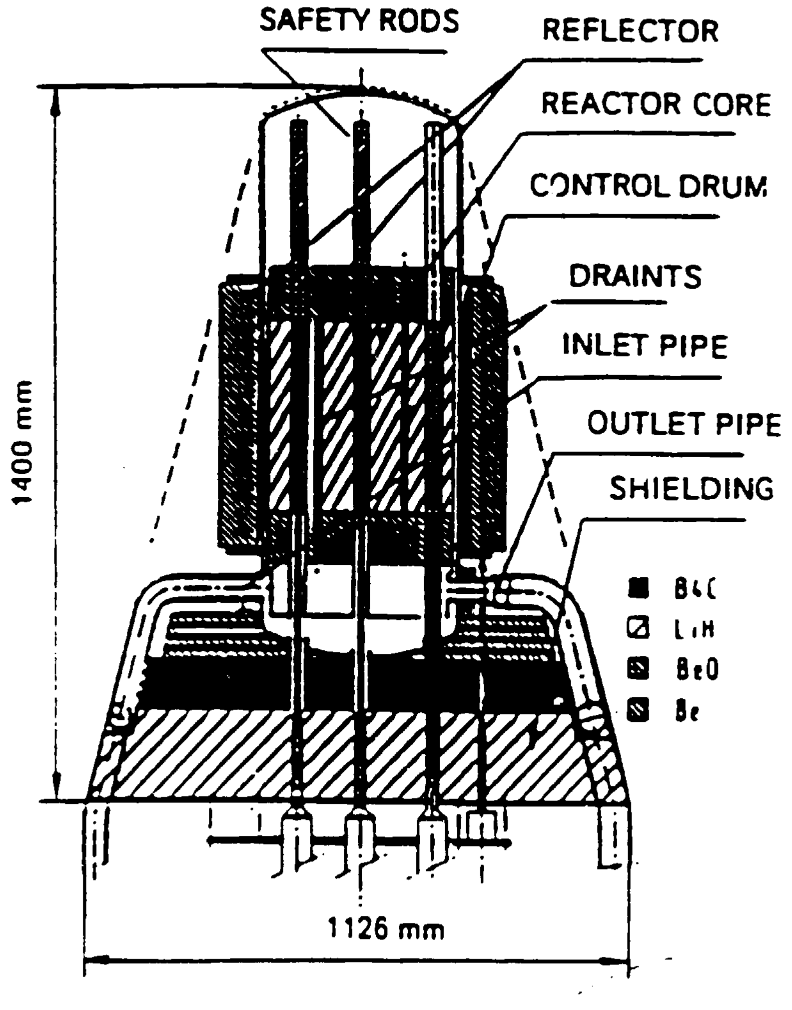

Figure I6
ERATO: Reactor block (UC_2/HeXe/HRA 850 °C).

I7

Format	Description
Title	LMCPBR
Application of the reactor	SNR, power source, moon-base
Reactor type	LMR/HTR
Power output	Small power/10 MWt
Organization (name)	Toshiba/JAERI, Japan
Development status	Studies
Description	A concept of a power generating system with a liquid metal cooled pebble-bed reactor with a thermal output of 10 MWt has been studied by the Toshiba & JAERI team for use in a moon base. This reactor concept is based on the HTGR fuel technology and the liquid metal cooling technology. The designed system consists of 71 lithium cooled pebble beds, a fast spectrum, drum-controlled reactor, a liquid metal steam generator, a turbogenerator, and a panel heat rejection system. The secondary coolant is potassium which is vaporized in a steam generator and rotates a turbine-generator. Both coolants are circulated by electro-magnetic pumps. The LMCPBR plant used the potassium Rankine cycle. A reactor vessel contains an integrated potassium boiler for reducing the total mass as much as possible. As a reactor structure material, Nb-Zr alloy is used considering the material strength at high temperatures. The total system mass is about 17 tonnes.
Schedule	No LMCPBR plant has been ordered yet. A conceptual design study is under consideration.

LMCPBR Unit Data

Thermal power		10 MWt
Reactor core		
-	Diameter	1.1 m
-	Height	1.1 m
-	Uranium mass	503 kg
-	C/U ratio	35.8
Structural material		Nb-Zr alloy
Reflector		
-	Axial	
	- Material	BeO pebble (3 cm diam)
	- Thickness	25 cm
-	Lateral	
	- Material	Be
	- Thickness	28 cm
Control drum		
-	Diameter	24 cm
-	Number of drums	18
-	Position	In lateral reflector
-	Material	B_4C (80 % ^{10}B) and Be
Primary coolant		
-	Material	Lithium-7
-	Inlet temperature	1 123 K
-	Outlet temperature	1 203 K
-	Flow rate	3.9 m^3/min

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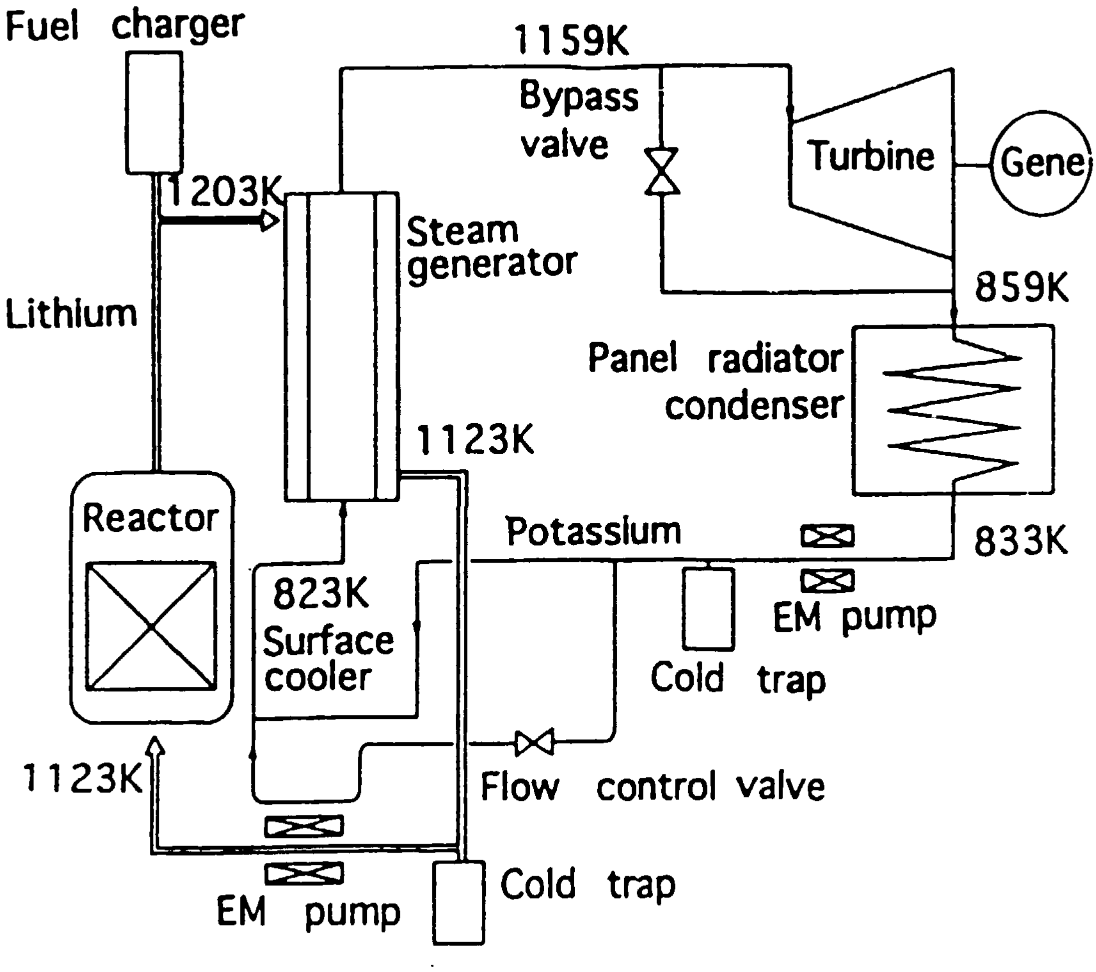

Figure I7
LMCPBR: Line diagram.

I8

Format	Description
Title	Rover/NERVA
Application of the reactor	SNR, propulsion
Reactor type	Prismatic solid core
Power output	Medium power/367 - 1 566 MWt
Organization (name)	Westinghouse/LANL, US
Development status	Basic design
Description	The basic Rover/NERVA reactor design consists of an axial-flow core 1.32 m long by 0.89 m in diameter supported in compression by regeneratively-cooled Inconel tie-tubes passing through unloaded (no uranium) support elements. The fuel elements are 19 hexagons 1.91 cm across flats. Fuel/ support element flow passages are nominally 0.254 cm in diameter. A radial beryllium reflector, 11.43 cm thick, houses 12 rotatable control drums that provide neutronic control. Fuel loading is varied radially to partially flatten power, and flow orifices at the inlet of each coolant passage match the flow to the local power. An aluminium inlet-end support plate, a shadow shield, and an aluminium pressure vessel are other major reactor components. Core diameter ranged between 53.3 cm (PEWEE 1) and 140 cm (PHOEBUS 2A). The flow passage diameter and fuel element size may, in principle, be changed. However, the element dimensions are governed by a complex set of performance and fabrication factors and the present dimensions are considered to be near-optimum. More than 50 000 fuel elements have been fabricated and tested during the 17 years of the Rover/NERVA program. The main efforts have been devoted to fuel fabrication particles and to insertion in graphite matrix.
Schedule	In 1973 the Rover/NERVA program was terminated. The future development of nuclear propulsion systems can be based on the technology and experience of the past efforts.

Rover/NERVA Unit Data

Type	XE	NERVA	Small engine
Thrust	245 kN	337 kN	72 kN
Specific impulse	6 970 m/s	8 085 m/s	8 575 m/s
Thermal power	1 141 MWt	1 566 MWt	367 MWt
Turbo pump power	5.1 MWt		0.9 MWt
Turbo pump speed	22 270 rpm	23 920 rpm	46 950 rpm
Pump discharge pressure	6.80 MPa	9.36 MPa	6.03 MPa
Engine flow rate	35.9 kg/s	41.9 kg/s	8.5 kg/s
Chamber temperature	2 270 K	2 360 K	2 695 K
Chamber pressure	3.86 MPa	3.10 MPa	3.10 MPa

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Nuclear rocketry review.
Fusion Technology, v 20, Dec, 1991, p 698-670.

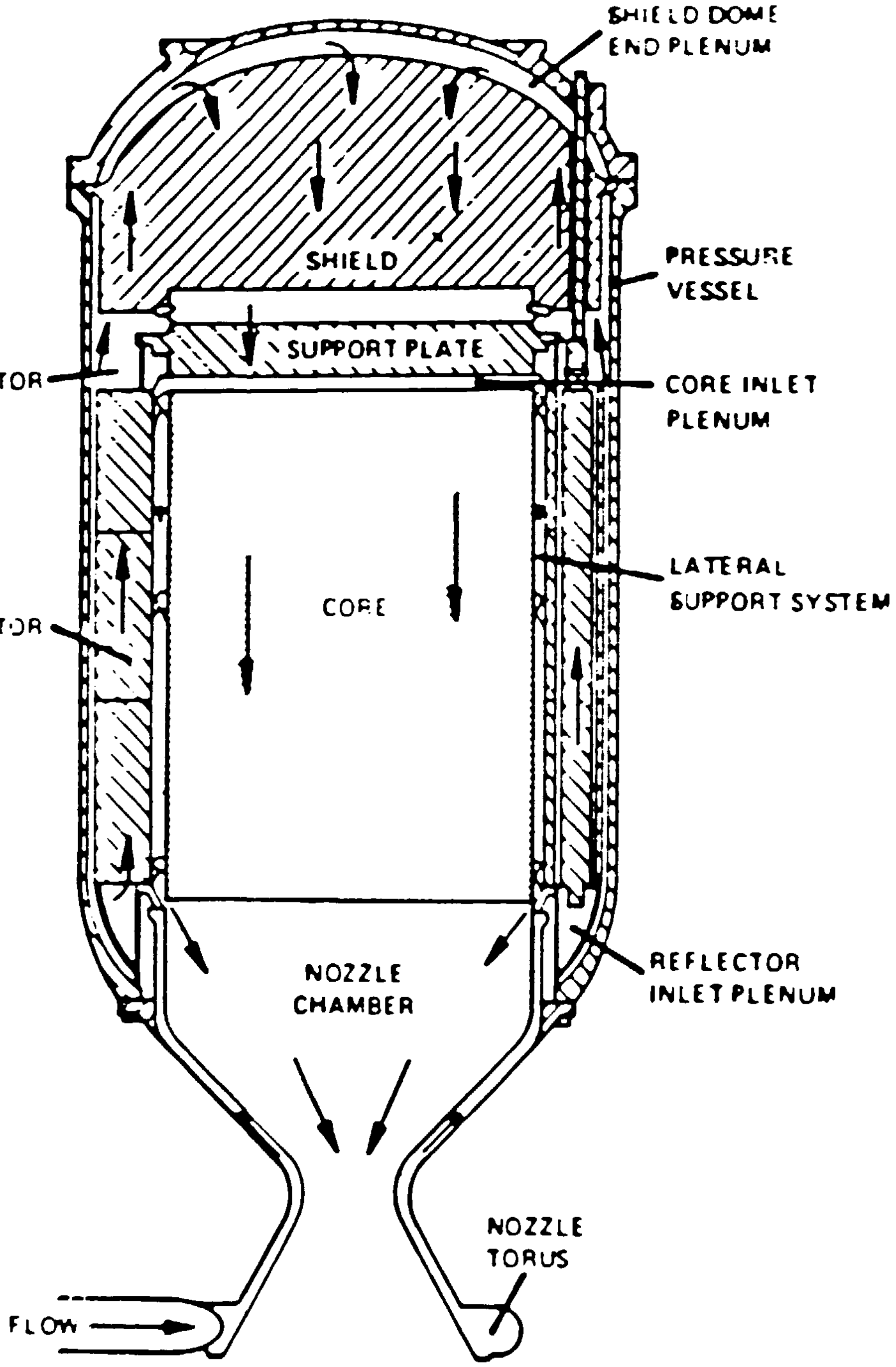

Fig. 5

Figure I8
Rover/NERVA: Reactor layout.

I9

Format	Description
Title	NPPS
Application of the reactor	SNR, power source/propulsion
Reactor type	Epithermal solid core reactor
Power output	Medium power/1 200 MWt
Organization (name)	RRC-KI, RF
Development status	Studies
Description	A Nuclear Power Propulsion System (NPPS) concept with the thermal output of 1 200 MWt has been developed by RRC-KI for a manned Martian mission. The advisability of using a nuclear power source in a space engine system is determined by the possibilities of the nuclear rocket engine realizing the specific impulse of 9 000 m/s in the propulsion mode using hydrogen as the working fluid, and also with a good thermal power capability and mass characteristics for the nuclear power engine system with a closed cycle in the power mode. One of the possible variations of the nuclear power engine system being considered is based on the concept of the nuclear rocket engine and the turbine electric engine converters. In the engine mode, hydrogen is used as a coolant and is heated in the reactor to a temperature of 2 600 - 2 900 K. In the power engine mode (Brayton cycle) the working fluid is a mixture of helium and xenon. Research results have shown that the most acceptable system is an intermediate (epithermal), solid-core reactor of uranium carbide, niobium and zirconium, with a reflector made of beryllium.
Schedule	This concept was based on basic theoretical research conducted in the former Soviet Union and abroad, and also on neutron research on critical assemblies, power tests on reactor facilities, and on prototype reactor installations. No current development.

NPPS Unit Data

Power propulsion scheme		3 - 4 modules
Module reactor configuration		Channel housing
Nuclear fuel		Solid solution of uranium, niobium, zirconium carbides
Propulsion mode		
-	Propellant	Hydrogen
-	Thermal power	not over 1 200 MWt
-	Specific impulses	$(8 - 9) \cdot 10^3$ m/s
-	Coolant flow rate	22 - 25 kg/s
-	Coolant outlet temperature	2 600 - 2 900 K
Power mode		
-	Coolant	Xenon and helium mixture (1 - 3 % He)
-	Energy conversion system	Turbomachine (Brayton cycle)
-	Electrical power	50 - 200 kWe
-	Coolant maximum temperature	1 200 K
Radiation environment		
-	Crew exposure to NPPS radiation	Not over 10 rad
-	Estimated doses in the instrumentation compartment:	
	- γ-radiation	Not over 10^6 rad
	- Fast ($\leq$0.1 MeV) neutrons influence	Not over 10^{13} n/cm^2
-	Radiation energy release in hydrogen tanks during propulsion mode	Not over 10^{-4} W/cm^3
NPPS total mass		50 - 70 Mg

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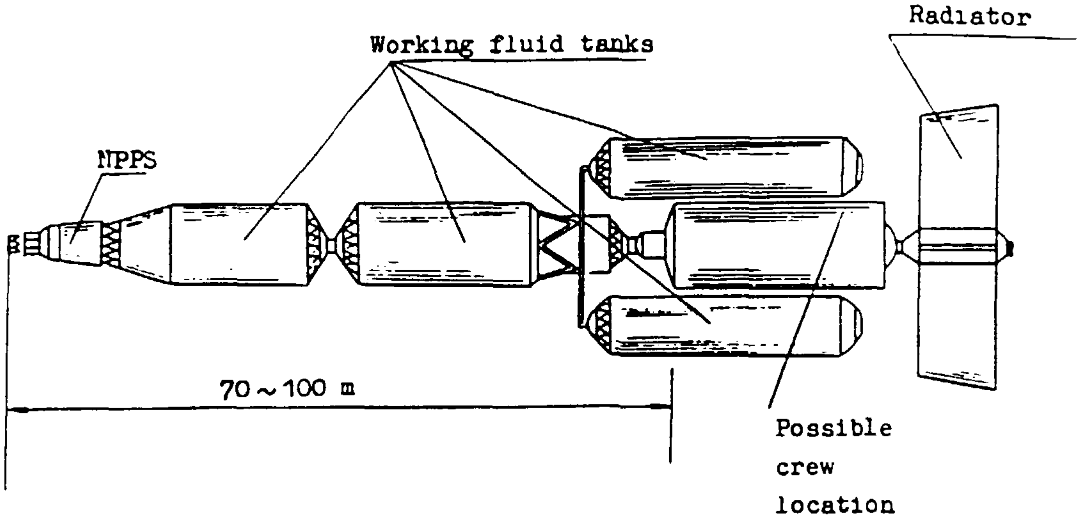

Figure I9
NPPS: Schematic.

J Subcritical Accelerator-Driven Waste Transmutation and Energy Generation Systems

J0 Introduction

Accelerator-driven waste transmutation and energy generation have gained renewed interest through studies on different long-term waste management options. The possibility of a future global power production system that is free from the discharge of long-lived radioactive products (TRU and fission products) is under consideration in France, the USA, Japan, Russia, Sweden and Switzerland.

The initial steps deal with the analysis of neutron economy to evaluate and compare the transmutation potential of different nuclear power systems as well as losses in the reprocessing, partitioning and fuel fabrication. The potential decrease of radiotoxicity is also an important subject for these studies.

It has been suggested that accelerators could be a part of advanced systems in order to produce energy and additional neutrons to transmutate waste nuclides or breed fertile material to fissile nuclides (Th-232 to U-233 or U-238 to Pu-239).

The blanket elements associated with an accelerator are similar to a fission reactor core concerning the neutron fluxes, heat generation and the radio-active content. Such a system has the advantage of operating in a subcritical mode. Nevertheless problems with loss of coolant are the same because the decay heat is similar and related to the power production.

From the economical point of view it has the disadvantage of the additional investment of the accelerator and that a significant fraction (10 - 30 %) of the produced electricity will be required to operate it.

Such a hybrid system with a high transmutation rate will be able to support the burning of TRU and long-lived fission products in the final stage of nuclear fission power assuming that the toxicity of spallation products is negligible.

Substantial technical development will be necessary to achieve an acceptable solution for the containment, the interface between the accelerator vacuum system and the target/blanket as well as accelerator parameters.

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1994, OECD, Report NEA/NSC/DOC(94)11.

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1994, USA, Report GAO/RCED-94-16.

J1

Format	Description
Title	Energy Amplifier
Application of the reactor	NPP
Reactor type	SADS/PWR (HTGR)
Power output	Small power/200 MWt
Organization (name)	CERN, Switzerland
Development status	Studies
Description	The concept of energy amplification to extract nuclear energy with the help of an accelerator-induced nuclear cascade with the 200 MWt thermal output is presented by CERN. The beam of high energy particles is directed into a fuel material target. Thorium is used as breeding fuel. The neutrons thereby produced are slowed down by the moderator medium and multiplied under subcritical conditions by the breeding and fission processes. The Energy Amplifier design consists of two main separate parts: medium current (1 - 10 mA), medium energy (1 GeV), proton accelerator feeding a subcritical assembly consisting of Th and moderated media (H_2O or graphite). Under conditions of moderate neutron flux ($10^{14}\ n{\cdot}cm^{-2}{\cdot}s^{-1}$), the concentration of fissile U-233 which is bred from Th is stable at about 1.3 %. The U-233 produces energy fission and is continuously regenerated in situ without resorting to any chemical separation. It is shown that the energy produced is several times larger than that required to power the proton accelerator. This system will result in very small quantities of Plutonium and higher actinide waste. Based on the abundant and inexpensive resource which is natural Thorium, this system can be built using present day reactor technology according to the proponents.
Schedule	No Energy Amplifier reactor design has been ordered yet. A conceptual study is in progress.

Energy Amplifier Unit Data

Accelerator type	LINAC	Cyclotron
Gain	40	40
Beam kinetic energy	0.8 GeV	0.8 GeV
Average current	6.25 mA	6.25 mA
Beam input	5.0 MW	5.0 MW
Energy amplifier type	PWR	HTGR
Fuel cycle	Th-^{233}U	Th-^{233}U
Fuel type	ThO_2-pins	ThO_2/ThC pebble with Pb/Pb-Bi target
Neutron flux	10^{14} n·cm^{-2}·s^{-1}	10^{14}n·cm^{-2}·s^{-1}
Capacity	200 MWt/60 MWe	200 MWt/80 MWe
Coolant temperature, - inlet/outlet	291 °C/322 °C	435 °C/710 °C
Coolant pressure	154 bar	58 bar

References

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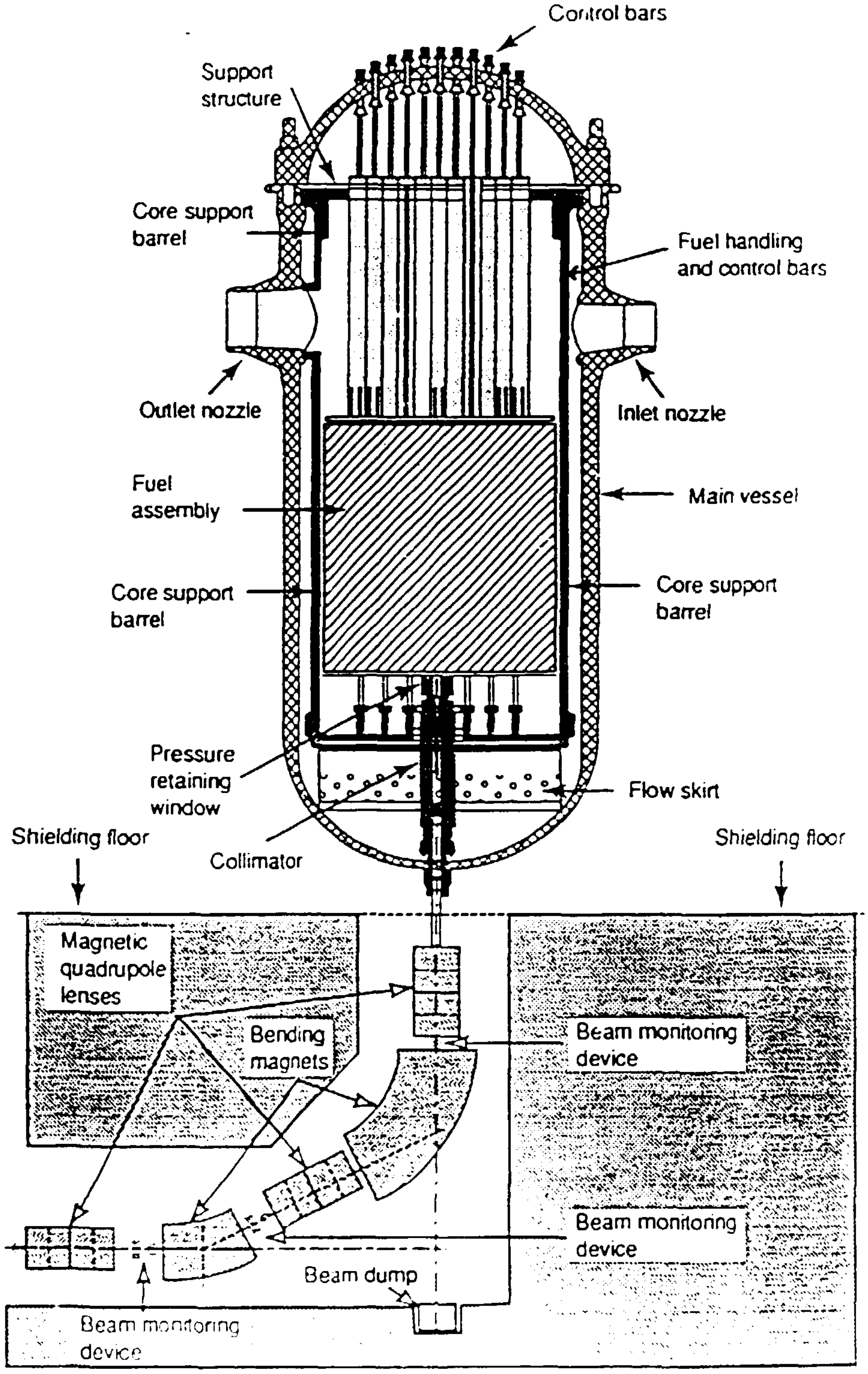

Figure J1
Energy Amplifier: Schematic (PWR-blanket version).

J2

Format	Description
Title	PHOENIX
Application of the reactor	TWS
Reactor type	SADS/LMR
Power output	Large power/3 600 MWt
Organization (name)	BNL, US
Development status	Studies
Description	Three accelerator-driven concepts operated in a subcritical mode are studied in BNL. BNL has focused the concepts just on the minor actinides and long-lived fission product incineration. The PHOENIX concept assumes a large Linac that can produce a 104-mA beam of 1.6 GeV protons. A multiple module concept was developed for the PHOENIX subcritical lattice. Each module resembles the core of the Fast Flux Test Facility (FFTF) with the minor actinides formed into oxide fuel rods, replacing the uranium and plutonium in the FFTF fuel. The fuel rods are cooled using liquid sodium and are bundled into 217 pin assemblies, with 124 such assemblies making up a 450 MWt target module. One to eight of these target modules are aligned in front of the proton beam, depending in part on how much of the "fuel" is available at any given time. A current of 104 mA is sufficient to drive with k = 0.9 subcritical lattice at 3 600 MWt. The proposed machine, based on the described PHOENIX concept, would transmutate the minor actinides and the iodine produced by 75 LWRs and would generate usable electricity of 850 MWe. The second BNL concpet with multi-segmented cyclotron injects 2.0 - 5 mA beam of 1.5 GeV protons into the lead target located at the center of the MOX-fuelled fast reactor at a slightly subcritical mode. The third BNL concept uses a multi-segmented cyclotron to inject 4 - 8 mA beam of 1.5 GeV protons into the lead spallation target located at the center of the fast spectrum core, which is loaded with nitride coated particle fuel.
Schedule	The conceptual design study is not completed and the R&D program is under way.

MOX Fuel Core System Unit Data

Accelerator	
- Type	Multi-segmented cyclotron
- Particle	Proton
- Energy	1 500 MeV
- Current	2.0 - 5.0 mA
Target	
- Equivalent diamater	10 cm
- Equivalent height	75 cm
- Target material	Pb
- Cooling material	He
Subcritical core	
- Equivalent diameter	180 cm
- Equivalent height	93 cm
- Material composition Fuel/Clad+Structure/Coolant/Moderator	35/34/41/0 v/o
- Chemical form of fuel	Oxide
- Materials of coolant and moderator	Na / -
- Averaged fresh fuel composition U/Pu/Np, Am,Cm/LLFP	73/22/5/0 w/o
- Isotopic composition of Pu $^{238}Pu/^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu$	0/58/24/14/14 w/o
- Isotopic composition of MA $^{237}Np/^{241}Am/^{242}Am/^{243}Am/^{243}Cm/^{244}Cm$	53.6/23.1/0/17.4/0/5.9 w/o
- Averaged composition of LLFP $^{99}Tc/^{129}I$	94/6 w/o
System characteristics	
- Effective multiplication factor: k_{eff}	0.98 - 0.99
- Thermal power in core	700 MWt (280 MWe)
- Power density: average	930 W/cm^3

References

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Accelerator-driven subcritical target concept for transmutation of nuclear wastes.
Nuclear technology, 1993, Vol 101, January, pp 1-17.

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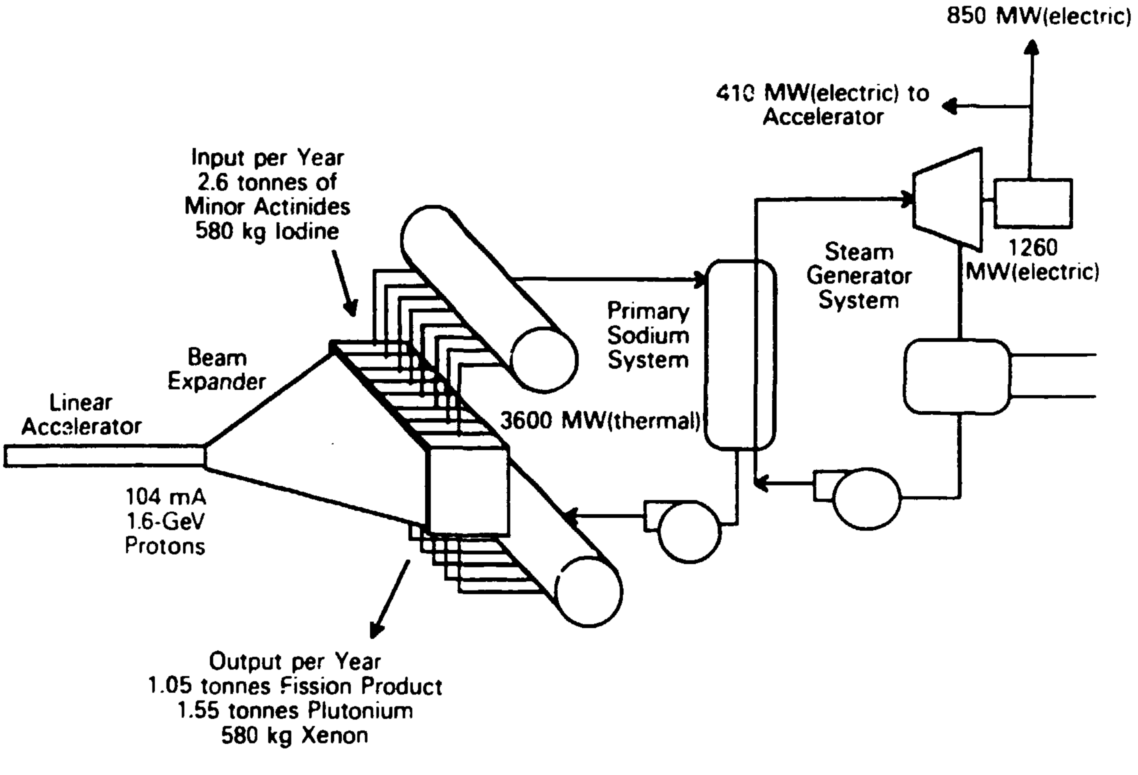

Figure J2
PHOENIX: Schematic.

J3

Format	Description
Title	JAERI-TPC
Application of the reactor	TWS
Reactor type	SADS/MSR (LMR)
Power output	Medium power/820 MWt
Organization (name)	JAERI, Japan
Development status	Studies
Description	In Japan the high-level radioactive waste incineration problem is a part of the OMEGA project (Options for Making Extra Gains from Actinides and Fission Products) proposed by The Science and Technology Agency (STA) in 1988. Three types of accelerator-driven transmutation systems are under development at JAERI. In the JAERI-TPC concept accelerator injects proton beam of 1.5 GeV, 40 mA into the tungsten target located at the center of the reactor core which is loaded with minor actinides and other long-lived nuclides. This system is capable of increasing about 260 kg per year of minor actinides. The core design of the JAERI-TPC concept is based on solid-state pin-bundle type fuel elements with sodium cooling. It offers some advantages like high power density, and high temperature operation under normal pressure. A study is also in progress on the system with 1.5 GeV, 25 mA proton beam accelerator and hard spectrum chloride molten salt assembly which is intended for transmuting not only minor actinides, but also fission products. This concept would be a continuous processing system. In the third concept accelerator injects 1.5 GeV proton beam of 20 mA into eutectic alloy (Np-Pa-Ce-Co-Tc) target core with graphite blanket, which is cooled by molten salt fluorides.
Schedule	The conceptual design phase is not completed.

JAERI-TPC Unit Data

Proton beam current			39 mA
Actinide inventory			3 610 kg
k_{eff}			0.89
Number of neutrons per incident proton			40 n/p
Number. of fissions			
-	(>15 MeV)		0.45 f/p
-	(<15 MeV)		100 f/p
Neutron flux			4×10^{15} n/cm^2/s
Mean neutron energy			690 keV
Burnup rate			250 kg/year
Thermal output			
-	Fuel		800 MWt
-	Tungsten		20 MWt
	Total		820 MWt
Power density			
-	Maximum		930 MW/m^3
-	Average		400 MW/m^3
Linear power ratio			
-	Maximum		61 kW/m
-	Average		27 kW/m
Coolant temperature			
-	Outlet	maximum	437 °C
		average	430 °C
Maximum temperature			
-	Fuel	center	890 °C
		surface	548 °C
-	Clad	inside	528 °C
		outside	484 °C

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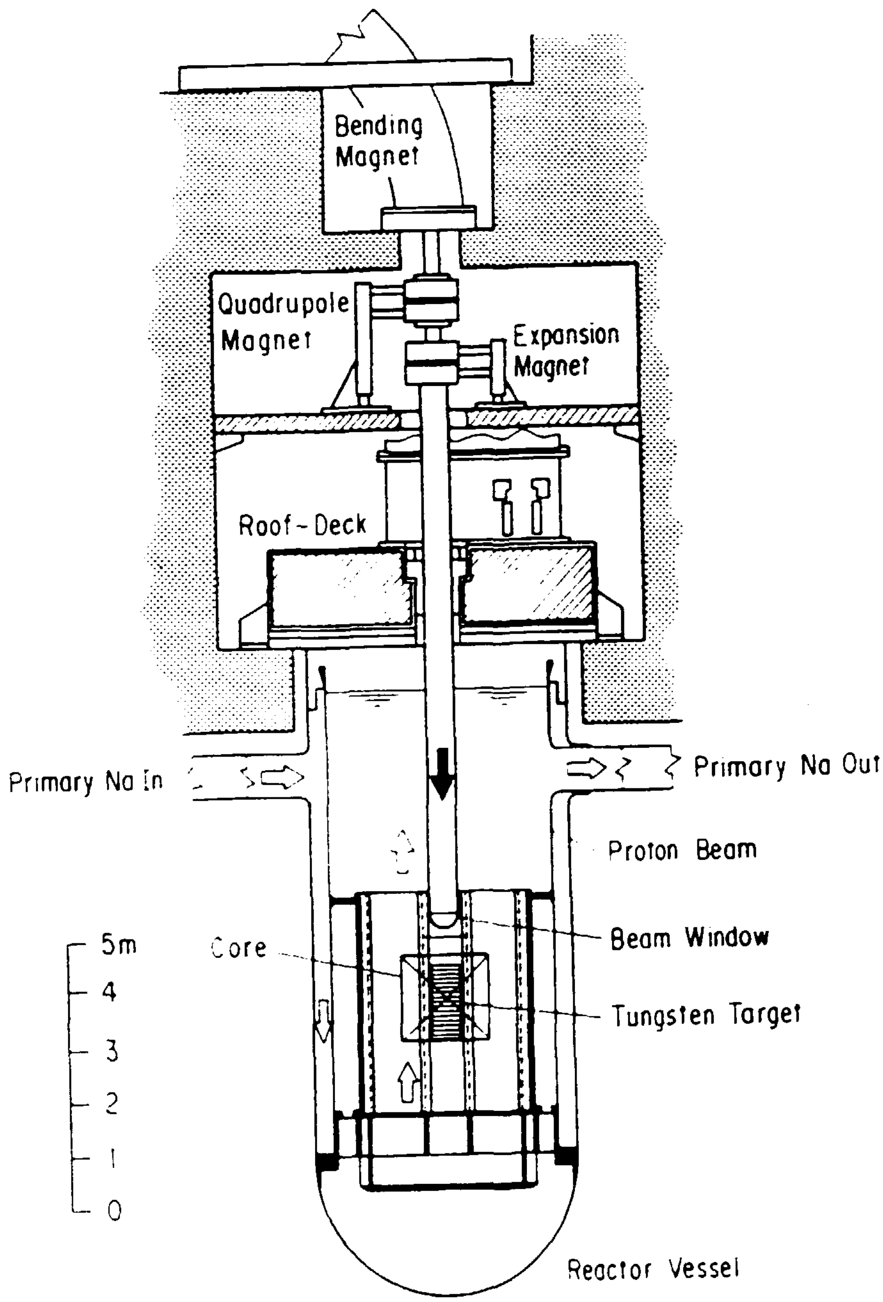

Figure J3
JAERI-TCP: Schematic.

J4

Format	**Description**
Title	ATW/ABC
Application of the reactor	TWS/MSR (LMR)
Reactor type	SADS
Power output	Large power/3 000 MWt
Organization (name)	LANL, US
Development status	Studies

Description

In recent years, accelerator-driven production of tritium (APT), weapon plutonium utilization (ABC), energy production (ADEP) and transmutation of waste (ATW) systems stimulated interest in LANL as a technologically possible method.

The most promising design option for an advanced ATW/ABC/ADEP concept uses 0.8 GeV, 90 mA proton beam LINAC; a fluid lead/or lithium target and a multiplying blanket which is graphite-moderated and molten salt-cooled. The systems operate at a value for k_{eff} of between 0.9 and 0.95. The ADEP system burns commercial waste at this rate and also generates 2 250 MWt in the Th-233U cycle while also burning the waste origniated from thorium. On-line separation would be used for fuel processing. The lithium target offers efficient neutron production (estimated at 28 neutrons per 800 MeV deuteron) and low absorption of blanket neutrons because of its small size and presence of the fission product transmutation region surrounding it. This target produces a very small amount of heavy spallation products.

The high intense thermal flux 10^{16} n/(cm^2·s) in the blanket of the proton accelerator allows for the effective transmutation of main waste actinide Np-237 through conversion into fissile Np-238, because in a thermal flux of a typical power reactor Np-237 is converted into neutron poison in low intense neutron irradiation (~10^{14} n/(cm^2·s))

The principle features of the ATW/ABC/ADEP concept are the following:

o Subcritical operation

o Low inventory of actinides and fission products: high thermal neutron flux with no dilution in external heat exchangers. Nonradioactive coolant. No spallation products in target. Low pressure

o Negative temperature and void reactivity coefficient

o Passively safe in the event of a loss of coolant accident. Large thermal inertia and good heat conductivity. Molten salt can be drained away without external intervention

o Proliferation-resistant fuel cycle. Uranium enrichment lower than 20 %. Thorium breeding without reprocessing

Schedule The conceptual study is not completed.

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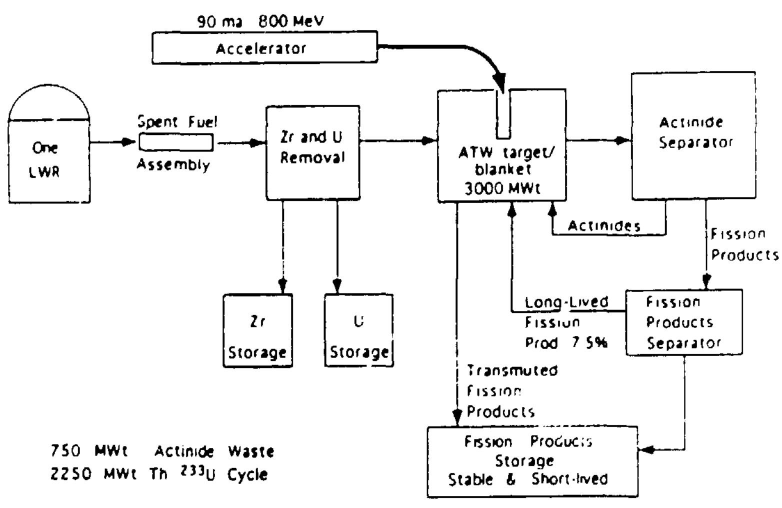
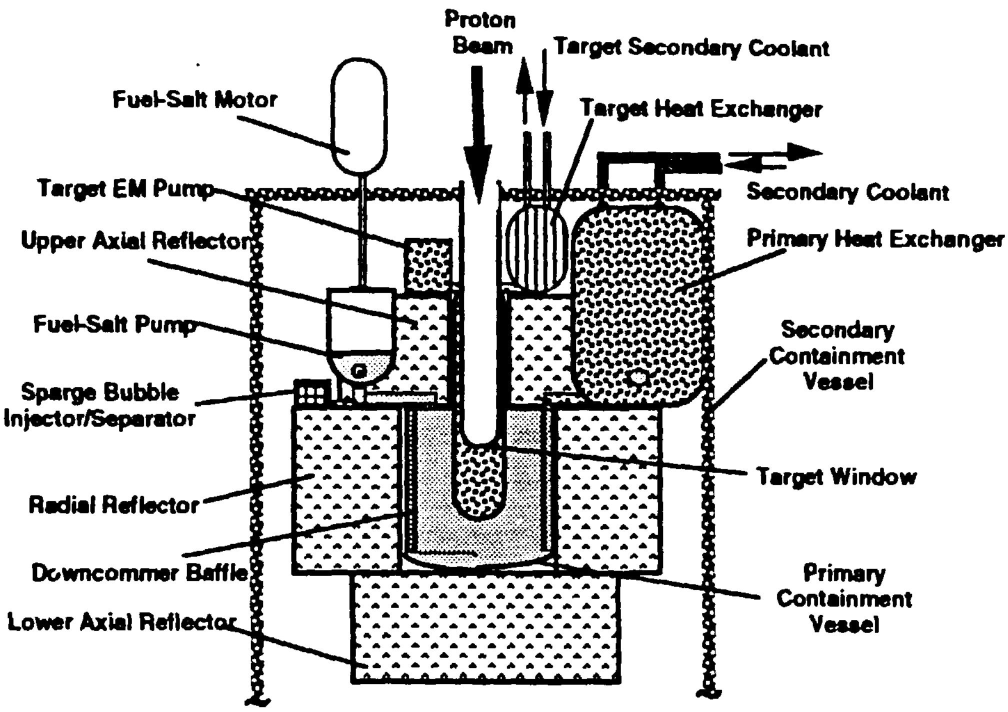
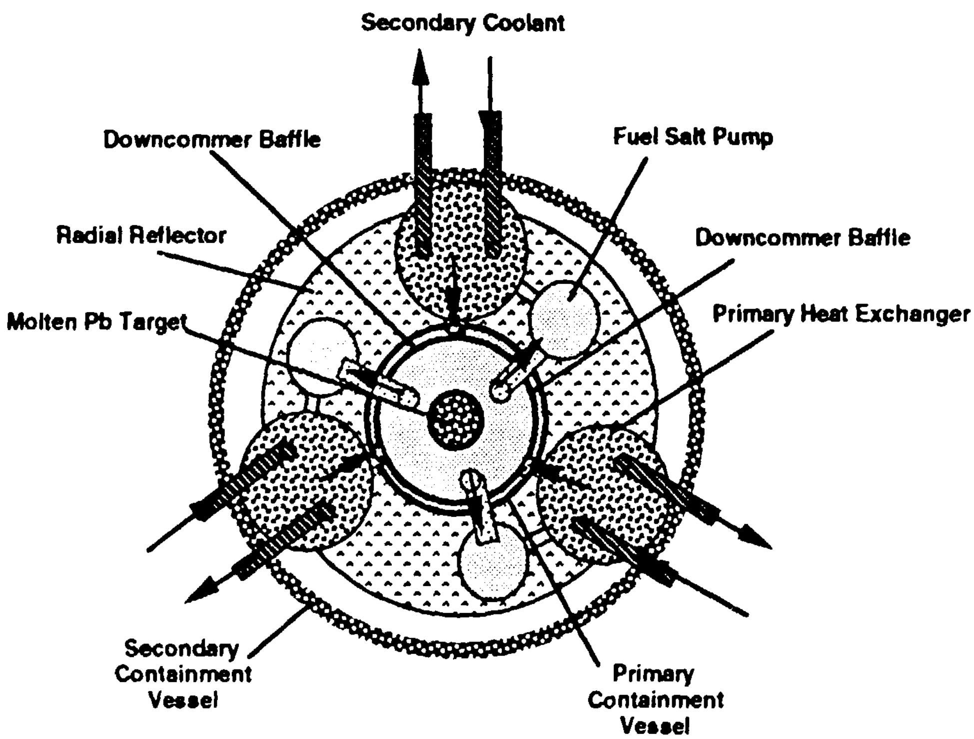


Figure J4 (2 pages)
ATW/ABC: Schematic flow sheets.


Proton Beam
Target Secondary Coolant
Fuel-Salt Motor
Target Heat Exchanger
Target EM Pump
Secondary Coolant
Upper Axial Reflector
Primary Heat Exchanger
Fuel-Salt Pump
Secondary Containment Vessel
Sparge Bubble Injector/Separator
Target Window
Radial Reflector
Downcommer Baffle
Primary Containment Vessel
Lower Axial Reflector


Secondary Coolant
Downcommer Baffle
Fuel Salt Pump
Radial Reflector
Downcommer Baffle
Molten Pb Target
Primary Heat Exchanger
Secondary Containment Vessel
Primary Containment Vessel

J5

Format	**Description**
Title	AMSB (Accelerator Molten Salt Breeder)
Application of the reactor	TWS, NPP
Reactor type	SADS/MSR
Power output	Medium power/1 000 - 2 100 MWt
Organization (name)	Tokay University, Japan
Development status	Studies

Description

The AMSB plant design composed of a proton accelerator and molten-fluoride target system as a fissile breeder is considered at Tokay University (Japan).

AMSB is composed of three parts: an 1 GeV-300 mA Linac, a molten-fluoride target/blanket system, and a heat transfer and electric power recovery system. The size of the target $LiF-BeF_2-ThF_4$ salt bath is 4.5 ~ 5 m in diameter and 7 m in depth. To keep them smaller in size, a comparatively slow proton of 1 GeV was chosen. The inside of the breeder vessel made of Hastelloy N (Ni-Mo-Cr alloy) is covered by thick graphite blocks immersed in salt. The target salt is introduced from the upper part forming a vortex of about 1 m in depth of salt. The proton beam is directly injected into the off-centered position near the vortex bottom, saving the neutron leakage and improving the heat dissipation.

The key features of the AMSB concept are the following:

o The target/blanket system is subcritical, and is not affected by radiation damage

o The heat removal could be managed by the dynamic salt, which is diluted in heavy isotopes

o The shuffling is automatic

Unsolved technological problems seem to be:

o High current Linac development

o Proton beam injection part engineering

Schedule

The conceptual study has not been completed.

AMSB Unit Data

Type	Standard type	High-gain type
Proton beam	1 GeV, 300 mA	1 GeV, 300 mA
Target salt	$^7LiF-BeF_2-ThF_4$	$^7LiF-BeF_2-ThF_4-^{233}UF_4$
Salt temperature, inlet/outlet	580 °C/680 °C	580 °C/680 °C
Molten salt loading	243 t	126 t
Neutron production per proton	25 ~ 40	36 ~ 58
U-233 production	(0.57 ~ 0.92) t/year	(0.82 ~ 1.33) t/year
Fissile inventory	(0 ~ 0.5) t	3.4 t
(doubling time)	(0 - 1 yr)	(2 ~ 4 years)
Incineration rate		
fission	~46 kg/yr	~174 kg/yr
spallation	~46 kg/yr	~46 kg/yr
Thermal output	1000 ~ 1500 MWt	1400 ~ 2100 MWt
Electric output	430 ~ 650 MWe	600 ~ 900 MWe
Linac consumption	600 ~ 700 MWe	600 ~ 700 MWe

References

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Plutonium (TRU) transmutation and ^{233}U production by single-fluid type accelerator molten-salt breeder (AMSB).
Proceedings International Conference on Accelerator-Driven Transmutation Technologies and Applications, Las Vegas, NV, July 25-29, 1994.

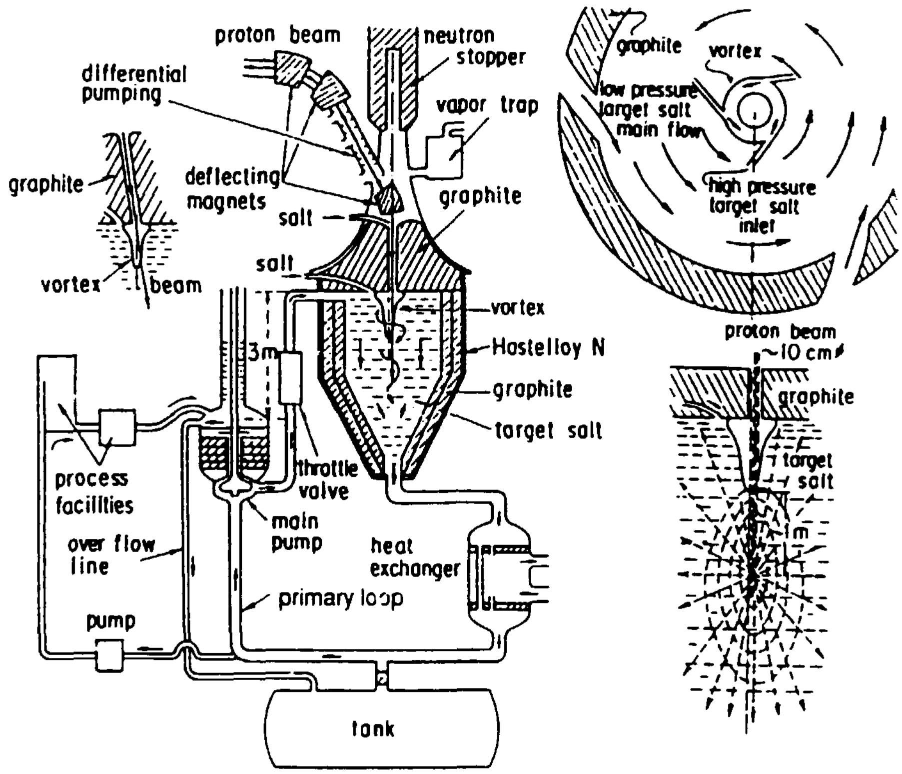

Figure J5
AMSB: Schematic.

J6

Format	Description
Title	BBR (Breeder-Burner Reactor Subcritical System)
Application of the reactor	TWS
Reactor type	SADS/MSR
Power output	Large power/5 000 MWt
Organization (name)	CEA, France
Development status	Studies

Description

Hybrid system Breeder-Burner Reactor Subcritical System (BBR) is investigated at CEA, mainly from a conceptual point of view, in order to assess its potential to transmute radioactive wastes (mainly long-lived fission products, LLFP) and its potential to insure a minimal long-term radiological risk related both to the fuel inventory inside the system and to the full fuel cycle (mass flows, reprocessing, transport waste disposal).

The system under study uses:

- o The Th cycle
- o High fuel burn-up (~45 % of heavy atoms)
- o Fast neutron spectrum and a flux level ~$2 \cdot 10^{15}$ n/s/cm^2
- o Subcriticality in the range of $k_{eff} \approx 0.85$
- o Neutron balance improvements (e.g. leakage decrease)
- o Fuel breeding (to compensate for reactivity swing)

The system has an external neutron source via proton beam (1.5 GeV, 270 mA), and the fuel of the subcritical reactor acts as target and is made of molten salts (Th + 3 % Pu-239) Cl_2 + Pb Cl_3 (where Pu-239 can be replaced by 5 % TRU), with the addition of Tc-99. The choice of molten salt chlorides is not essential, but taken as an example of a high burn-up configuration. The electric power of the system is 2 GWe, with a maximum specific power of ~70 W/cm^3. The fuel lifetime is 50 years with reloading intervals of 10 years.

The following rates of transmutation are obtained: Pu-239: 0.55 t/GWe/yr; Tc-99: 0.062 t/GWe/yr

The decrease of the waste toxicity as compared to the feed fuel is a factor of ~40 for "long" time intervals (10^2 - 10^6 years) and ~ 1 500 for "short" tiem intervals (10^2 -10^4 years).

Schedule

Conceptual studies are under way now.

BBR Unit Data

Accelerator

- Type	Proton LINAC
- Energy	1.5 GeV
- Current	270 mA

Target/Core

- Diamater	495 cm (all core)
- Height	600 cm
- Cooling material	60 (Th+3% ^{239}Pu)Cl_2 -40% $PbCl_3$ mol %
- Averaged fresh fuel composition Th/Pu/^{99}Tc weight ratios	35/1.1/1.85
- Isotopic composition of Pu	100 % ^{239}Pu, no TRU
- Isotopic composition of LLFP	100 % ^{99}Tc

System characteristics

- K_{eff}	0.85
- Core thermal power	5 000 MWt
- Power density: average/max	40/70 W/cm^3
- Neutron flux averaged in core	2×10^{15} n/cm^2/s
- Fuel (target) dwelling time	50 years
- Burnup reactivity swing	0.003 %δ k/k
- Burnup	45 % heavy atom
- Reloading interval	10 years

References

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Waste transmutation with minimal fuel cycle long-term risk.
Proceedings International Conference on Accelerator-Driven Transmutation Technologies and Applications.
Las Vegas, NV, July 25-29, 1994.

List of Abbreviations and Glossary of Terms

A

ABB	Asea Brown Boveri
ABWR	Advanced Boiling Water Reactor, see BWR
Accelerator	A device that increases the velocity and energy of charged particles such as electrons and protons; also referred to as a particle accelerator. In a "linear" accelerator, particles are accelerated in a straight path
Actinides	The elements with atomic numbers above 88 (actinium, element 89). The actinides series includes uranium, atomic number 92, and all the man-made transuranic elements
Active component	Any component that is not passive is active
Active system	Any system that is not passive is active
ADP	Advanced Double Pool reactor
ADS	Automatic Depressurization System
AEC	(U.S) Atomic Energy Commission
AECL	Atomic Energy of Canada Limited. A designer/supplier of nuclear reactors
AGR	Advanced Gas-Cooled Reactor
ALARA	As low as reasonable achievable
ALMR	Advanced Liquid Metal Reactor
ALSEP	Apollo Lunar Scientific Experiment Package
ALWR	Advanced Light Water Reactor, see LWR
AMSB	Accelerator Molten Salt Breeder
amu	Atomic mass unit
ANL	Argonne National Laboratory, US
AP	Advanced Passive
APWR	Advanced Pressurized Water Reactor, see PWR
ATR	Advanced Thermal Reactor: A heavy water moderated, light water-cooled reactor
ATWS	Anticipated Transients Without Scram
Availability Factor	The availability factor of a nuclear unit or station is the ratio of time when energy can be produced to the total time
AWTS	Accelerator-driven Waste Transmutation System

B

Base Load	The minimum load produced over a given period. A station used for base load is a station that is normally operated when available to provide power to meet the minimum load demands
BDA	Beyond Design Accident
BMDO	Ballistic Missile Defence Organization, US
BN	Russian version of sodium cooled Fast Breeder Reactor
BNL	Brookhaven National Laboratory
BOL	Beginning Of life
BOM	Beginning Of mission
BOP	Balance Of Plant
Breeder Reactor	A nuclear reactor that produces more fissile material than it consumes. In fast breeder reactors, high-energy (fast) neutrons produce most of the fissions, while in thermal breeder reactors, fissions are principally caused by low-energy (thermal) neutrons. See Fast reactor
Breeding Ratio	The conversion ratio when it is greater than unity. A high breeding ratio results in a short doubling time
BWR	Boiling-Water Reactor. A light-water reactor that employs a direct cycle; the water coolant that passes through the reactor is converted to high pressure steam that flows directly through the turbines

C

CANDU	CANadian Deuterium Uranium reactor: A type of heavy water reactor
Capacity Factor	See Load Factor
CAREM	Conjunto Argentina de Reactores Modulares
CCV	Cooldown Control Valve
CDA	Control Drum Actuator
CE	Combustion Engineering
CEA	Commissariat à l'Energie Atomique
CEC	Commission of the European Communities
Charge of a reactor	The fuel placed in a reactor
CHP	Combined Heat and Power

Conversion as used in reactor technology	The ratio between the number of fissile nuclei produced by conversion to the number of fissile nuclei destroyed. If the ratio for a given reactor is greater than one, it is a breeder reactor; if it is less than one, it is a converter reactor
Conversion, chemical	The operation of altering the chemical form of a nuclear material to a form suitable for its end use
Converter reactor	A reactor that produces some fissile material, but less than it consumes
Coolant	The medium in a nuclear reactor which removes heat from the reactor core where fission occurs and heat is produced, and transfers it to systems which convert the heat into steam
COPUOS	(UN) Committee On Peaceful Uses of Outer Space
Critical	Capable of sustaining a nuclear chain reaction
CRU	Combined Rotating Unit
CWCS	Containment Water Cooling System

D

DARPA	Defense Advanced Research Project Agency
DBA	Design Basis Accident
DEC	Direct Energy Conversion
Decay Heat	The heat produced by the decay of radioactive nuclides
DIPS	Dynamic (Radio)isotope Power System
DNHPP	Decentralized Nuclear Heating Power Plant
DOE	US Department of Energy

E

E-MAD	Engine Maintenance, Assembly and Disassembly
EASEP	Early Apollo Scientific Experiment Package
ECCS	Emergency Core Cooling System
EFR	European Fast Reactor
EHRS	Emergency Heat Removal System
EM	Electromagnetic
EMP	Electromagnetic Primary Pump

EMP	Electromagnetic Pump
Enriched fuel	See Fuel, enriched
Enrichment	i) the fraction of atoms of a specified isotope in a mixture of isotopes of the same element when this fraction exceeds that in the naturally occurring mixture; ii) any process by which the content of a specified isotope in an element is increased
EOL	End Of Life
EOM	End Of Mission
EOT	Earth Orbital Terminal
EPA	(U.S.) Environmental Protection Agency
EPR	European Pressurized Reactor
EST-1	Engine Test Stand Number One
eV	Electron volt
EVA	Extravehicular Activity
EWST	Emergency Water Storage Tank

F

Fail-safe	The term describes the behaviour of a component or system, following a failure (either internal or external). If a given failure leads directly to a safe condition, the component or system is fail-safe with respect to that failure
Fast neutrons	See **Neutron, fast**
Fast reactor	A nuclear reactor in which no moderator is present in the reactor core or reflector. So the majority of fissions are produced by fast neutrons. If a fertile species is present in the fast reactor core or in the blanket surrounding the core, it will be converted into fissile material by neutron capture. When more fissile material is produced than is used to maintain the fission chain, the reactor is called a breeder
Fault-/error-tolerant	(also called forgiving). The term fault-/error-tolerant, also called forgiving, describes the degree to which equipment faults/human inaction (or erroneous action) can be tolerated
FBR	Fast Breeder Reactor (see Fast reactor)
FDA	Final Design Approval
FEPS	Federation of Electric Power Companies

FFTF	Fast Flux Test Facility
Fissile	i) of a nuclide: capable of undergoing fission by interaction with slow neutrons;
	ii) of a material: containing one or more fissile nuclides
Fission products	Nuclides produced either by fission or by the subsequent radioactive decay of the nuclides thus formed
Fissionable	i) of a nuclide: capable of undergoing fission by any process;
	ii) of a material: containing one or more fissionable nuclides
FOAKE	First-Of-A-Kind-Engineering
Foolproof	Safe against human error or misguided human action
Fossil fuel	A term applied to coal, oil and natural gas
FP	Fission Product(s)
FPSE	Free Piston Stirling Engine
FSAR	Final Safety Analysis Report
Fuel cycle	The sequence of processing, manufacturing, and transportation steps involved in producing fuel for a nuclear reactor, and in processing fuel discharged from the reactor
Fuel, enriched	Nuclear fuel containing uranium which has been enriched in one or more of its fissile isotopes or to which chemically different fissile nuclides have been added
Fuel, nuclear	Material containing fissile nuclides which when placed in a reactor enables a self-sustaining nuclear chain to be achieved
Fuel reprocessing	The chemical or metallurgical treatment of spent (used) reactor fuel to recover the unused fissionable material, separating it from radioactive waste. The fuel elements are chopped up and chemically dissolved. Plutonium and uranium and possibly other fissionable elements are then separated for further use

G

GA	General Atomic
GCR	Gas-Cooled Reactor
GE	General Electric
GEO	Geostationary Earth Orbit, Geosynchronous Orbit
GHR	Gas-Cooled Heating Reactor

GIS	Graphite Impact Shell
GLFC	Graphite Lunar Module Fuel Cask
GmbH	German term signifying "Limited"
Grace period	The period of time which a safety function is ensured without the necessity of personnel action in the event of an incident/accident
GSOC	Geosynchronous Orbit Complex
GWe	Gigawatt (10^9 watts) electric

H

Half-life	The period of time required for the radioactivity of a substance to drop to half its original value; the time that it takes for half of the atoms of a radioactive substance to decay. Measured half-lives vary from millionths of a second to billions of years
Heavy water	Deuterium oxide (D_2O): water containing significantly more than the natural proportion (1 in 6500) of heavy hydrogen (deuterium) atoms to ordinary hydrogen atoms
HEO	High Earth Orbit
HEU	Highly-Enriched Uranium
HR	Heat Rejection
HSBWR	Hitachi Small Boiling Water Reactor
HTR or **HTGR**	High-Temperature Gas-Cooled Reactor. A graphite-moderated, helium-cooled advanced reactor that utilizes low enriched uranium
HTTR	High Temperature Test Reactor
HWR	Heavy-Water Reactor. Heavy water is used as a moderator in certain reactors because it slows down neutrons effectively and also has a low cross section for absorption of neutrons
HX	Heat Exchange
Hydropress EDO	Reactor Plant Design Bureau, Podolsk, Russia

I

I&C	Instrumentation & Control system
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IEA	International Energy Agency
IFR	Integral Fast Reactor
IHS	Intermediate Heat Exchange
IIASA	International Institute for Applied System Analysis
INET	Institute Nuclear Energy Technology, China
Inherent safety characteristics	Safety achieved by the elimination of a specified hazard by means of the choice of material and design concept
INSRP	Interagency Nuclear Safety Review Panel
Isotopes	Nuclides having the same atomic number (i.e. identical chemical element) but different mass numbers; e.g. 92-Uranium-235 and 92-Uranium-238. Isotopes have the same chemical properties but slightly different physical properties
IUS	Inertial Upper Stage

J

JAERI	Japan Atomic Energy Research Institute

K

KEPCO	Kansai Electric Power Company
keV	Thousands electron volts
kT	Kiloton
kW	Kilowatt
kWh	Kilowatt hour (10^3 watt hour)

L

LANL	Los Alamos National Laboratory
LCP	Large Communications Platform
LEO	Low Earth Orbit
LET	Linear Energy Transfer
LEU	Low Enriched Uranium
LH_2	Liquid hydrogen
LiH	Lithium hydrogen
LINAC	Linear Accelerator
Linear accelerator	A long straight tube (or series of tubes) in which charged particles (ordinarily electrons or protons) gain in energy by action of oscillating electromagnetic fields
LLFP	Long Lived Fission Products
LMCPBR	Liquid Metal Cooled Pebble Bed Reactor
LMFBR	Liquid Metal Fast Breeder Reactor. A fast reactor that employs liquid metal (sodium) as a coolant. The sodium in the primary loop passes through the reactor and transfers its heat to sodium in a secondary loop. This sodium then heats water in a tertiary loop which produces steam and drives a turbine. See also Fast Reactor
Load Factor	The load factor of a nuclear unit or station for a given period of time is the ratio of the energy that is produced during the period considered to the energy that it could have produced at maximum capacity under continuous operation during the whole of that period. Also called Capacity Factor
LOCA	Loss Of Coolant Accident
LOF	Loss Of Flow
LOHS	Loss Of Heat Sink
LOT	Lunar Orbital Terminal
LRC	Lunar Resources Complex
LWGR	Light Water Graphite Reactor. A reactor that uses ordinary water as a coolant and graphite as a modertor and utilizes slightly enriched Uranium-235 (e.g. Russian multi-channel RBMK reactor plant)
LWR	Light-Water Reactor. A nuclear reactor that uses ordinary water as both a moderator and a coolant and utilizes slightly enriched Uranium-235 fuel. There are two commercial LWR types: the boiling-water reactor (BWR) and the pressurized water-reactor (PWR)

M

MAP	Minimum Attention Plant
MARS	Multipurpose Advanced Reactor Inherently Safe
MCP	Main Circulation Pump
MeV	Million electron volts
MHTGR	Modular HTGR, see HTR
MHW	Multihundred watt
Minor actinides	The transuranic elements minus plutonium. Usually this term is used to refer to neptunium, americium, and curium. Some also refer to these as the "minor" transuranics. Plutonium is the dominant transuranic, but these minor transuranics contribute comparable radioactivity in spent fuel
Moderator	A material, such as ordinary water, heavy water, beryllium, graphite and some others used in a nuclear reactor to slow down fast neutrons so fissile nuclei can more easily and efficiently capture them, thus increasing the likelihood of further fission
MOX fuel element	Mixed Oxide fuel element. Fuel element in which fuel is an intimate mixture of uranium and plutonium oxides
MPD	Maximum Permissible Dose
MSBR	Molten Salt Breeder Reactor, see MSR
MSPWR	Mitsubishi Simplified PWR
MSR	Molten Salt Reactor. A nuclear reactor that uses fluid fuel, e.g. molten salt fluorides of Li, Be, Th, U.
MW	Megawatt
MWe	Megawatt (10^6 watts) electric
MWh	Megawatt hour (10^6 watts hour)
MWt	Megawatt (10^6 watts) thermal

N

NaK	Sodium-potassium eutectic mixture
NASA	(U.S.) National Aeronautics and Space Administration
NC	Natural Circulation
NCV	Nozzle Control Valve

NEA	OECD Nuclear Energy Agency
NEP	Nuclear Electric Propulsion
NEPSTP	Nuclear Electric Propulsion Space Test Program
NERVA	Nuclear Engine for Rocket Vehicle Application
Neutrons, fast	Neutrons of kinetic energy greater than some specified value. This value may vary over a wide range and will be dependent upon the application, such as reactor physics, shielding, or dosimetry. In reactor physics the value is frequently chosen to be 100 000 eV (electron-Volt)
Neutrons, slow	Neutrons of kinetic energy less than some specified value (see neutrons, fast). In reactor physics the value is frequently chosen to be 1 eV
Neutrons, thermal	Neutrons in thermal equilibrium with the medium in which they exist
NF	Nuclear Furnace
NHP	Nuclear Heating Plant
NHPD	Nuclear Heat and Power Plant
NPP	Nuclear Power Plant. A reactor or reactors together with all structures, systems and components necessary for the production of power (i.e. heat or electricity)
NPPS	Nuclear Power Propulsion System
NRC	(Nuclear Regulatory Commission): US body regulating the use of nuclear energy
NRDS	Nuclear Rocket Development Station
NSSS	Nuclear Steam Supply System
NTS	Nevada Test Site
Nuclear energy	Energy released in nuclear reactions or transitions
Nuclear fuel	See Fuel, nuclear
Nuclear reaction	A reaction involving a change in an atomic nucleus, such as fission, fusion, neutron capture, or radioactive decay, as distinct from a chemical reaction, which is limited to changes in the electron structure surrounding the nucleus
Nuclear reactor	A device in which a fission chain reaction can be initiated, maintained, and controlled. Its essential component is a core containing fissionable fuel. It is sometimes called an atomic "furnace"; it is the basic machine of nuclear energy
Nuclide	Any species of atom that exists for a measurable length of time. The term is used synonymously with isotope. A radionuclide is a radioactive nuclide.

O

OECD	Organisation for Economic Co-Operation and Development
OKBM	Reactor Plant Design Bureau, Nizni-Novgorod, Russia
OMEGA	Options for Making Extra Gains from Actinides and Fission Products
OTV	Orbital Transfer Vehicle

P

Passive component	A component which does not need any external input to operate
Passive system	Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation
PBIS	Pressure Balanced Injection System
PbTe	Lead telluride
PC	Power Convertor
PCCS	Passive Containment Cooling System
PCI	Pellet Clad Interaction
PCIV	Prestressed Cast Iron Vessel
PCRV	Prestressed Concrete Reactor Vessel
PCS	Power Conversion System
PCV	Pressure Containment Vessel
Peak load	The maximum load produced by a unit or group of units in a stated period of time. A station used for peak load generation is a station that is normally operated to provide power during maximum load periods only
Pebble bed reactor	A reactor, which utilizes spherical fuel elements
Per capita	Per unit of a population
PHRS	Passive Heat Removal System
PIUS	Process Inherent Ultimate Safety Reactor
Plutonium	A heavy, radioactive, man-made metallic element with the atomic number 94, created by absorption of neutrons in uranium-238. Its most important isotope is plutonium-239, which is fissile
PPO	Pressed Plutonium (-238) Oxide

Primary energy The energy content of fuels before they are processed and converted into forms used by the consumer. Primary energy refers to energy in the form of natural resources: water flowing over a dam, freshly mined coal, crude oil, natural gas, natural uranium. Only rarely can primary energy be used to supply final energy; one of the few forms of primary energy that can be used as final energy is natural gas

Proton A particle with a single positive unit of electrical charge and a mass that is approximately 1.840 times that of the electron. It is the nucleus of the hydrogen atom and a constituent of all atomic nuclei.

PSA Probabilistic Safety Analysis

PSI Paul Scherrer Institute, Switzerland

PSM Power System Module

PSOV Propellant Shutoff Valve

PUC Public Utility Commission

PUREX process The plutonium and uranium extraction (PUREX) process is an aqueous process used in several foreign commercial and U.S. defense programs for separating out elements in spent nuclear fuel

Pyroprocessing Nonaqueous processing carried out at high temperatures. An example of this is the relatively new technology being developed for reprocessing

PWR Pressurized-Water Reactor. A light-water moderator and cooled reactor that employs an indirect cycle; the cooling water that passes through the reactor is kept under high pressure to keep it from boiling, but it heats water in a secondary loop that produces steam that drives the turbine

R

R-MAD Reactor Maintenance, Assembly and Disassembly

rad Radiation absorbed dose

Radioactive Referring to the spontaneous transformation of one atomic nucleus into a different nucleus or into different energy states of the same nucleus

Radioactive decay The spontaneous transformation of one atom into a different atom or into a different energy state of the same atom. The process results in a decrease, with time, of the original number of radioactive atoms in a sample

Radioactive waste	The unwanted radioactive materials formed by fission and other nuclear processes in a reactor or obtained in the processing or handling of radioactive materials. Most nuclear waste is initially in the form of spent fuel. If this material is reprocessed, new categories or waste result: high-level, transuranic, and low-level wastes (as well as others)
Radioisotope	A radioactive isotope. An unstable isotope of an element that decays spontaneously, emitting radiation. Radioisotopes contained in the spent fuel resulting from the production of nuclear power generally fall into two categories: fission products and transuranic elements (known as transuranics, actinides, or TRU), and activation products produced by neutron absorption in structural material in the spent fuel
RBE	Relative Biological Effectiveness
RBMK	See **LWGR**
RBR	Rotating Bed Reactor
RCC	Reinforced Concrete Containment
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RDIPE	Research and Development Institute of Power Engineering, Moscow, Russia
Reactor core	The central portion of a nuclear reactor containing the fuel elements and the control rods (and part of the coolant and moderator), where most of the energy is produced
Recycling	The reuse of fissionable material, after it has been recovered by chemical processing from spent reactor fuel
rem	Roentgen equivalent man
Reprocessing, fuel	A generic term for the chemical and mechanical processes applied to fuel elements discharged from a nuclear reactor; the purpose is to remove fission products and recover fissile (uranium-233, uranium-235, plutonium-239), fertile (thorium-232, uranium-238) and other valuable material
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
RRA	Rolls-Royce and Associates
RRC-KI	Russian Research Centre "Kurchatov Institute", Moscow, Russia
RTG	Radioisotope Thermoelectric Generator
RVACS	Reactor Vessel Auxiliary Cooling System

S

SADS	Subcritical Accelerator-Driven System
Safeguards	Term used to refer to the total set of international verifications, observations, etc, which together constitute a determination that nuclear materials (or, in some international agreements, facilities or other materials) have not been diverted from nuclear power programmes to the production of nuclear weapons
SAFR	Sodium Advanced Fast Reactor
SBWR	Simplified or Small BWR. See BWR
SDR	Slowpoke Demonstration Reactor, see SES
SECURE	Safe Environmentally Clean Urban Reactor
SES	Slowpoke Energy System
SCLWR	Steam Cooled Light Water Reactor
SG	Steam Generator
SiGi	Silicon germanium
Simplified design	A system designed with a minimum number of components to achieve the related safety function and relying as little as possible on support systems
SIR	Safe Integral Reactor
SIS	Safety Injection System
SMF	Space Manufacturing Facility
SMR	Small and Medium-sized Reactors
SMS	Ship Mobile System
SNAP	Space Nuclear Auxiliary Power program, US
SNPS	Space Nuclear Power Source
SNR	Space Nuclear Reactor
SPAR	Space Power Advanced Reactor
Spent fuel	Nuclear reactor fuel that has been irradiated (used) to the extent that it can no longer effectively sustain a chain reaction and therefore has been removed from the reactor for disposal. This radiation fuel contains fission products, uranium, and transuranic isotopes
SPR	Advanced Space Nuclear Power Program
SPS	Satellite Power System
STA	Science and Technology Agency, Japan

STAR-C	Space Thermionic Advanced Reactor Compact, US
STS	Space Transportation System (Space Shuttle)
Subcritical	Not capable of sustaining a nuclear chain reaction, but involving some degree of multiplication of neutrons

T

T-111	Tantalum alloy (Ta-8W-2Hf)
TAGS	Silver antimony germanium telluride
Target	Material subjected to particle bombardment (as in an accelerator) in order to induce a nuclear reaction
TBCV	Turbine Bypass Control Valve
TCT	Thermionic Critical Technology program, US, see SNP
TE	Thermoelectric
TFE	Thermionic Fuel Element
TI	Thermionic
TITR	Thermionic Test Reactor
TOP	Transient Over Power
TOSBWR	Toshiba Simplified Boiling Water Reactor
TPC	Transmutation Plant System
TPS	TRIGA Power Safety System, see DNHPP
Transmutation	The transformation (change) of one element into another by a nuclear reaction or series of reactions
Transuranic	An element above uranium in the Periodic Table of elements - that is, one that has an atomic number greater than 92. All transuranics are produced artificially (during a man-made nuclear reaction) and are radioactive. They are neptunium, plutonium, americium, curium, berkelium, californium, einsteinium, fermium, mendelevium, nobelium, and lawrencium
TRIGA	Training of personnel, nuclear Research, Isotope production, General Atomic
TRU	Transuranium elements
TRUEX	A chemical solvent process under development to extract transuranics from high-level waste
TSCV	Turbine Shutoff Control Valve

TVA	Turbine Valve Actuator
TWh	Terrawatt (10^{12} watt) hour
TWS	Transmutation Waste System
TZM	Molybdenum alloy (Mo-0.6, Ti-0.1, Zr-0.035)

U

U-ZrH	Uranium zirconium hydride
UO_2	Uranium dioxide
Uranium	A radioactive element of atomic number 92. Naturally occurring uranium is a mixture of 99.28 per cent uranium-238, 0.715 per cent uranium-235, and 0.0058 per cent uranium-234. Uranium-235 is a fissile material and is the primary fuel of nuclear reactors. When bombarded with slow or fast neutrons, it will undergo fission. Uranium-238 is a fertile material that is transmuted to plutonium-239 upon the absorption of a neutron.
USR	Ultimate Safety Reactor, see MSR

V

VVER	Russian version of PWR plant

W

W	Westinghouse
WANO	World Association of Nuclear Operations
Waste separation	The dividing of waste into constituents by type (for example, high-level, medium-level, low-level) and/or by isotope (for example, separating out plutonium and uranium). The waste may be separated by a chemical solvent process such as PURES or by any of a number of other chemical or physical processes
WEC	World Energy Council
WOCA	World Outside CPE Areas (MEM ou Monde à Economie de Marché)
WRI	World Resource Institute

X

XE Experimental Engine

Z

ZrH Zirconium hydride

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