Conf-750614--2

DOMESTIC PARTICIPATION IN A NUCLEAR POWER PROJECT

by

J.A. Lane*

for presentation at the IAEA Interregional Seminar
on Nuclear Power Planning
Kingston, Jamaica
June 9-20, 1975

Importance of Local Participation

1. Although most nuclear plants ordered to date by developing countries have been on a turn-key basis, local industry plays a very important role especially in the construction and commissioning phases of a project. For example, during the construction of Kanupp by the Canadian GE ltd., the maximum number of Canadians at any time was 45 compared to 2350 local personnel. Since the field erection labor and construction services constitute about one-third of total plant costs, domestic participation can be quite high even on a turn-key job.

2. Manufacture of nuclear plant components by local firms offers additional advantages over merely participating in plant construction. These are (a) possible savings in component cost, (b) decreased foreign exchange requirement, (c) increased self-sufficiency in supply of parts, (d) upgrading of country's manufacturing capability, and (e) increased opportunities for development skilled manpower.

The IAEA Technical Assistance Mission to Brazil (1971)

1. A team of 4 experts was selected by the Agency to carry out a preliminary assessment of the capability of Brazilian industry to manufacture nuclear plant components for a second nuclear plant in the country, assuming that the first would be built as a turn-key project. This team spent the months of April and May 1971 in Brazil carrying out the assignment.

*Research sponsored by the U.S. Energy Research and Development Administration under contract with the Union Carbide Corporation.

2. A Brazilian 5-man working group was formed consisting of representatives of the electric industry (Eletrobras), the National Nuclear Energy Commission (CNEN), the Institute for Radioactive Research (IPR), and Institute for Nuclear Engineering (IEN). Prior to the mission, this group developed a list of 23 key industries most likely to be involved in a nuclear program. This greatly facilitated the work of the Mission.

3. Because of the limited time and manpower involved (18 man-months), the study was limited to PWR's and BWR's currently available from commercial suppliers. Only components and associated facilities making up the nuclear steam supply system (NSSS) and radioactive waste treatment system were considered.

4. As a first step, representatives of each of the selected industries were invited to CNEN headquarters for a 6-day briefing on the characteristics of light water reactors. Each major component was described with the aid of working drawings. Specifications and lists of components were given to the representatives. Questionnaires were also given to each company requesting data on the capacities of available production equipment.

5. One-half- to one-day visits were made to each industry to evaluate individual manufacturing capabilities. Each visit started with a meeting with key people to discuss the questionnaire. This was followed by a plant tour and then a final meeting to answer final questions.

6. The final three weeks of the study were devoted to an analysis of the information gathered and preparation of the report. This included a list of each company's production equipment and a cross list of potential suppliers of each important component.

7. Table 1 lists the actual companies that were evaluated along with the main production line of each company. This serves to illustrate what kinds of firms are most likely to be involved in the manufacture of nuclear plant components.

8. As a result of the study, it was concluded that the industries considered had the capability of manufacturing essentially all of the

smaller items of a nuclear steam supply system which operate more or less under conventional conditions. It was estimated that these constitute about 25% of the NSSS cost. Major components such as the reactor vessel, the main circulating pumps and steam generators, however, will have to be imported for several years until the engineering knowhow, the materials of construction, and the required fabrication machinery are available in Brazil.

9. Because of the very preliminary nature of the study, the IAEA Report to the Government of Brazil was given only very limited distribution and is not generally available.

The Bechtel Study (1973)

1. Early in 1973, the Brazil Nuclear Technology Company (CBTN) commissioned Bechtel Overseas Corporation to perform an in-depth investigation of the Brazilian industry with the purpose of identifying its present and potential capability to manufacture components of LWR nuclear plants. The study required about 20 man-years of effort and required 12 months to complete.

2. According to CBTN specifications, the following tasks were performed.

a. A four-volume Component Encyclopedia was prepared containing technical descriptions of some 1464 components of a typical, modern nuclear power plant having a nominal electrical capacity of 1,000,000 kilowatts.

b. An extensive library of drawings, textbooks, manuals, codes, standards, special reports, and U.S. manufacturers' product catalogs was established to complement the Component Encyclopedia and enhance "the transference of technical knowhow," which was one of the secondary objectives of the survey.

c. An extensive research was made of modern processes and machinery used in the manufacture of nuclear power plant components. With the assistance of independent consultants, a separate volume was prepared containing lists of typical manufacturing equipment together with photographs and manufacturing process flow charts to aid in capability analyses and deficiency determination.

d. A study of Brazilian manufacturing firms was made by Bechtel's Brazilian associate, Montor S.A., to determine potential candidates for nuclear power plant component manufacture. The detailed selection process yielded a representative group of 79 firms for the inspection program.

e. The 79 firms were inspected by five teams of qualified CBTN, Bechtel, and Montor engineers and inspectors utilizing specially composed inspection questionnaires and instructions. The inspection data are presented in four volumes of typed questionnaires which include the individual capability analyses performed with assistance of CBTN and outside consultants for each form.

3. The potential domestic participation in terms of % of dollar value of components is shown in Table 2. The percentages shown do not include the foreign content in the form of special materials and services. These costs amount to 3%, 6%, and 9% of the subtotal costs for the respective three phases.

4. The ability of a developing country, such as Brazil, to manufacture the balance of plant equipment is shown by Table 3.

5. In terms of numbers of components involved, the results of the Bechtel study indicated a first stage capability to manufacture 829 of the 1464 components listed in the above mentioned Encyclopedia. By correcting minor equipment deficiencies, the number could be increased to 1087 in the second stage. Third stage capability would extend to 1257 components.

6. Some 207 components were considered unlikely to be manufactured in Brazil in the foreseeable future for one or more of the following reasons: new factory required, excessive investment in relationship to assumed market, components linked to warranties by suppliers of imported systems, and feasibility too much subject to speculation.

7. Among the components considered unlikely to be manufactured in Brazil for the above reasons are the following: plant computer, seismic instrumentation, electronic signal converters, special piping hangers, electrical penetration assemblies, the main turbine-generator set and most of its auxiliaries, radwaste and boric acid packages, special pumps, large auxiliary steam turbines, and most of the components contained in or associated with the nuclear steam supply system (reactor vessel and internals, reactor tools and drive mechanisms, steam generators, pressurizer, coolant or recirculating pumps and motors, nuclear instrumentation).

8. The study generated a considerable volume of written background information, data, reports, etc. Volumes 1-4 constitute the Component Encyclopedia, Volume 5 covers the selection of manufacturers, Volumes 6-9 contain the plant inspection data (restricted distribution) and Volume 10 is the final report. An executive summary was also prepared. The availability of any of this information is uncertain; however, the executive summary can probably be obtained from CBTN.

The Pakistan Study (1974)

1. P. Butt of the Pakistan Atomic Energy Commission presented a paper at the CENTO Nuclear Energy Symposium, Ankara, Turkey, June, 1974, which summarizes the results of a PAEC study of local manufacturing capabilities for components of a heavy water reactor.

2. Since the 137 MW KANUPP reactor of the CANDU type has been operating in Karachi since 1972, this reactor was used as the basis of the evaluation. The KANUPP components were extrapolated to a plant of 500-600 MW capacity since plans are to set up a nuclear plant of this size by 1981.

3. For the purpose of the study components which could be manufactured locally were divided into 3 categories as follows:

Category A - Components that can be designed and built locally.

Category B - Components that can be built locally with imported designs.

Category C - Components that can be built locally using imported designs, the assistance of foreign personnel and possibly imported specialized machines and equipment.

4. Components of a CANDU reactor which might be manufactured in Pakistan are listed in Table 4 along with category and estimated cost. The costs shown do not include site erection costs, civil works costs, instrumentation costs or any indirect costs. More details are given in Mr. Butt's paper.

Influence of Type of Reactor on Domestic Participation Capability

1. None of the three studies just described made any attempt to evaluate the influence of type of reactor on the extent of domestic participation in a nuclear project. Although in the IAEA study both the BWR and PWR components were described to industrial representatives, it was not possible to compare these two reactors in the limited time available. When the Bechtel study was undertaken, a decision to build a PWR in Brazil had already been made. Thus the Bechtel evaluation concentrated on this type of reactor. Finally, the PAEC study looked only at a heavy water reactor because their experience was limited to this type of reactor. Thus there is no study available which shows the influence of reactor type on the extent of domestic participation.

2. Although it is not possible to make such a comparison in this seminar, a brief description of commercially available reactor types may be useful as a means of identifying broad areas where the choice of reactor type may be important from the standpoint of local manufacture. Descriptions of these reactors are given in the following paragraphs.

Boiling Water Reactor (BWR)

1. A schematic of the BWR Nuclear Steam Supply System is shown in Figure 1. As seen in this figure, the nuclear fuel assemblies are arranged inside a core shroud in the reactor vessel. Water boils in the core, and a mixture of steam and water flows out the top of the core and through steam separators at the top of the core shroud. Steam from

the separators passes through dryers to remove all but traces of entrained water and then leaves the reactor vessel through pipes to the turbine generator. Water from the steam separators and water returned from the turbine condenser mix, flow downward through the annulus between the core shroud and the reactor vessel, and return to the bottom of the core. Because the energy supplied to the reactor coolant (water) from the hot fuel is transported directly (as steam) to the turbine, the BWR system is termed a "direct cycle" system. The pressure in a typical BWR is maintained at about 1000 pounds per square inch (70 bar); at this pressure water boils and forms steam at about 545°F (285°C).

2. Details of the reactor vessel and internals for a typical BWR are shown in Figure 2. Steam flows from the reactor vessel to the turbine-generator in multiple main steam lines. The head of the vessel and the steam separators and dryers are removable for refueling the core. Neutron-absorbing control and safety elements in the reactor core are connected to rods that pass through fittings in the bottom head of the vessel and are operated by hydraulic drives mounted below the vessel. Because the reactor heat output is sensitive to the rate-of-flow of coolant through the core, partial control of the power is effected by varying the driving flow to the pumps that can recirculate some of the water through the core.

3. Modern BWR's employ primary and secondary containment such as shown in Figure 3. The primary containment employs a "drywell," enclosing the entire reactor vessel and its recirculation pumps and piping. It is connected through large ducts to a lower-level pressure-suppression chamber which stores a large pool of water. Under accident conditions, valves in the main steam lines from the reactor to the turbine-generators would automatically close, and any steam escaping from the reactor system would be released entirely within the drywell.

4. The secondary containment system is the reactor building that houses the reactor and its primary containment system. The most advanced BWR plants use a separate free-standing leak-tight containment shell inside of a sealed building which provides a further barrier to the escape of gaseous effluents, as well as a shielding to further reduce the escape of radiation emanating from the reactor proper.

Pressurized Water Reactor (PWR)

1. Unlike the direct in-vessel boiling of BWRs, all PWRs employ dual coolant systems for transferring energy from the reactor fuel to the turbine and are called "indirect cycle" systems. The high-pressure circuit comprising the reactor vessel, piping, the necessary pumps, and the inner tube-side of the steam generators is termed the "primary system;" the lower pressure circuit is called the "secondary system." (A schematic arrangement of a 1000-MWe PWR system, with four steam generators and one pump for each steam generator, is shown in Figure 4.

2. The pressure maintained in a typical large PWR system, about 2250 psi (150 bar) permits water to be heated to about 650°F (340°C) without boiling. The high-pressure water, heated to an average temperature of around 315°C, is piped out of the reactor vessel into two or more steam generators. Heat from the high-pressure reactor coolant water is transferred through heat exchanger tubes into a secondary stream of water at considerably lower pressure and temperature than the former and causes the water of the secondary stream to boil and produce steam for the turbine.

3. A cutaway view of a typical PWR reactor vessel and its internals is shown in Figure 5. The vessels have removable top heads (for refueling) provided with fittings to accommodate the mechanisms for driving neutron-absorbing rods into and out of the core to control the nuclear chain reaction.

4. Most present-day PWR containments are constructed of reinforced concrete with a steel liner (Figure 6). Refinements in containment technology are still being made, and containment systems vary widely from plant to plant. For example, in some PWR plants, the containment space is kept at slightly below atmospheric pressure so that leakage through the containment walls would, at most times, be inward from the surroundings. Other systems have double barriers against escape of material from the containment space.

5. Two kinds of additional measures are taken in PWR plants to minimize the potential for escape to the environment of any accidental release of radioactive materials. In some plants, cold-water sprays are provided to condense the steam resulting from a major escape of primary system coolant into the containment; in other plants, stored ice is used for this purpose.

High Temperature Gas Cooled Reactor (HTGR)

1. The HTGR operates on the thorium-uranium fuel cycle. Helium is used as the reactor coolant, and graphite is the moderator and core structural material. The fuel is a mixture of thorium and highly enriched uranium particles coated with thin layers of pyrolytic graphite. These particles are then bonded into fuel rods and inserted into large blocks of graphite. The use of helium as a coolant has the fundamental advantages that the coolant always remains in the same phase and is chemically inert. However, because of its relatively poor heat conduction properties, 50 bar pressures must be used. The graphite is used both as a moderator and a core structural material; however, a potential disadvantage of graphite is its tendency to react with steam (which might enter the reactor core if there should be a leak in a steam generator).

2. The characteristics of the HTGR NSSS are illustrated by Figure 7 showing a vertical section of the 330 MWe Ft. St. Vrain reactor.

3. The most striking feature of this reactor type is the Pre-stressed Concrete Reactor Vessel. The PCRV contains the reactor core and entire primary coolant system, including steam generators and helium circulators. The PCRV also serves as the primary coolant system pressure boundary and provides the necessary biological shielding. In this design, the steam generator module is located beneath the core. In the 1160 MWe capacity designs, the PCRV consists of a central cylindrical cavity containing the core, surrounded by six cavities containing the steam generators and main helium circulators and by three smaller cavities containing the auxiliary gas circulators and heat exchangers.

4. The cavities and all penetrations are lined with welded carbon steel, which acts as a leak-tight barrier. The top head above the central cavity of the PCRV contains a number of penetrations that house control rod drives, the reverse shutdown system, and the core orificing mechanism.

5. Each of the six primary circuits in an 1160-MWe plant is equipped with a helium circulator. Each circulator consists of a single-stage axial flow helium compressor and a single-stage steam turbine drive. The circulators are water-lubricated and have a helium buffer seal that is designed to prevent helium leakage from the primary coolant or water in-leakage to the coolant.

6. Each steam generator consists of a single helical tube bundle arranged in an annulus of a center duct. Helium leaves the core at 760°C and enters the steam generator. The resultant steam generator outlet conditions are 510°C and 160 bar. This results in a net plant efficiency of 39%.

Heavy Water Moderated and Cooled (HWR)

1. A heavy water moderated and cooled reactor fueled with natural (unenriched) uranium (CANDU) is being sold commercially by the Canadians. This concept, shown schematically in Figure 8, features horizontal pressure tubes and on-line refuelling. The pressure tube approach has the advantage from a local participation standpoint that no thick walled stainless steel clad pressure vessel is required. The s.s. calandria vessel, though 10 m in diameter and 11 m long, is only about 2 cm thick and could be produced locally.

2. In a typical CANDU design, the heavy water coolant, at about 95 bar, passes through the pressure tubes where it is heated to about 290°C. Steam at 250°C and 40 bar is generated by the heavy water coolant in large U tube heat exchangers with an overall plant efficiency of 29%.

3. Two vertical sections through the 137 MW KANUPP reactor which is of the CANDU type are shown in Figure 9 to show the arrangement of components. Details of these components are given in Butt's paper.

Steam Generating, Heavy Water Moderated Reactor (SGHWR)

1. The SGHWR uses heavy water, at low pressure, as the moderator and boiling light water as the coolant. The separation of the two fluids is achieved by means of Zircaloy pressure tubes. The fuel can be either natural or slightly enriched (∿ 2%) uranium dioxide, sealed in Zircaloy tubes. This reactor type employs standardized isolated hot channels of the same length and steam generating conditions for a large range of power sizes.

2. A 100 MWe prototype plant fueled with slightly enriched UO_2 is in operation in Great Britain and shown schematically in Figure 8. It contains the same hot channels which would be required for a 600 MWe unit. Thus the British are in a position to offer slightly enriched SGHWR's up to 600 MWe on a firm basis. Since large plants are not in full operation, some of the design features and costs of construction and operation remain to be demonstrated.

Comparison of Reactor Types

1. Despite the fundamental differences in the design and operating characteristics of the BWR and PWR, it is unlikely that the extent of domestic participation would be greatly different for these reactor types. In both cases, the large components associated with the nuclear steam supply system such as the pressure vessel, steam generator pressurizer valves and main circulating pumps would have to be imported.

2. In the case of the HTGR it is possible that the prestressed concrete reactor vessel could be assembled by a local firm with the assistance of foreign experts. Here again, however, the helium circulators, heat exchangers and other major components of the NSSS would be imported.

3. The CANDU HWR and SGHWR, both being of the pressure tube type of reactor, offer the possibility that the calandria, pressure tubes, end fittings, etc., could be produced locally. The large NSSS components (main circulating pumps, high pressure valves, refuelling machines, and heat exchangers, etc.) would be imported. The extent of domestic participation should be roughly the same for these two reactor types.

Steps to Evaluating Domestic Participation Possibilities

1. In order to carry out a study of the capability of local industry to participate in a nuclear power project, the following steps should be undertaken:

a. Organize a local working group consisting of representatives of the Government, utility and nuclear authority.

b. Develop or obtain technical descriptions of components of each nuclear reactor type of interest.

c. Establish requirements of manufacturing equipment required to produce each component.

d. Collect information on manufacturing capabilities of industrial firms in country.

e. Select industries to be inspected.

f. Organize inspection teams.

g. Compose inspection questionnaires for evaluating production capabilities and quality assurance and quality control practices in each inspected industry.

h. Carry out inspections.

i. Collate and analyze results of inspections.

2. The IAEA is in a position to undertake preliminary studies of domestic participation in a nuclear project in the form of Technical Assistance Missions. More detailed studies would have to be done by the country concerned, possibly using the services of an Architect-Engineering firm or some other type of consulting firm.

Table 1

Brazilian Industries Visited by the IAEA Mission

Company Name	Parent Company	Main Production Lines
Mecanica Pesada	Schneider-Creusot, France MAN, FRG	Heavy machinery
Brown Boveri	Brown Boveri, Switzerland	Heavy machinery
Charleroi	ACEC, Belgium	Transformers, switchgear
Cobrasma	--	RR, automotive, chemical plant
Nordon	Nordon & Cie, France	Chemical plant
Confab	--	Industrial plant, piping
Cornersol	--	Valves
KSB	Klein, Schanzlin & Becker, FRG	Pumps
General Electric	GE, USA	Transformers, generators
Bardella	--	Heavy machinery
Jaragua	--	Chemical, petrochem. plants
Bopp-Reuther	Masoneilan Int., USA	Valves
Villares Equipment Division	--	Heavy machinery
Villares Steel	--	Alloy steels
Mannesman	Mannesman AG, FRG	Piping
Usiminas	--	Iron ingots, steel plates
CBC	Mitsubishi, Japan	Boilers, tanks
Ishikawajima	Ishikawajima, Japan	Ships, diesel engines
CBV-Microlab	--	High precision parts
CSN	--	Steel
Worthington	Worthington, USA	Pumps
Sulzer	Sulzer, Switzerland	Refrigerator, a/c
Coemsa	Ansaldo, Italy	Transformers, steel structures

Table 2

Potential Contribution of Brazil Industry to Construction of 1100 MW PWR

Item	U.S. $millions (1973 levels)	Local Contribution, % First Stage 1973-74	Second Stage 1975-77	Third Stage 1980-82
1. NSSS + Auxiliaries	48.5	2.9	14.7	21.2
2. Turbine Generator + Auxiliaries	40.9	3.1	6.6	7.3
3. Equipment for Balance of Plant	45.0	40.3	55.0	61.2
4. Field Erection Labor for Items 1-3	31.9	96.2	96.2	96.2
5. Civil/Structural Work	42.3	88.7	95.5	95.5
6. Construction Facilities, Site Services, Plant Startup	31.4	95.5	95.5	95.5
Subtotal	240.0	50	56	59
7. Engineering Procurement, Const. Management Services	42.6	0	0	0
8. Contingency Allowance	27.4	0	0	0
Total	310.0*	38	44	46

*Costs exclude transmission lines, construction camps, taxes, duties, interest during construction, insurance, nuclear fuel, land, spare parts and operator training.

Table 3

Brazilian Manufacturing Capability in

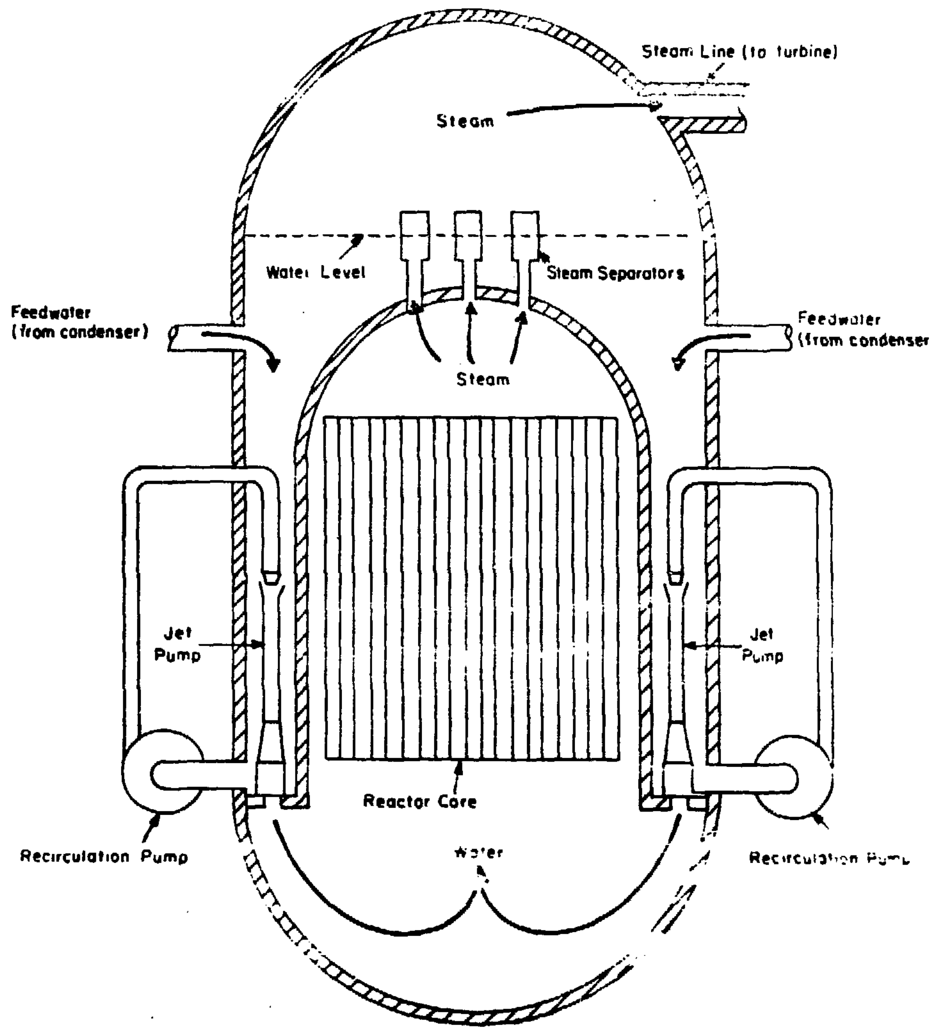
Balance of Plant Equipment

Cost Elements	Estimated Cost (U.S. $Million)	Possible Brazilian Industry's Contribution		
		First Stage (1973-1974)	Second Stage (1975-1977)	Third Stage (1980-1982)
A. Rotating Mechanical Equipment	5.1	54	72	73
B. Tanks, Vessels, Heat Exchangers	10.4	87	88	89
C. Mechanical Handling Equipment	1.3	96	96	96
D. Process Piping Systems	12.3	11	38	60
E. Instrumentation and Control	3.5	13	44	54
F. Power Transformers and Isophase Bus	4.0	71	96	96
G. Cables Systems and Penetrations	1.8	75	75	75
H. Other Electrical Equipment	1.9	18	97	97
I. Heating, Ventilating, Air Conditioning Equipment	4.7	66	76	99
J. Total BOP Equipment	45.0	50	69	78
K. Foreign Content Included in Item J	--	10	14	16

Table 4

Components of a 600 MW CANDU Type Reactor
Which Might be Manufactured Locally in Pakistan

Item	Categories	Estimated Cost ($million)
Structures and Shielding Doors	A	3.0
Reactor Structures	A,B,C	4.0
Moderator and Helium System	A,C	0.15
Primary Heat Transport System	A	0.03
Auxiliary Systems	A,B	0.2
Fuel Handling System	A,B	0.3
Fuel (First Charge)	B	8.0
Boiler Steam & Water System	A,B,C	2.5
Intake Cooling System	A	0.5
Process Water System	A,B	1.1
Fire Fighting System	A	0.02
Water Treatment Plant	A,B	0.3
Ventilation System	A,B	0.5
Water Storage Tanks	A	0.1
Heavy Water Upgrading Plant	A	1.0
Reactor Building Dryers	A	0.3
Active Drainage Tanks	A	0.1
		22.1



SCHEMATIC ARRANGEMENT OF BWR NSSS

Figure 1

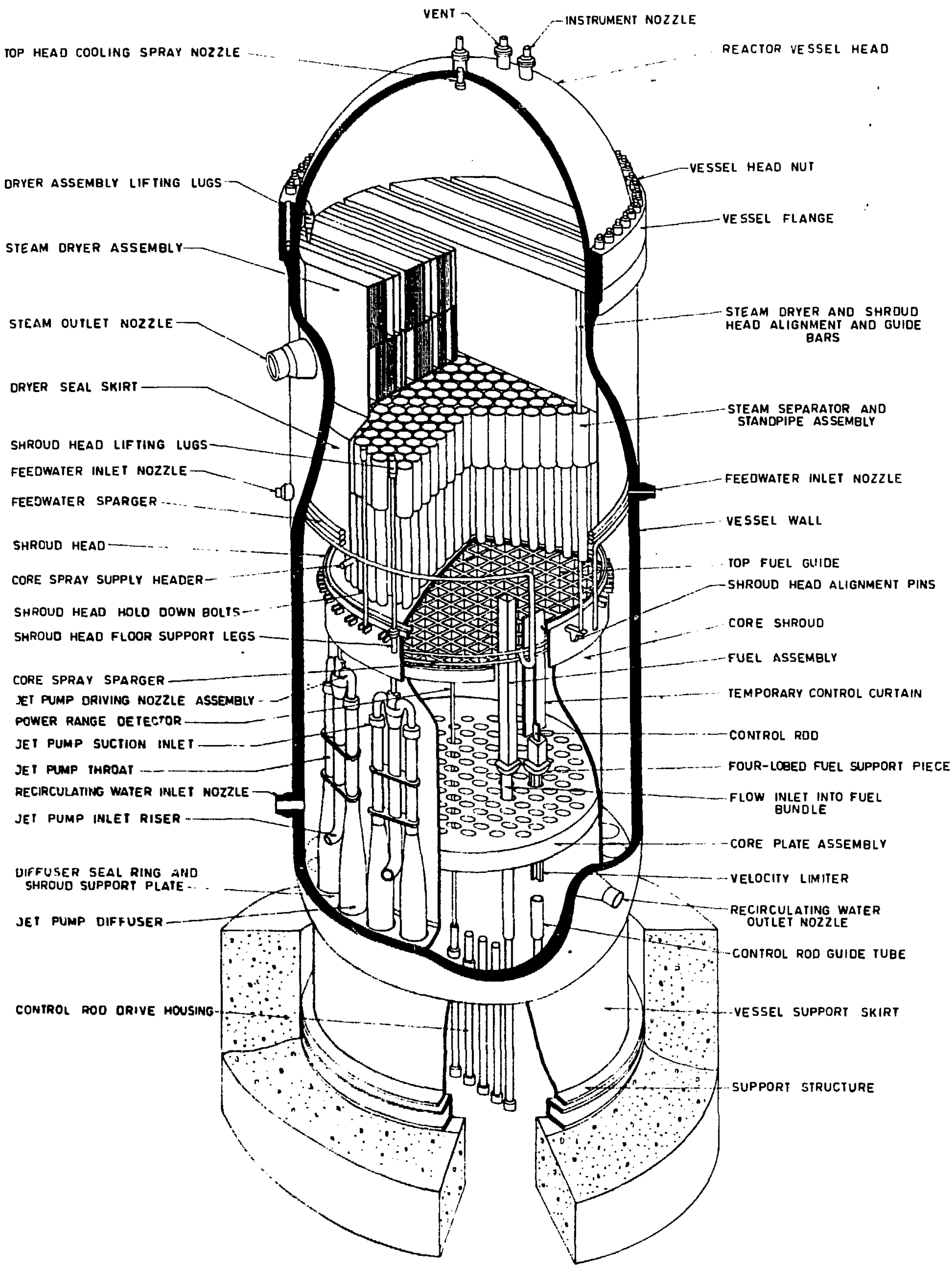


Figure 2

BWR Pressure Vessel and Internals

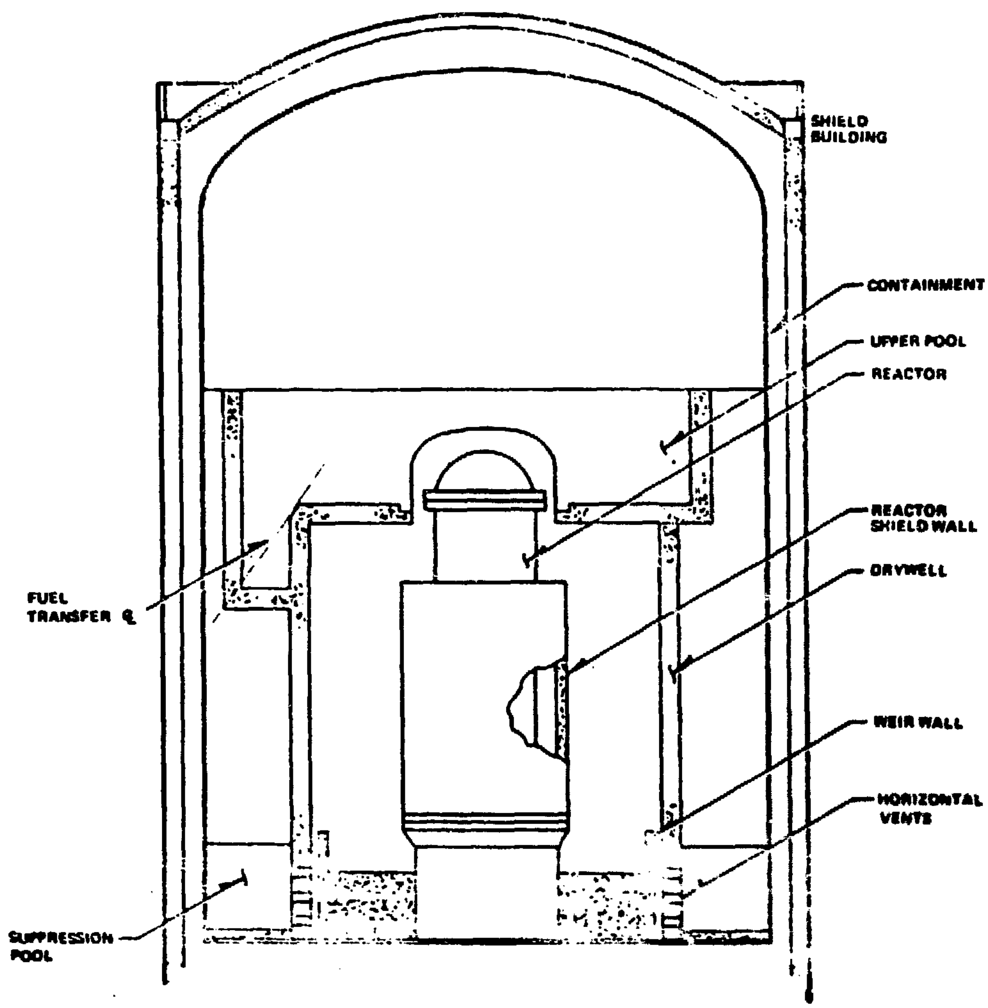
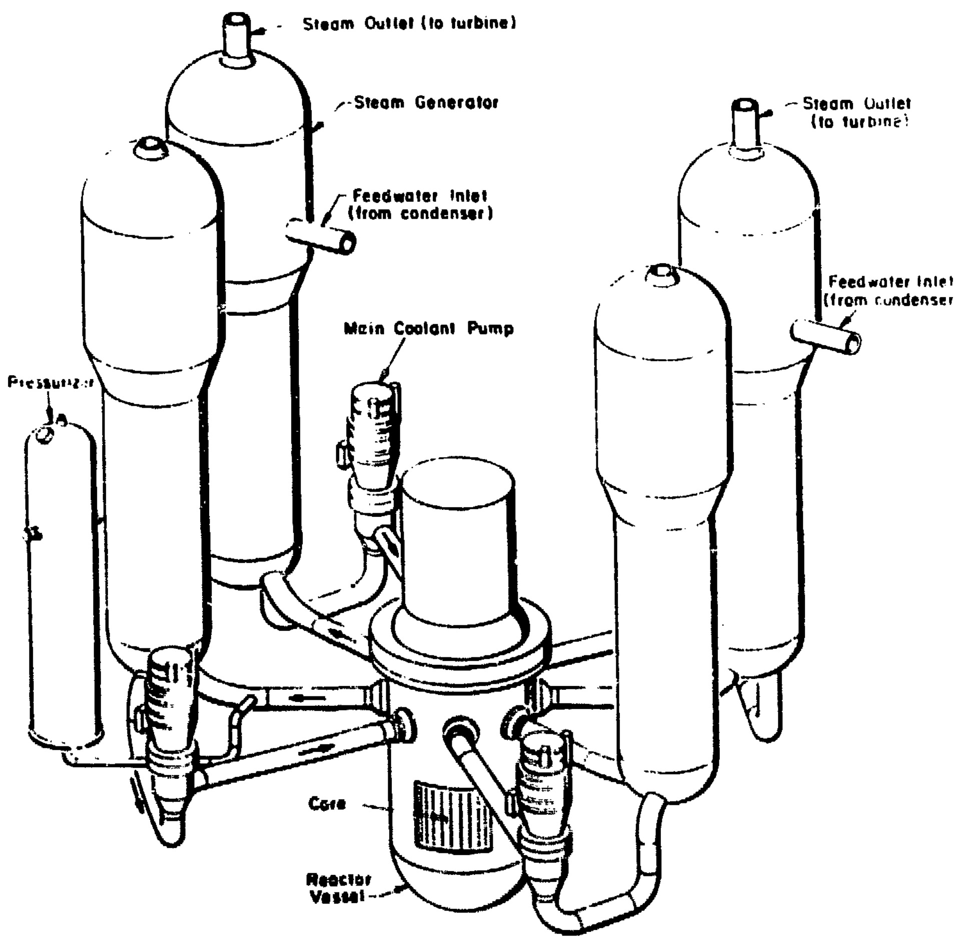


Figure 3

Multiple Containment System for Recent Large BWR Plants



SCHEMATIC ARRANGEMENT OF PWR NSSS

Figure 4

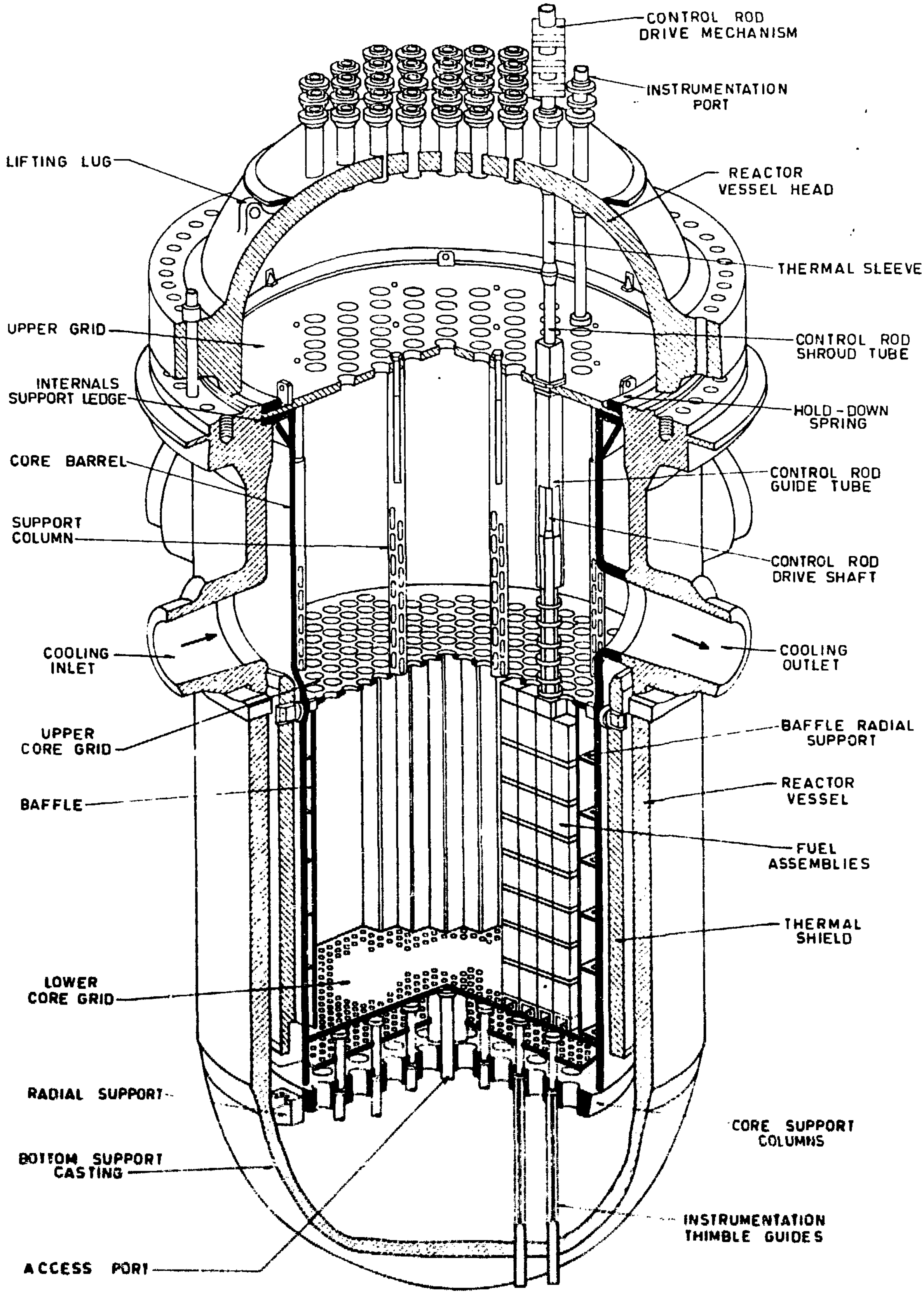
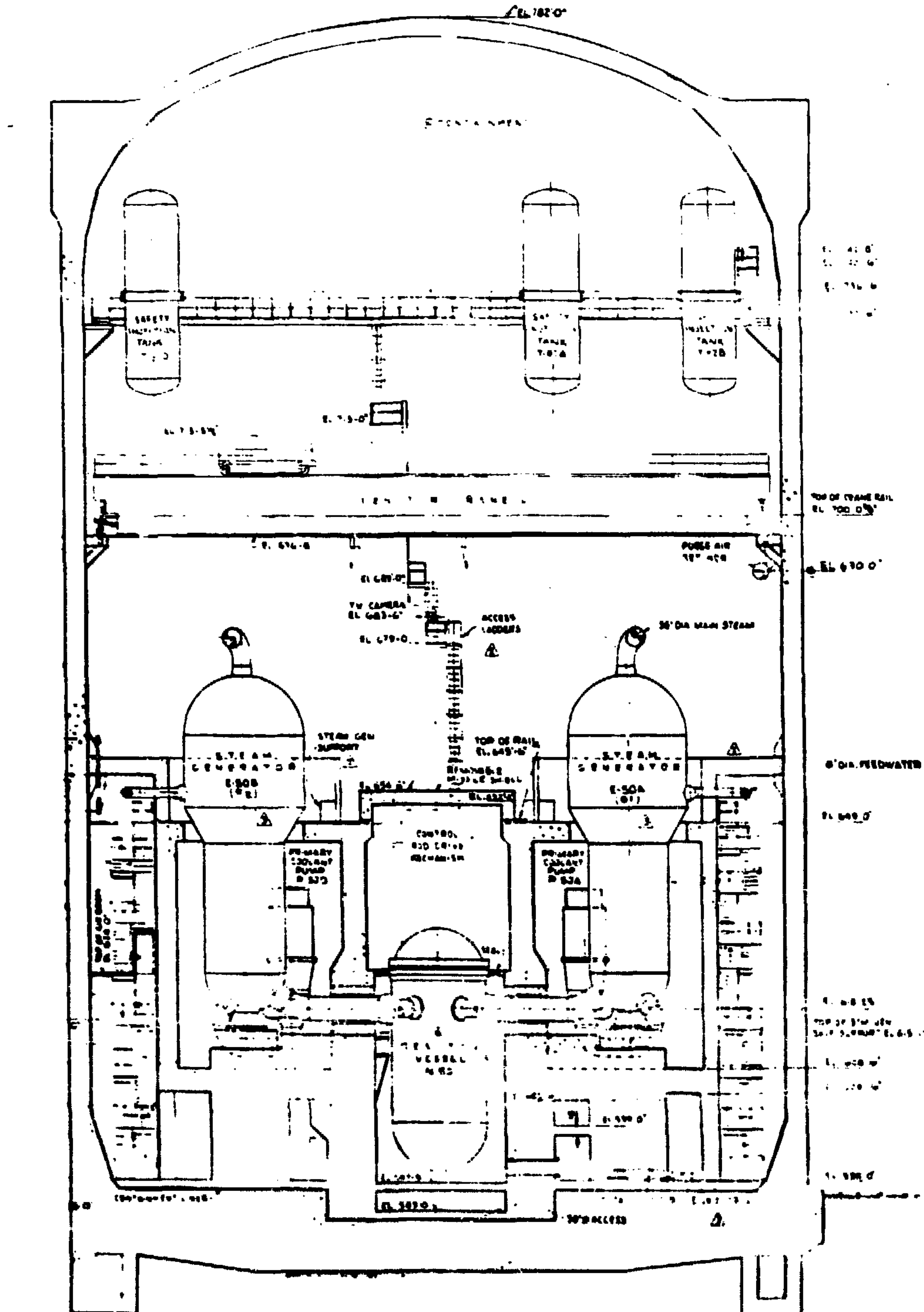


Figure 5

PWR Pressure Vessel and Internals



PWR CONTAINMENT

Figure 6

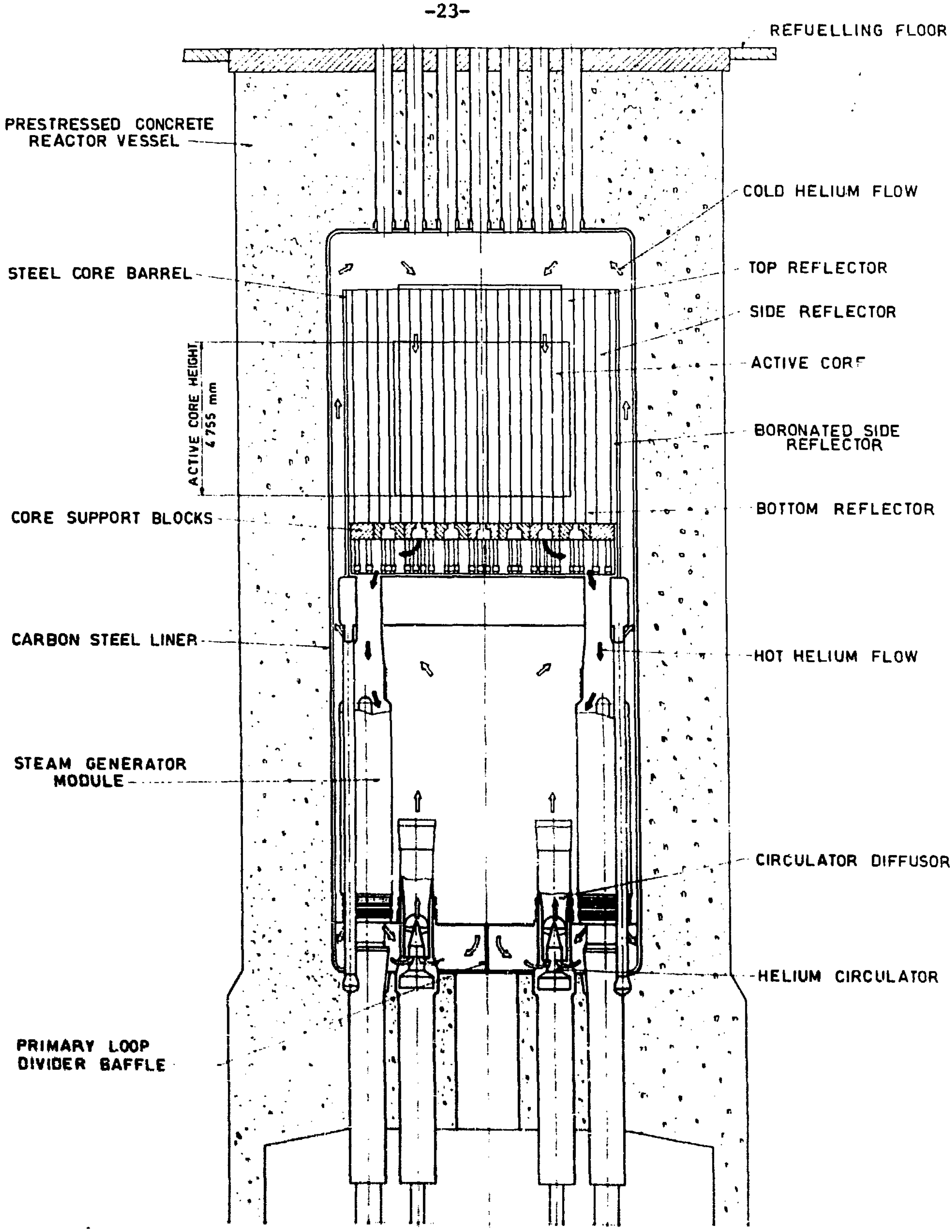


Figure 7

HTGR Pressure Vessel and Internals

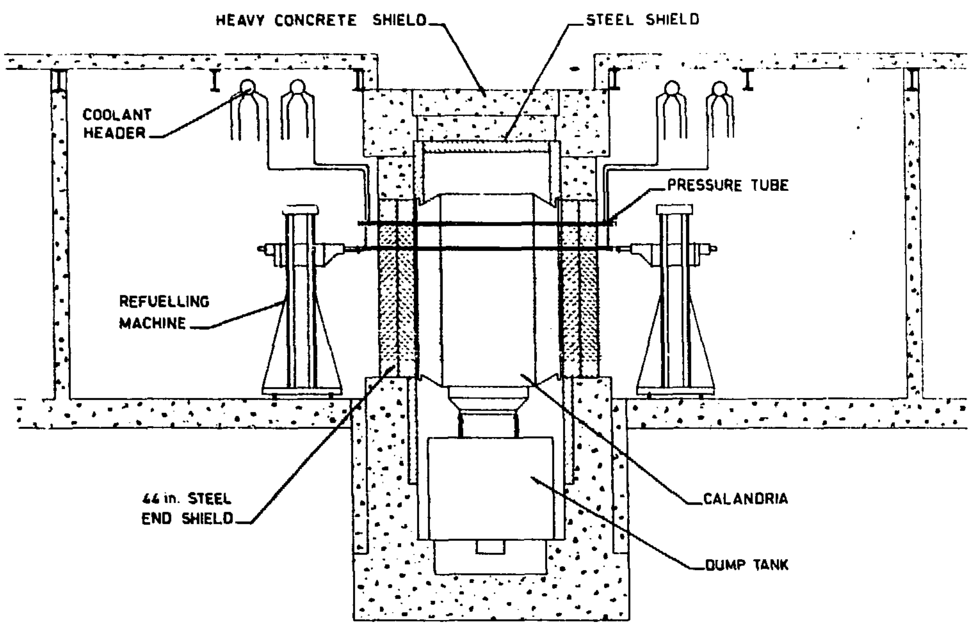


Figure 8

Schematic of CANDU Heavy Water Reactor

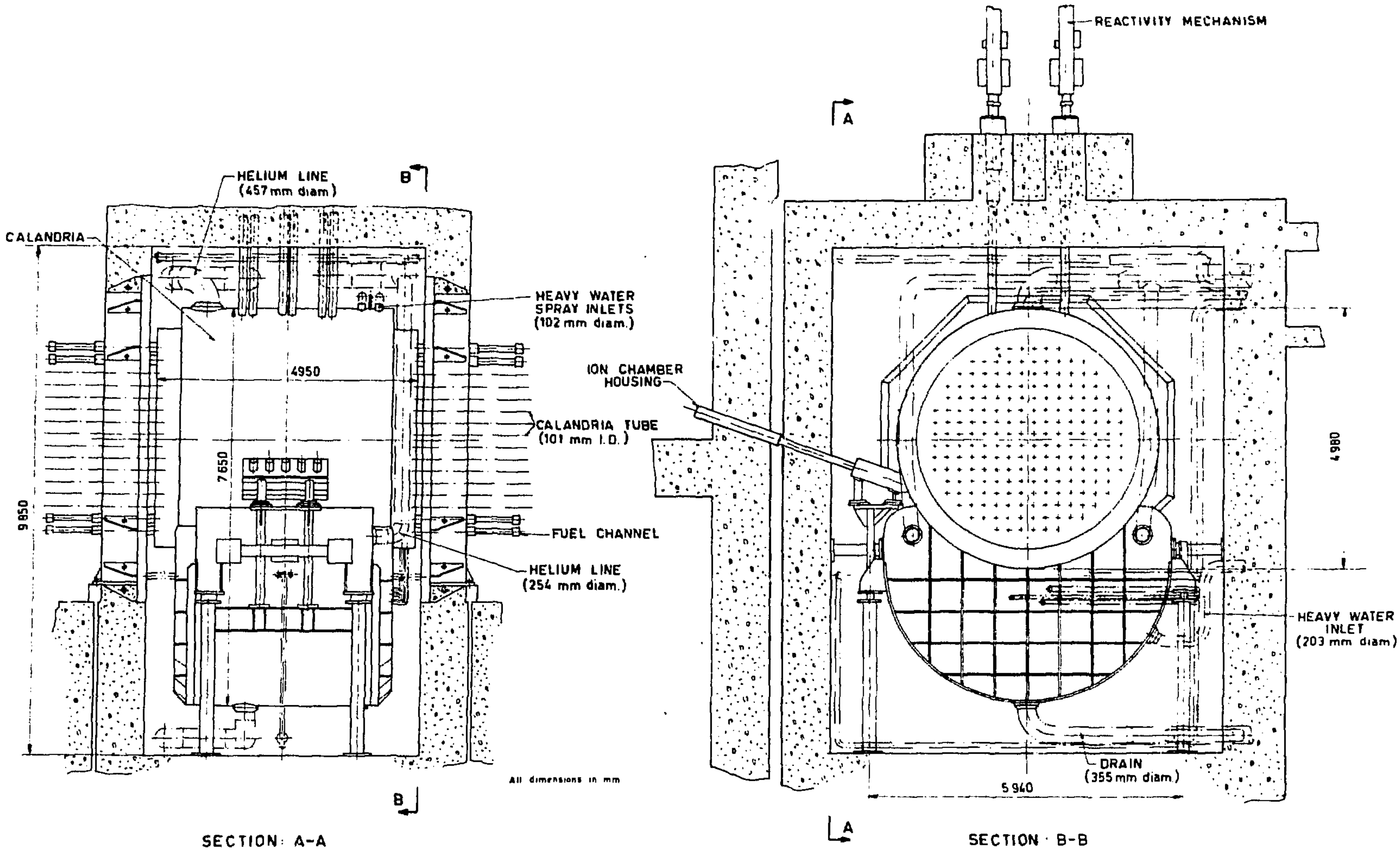


Figure 9
Vessel and Internals of KANUPP Reactor

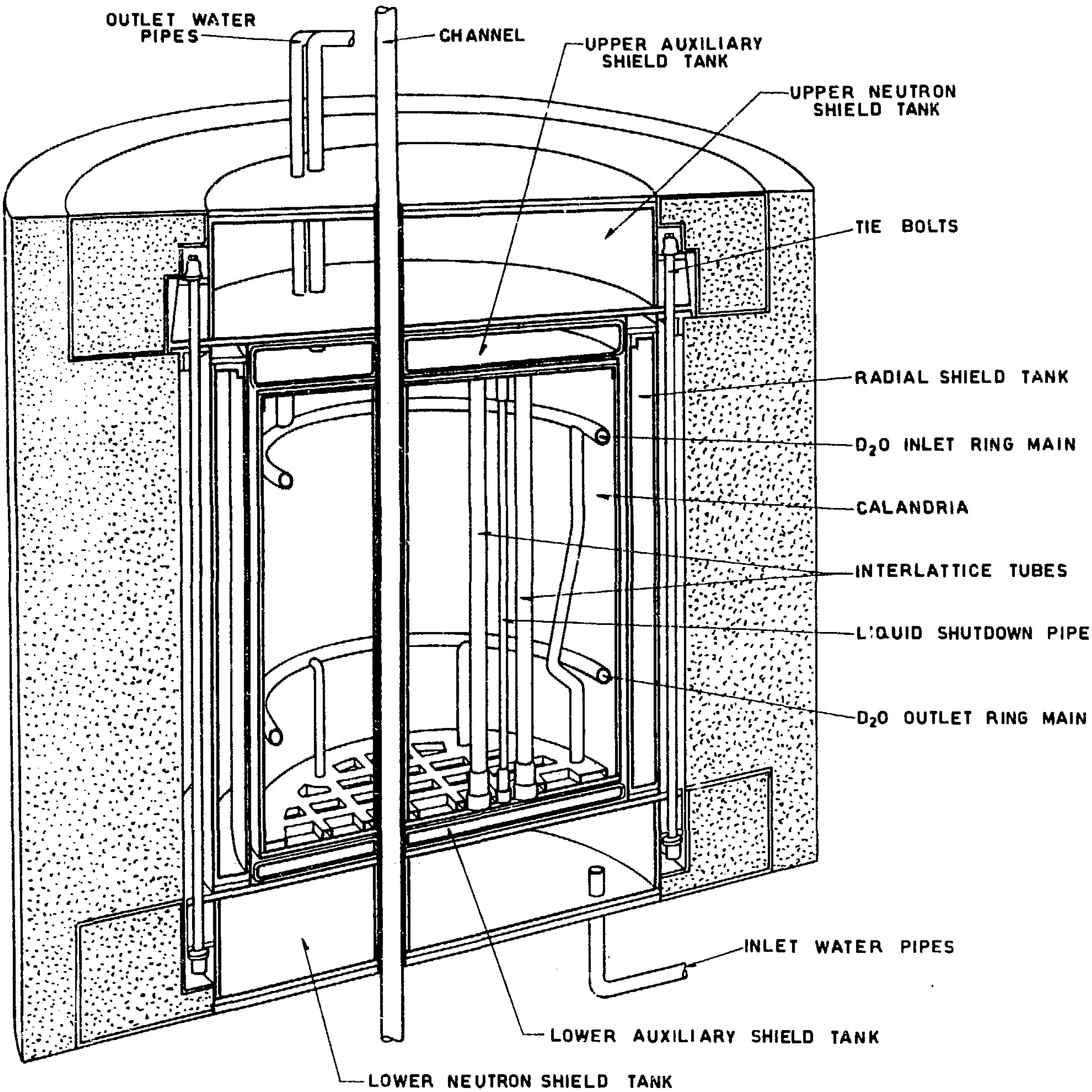


Figure 10

Schematic of SGHWR Prototype