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Survey of Light-Water-Reactor Designs to be Offered in the United States

Irving Spiewak

Report prepared by

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712 S. Main Street
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under
Purchase Order No. 11X39051V

for
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
operated by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U.S. DEPARTMENT OF ENERGY
under Contract No. DE-AC05-84OR21400

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
NTIS price codes—Printed Copy: A04; Microfiche A01

ORNL/TM-9948

SURVEY OF LIGHT-WATER-REACTOR DESIGNS TO BE OFFERED IN THE UNITED STATES

Irving Spiewak, Consultant

Date Published - March 1986

NOTICE: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

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ABSTRACT

ORNL has conducted a Nuclear Power Options Viability Study for the Department of Energy. That study is primarily concerned with new technology which could be developed for initial operation in the 2000-2010 time frame. Such technology would have to compete not only with coal options but with incrementally improved commercial light-water reactors. This survey reported here was undertaken to gain an understanding of the nuclear commercial technology likely to be offered in the late 1980s and perhaps beyond.

The three U.S. vendors actively marketing NSSSs are each developing a product for the future which they expect to be more reliable, more maintainable, more economical, and safer than the present plants. These are all essentially 3800-MW(t) designs, although all are studying smaller plants. They apparently will be offered as standard prelicensed designs with much larger scope than earlier NSSS offerings, with the possibility of firm prices.

Westinghouse with Mitsubishi Heavy Industries is developing a completely new design (APWR) to be built initially in Japan, hopefully for operation by the mid-1990s. Westinghouse is making a strong effort to have the APWR licensed in the U.S. as a standard plant.

Combustion Engineering (C-E) is evaluating potential improvements to the System-80 standard design (CESSAR) that has already received final design approval by the NRC.

General Electric (GE), with Hitachi and Toshiba, is developing a new design (ABWR) that incorporates advanced features which have been proven by the worldwide BWR suppliers. The ABWR is to be built initially in Japan, but the design could be adapted to the United States.

Westinghouse, C-E, and GE have done some conceptual evaluation of reactors in the 600-MW(e) class. The Westinghouse concept is a two-loop plant intended for factory assembly in a shipyard and delivery to a site by barge. The GE concept is a modification of the ABWR with some additional passive safety features. The C-E designs range from scaled-down System-80s to small natural circulation PWRs. These concepts may be of interest to DOE or EPRI as "small" reactors.

1. INTRODUCTION

ORNL has conducted a Nuclear Power Options Viability Study for the Department of Energy. That study was primarily concerned with new technology which could be develcped for initial operation in the 2000-2010 time frame, including gas-cooled and sodium-cooled options.

Such new technology would have to compete not only with coal options but with incrementally improved commercial light-water reactors. The survey of this report was undertaken to gain an understanding of the nuclear commercial technology likely to be offered in the late 1980s. If commercially successful, it might continue to be offered in the 1990s and beyond.

The survey builds on information obtained by the author in carrying out an earlier study of nuclear technology options. The earlier data were updated through visits to Combustion Engineering (C-E), General Electric (GE), and Westinghouse (W) during July-September 1984. Some additional insights were gained by the author through visits to several utilities, the Nuclear Regulatory Commission (NRC), and the Electric Power Research Institute (EPRI) over this period.

Vendor personnel were helpful in responding to questions of clarification posed during the writing of the report. Each vendor reviewed the portion of the report dealing with his products as a further check of accuracy.

The report is divided into three major sections, each dealing with one of the three vendors canvassed. Each section is subdivided to cover a general description of the concepts, their commercial status, strategy for control of costs, safety-related parameters, and licensing considerations. Finally, there is a brief overall summary.

2. ADVANCED PWR DESIGNS OF THE WESTINGHOUSE ELECTRIC COMPANY

The product which Westinghouse currently offers is a two- to four-loop design typified by the Phillipine plant (two-loop) or the Callaway plant (four-loop). The capacity is in the range 600 to 1250 MW(e). The proposed Sizewell-B design is the most recent of that class and is discussed in Sect. 2.1.

Westinghouse is developing a more advanced design (APWR) for a group of Japanese utilities. The APWR is a 1350-MW(e) design. This report presents the publicly available data on the APWR (Sect. 2.2).

Westinghouse is exploring novel configurations for its two-loop plant to reduce cost and construction schedule. A barge-mounted concept is described in Sect. 2.3.

2.1 THE SIZEWELL-B REACTOR

In the early 1970s, Westinghouse, together with a group of U.S. utilities and Bechtel, evolved a standard PWR design, Standardized Nuclear Unit Power Plant System (SNUPPS). SNUPPS is a conventional Westinghouse four-loop reactor with vertical U-tube steam generators and is typical of today's U.S. plants (Fig. 2.1). The first plant of this series is Callaway, scheduled for operation in early 1985.

SNUPPS incorporates two independent reactor shutdown systems with a backup emergency boration system, two high-pressure and two intermediate-pressure emergency core cooling pumps, and three diverse auxiliary feedwater supply systems to help remove residual heat from the steam generators. Most of these features are present in other Westinghouse LWRs, and SNUPPS may be regarded as being typical of modern Westinghouse PWRs. The core-melt probability of these plants as calculated by probabilistic risk assessment (PRA) is usually in the range 10^{-4} to 10^{-5} per reactor year.

An improved version of the SNUPPS design, and one in which the greatest attention has been given to safety, was prepared by the Central Electricity Generating Board (CEGB) in the United Kingdom for a plant designated Sizewell-B. To meet the stringent requirements posed by the high population density in the vicinity of the site, the CEGB made an intensive five-year study of the reactor safety problems and the measures that might be taken to minimize the probability of an accident. This study was exceptionally thorough and included the exploration of a wide range of possibilities and much new design work.

The Sizewell-B design has added the following features to enhance safety above that of the SNUPPS (Callaway) design:

1. Four high-pressure safety injection (HPSI) pumps dedicated to safety, each with heads lower than 2000 psi and with higher flow volumes than Callaway's. The actuation of the HPSI pumps will

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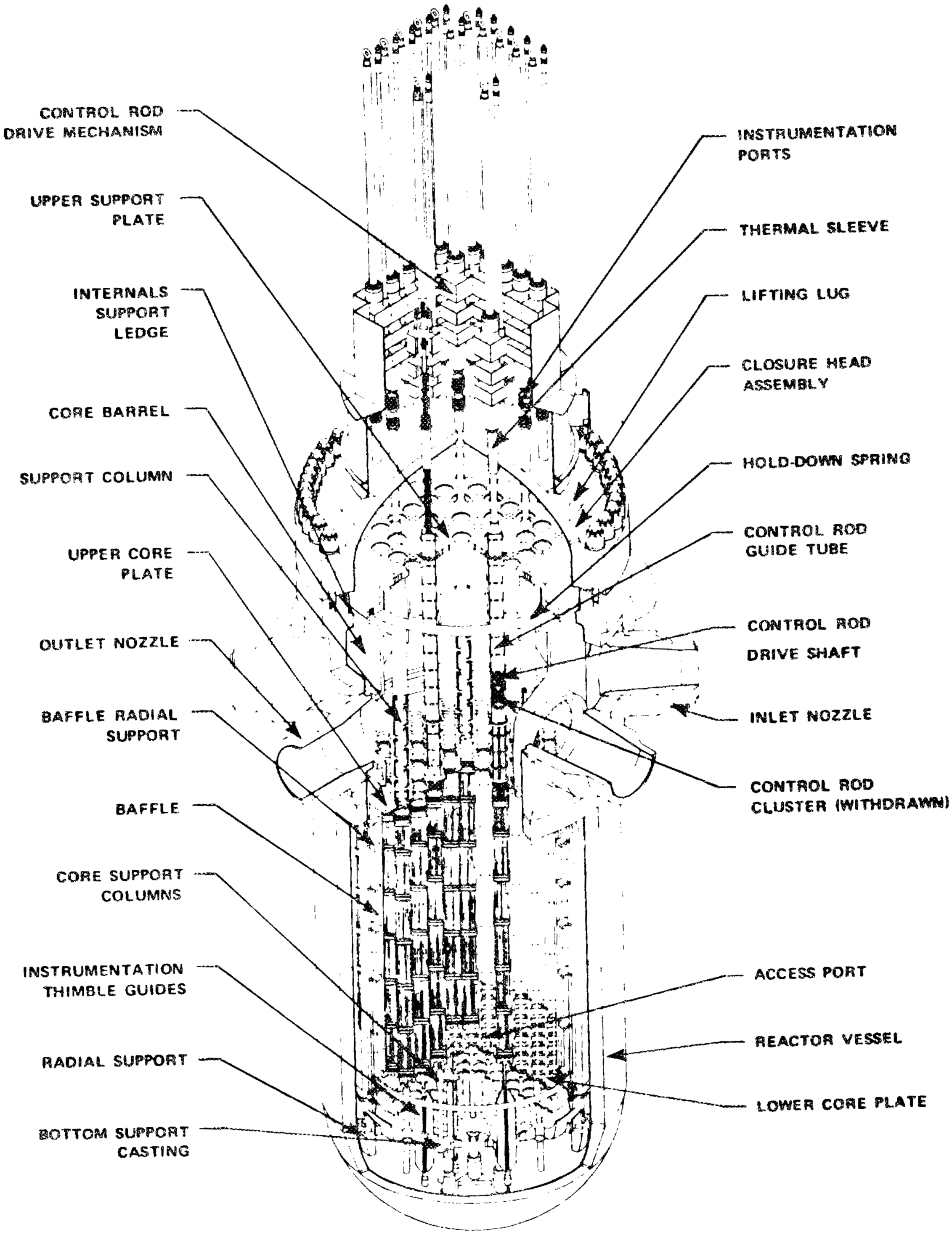

Fig. 2.1 Reactor vessel internals of large PWR plants.

Source: Westinghouse Electric Corporation, Systems Summary of a Westinghouse Pressurized Water Reactor Nuclear Power Plant, (Pittsburgh, Pennsylvania, 1971), p. 40 (reproduced with permission).

automatically shut down the higher head charging pumps, thus preventing overpressurization in overcooling transients.

2. Four accumulators, any two of which are sufficient for core cooling at the 600 psi pressure range (instead of the required three at Callaway).

3. Four low-pressure pumps to recirculate water for core cooling at low pressures and for the containment sprays. These pumps are dedicated to residual heat removal. In addition, the high-pressure HPSI suction is automatically switched to the containment sump when the refueling water storage tank is low. In older Westinghouse reactors, including SNUPPS, such switching to this backup source of water must be done manually.

4. An additional steam-driven auxiliary feed pump, in addition to the two electric pumps used in SNUPPS. All the pumps are farther apart than at Callaway and are, therefore, less subject to common-mode failure.

5. Four diesel generators (instead of two) to provide emergency power in the case of loss of off-site power.

6. A microprocessor-based reactor protection system backed up by a secondary protection system based on solid-state switches.

7. An emergency boration system as a backup reactor trip system to cope with anticipated transients without scram.

8. An extra diesel-driven emergency charging pump to make up for pump seal leakage during station blackout.

9. An additional isolation valve between the high-pressure reactor cooling system and the low-pressure residual heat removal system to minimize the chance of the containment bypass accident sequence (the so-called V sequence).

10. Connections to provide water from fire pumps to containment safety features.

11. Construction of ring forgings with no major welds in the beltline region of the reactor pressure vessel to minimize the chance of vessel brittle failure due to irradiation and overcooling transients.

12. A secondary containment vessel to further reduce the probability of an escape of radioactive material to the environment.

The PRA for the Sizewell-B reactor gives a mean core-melt probability of 1.1×10^{-6} per reactor year (Table 2.1), about two orders of magnitude below that of a typical U.S. reactor. The risk is dominated by loss-of-coolant accidents. The probability of a large release of radioactivity is estimated to be 3×10^{-8} per reactor year. The

Table 2.1. PRA for the Sizewell-B reactor core melt by initiating event

Initiating event	Core melt frequency	Percentage of total core melt frequency
Large LOCA	1.83E-07	15.8
Medium LOCA	2.58E-07	22.2
Small LOCA	3.83E-07	33.0
Steam generator tube rupture	1.91E-08	1.6
Secondary side break inside containment	2.32E-08	2.0
Secondary side break outside containment	3.54E-08	3.0
Loss of main feedwater	1.58E-08	1.4
Closure of one MSIV	5.71E-11	<0.01
Loss of RCS flow	8.11E-11	<0.01
Core power excursion	5.11E-12	<0.01
Turbine trip	8.36E-10	0.07
Spurious safety injection	1.44E-10	0.01
Reactor trip	8.54E-10	0.07
ATWS	1.37E-07	11.8
Loss of offsite power/ turbine trip	6.03E-09	0.5
Interfacing systems LOCA	2.37E-09	0.2
LOCA beyond capacity of ECCS	1.00E-07	8.6
TOTAL	1.16E-06	100.0

cumulative impact of the measures designed to improve safety beyond that of the standard SNUPPS design has been estimated to increase the power plant capital cost about 20%.

2.2 THE ADVANCED PRESSURIZED WATER REACTOR (APWR)

2.2.1. General Description

The stated objectives of APWR are:
- improved operability,
- improved availability,
- low economic and public risk,
- reduced occupational exposure, and
- reduced capital and operating costs.

The APWR contains a number of innovations when compared to earlier Westinghouse four-loop designs:

- The reactor vessel is longer, with the coolant nozzles much higher above the core. (Compare Fig. 2.2 with Fig. 2.1). This provides additional margin for loss-of-coolant accidents.

- The core contains 193 fuel assemblies, each containing 19 × 19 rods. This provides stretch capability to 4200 MW(t) but also represents more conservative thermal design margins (reduced power density) than used in earlier designs.

- The neutron spectrum is controlled through movable zirconia rods which displace water. About 15% of the moderator water is displaced. Secondary control is with gray rods. This control system reduces fuel cost and reduces or eliminates use of dissolved boron.

- There is a radial neutron reflector (steel) to reduce fuel cost and reactor vessel fluence.

- The containment vessel is a large 197-ft diameter steel spherical shell, similar to the KWU designs or the TVA Yellow Creek design. This approach leads to additional space for maintenance and improved working conditions for construction. (A spherical shell provides more volume per unit weight or cost of structure compared to a cylindrical shell.)

- The steam generator design is being improved to contain relatively corrosion-resistant, low-cobalt Inconel-690 tubes, improved sludge collection, presleeved tubes at the tube sheets, and corrosion-resistant tube sheet and tube support materials.

- A large pressurizer is being supplied to provide for full load rejection without actuating pressure relief valves.

- The instrumentation and control system uses microprocessors, multiplexing, fiber optics, and cathode-ray tube displays. Four channels are provided, with automatic testing of one channel performed while the remaining three provide two-of-three coincidence monitoring. The status of all sensors is communicated to the plant computer for diagnostic and display purposes.

- Control and safety systems are separated to increase reliability and to reduce risk of common-mode failures.

The APWR is estimated to achieve savings of 23% in uranium utilization, 30% in enrichment work, and 20% in overall fuel cost. Most of these savings are obtained through the spectral shift control feature.

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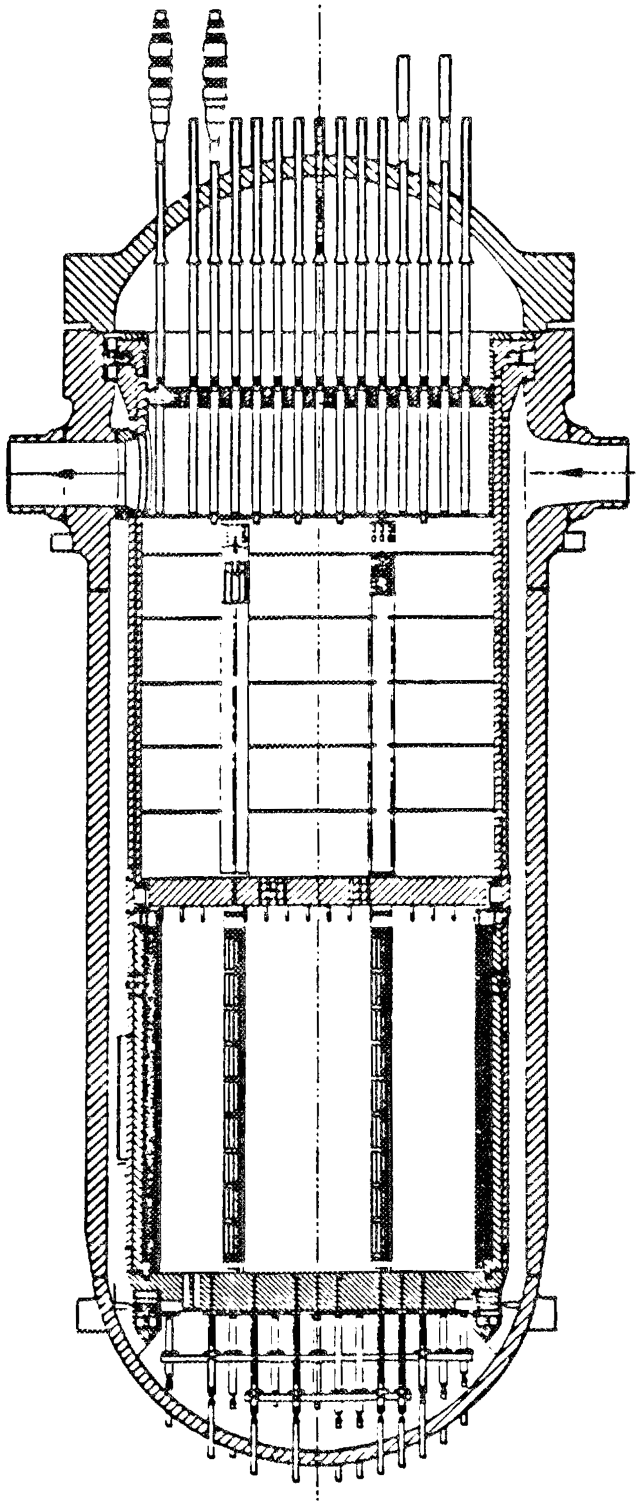

Fig. 2.2 The APWR vessel.

2.2.2 Commercial Status

The APWR development is supported through a seven-party contract signed (August 1982) by five Japanese utilities (headed by Kansai), Mitsubishi Heavy Industries (MHI), and Westinghouse (W). The $150 million development costs are shared by the utilities, the Japanese Government (MITI), MHI, and Westinghouse. Westinghouse and MHI are to develop a total plant design, including the verification testing of major components.

Design responsibilities are shared as follows:

W - Core, reactor vessel internals, reflector, displaced rod drives, refueling equipment, reactor pumps and coolant system, and transient/accident analysis.

MHI - Reactor vessel, pressurizer and piping, control room, electrical systems, and plant layout.

W/MHI - Design integration, steam generator, and fluid systems.

The preliminary design of the APWR is virtually complete. The intermediate design is scheduled to be completed by the end of 1985.

Testing of core, reactor, and steam generator components have been underway in Japan and in the U.S. These should be completed by 1987.

Westinghouse hopes for site selection in 1985, a firm order by 1986, and operation by 1993-94.

Westinghouse has initiated other activities intended to open U.S. and European markets to the APWR. This includes design review, with Westinghouse feedback, by a group of U.S. utilities, a group of Belgian utilities, and an architect-engineer. In addition, Westinghouse has initiated a licensing program with the NRC (Sect. 2.2.5). The intent of this effort is to establish an approved standard design in the U.S. The standard design is expected to be similar to the Japanese design except for reduced seismic level. Approximately 70% of the plant drawings would be available at the construction permit stage.

Westinghouse has offered firm-price contracts for about 80% of a plant abroad, and would be prepared to offer firm price and schedule in the U.S. for similar scope given a suitable licensing climate. Westinghouse can supply total project management, the nuclear steam supply system, safety-related balance-of-plant (BOP) systems, the control room with all associated systems, design and construction of the nuclear-related portions of the plant, assuming support from subcontract architect-engineer-constructors. Some U.S. firms are willing to provide firm-price subcontract services.

Westinghouse is also prepared to carry out a lesser role, depending on the preferences of the utility customer.

2.2.3 Cost Consideration

The capital cost target for APWR is a 15% reduction in capital costs [$/kW(e)] relative to earlier Japanese nuclear plants. This will be a difficult target since there are added costs due to a number of new safety features, described in Sect. 2.2.4. Features which contribute to a reduced capital cost include:

- simplified fluid systems designs,
- elimination of some systems, e.g., the boron systems may be eliminated or reduced in scope,
- elimination of safety-system interconnections,
- multiplexing the instruments and controls,
- a nonsafety start-up feedwater system,
- standard design supplying 70% of the drawings at the beginning of construction,
- benefits of scale of 1350-MW(e) plant relative to smaller current plants.

The design program has not been completed, but Westinghouse believes it is making progress toward the capital cost goal.

The fuel cost target for APWR is a 20% reduction from present cost levels. This appears to be achievable from the spectral shift control and the use of reflectors.

An ambitious plant availability target of 90% is to be achieved by the following means:

- refueling cycle extended to 18-24 months,
- refueling/maintenance outage reduced from 75 days (typical for Japan) to 45 days through extensive automation of fuel handling, fuel inspection, and steam generator inspection,
- improved steam generators,
- more rugged fuel assemblies,
- full-load rejection capability (further reducing the low Japanese occurrence of trips),
- on-line testing and calibration of instruments.

2.2.4 Safety Considerations

The overall safety philosophy of the APWR is similar to that of earlier four-loop Westinghouse designs such as SNUPPS but the details have been substantially strengthened, as follows:

- The increased volume of primary coolant in the reactor vessel above the core increases the time available to deal with loss of coolant.

- Lower core power density increases safety margins.

- There are four complete trains of mechanical equipment in the safeguard system.

- A large emergency water storage tank is provided inside containment as the water source for four safety injection pumps. This storage tank automatically gets steam generator tube rupture flows.

- Containment sumps are kept filled with water to increase available heat capacity.

- Safety and control systems are separated to increase reliability and reduce common-mode failures.

- Four separate and hardened compartments are provided to house high- and low-pressure safety injection pumps. This feature reduces the likelihood of radioactivity release to the atmosphere and makes sabotage of the safety systems very difficult.

- The control room is improved, with improved diagnostic capabilities.

- The larger pressurizer and core provide for improved response to transients.

- The large dry containment vessel is conservatively designed.

- The steam generator secondary side water inventory is controlled automatically.

- There is injection of pressurized water to reactor coolant pump seals.

- The reactor vessel neutron fluence is reduced.

- The overall improvements in plant availability, reliability, and maintainability translate into improved safety.

The improved safety protects not only the public but also the utility investor.

Westinghouse has performed a comparative PRA of the APWR and a conventional PWR for internal events. The results are shown in Fig. 2.3. The internal risk in the APWR appears to be dominated by the steam-generator tube break accident, itself at the very low level of 10^{-7} core melts/reactor year. The reported total risk from internal events of less than 2×10^{-7}/reactor year is well below Westinghouse's target of 1×10^{-6}/reactor year overall risk from the APWR. External event analysis, which is site specific, will be carried out later.

The APWR has an ambitious target of 100 man-rem/year, well below the levels experienced at Japanese PWR plants in the past (Fig. 2.4) and even farther below U.S. experience (Fig. 2.5). The plant features to achieve this target are as follows:

- fewer refueling/maintenance outages,
- use of low-cobalt materials, especially in steam generator tubing,
- more reliable equipment,
- greater use of automation in inspection and steam generator tube repair,
- better plant layout, greater accessibility for maintenance and use of shielding.

In an earlier version, the APWR had a dedicated, passive steam condenser to provide decay heat removal and to deal with steam generator tube ruptures. This has now been replaced with the in-containment emergency water storage tank and an emergency feedwater system. It was stated that the deficiencies of the passive steam condenser relative to the present design were:

- It was slightly more expensive.
- It reduced risk by a less-than-expected factor, especially for steam-line breaks.
- It lacked diversity.
- U.S. utilities were negative about it.
- The ACRS supported the passive system but considered the active system acceptable.

2.2.5 Licensing Considerations

Westinghouse is pursuing the licensing objectives of preliminary design approval by NRC for the APWR by 1986 and final design approval by 1988 through rulemaking. At that point a standard "nuclear power block" design would be preapproved, together with complete specification of residual safety interface requirements.

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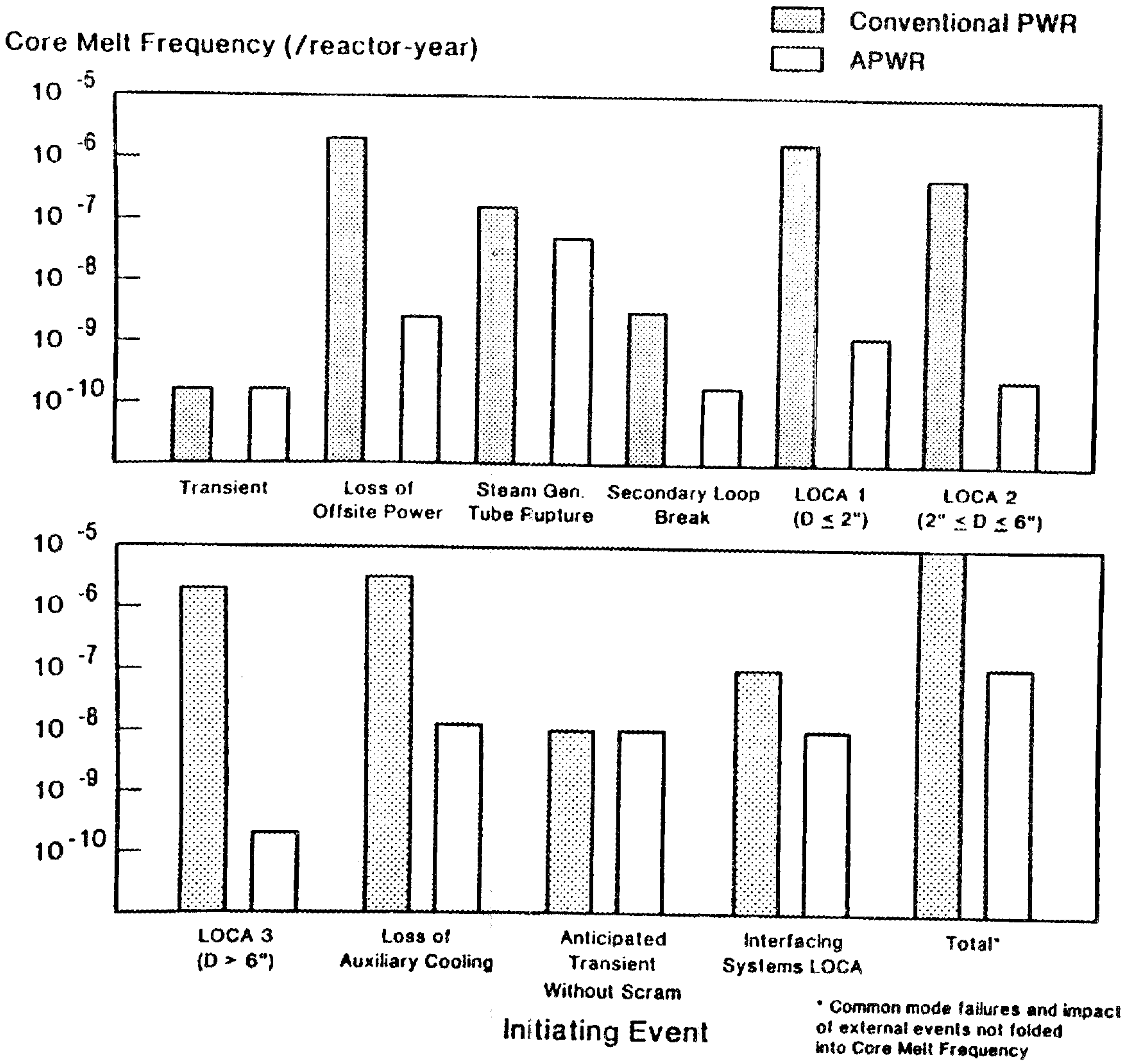

Fig. 2.3 Core melt frequency, conventional PWR and advanced PWR.

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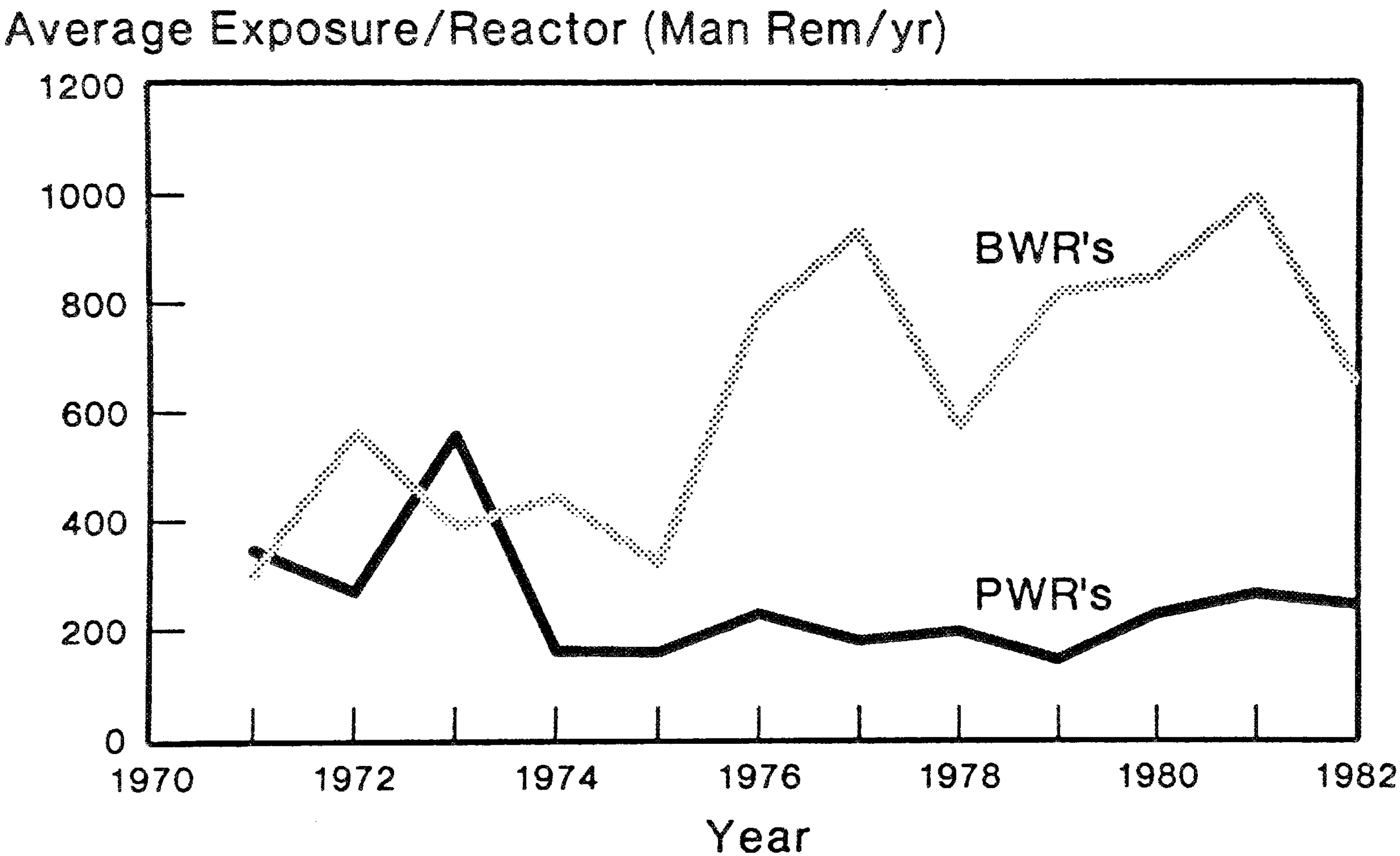

Fig. 2.4 Estimate of radiation exposure in Japanese reactors.

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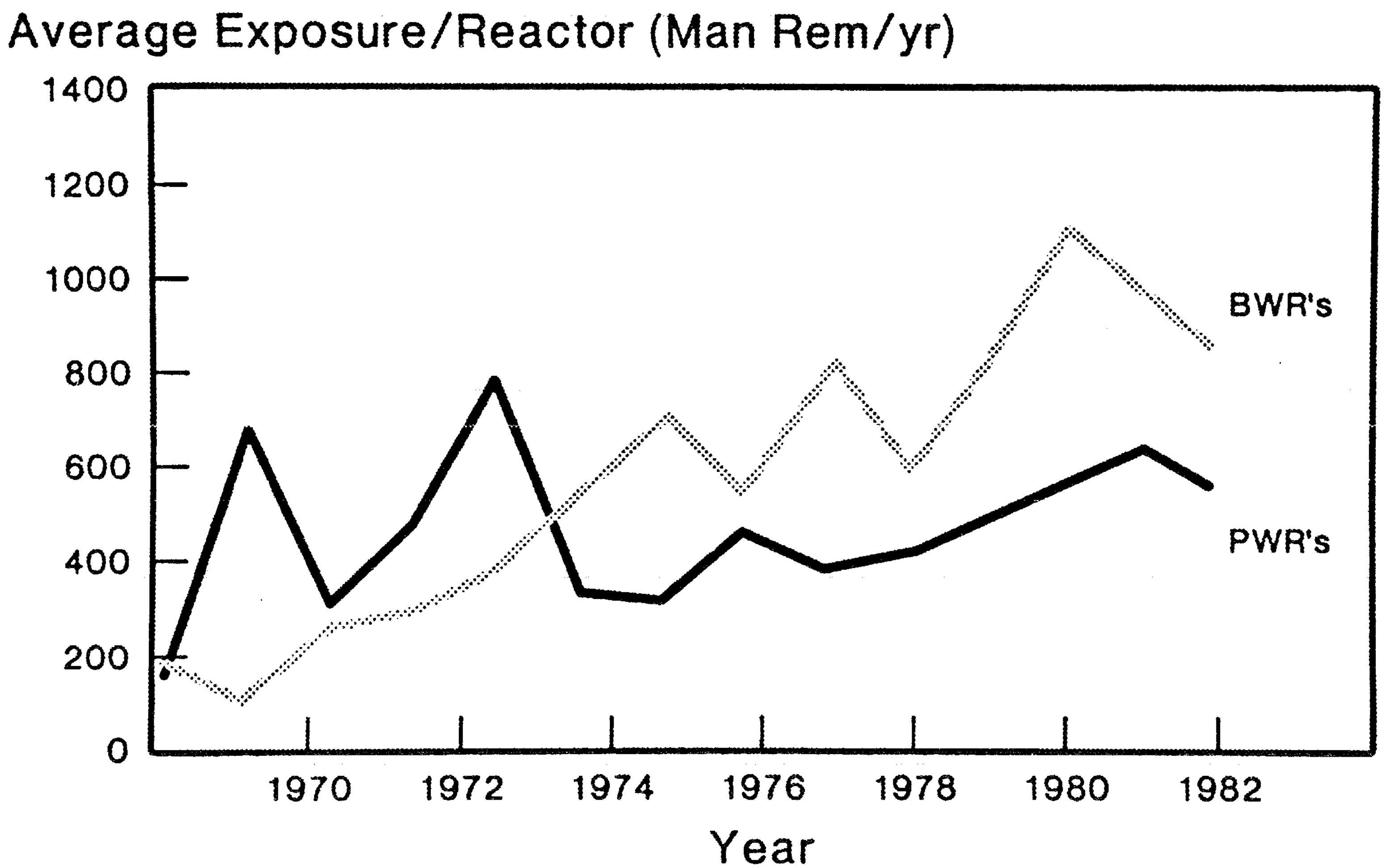

Fig. 2.5 Estimate of radiation exposure in U.S. PWRs and BWRs.

Currently, the NRC staff and the ACRS are involved reviewing "modules" of the APWR. These modules (of which 16 will constitute the complete nuclear power block) address interface requirements in each case. The sum of the 16 modules, with some connecting language, will constitute a Preliminary Safety Analysis Report.

In total, Westinghouse will address all applicable NRC regulations, TMI issues, unresolved safety issues, generic issues, the proposed severe accident policy, and will include a PRA. In the event of a firm plant order, the detailed plant design could result in a Final Safety Analysis Report and a NRC-approved standard design.

2.3 THE TWO-LOOP BARGE-MOUNTED PLANT

The Westinghouse two-loop plants of 500- to 600-MW(e) rating have had exceptionally good performance relative to other Westinghouse and other LWR plants generally. Both capital cost and overall power cost experience have been favorable relative to experience with larger plants, in apparent defiance of scaling laws. (One factor responsible for the good performance may be the coincidence that these plants were purchased by utilities that have displayed high-quality management and operation.) Westinghouse has recently designed and built a two-loop plant in the Philippines and, therefore, has an up-to-date two-loop design.

Westinghouse has carried out studies of barge-mounted two-loop plants, most recently for EPRI. An earlier version placed the nuclear and turbine plants on two separate barges which could be floated to the site and permanently emplaced in horizontal tunnels carved out of a hillside. The current version has only the nuclear plant on the barge (Figs. 2.6 and 2.7). The overall size of the facility is comparable to that of a merchant vessel.

The objectives of this approach are low cost, short construction schedule (four years), factory quality assurance (at a shipyard), and low risk to the buyers. The plants would be built to a prelicensed standard design. The nuclear island would be factory made and transported to the site for permanent emplacement. The balance of plant would be built on site.

Preliminary studies indicate the concept would be competitive in cost with large nuclear plants, and operable by the mid-1990s (Fig. 2.8).

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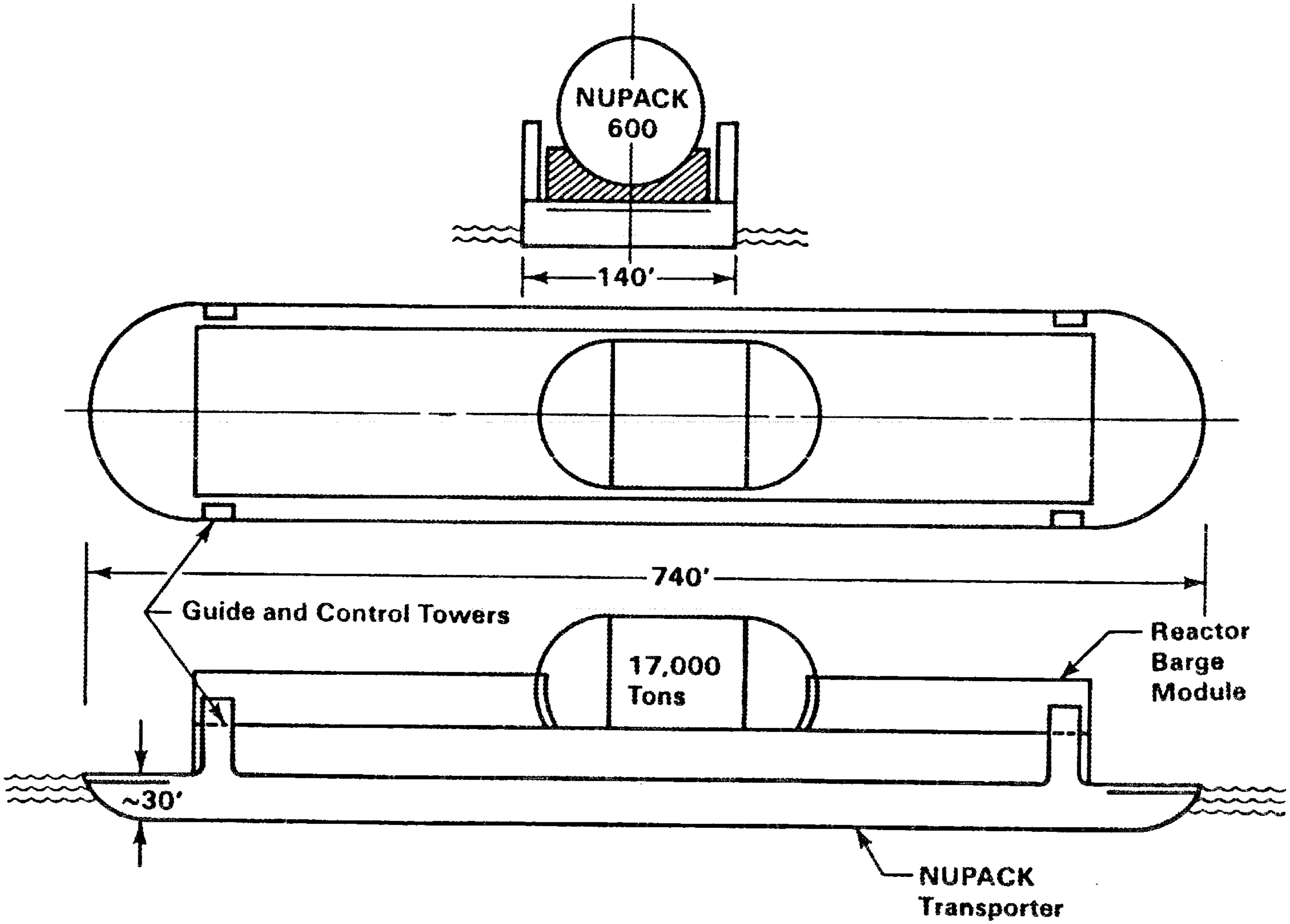

Fig. 2.6 Submergible Barge NUPACK Transporter

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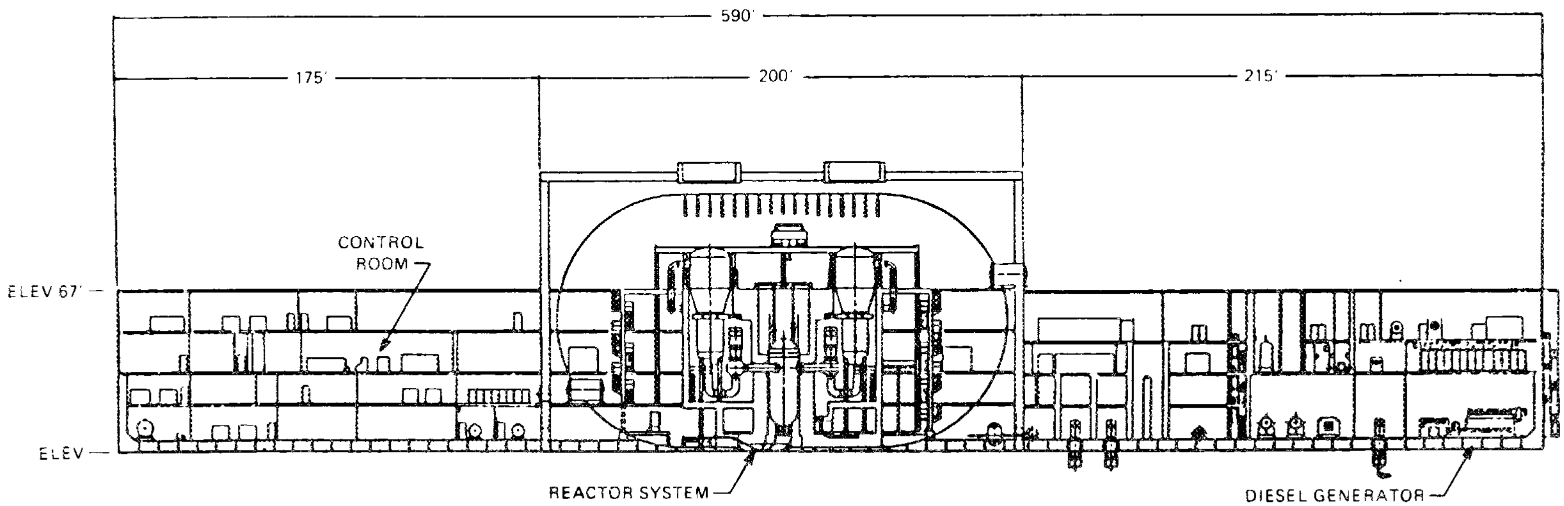

Fig. 2.7 NUPACK 600 reactor barge module.

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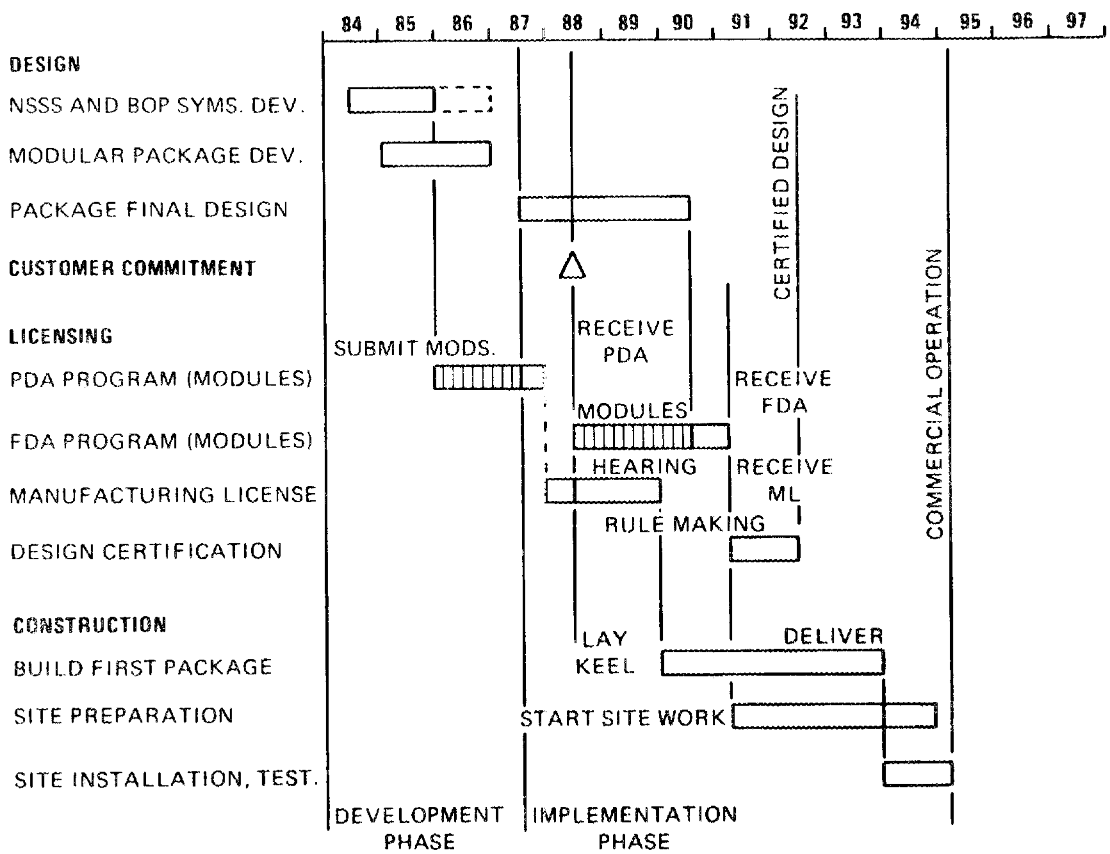

Fig. 2.8 First modularized plant schedule - commercial prototype.

3. ADVANCED PWR DESIGNS OF COMBUSTION ENGINEERING, INC.

The product which C-E currently offers is the System-80 nuclear steam supply system (NSSS). Designs of 3800 and 2800 MW(t) are available. The first System-80 plant to be completed will be Palo Verde which contains three 3800 MW(t) [1270-MW(e)] units. The System-80 design has received final design approval from the NRC via the C-E Standard Safety Analysis Report (CESSAR). System-80 is described in Sect. 3.1.

C-E's efforts on future plants are concentrated on upgrading certain aspects of the System-80 design to increase reliability, decrease costs, and provide greater assurance of safety. To a certain extent, modifications may be required to satisfy new NRC regulations, but the primary driving force for changes would be innovation based on construction and operating experience. The further evolution of the System-80 design is described in Sect. 3.2.

3.1 THE SYSTEM-80 STANDARD PLANT

3.1.1 General Description

The large System-80 plant is a PWR rated at 3800 MW(t). The design has evolved from earlier C-E designs, and contains two steam generators and four primary pumps (Fig. 3.1). The standard CESSAR plant can be accommodated in a cylindrical containment, as at Palo Verde, or in a spherical shell, as at the canceled TVA Yellow Creak plant (Fig. 3.2). The advantages of a spherical containment were described in Sect. 2.2.1.

The System-80 core utilizes the increased area of the 16 × 16 fuel assembly. Parameters are listed in Table 3.1. Substantially more control element fingers are provided to increase the flexibility of control during operation.

The size of the pressurizer is large relative to earlier designs to permit the system to respond to loss of electrical load without excessive primary system pressure. The System-80 plant does not require power-operated relief valves such as the one which failed to close during the TMI-2 accident.

The System-80 offering includes a four-channel reactor protection system. A two-of-four coincidence is required to actuate safety systems. When one channel is taken off line for testing, the remaining channels provide two-of-three coincidence. System-80 also contains both a plant computer and a core-monitoring computer.

One option offered with System-80 is the Nuplex-80 control center. This design, included as part of the Yellow Creek plant design, makes use of solid-state components, cathode-ray tube displays, diagnostics to assist the operator, and signal multiplexing.

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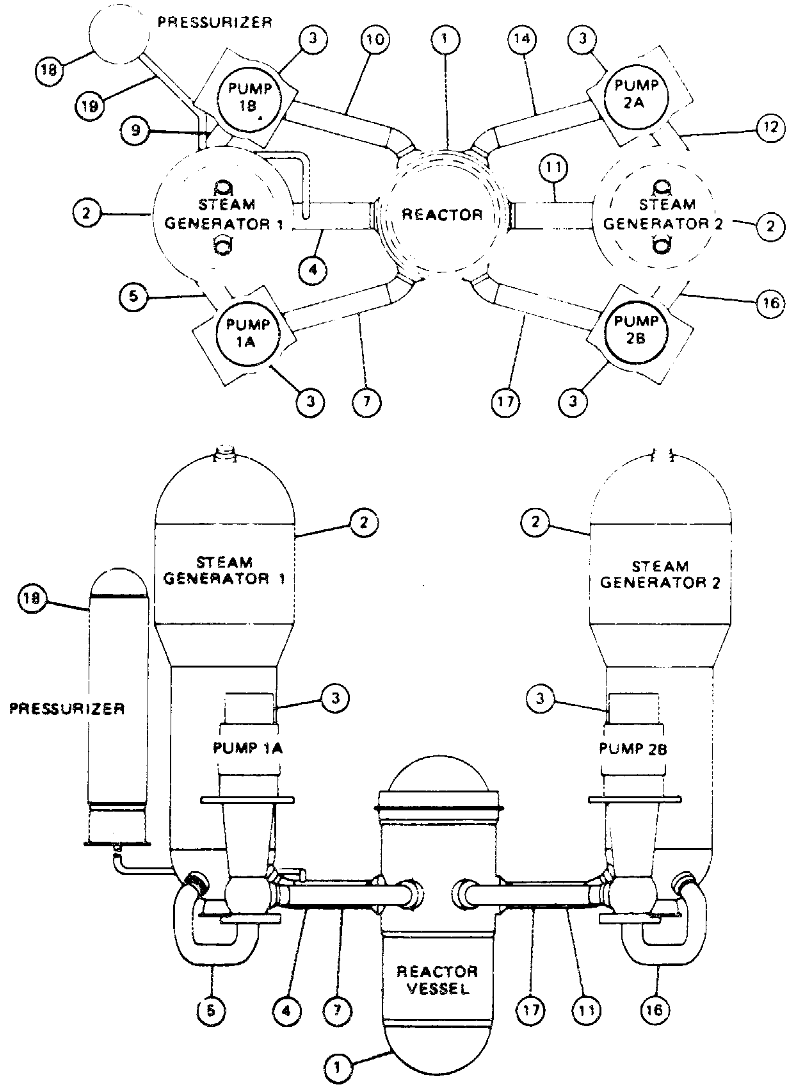

BILL OF MATERIAL		
ITEM No.	QUAN.	NAME
1	1	REACTOR VESSEL ASSY'
2	2	STEAM GENERATOR
3	4	REACTOR COOLANT PUMP
4	1	STEAM GENERATOR No. 1 INLET PIPING
5	1	PUMP 1A SUCTION PIPING
7	1	PUMP 1A DISCHARGE PIPING
9	1	PUMP 1B SUCTION PIPING
10	1	PUMP 1B DISCHARGE PIPING
11	1	STEAM GENERATOR No. 2 INLET PIPING
12	1	PUMP 2A SUCTION PIPING
14	1	PUMP 2A DISCHARGE PIPING
16	1	PUMP 2B SUCTION PIPING
17	1	PUMP 2B DISCHARGE PIPING
18	1	PRESSURIZER
19	1	PRESSURIZER SURGE LINE

Fig. 3.1 System-80 primary system arrangement.

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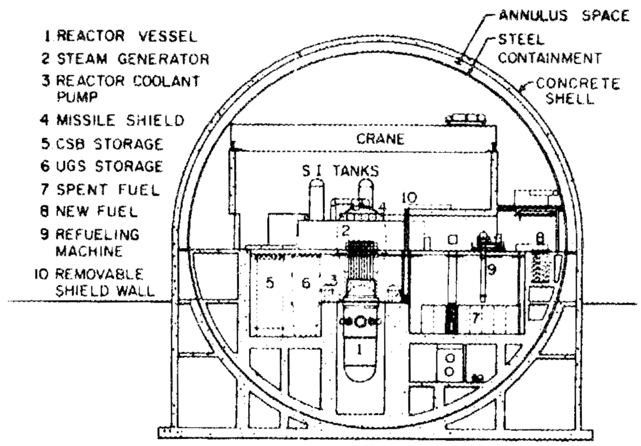

Fig. 3.2 System-80 spherical containment.

Table 3.1. Comparison of 3817 MW(t) system-80 NSSS with previous 2700 MW(t) NSSS

	Calvert Cliffs	System-80
Core power, MW(t)	2,700	3,800
Number of fuel assemblies	217	241
Active length of core,		
in.	136.7	150
(m)	(3.472)	(3.810)
Power density, kw/L	82.8	95.9
Fuel assembly dimensions,		
in.	7.98 × 7.98	7.98 × 7.98
(mm)	(202.7 × 202.7)	(202.7 × 202.7)
Number of fuel rods per assembly	176(14 × 14)	236(16 × 16)
Fuel rod OD,		
in.	0.440	0.382
(mm)	(11.2)	(9.7)
Clad thickness to OD rate	6.4×10^{-3}	6.5×10^{-3}
Maximum linear power density		
kw/ft	15.5	12.5
(kw/m)	(50.8)	(41.0)
Average		
kw/ft	6.2	5.2
(kw/m)	(20.8)	(17.06)
H_2O/UO_2 volume ratio	1.96	2.02
Specific power, kw/kgU	32.4	37.0
Number of control rod drives	77	89
Number of control element fingers	385	708
Primary system pressure,		
psia	2,250	2,250
(bars)	(155)	(155)
Reactor coolant average temperature,		
F	572	594
(C)	(300)	(312)
Coolant flow rate,		
gals/min	370,000	445,600
m^3/h	(84,035)	(101,205)
Maximum core heat flux,		
Btu/h ft^3	457,600	425,400
(W/cm^3	(144)	(134)
Minimum DNBR	1.82	2.13
Secondary steam pressure		
psia	850	1,070
(bars)	(59)	(74)

3.1.2 Commercial Considerations

The System-80 design has been offered commercially for about 10 years. However, most of the plants ordered have subsequently been canceled. The three Palo Verde units should provide an operating base to support future orders.

The CESSAR standard plant design has final design approval from the NRC staff and, once the NRC has completed its Severe Accident Policy, C-E will probably request rulemaking by the Commission. At lease three alternate standard BOPs are available to support CESSAR. Purchase of a standard plant is expected to contribute to a predictable construction schedule and reduced engineering and construction costs.

C-E has offered a firm-price bid for a plant in Taiwan with a supporting architect-engineer and is pursuing other foreign business. C-E is considering a much greater scope of supply than in the past, including feedwater systems, condensers, cooling systems, and containment (all safety-related systems). There is a realization that the integrated plant must be carefully designed and constructed to maximize reliability and maintainability - and thereby achieve safety and reasonable costs.

The largest improvements in capital cost would be achieved through high-quality management during construction and licensing stability to preclude design changes. This includes use of standard designs, application of new computer tools, and scheduling, with a strong focus on minimizing the schedule. Scheduling delays are automatically translated into higher labor costs for the on-site labor force.

C-E plants have operated with higher availability, on the average, than other U.S. light-water reactors. The vendor attributes this to design conservatism and ease of maintenance. C-E believes that a 10-point improvement in availability is achievable in their future plants through better design integration and improved materials selection.

3.1.3 Safety Considerations

The principal safety functions within the System-80 design are as follows:

- reactivity shutdown,
- emergency water supply (safety injection),
- residual heat removal system.

The reactivity shutdown is actuated by the Reactor Protection System which incorporates the core protection computer that determines whether there is insufficient local departure from nucleate boiling margin or excessive local power density. There are two groups of shutdown rods, either of which can reduce the reactor power to zero. In addition, the reactor can be shut down by injecting boron.

The emergency water supply (to cope with loss-of-coolant events) consists of two fully redundant injection trains, each consisting of two pressurized (with 600 psi nitrogen) safety injection tanks located in containment, an external water storage tank, a high-pressure pump, and a low-pressure pump. The pump suctions are automatically switched over to the containment sumps after the external storage tank is drained.

Decay heat removal is initially provided by the steam generators. Long-term residual heat removal is accomplished with the shutdown cooling system following reduction of primary system temperature and pressure. Low-pressure safety injection pumps circulate water to heat exchangers which, in turn, reject heat to the ultimate heat sink.

C-E believes that the System-80 design has pioneered many features which contribute to the safety of the plant. These features are now being offered by other vendors as well, either as backfits or incorporated into new designs. These features include:

- greater core thermal margin,

- large pressurizer volume to absorb loss of electrical load,

- improved secondary-side steam generator materials (stainless steels),

- use of two-of-four coincidence to actuate safety systems with one channel available for off-line testing,

- use of a core-monitoring computer to continually monitor core thermal-hydraulic parameters,

- advanced control room.

C-E uses PRA techniques in some cases to evaluate and compare alternatives for a given plant. They do not have a complete PRA for a System-80 plant at this time.

3.2 FURTHER EVOLUTION OF THE SYSTEM-80 DESIGN

For the 1990s, C-E plans to offer an improved version of System-80 with the following features likely to be included:

- an integrated NSSS/BOP, with the reactor vendor being responsible for a much larger portion of the plant than at present.

- simplification, to the extent possible,

- emphasis on reliability/maintainability,

- higher quality steam generators; improved steam generator materials,

- higher quality heat exchangers and condensers to avoid ingress of contaminants to the steam generators,

- upgraded control room and instrumentation,

- fewer pipe supports (seismic and pipe-whip criteria),

- optional full-pressure decay heat removal system,

- reactor pressure vessel design that greatly reduces impact of neutron fluence,

- improved feedwater systems and control systems,

- design to avoid spurious trips,

- fully replaceable major equipment.

C-E officials believe it would be desirable, and probably cost-effective in a broad sense, to provide conservative features addressing public concerns even if current features are believed to be safe enough. For example, the following should be considered:

- lower power density to provide greater safety margin,

- larger coolant inventory-to-power ratios to provide a more forgiving transient response,

- design shutdown systems to avoid ATWS,

- improved emergency power systems.

Supply of conservative systems to accomplish the needed safety functions should allow simplification or elimination of more complex systems now used to accomplish those functions. The improved features will pay for themselves, in C-E's view, through improved plant reliability and maintainability, which correlate with safety. However, there is a competitive problem in that various vendors may offer less conservative plants at more competitive prices. C-E officials believe that it would be desirable for utilities to recognize and support conservative (but hopefully simplified) designs that are likely to lead to lower costs in the long run.

4. ADVANCED BWR DESIGNS OF THE GENERAL ELECTRIC COMPANY

The product which GE currently offers is the BWR/6 with Mark III containment. Designs are available in the range 600 to 1300 MW(e), but most plants are at the upper end of the range; for example, Grand Gulf is listed at 1250 MW(e) and Kuosheng 1 and 2 at 950 MW(e). The GE Standard Safety Analysis Report (GESSAR) standard plant is a BWR/6-Mark III (Sect. 4.1).

GE is developing a more advanced design (ABWR) in cooperation with its Japanese licensees and the Tokyo Electric Power Company. The ABWR is a 1350-MW(e) design. This report presents data on the ABWR which has been made available (Sect. 4.2).

GE is exploring a lower power BWR concept that is intended to make use of passive safety systems. This concept is also described here (Sect. 4.3).

4.1 THE GESSAR STANDARD PLANT

As stated above, the GESSAR is a BWR/6-Mark III plant. It is a direct-cycle boiling-water reactor with steam generated in the core conveyed directly to the turbine. Figure 4.1 displays the reactor assembly.

A principal design feature of BWRs is the existence of a natural circulation flow path within the reactor vessel so that there is adequate coolant flow capacity for removing the afterheat from the core by natural thermal convection as long as the water inventory in the reactor vessel is maintained at the proper level. Further, the BWR primary system can be depressurized rapidly; low-pressure as well as high-pressure pumps can, therefore, be employed to maintain the proper water level in the reactor vessel. There are a total of 13 pumps in the BWR/6, 11 of which can individually handle nonbreak transients.

The Mark III containment (Fig. 4.2) is the latest version of the pressure suppression containment used on all but the very earliest BWRs. Pressure suppression containment systems employ a large pool of water (the suppression pool) and a system of vents leading from the reactor cavity to the suppression pool. Following a postulated reactor primary system rupture, steam and fission products released from the reactor will be channeled to the suppression pool. Experiments show that the bubbling of these materials through the pool should remove over 99% of the iodine and particulate fission products from the vented gases. Thus, the vast majority of the radioactive fission products that might escape from the core in the event that severe core damage did occur would be retained within the primary containment system. The Mark III containment building has been enlarged and strengthened relative to earlier designs, and potential bypassing of the suppression pool has been eliminated.

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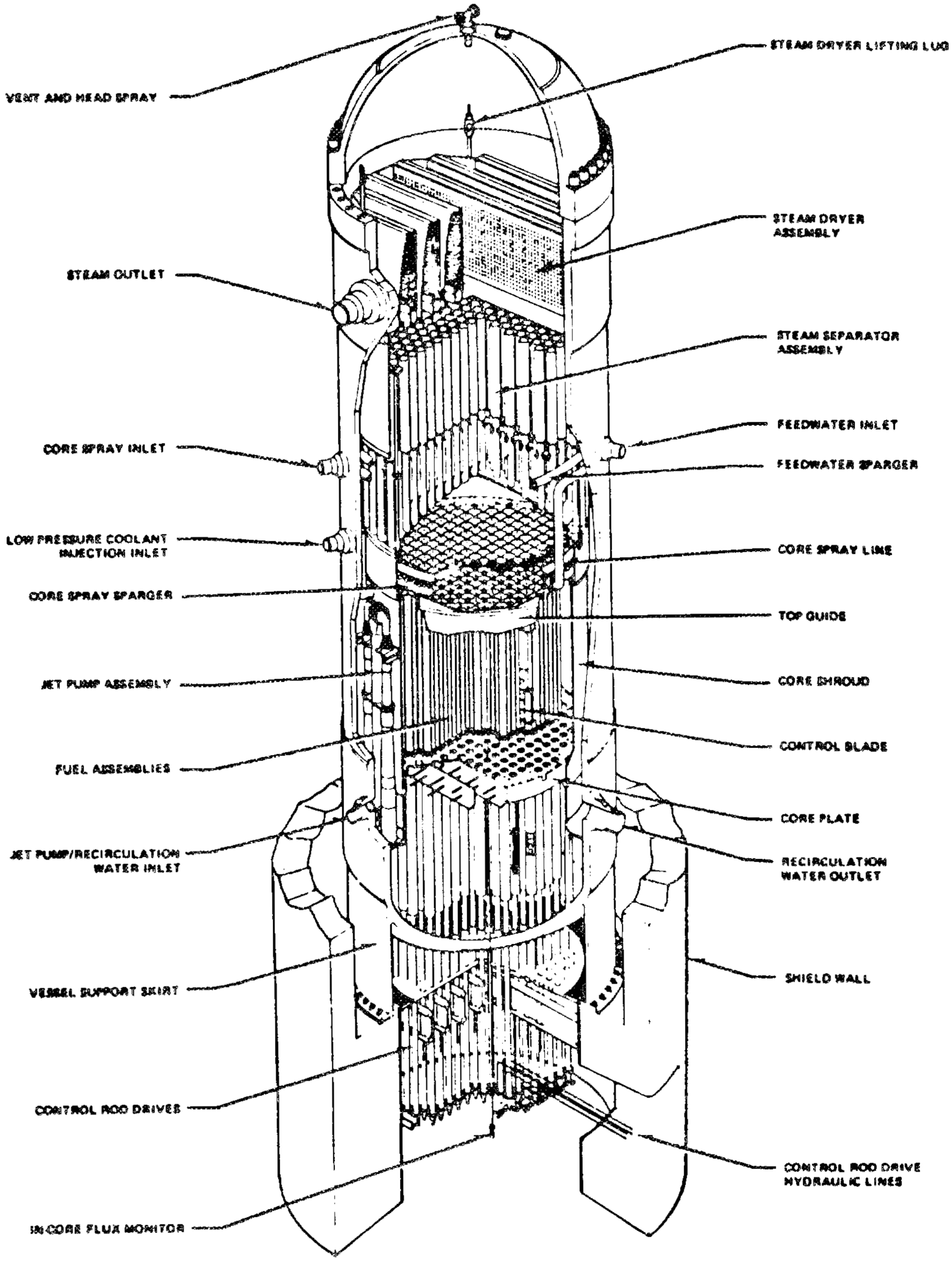

Fig. 4.1 BWR/6 reactor assembly.

Source: General Electric Company, BWR/6 General Description of a Boiling Water Reactor (Rev.) (San Jose, California, 1980), p. 2.2 (reproduced with permission).

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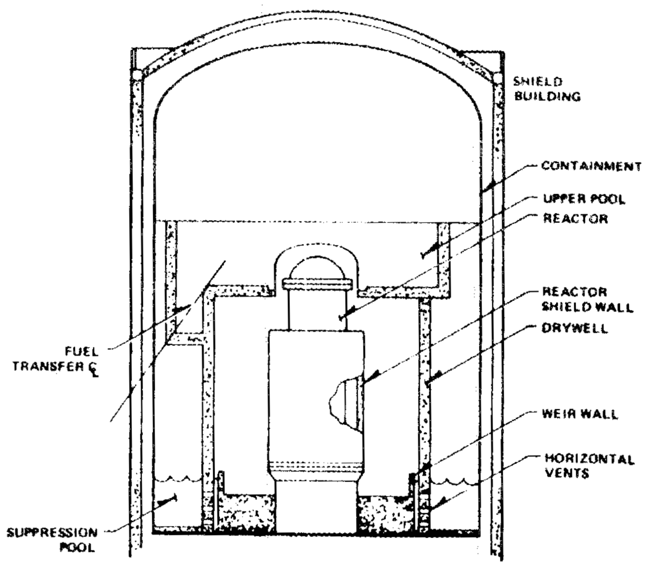

Fig. 4.2 Mark-III containment schematic.

The general strategy for dealing with a severe accident challenge in a BWR is as follows:

- First, to insert the safety rods and to confirm that has occurred.

- Second, to maintain water level in the vessel either with high-pressure sources or, alternatively, by depressurizing to the suppression pool via relief valves and using low-pressure pumps.

- Third, to establish a heat sink either in the main condenser or, if necessary, using the suppression pool. In principle, steam from the suppression pool could be vented to atmosphere and makeup water supplied from a fire truck to the reactor vessel.

Operating BWRs have suffered from unavailability due to fuel failures and, more recently, due to recirculating pipe cracking. At this point, it appears that these problems have been overcome. Redesigned BWR fuel has much greater tolerance to load changes and long burnup. Pipe cracking is avoided by use of nuclear-grade (low-carbon) stainless steel, careful control of weld stresses, and optional hydrogen addition to the primary coolant.

The GESSAR application submitted by GE to NRC is intended to satisfy 10CFR50 criteria, post-TMI modifications, and to address the unresolved safety issues. The post-TMI modifications are as follows:

- improved emergency procedure guidelines - to provide plant operators with concise procedures to follow during an emergency,

- improved safety/relief valve position indication - to provide a faster, more direct indication of an open valve to the plant operator for appropriate action,

- improved post-accident sampling capability - provisions for obtaining a post-accident "grab sample" of reactor water and containment air at an accessible location to facilitate assessments of core damage,

- improved containment instrumentation - to monitor containment pressure, radiation level, and suppression pool water level following an accident,

- improved effluent monitors - to provide capability to monitor plant effluents over full range from normal to accident conditions,

- control room improvements - to provide an improved man-machine interface in the control room and facilities for responding to an emergency. Examples of improvements include:

(1) a Plant Safety Parameter Display to provide key plant safety parameters in a clear, concise format; (2) an on-site Emergency Response Center for coordination of emergency response; and (3) a Nuclear Data Link to provide key plant safety information to the NRC,

- auto-restart of high-pressure core spray - to provide automatic restart at low reactor water level in the event the operator takes manual control of the system and subsequently fails to maintain water level,

- auto-depressurization for nonbreak events - to provide automatic depressurization logic for nonbreak events (e.g., loss of feedwater) accompanied by failure of all high-pressure cooling systems.

The GESSAR application has been approved by the NRC staff and is awaiting approval by the Commission.

GE has performed a PRA in support of the GESSAR program (see Table 4.1), with a reported core-melt probability from internal sources of 4.7×10^{-6}/reactor year. The risk is dominated by loss of off-site power events.

Table 4.1. BWR/6 PRA results: Breakdown of the assessed frequency of core damage per reactor year

Event description	Frequency of core damage per reactor year	Percent of core damage probability
Transients		98.0
- Loss of off-site power	4.1×10^{-6}	(88)
- All others	5×10^{-7}	(10)
Loss of heat removal	2×10^{-8}	0.4
ATWS	5×10^{-8}	1.1
LOCA	2×10^{-9}	0.04
TOTAL	4.7×10^{-6}	

4.2 THE ADVANCED BOILING WATER REACTOR (ABWR)

4.2.1 General Description

The stated objectives of the ABWR are:

- improved operability,

- improved capacity factor,

- improved safety and reliability,

- reduced occupational exposure,

- reduced capital and operating costs,

- maximum use of technology demonstrated on U.S., Japanese, Swedish, and German BWRs.

The most salient innovation in the ABWR is the replacement of the external recirculating pumps and loops by 10 internal sealless coolant pumps (Fig. 4.3). KWU and ASEA-Atom have utilized internal pump recirculation systems in their recent BWRs. Full power of 1350 MW(e) can be achieved with 9 of 10 pumps. Advantages claimed for this approach include:

- elimination of all major pipe nozzles in core region and below core,

- elimination of large break loss-of-coolant accident concerns,

- shorter construction schedule,

- more space in drywell,

- less maintenance, less in-service inspection, and reduced occupational exposure,

- reduced recirculation system pumping power,

- reduced plant cost,

- improved availability/reliability.

The ABWR has provisions for varying the recirculating water flow rate to control the neutron spectrum (hydraulic spectral shift). It is expected that the fuel burnup will be 38,000 MWD/tonne. By these and other means, the fuel utilization can be improved sufficiently to reduce fuel-cycle costs by about 20% relative to present Japanese practice.

It is also proposed to improve the load-following characteristics of the BWR. The fuel has been improved over earlier designs so that it appears that power reductions from 100 to 70% will be possible through flow controls only and as much as 50% may be possible using both flow control and control rods. Fine motion control rod drives contribute to maneuverability.

Table 4.2 summarizes the key changes in going from the GESSAR to the ABWR designs. The ABWR instrumentation and control system has four independent channels, any one of which can be calibrated automatically while the remaining channels retain a two-out-of-three coincidence feature. The ABWR containment design has not been frozen as yet; a likely option is a modified Mark II-type. The containment would retain the feature of venting the drywell through the pool, characteristic of the Mark III design.

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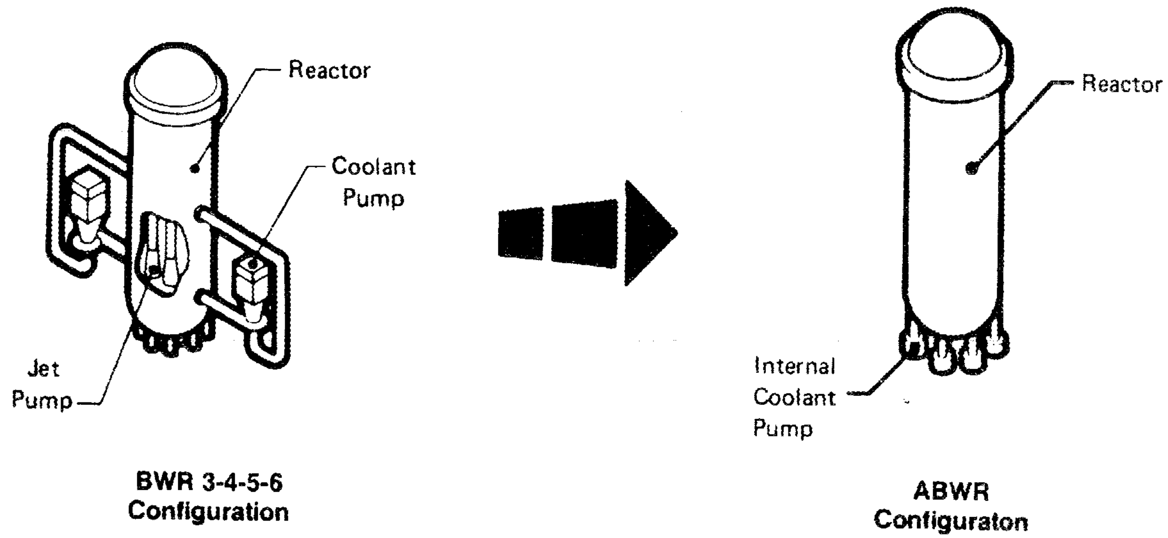

Fig. 4.3 BWR recirculation configurations.

Table 4.2. Key differences between ABWR and GESSAR designs

Plant feature	GESSAR	ABWR
Recirculation system	External pumps Flow control valve	Internal pumps Variable-speed, solid-state power supply
Control rod drives	Hydraulic	Electric/hydraulic fine motion
Emergency core cooling	Three divisions 1 high-pressure spray 1 low-pressure spray 3 low-pressure flooders 1 steam-driven reactor-core isolation cooling system	Three completely separate divisions 2 high-pressure sprays 1 steam-driven reactor core isolation cooling system 3 low-pressure flooders
Core spray sparger	Peripheral ring	Overhead
Decay heat removal	2 steam-condensing heat exchangers	3 wetwell/drywell heat exchangers
Control of reactor flow, feedwater and pressure	Analog	Digital
Transmission of control and safety signals	Wires	Multiplexed
Containment	Horizontal vents Steel Open pool Air	Vertical vents Concrete Covered pool Inerted
Steam bypass capacity	35%	25 to 35%
Fuel transfer	Inclined tube	Cask lift

4.2.2 Commercial Status

The BWR/5-Mark II is the current standard BWR design in Japan. Its features and safety characteristics have been extensively tested. It is expected that the ABWR will be selected as the next Japanese standard design. The ABWR project is a joint effort of GE, Toshiba, and Hitachi. It is sponsored by six Japanese utilities under the leadership of Tokyo Electric Power Co. MITI is doing some independent testing and evaluation of key concepts.

Phase I of the ABWR was a plant definition/feasibility study undertaken by the worldwide BWR suppliers in 1978-79. Phase II consisted of more detailed technical evaluations, component testing, and development. Technical evaluations were completed in 1983. While many of the test programs were also completed in 1983, some will continue into 1987. Phase III is now underway and consists of design optimization to be completed June 1985. Given a favorable project decision by the utilities (1986), licensing would proceed from 1986 to 1989, a construction permit obtained by 1989, and the lead plant would be in operation by 1994.

GE is designing the ABWR to meet most current U.S. requirements. Discussions with the NRC are planned to introduce the ABWR design. A favorable response is expected since most features are based on proven BWR technology.

It is too early to speculate on whether GE will offer firm-price bids on the ABWR and precisely what would constitute the scope of supply. Presumably the GESSAR approach would be representative of what might be expected.

4.2.3 Cost Considerations

The capital cost target for ABWR is to limit capital cost of the 1350-MW(e) plant to not exceed that of an 1100-MW(e)BWR/5-type plant built in Japan. Features which contribute to a reduced capital cost include:

- elimination of the recirculation piping,

- multiplexing of control and instrument cables,

- less crowded containment, allowing rapid construction schedule.

The design program has not been completed but GE expresses confidence that the cost targets will be achieved.

The fuel cost target for ABWR is a reduction of 20% from current BWR fuel costs. This would be achieved through extended burnup and hydraulic spectral shift control.

GE believes that the Japanese should be able to achieve plant availabilities of over 80% from the ABWR. Factors which contribute to high availability include:

- 18- to 24-month refueling cycle,
- on-line testing and calibration of instruments,
- elimination of inspection and maintenance of the recirculation piping,
- reduced refueling/maintenance outage,
- less crowded containment, facilitating maintenance,
- capability for achieving full power in two hours from hot standby (versus eight hours in present plants).

4.2.4 Safety Considerations

The overall safety philosophy of the ABWR is similar to that of the GESSAR. Those features listed in Table 4.2 and in the last paragraph of the previous section generally contribute to enhanced safety and reduced risk. The ABWR has greater diversity of safety systems and more complete separation from operational systems compared to GESSAR. Each depressurization system valve is actuated by a dedicated air storage tank. The suppression pool has sufficient heat capacity to accommodate up to 24 hours of decay heat from the reactor. There are sufficient on-site steam and power sources to keep the core covered for 8- to 10 hours after a station blackout accident, giving the operations staff that time to restore feedwater flow.

GE is considering a procedural change whereby venting of steam from the suppression pool to the atmosphere would be done at the discretion of the operators to avoid excessive pressure buildup in containment during certain accident sequences.

GE's summary of the factors contributing to reduced risk is given in Table 4.3. They have performed a comparative PRA of ABWR versus BWR/6 for internal events. According to this analysis, the ABWR risk of severe accident from internal events is about a factor of 10 better than GESSAR's.

The ABWR should be improved relative to GESSAR in the area of personnel radiation exposure as well. Factors reducing exposure include:

- less frequent refueling,
- elimination of recirculation piping maintenance and inspection as a source of radiation,
- better layout for maintenance,
- more automated servicing.

Table 4.3. Factors contributing to reduced ABWR risk

Improved core cooling:
- More high-pressure coolant pumps
- Motor-driven and turbine-driven pumps
- Turbine-driven feedwater pumps

Lower initiating event frequency
- Fault-tolerant digital controllers
- Solid-state control logic

Reduced LOCA probability
- Elimination of recirculation piping and pump

Improved scram system reliability
- Diverse system (electrical + hydraulic drives)
- Scram discharge volume elimination

Improved heat removal capability
- More pumps and heat exchangers

Others
- Improved diesel generator* spatial separation
- Suppression pool feature retained

*Author's note: About 5 MW per redundant set of safety systems is required to drive core-cooling and emergency auxiliaries.

4.3 THE LOWER POWER BWR

GE has performed some preliminary design of a small BWR concept, 200- to 600-MW(e). The objectives of this study are to evaluate a system which would

- accomplish safety functions in a simpler manner than BWR/6,
- be more tolerant of transients,
- require no operator action for a long period after a transient,
- address severe accident possibilities,
- have high availability,
- utilize existing technology base insofar as possible.

The concept is illustrated in Fig. 4.4. A diagram of the vessel is shown in Fig. 4.5. High-pressure recirculation of water through the

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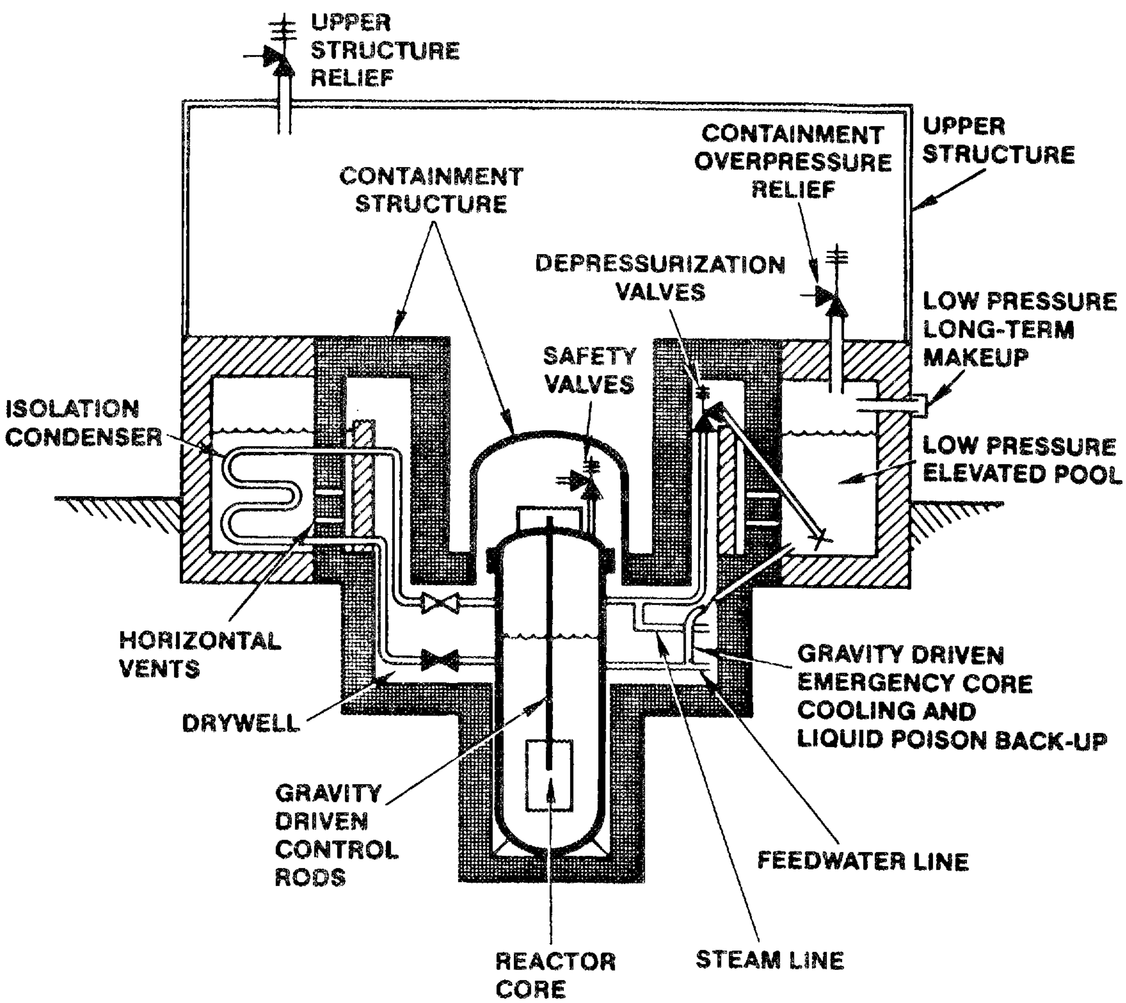

Fig. 4.4 The lower power BWR concept.

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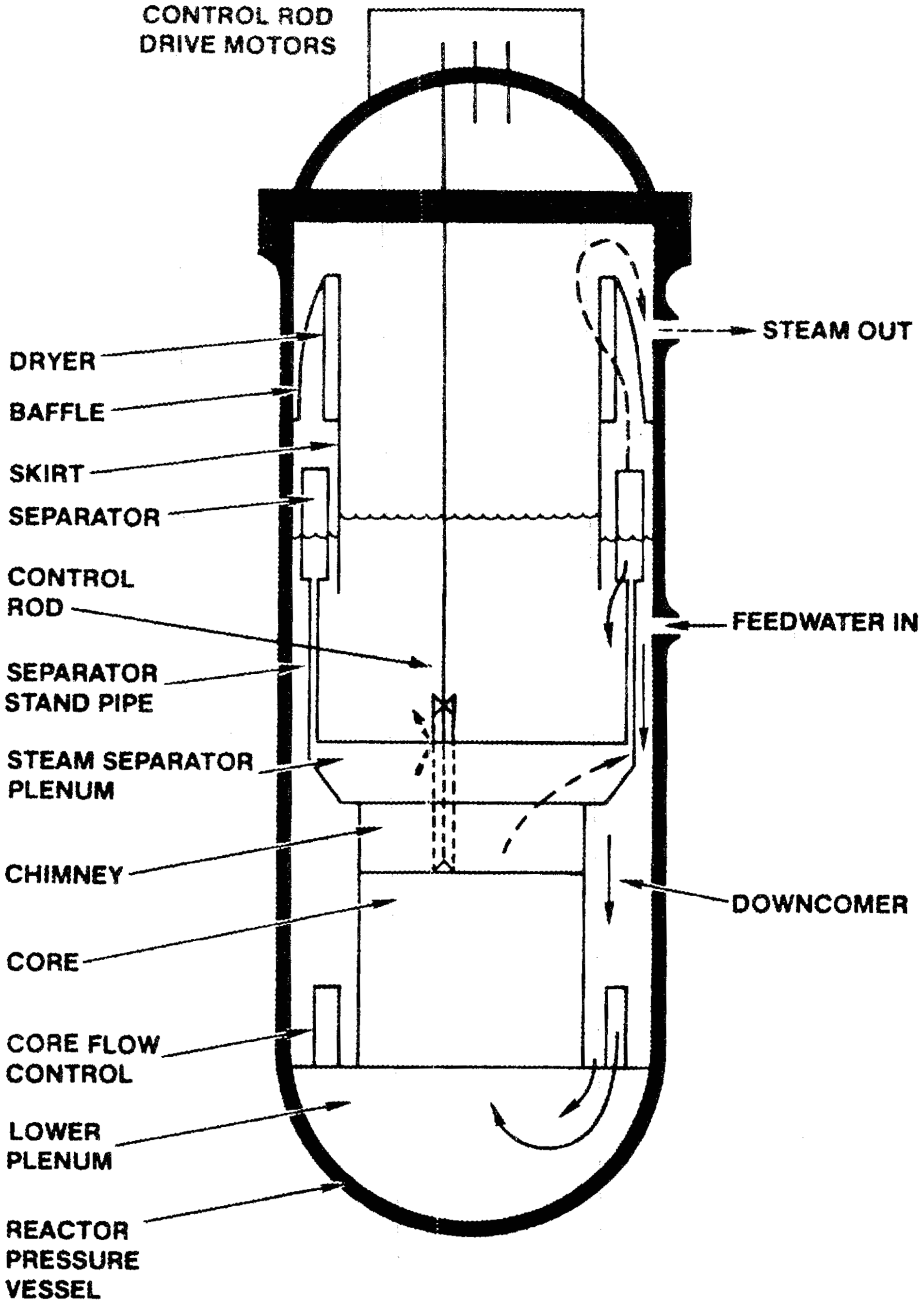

Fig. 4.5 Power production in the reactor vessel.

core is achieved by natural convection.* The control rod drives are mounted at the top of the vessel;* therefore, steam separators and dryers must be positioned peripherally outside the circumference of the rods.

4.3.1 Safety Systems

Safety systems are generally more passive than those of the large BWR designs. Safety rods are either gravity or hydraulically actuated. There is an isolation condenser which transmits decay heat from core to suppression pool by natural circulation. Steam is normally to be admitted to the isolation condenser; about 4% of the reactor heat would thereby be used to preheat feedwater. A steam-driven jet supplies makeup from the condensate storage tank to the reactor vessel to remove decay heat (Fig. 4.6); it is proposed to operate this system during normal power operation as well. In an emergency, the reactor can be blown down to the pressure of the elevated borated suppression pool and decay heat removed by natural circulation. In that mode, there is sufficient heat capacity to prevent core damage for three days without operator intervention. This feature eliminates dependence on emergency diesels.

The valves required to perform safety functions fail in a "safe" direction.

The large steam surge volume in the reactor vessel tends to damp pressure surges.

The small reactor accommodates a limited amount of spectral shift flow control, less than for a standard BWR.

The lower power BWR has not been developed to the extent where a cost or market evaluation can be made.

4.3.2 The Modified Small BWR

Henry E. Stone has provided (1984) the following additional comments:

"We have selected a power level of about 600 MW(e) for further evaluation. At this power, plant cost studies indicate that forced circulation is a better choice for providing core flow than natural circulation. We have, therefore, opted to follow the ABWR approach and use internal recirculation pumps, as noted in Sect. 4.2. This eliminates the need to develop the natural circulation core flow control device (Fig. 4.5). It also allows core flow (and therefore core power) control over a wider range and retains the spectral shift flow control capability.

*The small GE BWR concept has been modified since the preparation of this report to be cooled with in-vessel recirculating pumps similar to those of the ABWR, and the control rods mounted at the bottom of the vessel. The modified concept is described in the "Nuclear Power Options Viability Study" report.

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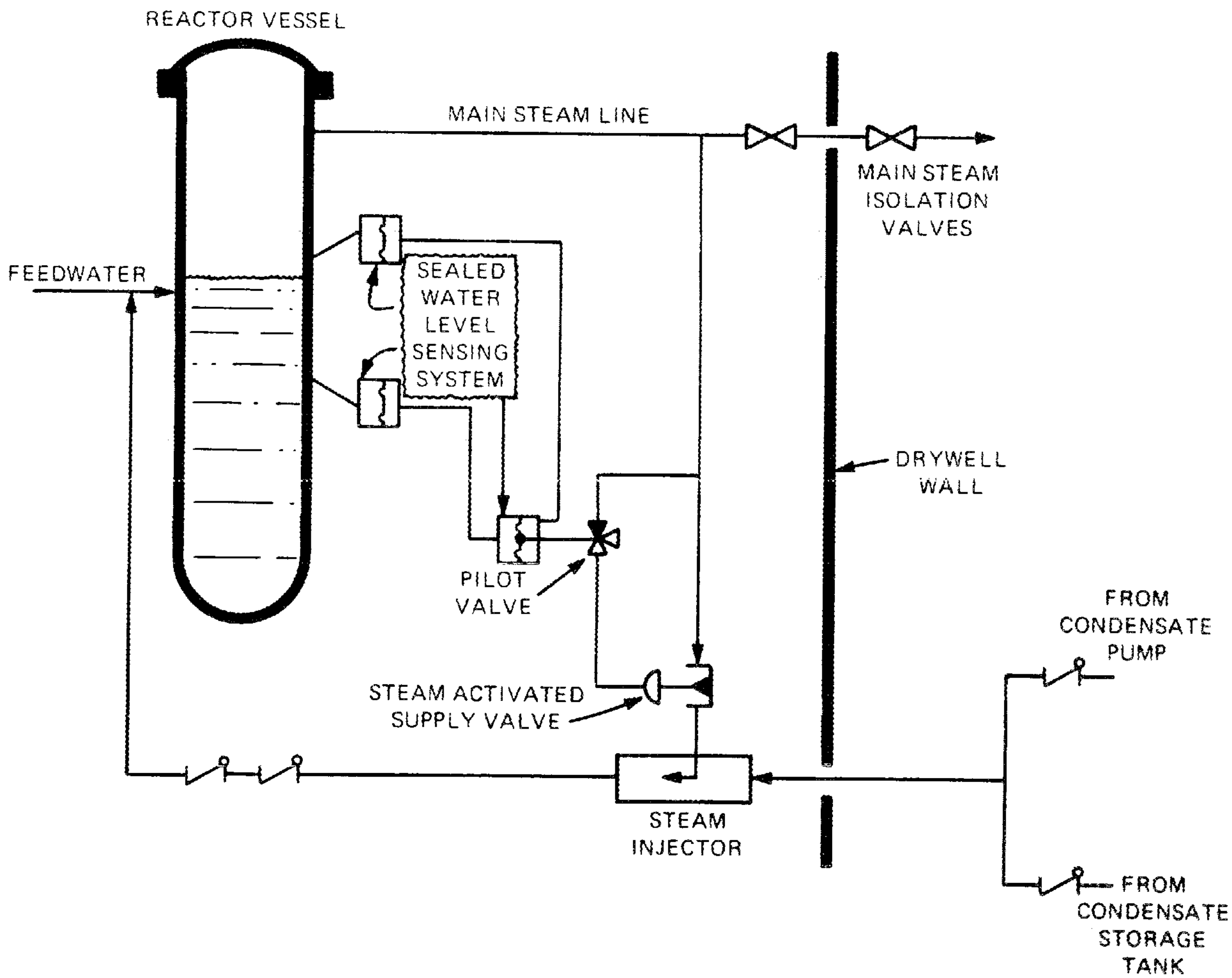

Fig. 4.6 Steam injector system.

"Our cost studies also indicate that there is a very small economic incentive associated with top-mounted, control-rod drives. We have, therefore, chosen bottom-mounted drives, similar to choice used in ABWR. This reduces the development effort required."

"With these changes in the concept, we have a higher confidence in reaching our objectives. We think that a four- to five-year construction period can be realized and that busbar costs can be competitive with a plant using U.S. coal."

5. SUMMARY

The three U.S. vendors actively marketing NSSSs are each developing a product for the future which they expect to be more reliable, more maintainable, more economical, and safer than the present plants. These are all essentially 3800-MW(t) designs although all are studying smaller plants. They apparently will be offered as standard prelicensed designs with much larger scope than earlier NSSS offerings, with the possibility of firm prices.

Westinghouse with Mitsubishi Heavy Industries is developing a completely new design (APWR) to be built initially in Japan, hopefully for operation by the mid-1990s. Westinghouse is making a strong effort to have the APWR licensed in the U.S. as a standard plant.

Combustion Engineering is evaluating potential improvements to the System-80 standard design (CESSAR) that has already received final design approval by the NRC.

General Electric, with Hitachi and Toshiba, is developing a new design (ABWR) that incorporates advanced features which have been proven by the worldwide BWR suppliers. The ABWR is to be built initially in Japan, but the design could be adapted to the United States.

Westinghouse, C-E, and GE have done some conceptual evaluation of reactors in the 600-MW(e) class. The Westinghouse concept is a two-loop plant intended for factory assembly in a shipyard and delivery to a site by barge. The GE concept is a modification of the ABWR with some additional passive safety features. The C-E designs range from scaled-down System-80s to small natural circulation PWRs. These concepts may be of interest to DOE or EPRI as "small" reactors.

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