ANALYSIS OF THE TECHNOLOGICAL DIFFERENCES BETWEEN STATIONARY & MARITIME NUCLEAR POWER PLANTS

by

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ABSTRACT

This thesis presents the results of a comparison between maritime and stationary reactors with the aim of determining the technological differences among them, the factors which cause these differences and the possibility of closer integration of their respective design and operations.

The comparison is made between anintegral-type PWR for maritime application and a loop-type PWR for stationary application. Four main factors that differently affect the behavior of the reactor in both land-based and ship-based applications are considered. They are: operational, environmental, safety and economic factors.

The results are given in terms that compare the systems performance, constraints and major technological differences. Some consideration about how to diminish or to avoid these differences are also presented.

Insofar as the possibility of maritime reactor application is concerned, it is concluded that the shipowners are still unlikely to invest in nuclear ships, not only a result of concern with technological reliability but because of a combination of assumed uncertainties and risks which are not technological, and the fact that for many applications, it is as yet assumed not to be economically superior.

Thesis Supervisor..................Ernst G. Frankel
Title ....Professor of Ocean Systems

A C K N O W L E D G E M E N T S

The author wishes to express his sincere gratitude to the Argentine Navy for the opportunity of pursuing the course of studies at Massachusetts Institute of Technology that has led to this thesis.

To Professor E. G. Frankel of the Department of Ocean Engineering and Professor Manson Benedict of the Department of Nuclear Engineering, the author wishes to give thanks for offering much good guidance.

But most especially, to my wife Ana Maria and my kids for their patient encouragement.

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CHAPTER I

INTRODUCTION

Stationary nuclear power plants have been built or ordered nearly exclusively in the range between 800 to 1200 Mw(e) during recent years. This is due to the economics of scale in power generation costs with the size of the nuclear plants as well as the increasingly large concentration of energy demand in highly populated industrial areas which most new nuclear plants serve.

However, there are likewise requirements for small and medium nuclear power units for several reasons. Among these are:

1. Nuclear propulsion of merchant vessels
2. Nuclear reactors with a power level of 200-500 Mw(t) that appear compatible with energy requirements of some single industrial plants.
3. Land-based power units for electric supply in remote areas.
4. Nuclear heat generation for seawater desalination to produce drinking water and water for irrigation in agriculture.
5. Nuclear plants to meet military installation energy requirements.
6. Small-to-medium reactors for electric power production in developing countries.

It is most interesting to note that medium and small size units have been as well tested as prototype plants for larger power units.

Whether small reactors become a viable means of supplying industrial energy depends to a large extent on the number of units per year demanded by the market. If the potential market is only one or two units annually and if each of these is a custom engineered system, then the cost of small reactors is likely to be prohibitive for most commercial applications. If a number of applications for essentially the same reactor technology could be identified, the prospects for economical small reactors would be substantially improved.

In the case of nuclear ship propulsion, there is a particular need for direct research and development to assure economic competitiveness of such small units.

In the past, nuclear propulsion for merchant ships has been unable to compete with conventional propulsion because of the relatively low power requirements of most ships and comparatively low percentage of full power utilization. However, container ships with propulsion plant outputs of between 80,000 and 120,000 shp are already in operations and ultra-large crude carriers(ULCC) with load capacity of 600,000 ton are definitely beyond the economics threshold of nuclear propelled ships.

Considering that small reactors, under conditions described above, would be economically suitable, it is important to emphasize the technological differences between a ship-based reactor plant and a land-based reactor plant.

The technology of nuclear plants can be defined by four basic factors:

I Operational Factors

II Environmental Factors

III Safety Factors

IV Economic Factors

These four basic factors are closely correlated among types and applications of nuclear plants. These factors affect differently the conditions of design, performance and operation of the nuclear plants, depending on the use of the nuclear plant on land or ship based. Moreover, as these conditions depend on the reactor type used in each case, a brief description of the past and present situation of the nuclear ship propulsion is made.

In conclusion, the most suitable type of reactor for maritime application was found to be the integral PWR.

In the subsequent chapters, the technological differences between an integral PWR reactor for maritime application and a loop-type PWR for stationary application are established, fulfilling the main purpose of this thesis.

CHAPTER II

SURVEY OF NUCLEAR POWERED SHIP DEVELOPMENT

Four non-military nuclear powered ships have been built up till now. Chronologically, these are: the USSR icebreaker "LENIN" which started to operate in 1959; the U.S. flag combination cargo-passenger ship N.S. "SAVANNAH" which operated from August, 1962 to July, 1970 and now is in lay-up; the West German ore carrier N.S. "OTTO HAHN", delivered in 1968 and still in service, and the Japanese cargo vessel N.S. "MUTSU" completed in early 1973 but not yet operational. See Figures 2.1 and 2.2.

The total number of nuclear ships in operation by all nations in 1972 is given in Jane's Fighting Ships as one merchant ship, one icebreaker, and 221 naval vessels. These are listed by type and flag in Table 2.1 and by engineering features in Table 2.2.

The following data were presented by Vice Admiral H.G. Rickover before the U.S. Congress (Ref. 12). In 1972, the United States had in operation 104 nuclear powered submarines which had a combined operating experience of 900 ship-years* and an estimated distance (through 1970) of over 16 million miles.

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*One ship-year is equivalent to the operation (including in-port and shipyard time) of one ship for one year, calculated from the data of launching.

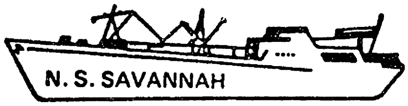
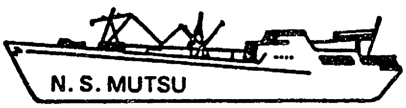
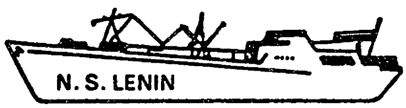
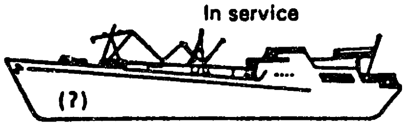
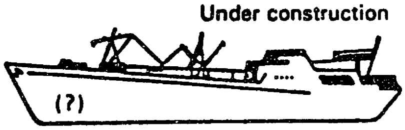
U.S.A.  Laid-up	• N.S. SAVANNAH laid-up after 455,000 miles and 8 years of operation. • MARAD'S advanced nuclear propulsion system competitive now at high power levels. • Growing industry interest.
GERMANY  In service	• Government and industry aggressively pursuing international market. • Construction decision appears imminent on two 80,000 SHP containership jointly with JAPAN. • R & D continuing on advanced reactor technology for 1980-1990 propulsion systems.
JAPAN  Under construction	• Very aggressive government-industry program. • Construction decision appears imminent on two 80,000 SHP containerships jointly with GERMANY. • Projecting market of 300 nuclear containerships in period 1980-2000.
RUSSIA  In service  Under construction  Under construction	• N. S. LENIN was world's first nuclear powered ice breaker. • Two advanced nuclear powered ice breakers reported under construction. • Commercial propulsion activity not known.
NORWAY SWEDEN NETHERLANDS ITALY FRANCE UNITED KINGDOM	• DESIGN STUDIES ONLY • Varying degrees of interest and capability.

FIGURE 2.1 WORLDWIDE NUCLEAR PROPULSION ACTIVITY IN 1971

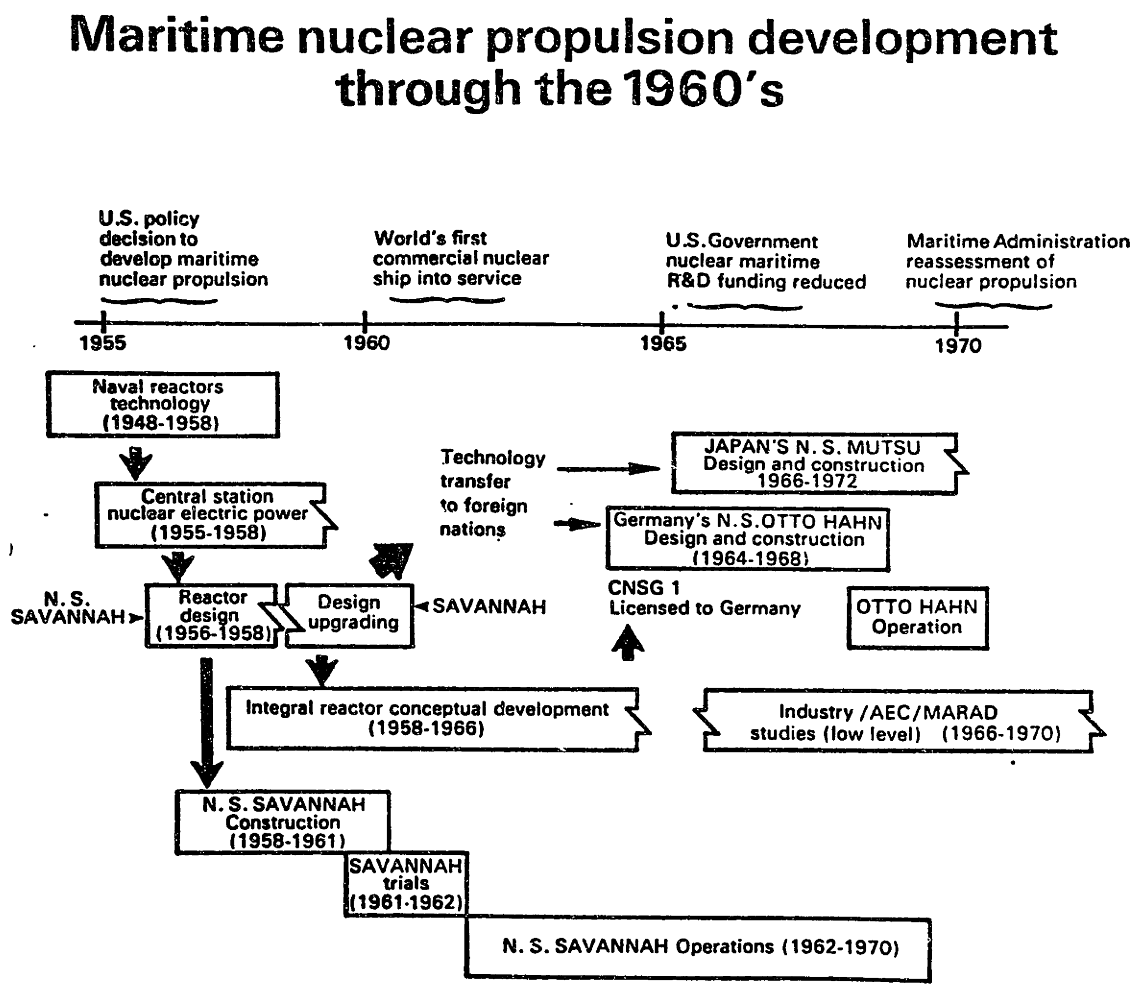

Maritime nuclear propulsion development through the 1960's
U.S. policy decision to develop maritime nuclear propulsion
World's first commercial nuclear ship into service
U.S.Government nuclear maritime R&D funding reduced
Maritime Administration reassessment of nuclear propulsion
1955
1960
1965
1970
Naval reactors technology (1948-1958)
Central station nuclear electric power (1955-1958)
N. S. SAVANNAH
Reactor design (1956-1958)
Design upgrading
SAVANNAH
Technology transfer to foreign nations
JAPAN'S N. S. MUTSU Design and construction 1966-1972
Germany's N.S.OTTO HAHN Design and construction (1964-1968)
CNSG 1 Licensed to Germany
OTTO HAHN Operation
Integral reactor conceptual development (1958-1966)
Industry /AEC/MARAD studies (low level) (1966-1970)
N. S. SAVANNAH Construction (1958-1961)
SAVANNAH trials (1961-1962)
N. S. SAVANNAH Operations (1962-1970)

FIGURE 2.2

TABLE 2.1

PRESENT & PROJECTED NUMBERS OF
NUCLEAR-POWERED SHIPS, JANUARY 1973[a]

Category	Active[b]	Planned[c]	Projected totals
Naval			
Submarines			
United States	104	20	140-170
Soviet Union	95	?	140-170
United Kingdom	12	3	?
France	3	2	?
Total Submarines	214	45[d]	280-340
Surface Combatants			
Aircraft Carriers (U.S.)	2	1	
Cruisers (U.S.)	1	-	
Frigates (U.S.)	4	3	
Total Surface Combatants	7	4	21[e]
Icebreakers (Soviet Union)	1	2	
Merchant - Experimental			
SAVANNAH (U.S.; laid up)	-		
OTTO HAHN (West Germany)	1		
MUTSU (Japan; not yet operational)	1		

[a] The "Active" and "Planned" entries for submarines, surface combatants, and icebreakers are from Blackman, Raymon V.B (ed.), Jane's Fighting Ships, 1972-1973 (New York: McGraw-Hill, 1972). The "Projected Totals" for submarines and surface combatants represent the Panel's projections and do not necessarily represent the views of Jane's Fighting Ships or the countries involved.

TABLE 2.1 (Continues)

b/ Includes those launched but not commissioned.

c/ Under construction or authorized.

d/ Assumes the Soviet Union and United States have comparable construction rates, 3 to 5 submarines per year. The Soviet rate may be somewhat higher.

e/ Assumes total will be about twice the presently proposed U.S. number: 1 nuclear powered cruiser plus 4 nuclear-powered task forces, each consisting of 1 carrier and 4 frigates.

TABLE 2.2

ENGINEERING FEATURES OF NUCLEAR-POWERED SHIPS[a/]

Category	Name	No. of (PWR)[a/] Reactors	No. of Shafts	Total Shaft Horsepower
Naval				
Submarines		1	1	15,000 - 30,000
Surface Combatants				
Carriers	ENTERPRISE	8	4	280,000
	NIMITZ, EISENHOWER	2	4	260,000
Cruiser	LONG BEACH	2	2	80,000
Frigates	BAINBRIDGE, TRUXTON	2	2	60,000
Icebreakers	LENIN	3	3	44,000
	ARKTIKA[c/]	2	?	30,000
Merchant	SAVANNAH[d/]	1	1	22,000
	OTTO HAHN	1	1	10,000
	MUTSU[c/]	1	1	10,000
	Projected			60,000 - 240,000+

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a/ Source, for naval vessels and icebreaker, is Jane's Fighting Ships, 1972-1973, op. cit. (See notes, Table E-1)

b/ PWR = pressurized water reactor

c/ Not yet operational

d/ In lay-up

Comparable figures for the seven U.S. nuclear-powered surface combatants are 48 ship-years through 1972 and 1.5 million miles through 1970. The United States has lost two nuclear-powered submarines at sea. The THRESHER sank on April 10, 1963 in the Atlantic Ocean in water 8,500 feet deep; the SCORPION sank between May 21 and 27, 1968 also in the Atlantic in more than 10,000 feet of water. It is suspected that the Soviet Union also has lost one or two nuclear powered submarines.

Also in operation today are the nuclear-powered Soviet icebreaker LENIN and the West German ore carrier OTTO HAHN. Both the SAVANNAH and the OTTO HAHN are experimental ships, as is the MUTSU. Thus, while neither West Germany or Japan is permitted to acquire nuclear-powered (or other) warships, each has built a nuclear-powered merchant ship to gain experience in this means of propulsion.

All the existing nuclear-powered ships use water-cooled reactors and all those in the U.S. nuclear fleet, and perhaps others as well, use pressurized water reactors (PWR).

The PWR is available in two general forms: the loop or dispersed type (Figure 2.3) as used, for example, in all land-based civil power stations or in the N.S. SAVANNAH and the integral type (Figure 2.4) as installed, for example, in the German prototype ship, the "OTTO HAHN". The integral type differs from the dispersed type in that heat exchangers and primary coolant pumps are within the main pressure vessel.

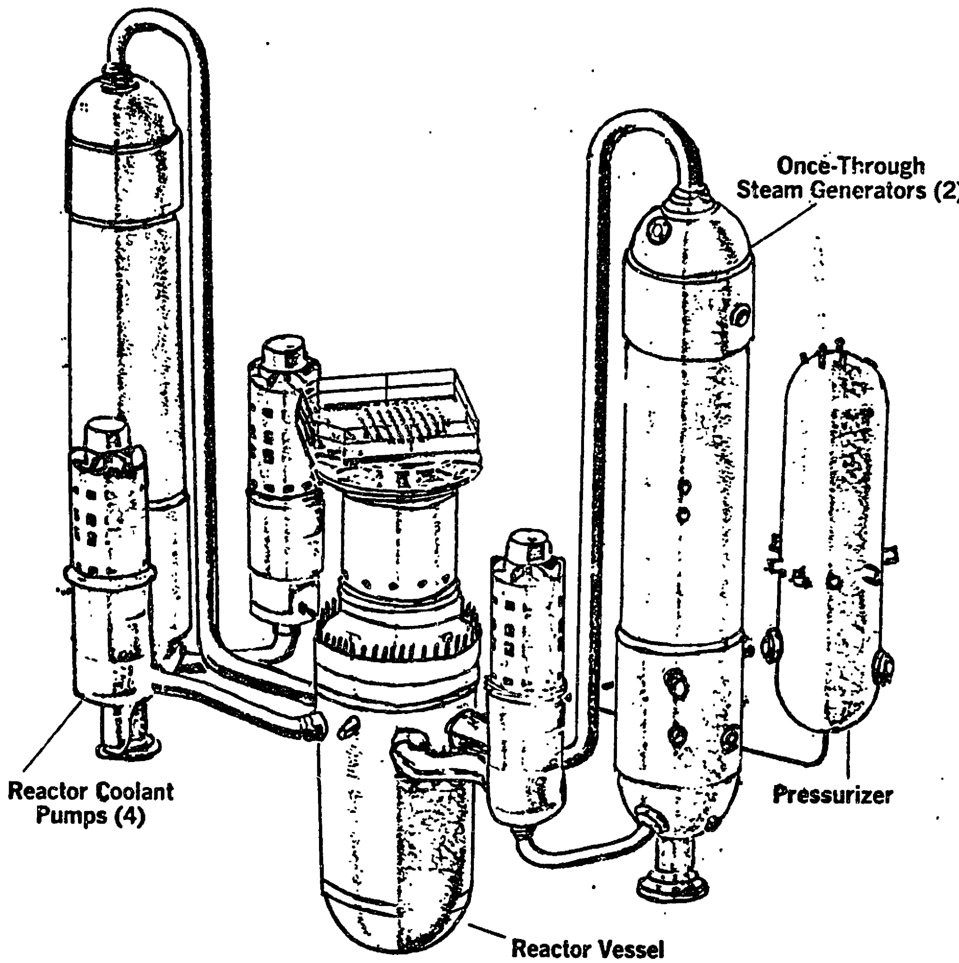
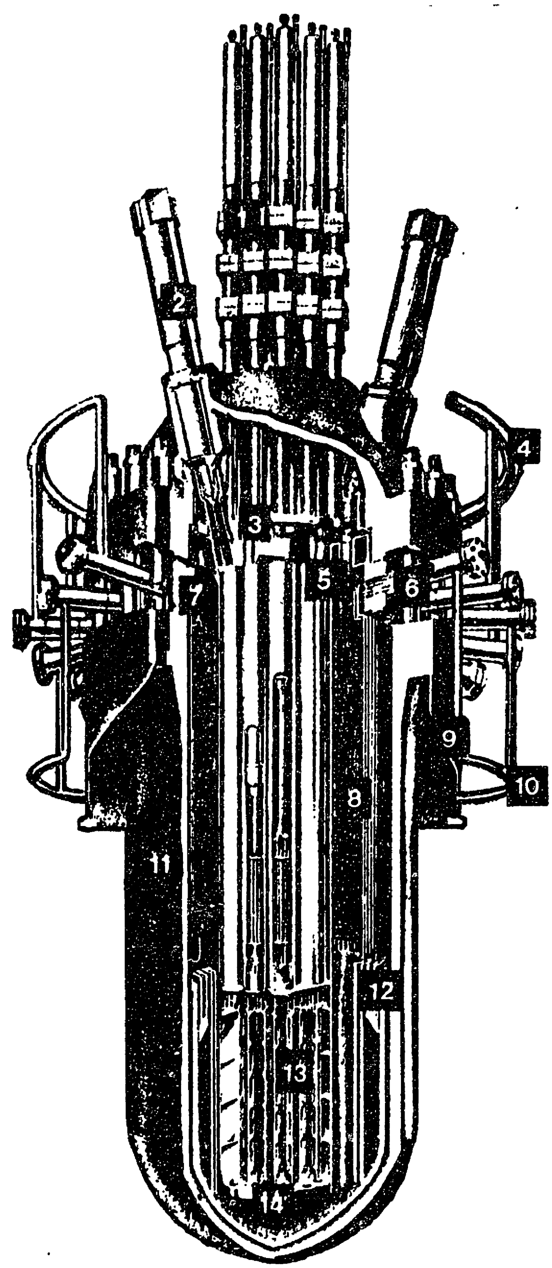


FIGURE 2.3 TYPICAL LOOP-TYPE PWR



1 Control mechanisms
2 Primary pump
3 Pre-tensioning assemblies
4 Steam main
5 Core support structure
6 Tube plate
7 Heat exchanger collar
8 Helical once through heat exchanger
9 Support skirt
10 Feed water main
11 Reactor vessel
12 Thermal shields
13 Fuel assemblies
14 Core support grid

FIGURE 2.4 TYPICAL INTEGRAL PWR

The integral PWR used for the OTTO HAHN has been in service for over six years. The "OTTO HAHN" has demonstrated the feasibility of PWR units up to 10,000 shp, the SAVANNAH (USA) up to 22,000 shp (loop-type) and the LENIN (USSR) with twin reactors up to 40,000 shp (loop-type).

Further design studies in Germany, Japan, the United Kingdom and the United States have confirmed the feasibility of the integral concept in the 40,000 - 110,000 shp range. Of the several variants that have been proposed, the EFDR system and the CNSG IV are in the most advanced stages.

The EFDR system (Figure 2.5), an integral PWR design with self-pressurization, is a development of the design proved in the "OTTO HAHN".

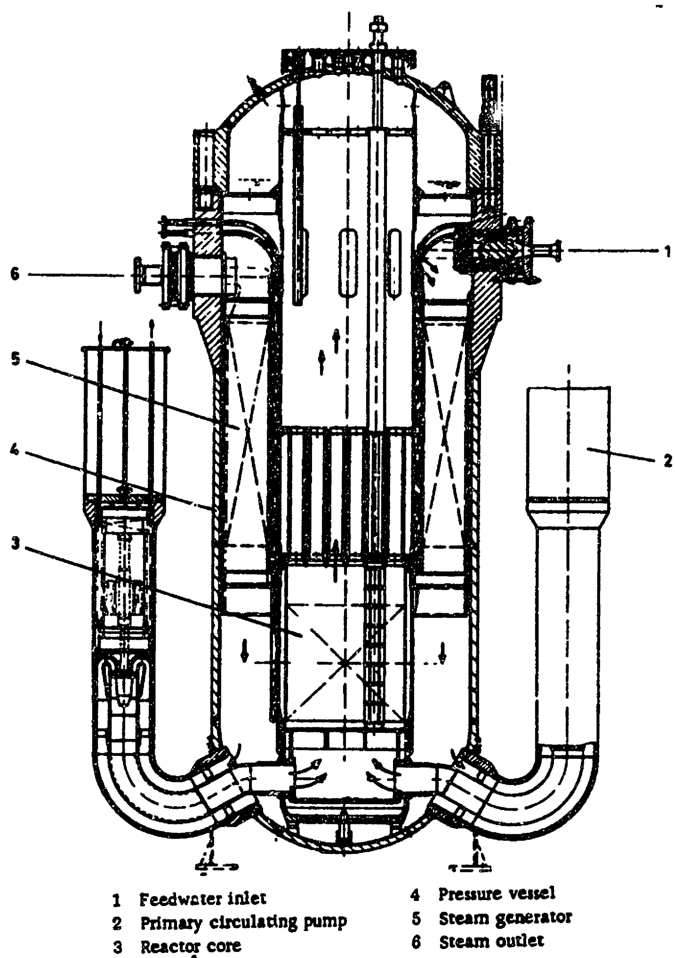
The CNSG (Consolidated Nuclear Stem Generator) is an integral PWR with separated pressurizer, proposed by Babcock and Wilcox (Figure 2.6) to the U.S. Department of Commerce, Maritime Administration for supply power to an ultra-large crude carrier (ULCC).

The only but major differences between these two integral PWR's are based on the facts that the German type is self-pressurized and the CNSG (Figure 2.6) uses an independent pressurizer connected to the pressure vessel by small diameter piping.

The present state of technology and experience thus suggests the integral PWR type as the most suitable for marine applications.

EFDR: Entwicklung Fortschrittlicher Druckwasser Reaktor
: Development of Advanced PWR

FIGURE 2.5 MARINE EFDR PRIMARY SYSTEM SELF-PRESSURIZED



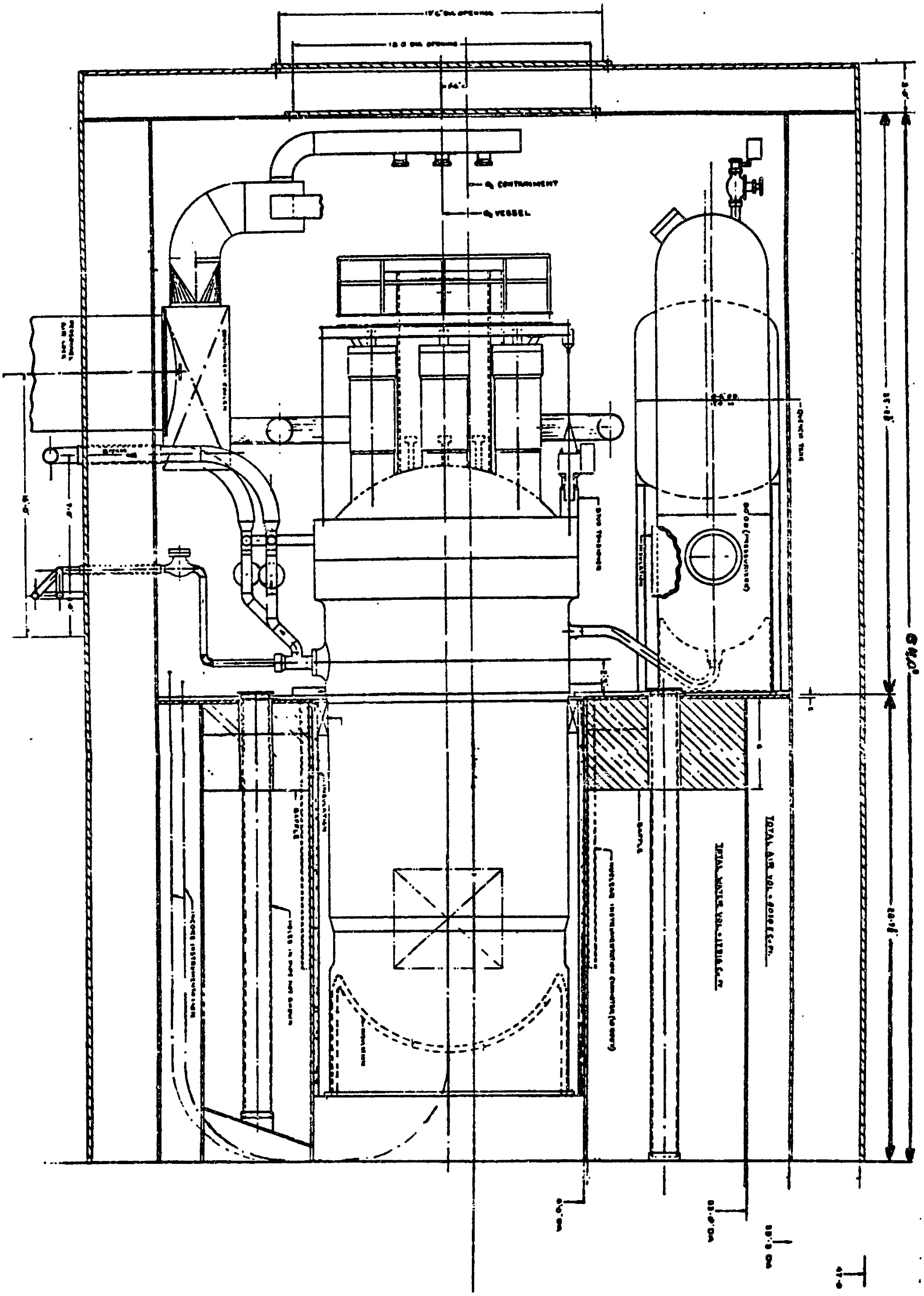


FIGURE 2.6 MARITIME INTEGRAL PWR WITH SEPARATED PRESSURIZER (CNSG)

Comparing the integral design with the loop-type design of pressurized water reactor, the following advantages for the integral type can be pointed out:

1. The robust construction and high availability of the integral system has been proven on numerous voyages of the "OTTO HAHN". The inherent stability and easy control of the reactor system are of special importance for a nuclear ship. This is an advantage for automatization.
2. The integral design seems to be economical
   - The integration of the primary system into the pressure vessel leads to a weight and space-saving construstion.
   - It is not necessary to connect primary components by means of high pressure pipes with large diameter characteristic of the loop-type design. There are no problems with the support and the thermal compensation as in the loop system.
   - The compact primary system can be enclosed in a compact containment of smaller diameter. Therefore, the integral system is favorable for containment with a pressure-suppression system.
   - Pre-assembling of larger parts is possible as known from experiences of the N.S. "OTTO HAHN". The internals including the steam generator are assembled in the factory's workshop in the pressure vessel under clean conditions. The complete

pressure vessel with internals will then be brought as a whole onboard the ship for installation into the containment.

3. From the safety point of view, the integral system is favorable because of the following:
   - There exist no large diameter primary pipes as in the loop-type. The only potentially critical occurrence is a break in the small pipes to the water make-up system in the EFDR type or the connection pipe to the pressurizer in the CNSG.
   - At a LOCA (Loss of Coolant Accident) due to such a failure, the steam flow rate into the containment is much smaller compared to a corresponding accident of a pressurized loop-type reactor. The resulting stress for the internals of the pressure vessel, especially for the core structure is insignificant. The water level above the core will sink slowly. The level can be maintained by a reasonable emergency primary water feed system. Corresponding to the small flow rate during the LOCA, easier conditions result for the layout of the safety containment leading to a less heavy construction.
   - The so-called cold water accident cannot occur.
   - The integral system has the possibility of natural circulation by convection in the primary system.

There exists a limit to the size of integral systems due to manufacturing difficulties of the reactor vessel in the power range of more than about 200 Mw(e). However, this is not a problem for maritime applications since 200 Mw(e) greatly exceed the maximum power requirement of the biggest ship that can be designed up today (See Table 2.3).

Hereafter, the technological differences described between a maritime reactor and a stationary reactor will be based essentially on an Integral PWR (Figure 2.7) used in ship propulsion and a Loop-Type PWR (Figure 2.8) used as a land-based plant.

TABLE 2.3

MAXIMUM CARGO CAPACITIES, BY SHIP TYPE, OF SHIPS BUILDING OR ON ORDER, MID-1973[a/]

Ship Type	Cargo Deadweight (Long Tons)	Shaft Horsepower[b/]	Nation & Shipyard
Dry Bulk:			
Bulk/Ore	115,900	23,200	Japan: Mitsui (Tamano Works)
Bulk	145,000	32,000	West Germany: Blohm & Voss (Hamburg)
Ore	164,000	32,000	Japan: IHI[c/] (Aioi Shipyard)
Combination Bulk:			
OBO[d/]	168,000	28,000	Japan: Sumitomo (Tokyo)
Ore/Oil	275,000	38,000	Sweden: Eriksberg (Gothenburg)
Tanker	540,000	64,800	France: Chantiers de l'Atlantique (St. Nazaire)

a/ "Vessels of 1,000 or more Gross Tons Building or on Order in World's Shipyards, as of April 1, 1973," Marine Engineering/Log, Annual Yearbook Issue, June 15, 1973, pp. 125 ff.

TABLE 2.3 (Continuation)

b/ This is the corresponding shp for the maximum-deadweight ship in each category - not necessarily the maximum shp for that category.

c/ Ishikawajima-Harima Heavy Industries

d/ Ore/bulk/oil carrier

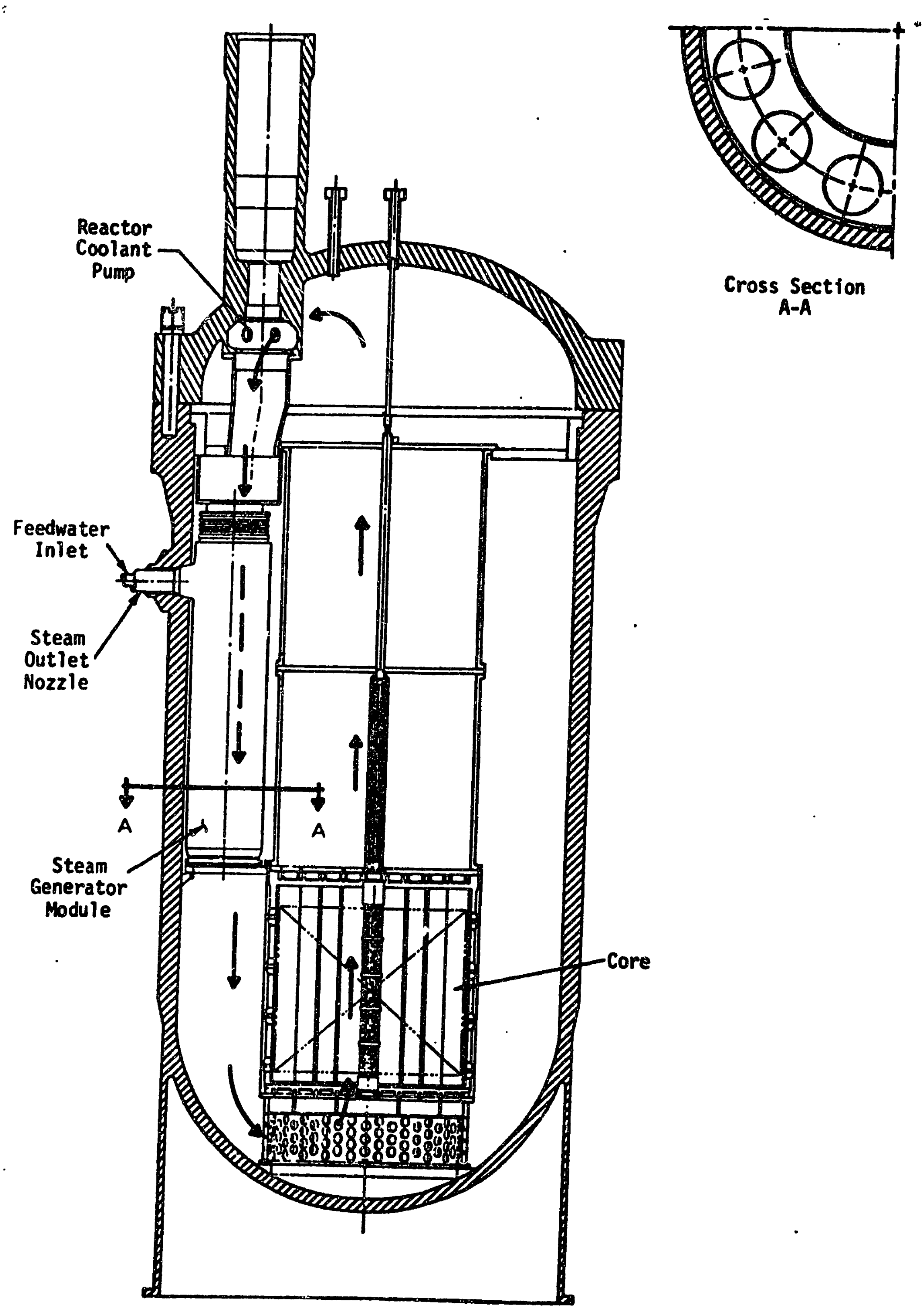


FIGURE 2.7 ELEVATION VIEW OF INTEGRAL CNSG IV A FOR MARITIME APPLICATION

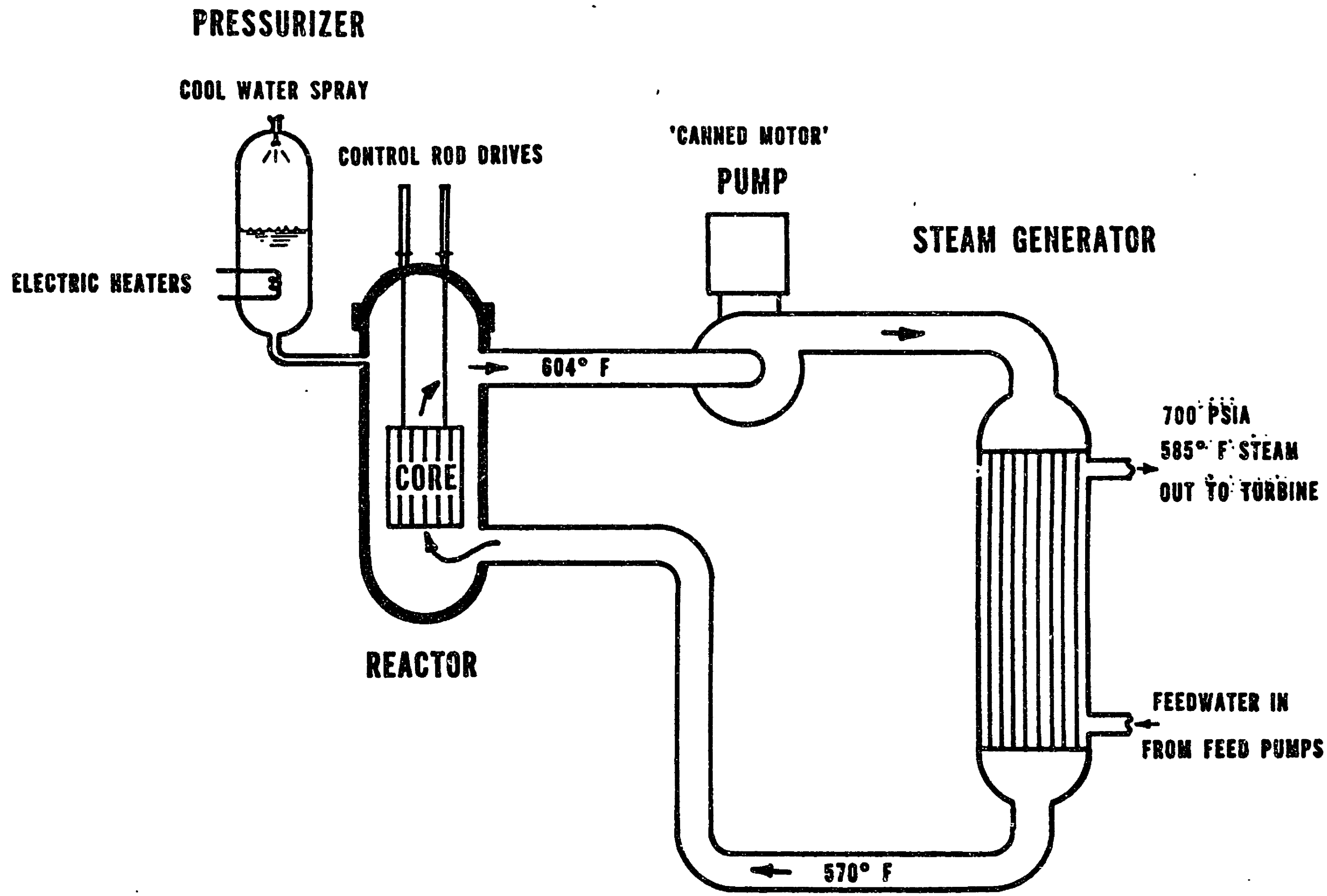


FIGURE 2.8 SIMIPLIFIED LOOP-TYPE PWR DIAGRAM

CHAPTER III

TECHNOLOGICAL DIFFERENCES DUE TO OPERATIONAL FACTORS

3.1 Introduction

It is of interest to make a comparison, from the standpoint of operational factors, of the technological differences of a nuclear ship with those of a stationary steadily operated power reactor. Since nuclear ships are mobile, they are operated under widely varying circumstances compared with stationary steadily operated power reactor.

There are many more start-up and shut-down as well as widely variable needs for power in maneuvering which results in more stress and strain on the machinery.

The maneuverability that a maritime reactor demands is bigger than that which a stationary reactor requires.

3.2 Maneuverability

Due to the very stringent maneuvering requirements in the maritime plant and the high differential control rod worth, the reactor control system design for a nuclear ship board application offers some peculiar difficulties in comparison with land base-load plants.

The load variation requirements which can be up to 50 times more rapid than required for land-base load plants are difficult to meet due to the following:

(1) poor heat transfer characteristics of a ceramic fuel,

(2) the step by step control rod movement provided by the magnetic jack type control rod drive mechanisms used in stationary reactor, and

(3) the high incremental worth of control rod.

The use of this control type may result in an oscillatory control rod motion in an attempt to maintain equilibrium within power and temperature dead bands.

It follows from the above consideration that a magnetic jack control rod drive mechanism cannot be used in reactors for maritime use. Since the electro-hydraulic control rod drive system has had a lot of operational, maintenance, economic and safety problems [this system was used by N.S. SAVANNAH], the only type of control rod drive mechanism available is the roller-nut drive mechanism. Thus, while the stationary reactor uses magnetic jack and roller-nut drive mechanisms to drive their control rod assemblies, the maritime reactor uses only the roller-nut drive mechanisms. This roller-nut drive mechanism differs also in some part of the design from the roller-nut drive mechanisms used in stationary reactors.

It also follows that a reactor control system for a shipboard plant should have such characteristics that it acts promptly and efficiently in the first part of an operational transient, in order to obtain a prompt gross regulation, and should be able to perform a fine regulation so that it can be prevented from approaching the unsafe conditions and oscillation around the equilibrium. The conventional PWR reactor control system, used in the stationary reactor, employs as basic control variable an average coolant temperature signal. This signal is inadequate to fulfill the fine regulation. In maritime reactors besides the usual temperature error, a signal is needed to "feel" the transient as soon as it is starting and "disappear" at the end of the load variation when only the temperature error signal should be active. The difference between turbine load request (approximately proportional to secondary steam mass flow rate) and core generated power (proportional to neutron flux) is then employed as prompt signal.

Due to the differences in ramp load changes under automatic control, the reactor coolant system must provide different conditions in such a way that relief valve or turbine by-pass action must be avoided.

Because of the same problem the maritime reactor is controlled manually at power level less than 20% of the total power while a stationary reactor is controlled manually at power level less than 15% of the total power.

Since power maneuvering in maritime applications will be more severe than for stationary, the transient used in the design and fatigue analysis of the components of the reactor cooling pressure boundary will be different and, therefore, the operating transient design cycles for normal upset, emergency, faulted and testing conditions will be very different from maritime to stationary and will be different depending on the type of ship (tanker, container, bulk).

3.3 Stationary Reactor Control Rod Drive Mechanisms

The control rods assembly (Figure 3-1) in the stationary type PWR are driven by two types of mechanisms: the magnetic jack and the roller-nut.

3.3.1 Magnetic Jack Drive Mechanism

The full length control rod clusters are positioned by latch type magnetic jack drive mechanisms (Figure 3-2). The full length control rod cluster consist of five major components:

- Pressure Housing
- Operating Coil Stack
- Interval Latch Assembly
- Position Indicator Coil Stack
- Control Rod Cluster Drive Shaft

All moving components are contained in a stainless steel pressure housing attached to a head adapter. The adapter is welded to the reactor vessel head constituting an integral part of the vessel. The housing is completely free of mechanical seals, electric lines and hydraulic penetration.

The operating coil stack consists of three independent coils installed over the outside of the pressure housing. The stack can be removed and replaced while the reactor is pressurized, because of rests on a shoulder of the pressure housing without any mechanical attachment.

Inside the pressure housing the internal latch assembly of the mechanism, consisting of gripper latches and armatures is located.

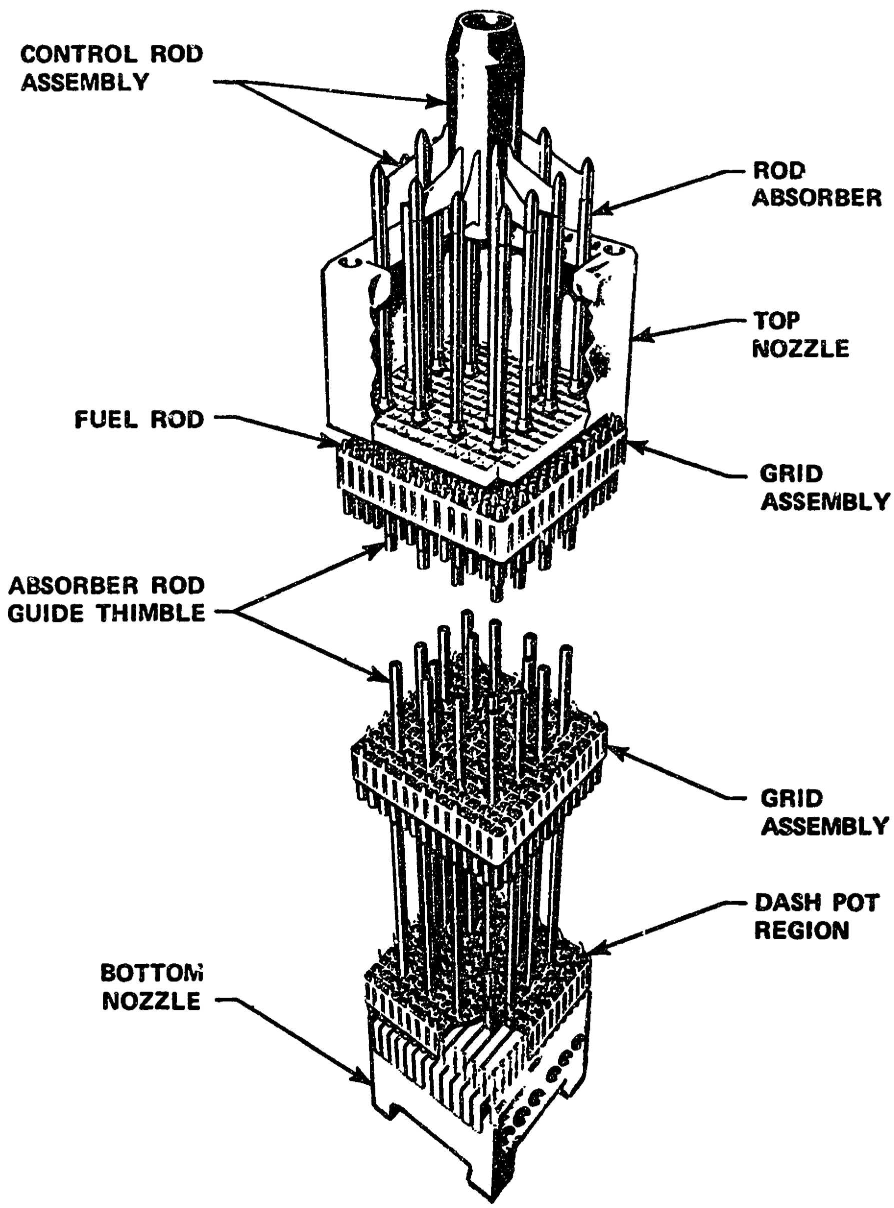


FIGURE 3.1 CUTAWAY OF TYPICAL ROD CLUSTER CONTROL ASSEMBLY OF STATIONARY PWR'S

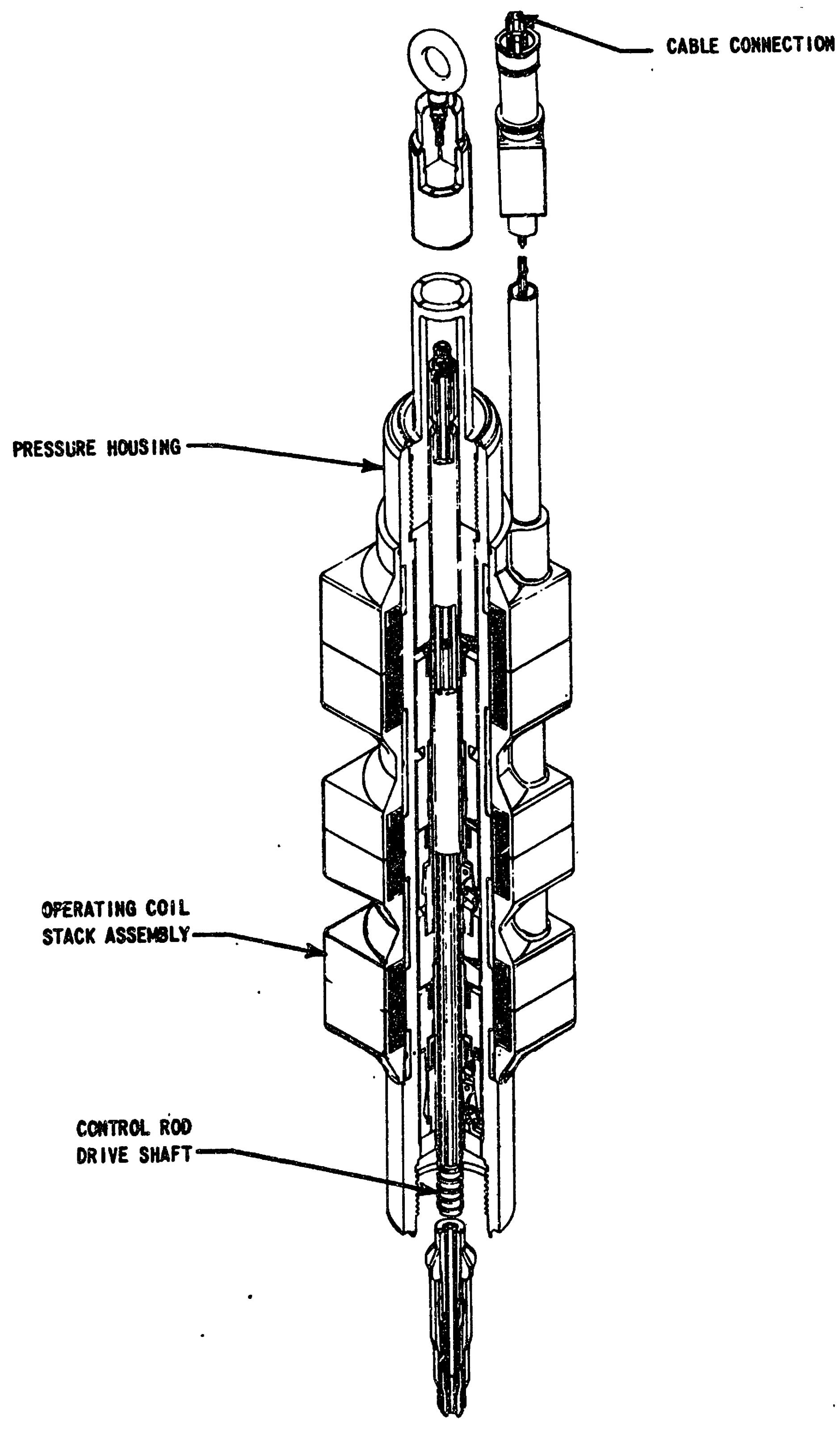


FIG. 3.2 CUTAWAY OF FULL LENGTH CONTROL ROD DRIVE MECHANISM OF STATIONARY PWR'S

During normal plant operation, the drive mechanism serves only to hold in position the control rod assembly that has been withdrawn from the core. Under this condition, only one coil is energized to engage its respective latches with the drive shaft. If power to this coil is cut off, either deliberately in the reactor trip or due to an accident power failure, the control rod cluster instantly falls by gravity into the core.

Circumferential grooves in the drive shaft engage the latches of the drive mechanism. A coupling attached to the lower end of a drive shaft connects to a control rod cluster and measures are provided for remotely engaging or disengaging this coupling from the control rod cluster.

3.3.2 Roller-Nut Type Drive Mechanism

The part length control rod cluster are positioned by roller-nut type drive mechanism (Figure 3-3) mounted on the reactor vessel head. These consist of the following components:

- Pressure Housing
- Internal Rotor Assembly
- Position Indicating System
- Drive Shaft Assembly

The pressure housing consists of the rotor assembly housing and the rod travel housing. The internal rotor assembly is the operating center of the mechanism. The rotor assembly is free to rotate within the pressure housing

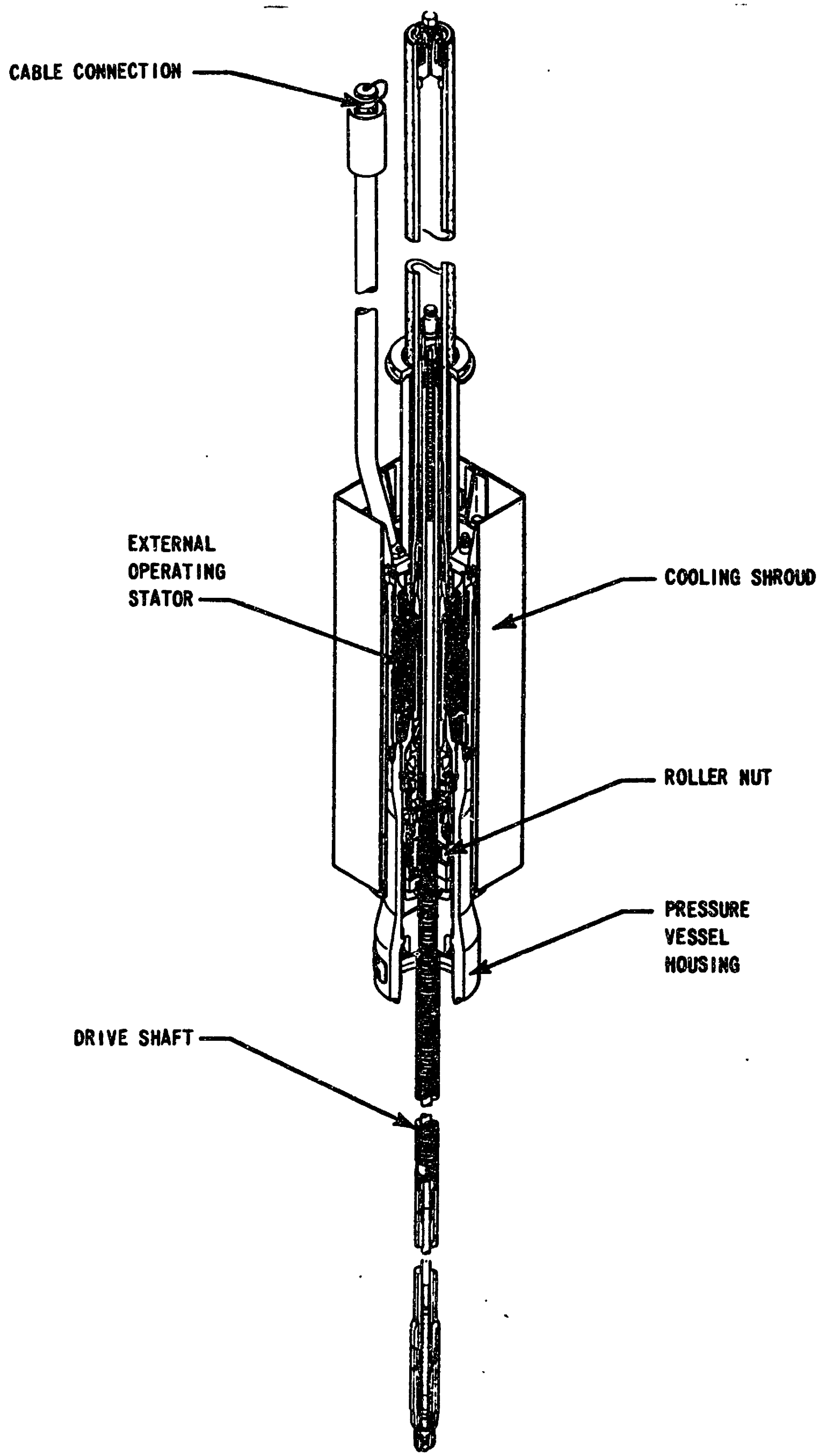


FIGURE 3.3 CUTAWAY OF ROLLER-NUT TYPE DRIVE MECHANISM OF STATIONARY PWR'S

between a lower thrust bearing and an upper radial bearing. Five free-rotating roller-nuts are held captive in the lower cylindrical portion of the rotor and are counted to match the lead angle of the drive rod threads. As the internal rotor assembly rotates, the roller-nuts turn within the thread of the drive rod, translating vertical motion to it, much as a turning nut would cause a bolt to raise or lower in a slot which prevents the bolt from rotating. The stator provides rotational energy to the rotor assembly after first releasing the brake which is in the braking position with the motor de-energized.

A coupling attached to the lower end of the drive shaft connects to the spider body at the top of the control rod. This coupling can be engaged or disengaged remotely from the spider body.

The unique feature of the part-length drive is its ability to hold and maintain control rod position in the event of a power interruption or a complete power failure. Thus, the part length rods remain in position during a reactor trip.

3.4 Maritime Reactor Control Rod Drive Mechanism

The control rod drive mechanisms now used in maritime reactors (Figure 3-4) are based on roller-nut type control rod drive mechanisms similar to that used to drive the part length control rod assembly in land-based reactors. Modifications will adapt these drive mechanisms for the special requirements presented by shipboard operating conditions.

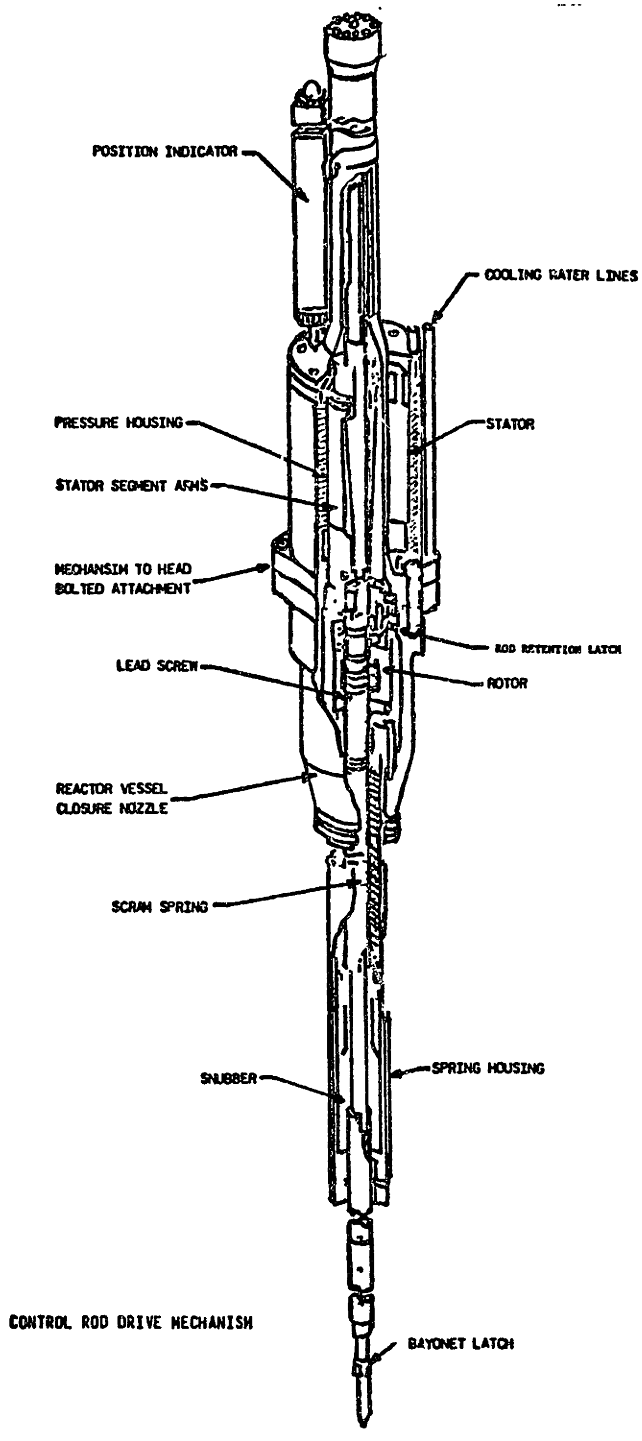


FIGURE 3.4 ROLLER-NUT TYPE CONTROL ROD DRIVE MECHANISM FOR MARITIME REACTOR

The control rod drive mechanism consists of a motor tube assembly, a lower mechanism assembly and a lead screw and extension assembly. The motor tube houses a rotor assembly. The motor tube is closed on the upper end with a closure and vent assembly and is part of the primary pressure boundary. An external motor stator surrounds the motor tube, and position indicators are arranged outside the upper extension of the motor tube. The control rod drive output element is a non-rotating, translating lead screw coupled to the control rod. This is the location of the main difference from the stationary reactor which is not only due to an operational problem but an environmental and safety problem as we will see later.

The screw is driven by separating antifriction roller-nut assemblies attached to segment arms. They are magnetically rotated by the motor stator located outside the pressure housing. Stator current causes the separating roller-nut assembly halves to close and engage the lead screw. Mechanical springs disengage the roller-nut halves from the screw in the absence of current. For rapid insertion, the nut parts separate and release the lead screw and control rod. A mechanical spring, aided by gravity, forces the control rods into the core. A hydraulic snubber located at the end of the spring housing decelerates the control rod assemblies to a slow speed near the full-in position.

CHAPTER IV

TECHNOLOGICAL DIFFERENCES DUE TO ENVIRONMENTAL FACTORS

4.1 Introduction

Nuclear ships are operated in many different environmental situations. Ships at sea are subject to various kinds of motion, such as heaving, rolling, pitching, yawing, surging, and swaying, each of which will give rise to acceleration forces on the ship structure. Spectrum analysis of ocean waves indicates that regular cycle motions in a ship at sea are of quite small amplitude and will not give rise to the serious repetitive accelerations which could affect the functioning and stability of a nuclear reactor.

In heavy weather, however, the resulting acceleration forces on a shipboard reactor and its systems are the most severe of those expected from any source. The frequency of encountering heavy weather at sea is not a matter of precise determination. The nature and extent of the forces prevalent during extreme conditions of heavy weather are also difficult to ascertain and difficult to measure. Thus, the reactor and the ship are designed in accordance with conservative specifications for roll, pitch, heave and "g" (acceleration) forces which should seldom be encountered in service.

Ships can be submitted to deceleration forces in the case of collision or grounding. The generated power in a marine propulsion plant is much smaller than in a land-based station. The ship is submitted to vibration due to propellers and its own frequency. The ship has plenty of water surrounding it compared with stationary plant, but this water is saline.

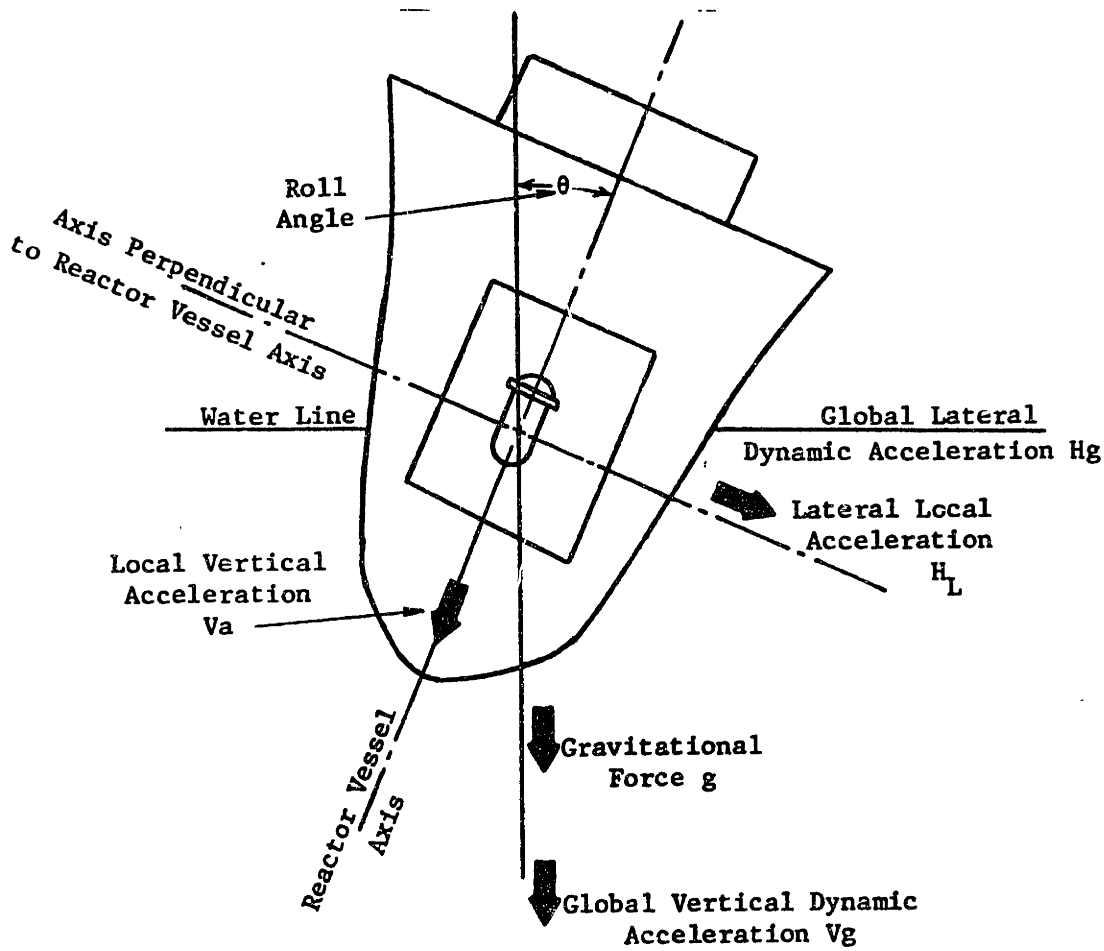
4.2 Ship Motion & Accelerations

Due to wave-induced ship motions, the reactor installation is subject to accelerations in all directions (see Figure 4.1). For this reason, special consideration must be given to:

1. Dynamic Behaviour of the Reactor System
2. Strength Calculations of the Reactor Vessel Support & Other Components of the Installation.

Ship motion induces forces on ship components due to the accelerations of pitching, rolling and heaving (and to a le-ser extent, yawing, surging and swaying), and to the inertial effect of the mass of various parts. The maximum velocity occurs at the midpoint of the motion and diminishes to zero at the end of the motion. The maximum acceleration for each motion occurs when the velocity is zero, and the maximum dynamic effect in the direction of the motion occurs at the end of the motion. Therefore, a centrifugal force may be developed in a ship with a very fast roll period. This force will be in a line between the component and the center of roll. In large ships forces due to yawing, surging and swaying are not expected to be significant.

It should be clearly understood that the dynamic forces considered are limited to those due to accelerations arising



$$Va\ (max) = Vg\cos\theta + 32.2\cos\theta + Hg\sin\theta$$

$$Va\ (min) = 32.2\cos\theta - Vg\cos\theta - Hg\sin\theta$$

$$H_L\ (max) = Vg\sin\theta + 32.2\sin\theta + Hg\cos\theta$$

FIGURE 4.1 DIRECTIONS OF VARIOUS COMPONENTS OF INERTIAL FORCES SEEN BY REACTOR DUE TO SHIP MOTION from ship motions. To these forces the static force components due to gravity must be added or substracted to obtain the net force acting on a particular part of the ship.

4.2.1 Forces Due to Pitching

Pitching is the rotation of the ship about a horizontal axis perpendicular to the direction of motion of the ship. The precise location of the pitch axis is difficult to establish since it is a function of ship mass distribution and water plane characteristics that in seaway vary with wave action and ship motion. Forces due to pitching are tangential to the arc of rotation and are calculated with the ship in the maximum up-pitch or down-pitch condition. The rotation is assumed to be a simple harmonic motion. The maximum acceleration due only to pitch should be for one particular ship calculated on the basis of simple harmonic motion with a roll period for seven seconds and a single amplitude roll of six degrees.

4.2.2 Forces Due to Rolling

Rolling is the rotation of a ship about a horizontal axis parallel to the ship's center line. Its location, for the same reason as in pitch, is difficult to establish. Forces due to rolling are tangential to the arc of rotation of the item being studied and arc maximum when the ship is in the maximum rolled angle. The rotation is assumed to be a simple harmonic motion.

The maximum accelerations due only to roll should be calculated on the basis of simple harmonic motion and the ship's natural period with roll angles between zero and 45 degrees single amplitude. In determining the total roll-induced loads when the equipment is above the roll center, the acceleration component is additive to the deadweight component and the natural period of roll should be the shortest that can occur in any condition of ship's loading. In determining the total roll-induced loads when the equipment is below the roll center, the acceleration component is substracted from the deadweight component and the natural period should be the largest that can occur, thus minimizing the negative acceleration effect.

4.2.3 Forces Due to Heaving

Heaving is the vertical translation of the ship's center of gravity. The amplitude and the natural period of heave are established by the ship's mass and waterplane characteristics. Acceleration due to heave is calculated by assuming a simple harmonic motion with maximum acceleration occurring at the extremities of the heave.

The maximum acceleration due only to heave should be calculated on the basis of a heave of L/80 half amplitude in a period of eight seconds where L is the length of the ship.

4.2.4 Forces Due to Combined Motion

The probability of experiencing ship motion that would cause all previously described maximum effects to be additive is considered to be small. Heave and pitch periods are close and in a regular wave train the motion of heave and pitch can be coupled with a lag of up to 100 degrees. However, a directly additive effect of pitch and heave might occur.

Roll is independent of pitch and heave and the maximum effect will coincide with the maximum effect of heave and pitch randomly. The pitch and heave angles and periods are used as stated previously. A single amplitude roll of 30 degrees in a period of ten seconds should be used for the roll force.

The classification societies* require that the nuclear reactor plant will operate satisfactorily under all seagoing conditions.

4.2.5 Differences in Design Rules of Today

In accordance with the rules of the American Bureau of Shipping, a reactor installation must operate satisfactorily under the following ship inclinations and accelerations.

Vessel Accelerations: Vector sum of maximum acceleration due to heave, pitch and roll.

Athwartship Accelerations: Vector sum of maximum acceleration to roll, sway and yaw.

Fore and Aft Acceleration: Not less than 1g.

Roll Motion: 30° to each side.

Pitch Motion: 10° up and down.

Permanent List 15° to port or starboard.

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*U.S.C.G. United States Coast Guard
A.B.S. American Bureau of Shipping

Permanent Trim	5° down by head or stern.

The reactor safety system must operate with:

Roll Motion	45° to each side.
Pitch Motion	12° up and down.
Permanent List	45°
Trim	10° down by head or stern.

Furthermore, a scram must occur when the list is greater than 45° or the trim greater than 12° or the containment is flooded.

This guide was developed by American Bureau of Shipping (ABS) in 1962 after construction of the N.S. SAVANNAH, and since no United States nuclear merchant ship has been constructed in the interim and has not been tested against actual design and construction. Now, it has become necessary to carefully review this guide for direct applicability to the ships and power plant to be developed and built.

To obtain more knowledge about the accelerations to which vessels are subject, theoretical investigations model testing and measurements onboard ships have been performed. As a result of these investigations, it can be said that the accelerations to which reactor plants are subject depend on:

a) the place in the ship of the reactor plant

b) ship's speed

c) size of the ship

Accelerations produced by ship motions affect the reactor significantly, particularly in the fuel element and control rod drive mechanism area.

Based on the calculation principles of rolling, pitching and heaving described above, studies made* for large ships such as are necessarily under consideration with nuclear propulsion arrived to the following preliminary loads.

Condition	Acceleration,g	Cycles/25 yrs.	Period in Secs. Est.	A.B.S.
45 deg. roll	0.70 lat.	125	14	10
30 deg. roll	0.50 lat.	3,125	14	10
15 deg. roll	0.26 lat.	75,000	14	10
10 deg. roll	0.17 lat.	3,125,000	14	10
5 deg. roll	0.1 lat.	6,250,000	14	10
6 deg. pitch	0.23 vert.	312,500	7	7
12 ft. heave	0.22 vert.	312,500	8	8
Combination Motion	1.53 vert.up	625		
Combination Motion	-0.43 vert.down	625		

NOTE: Amplitude are all single amplitude

Periods are for complete cycles

Acceleration includes deadweight except for pitch and heave

Roll periods are for adverse loadings

ABS (American Bureau of Shipping

EST (Estimation)

Cycles have been estimated based on probable trade routes and expected sea conditions over the ship usage periods on the routes.

German Lloyd (GL) requires for support structure and safety system a value of 1g for additional accelerations. For consideration of shock stresses, German Lloyd and Det Norske

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*G. G. Sharp Company, Naval Architects, New York, under subcontract to B&W.

Veritas requires a capability to sustain a load of the order of 1g while Lloyds Register very conservatively requires a load of 3g.

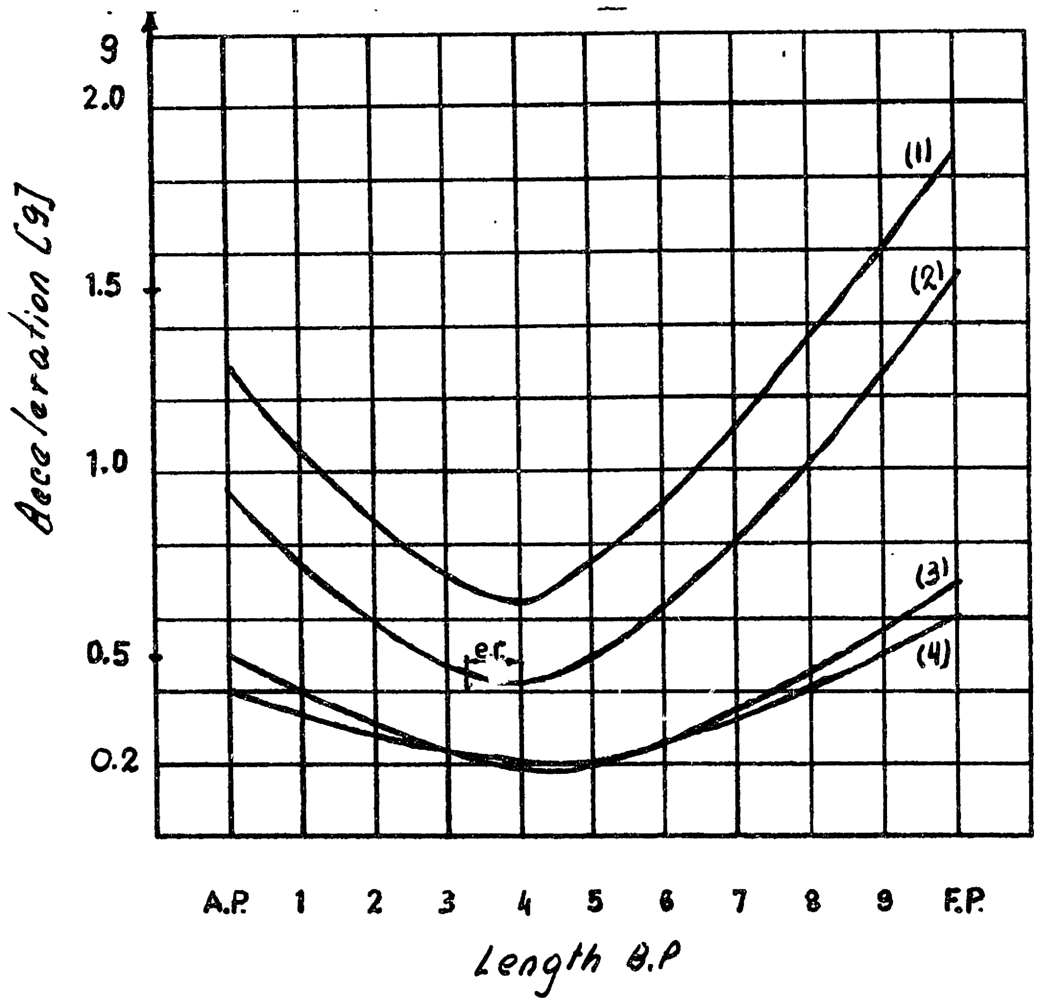
On the other hand, a study made by Reactor Centrum Nederland (see Figure 4.2) found a maximum vertical acceleration of 0.43g for a reactor installation in a long container vessel.

Figure 4.3 shows measurements which have been made on the N.S. "OTTO HAHN" and calculation made for a fast containership. The measurements in the N.S. "OTTO HAHN" have been made under most adverse conditions which until now have been met by the ship: wind forces of ten Beaufort from ahead, full power and wave heights of 26 to 30 feet. The acceleration values were 0.6g on the fore end and 0.4g on the after end of the ship. In the reactor area ±0.2g were measured.

Figure 4.4 shows a typical longitudinal distribution of additional accelerations generated by combined pitching and heaving in OTTO HAHN.

These measurements should show that for big ships, as reactor ships in any case will be, those values established by A.B.S., G.L., N.V. and LL.R and that calculated by B&W for the CNSG are far over-estimated.

Anyway, it will be very important to find a realistic value of acceleration since the use of lesser values can conduct to a serious failure of the reactor plant and an over estimated one can represent only unnecessary cost increases,



	Length (m)	Displacement (ts)
Container ship	268	51,000
Tanker	245	87,500
Japanese Model	116	10,320
N.S. "OTTO HAHN"	172	25,180

(1) Japanese-Model calculated & measured value

(2) Large Container Ship. Calculated value by Reactor Centrum Nederland

(3) Tanker Ship. Calculated Value by Reactor Centrum Nederland

(4) "OTTO HAHN" measured values

FIGURE 4.2 COMPARISON OF MEASURED & CALCULATED VERTICAL ACCELERATION DUE TO SHIP MOTION

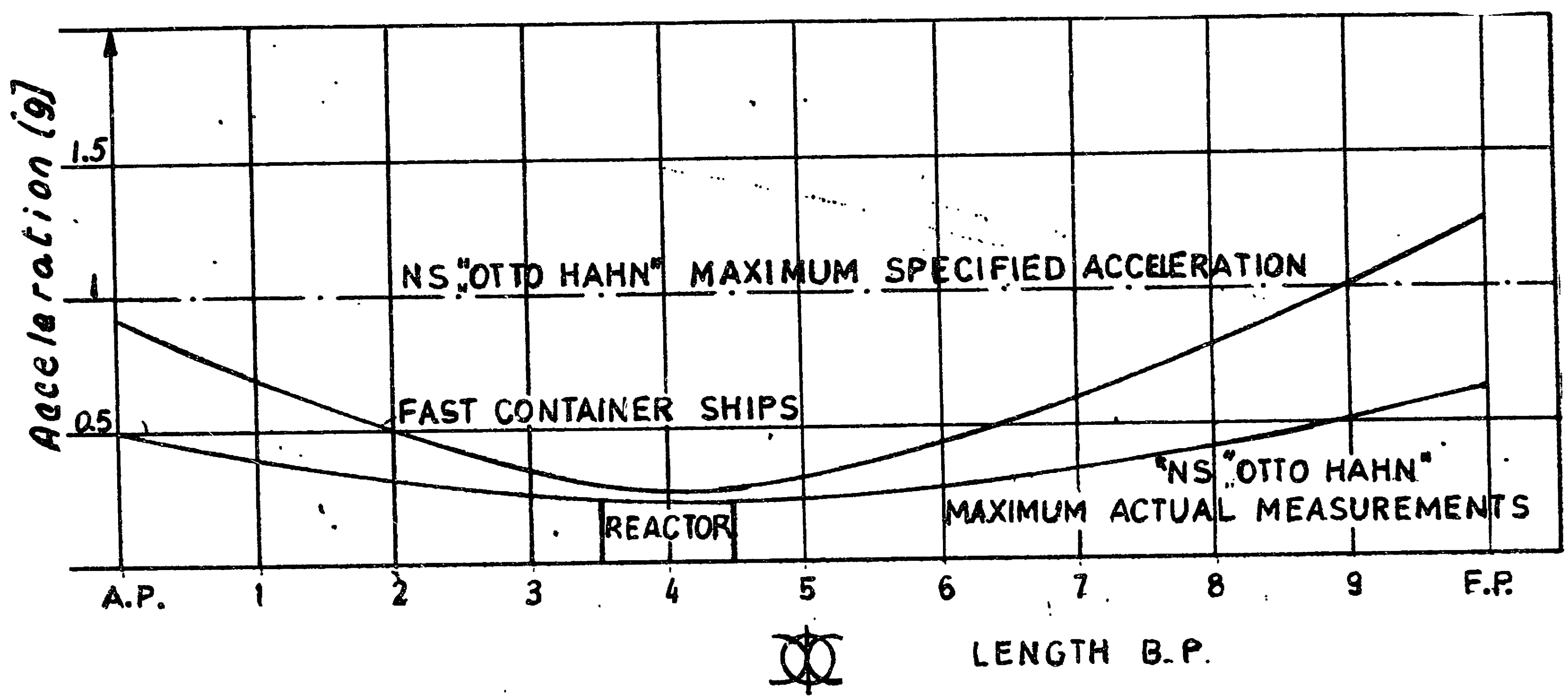


FIGURE 4.3 VERTICAL ACCELERATION OVER RELATIVE LENGTH OF SHIP

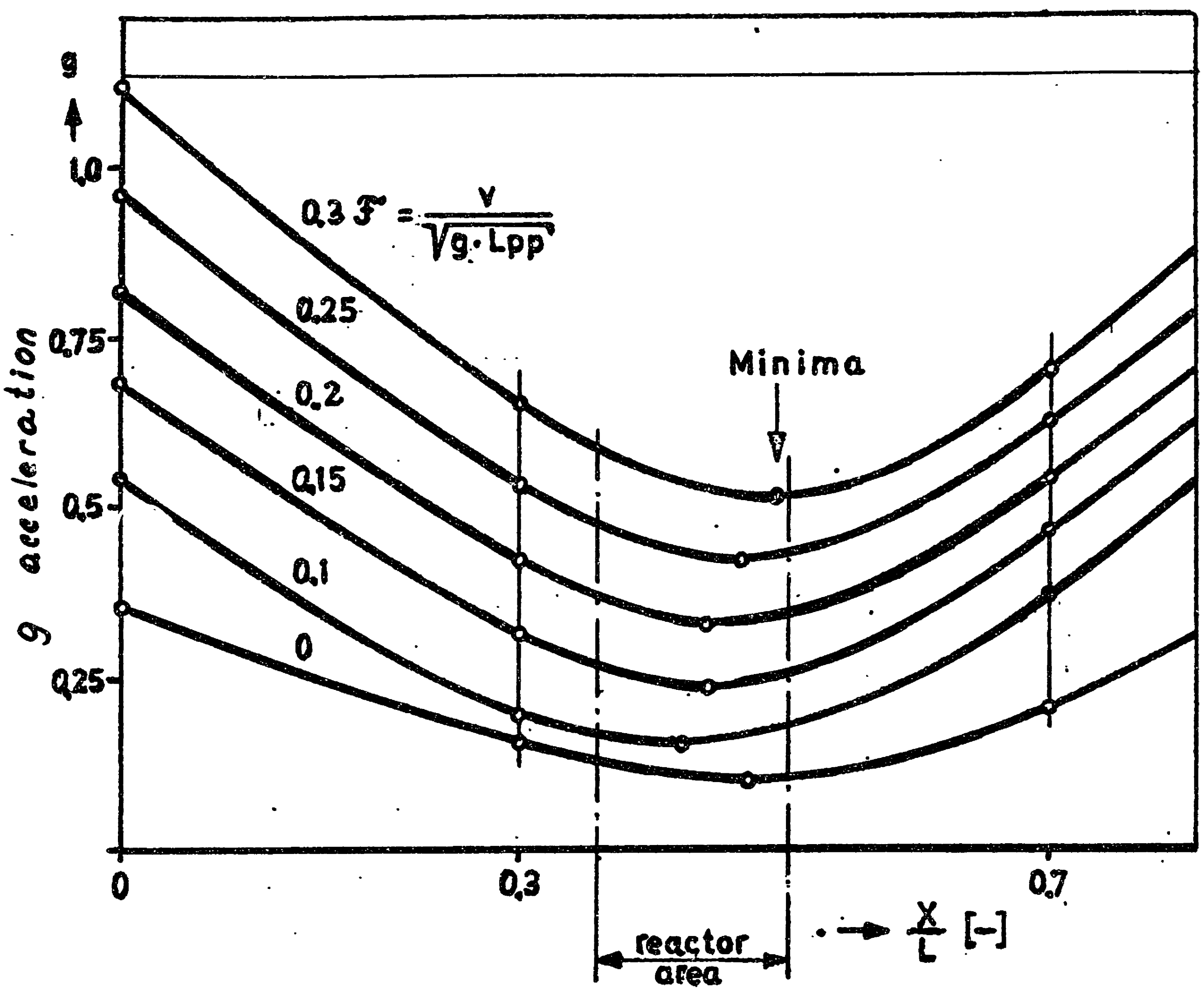


FIGURE 4.4 HEAVE & PITCH ACCELERATIONS FOR N.S. OTTO HAHN BASING ON CALCULATIONS OF INSTITUT FUR SCHIFFBAU, HAMBURG

because, for instance, of stronger and heavier support structures than are required.

4.3 Reactor Parts Affected by Ship Motion

The ship motion will affect the design of

- Steam Generator
- Control Rod Drive System
- Foundation and Retaining Structures
- Fuel Rod Assembly
- Reactor Vessel with RCS, Internals & other Accessories
- Reactor Coolant Piping
- The Pressurizer
- Reactor Coolant System Instrumentation
- Containment
- Control Rod Assembly

4.3.1 Steam Generator

It is necessary to attain a steam generator design able to assure good steam generator performance also in presence of wave motion. To do that, it is necessary to develop and use digital computer codes to define steam generator steady state and transient operating conditions including the ship motion effect. Natural circulation boiling water systems for shipboard application should be properly designed to avoid adverse effects of gravity field variation due to ship motion on the stability of fluid circulation.

In the stationary nuclear plants in the PWR system are used the "U" type steam generator (Fig. 4.5 & .6) that fall in

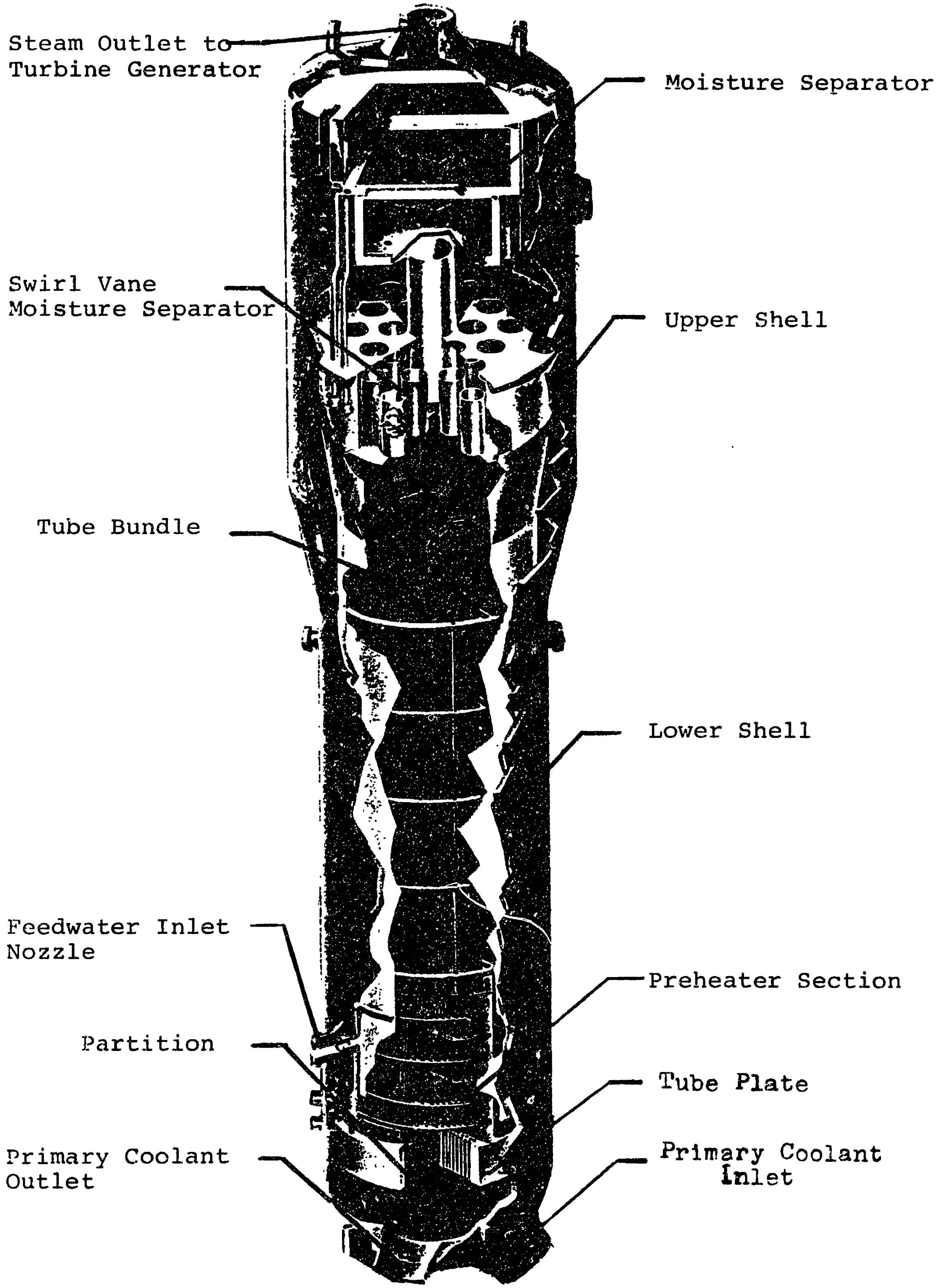


FIGURE 4.5 CUTAWAY OF A TYPICAL STATIONARY PWR STEAM GENERATOR

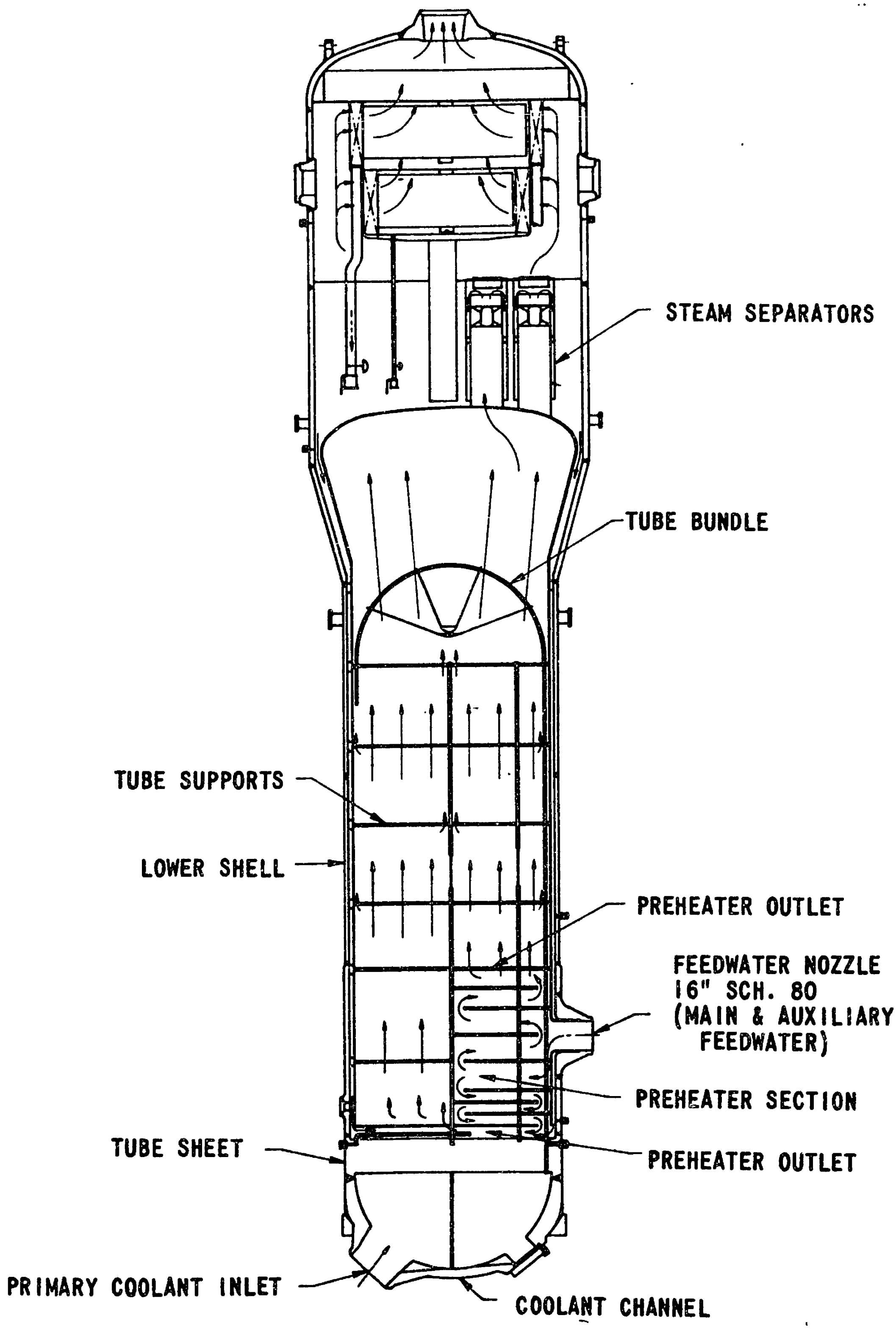


FIGURE 4.6 SIMPLIFIED TYPICAL STATIONARY PWR STEAM GENERATOR DIAGRAM

the category of boiling water reactor natural circulation systems that was adopted in N.S. SAVANNAH, N.S. LENIN and the theoretical design of Enrico Fermi. It consists mainly of two basic regions: a circulation loop and a steam drum. Such a two-phase natural circulation system could present flow oscillations causing loss of heat transfer efficiency, large level fluctuations in the downcomer, pulsation in the steam supply, etc., which could arise moreover from the ship motion perturbations whose effects could be to force a barely stable unit into the region of instability.

To study this ship motion perturbation effect, it is necessary to develop, besides codes already available for stationary reactor steam generator, special code that permits the analysis of the thermohydraulic stability of the recirculation loop in presence of rolling, pitching and heaving motions.

In the integral reactor type used in OTTO HAHN and developed in the CNSG design, the steam generator is fairly different from that used in loop reactor type. While in the loop type, only two steam generators located outside of the pressure vessel are used, the last conception of B&W, the CNSG IV A use steam generator comprises of twelve straight tube-and-shell cylindrical modules located inside of the pressure vessel, in the annulus above the top level of the core (Fig. 4.7 & .8). Each steam generator module utilizes one feed line with the latter arranged inside the steam line, thus requiring only twelve reactor vessel penetration. The once-through steam generator incorporates counter flow heat

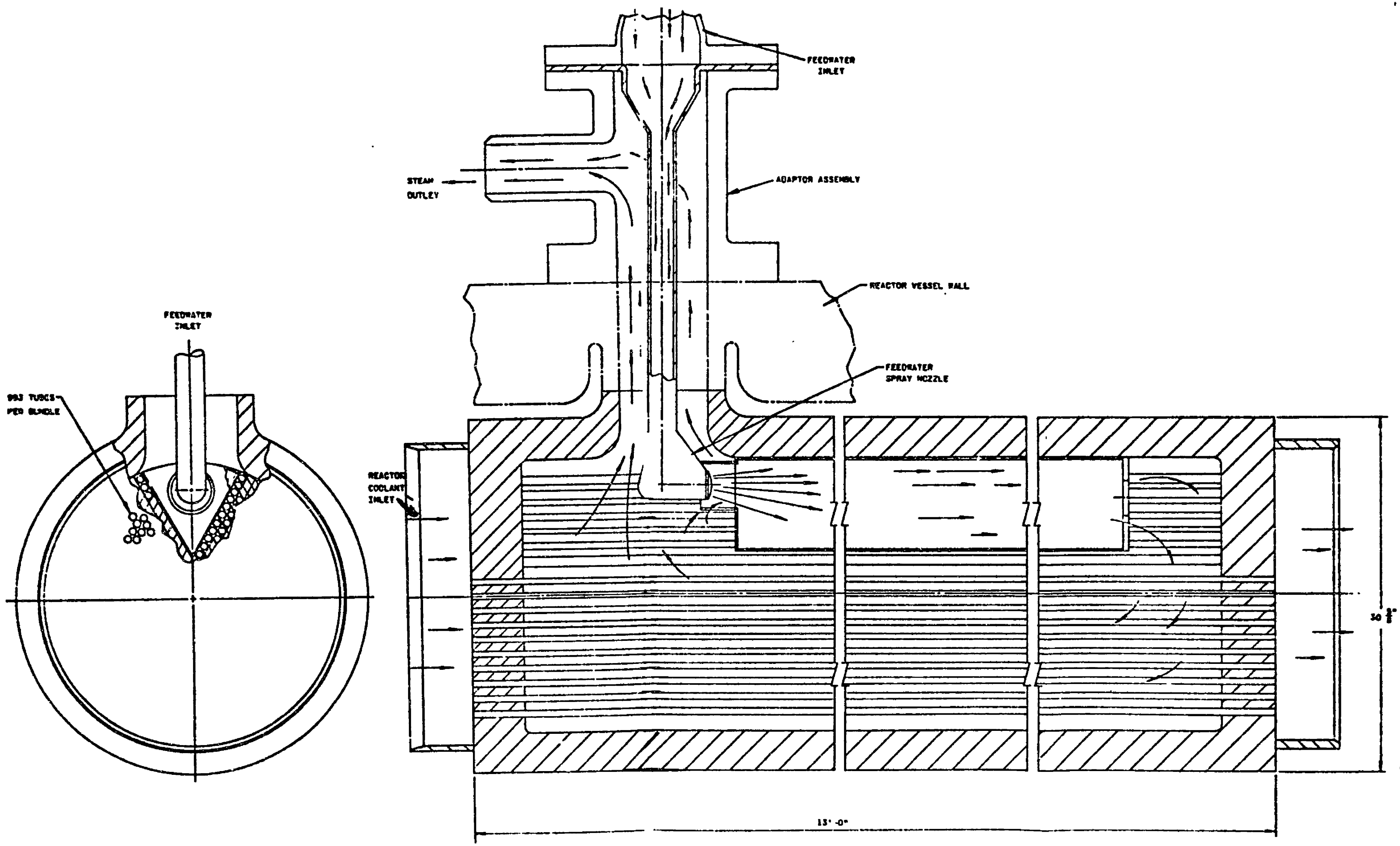


FIGURE 4.7 STEAM GENERATOR MODULE FOR INTEGRAL CNSG IV A

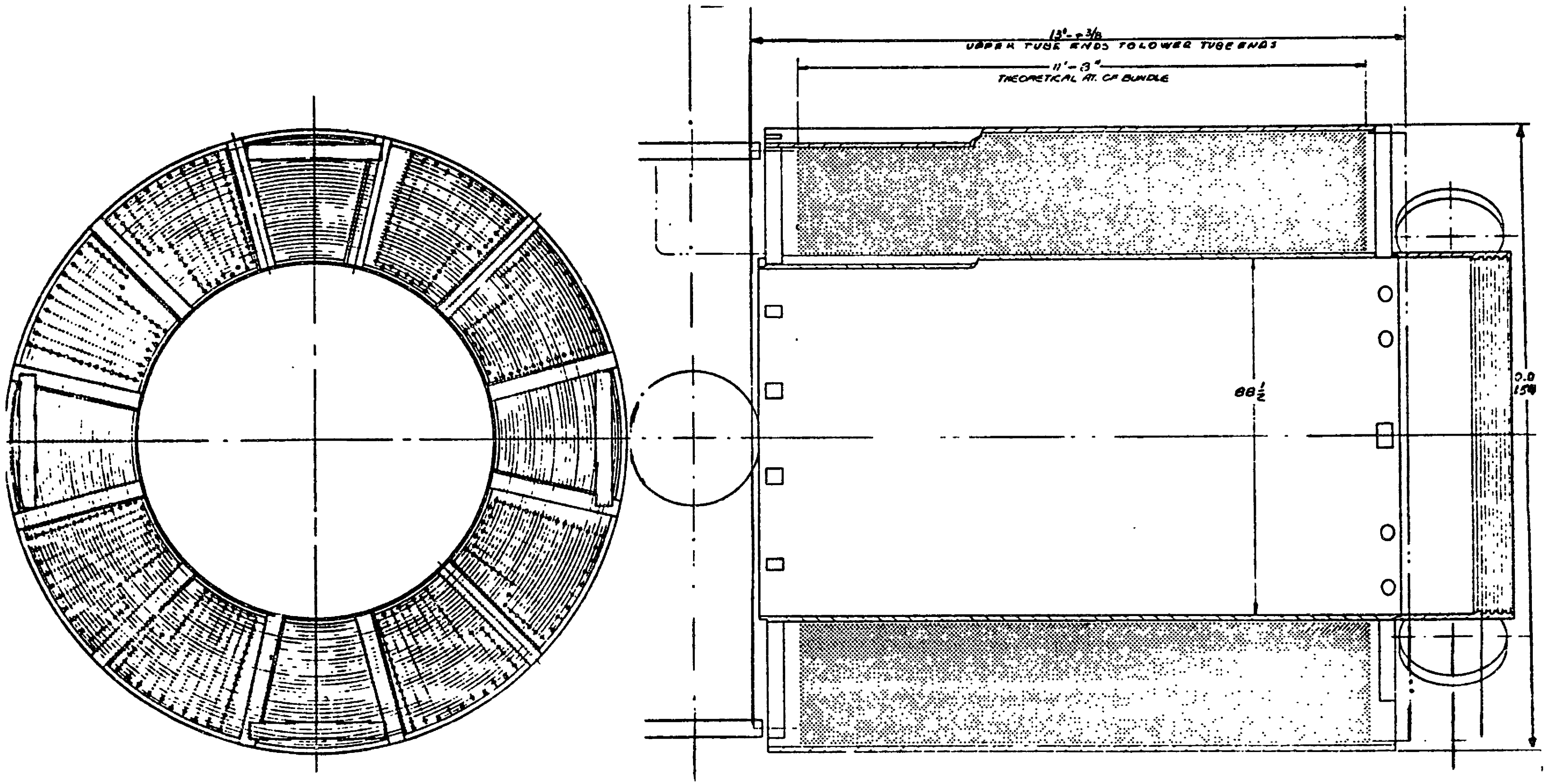


FIGURE 4.8 CUTAWAY OF STEAM GENERATOR MODULE FOR INTEGRAL CNSG IV A

transfer with shellside boiling to produce steam at a constant pressure. The reactor coolant system operates at a constant average temperature over the normal load range. This concept avoids the possibility of flow oscillations which arise from ship motions perturbations.

4.3.2 Control Rod Drive System

The control rod drive system must be designed to operate under the following conditions:

1. A continuous oscillation roll having a roll center an appropriate distance below or above the top of the pressure vessel, a determinate roll amplitude to each side from the vertical, and a roll period of determined seconds.
2. A continuous oscillation pitch having a period in seconds and when combined with heaving, imposes an additional load not to exceed a certain value of "g" in the vertical direction.
3. Fore and aft acceleration not exceeding a certain "g" value.
4. A permanent list of certain degrees and a permanent trim of determined degrees.

The control rod drive mechanism must be capable of powered insertion of control rod at ship angles greater than 45°. At ship angles less than 45°, normal control rod trip will be utilized to shut down the reactor. Above 45°, the trip time increases and at some angle between 45° and 90°, rod motion may stop entirely. Once control rod motion stops, a powered insertion must be utilized to drive the control rods to their full-in position (major detail then in SAFETY).

The main design differences between the roller-nut type drive mechanism for STATIONARY or MARITIME to avoid the ship motion safety problem is located in the MARITIME type(described in point 3.4) so that the lead screw is driven by separating anti-friction roller-nut assemblies attached to segment arms which for rapid insertion, separate into halves and release the lead screw and control rod. While in the STATIONARY type, the roller-nut is fixed. Another difference is that in MARITIME a mechanical spring is attached to aid gravity in forcing the control rod into the core at any ship inclination angle.

4.3.3 Foundation and Retaining Structures

The containment vessel, the reactor pressure vessel, and the biological shielding must be supported by foundations designed to distribute the load over the ship's structure as evenly as possible. The supports must be designed to withstand the dynamic effects of the ship's motion and, in addition, must be capable of keeping the containment, pressure vessel, and shielding in place at all angles of list, including the completely capsized condition.

The static system of the containment bedding of N.S. OTTO HAHN is shown in Figure 4.9 as thus in Figure 4.10, the containment and pressure vessel supports are showed.

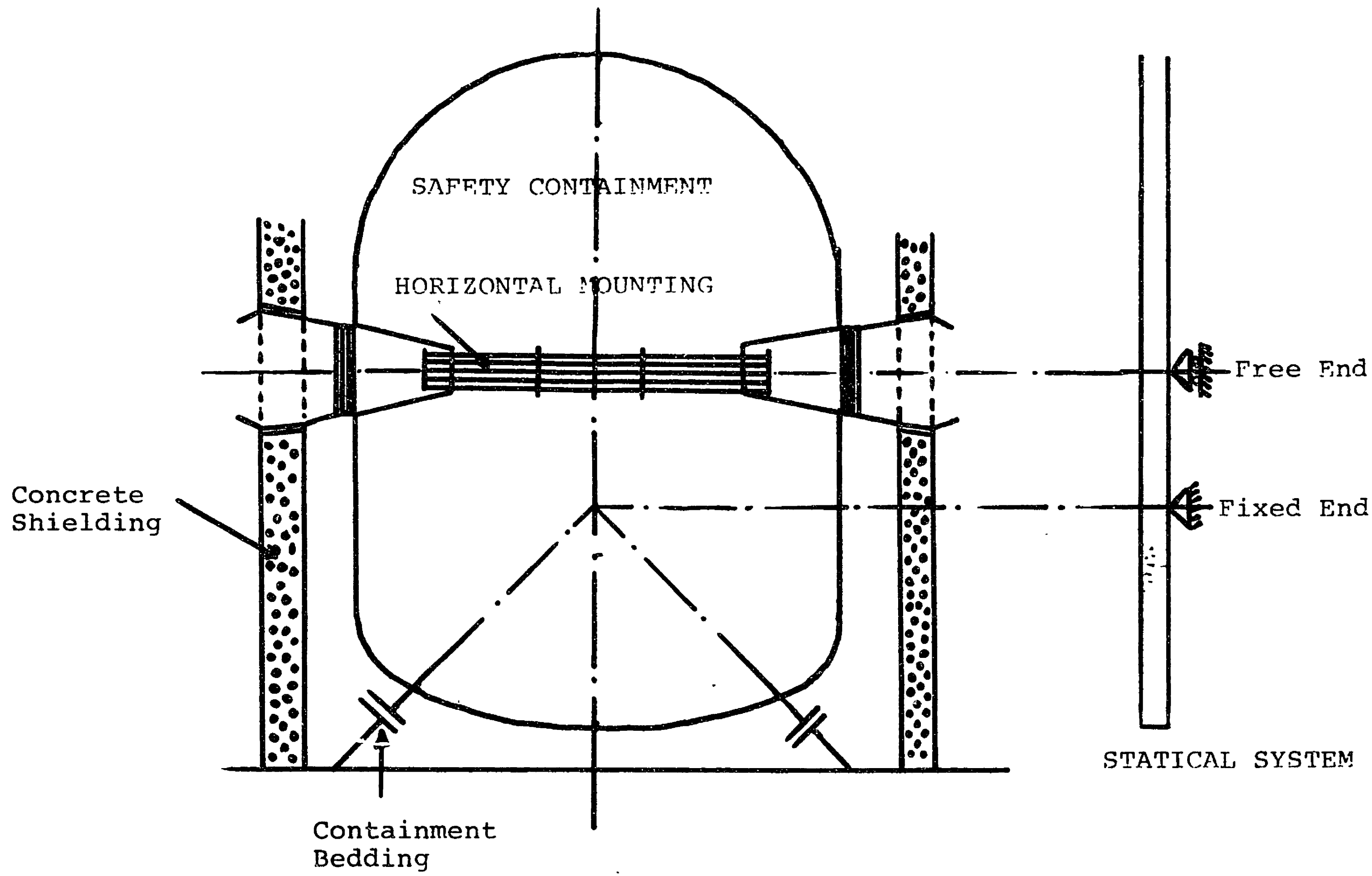
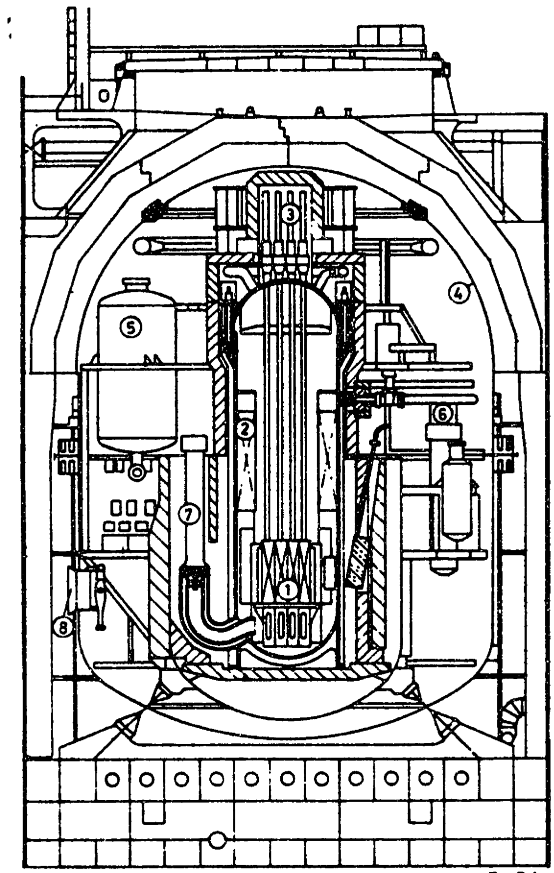


FIGURE 4.9 BEDDING SYSTEM OF SAFETY CONTAINMENT OF N.S. OTTO HAHN



1 Reactor core
2 Steam generator
3 Control rod drive
4 Containment
5 Pressure relief tank
6 Steam outlet/feedwater inlet
7 Primary circulating pump
8 Flooding valve

FIGURE 4.10 CONTAINMENT & PRESSURE VESSEL SUPPORT FOR N.S. "OTTO HAHN"

Comparing those figures with Fig. 4.11, 4.12 & 4.13, we can see how different are the support of the reactor pressure vessel, steam generator and reactor coolant pump of stationary reactors plant from that of maritime reactors.

4.3.4 Fuel Rod and Fuel Rod Assembly

The vertical acceleration to which a fuel rod can be submitted due to heaving motion could cause vertical pellet movement especially at the beginning of the fuel life. This movement could bring problems in the core power distribution. To avoid the probability of this movement, the spring design that maintains the pellets in position is very important. Relative motion between the fuel rod and the spacer grid due to ship motion may induce fretting of the cladding. It is necessary to insure that this fretting is not excessive.

4.3.5 Reactor Vessel, Internals and Other Accessories

The reactor internals must be designed to support static loads of the core and the added loads imposed by the effects of ship roll, pitch and heave. These loads are transmitted to the pressure vessel in the integral reactor type at three locations: the head, the steam generator support, and the upper vessel flange. The core barrel assembly that surrounds the core and whose function is to direct the coolant through the core must also take side loads due to fuel assembly deflection caused by ship roll. These side loads are transmitted to the vessel wall through horizontal supports.

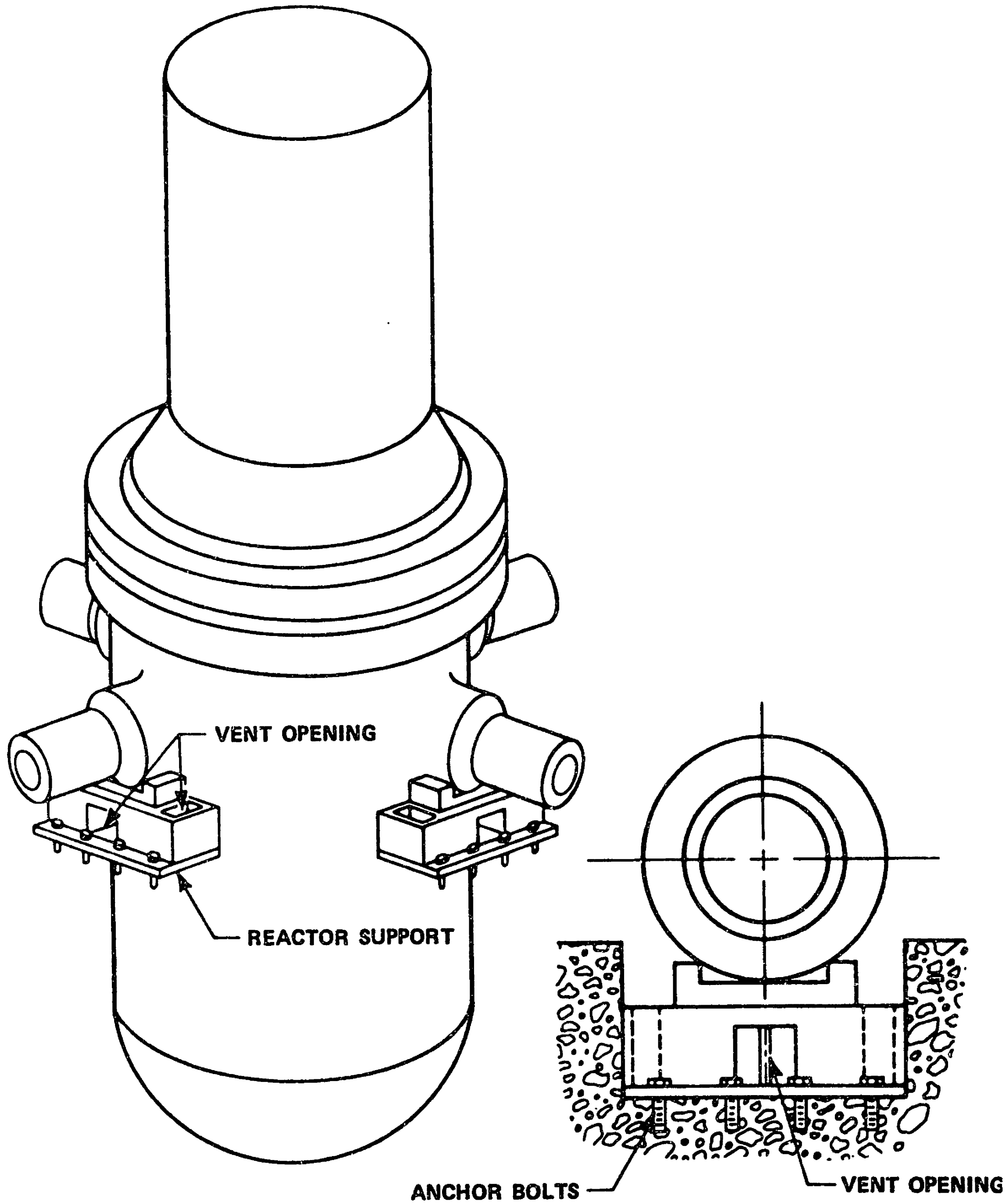


FIGURE 4.11   REACTOR VESSEL SUPPORTS FOR STATIONARY PWR PLANT

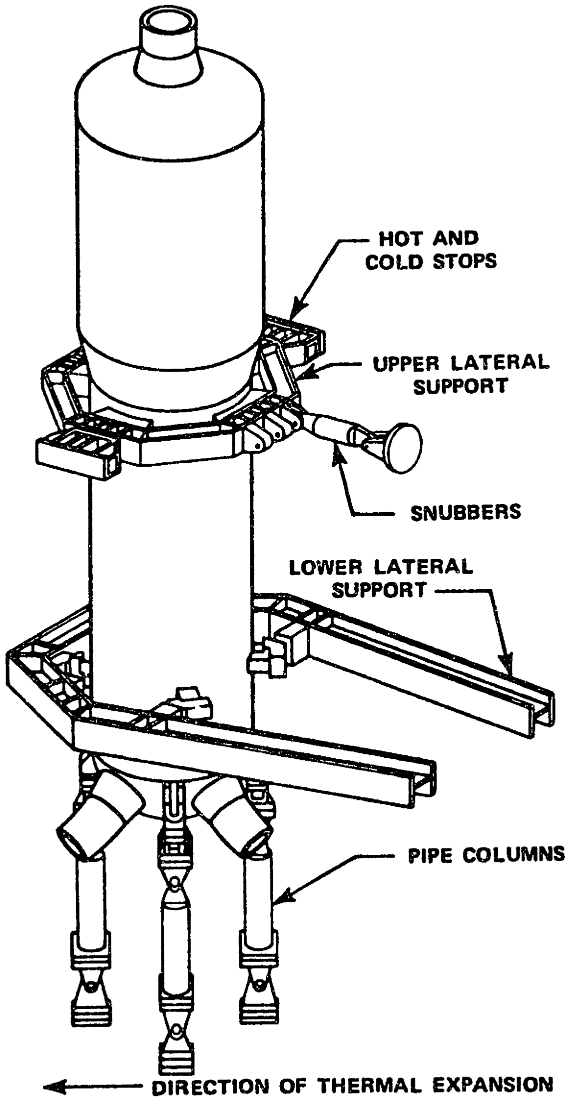


FIGURE 4.12 STEAM GENERATOR SUPPORTS FOR STATIONARY PWR PLANT

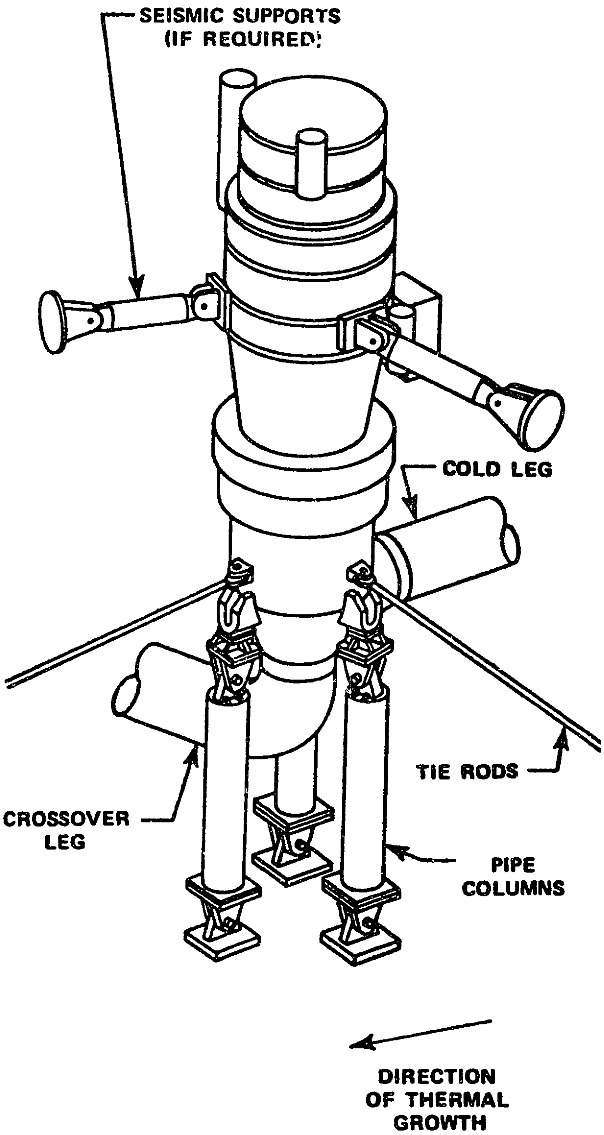


FIGURE 4.13 REACTOR COOLANT PUMP SUPPORTS FOR STATIONARY PWR PLANT

The reactor vessel with the reactor coolant system, internals and other accessories in the integral reactor type are supported vertically by means of specially designed steam nozzles. Lateral restraining devices must be added near or at the bottom of the reactor vessel if required to limit vessel motions caused by ship motions.

4.3.6 Reactor Coolant Piping

Design must include consideration of the structural effects resulting from ship motion. In the integral reactor type, the main steam lines are designed with fast-acting motor-operated isolation valves, and in its design, it is necessary to consider the structural and operational effects resulting from ship motion. Snubbing-type supports must be provided to support against ship motion.

4.3.7 The Pressurizer

The various loadings induced by ship motion must be considered among the total stresses in the pressurizer design.

4.3.8 Reactor Coolant System Instrumentation

In the maritime reactor within the instrumentation of the reactor coolant system which provides for the measurement and control of process variables needed for proper reactor operations and protection, it is necessary to have sensors for ship motion. These sensors utilize a gyroscopic platform for their measures. These measures are essential since ship roll is one of the types of trips to be considered. Ship roll is measured to determine whether the vessel has rolled so far to one side or the other that it will continue to roll until it capsizes.

4.3.9 Control Rod Assembly

In stationary reactors, some contact between control rods and the fuel assembly guide tubes is expected. This contact in maritime reactors is incremented because of the ship motion that will contribute to analyzing the deformation of the fuel assembly and its guide tube as a result of ship motion excitation.

4.3.10 Containment

The containment structures must be provided to maintain in place its internal components, and components necessary for mitigation of the accident consequences regardless of the ship's orientation. It must be safely designed to withstand structural gravity loads (normal and capsized ship attitude) and operating loads due to ship motion.

To design a containment for a reactor ship, it is necessary to include among all credible conditions of loading the normal loads that vary with intensity and occurrence due to ship motion.

The containment support and sway bracing are designed to support the weight imposed by the containment and its contents with the ship in any position.

In the pressure suppression pool containment type, for maritime application, the wet well is subdivided into identical and discrete compartments. This design is made with the purpose to avoid the problem of the liquid mirror that only one compartment could build up since the wet well volume is only 65% filled. In that case, stability and operational problems could occur due to the ship motion effect.

4.4 Reactor Parts Affected By Ship Vibrations

The major vibration frequencies are as follows:

- Propeller frequencies caused by alternating pressure fields impinging on the hull as the propeller rotates. The frequency is equal to the number of blades times the revolutions per minute.
- Hull natural frequency is the frequency of the hull acting as a simple beam. It is dependent on hull inertia, form, and mass with several values, each associated with a specific mode of vibration.

These vibrations affect different components of the reactor such as:

- Safety features actuation systems
- Foundations
- Reactor internals
- Containment
- Reactor coolant piping
- Main steam line isolation system
- Pressurizer
- Reactor protection equipment

4.4.1 Safety Features Actuation System

Monitors reactor coolant and containment pressures to detect loss of integrity of the reactor coolant system pressure boundary. This system is being designed to withstand expected vibrations under normal operating condition as well as transient conditions due to environmental consideration such as weather.

4.4.2 Foundations

The foundations of the reactors components, such as containment, pressure vessel, shielding, pumps, etc., shall be non-resonant with major frequencies predicted and/or encountered. The only significant source of induced vibratory motion is the propellers. Then, all components and their foundations must be non-resonant with propeller induced frequencies at full power. The frequencies of interest are those resulting from normal propeller shaft revolution times the number of propeller blades and the first harmonic of this frequency (twice blade frequency). Because of necessity (the propeller operates over a range of speeds), it may not be possible to avoid passing through resonance points associated with some components such as main propulsion gear torsional frequencies.

For example, this condition is particularly true of torsional frequencies in the main propulsion gearing train. In this case the components affected, or their foundation, are tuned so that resonance occurs at low propeller speed where induced forces are negligible.

4.4.3 Reactor Internals

The internals must be designed to withstand the dynamic loads resulting from the vibration loads imposed by external sources, such as the ship propellers and auxiliary equipment.

4.4.4 Containment

It must be designed to safely withstand the operating loads due to vibrations.

4.4.5 Reactor Coolant Piping

In its design must be included consideration of the structural effects resulting for vibration.

4.4.6 Main Steam Isolation Line

For the design of the fast-acting motor operated isolation valves, it is necessary to consider the structural and operational effects resulting from ship vibration. In the integral reactor type (CNSG) snubbing-type supports are provided to isolate valves from random vibrations.

4.4.7 Pressurizer

Its design must consider any structural effect as a result of vibration, whether induced by operational or environmental conditions.

4.4.8 Reactor Protection Equipment

It will be qualified according to the vibration requirement. Based on experience with numerous ships, this requirement must cover the frequency range of zero to 100 H_z at amplitudes from 0.4" (low frequency) to 0.002" (high frequency). With larger ships, this requirement diminishes.

CHAPTER V

TECHNOLOGICAL DIFFERENCES DUE TO SAFETY FACTORS

5.1 Introduction

Safety considerations for ships must be differentiated from the safety considerations for land-based nuclear power plants because:

1. The possible type and kind of accidents on a ship are fairly different from that of land-based plants.
2. The loss of propulsion power for a ship can result in unsafe conditions for the ship, cargo and crew.
3. The ship will operate for the large majority of the time at sea, away from populated areas, which significantly reduces the risk to the general public.
4. The proximity of the living quarters aboard ship to the reactor plant means that, unlike a land-based establishment, the operational personnel and the rest of the ship's crew will be subjected to some radiation outside their working hours.

5.2 Type & Kind of Major Different Accidents

A nuclear power plant, both stationary and maritime, can be subjected to two types of accidents.

Reactor Internal Accident

For both stationary and maritime, the major credible accident adopted is that due to the LOCA (Loss of Coolant Accident). The only difference in this case is in the type of reactor used (loop or integral type) and not whether they are land or ship-based.

Reactor External Accident

Here the matter is quite different. In stationary, the critical load conditions are given by the Earthquake loads and some time where the location is not in an Earthquake-prone region; by the wind forces. In maritime the critical conditions are given by nautical accidents as collision, grounding, capsizing and sinking.

5.3 Safety Areas of Nuclear-Powered Ship

Two separate areas of safety must be considered for a nuclear powered ship: reactor safety area and ship safety area.

5.3.1 Reactor Safety Area

Reactor safety is provided by the reactor protection system which protects the reactor from damage by initiating a reactor trip. The reactor trip results in dropping the control rods into the reactor core under the forces of gravity and the scram springs acting in the control rod drive mechanism. Due to the functional and gravitational forces, the reactor trip is effective in a specific band of ship angles. However, this band of ship angles covers all normal and anticipated ship angles that will be encountered during ship operation. Only major ship accidents, such as severe collision, grounding, flooding and sinking could cause the ship to permanently attain an attitude outside the band of angles where the reactor trip is not effective. In addition, these severe ship accidents that could cause an excessive ship attitude are highly unlikely and would allow ample time for a manual trip before the excessive attitude would be reduced.

5.3.2 Ship Safety Area

Ship safety requires a highly reliable propulsion plant to prevent loss of ship, cargo, and crew in extreme conditions where propulsion power for ship maneuvering is vital. The plant control system contributes to a higher power continuity by maneuvering the plant to avoid the need for protective action. This control system, which has a much broader scope of plant control than a single-channel central station system, is a redundant automatic control system and provides control for normal, off-normal, and faulted conditions. A principal difference between the maritime reactor plant control system and the stationary reactor control system is the use of digital hardware and software for implementation of control policies. The maritime plant control system provides redundant control channels and employs more sophisticated control techniques for normal control and control of many off-normal and faulted conditions.

These differences in the maritime control system are directed toward power continuity of the ship board propulsion plant.

The consequences of the loss of propulsion power resulting in the real possibility of loss of the ship, cargo and life, must be weighted against the possibility of reactor damage resulting from possible but unlikely initiation of a major reactor accident during momentary ship operation at attitudes where a trip is not fully effective.

The nuclear plant must be designed with due consideration to continue operation with defects when the loss of propulsion to the ship would produce consequences more severe than operation with the defects. Thus, certain events may be postulated for which loss of propulsion produces consequences more severe than continued operation for a limited period of time with the defect. Necessary equipment and systems are provided for reduced power operation. Every ship is provided with an auxiliary propulsion system such as boiler, diesel engine, etc., enabling the ship to move by itself in case of loss of reactor power. Also, the plant control system is designed to satisfy this criterion.

Finally, the ship design should be such that to minimize the risk of nautical accident. Nevertheless, in case of occurrence of such accidents, protection of the reactor should avoid any damage to it.

5.4 Differences Among Subassemblies for Maritime & Stationary Reactors

5.4.1 Reactivity Control Aspects

The major differences in reactivity control systems are due to the impossibility of use of chemical shim in maritime reactors.

5.4.1.1 Maritime Reactor Reactivity Control

In maritime reactors the reactivity is controlled by normal and transient operation by control rod assemblies (CRAs), burnable poison rod assemblies (BPRAs) and lumped burnable poison rods (LBPRs) integral with each fuel assembly.

Chemical shim control is presently not a feature of normal reactivity control of the maritime design because the ship sinking accident might result in seawater inleakage and result in reactivity addition in the core. However, German designers still believe that chemical shim is permissible for maritime application. Thus, in CNSG sufficient control rod assembly worth is available to shut the reactor down with at least a 1% $\Delta K/K$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position and with no soluble boron in the primary coolant.

Sufficient CRA worth is also available to shut the reactor down with at least a 1% $\Delta K/K$ margin in the cold condition at any time during the life with no soluble boron present.

The reactivity worth of a CRA and the rate at which reactivity can be added are limited to ensure that credible reactivity accident cannot cause a transient capable of damaging the reactor coolant system or causing significant fuel failure.

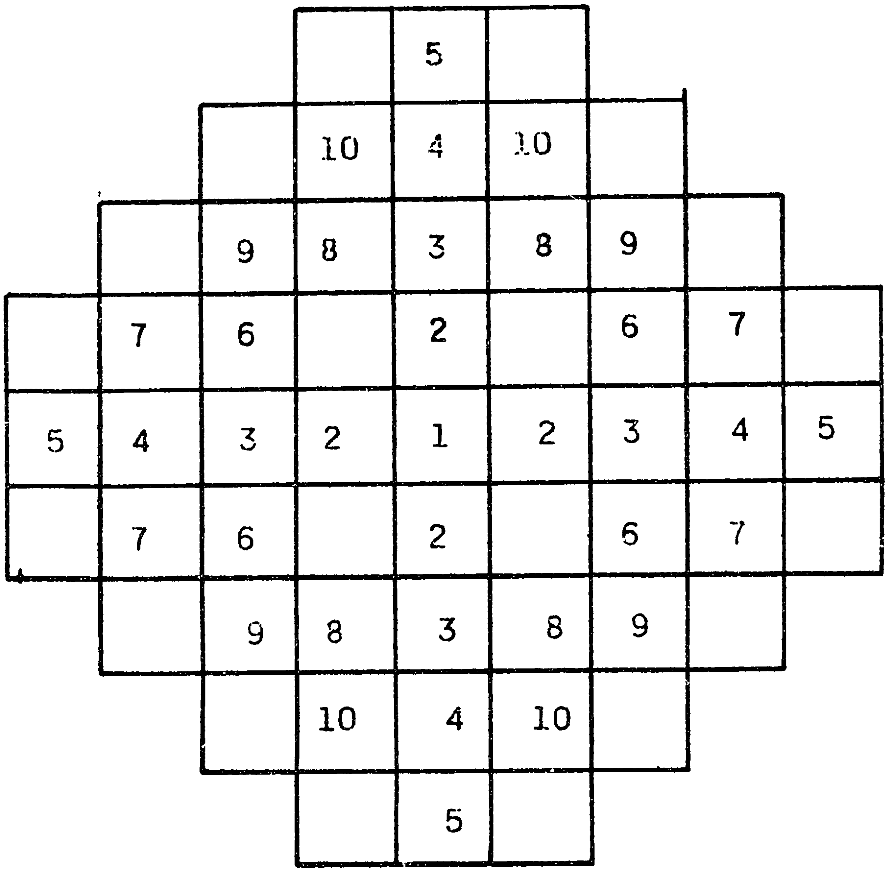
In the design of the CNSG of B&W, a boric acid addition system is foreseen only to be used to add soluble boron to the reactor coolant to ensure the reactor shut down with at least a 1% $\Delta K/K$ margin in the cold condition in the unlikely event it is necessary to cool the reactor to ambient temperature with rod malfunctioning (i.e., the most reactive control rod stuck in the fully withdrawn position). The main reason to use this system is to fulfill, as a second reactivity control system, the requirement of the general Design Criteria for Nuclear Merchant Ship that in it criterion 26 establish -

> "Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, etc. The second reactivity control system shall be capable of reliably controlling the rate of reactivity change resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions".

The CRAs (Figure 5.1) are of safety or regulations type according as its functions.

The reactivity of the core under regulating operating conditons is controlled by the CRAs. Control rod assemblies are used to control reactivity components due to the moderator temperature deficit, equilibrium xenon and samarium, transient xenon, doppler deficit, shut down margin, xenon undershoot, and fuel burnup and fission product buildup.

Fixed burnable poison , provided as LBPR and BPR, is used for maritime service in lieu of soluble poison because,



BANK	NO. OF RODS	PURPOSE
1	1	REGULATING
2	4	REGULATING
3	4	REGULATING
4	4	REGULATING
5	4	SAFETY
6	4	SAFETY
7	4	SAFETY
8	4	SAFETY
9	4	REGULATING
10	4	SAFETY

FIGURE 5.1 CONTROL ROD LOCATIONS FOR INTEGRAL CNSG IV

as was said before, in that way fixed poison cannot be lost under accident condition as could be in core of soluble poison.

In the fuel assemblies containing no CRAs, a BPRA is inserted (see Figure 5.2) into the guide tubes. The BPRAs are latched in place and are not moved during the fuel cycle. These assemblies absorb excess reactivity at beginning of life. Their absorption capability decrease until, at end of life, they have almost no effect on the core reactivity.

Each BPRA has a certain number of poison rods (20 rods in CNSG IV), a stainless steel spider, and a coupling mechanism. The coupling mechanism and the rods are attached to the spider. The basic assembly is similar to the control rod assembly (see Figure 5.3).

Each BPR (see Figure 5.4) has a section of burnable poison of sintered pellets (in CNSG IV are Al_2O_3 B_4C). The poison concentration is not varied along the length of the rod. The poison section is axially located by internal spacers and is designed to permit differential axial expansion of the cladding and poison section.

LBPRs (Figure 5.5) are separate rods of burnable poison, which are dispersed in the fuel rod array. The LBPRs contain variable concentrations of burnable poison. [In CNSG this burnable poison is boron carbide (B_4C) dispersed in an alumina (Al_2O_3) pellet.] Burnable poison concentration varies radially and axially within the core (see Figure 5.6). LBPRs have three axial zones. All rods in a given fuel

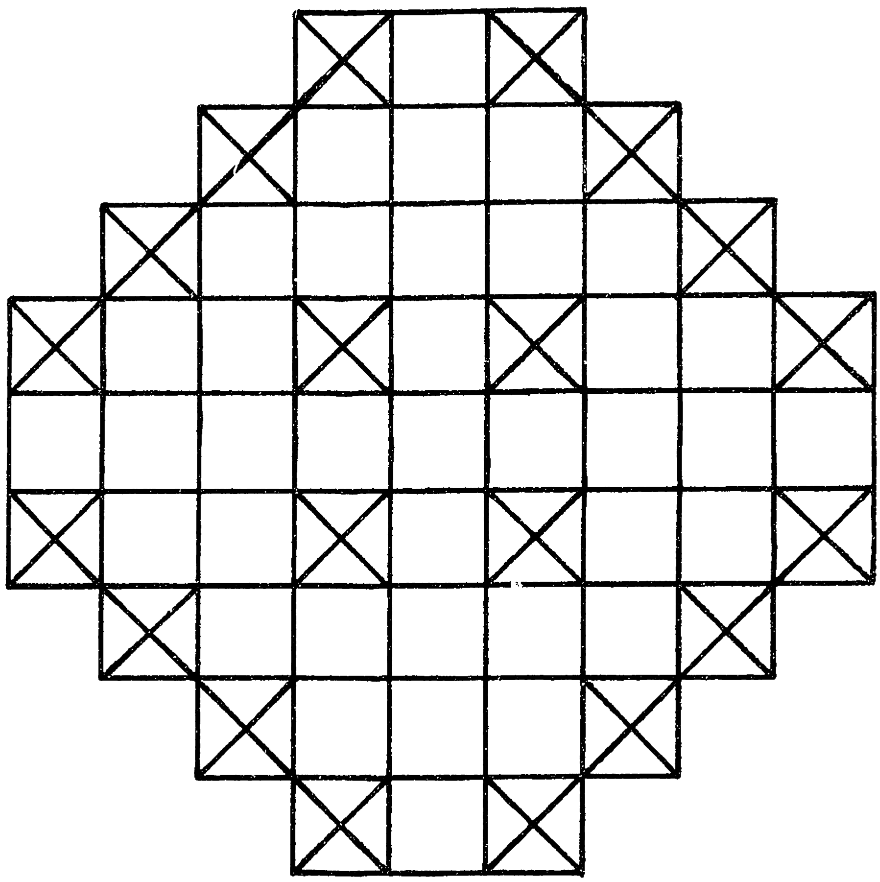


FIGURE 5.2 LOCATION OF FUEL ASSEMBLIES CONTAINING BURNABLE POISON ROD ASSEMBLIES IN INTEGRAL CNSG IV

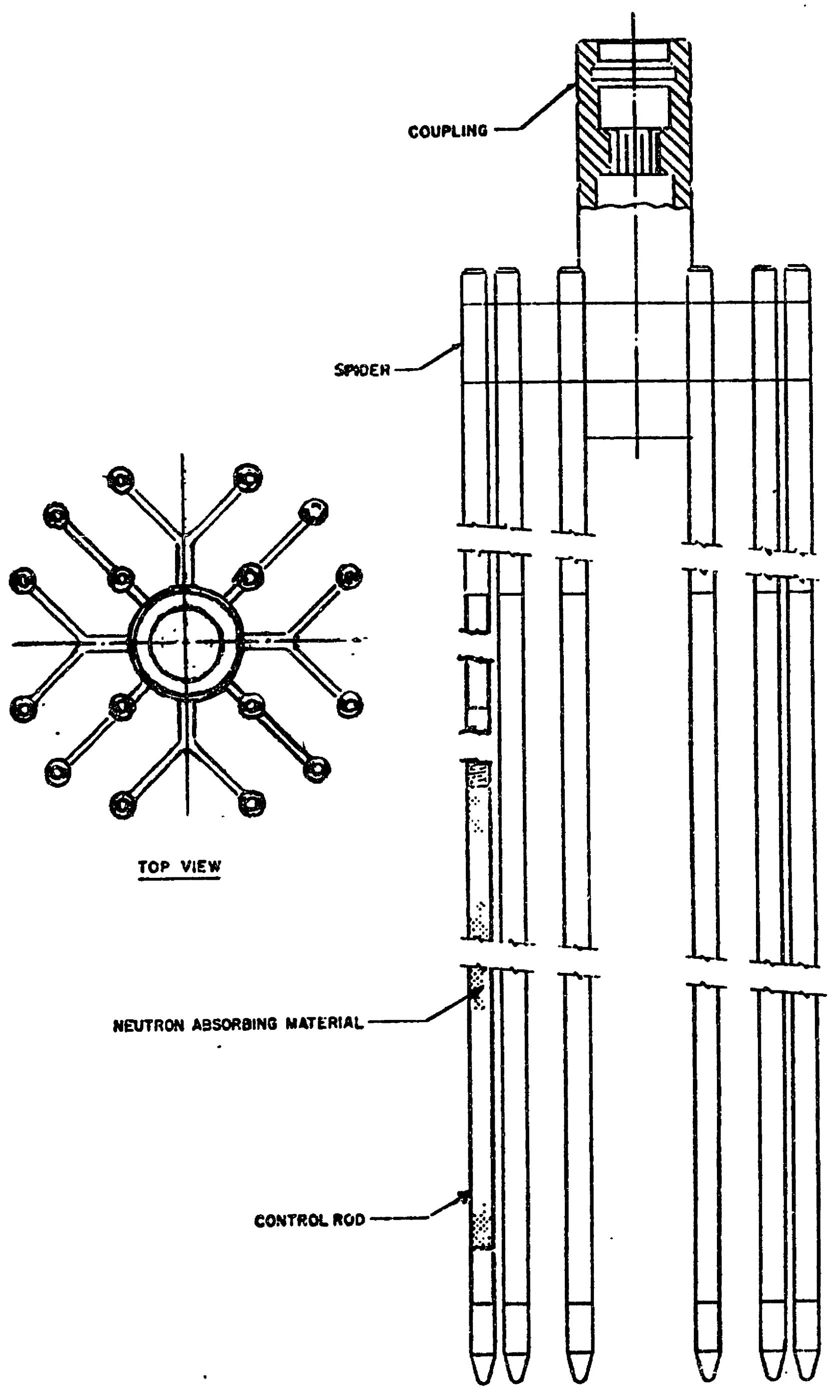


FIGURE 5.3 CONTROL ROD & SPIDER ASSEMBLY OF THE INTEGRAL CNSG IV

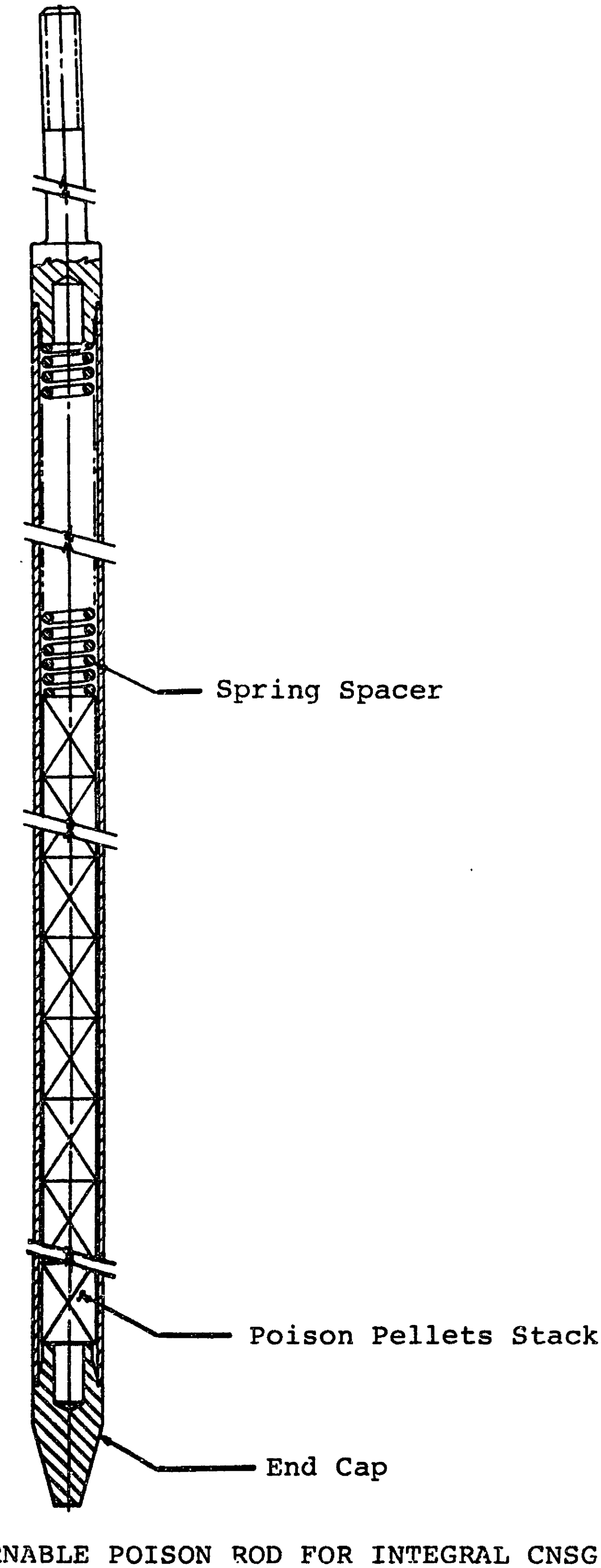


FIGURE 5.4 BURNABLE POISON ROD FOR INTEGRAL CNSG IV

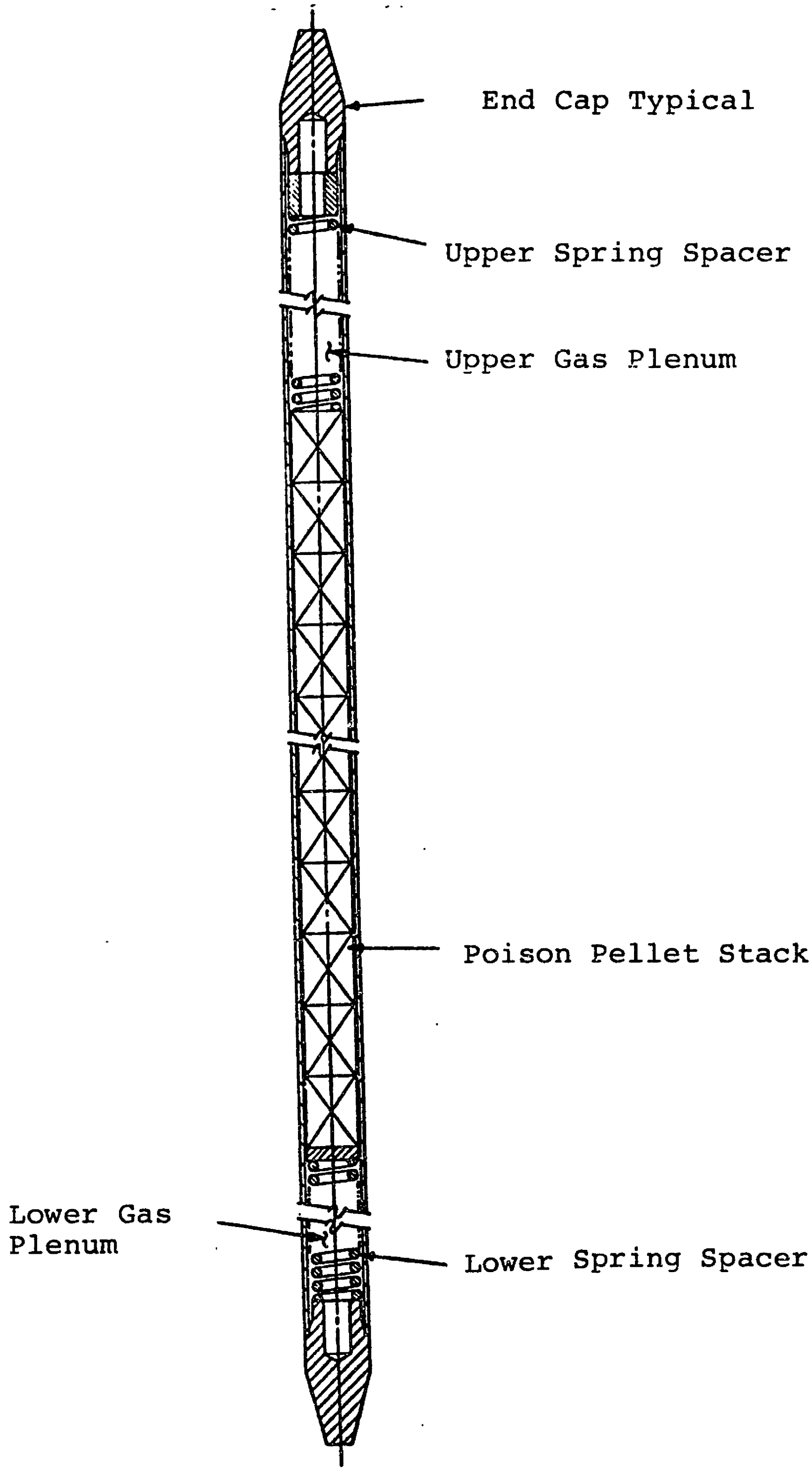
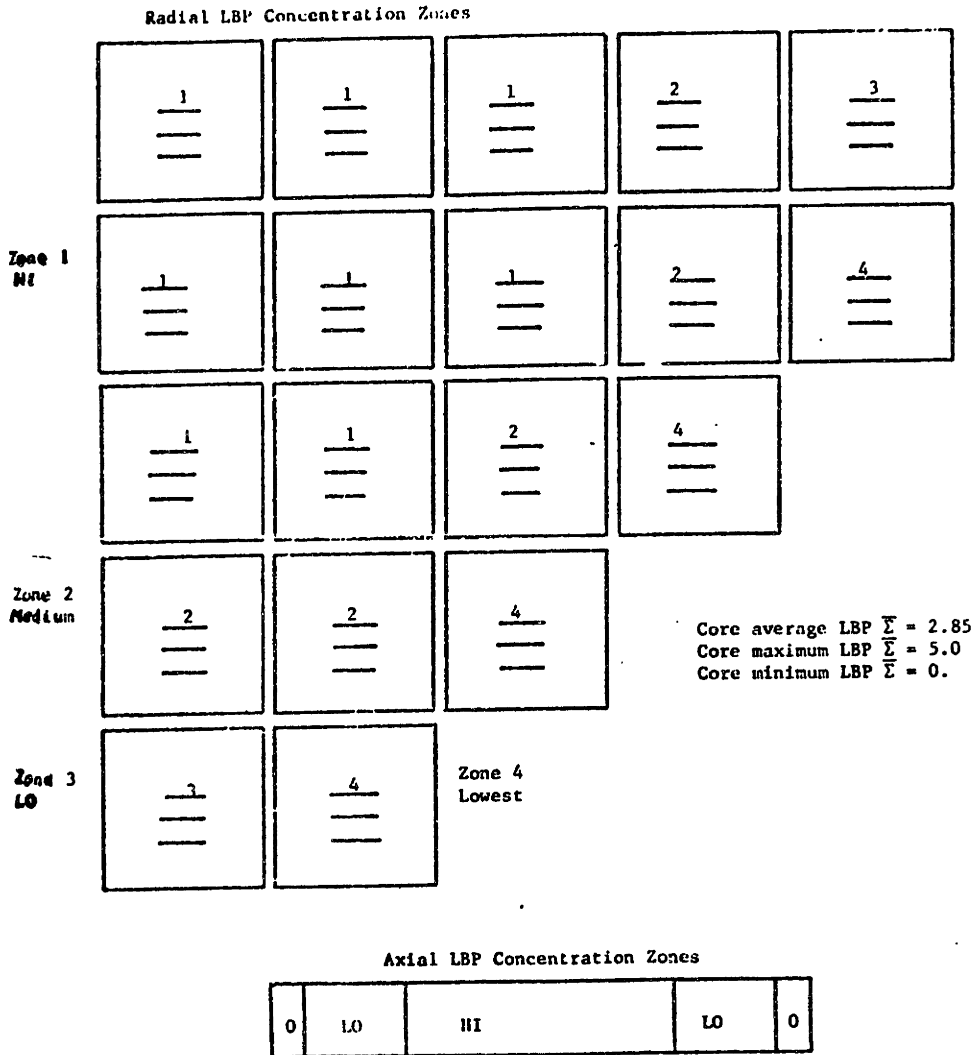


FIGURE 5.5 LUMPED BURNABLE POISON ROD (LBPR) FOR INTEGRAL CNSG IV



NOTE: LBP $\bar{\Sigma}$ is used purely as a labeling device to express the ^{10}B content in the burnable poison rods. It is representative of the Maxwellian-averaged microscopic absorption cross section of ^{10}B at 600 F (~2400 barns). The heterogensous ^{10}B number desnity N(^{10}B) = Σ_a/2400 atoms/barn-cm.

FIGURE 5.6 AXIAL & RADIAL LBP DISTRIBUTIONS IN THE INTEGRAL CNSG IV fuel assembly have the same boron concentrations. The LBPRs are similar in design to fuel rods except that internal spacers and springs are designed specifically for LBPRs.

In case Boron-10 is used as burnable poison, sufficient plenum volume must be provided to ensure that helium released from boron-10 fission does not cause stress that exceeds allowable design limits.

LBPR end caps differ in shape from fuel rod end caps to ensure a distinctive difference in the appearance of the fuel rod and LBPR. This permits verification that they are placed correctly in each fuel assembly. Each LBPR and pellet must be marked with a symbol to permit verification that the proper concentration is used at each position.

5.4.1.2 Stationary Reactor Reactivity Control

In stationary reactors, reactivity control is provided by neutron-absorbing control rod and by a soluble chemical neutron absorber (boric acid) in the reactor coolant.

Neutron-absorbing control rod provides reactivity control to compensate for more rapid variations in reactivity such as:

(1) fast shutdown
(2) reactivity changes associated with changes in the average coolant temperature above hot zero power
(3) reactivity associated with any void formation, and
(4) reactivity changes associated with the power coefficient of reactivity.

To fulfill these requirements the rods are divided into two categories and according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control or regulating group of rods. As was seen in Chapter III, the part length control rods fulfill this function.

The remaining rods, which provide shutdown reactivity, are termed shutdown rods and are the same full length rods as the control rod named in Chapter III.

Chemical Neutron Absorber

Its concentration is varied as necessary during the life of the core to compensate for:

(1) changes in reactivity which occur with change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions

(2) changes in reactivity associated with changes in the fission product poison xenon and samarium

(3) reactivity losses associated with the depletions of fissile inventory and build-up of long-lived fission product poisons (other than xenon and samarium)

(4) changes in reactivity due to fixed burnable poison burnup.

5.4.2 Differences in the Movable Control Rod Assembly Worth Design

Control of excess reactivity by movable control rod assemblies in the CNSG IV is shown in Table 5.1. That table contains an item 'transient xenon' which is not needed for a stationary reactor that can be confirmed, if we compare the functions of the stationary and maritime reactivity control systems described in point 5.4.1.1.

This core "transient xenon" reactivity excess is required in marine reactor to override the negative reactivity due to xenon build-up following a power decrease from full to zero power or reactor shut-down. This excess of reactivity permits the reactor to be started up within a reasonable time after its shutdown, overriding the negative reactivity barrier imposed at that interval by the xenon buildup (see Figure 5.7). In a maritime reactor that condition is very important because, sometime, the ship security is more important than the reactor safety as was said in the second consideration of point 5.1.

5.4.3 Control Rod Assembly Safety Powered Insertion System

In the integral reactor type (CNSG), to provide maximum power continuity without sacrificing reactor safety, a powered insertion system is used, in addition to the reactor protection system, which has no attitude trip. This system automatically and simultaneously inserts all control rods using the rod drive mechanisms for the condition of excessive ship attitude. It has the capability to insert all rods and

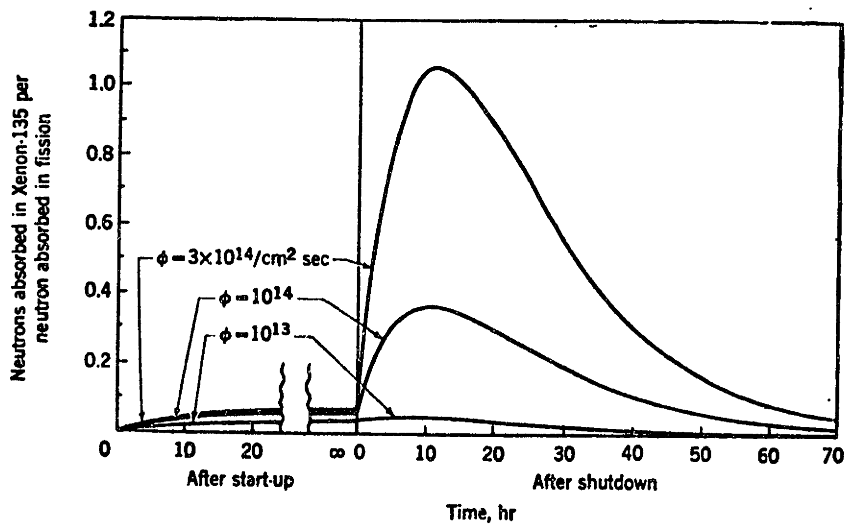


FIGURE 5.7 XENON POISON RATIO DURING REACTOR OPERATION AT CONSTANT FLUX AND AFTER SHUTDOWN.

TABLE 5.1

BOL REACTIVITY CONTROL DISTRIBUTION

Controlled by Movable Control Rod Assemblies (CRA)	Reactivity, % $\Delta k/k$
Moderator temp deficit (68 to 505F)	2.2
Moderator temp deficit (0 to 100% power, 505 to 589F)	1.8
Equilibrium xenon and samarium	1.5
Fuel burnup and fission product buildup	3.2
Transient xenon	1.6
Doppler deficit (0 to 100% power)	0.7
Shutdown margin	1.0
Controlled by Burnable Poison	
Fuel burnup and fission product buildup	18.0

insures a plant shutdown for any ship attitude. This powered insertion is considered a ship-related engineered safety feature because it is actuated only in the highly unlikely case of a major ship accident where an excessive attitude exists and a normal trip has not occurred.

In summary, this system has the capability to insert all control rods at any ship attitude and to remove the need for an attitude trip, affording greater power continuity and thereby greater ship safety.

The powered insertion system is a subsystem of the rod drive control system and provides the capability to shut the reactor down at excessive ship angles. For normal operation, anticipated transient and emergency conditions for which the ship attitude is not excessive, the spring-assisted gravity trip is the primary shut-down means. But, if the ship attitude should exceed the set point value, as it might during severe collision, flooding, and sinking, before normal gravity trip occurred, then, the powered insertion system would simultaneously insert all control rods. The powered insertion system is automatically initiated by the plant protection system when the ship attitude detectors exceed the set point value. The set point value will depend on the ship's characteristics and on the rod drive mechanism angle trip tests. Also, the powered insertion rod speed will depend on core performance and accident analysis.

The powered insertion system is capable of fully inserting all control rods at any ship angle. Also, a special purpose power insertion battery is provided to allow complete insertion of all rods without the need for normal ship's service power. The rod control equipment and battery are designed to operate at all ship attitudes and are located on the ship so that complete insertion would be accomplished before flooding could occur.

In summary, the powered insertion system provides a positive shutdown capability at all ship attitudes for which normal gravity trip is ineffective.

The powered insertion function is actuated by the plant protection system or manually and is physically performed by the rod drive mechanisms and rod control system. The powered insertion system can be initiated automatically by the plant protection system or manually from either the control console or the hot and cold shutdown panel. Automatic initiation results from an excessive ship attitude as detected by the plant protection system ship attitude detectors. Then when the plant protection system initiates a powered insertion, the events occur as follows:

- The powered insertion battery is connected to the rod drive motor bus when two parallel circuit breakers are closed.
- The redundant "insert only" control circuits are energized to power "insert all" control rods.
- When all the control rods in each group are fully inserted, the botton limit switches stop the group's insertion action.

- The rods remain latched (roller-nut) engaged with the leadscrew) until stators are de-energized.

Once the rods are fully inserted, a mechanical latch prevents any rod-out motion in the latched or unlatched condition as might occur in case the ship capsizes.

Manual operation is necessary to return the rod control to the normal mode of operation.

The powered insertion system is redundant. It has:

- Powered Insertion Battery
- Powered Insertion Breakers
- Rod Drive Motion Modules
- Powered Insertion Control - A separate powered insertion redundant control circuit is provided.
- Full-In Limit Switch

The following display is provided on the control console:

- Powered Insertion Battery Low-Charge Alarm
- Powered Insertion Actuation Alarm
- Individual Absolute Rod Position Indication & Rod Bottom Lights

The powered insertion system, which includes the control circuits, drive motor modules control rod and drives and battery, is self contained. Once initiated by the plant protection system, all rods are fully inserted without support from other systems.

5.5 Differences in the Reactor External Protection

Besides the protective measures against potential internal failures due to both reactor internal or reactor external accidents, there are also special arrangements directed towards the limitation of the consequences of external mishaps which are to be taken into account in the maritime environment.

5.5.1 Collision

Collisions are the most frequent of marine accidents and cannot be totally prevented through design even if the most sophisticated collision avoidance equipment is installed. We can differentiate here between two kinds of protection, passive collision protection and active collision protection.

5.5.1.1 Passive Collision Protection

A nuclear ship must be capable of coping, in almost all conceivable circumstances, with the shock resulting from a collision with another ship without impairment of the reactor space. In order to satisfy this requirement, the following conditions are to be fulfilled:

- The greater possible distance between the hull side and the longitudinal bulkhead of the reactor space.
- A heavy steel structure between these two boundaries since it is the deformation of that structure which would absorb a part of the kinetic energy of the ramming ship.
- A suitable arrangement of that structure in order to achieve a deformation entailing the greatest energy consumption.
- To avoid any impairment of the containment vessel by ruptured parts.

For a particular ship, the first point provides little margin since the breadth of the ship is already defined as well as that of the reactor space.

It may be noted that protection increases with the breadth of the ship. In this respect, the nuclear ship currently in operation, or some being built, are not in an especially good position since their sizes are somewhat small.

It is obvious that the importance and the arrangement of the structural parts which resist the effects of a collision are notably different whether the lateral spaces are used or not. If these spaces are void, it would be possible to incorporate a greater mass of resisting steel.

Another point of view is the distribution of the collision probability over the ship length. Different authors have evaluated and investigated the existing statistical material with the result that the probability of collision is generally reduced from the forebody to the afterbody. Based on this knowledge, the location of a reactor plant should be provided for at the afterbody. Here also the bending moment resulting from the impact of a collision is a minimum in the region aft of 1/4 L.

Roughly the collision protection structures can be divided into two types: the energy absorbing type and the resisting type.

The first type uses a combined structure of web frames, web stringers and a number of decks in the ship's side which provide deformable material to absorb the collision energy on a relatively long penetration way. This measure can be supported by strong decks which cut the forebody of the ramming ship using the cut bow structure of the opponent also for energy absorption.

The impact accelerations are small and estimated far below 1g. Figures 5.8, .9 & .10 give some examples of collision protection structures of N.S. SAVANNAH, N.S. MUTSU, and N.S. OTTO HAHN.

The second type consists of a strong side structure (Figure 5.11) which is able to destroy the forebody of the ramming ship without severe damage at the stricking vessel. The impact accelerations for the stricken vessel are higher so that a break through of the ship cannot be excluded absolutely. But, if it would happen, the vessel would break through fore or aft of the reactor region due to the strong protection structures in the reactor area. This is to be preferred instead of damage of the reactor itself. The two parts of the vessel would float due to the narrow subdivision. The resisting type was a German proposal for a collision protection of a nuclear powered containership.

Although there is no provision in the recommendation of classification societies on model testing, in order to verify the adequancy of collision resisting structures, there have been several investigations in that field conducted mainly in Japan, in Germany (see Figure 5.12) and in Italy.

Even though the importance of tests on models is not to be under-estimated, it is also necessary to formulate clearly

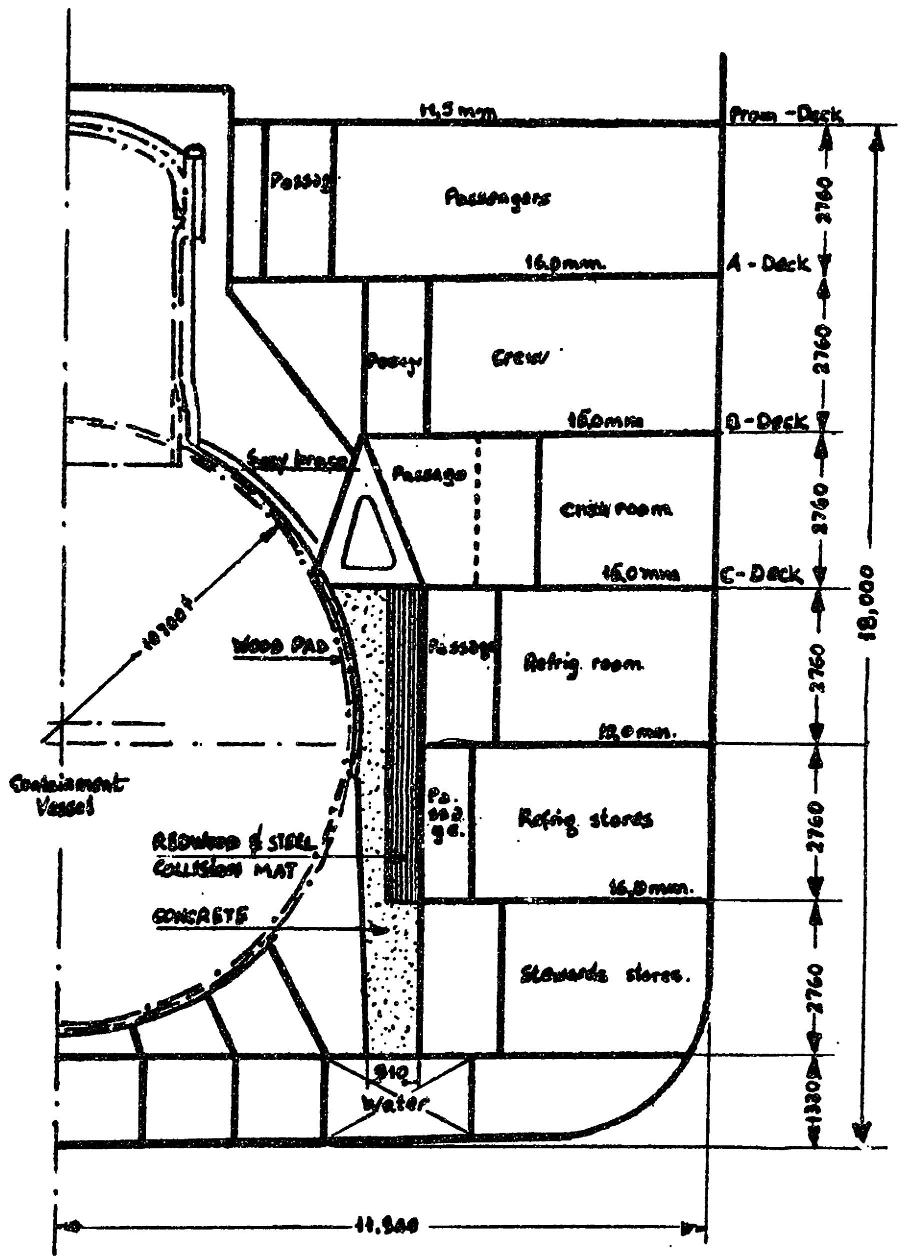


FIGURE 5.8 COLLISION PROTECTION OF
N.S. "SAVANNAH"

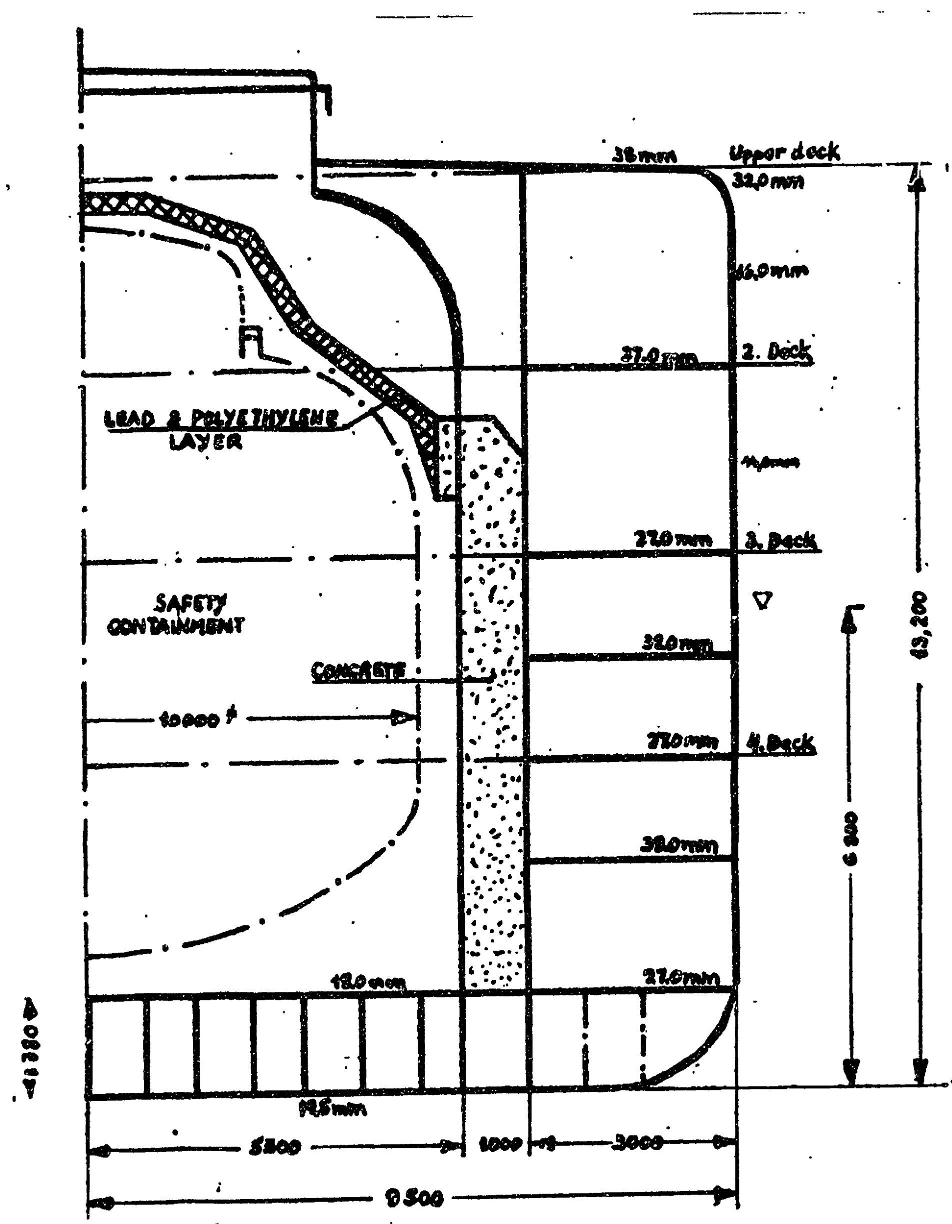


FIGURE 5.9 COLLISION PROTECTION
OF N.S. MUTSU

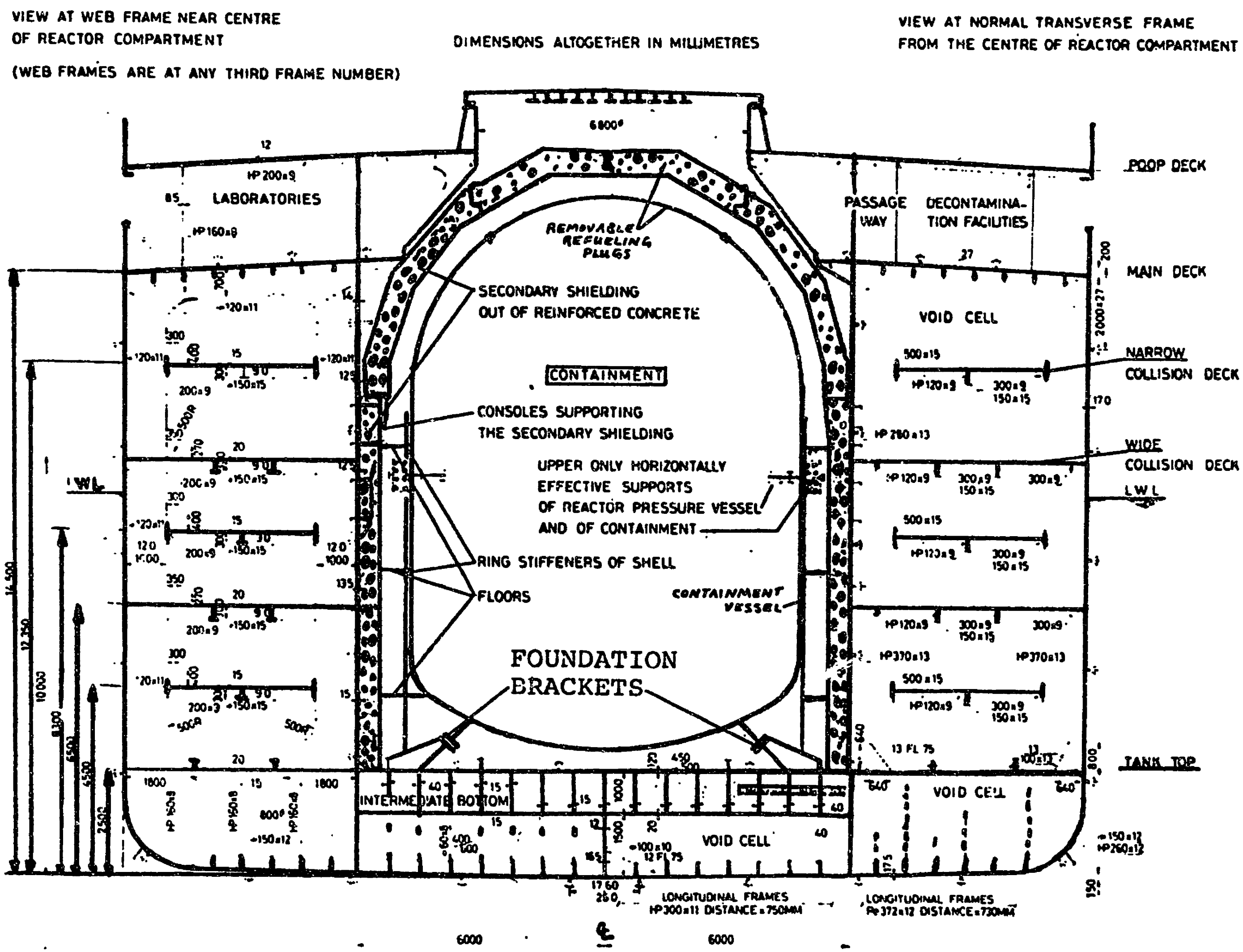


FIGURE 5.10 DETAILS OF N.S. "OTTO HAHN" REACTOR AREA STRUCTURE

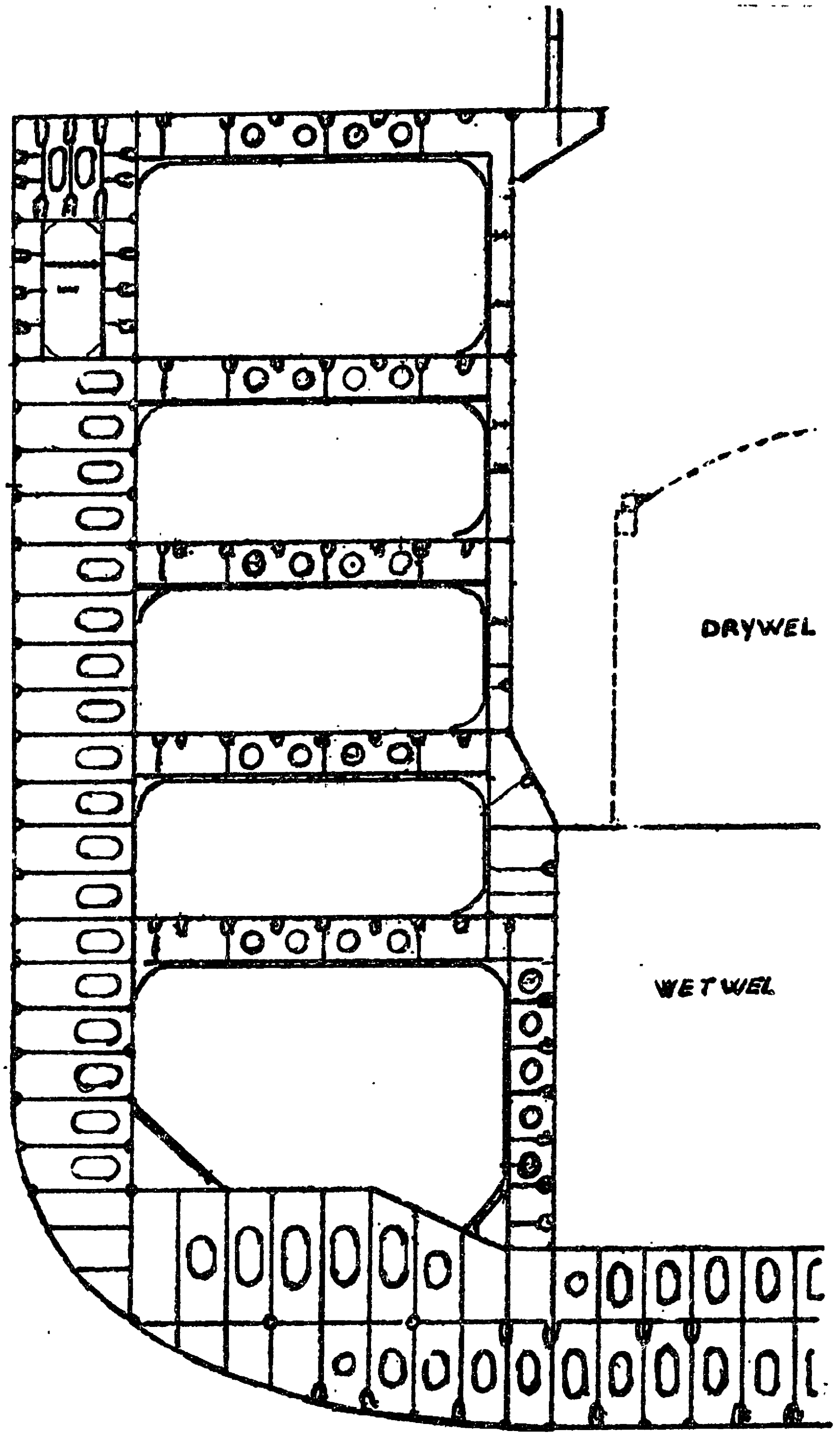


FIGURE 5.11 COLLISION PROTECTION, RESISTING TYPE

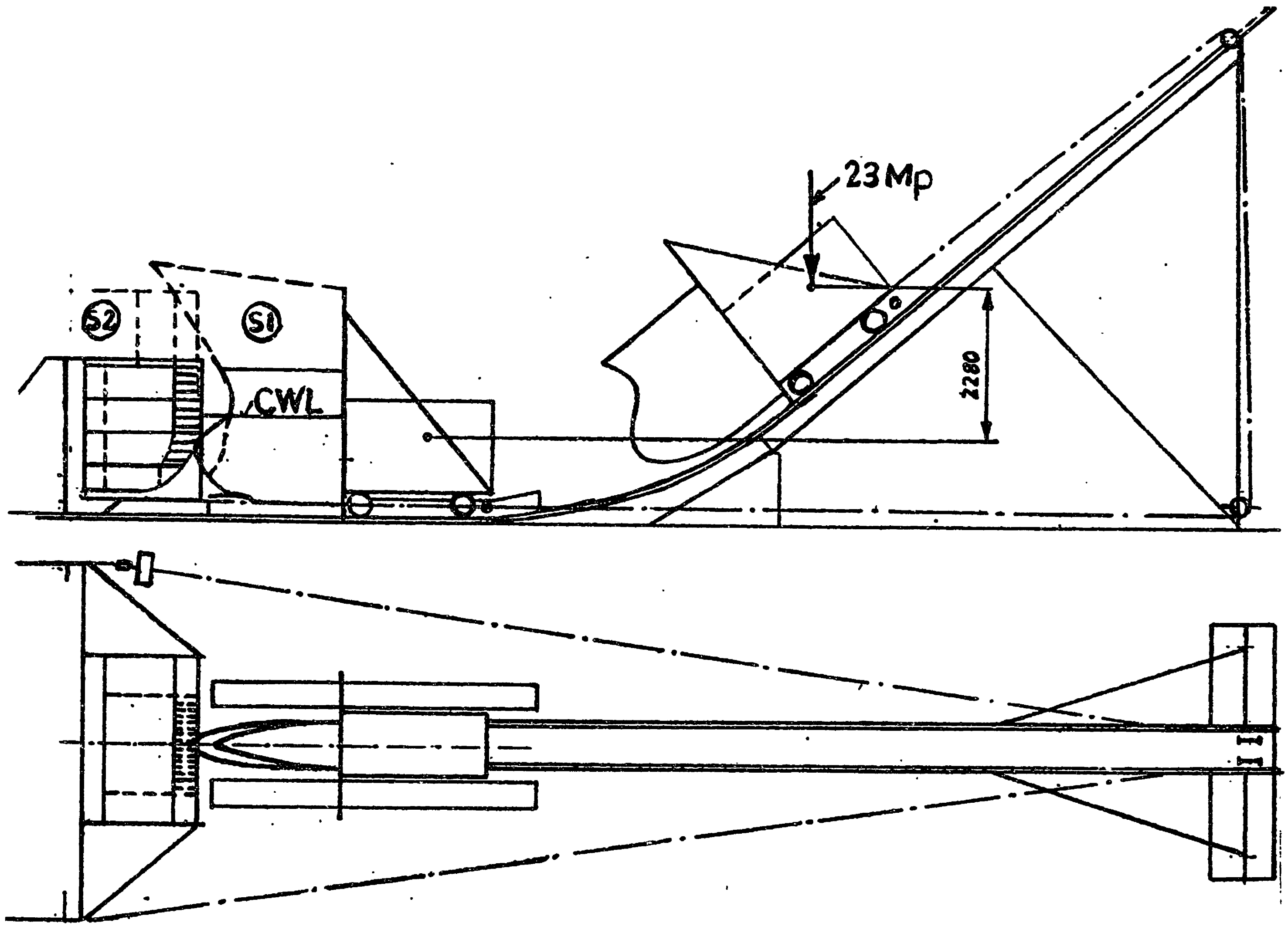


FIGURE 5.12 GERMAN COLLISION TEST FACILITY

what they are aimed at to put the problem in a general perspective. It must be acknowledged that an absolutely safe protection is not achievable and that a certain amount of risk is to be accepted. The gist of the problem thus lies in the evaluation of that risk and in assessing whether it is permissible or not.

Thus, the most efficient structure is to be devised through an adequate choice of its arrangement, of the scantling and of the material used always without undue cost.

5.5.1.2 Active Collision Protection

Collisions are largely attributable to human error and poor visibility. The best protection against collisions is to try to prevent collisions generally. Possible measures are having well trained nautical crews which can be achieved among other measures with ship simulators, and by the installation technical devices for detecting and tracing potential collision opponents, combined with a computer for calculation of the optimal change of course.

B&W units, CNSG have designed a suitable detection and warning system that can reduce the risk of collision and that can be installed aboard the ship. It is called the collision avoidance system. It works in conjunction with the navigation radar. This system can track, target course vectors, display potential areas of danger, display potential points of collision, provide automatic warning of an approaching target, and plot a safe course for the ship.

Three types of prevention are provided: a collision warning, a collision alarm and a lost-target return warning.

A collision warning occurs whenever a tracked target will pass within one mile in less than 20 minutes.

A collision alarm occurs whenever a tracked target will pass within 0.5 mile in less than five minutes.

A lost-target return warning occurs if a tracked target is not getting radar hits on 20% of the gate scans.

A lateral collision acceleration of 0.5g was estimated by the naval architects as being maximum for large ships such as are necessarily under consideration with nuclear propulsion (>900 ft., >10,000 tons).

As a consequence of a collision and depending upon the type of ship nuclear propulsion (bunker, container) and the type of the other ship or ships involved in the collision, it is very probably that a fire might start. If by chance one or both the ramming and the rammed ships are oil tankers, the fire probability is very high.

Thus, the position of the reactor plant within the ship can play an important role as long as it is isolated from the possible fire. Even the best fire fighting equipment that can be put on oil tankers generally is not enough to control fires on oil tankers after fire starts, especially from a strong collision.

In a ship it is possible to minimize the probability of accidents (collision) by avoiding congested port areas and changing course or speed to avoid arrival in congested zones during periods of bad weather. In the case of the Ultra Large Crude Carrier, the deep draft and large overall dimensions preclude its entry into such congested areas. Generally, they will be on service between deepwater offshore terminals, a situation that presents the minimum risk of collision since most accidents occur while approaching busy harbors especially when it is necessary to sail a large channel for harbor arrival.

Another way is to confine the navigable water for sailing in only one direction as it is recommended for the English Channel, and not to forget the efforts to improve the maneuvering capacity.

All these measures are part of the active collision protection which, combined with passive collision protections, can minimize the danger of damaging the reactor by collisions.

5.5.2 Grounding

Besides collision protection, attention must be paid to the grounding of ships.

The effects of grounding are crushing of the structure and shock. Experiences from groundings in the past indicate that the shock resulting from grounding is so minor as to be negligible.

5.5.2.1 Active Grounding Protection

This danger can be minimize by a nautically trained crew, supported by high maneuvering capacity of the ship at every condition.

As a deterrent to grounding, the ship must have a fathometer depth-sounder system that includes a shallow depth alarm as an accessory. An alarm is sounded on the bridge if the depth of water under the ship is less than a preset minimum. The system cannot ensure the safety of the ship, but it is an effective navigational aid and like the collision avoidance system for collision protection can reduce the risk of grounding, if properly used.

5.5.2.2 Passive Grounding Protection

A constructive precaution is the location of the reactor as high as possible but there are often limitations from large dimensions of the reactor or from ship stability factors.

Another possibility is a raised double bottom in the reactor area, which can be divided in height. The upper support bottom is able to support the reactor even in the case of destroyed lower ship's double bottom.

While the upper support bottom consists of a strong grid structure, the lower part consists of the normal bottom construction corresponding to classification standards. The deformation should be limited only on the lower bottom part. This construction principle has been used for N.S. OTTO HAHN (see Figure 5.10).

Since most groundings are bow-on, then the better position of the reactor, like for the collision protection, will be if it is sufficient aft in the ship. If grounding did occur, or if the ship struck an uncharted object involving the after portion of the ship, the effect on the reactor would still be expected to be minor.

In the event that the containment vessel were disturbed, the greatest source of concern for a nuclear plant such as that installed in N.S. SAVANNAH would normally be the distortion of the coolant piping. However, both in the N.S. OTTO HAHN and CNSG installation this danger is reduced with the reactor coolant system design (integral type).

The reactor vessel contains the core, the pumps and the steam generator. In the OTTO HAHN, since it is self pressurized, it doesn't have any piping at all. In the CNSG, it has piping that connects pressure vessel with pressurizer, but it is small compared with the loop type reactor coolant piping.

The most serious grounding situation is the grounding of the ship in unprotected waters or on a shoreline where the effects of surf and wind would pound the ship, cause pivoting upon the point of contact, and ultimately break up the ship. In the unlikely event of such an occurrence, the containment vessel would not be subjected to stress in excess of the impact stress, and it would remain intact. Owing to the heavier stringers and deck plating used as collision protection, the ship would break under such circumstances at points other than the structurally high-strength section in the way of the reactor containment. For this situation the resisting type structure, presented in the German proposal for a collision protection, would be satisfactory.

The effects of groundings are also affected by the type of surface struck, that is, smooth or jagged on a relative basis, such as sand bars versus rock. For the smooth surface grounding the available power coupled with ballast or cargo trimming would be sufficient to free the ship before it reached dangerous 45 degree list. The same is true for the jagged grounding provided the hull plating is not hooked fast.

The classification societies require only a bottom height in the reactor area which does not exceed the double height necessary for the corresponding ship size. It should be attainable that in case of a severe stranding on sharp edged rocks an imaginable break-through does not occur directly in the reactor area.

A lateral grounding acceleration of 0.5g was estimated by the naval architects as being maximum for a ship of the size under current study (>900 ft., >50,000 tons).

The heat sink for the cooling system is the water surrounding the ship, and this water is piped into and out of the ship via "sea chests". To ensure that a supply of cooling water is available even in the case of grounding, sea chests are fitted high and low both starboard and port. These are interconnected through appropriate isolation valves and serve all auxiliaries requiring cooling water.

5.5.3 Flooding & Sinking Problems

Two consequences are of major concern relative to flooding and sinking: the safety of personnel aboard and insurance that the sunken nuclear plant does not constitute an environmental hazard. The most likely cause of flooding and sinking is a collision where the nuclear ship is struck by another ship. In the case of the CNSG where in accordance with USCG regulations, the ship is designed to withstand two compartments damage and flooding, any two adjacent compartments may be flooded and the ship will remain afloat with adequate stability. The consequences of flooding damage to all possible critical combinations of these compartments were examined and acceptable GM, draft, trim, and angle of heel resulted in each case.

In spite of the manifold safety measures, the sinking of a nuclear powered merchant ship cannot be excluded absolutely.

If this improbable case would happen, the safety containment must be protected against collapsing by the rising water pressure for two main reasons.

First, the safety pressure vessel has to maintain its integrity also in case of sinking. Second, parts of the collapsing safety vessel could destroy important components of the reactor and to initiate a nuclear accident.

It is also important to get water into the safety containment in order to remove the decay heat from the reactor after shut down.

For all these reasons flood openings must be provided for. Possible flood openings are valves or flaps as in the case of N.S. SAVANNAH and N.S. OTTO HAHN or rupture discs in connection with butterfly valves or some combination of that in the reactor compartment and reactor containment.

The area of flood openings depends on the free inner volume of the containment and compartment, the sinking velocity of the ship, which is individual for every vessel at different loading condition, the flow contraction coefficient depending on the opening construction and the pressure level where the flooding process starts.

The flooding process can be divided into two phases: the phase of slow sinking where the ship compartments are flooded gradually, (within minutes or hours), and the phase of fast sinking which takes only a few seconds and to which special attention must be paid. Figure 5.13 shows typical different pressure curves, differences between outer and inner pressure of the safety vessel at the period of fast sinking in dependence of increasing flood opening areas. These differences are valid only for one individual ship and one loading condition. The chosen opening area determines the design pressure of the safety enclosure. It is the matter of outer pressure which must be considered by lay out of cylindrical or spherical shells. Different classification societies recommend an outer pressure of about 3 atm for the containment design.

Due to its installation aboard a ship, the containment and the component and machinery within it must be designed to remain in place regardless of orientation if the ship should capsize and eventually sink. Thus, using automatic

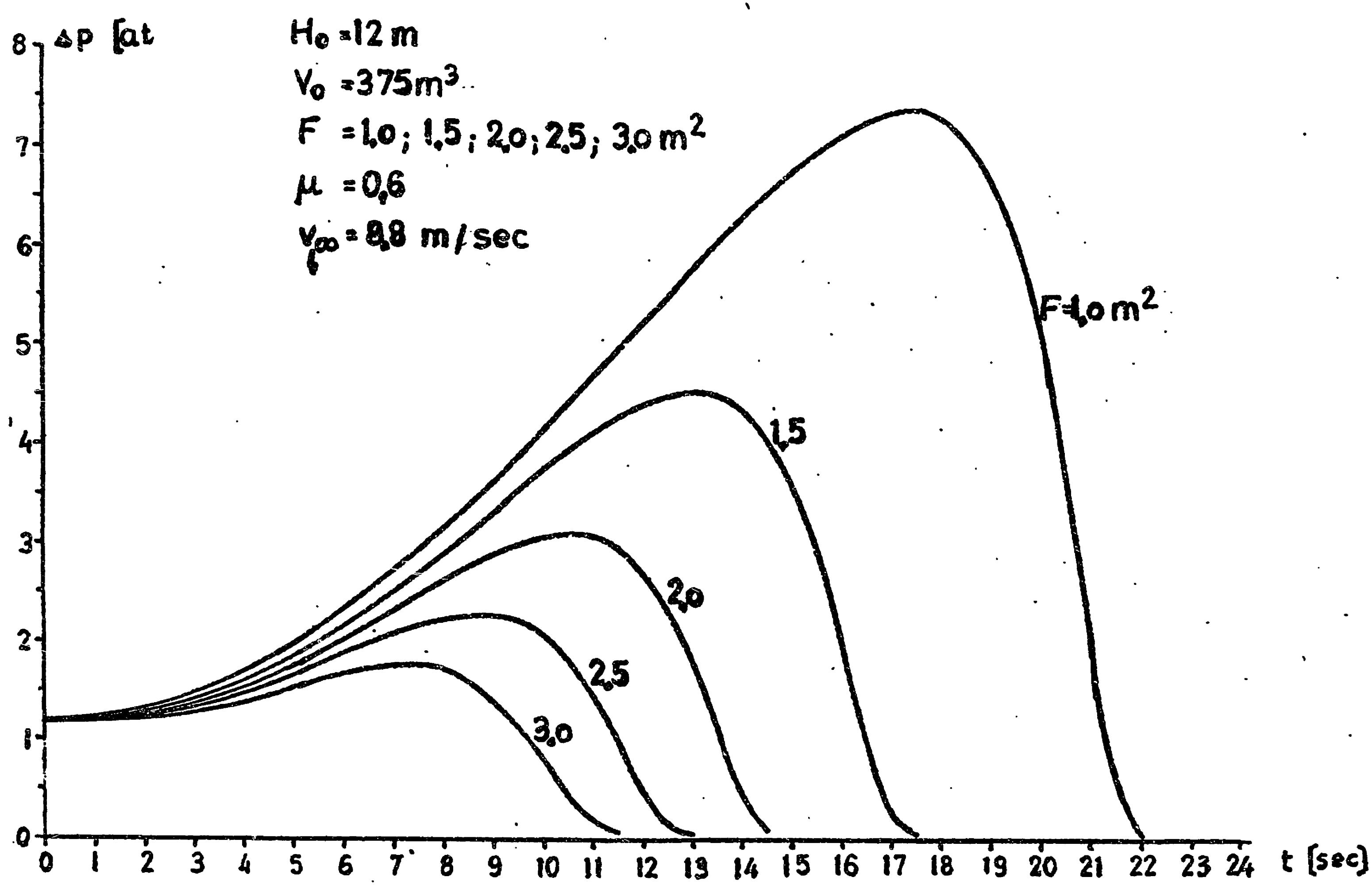


FIGURE 5.13 TYPICAL DIFFERENT PRESSURE CURVES OF THE SAFETY VESSEL FOR FAST SINKING

pressure valves, which allow flooding to relieve the pressure gradient, the containment vessel must remain intact without collapse due to external pressure if the ship sinks.

The inward flooding valves used in CNSG are held closed by a bolt which has been reduced in cross section along a portion of its length. This bolt holds the valve shut until the external pressure gradient caused by a sinking accident cause tensile failure in the bolt's reduced cross section. The valve then opens and allows the containment to flood and the pressures to equalize. A spring closes the valve and holds it shut when the pressure is equalized inside and outside the containment.

The valves must meet the following requirements:

- Open fully at the set pressure
- Remain open until pressure equalize
- Have sufficient capacity to prevent the rate of change in the external pressure from exceeding that of the internal pressure, so that the resulting pressure gradient would not cause failure of the containment shell
- Reclose once the pressure has equalized
- Valve material must be compatible with a seawater environmental
- Valve must be capable of reopening if the sunken ship shifts and settles to a greater depth

To assure that the possible thermal expansion of seawater inside the containment, owning to the stored energy from hot components and decay heat from the core that can it pick up especially if it is combined along with a LOCA, will not cause an over pressure,it is necessary to fit a relief valve outside the containment that will relieve to the reactor compartment. Inside the containment upstream of the relief valves would be a rupture disc. Both rupture disc and relief value will be designed to relieve at determined containment pressure.

If unavoidably sinking occurs and an accident in the pressure vessel, such as collapse due to failure of the inward flooding valves of the containment to open, then a seawater inleakage to the primary cooling system might occur. If chemical shim is used to control the core reactivity, this seawater inleakage would wash out the soluble boric acid solution and a dangerous reactivity addition into the core would occur. A consequence of that very improbable possibility is that, in the CNSG, the use of chemical shim as regular reactivity control system is avoided. The boric acid addition systems is only foreseen, as was said in 4.1, as utmost case to shut the reactor down.

5.5.4 Capsize Problems

As a consequence of a flooding or sinking accidents, a ship can capsize. To avoid the consequences of a ship capsize accident, some reactor parts must be designed and assemblied in a different way from that used for a stationary reactor. Thus, the reactor internals must be designed to transmit the entire weight of the core and internals to the upper vessel head without exceeding the material strength values for a faulted condition.

The foundations for reactor containment vessel and its associated biological shielding must be designed to retain these structures in place in all ship orientation including absolute ship capsized condition.

The biological shielding surrounding the containment is mounted in CNSG on a flanged foundation to provide lateral support at the base and, in the case of a complete capsize, a retaining ring at the top of the shielding keeps it in place by means of tie rods to the foundation.

SHIP COLLISION & GROUNDING SAFETY INTERLOCK ANALYSIS

Interlock	COMMENTS & CONSEQUENCES	Applicable Design Criteria
Normal Decay Heat Valve	Piping, valves & initiating safety instrumentation must be designed to continue functioning despite of a collision or grounding event.	USCG Port 56
High Pressure Injection System	Instrumentation, valves & piping must be designed to continue functioning despite of collision or grounding event.	"
High & Low Pressure Injection Change-Over	Piping, valves, make-up tank, & instrumentation must be designed to continue functioning in the event of collision or grounding.	"
Boric Acid Addition System	Piping & valves must be designed to continue functioning on the event of collision or grounding.	"
Liquid Waste System Discharge	Pipe rupture upstream of the isolation valves releases radioactive liquid inside station in shielded areas. Rupture downstream or between series valves still leaves system discharge isolated.	"
Gaseous Waste System Discharge	System, including valves, piping and tank must be designed to continue functioning in the event of collision or grounding.	"

CHAPTER VI

TECHNOLOGICAL DIFFERENCES DUE TO THE ECONOMIC FACTORS

6.1 Economic Factors

Volume and weight reduction play a very important role in the maritime applications reactors because less volume and weight for the same power, implies more cargo carrying capability and therefore more profit potential. In stationary plants, this problem does not exist.

The refueling operation as well as refueling facilities are, due to the necessity of the volume and weight reduction, totally different from the stationary plants. Thus, whereas in maritime reactor plants a dry refueling system is used, in stationary reactor plants, a wet refueling system is used.

Another feature which must be looked is the fuel cycle duration. Whereas, long-life cores are needed for ship operation, they are not necessarily required for industrial usage where fuel economics is more important.

6.2 Volume & Weight

From the economic point of view, two desirable characteristics necessary for ship board power systems are compactness and lightweight. To fulfill these requirements many reactor power plant components are designed differently, depending upon their application in stationary maritime reactors. From this point of view, the main differences fall upon the following categories:

1. Containment System

2. Refueling System
   - Refueling Facilities
   - Refueling Technique
   - Refueling Procedure
   - Refueling Time

3. Radiation Shielding System

It is important to emphasize that those requirements also put nuclear ship reactors at a disadvantage with those of stationary plants:

- Because of space limitations, the machinery of the working part is much more crowded than in stationary land-based plants.
- Because of weight and space requirements, the ship, spare parts and maintenance facilities are kept to a minimum.

6.3 Containment System for Integral PWR

Under the design criterion of compactness and lightweight, the better containment system to be selected is a pressure-suppression type. The pressure-suppression containment type reduces the overall size of the envelope around the reactor and minimize weight by keeping the shell design pressure down.

Because the pressure-suppression containment concept has heretofore been employed only in stationary BWR power plants, it is necessary to evaluate its performance for PWR in its maritime application under simulated accident and ship attitude conditions.

So far, only Babcock and Wilcox, in its CNSG, have designed such a type of containment. The other ships have always used a dry containment type. From Table 6.1, it can

TABLE 6.1 COMPARISON OF CONTAINMENTS FOR DIFFERENT NUCLEAR REACTOR PLANTS

	S SAVANNAH	H OTTO HAHN	I MUTSU	P CNSG	S PMNP-Platform Mounted Nuclear Plant	Typical Stationary PWR Westinghouse
Reactor Power (Mwt)	80	38	36	312	3411	3411
Containment Shape	Horizontal Cylinder w/ hemispherical	Vertical Cylinder	Vertical Cylinder	Vertical Cylinder	Vertical Cylinder	Vertical Cylinder
Containment Height	51' 10 3/4"	42'7"	34'7"	64'	102'	205'9"
Containment Diameter O.D	35' 4/3/4"	--	--	--	--	--
I.D		29	32'10"	38'	120'	124'
Containment Length	50'8.5"	--	--	--	--	--
Containment Type	Dry	Dry	Dry	Pressure-Suppression	Dry	Dry (ice condenser)
Containment Wall Thickness	1. 1/4 " (at hemispherical end) 3 3/4" (cylindrical portion	1.18' & 1.57 (Platting & botton) 0.78' (top)	1.42' (cylinder portion) 2.36' (spherical portion)	1.25'	--	3.5' cylinder port 2.51 dome
Containment Gross Volume (Cu. ft)	~52,250	~28,130	~29,300	~72,600	~1,155,000	2,500,000

TABLE 6.1 COMPARISON OF CONTAINMENTS FOR DIFFERENT NUCLEAR REACTOR PLANTS (Continues)

$\frac{\text{Gross Volume}}{\text{Reactor Power}} \left(\frac{\text{Cu. ft.}}{\text{Mwt}}\right)$	$\sim$650	$\sim$740	$\sim$810	$\sim$230	$\sim$340	$\sim$730

be seen that maritime containment designs have progressed, looking for volume and weight reduction, especially when, as in the case of CNSG, the reactor was designed for a commercial ship more than a research ship as in the case of SAVANNAH, OTTO HAHN and MITSU.

Figures 6.1, 6.2, 6.3, & 6.4 show how the containment shapes were changing to reach better performance in the sense of reduced volume and weight. On the other hand, while in the PWRs for maritime applications, the containment used should be a pressure-suppression type, for stationary application it is used a dry containment. As can be seen, from Figures 6.5, 6.6, 6.7, & 6.8 , for stationary PWR plants to keep the containment lower is not a design requirement.

One of the most important technological difference between these two types of containment is connected with one of the containment system components, the heat removal system. Thus, in the pressure-suppression containment system heat removal is performed by:

1. The Containment Cooling System: it is the primary system that provides heat removal after an accident.
2. The Suppression Pool-Cooling & Emergency Spray System
3. Decay Heat Removal System: this system is preferably designed for heat removal in reactor shutdown and refueling.

In the dry containment, heat removal is performed by:

1. Containment Fan Cooler System. This is not a safety system. It works in any condition but especially in normal

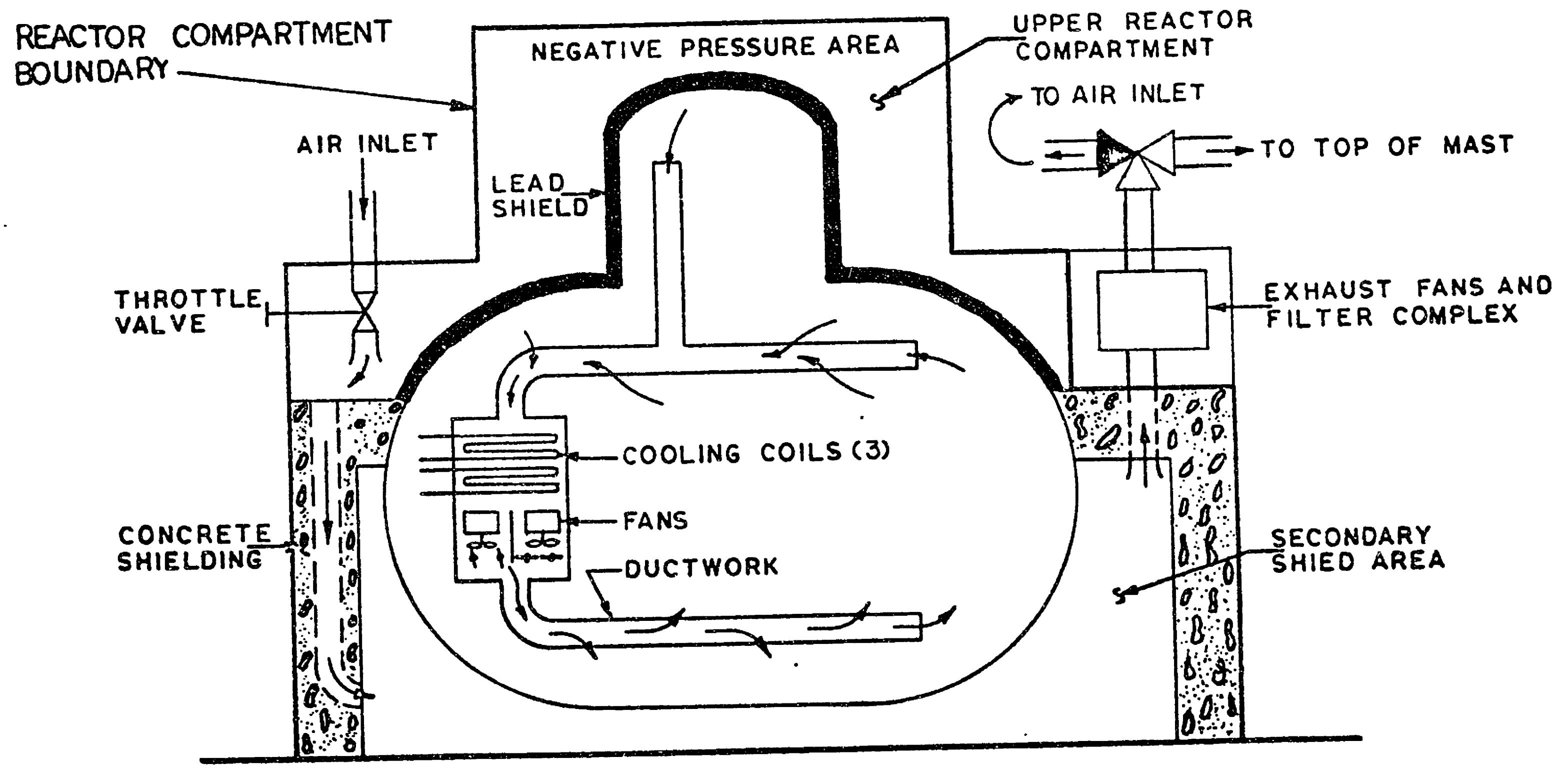


FIGURE 6.1 SIMPLIFIED DRAWING OF THE CONTAINMENT SYSTEM OF N.S. SAVANNAH

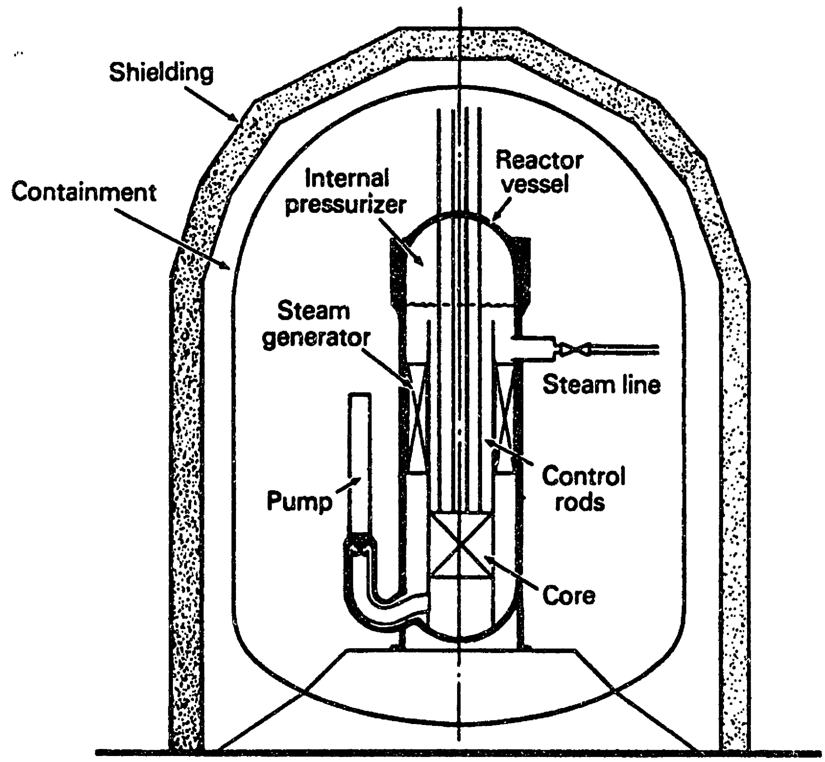


FIGURE 6.2 SIMPLIFIED DRAWING OF THE CONTAINMENT OF N.S. "OTTO HAHN"

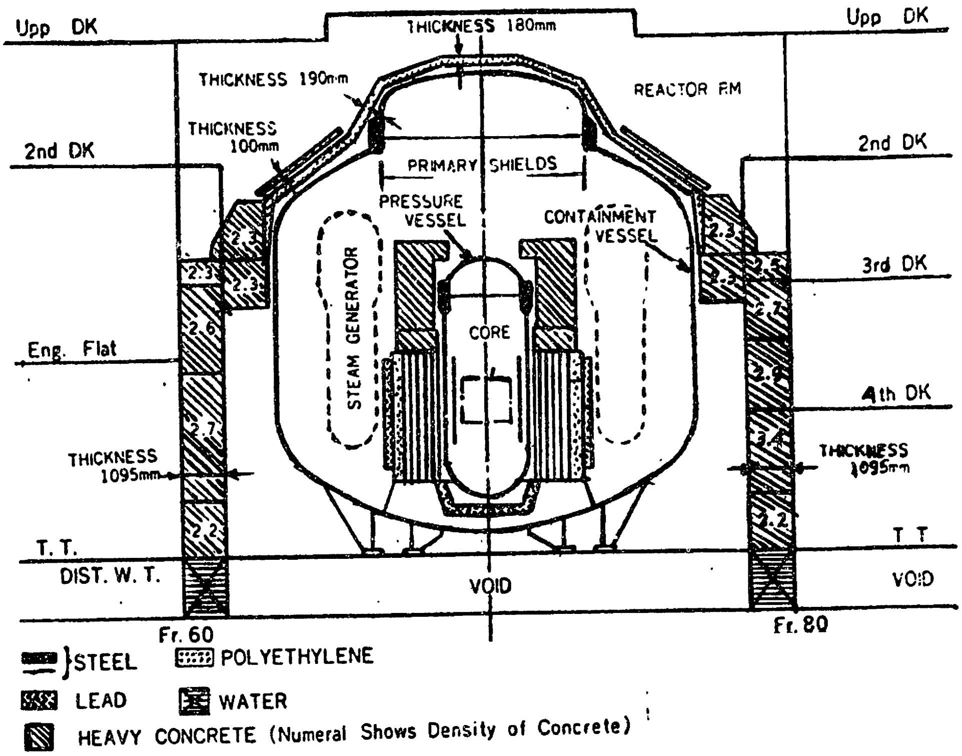


FIGURE 6.3 N.S. MUTSU REACTOR CONTAINMENT VESSEL AND RADIATION SHIELDING DETAILS

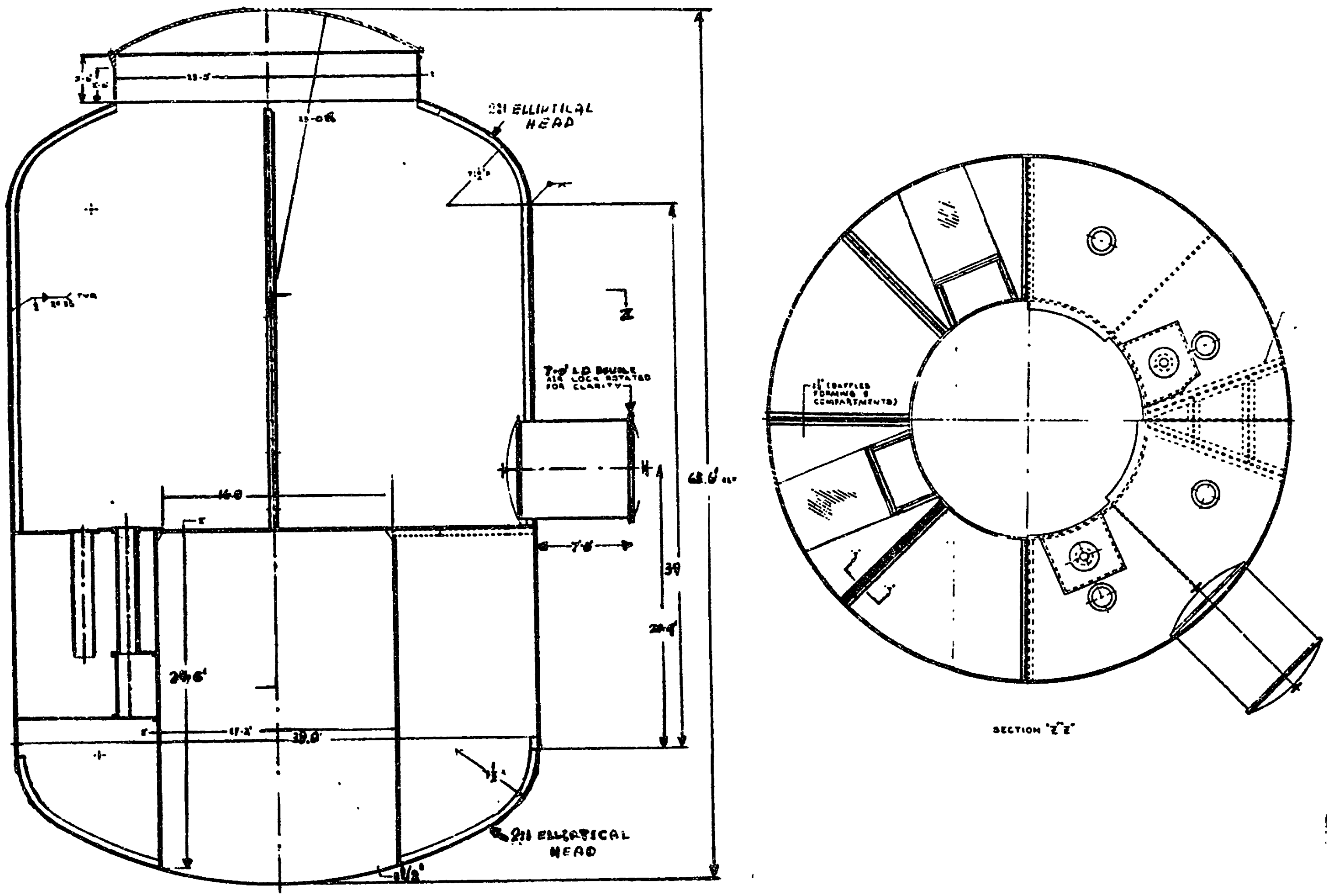


FIGURE 6.4 CONTAINMENT OF INTEGRAL CNSG IV

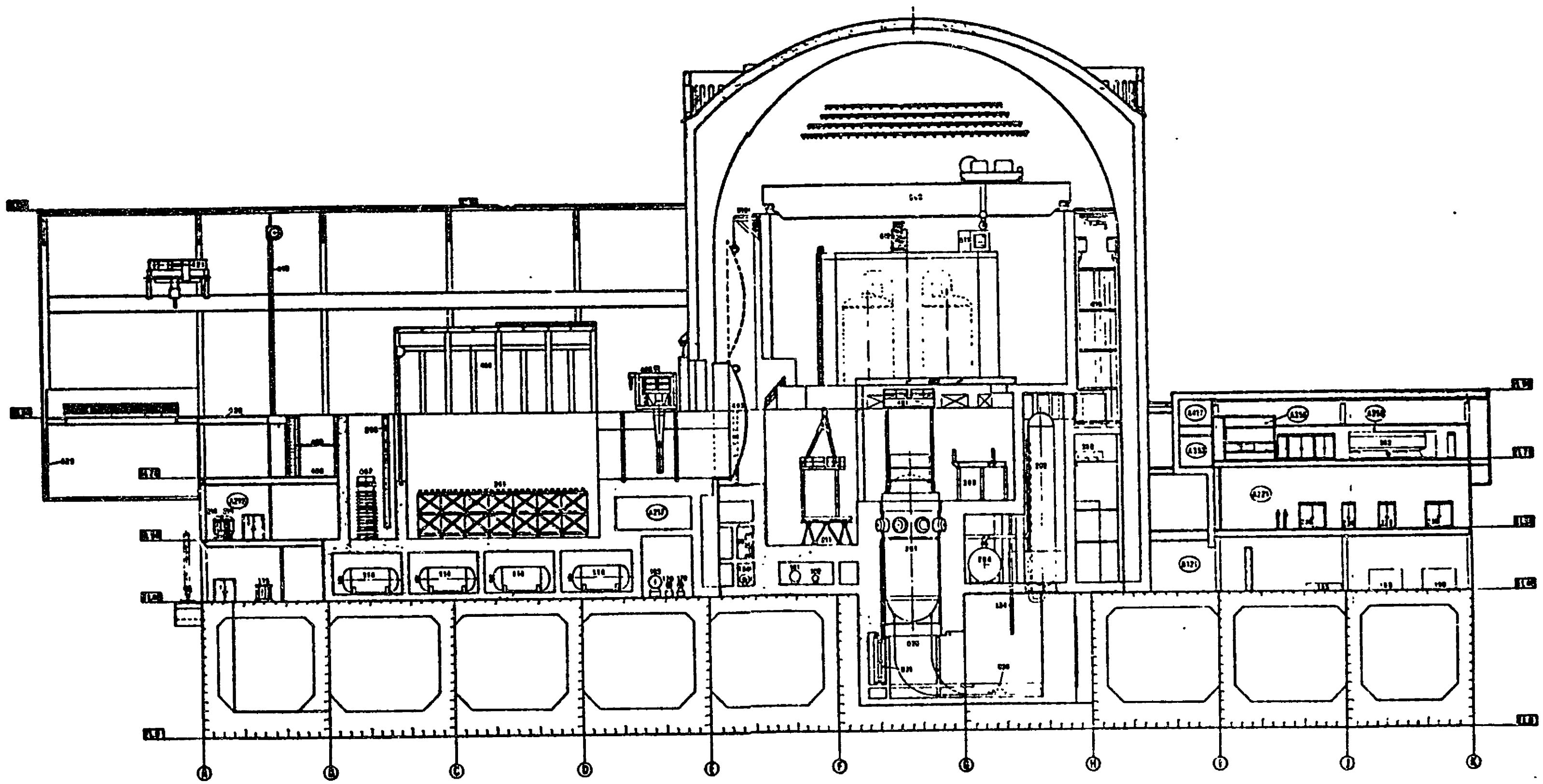


FIGURE 6.5 CONTAINMENT OF A PWR FOR A PLATFORM MOUNTED NUCLEAR PLANT

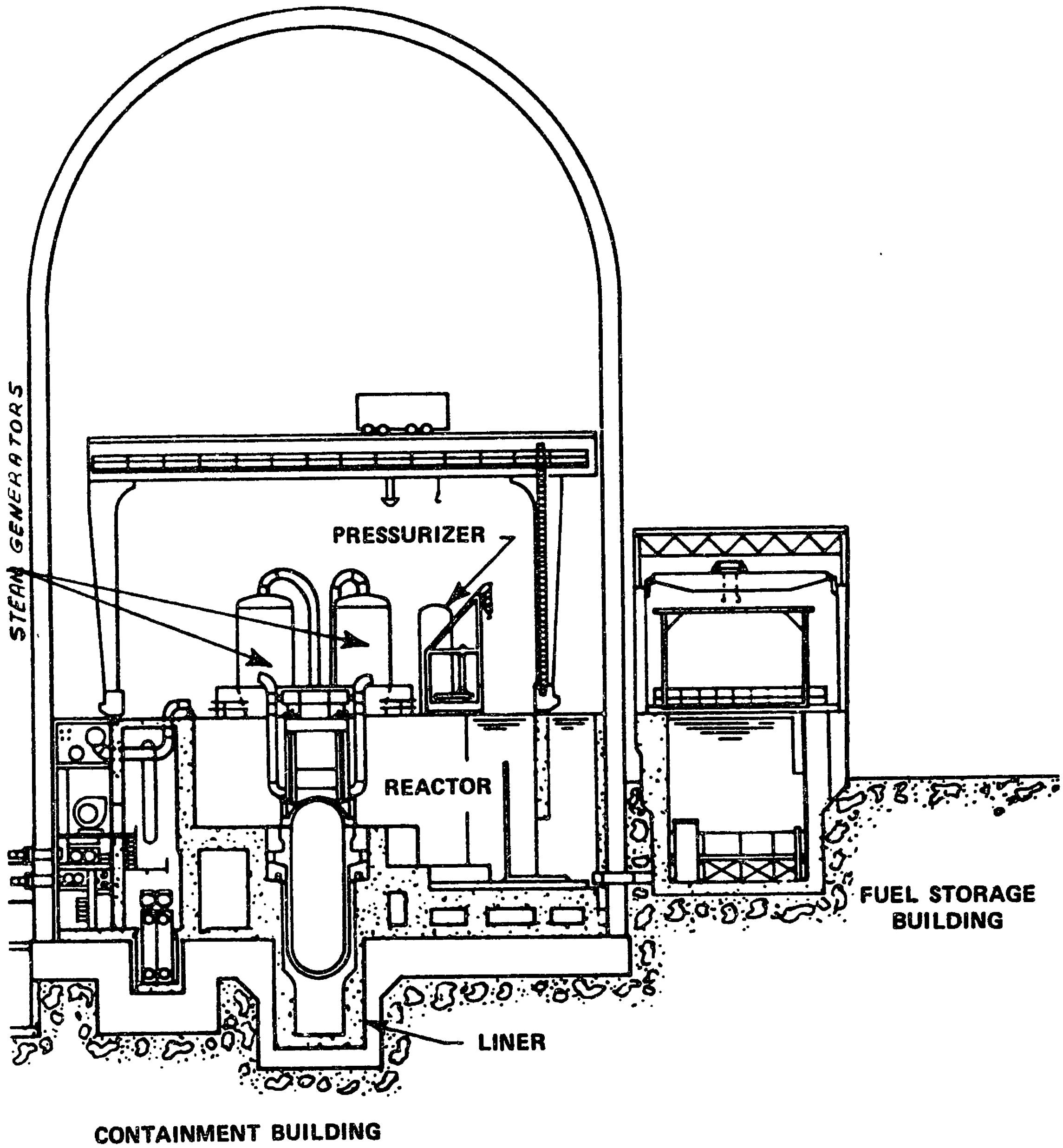


FIGURE 6.6 CONTAINMENT OF A TYPICAL STATIONARY PWR PLANT

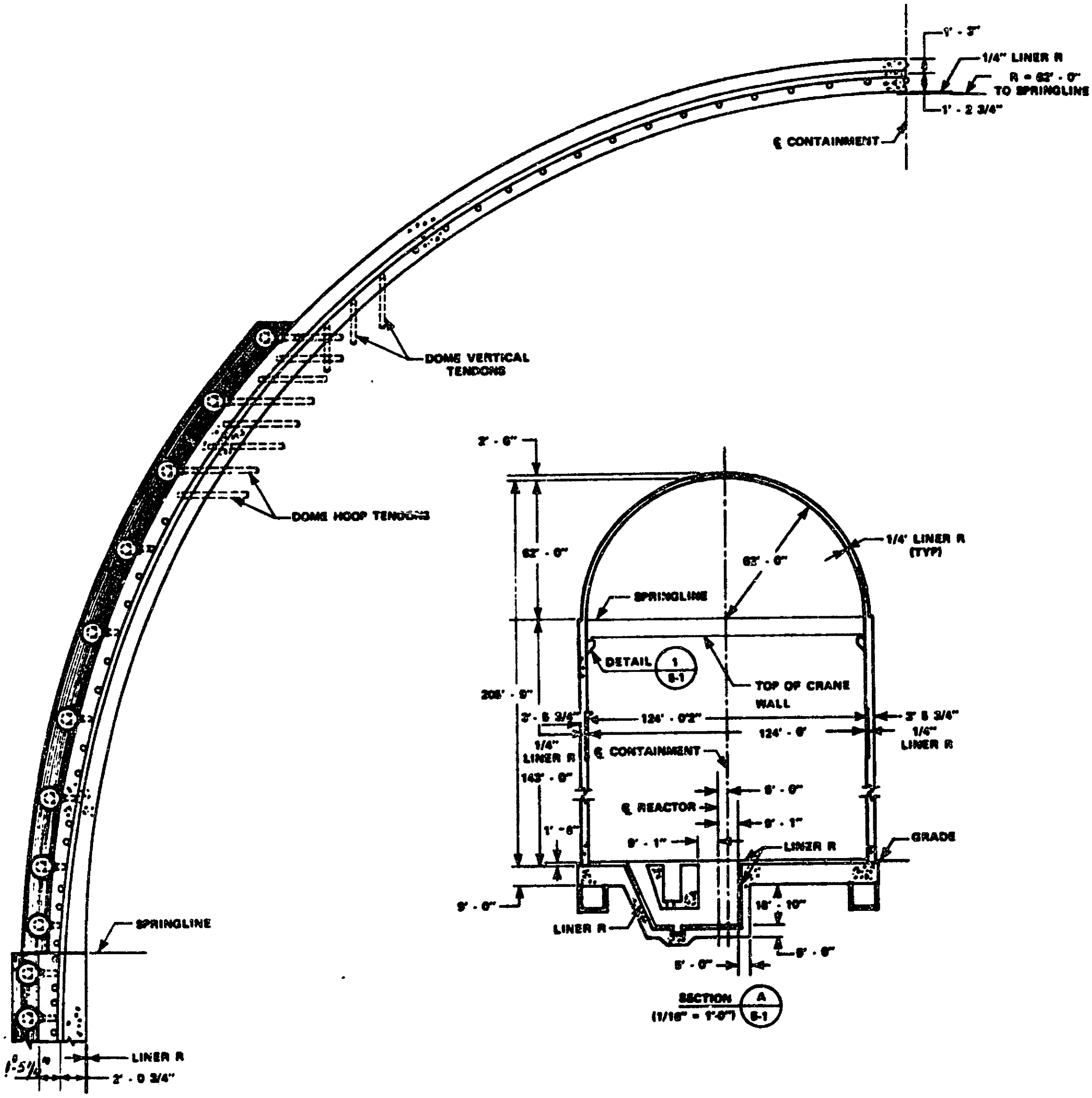


FIGURE 6.7 CONTAINMENT CONSTRUCTION OF A STATIONARY PWR PLANT

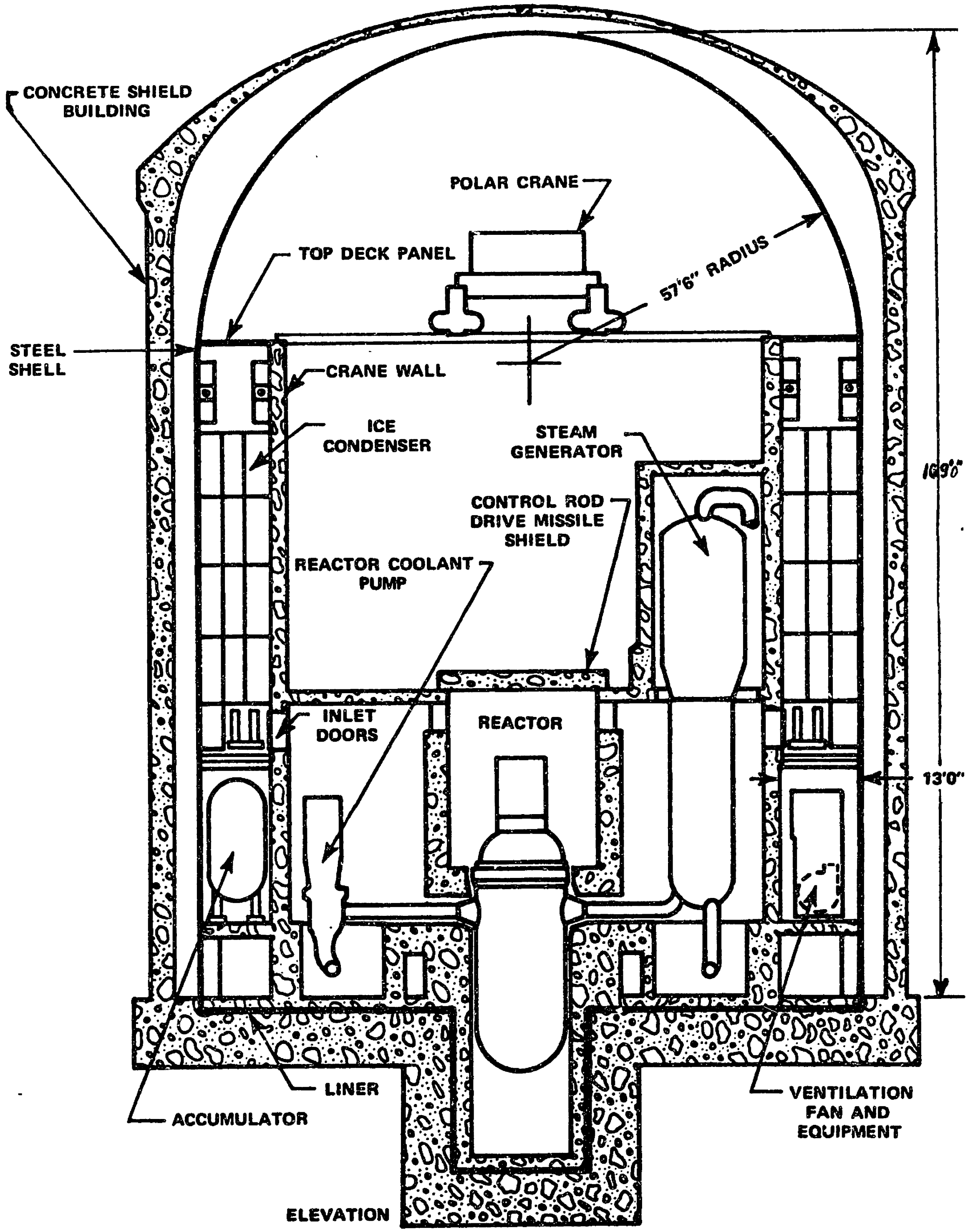


FIGURE 6.8 CROSS SECTION OF ICE CONDENSER CONTAINMENT OF STATIONARY PWR

and after LOCA conditions.

2. Separate Containment Spray System Train whose components operate in sequential modes.

3. Ice Condenser Reactor Containment System.

Although the containment cooling systems of both pressure-suppression containment and dry containment are similar, in the last the cooling system provides jointly cooling and air purification while pressure-suppression provides only cooling. This is due to the fact that in pressure-suppression the purification is provided separately by the containment air purification system.

The other two components of heat removal systems differ completely in their design. Thus, the ice condenser reactor containment system used in dry containment could never be used in a maritime reactor containment due to its great volume (see Figure 6.8).

6.4 Refueling System

6.4.1 Refueling Facilities

Due to the problems of volume and weight, the new and spent fuel in maritime reactors is stored outside of the ship, in an outside facility. This is because a ship board self-serving facility is considered impractical on nuclear merchant ships due to the wasted cargo space that the facilities would occupy and the expense associated with duplication of facilities on every ship.

There are presently two different concepts of facilities that can be considered for use in maritime applications. These concepts are:

1. Permanent shore-based service facility

2. Barge mounted mobile service facility

To choose between these two concepts, it is necessary to analyze recommended criteria such as:

- The harbor and channel leading to the site should have sufficient depth and maneuvering room to accommodate the vessels to be using the facility.
- The site selected should be reasonably convenient to the vessel's normal cargo loading port and trade route.
- The site selected should be in a shipyard or within towing distance of one capable of supporting maintenance, repair, and dry docking of the ship.
- The site should be capable of enclosure by fence to provide access control and ideally should be located in an area of relatively low population density.
- Seismological, hydrological, and meteorogical conditions encountered in the area and their effect on refueling operations should be considered.
- The pier area should not experience large tidal ranges, current speed or surface disturbances.

6.4.1.1 Permanent Shore-Based Refueling Facility

Assuming that all site locations meet the previous criteria, there are for permanent shore-based refueling facilities the following possible locations:

- A site completely separate from a shipyard
- A nuclear repair or construction yard.

- A Non-nuclear repair or construction yard.

Each one has its advantages and disadvantages.

The location outside a shipyard results in the most extensive and costly facility, since many of the shipyard piers, cranes, shops, storage areas, and trained personnel normally available in a shipyard will have to be duplicated. But, it could be necessary in case shipyard facilities were not available at the port most convenient for the ship owner. Whenever the number of ships that will use this facility economically justifies its construction, facility location at a nuclear ship construction yard has the advantage that many of the required facilities would already be present.

If the facility is located at a non-nuclear shipyard, it will be necessary to add at the normal facility others such as radioactive waste processing and storage, contaminated machine shop, decontamination facilities, new fuel storage area, and nuclear trained personnel since a shipyard location should allow any required maintenance, repair, or inspection of the ship to proceed simultanecusly with the refueling.

At the beginning a refueling facility location in a nuclear ship construction yard should be recommended as the preferred site.

6.4.1.2 Mobile Refueling Facility (see Figure 6.9)

This kind of facility would offer some advantages to an owner over a fixed-shore facility such as:

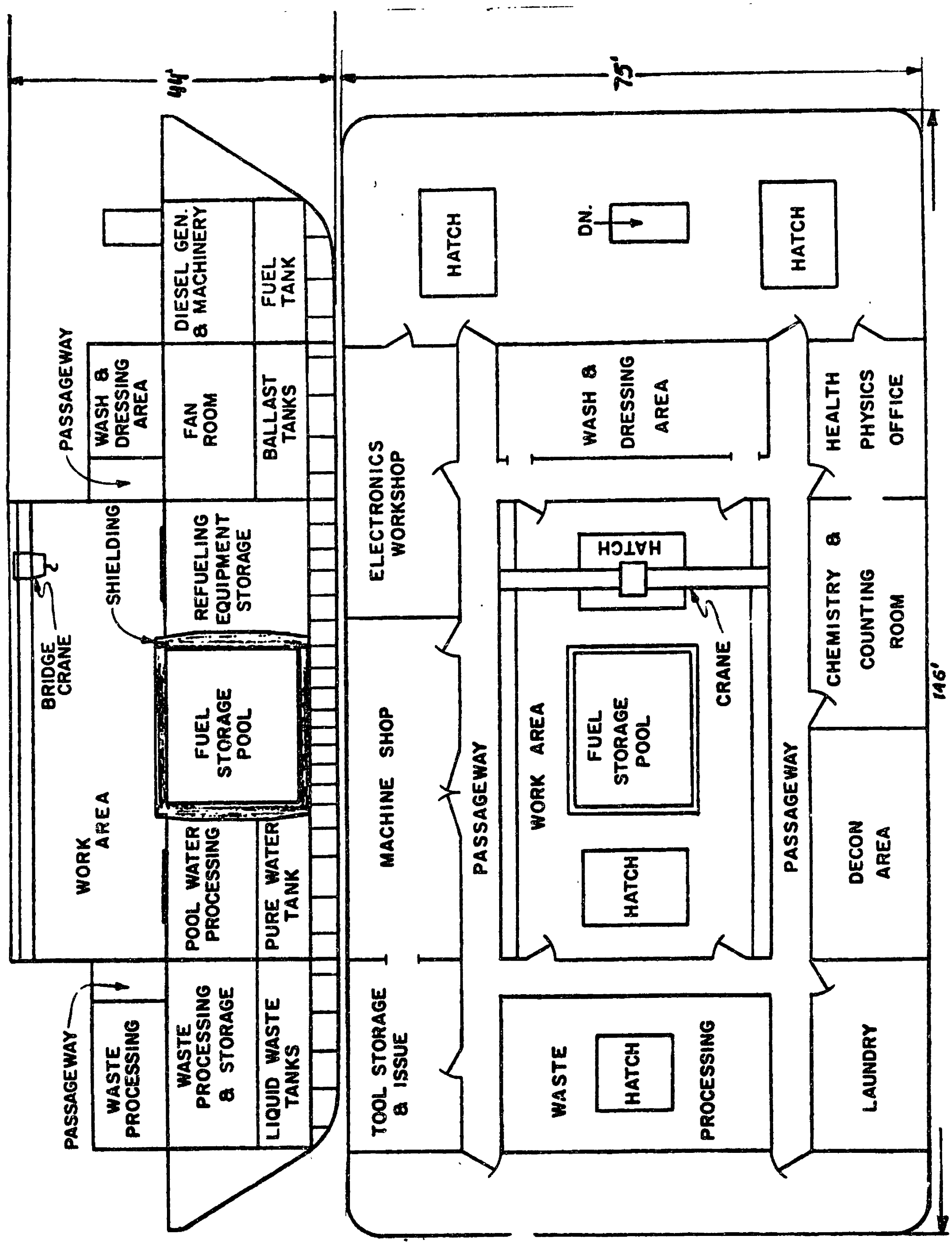


FIGURE 6.9 DIAGRAM OF BARGE MOUNTED REFUELING FACILITY

- The owner is not restricted to having refueling at one location that may not be convenient to all ships.
- A mobile facility would allow major nuclear repairs to be performed almost anywhere with minimum disruption of the nuclear ship's operating schedule.
- It gives to the owner the flexibility of changing a ship's trade routes without having to locate another shore based facility convenient to these new trade routes.
- It could be used to transport the spent fuel to or near the reprocessing plant, avoiding spent fuel transfer to another kind of transport and may be resulting in a cheaper and reliable transport system.

On the other hand, there are some disadvantages that are necessary and must be taken into account for a valid comparation such as:

- The fact that the initial and upkeep cost are greater
- Facility expansion would be different
- Licensing of a mobile facility would be more difficult than a fixed shore facility

6.4.2 Refueling Technique

6.4.2.1 Description

A fuel storage facility (service building) for a nuclear powered ship is similar in concept to a land-based unit. However, the actual fuel handling from ship to facility is entirely different in procedure and design.

The refueling procedures used in the refueling operation differ greatly from a stationary reactor to a maritime reactor since, due to the space and weight problem on ship reactor installation, the reactor plant refueling installation and refueling equipments used in both are fairly different.

The refueling scheme proposed for a maritime reactor is referred to as dry-type fuel removal as opposed to the wet-type removal used for all or most stationary plants. Briefly, the major difference in the two schemes is that the dry type utilizes a rotating index shield transfer cask arrangement for removing and transporting fuel to the spent fuel pool. The cask is filled with water during transfer and cask handling. A wet-type scheme utilizes a positioner bridge with a fuel handling tool, and fuel is transported entirely under water.

6.4.2.2 Equipment & Installation Description

Maritime Reactor Plant

Except in the case of OTTO HAHN, no other maritime reactor has special installation onboard for refueling operation. A heavy, portable lift crane, mounted either on the ship's deck, barge mounted refueling facility or in the dock of the permanent shore based service facility, can be considered for the removal or replacement of heavy equipments, such as the reactor vessel, closure head, internals transfer cask, and fuel transfer cask. In the case of a permanent shore-based serving facility, the equipment would be lowered from the ship to the service pier and placed on a mobile flat car for final transportation to the service building and spent fuel pool. In the case of Barge Mounted Refueling Facility, the equipment would be lowered from the ship to the Barge Facility directly. In addition to the heavy lift crane, equipment for the shipboard refueling operation can be separated into three main groups:

Group 1: Head Handling Equipment (See Figure 6.10)

This equipment comprises:

- Stud Handling Tools
- Lead Handling Slings which insure a true vertical lift of the closure head into and out of the containment.
- Lead Storage Stand which supports the closure head and provide shielding during storage, maintenance and in-service inspection.

Group 2: Internals Handling Equipment (See Figure 6.11)

This equipment comprises:

- Internals Transfer Cask which is a steel-lead structure designed to support the internals components and to provide shielding during storage and in-service inspection.
- Internals Cask Handling Sling which insure a true vertical lift of internals into the cask and subsequent removal of the cask from the containment.

Group 3: Fuel Handling Equipment (See Figure 6.12)

This equipment comprises:

- A Rotary Manipulator which is provided to accurately locate the fuel transfer cask over a particular fuel assembly and provide shielding. This component sits on the reactor vessel seal flange. It is a closed

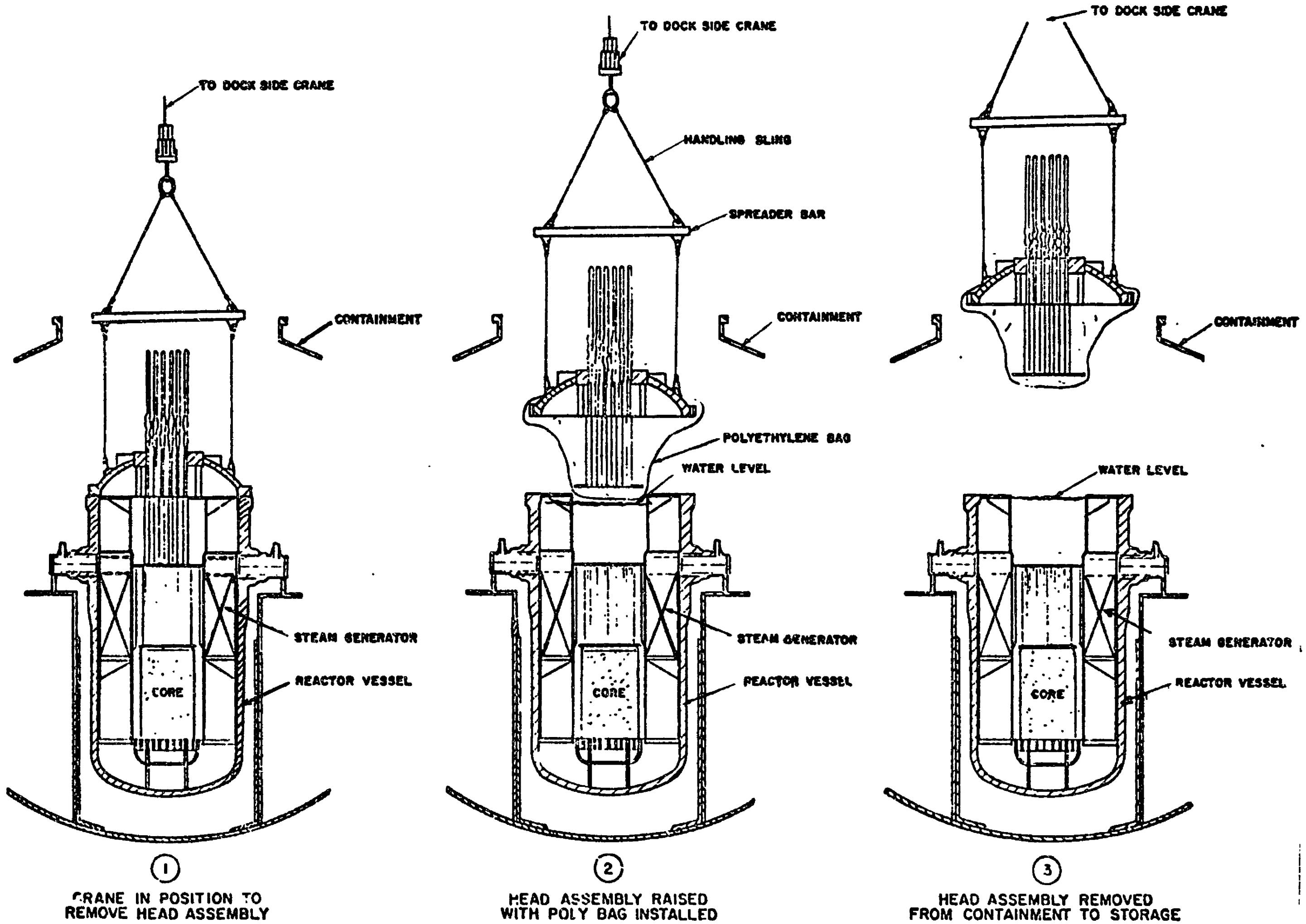


FIGURE 6.10 REACTOR VESSEL HEAD ASSEMBLY REMOVAL IN MARITIME REACTOR

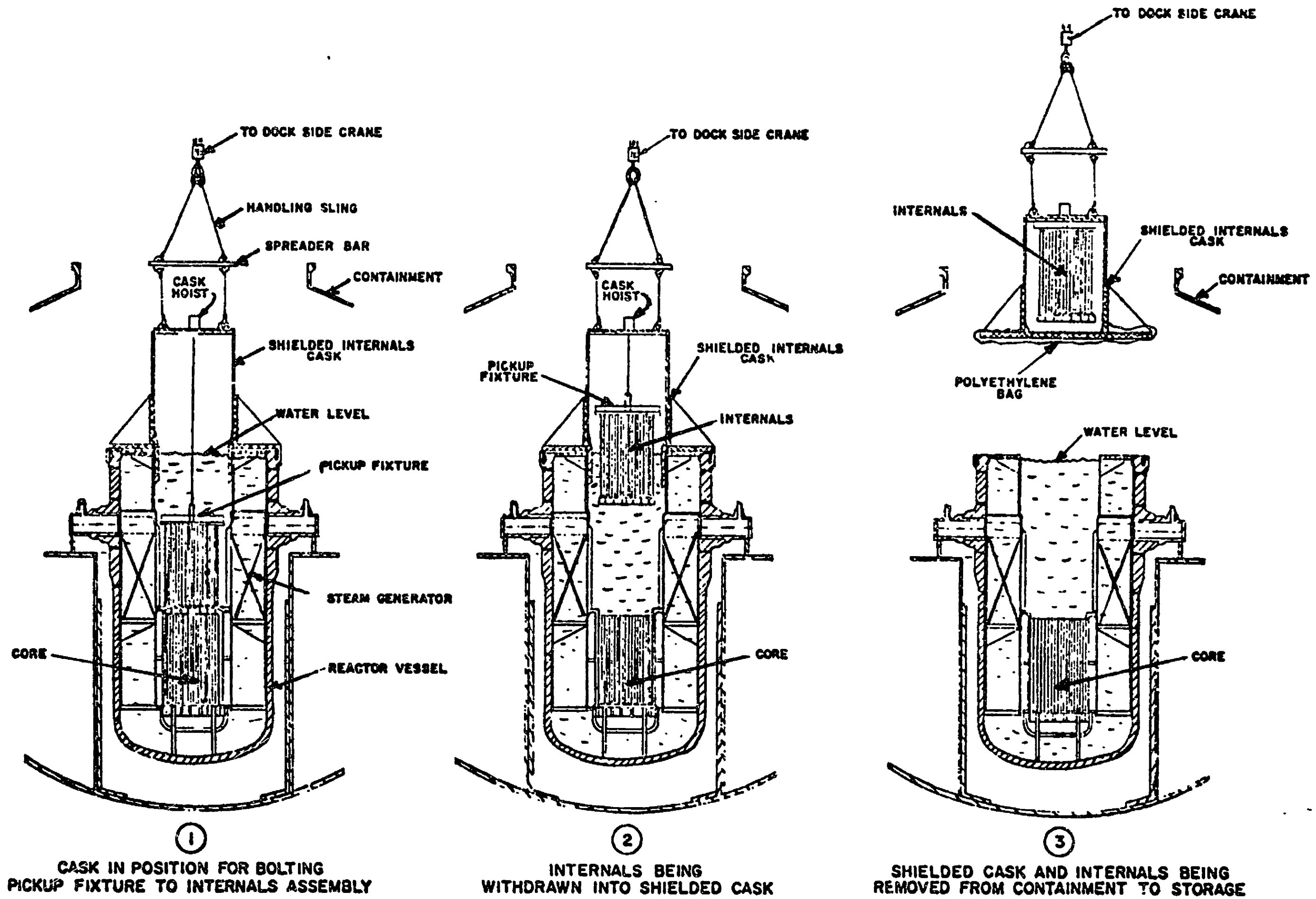


FIGURE 6.11 REACTOR INTERNALS REMOVAL IN MARITIME REACTOR

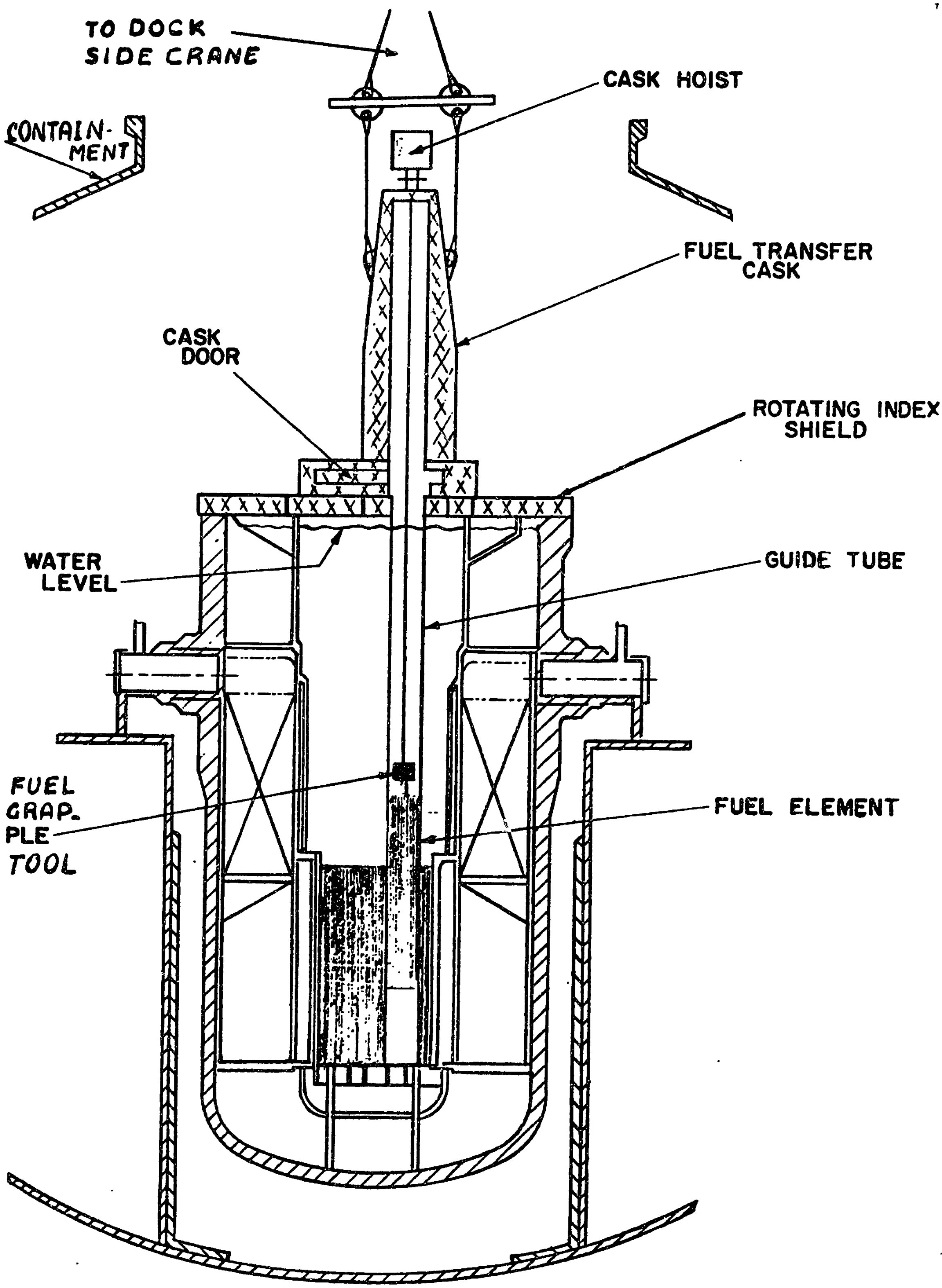


FIGURE 6.12 TRANSFER OF SPENT FUEL FROM THE CORE TO THE TRANSFER CASK FOR DRY REFUELING

component except for the access hole for fuel assembly extraction.

- Fuel Transfer Cask (see Figure 6.13) which is designed for transferring irradiated fuel elements and control rods between the reactor and the spent fuel storage pool.

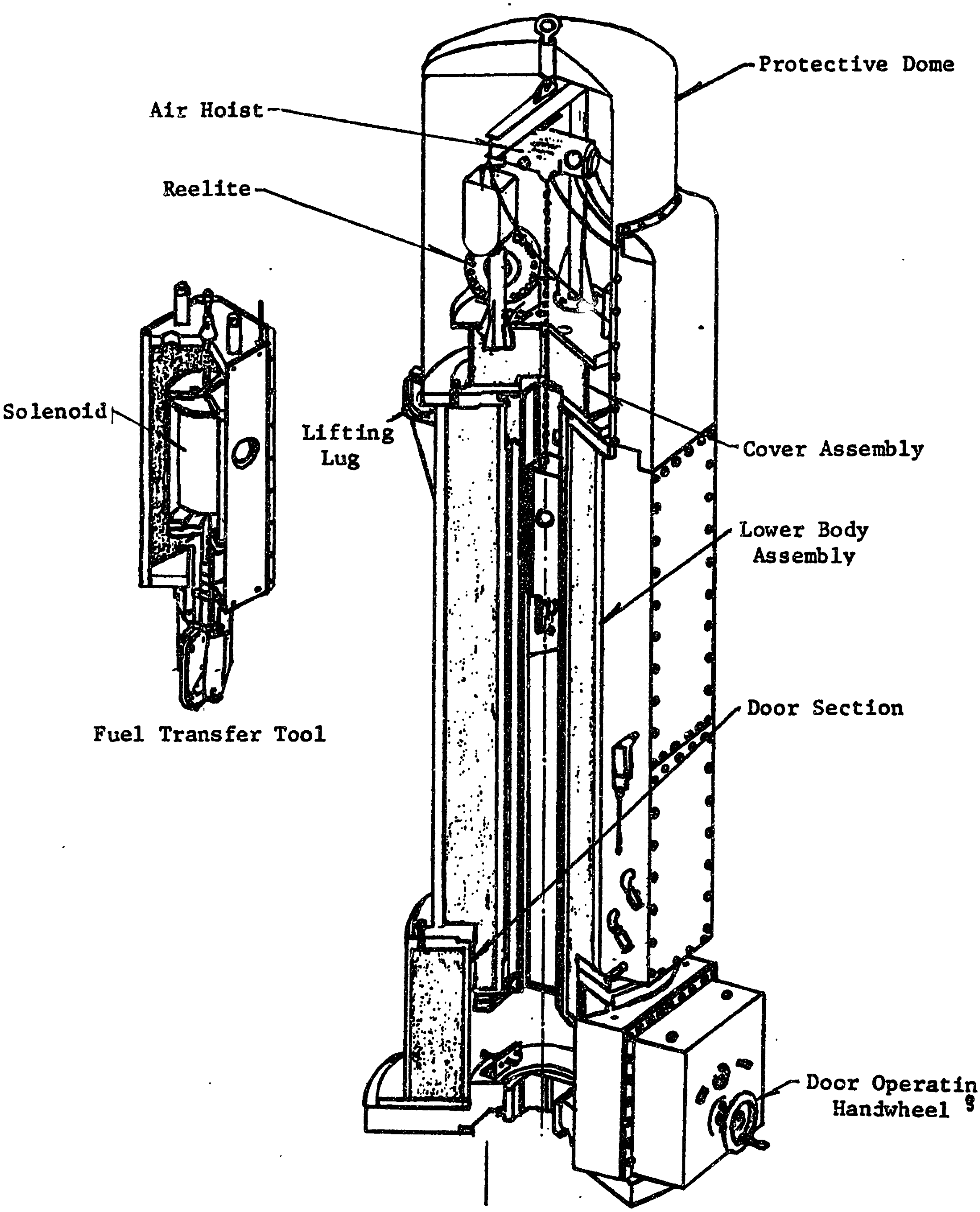
It has five basic subassemblies:

(a) Access body which is formed by three concentric steel cylinders jointed with steel end flanges. A ten-inch-thick annulus between the outer cylinders is filled with lead shielding. An annulus between the inner cylinders serves as a channel for the cooling water system. A square tube within the inner cylinder provides alignment for the pickup tool and the fuel assembly. When the spent fuel is being handled, the volume within the inner cylinder is filled with water. A shield door closes the bottom of the cask.

(b) A cover assembly which bolts to the upper flange of the cask body. It consists of a thick lead shield which also serves as a mounting platform for an air hoist that raises and lowers the pickup tools, and a spring-loaded reel for coiling the electrical cable that powers the pickup tool. A protective dome is bolted over the cover assembly.

(c) A door section which is a manually operated shield door which provides bottom shielding when the cask is in the

FIGURE 6.13 FUEL TRANSFER CASK USED IN DRY REFUELINC



lift position and closes the bottom of the cask when filled with a fuel assembly and cooling water.

(d) A pickup tool assembly which provides a positive means for grappling the fuel assembly and lifting it into the cask.

(e) A control panel which incorporates controls needed for operation of the pickup tool.

Stationary Reactor Plant

The fuel handling installation is generally divided into two areas: the refueling cavity and the spent fuel pit. The refueling cavity is totally contained inside of the reactor containment while the spent fuel pit is outside of the reactor containment. The area which presents a real difference between a stationary and maritime fuel handling installation is the refueling cavity. The refueling cavity is formed by a reactor cavity and a refueling canal.

Reactor Cavity is a reinforced concrete structure. It is filled with borated water for refueling. In that situation it forms a pool above the reactor. The cavity is filled to a depth that limits radiation at the surface of the water to acceptable levels during those brief periods when a fuel assembly is being transferred over the reactor vessel flange. The cavity is large enough to provide storage space for the reactor upper and lower internals, and for miscellaneous refueling tools.

Refueling Canal is a passageway that extends from the reactor cavity to the inside surface of the reactor containment. The canal is formed by concrete shielding walls extending upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity, thus providing the greater depth required for the fuel transfer system tipping devices and the rod cluster control changing fixture located in the canal. The refueling canal is connected to the spent fuel pit by a transfer tube.

The spent fuel pit area provides for the underwater storages of spent fuel assemblies and for control rods after their removal from the reactor. Its function doesn't differ from that of maritime reactor storage fuel installation.

The equipment for a stationary PWR refueling operation can be separated into the following two groups:

Group 1 - Reactor Vessel Handling Equipment

- Reactor Vessel Head Lifting Device which consists of structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operation.
- Reactor Internals Lifting Device which is a structural frame suspended from the overhead crane. The frame is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate.
- Manipulator Crane which is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The function is to transfer fuel

assemblies within the core and between the core and the fuel transfer system conveyor carriage.

- Vertical Mast: It consists of a tube (mast tube) which contains a long tube with a pneumatic gripper on the end. This gripper tube can be lowered from the mast to grip the fuel assembly. It is long enough so the upper end is still contained in the mast tube when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. While inside the mast tube, the fuel is transported to its new position.
- Reactor Vessel Stud Tensioners which are employed to secure the lead closure joint at every refueling.
- Guide Studs: Three guide studs are inserted into the reactor vessel flange during refueling. Their function is to guide the closure head off and onto the vessel, and to guide the internals into and out of the vessel.

Group 2: Refueling Cavity Handling Equipment

- Rod Cluster Control Fixture: This is mounted on the reactor cavity wall for removing rod cluster control element from spent fuel assemblies and for inserting them onto new assemblies. The fixture consists of two main components: a guide tube mounted to the wall for containing the rod cluster control element and a wheel mounted carriage for holding the fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the rod cluster control element and lifts it out of a fuel assembly.

- Upper Internals Storage Stand: This is a structural steel fixture used to support the upper internal package from its top flange when removed from the reactor vessel. It is installed in the refueling cavity and during refueling is underwater.
- Drive Shaft Unlatching Tool: Its function is to remove and assemble the control rod drive shafts to the rod cluster control assembly. All drive shafts are removed as a unit with the reactor vessel upper internals.
- Rod Cluster Control Thimble Plug Tool: This is a long-handled manually operated tool which is used in the refueling canal to remove and replace the thimble plug in a fuel assembly. When a rod cluster control element is transferred from one fuel assembly to another, a thimble plug is inserted, with this tool, in the fuel assembly from which the rod cluster control element was removed.

<u>Fuel Transfer System</u> (See Figure 6.14)

This system introduces an underwater conveyor car that runs on tracks extending from the refueling canal through the transfer tube in the containment wall and into the spent fuel pit. The conveyor car container accepts a fuel assembly (see Figure 6.14) in the vertical position from the manipulator crane. Then the fuel assembly is rotated to a horizontal position for passage through the fuel transfer tube, and

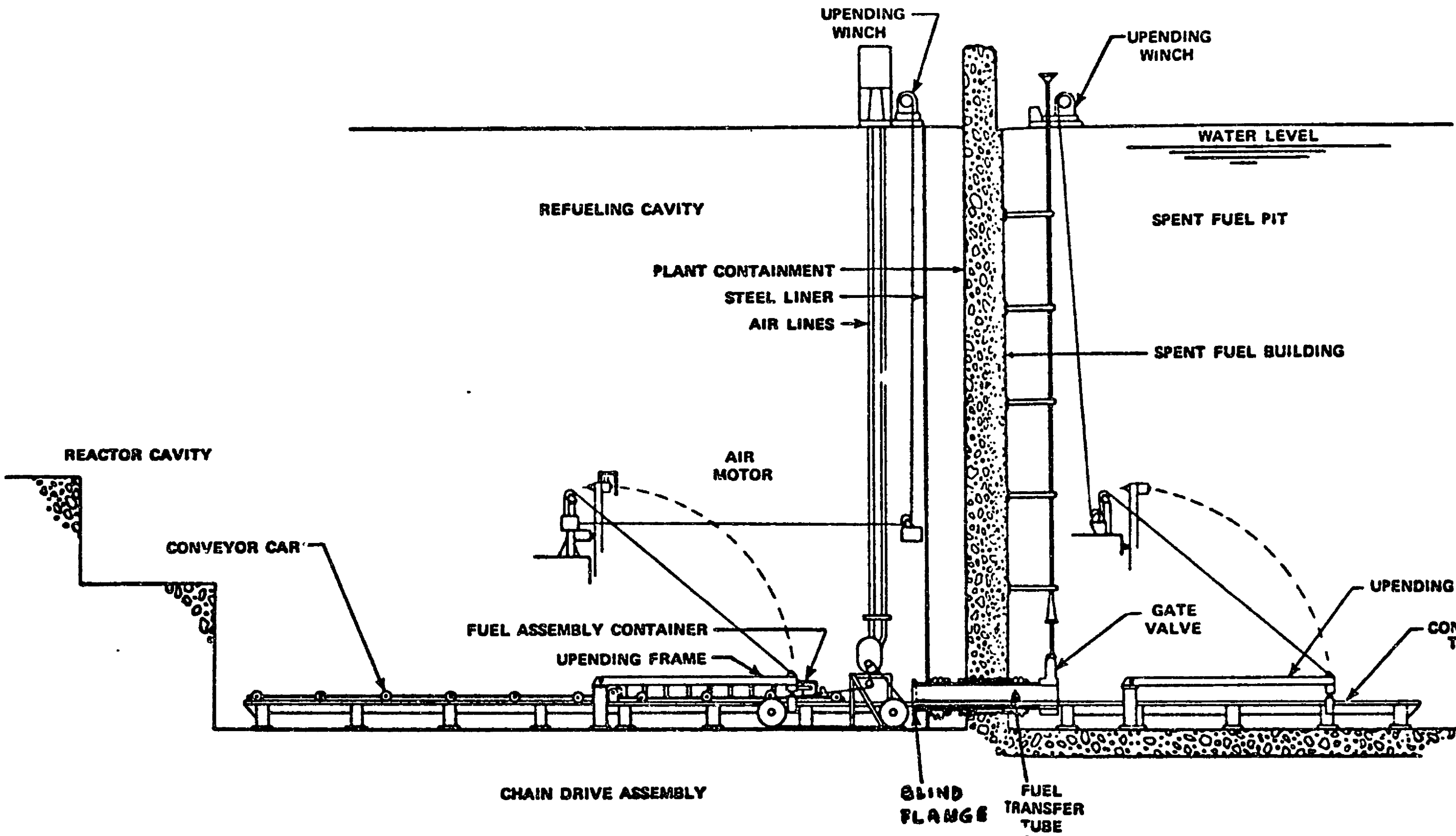


FIGURE 6.14 FUEL TRANSFER SYSTEM ARRANGEMENT IN TYPICAL STATIONARY PWR PLANT

finally rotated to a vertical position in the spent fuel pit for unloading. The conveyor car is stored in the spent fuel pit when the plant is in operation. At the end of the transfer tube, on the spent fuel pit side, there is a gate valve which, during reactor operation, is closed to seal the reactor containment. The transfer tube is sealed on the containment side by a blank flange.

6.4.3 Refueling Procedure

Maritime reactors are currently designed for operating periods of from two to five years between refueling. The time between refueling is dependent on the plant utilization and core design.

In order to compare stationary and maritime reactor refueling operation, an outline of the operations performed during a typical maritime reactor refueling and a typical stationary PWR reactor refueling will be presented.

6.4.3.1 Typical Maritime Refueling Procedure

This typical maritime refueling procedure is based on the N.S. SAVANNAH refueling technique which used a rotary index shield to position the fuel transfer cask over specific fuel assemblies and to provide shielding during fuel transfer into the cask.

The following are the various operations required in that typical refueling. These operations, per se, don't differ between stationary and maritime, but the procedure used in each one is different in some way.

Phase I - Preparation for Refueling

The reactor plant is shut down, cooled and depressurized, and the delay heat removal procedure is initiated. The containment vessel closure hatch is unbolted and transported to storage area. This procedure is not necessary in a stationary reactor since everything is made inside the containment.

Phase II - Reactor Disassembly

With all control rods already inserted (1) the rods are uncoupled from their drive mechanisms, (2) the control extensions are withdrawn to the full-out position, and (3) the control rod drive mechanism cables and cooling air ducts are removed and stored.

The water level in the reactor vessel is lowered below the vessel flange. Vessel head studs are removed and stored. The head is jacked up and a poly-ethylene bag is attached to the bottom. The head is then lifted from the containment vessel and transferred to a storage area (see Figure 6.10). Then, the reactor internals are removed (see Figure 6.11) in the following sequences:

- A reactor vessel flange protective ring is installed.
- A cask for handling the internals assembly is lowered into position at the protective ring.
- A pickup fixture is attached to the internals assembly.
- The internal assembly is raised by means of the pickup fixture until it is positioned within the cask.

- The loaded cask is lifted from the containment to the main deck where the bottom is closed and then transported to a storage area.

Phase III - Fuel Handling

The refueling sequences are started with a rotating index shield installed on the protective ring in preparation for fuel transfer. The index shield is positioned over the first fuel assembly and control rod to be unloaded (see Figure 6.12). The fuel transfer cask is then positioned and aligned on the rotating index shield over the open port.

A grapple tool is lowered from the cask to engage and connect with the fuel assembly. The fuel assembly is lifted into the cask until the assembly is in the "fuel-up" position.

The cask bottom door is closed, the cooling water lines are connected, and the cask is filled with water. The loaded cask is transferred to a dockside storage pool, the cask door is opened and the fuel assembly is lowered into a storage rack. The control rods removed, are used again in the new fuel.

Change-over of the control rod is accomplished underwater in the storage pool. The rod is first removed from the spent fuel assembly and then introduced into the new assembly. The transfer cask is positioned over a new fuel assembly. The new fuel assembly along with control rod is withdrawn into the cask.

The loaded cask is carried back to the ship, installed on the rotating index shield over the vacant core position and the fuel assembly is introduced into the core. This process is repeated until all spent fuel assemblies have been replaced with new fuel.

This process is for only one-batch, but for a two or more batch core the difference is that only one-batch is removed and replaced; the others are removed and placed in a different core location in accordance with the fuel management schedule. At the completion of the fuel transfer, the rotating index shield and transfer cask are removed and stored onshore.

Phase IV - Reactor Reassembly

This operation is the inverse of Phases I and II.

Phase V - Preoperational Checks, Tests & Startup

Pre-critical checks and core physics tests are conducted, and the reactor is brought critical.

6.4.3.2 Typical Stationary Refueling Procedure

This typical stationary refueling procedure is based on refueling technique of the Westinghouse PWR. As in maritime the refueling operation can be divided into five major phases: Phase I - Preparation for Refueling; Phase II - Reactor Disassembly; Phase III - Fuel Handling; Phase IV - Reactor Assembly; Phase V - Preoperational Checks, Tests and Startup.

Phases I, IV and V differ little from the maritime phases. For that reason only Phases II and III will be described here. In these phases procedures which differ from the same maritime refueling phases will be stressed.

Phase II - Reactor Disassembly

The reactor disassembly is conducted in the following steps:

1. The control rod drive mechanism cables and cooling air ducts are disconnected and moved to storage.
2. The reactor vessel head insulation is removed.
3. Upper instrumentation thermocouple leads are disconnected.
4. The incore instrumentation thimble guides are disconnected at the seal table (see Figure 6.15) and extracted upward.
5. The reactor-vessel-to-cavity seal ring is bolted down.
6. The reactor vessel head nuts are loosened using a stud tensioner and then both stud and nuts are removed and stored.
7. The reactor vessel head is unseated and raised by the plant crane.
8. The reactor cavity is filled with borated water to the vessel flange.
9. The head is slowly lifted while water is pumped into the cavity. The water level and vessel head are raised simultaneously, keeping the water level just below the head. Step 8 and 9 are different from that of the maritime procedure.
10. The reactor vessel head is removed to a dry storage area.

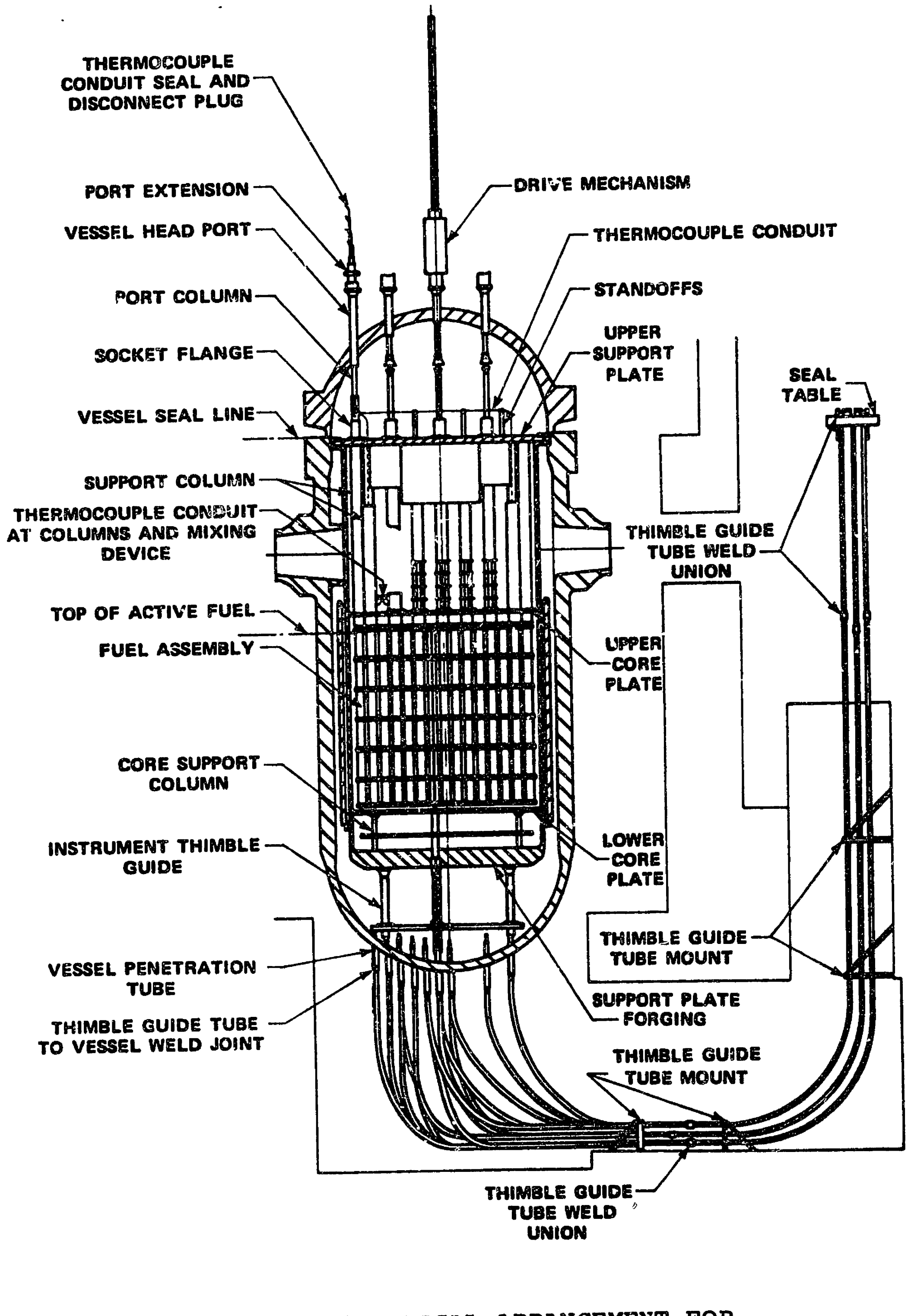


FIGURE 6.15 TYPICAL ARRANGEMENT FOR IN-CORE INSTRUMENTATION IN STATIONARY PWR PLANT

11. The control rod drive shafts are unlatched, using the drive shaft unlatching tool. The control rod drive shafts remain with the reactor vessel upper internals.

12. The reactor internals lifting rig is lowered into position by the plant crane. After the support plate of the upper internals and control rod clusters drive shafts are secured by the rig, they are lifted out of the vessel and stored in the underwater storage stand in the refueling cavity. For this procedure a cask is not used for handling the internals such as is used in the maritime reactor.

Phase III - Fuel Handling

In this phase the following steps are used.

Reactor Vessel Fuel Discharging

The crane is positioned over a fuel assembly in the most depleted region of the core. The fuel assembly is lifted to a predetermined height sufficient to clear the reactor vessel and still leave sufficient water-covering to eliminate any radiation hazard to the operating personnel. This is possible because of the type of installation and represents a significant differences from the maritime procedure.

Rod Cluster Control Removing

In case the removed assembly contains a rod cluster control, the assembly is placed in the rod control cluster changing fixture. The rod control cluster is removed from the spent fuel assembly and placed in a new fuel assembly or in a transferred spent fuel assembly.

Spent Fuel Transporting From The Reactor Vessel To Storage Plant

The mainipulator crane is moved to line up the spent fuel assembly with the fuel transfer carriage. The fuel assembly is inserted into the fuel assembly container of the carriage, which previously was tipped upright by the upending frame to permit that operation.

The container is placed horizontally by the upending frame and the carriage is moved through the fuel transfer tube to the spent fuel storage pit. The fuel assembly container is tipped upright and the fuel assembly is unloaded by the spent fuel handling tool attached to the spent fuel pit crane. The spent fuel assembly is placed in the spent fuel storage rack.

Core Fuel Recharging

Once the fuel assembly is discharged in the spent fuel storage pit, the conveyor car is moved back into the refueling canal. The new fuel assembly is brought from dry storage, lowered into the fuel transfer canal and loaded into the conveyor car. Partially spent fuel assemblies are moved from one region to another region of the reactor core. Any new fuel assembly or transferred fuel assembly that will be placed in a control position must be first placed in the control rod cluster changing fixture to receive a control rod cluster from a spent fuel assembly (step 2). The new fuel assemblies are loaded into the vacant region of the core.

Because all fuel handling procedures in the stationary reactor are made entirely underwater, the refueling operation is named a wet-type removal.

6.4.4 Refueling Time

Refueling periods, while occurring infrequently over a ship's life, will constitute possible non-revenue producing time. In addition, most of the direct costs involved are proportional to refueling outage time. It follows that outage time for reactor refueling must be held to a minimum to achieve the most favorable ship economics.

The technological differences between stationary and maritime reactors, derived from the maritime economic problem of volume and weight, represent an important increase of maritime refueling time compared with the stationary refueling time.

Nuclear reactor plants out of service for refueling operation, either maritime or stationary, depend upon:

- The amount of preparation for the refueling
- The amount of expertise refueling personnel gained through previous experience or training
- The number of problems encountered during the refueling
- How well the reactor is designed for easy refueling
- The number of fuel assemblies in the core

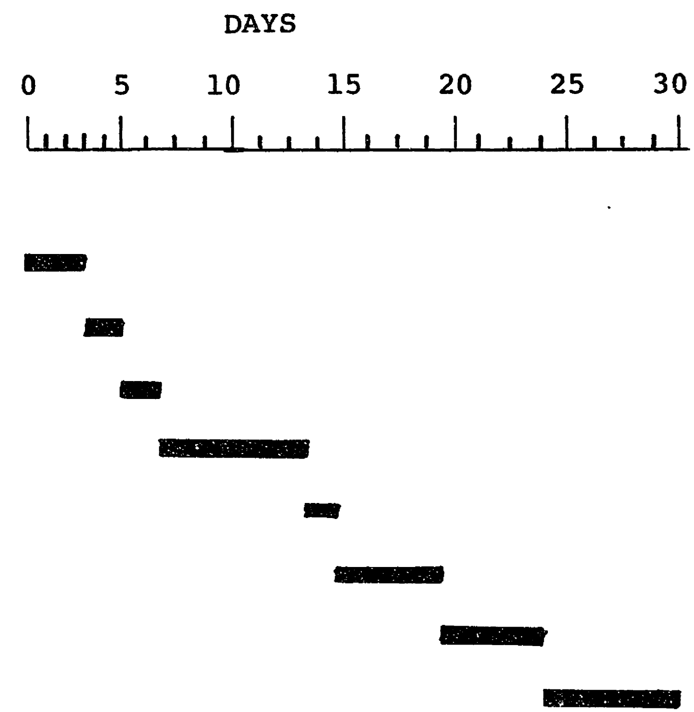
From all these points and the case under study (Technological Differences between Stationary and Maritime Reactor Application), the most important point is "how well the reactor is designed for easy refueling".

Fig. 6.16 presents a projected schedule of maritime reactor refueling operation. This projected schedule was developed by U.S. Maritime Administration. The schedule is based on a three-shift, seven day week, with no allowance made for inspections, equipment breakdown or any unforeseen problems which could disrupt a continuous operation. The fuel transfer time is based on a one-batch complete fuel replacement refueling.

Figure 6.17 shows a typical schedule of a stationary PWR refueling operation. Comparing these two figures 6.16 & 6.17 it can be seen:

1. While to remove and to storage the reactor vessel head on ship requires 48 hours, in stationary plant, this operation only takes eight hours.
2. Internal removal on reactor ship take 30 hrs , again four hours for stationary reactor.
3. Ship fuel transfer operation take 149 hours against 100 hrs for stationary reactor. Besides, we must point up the differences of fuel assemblies to be removed in one and another case. Thus, while in this stationary type of reactor, there are 193 fuel assemblies of which 64 are discharged as spent fuel and the rest are changed in position; in the maritime

FIGURE 6.16 MARITIME REACTOR REFUELING SCHEDULE ESTIMATES



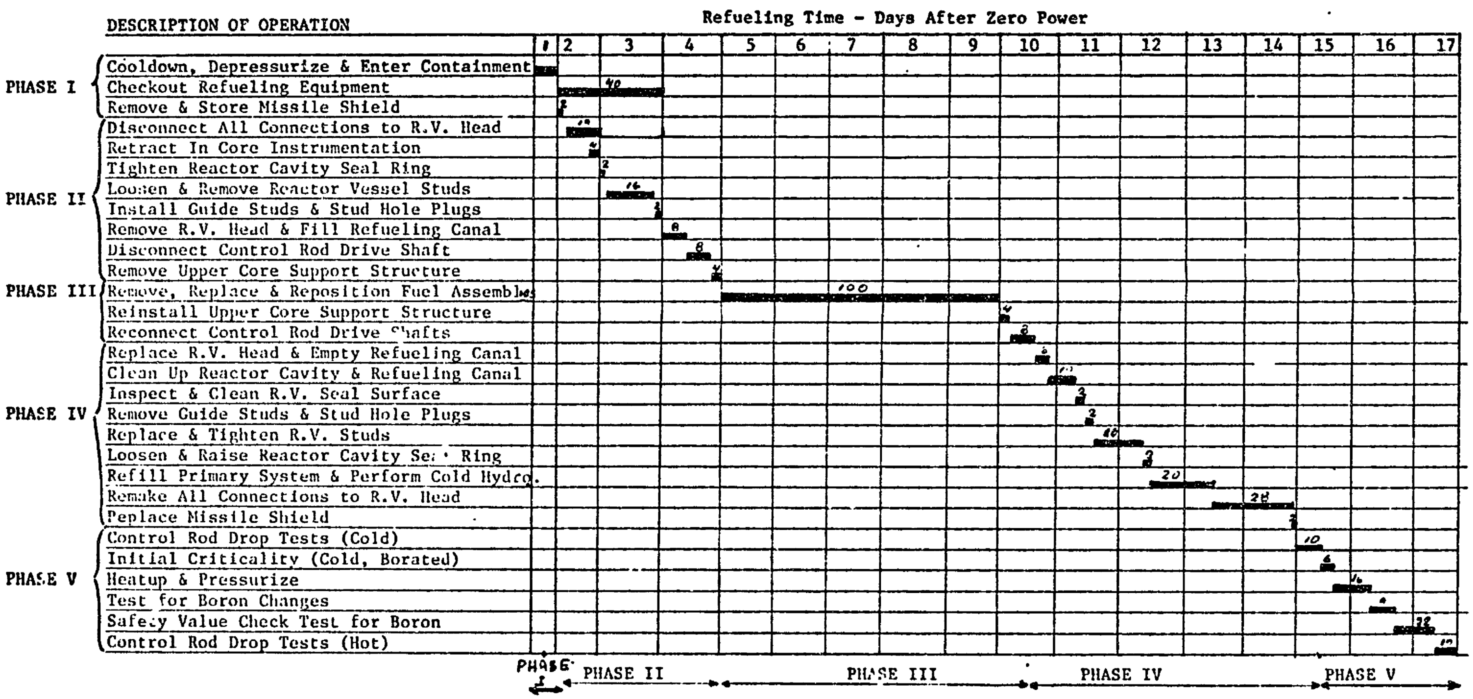
PHASE

1. Preparation (60 Hr)
2. Vessel Head Removal (48 Hr)
3. Internals Removal (30 Hr)
4. Fuel Transfer (149 Hr)
5. Internals Replacement (24 Hr)
6. Vessel Head Replacement (90 Hr)
7. Preparation for Startup (120 Hr)
8. Startup, Tests, Heatup (144 Hr)

Total Hours - 665

Total Days - 27.7

FIGURE 6.17 TYPICAL SCHEDULE OF STATIONARY PWR REFUELING



reactor there are a total of only 32 fuel assemblies which are replaced completely.

4. Internals replacements takes 24 hours in maritime reactors and 12 hours in stationary reactors.
5. Vessel head replacement takes 90 hours in maritime reactors and only 36 hours in stationary reactors.

One of the main causes of these differences in refueling time operation is due to the differences in the refueling installations and refueling procedures which cause, in maritime refueling, loss of time in the transportation of parts to and from the storage area. These differences in installations and procedures, as was said before, are based almost exclusively in the ship economic problem of reducing, as much as possible, volume and weight.

Also, it is possible to assert that although the 27.7 days of maritime reactor refueling period may be reduced by improvements in refueling equipment and installations design, by use of new refueling techniques and utilizing experienced personnel for all refueling operations; the maritime refueling period could never be reduced to the stationary refueling period given the limitations imposed on the refueling installations and equipment design by the ship volume and weight conditions.

6.5 Radiation Protection

6.5.1 Choice of Shielding Materials

Apart from shielding properties there are four factors governing the selection of materials for marine reactor shielding: weight, space, engineering and cost. A heavy shield, besides generally reducing the ship's carrying capacity, always constitutes a support problem.

The space taken up by a shield arrangement, as measured by the size of the reactor compartment to contain it, will be at the direct expense of cargo capacity. However in any case the penalty is difficult to assess. Thus, taking for example the design of the tanker, if a larger reactor compartment is to be fitted, a more economical ship will result if the hull is redesigned to have the original cargo capacity as well as the larger reactor compartment, rather than use the original hull and reduce the cargo capacity to make way for the reactor compartment. In other types of ships, carriers for example, where the cargo limitation is one of weight and not volume, the size of the shield is of little importance.

6.5.2 Comparison of Design Bases for Different Reactor Shieldings

To provide an idea of the greater importance of the weight of the shielding in the ship reactor design than in stationary reactor, the design bases for a ship reactor (CNSG), a Platform Mounted Nuclear Plant (PMNP) and a PWR for stationary plant are described.

6.5.2.1 CNSG Shielding Design Bases

The major design objectives are as follows:

1 - Normal Operation and Anticipated Operation Occurrences

To ensure that radiation dosages to crew members and to the general public are within the exposure limits set forth in 10 CFR 20 and 10 CFR 50 and that they are as low as practicable.

2 - Emergency Conditions

To ensure that crew members are adequately protected and to preclude undue hazards to the general public.

3 - Shield Optimization

To provide an efficient design that will afford maximum protection with minimum shield weight.

6.5.2.2. Platform-Mounted Nuclear Plant

PMNP - Shielding Design Bases

Shielding is provided to perform the following functions:

1. Assure that the radiation dose to all personnel on the platform as well as the general public is within the limits set forth in 10 CFR 20 during normal plant operation.
2. Provide a shielded living environment for all platform personnel, following the Design Basis Accident, within 10 CFR 20 limits.
2. Assure that "contained radiation sources" following the Design Basis Accident do not result in direct doses off site in excess of limits specified in 10 CFR 100.

6.5.2.3 Stationary Plant

Shielding Design Bases of Diablo Canyon Nuclear Plant

Shielding is provided for normal operation, maintenance, accident conditions and refueling. Maximum radiation doses for plant personnel are limited to those specified by AEC regulations.

All radiation and high radiation areas must be appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

To accomplish the foregoing, shielding must be arranged to protect personnel against direct gamma radiation streaming through the piping penetrations as well as against source of radioactivity with equipment and piping.

The shielding design is based on conservative source estimates and coordinated with the piping layout and valve locations. Reach rods must be provided where necessary to permit the operator to remain behind the shielding while operating valves.

Radiation shielding must be designed for operation at maximum calculated thermal power and to limit the normal operation radiation levels at the site boundary to below those level allowed for continuous nonoccupational exposure. The plant must be capable of continued safe operation with 1% fuel element defects.

The shielding provided assures that in the event of a hypothetical accident, the integrated off-site exposure due to the contained activity would not result in any harmful off-site radiation exposures.

The radiation shielding is further designed to permit continued operation of other units on this site in the unlikely event that a unit experiences the loss-of-coolant accident.

Shielding Design Based on Stationary Pilgrim Nuclear Power Station

Radiation Exposure of Individuals

The basis for the radiation shielding design for normal operation is 10 CFR 20. For the design basis accident, the station design is based on the guideline values of 10 CFR 100.

Radiation Exposure of Materials & Components

Materials and components are selected on the basis that radiation exposure as a result of the shielding design will not cause significant changes in their physical properties which adversely affect operation of equipment during their design life.

Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a design basis accident.

The following general radiation exposure limits were considered in the selection of materials.

Material	Approximate Damage Threshold
Teflon	$\geq 1.0 \times 10^{4}$ rads
Most thermoplastic & elastomers	1.0×10^{6} rads
Some thermoplastics	1.0×10^{7} rads
Ceramics	1.0×10^{10} rads
Metals	1.0×10^{11} rads

6.5.2.4 Conclusion

Thus, comparing those different shielding bases design, we can see:

1. That one of the major ship reactor shielding design objectives - the shield optimization to provide an efficient design that affords maximum protection with minimum shield weight - is not shared by any other reactor plant.
2. The stationary plant (Diablo Canyon Nuclear Plant) foresees shielding for refueling, but in maritime, due to a matter of weight, the shielding is furnished under separate cover.
3. In stationary plant (Pilgrim Nuclear Power Plant) the material is chosen for its resistance to radiation damage and not for its weight.

6.5.3 Maritime Reactors Sheidling Designs

In CNSG, primary shielding within the containment plant vessel and secondary shielding outside the containment vessel must be designed to provide optimum shielding with minimum weight. To reach this objective, process equipment, interconnecting pipes and waste collection tanks must be locally shielded so that the design dose rate in each zone is not exceeded. The reactor compartment equipment must be arranged so that local shielding requirements are minimal and thus reduce total shield weight.

Whenever practicable, equipment operation in high radiation levels must be avoided by using remote control devices. Occupancy of areas such as the reactor compartment must be minimal; such areas must be occupied primarily for required inspections. Shielding requirements for refueling must be accomplished utilizing portable shielding.

During nuclear power plant operation, the crew must not enter the containment vessel. Reactor personnel may enter the containment only under Health Physics supervision after the reactor is shutdown and the primary system is cooled.

Areas of the ship that may require a background radiation level that is lower than the zone design dose rate, must be provided with portable shielding as required. The shielding material and the thickness of shielding are determined by the operational design dose rates specified for all areas of the ship as well as the weight of the materials used. Following these general criteria the shielding design can present some differences depending upon the type of the reactor used (loop or integral) and the type of containment, etc.

Thus, the CNSG foresees in its shielding design two shields, primary and secondary shielding. The primary shielding includes all materials within the containment vessel as well as the associated vapor suppression tanks and adjacent water pool. Shielding above the vapor suppression pool is provided to reduce operating neutron and gamma radiation.

To block the neutron streaming path between the pressure vessel and the containment vessel wall, an annular shield is provided at the top of the vapor suppression pool.

The upper primary shielding also reduces intermediate energy neutron streaming in the pressure vessel carbon steel wall.

The secondary shielding is provided by the containment vessel bulkhead support structures and additional lead, polyethylene, steel or concrete is placed outside the containment vessel.

In the CNSG the basic secondary shield consists of a four foot thick vertical concrete cylinder with a two-foot-thick cap. The auxiliary and process equipment is contained in segmented areas of the reactor compartment with approximately two-foot-thick concrete wall.

In the Italian project of nuclear ship propulsion "Enrico Fermi", a PWR loop-type reactor (see Figure 6.18), the radial shield consists of heavy metal and water cylindrical shells and is divided in three main components:

- Thermal Shield (Iron/Water): has the main scope of reducing the radiation damage of the pressure vessel steel. In Table 6.2 the alternate iron/water layers inside the pressure vessel are summarized. The configurations represent the optimum arrangement iron/water to reduce the radiation damage of the vessel to a minimum value with minimum weight.

- Primary Shield (Iron/Water)(So-called Neutron Shield Tank): surrounds the pressure vessel and attenuates the radiations coming out from the same. It consists of a carbon steel walled vertical cylindrical tank, with annular horizontal cross section, containing concentrical alternate layers of iron base material and water. This combination makes a shield for both

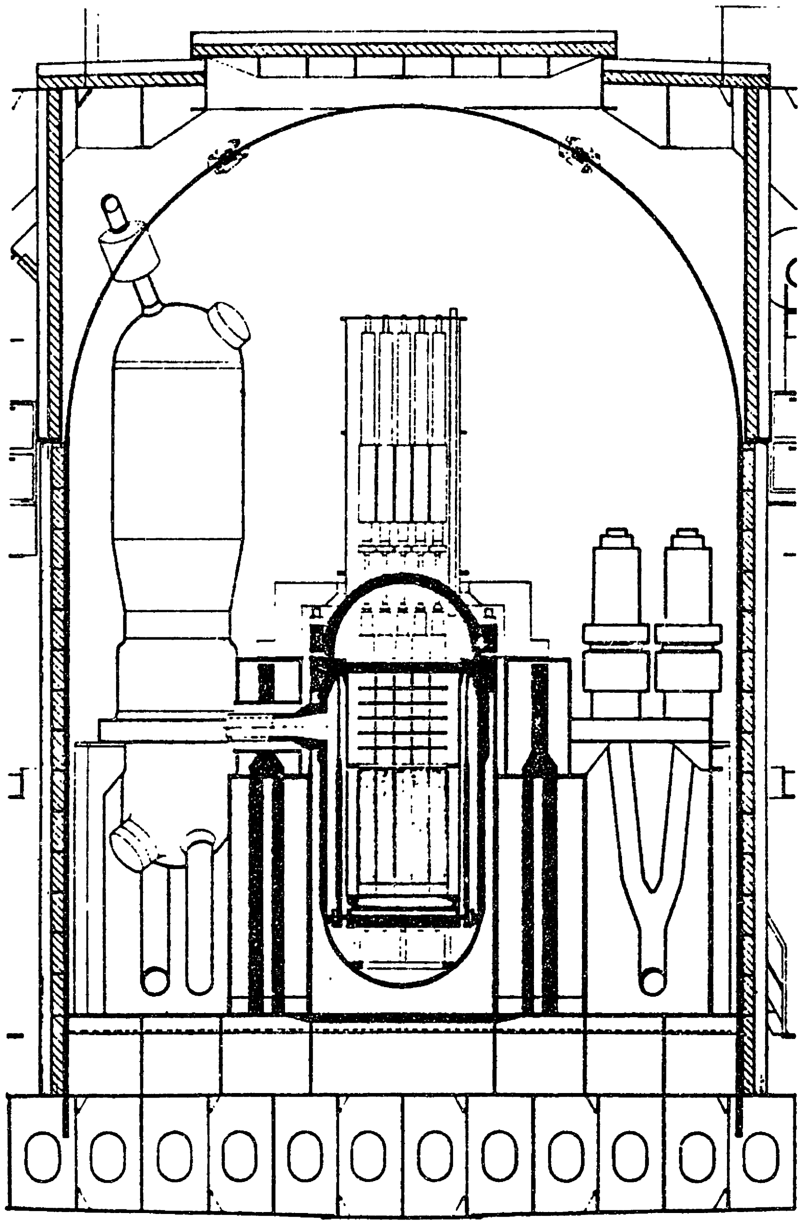


FIGURE 6.18 CORE, COOLANT LOOP & SHIELD OF A PWR REACTOR FOR THE ITALIAN SHIP PROPULSION PROJECT

TABLE 6.2 THERMAL SHIELD COMPOSITION (THICKNESS) OF ITALIAN NUCLEAR SHIP PROJECT

Stainless Steel (Baffle)	12,7 mm
Water	Variable (from 9,3 mm to 183,3 mm)
Stainless Steel	30 mm
Water	35 mm
Stainless Steel	80 mm
Water	102,5 mm

neutrons and gamma rays. Figure 6.18 shows the neutron shield tank and Table 6.3 summarizes the thickness of the internal iron and water layers.

It is necessary to point out here that from the search for the optimum primary shield configuration it results that the optimum primary shield, with regard to the weight, should be composed by a lamination of heavy metal located about the middle of the water tank. The present configuration gave more weight to the important mechanical support fuction of the neutron shield than to its shield function. The search made by the Italian Group to find the primary shield optimization consisted in evaluating the weight of various possible shield designs giving the same dose at specified points.

With regard to the requirements of shield construction and plant operation, twelve configurations of the primary shield were selected. For every one of these configurations, the necessary thickness of the secondary shield was computed in order to obtain the wanted value for the total dose and then the total weight was calculated. Figure 6.19 and Table 6.4 show some results of the former search. In Figure 6.19 can be seen the weight variation (ΔW), for the optimum configuration, as a function of the ratio of neutron radiation to total dose (D_n/D_n+D_γ). It can be seen that the minimum weight is found for $D_n/D_n+D_\gamma = 0.1$.

Table 6.4 reports a comparison among the various configurations for a particular value of the dose rate and for

TABLE 6.3 NEUTRON SHIELD TANK COMPOSITION (THICKNESS) OF ITALIAN NUCLEAR SHIP PROJECT

Lower Part	Water	320 mm
	Carbon Steel	150 mm
	Water	100 mm
	Carbon Steel	190 mm
	Water	315 mm
Upper Part	Water	310 mm
	Carbon Steel	200 mm
	Water	245 mm

FIGURE 6.19 ITALIAN NUCLEAR SHIP PROJECT. WEIGHT VARIATION FOR THE PRIMARY SHIELD CONFIGURATION AS A FUNCTION OF THE RATIO OF NEUTRON TO TOTAL DOSE

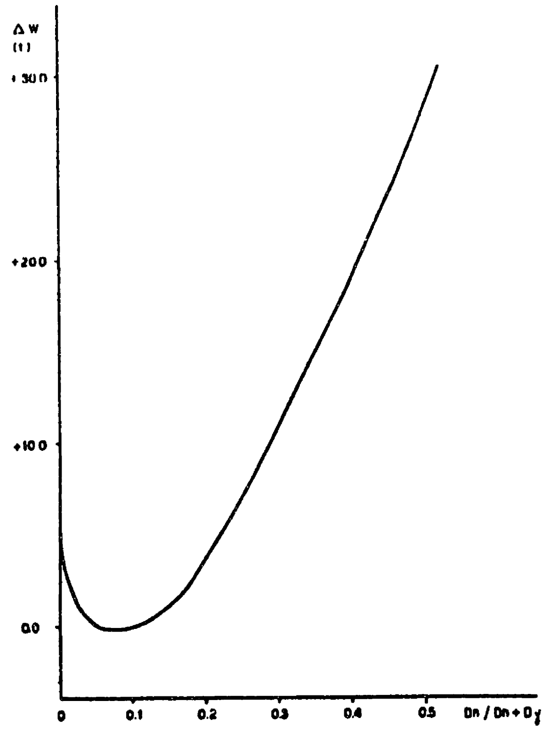


TABLE 6.4 LEAD & POLYETHYLENE THICKNESS OF THE SECONDARY SHIELD IN THE VARIOUS SOLUTIONS & WEIGHTS COMPARED TO ONES OF SOLUTION NUMBER 12 FOR A CONSTANT DOSE $D_n+D_\gamma=0.78$ MREM/H & $D_n/D_n+D_\gamma=0.1$

Italian Nuclear-Ship Project

Solution (Number)	t (cm) Lead	t (cm) Poly	W (Tons)
1	24.3	40.0	+ 184.5
2	24.5	40.0	+ 193.7
3	17.1	41.6	+ 31.2
4	20.2	49.0	+ 113
5	17.0	49.8	+ 54.6
6	20.6	34.0	+ 101.1
7	18.4	32.8	+ 64.9
8	18.1	31.8	+ 57.4
9	16.9	25.5	+ 11
10	18.0	15.8	+ 37.7
11	14.5	25.4	- 33.9
12	15.1	13.7	0

$D_n/D_n+D_\gamma = 0.1$. Thus, the weight variation ΔW and the lead and polyethylene thickness (t_{pb} and t_{poly}) for the secondary shield with respect to the optimum configuration (Nº12) is given. Comparisons performed for other ratios (D_n/D_n+D_γ) and for various dose values gave similar results to those of Table 6.4.

- Secondary Shield (Lead/Polyethylene): surrounds the containment vessel and attenuates, to the allowed dose limits, the radiation coming from the coolant loops (mainly gamma ray arising from activation) and escaping from the primary shield.

In this case the solution of minimum weight is one with the shield modulated in thickness so that on its outer surface the allowed dose level is constant.

This solution is not an easy one since the dose that reaches the different points of the primary shield is not constant. The dose is not constant due to: the configuration of the shield, the different types of attenuation to which the radiation is submitted along its way, and the different points and sources of radiation. This means that finding the minimum weight shielding required for the Maritime Shielding Reactor is a very difficult task. For a complete study, it is necessary to emphasize the interrelations between the shielding design (minimum shield weight for the allowed dose level) and the general mechanical design of the reactor primary and auxiliary systems.

The Italian research obtained the thickness which gave a constant dose on the shield by making a dose thickness plot for a map of points situated on the outer surface of the shield. Therefore, a dose calculation by the code QAD was performed on an array of twelve azimuthal points and eight axial points with lead thickness of the secondary shield from eleven cm to eighteen cm. On the other hand, they arrived at the result that the shield with constant thickness, which allows the same dose limits, is sixteen cm lead thick and that its weight is greater than the above shield by 5%.

6.5.4 A Cost Comparison of Various Shield Materials in a Marine Reactor

The following comparison of costs of various shield materials was carried out by Brown and Jackson (17), although both the cost and some of the designs considered are out of date, they do give a very good idea of the importance of the weight and the shielding total cost in the selection of the shielding material.

In that case an attempt was made to compare the cost of using a range of materials to shield a reactor system to the same radiation levels. Sufficient major design parameters had been fixed at that time to enable a reasonably realistic cost estimate to be made.

The basic design was for a 65,000 ton oil tanker, with a reactor plant capable of about 70 MWt output. A typical reactor vessel was assumed incorporating an internal thermal shield, and the primary shield was placed immediately outside it and designed to give a radiation level of 2.7 rem/h at the outer surface. The secondary shield , outside the containment vessel, was designed to give an attenuation factor of 100. This arrangement was adopted to give a radiation level of 1.5 rem/yr at the far side of the controlled area boundary at a distance of 20 ft. from the secondary shield. To date this dose rate is considered, as acceptable for a zone of continuous occupancy in the PSAR of the CNSG Iv (8).

An arbitrary 2000 curies of fission products from faulty fuel elements was assumed to be uniformly dispersed in the coolant inside the reactor. Since no account was taken of any radiation from heat exchangers, or holes through the main shields, it is possible to make a comparison with an integral reactor type.

Twenty arrangements were assessed, made up of the combination of four primary shield types with five secondary shields. The following designed parameters were used:

Core Flux	10^{13} cm^{-2} sec^{-1} at reflector boundary
Reflector	equivalent to 6 in. of water
Thermal Shield	3 in. of steel
Reactor Vessel	Diam. 7 ft. 6 in. I.D.
	Height 24 ft.
	Thickness 4 1/2 in

Containment Vessel Type: Vertical Cylinder

Containment Vessel Diameter - Varies with primary shield thickness

Containment Vessel Height 48 ft.

Containment Vessel Thickness: About 1.5 in.

The four primary shields studied were made of readily obtainable materials and contained elements of both high and low mass number. They are listed below:

1. Lead & Water - lead plates within water layer.
2. Concrete - homogeneous shield self-supporting.
3. Steel & Water - steel plates within water layer.
4. Steel, Boron/Steel & Water - steel plates within water layer, and a boron/steel plate being the penultimate layer in the shield.

The disposition of the various metal layers was chosen to give the best shielding results. In all cases the thermal shield and pressure vessel wall were taken into account as additional shielding.

The five secondary shields attenuate a radiation that was predominantly gamma rays and included some neutrons. They were:

a. Reinforced Concrete (Cylindrical Tank Type) - Arranged as a cylindrical tank around the containment vessel.
b. Reinforced Concrete (Cofferdam Type) - Situated at the reactor compartment boundary bulkheads.
c. Steel & Water - The steel was utilized to form a tank to contain the water as secondary shield situated at the reactor compartment boundary bulkheads.
d. Lead & Polyethylene - An inner lead layer attached to the containment, covered by the polyethylene.

e. Steel & Concrete - The steel placed as one layer at the innermost boundary of the shield covered by reinforced concrete.

For each arrangement, the cost and weight of the following items have been calculated: Primary Shield, Containment Vessel, Secondary Shield and Support Structures. A capitalized cost for loss of cargo was also included. It was assumed that the hull should remain unaltered and that the cargo capacity should change with the variation in reactor room length, but was uneffected by the shielding weight. The loss was calculated relative to the smallest arrangement considered, which was assumed to have zero loss. A capital charge of £150/tn was taken, based on voyages between the United Kingdom and the Persian Gulf via Suez.

Table 6.5 compares the relative weights and cost of each arrangement This table gives us a clear comparison of the total costs and weight of different shielding arrangements.

The figures for the lead and water and the steel and water primary shields illustrate the influence that shield size has on total cost. Here, it can be seen how the increase in the loss of cargo charge introduces the largest increase in the total.

The high cost of the boron steel primary shield, however, outweighs the comparatively modest cost of loss of cargo associated with it.

TABLE 6.5 COMPARISON OF WEIGHTS & COSTS OF SHIELD ARRANGEMENTS

PRIMARY SHIELDING		SECONDARY SHIELDING									
		Concrete (Cylindrical tank type)		Concrete (Cofferdam type)		Steel & Water		Lead & Polythene		Steel & Water	
	Item	Weight(t)	Cost( )	Weight (t)	Cost ( )	Weight (t)	Cost ( )	Weight (t)	Cost ( )	Weight (t)	Cost ( )
Lead & Water	Primary	325	70340	325	70340	325	70340	325	70340	325	70340
	Secondary	964	11870	1930	18610	1364	50580	884	312930	2022	264630
	Containment	164	71340	164	71340	164	71340	164	71340	164	71340
	Support	37	7350	9	1750	9	1750	12	2420	9	1750
	Loss of Cargo	--	30000	---	120000	---	150000	--	0	--	90000
	Total	1490	190900	2428	282040	1862	344010	1385	457030	2520	498060
Concrete	Primary	382	7040	382	7040	382	7040	382	7040	382	7040
	Secondary	1103	13630	2038	19750	1445	54720	1021	361900	2132	278850
	Containment	208	88400	208	88400	208	88400	208	88400	208	88400
	Support	38	7560	10	1960	10	1960	12	2420	10	1960
	Loss of Cargo	---	120000	---	200000	---	250000	---	85000	---	170000
	Total	1731	236630	2638	317150	2045	402120	1623	544760	2732	546250
Steel & Water	Primary	377	95340	377	95340	377	95340	377	95350	377	95340
	Secondary	1012	12460	1965	18990	1395	52740	921	326290	2057	268740
	Containment	178	75650	178	75650	178	75650	178	75650	178	75650
	Support	37	7380	9	1780	9	1780	12	2420	9	1780
	Loss of Cargo	---	70000	---	150000	---	180000	---	30000	---	120000
	Total	1604	260830	2529	341760	1959	405510	1488	529700	2621	561510
Steel, Boron/Steel & Water	Primary	310	150830	310	150830	310	150830	310	150830	310	150830
	Secondary	996	12220	1956	18950	1378	50760	907	322100	2048	267870
	Containment	173	74390	173	74390	173	74390	173	74390	173	74390
	Support	37	7360	9	1760	9	1760	2	2420	9	1760
	Loss of Cargo	---	50000	----	140000	---	170000	---	20000	---	111000
	Total	1516	294800	2448	385930	1870	447740	1402	569740	2540	604850

The secondary shields are divided into those placed around the containment and those at the reactor compartment bulkheads. The advantage offered by the latter is the facility to inspect and test the containment from outside, but this advantage is economically penalized by the greater loss of cargo charge that would have to be paid.

The weight of the secondary shields are the governing factor in the total weights and the arrangements,including the lead and polythlene secondary shield,are the lightest. Finally, the range of weights, however, is not so great as the range of costs.

6.5.5 Stationary PWR Shielding Design

In a stationary PWR plant, the shield against radiation is given by:

The Thermal Shield (See Figure 6.20)

An one-piece component, integral with the lower core support barrel assembly, forms an annular flow channel which routes the coolant to the core. The shield protects the vessel by attenuating much of the gamma radiation and some of the fast neutrons that escape from the core, and reduces thermal stresses in the vessel resulting from the heat generated by gamma energy.

The Containment

A continuous reinforced cylindrical concrete structure with a hemispherical roof and a flat foundation slab. The inside face of the concrete shell is steel lined to insure a

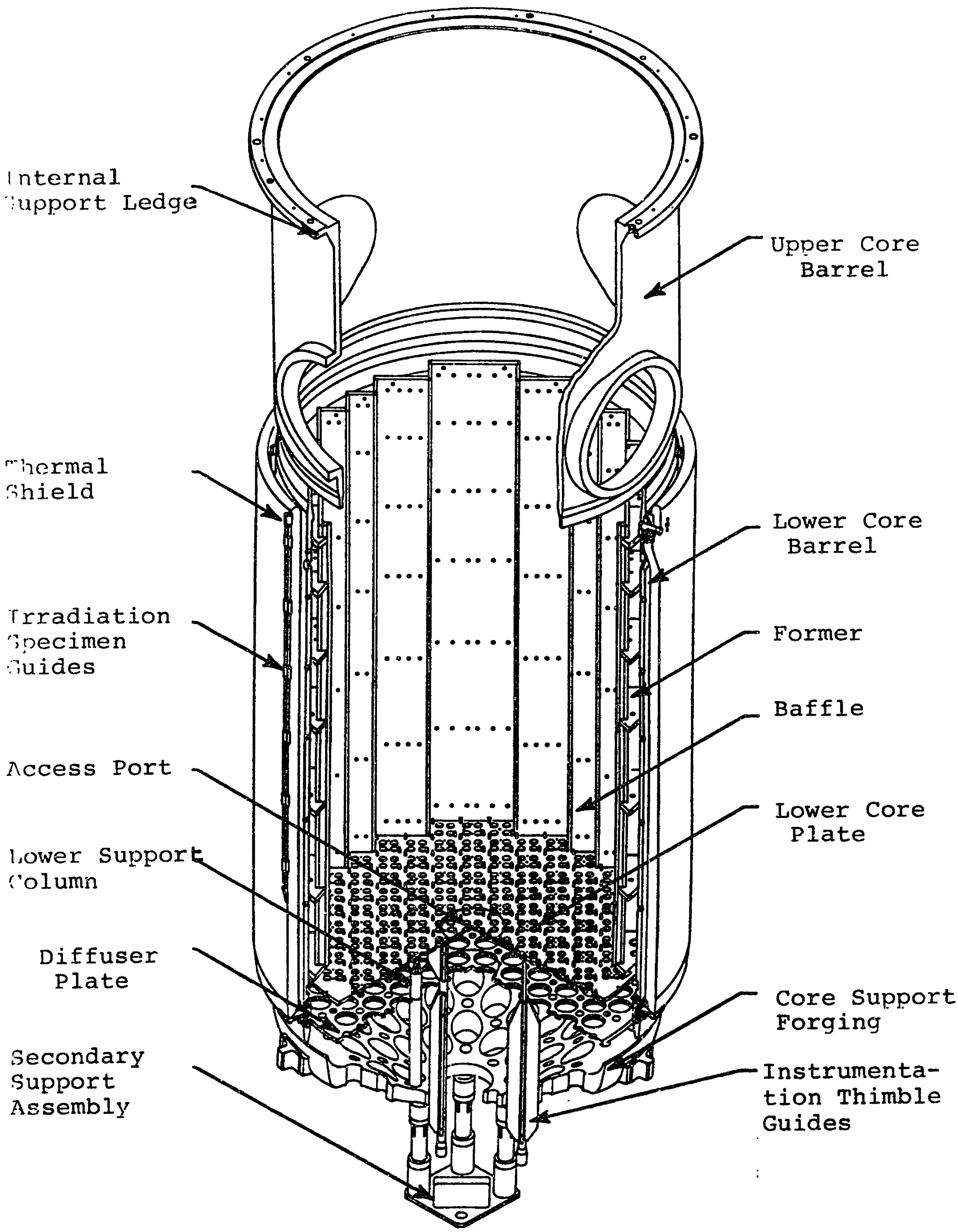


FIGURE 6.20 LOWER CORE SUPPORT STRUCTURE FOR STATIONARY PWR

high degree of leak tightness. The structure provides biological shielding ensuring that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded in both normal and accident situations. The approximate dimensions of the containment are:

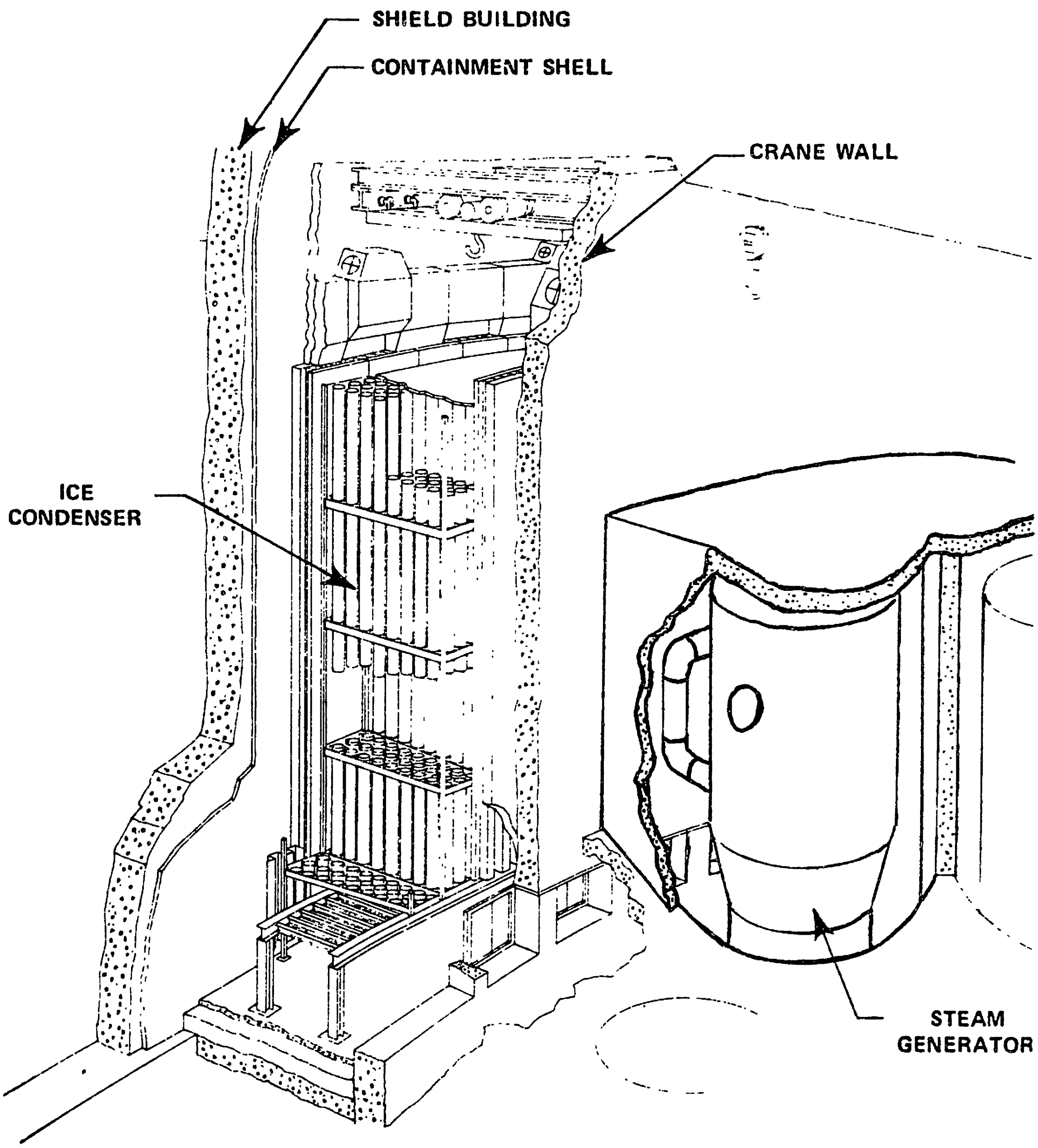
124 feet inside diameter, 203 feet inside height

3 1/2 feet wall thickness, and 2 1/2 feet down thickness

Representative details of the construction that are used are shown in Figure 6.7.

In Figure 6.21 it can be seen that, besides the shielding provided by the containment, there is another wall like crane-wall and a steam generator house wall which provide shield to the radiation. The same occurs with the different equipment that is inside the containment which in one way or another provides shielding against radiation.

FIGURE 6.21 ICE CONDENSER COMPARTMENT SHOWING SEPARATE SHIELD FOR STEAM GENERATOR, STATIONARY PWR



CHAPTER VII

MAJOR REACTOR ASSEMBLIES - SYSTEMS DIFFERENCES

So far, we have seen how different factors affect the design and behaviour of some parts of maritime reactor plants compared with stationary reactor plants. In this chapter, we will try to present the different reactor assemblies that present the most important technological differences between applications for stationary and maritime reactor, taking into account operational, environmental, safety and economic factors altogether.

7.1 Reactivity Control System Assembly

This assembly comprises both the control system plus the mechanisms that drive this system.

7.1.1 Maritime Reactor Design Restrictions

Due to operational factors, the magnetic jack control rod drive mechanisms cannot be used. Environmental-safety factors recommend to avoid the use of chemical shim as a regular reactivity control form. They also require extra systems to shut the reactor down in case of abnormal ship attitude. Economics-operational factors demand high differential control rod reactivity worth.

7.1.2 Maritime Reactor Reactivity Control System Assembly

In a nuclear shipboard reactor, the core reactivity is controlled basically by rod cluster control assemblies moved by only one type of drive mechanism. The control rod assemblies have two functions: safety or control. Their reactivity worth is different depending upon their function and position (see Tables 7.1 & 7.2). The total required reactivity worth depends on core design and core lifetime (see Table 7.3). In case of reactor shutdown all the assemblies are inserted.

The only type of drive mechanisms used is the roller-nut type which brings to the control rod the possibility to join both smooth insertion in normal condition and instantaneous insertion in safety conditions.

To provide instantaneous insertion of all control rod assemblies even in the ship capsize condition, this system is complemented by a scram spring and a powered insertion system. Major detail of this system is given separately in point 5.4.3. Then in this kind of reactor the control rod assemblies are used to control reactivity components due to the moderator temperature deficit, equilibrium xenon and samarium, transient xenon, doppler deficit, xenon undershoot, fuel burnup and fission product buildup and finally shutdown margin.

The control rod assemblies are complemented in normal reactivity control functions by fixed burnable poisons. These burnable poisons limit initial excess reactivity, flatten radial power distribution and, along with the control rods, control reactivity effects from fuel burnup and fission product build up.

Fixed burnable poison is used in maritime service instead of the regular soluble poison control system of stationary reactors.

TABLE 7.1

SCRAM & REGULATION GROUP WORTHS OF CNSG

	Reactivity, % $\Delta k/k$		Reactivity, Dollars	
	BOL	EOL	BOL	
Scram Worth				
Total for 20 CRAs	13.2	6.3	19.70	10.73
Maximum Differential	3.8	1.2	5.71	1.99
Regulating Worth				
Total for 17 CRAs	7.6	(a)	11.29	(a)
Maximum Differential	3.2	(a)	4.70	(a)

[a] At EOL all regulating rods are out of the core except bank 1. Total worth of bank 1 from EOL critical position = 0.5% $\Delta k/k$.

TABLE 7.2

CONTROL ROD BANK & CRA WORTHS OF CNSG

	Reactivity, % $\Delta k/k$		Reactivity, Dollars	
	BOL	EOL	BOL	EOL
Core Condition				
All regulating rods fully inserted, all safety rods out				
Maximum bank withdrawal worth	3.0	2.9	4.43	4.94
Maximum dropped rod worth	0.2	0.2	0.35	0.39
All rods fully inserted				
Maximum bank withdrawal worth	6.5	6.4	9.70	10.81
Maximum stuck rod worth[a]	3.5	3.4	5.17	5.76

[a] Conservative

TABLE 7.3

SHUTDOWN REACTIVITY ANALYSIS - HOT OF CNSG

	REACTIVITY			
	% Δk/k		Dollars	
	BOL	EOL	BOL	EOL
Required Movable Control Rod Worth				
Doppler deficit, 0-100%	0.7	0.9	1.04	1.61
Moderator deficit (505-589F)	1.8	2.4	2.64	4.07
Inserted transient CR worth	4.7	0.2	7.05	0.34
Possible reactivity feedback from xenon undershoot (below equil)	0.5	0.5	0.75	0.85
Total required rod worth	7.7	4.0	11.48	6.87
Shutdown Analysis				
Total calculated worth (37 rods)	20.8	20.4	30.99	34.65
Rod model correction (10%)	2.1	2.0	3.10	3.47
Rod nut effect	0.	1.8	0.00	3.05
Stuck rod worth (not reduced)[a]	3.5	3.4	5.17	5.76
Reduction in rod worth (589-505F)	1.6	1.6	2.39	2.71
Net rod worth available	13.6	11.6	20.33	19.66
Excess Control Rod Worth				
Net rod worth available minus total required rod worth	5.9	7.6	8.85	12.79

[a] Conservative

These burnable poisons are presented in the form of individual lumped burnable poison rod (LBPR) and burnable poison rod assemblies (BPRAs). The lumped burnable poison rods (LBPR) are separate burnable poison rods, dispersed in the fuel rod array (see Figure 5.5). They contain variable concentrations of burnable poison dispersed in mixed pellets. The burnable poison concentration varies radially and axially within the core. Each rod has three axial zones. All rods in a given fuel assembly have the same boron concentration.

Burnable poison rod assemblies (BPRAs) are inserted into the tube that corresponds to the control rod assemblies (CRAs) in those fuel assemblies containing no CRAs. The BPRAs are latched in place and are not moved during the fuel cycle. Since maritime reactors are batch refueling, these BPRAs are changed in each refueling. These assemblies absorb excess reactivity at beginning of life. Their absorption capability decreases until, at the end of fuel cycle, they have almost no effect on the core reactivity.

Each BPRAs has a number of burnable poison rods, a spider and a coupling mechanism. The coupling mechanism and the poison rods are attached to the spider. The basic assembly is similar to the control rod assembly. Each burnable poison rod has a section of sintered pellets containing boron, which serve as burnable poison. The boron concentration is not varied along the length of the rod. The poison section is axially located by internal spacer.

Soluble poison is not considered as a regular reactivity control system because in case of a ship sinking accident it might result in seawater inleakage and therefore, due to boric acid dilution, in reactivity addition in the core.

Soluble poison is used only as an utmost emergency system. In that case, soluble poison is supplied through a boric acid addition system. Thus, this system provide an attenuate means of removing excess reactivity from reactor core, but this boric acid supply system could not operate in case of seawater inleakage due to sinkage. The system can also provide means for maintaining the reactor in a subcritical condition in a cold, clean, stuck rod configuration.

7.1.3 Stationary PWRs Reactivity Control System Assembly

Normal reactivity control is provided by neutron absorbing control rods and by soluble chemical neutron absorber in the reactor coolant. The concentration of soluble chemical neutron absorber in the coolant is varied as necessary during the life of the core. Only in the first fuel cycle, burnable poison rods are used. This is needed because the first cycle contains more excess reactivity than subsequent cycles, due to the loading of fresh fuel. This excess of reactivity cannot be controlled by the soluble neutron absorber because to do that control, the boron concentration would be on the order of 1700 ppm and the moderator temperature coefficient would be positive. The tendency of the core to undergo azimuthal or diametral xenon oscillations is a function of the moderator temperature coefficient. Then to assure that the core is stable, the moderator temperature coefficient must be made sufficiently negative by the incorporation of fixed burnable poison rods in the first core.

Figure 7.2 shows the reduction in chemical shim brought about by the presence of this burnable poison. Figure 7.1 shows the approximate burnable poison rod locations in the first fuel cycle of a PWR four-loop-core. At the end of the first cycle, during refueling, the burnable poison assemblies are discarded.

The soluble chemical neutron absorber concentration is varied to control long-term reactivity changes such as:

1. Fuel depletion & fission product buildup
2. Cold to hot, zero power reactivity change
3. Reactivity change produced by intermediate-term fission products such as xenon and samarium
4. Burnable poison depletion

At the beginning of any cycle, a reactivity reserve equal to the depletion of the fissionable fuel and the buildup of fission product poisons over the specified cycle life must be "built" into the reactor. This excess reactivity is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant. Thus, the boron concentrations change from a maximum value at beginning of life depending on different conditions (see Table 7.4), to a minimum value of approximately 40 to 50 ppm at the end of each cycle. The end of design cycle life is defined to occur when the chemical shim concentration is essentially zero with

FIGURE 7.1 NUMBER & LOCATION OF BURNABLE POISON RODS OF A STATIONARY PWR PLANT

					9		9		9					
		8		12		20		20		12		8		
	8		20		12		12		12		20		8	
		20		20		16		16		20		20		
	12		20		20		16		20		20		12	
9		12		20		16		16		20		12		9
	20		16		16		16		16		16		20	
9		12		16		16		16		16		12		9
	20		16		16		16		16		16		20	
9		12		20		16		16		20		12		9
	12		20		20		16		20		20		12	
		20		20		16		16		20		20		
	8		20		12		12		12		20		8	
		8		12		20		20		12		8		
					9		9		9					

FIGURE 7.2 BORON CONCENTRATION VERSUS FIRST CYCLE BURNUP IN A TYPICAL STATIONARY PWR PLANT

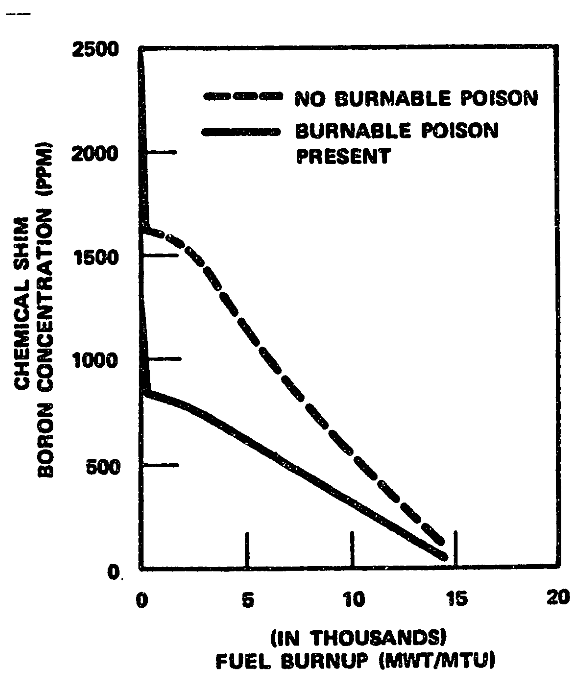


TABLE 7.4 REACTOR DESIGN COMPARISON TABLE FOR STATIONARY PWR PLANT

	17 X 17 Fuel Assembly		15 X 15 Fuel Assembly			
	Reference 4-Loop Plant Without Loop Stop Valves	Reference 4-Loop Plant With Loop Stop Valves	Reference 4-Loop Plant Without Loop Stop Valves	Reference 4-Loop Plant With Loop Stop Valves	McGuire Units 1 & 2	Trojan
Boron Coefficient in Primary Coolant, %Δρ/ppm						
Hot	1%/120	1%/120	1%Δρ/120	1%Δρ/120	1%Δρ/100	1%Δρ/100
Cold	1%/60	1%/60	1%Δρ/60	1%Δρ/60	1%Δρ/85	1%Δρ/85
Boron Concentrations (First Cycle with Burnable Poisons), ppm						
Cold, K_{eff}=0.99, RCCA withdrawn	1447	1447	<1500	<1500	<1500	1420
Hot, Poisoned, k_{eff}=1	894	894	<1200	<1200	<1100	950
Reactivity Requirements for RCCA, %Δρ(BOL/EOL)						
Control Requirements						
Power Defects	2.01/3.45	2.01/3.45	~1.70/~3.65	~1.70/~3.65	~1.70/~3.85	1.85/3.43
Rod insertion allowance	0.50/0.50	0.50/0.50	0.70/0.50	0.70/0.50	0.70/0.30	0.60/0.60
Total	2.51/3.95	2.51/3.95	~2.40/~4.15	~2.40/~4.15	~2.40/4.15	2.45/4.03
Estimated RCCA Worth						
All full length in	8.00/7.30	8.00/7.30	~9.74/~7.33	~9.74/~7.33	~9.74/~7.33	9.06/7.79
All but highest worth in	7.00/6.20	7.00/6.20	~8.44/~6.383	~8.44/~6.38	~8.44/~6.38	8.03/6.26
Estimated RCCA Credit with 10% uncertainty adjustment	6.30/5.58	6.30/5.58	7.60/5.75	7.60/5.75	7.60/5.75	7.23/5.63
Shutdown Margin Available	3.79/1.63	3.79/1.63	~5.20/≥1.60	~5.20/≥1.60	5.20/≥1.60	4.78/≥1.60

control rod present to the degree necessary for operational requirements.

The rod cluster control assemblies provide reactivity control for:

1. Fast shutdown
2. Reactivity changes associated with changes in the average coolant temperature above hot zero power
3. Reactivity associated with any void formation
4. Reactivity changes associated with power coefficient of reactivity
5. Reactivity changes associated with xenon oscillations

There are two types of control rod assemblies: Full length rod cluster control assembly and part length rod cluster control assembly.

The full length rod cluster control assemblies are divided into two groups: control and shutdown. The control group compensates for reactivity changes due to variations in operating conditions of the reactor, i.e., power and temperature. Both the control and shutdown groups provide adequate shutdown margin, which is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped assuming that the highest worth assembly remains fully withdrawn, assuming no changes in xenon or boron concentration or part length rod position.

The full length control rod assembly is driven by a magnetically operated jack mechanism. A magnetic jack is an arrangement of three electro-magnets which are energized in a controlled sequence by a power cycler to insert or withdraw rod control clusters in the reactor core in discrete steps. As was said before, due to operational factors this mechanism can not be used in a maritime reactor. This control rod drive mechanism is a trip design. Tripping can occur during any part of the power cycler sequencing if power to the coils is interrupted.

The function of the part length rods is to control axial neutron flux shape and axial xenon oscillations, should they occur. The xenon oscillation in the axial connection can be induced after the first refueling and only under certain circumstances by action of control rods and the shift of the axial density distribution with power. The part length rods are on manual control.

The axial position of the part length rods depends on the full length rod insertion, power level, xenon distribution, etc., since generally a considerable portion of the neutron power generation takes place in the region containing the part length rods, their effect on possible reactivity insertion rates and power shape is significant.

The part length rod cluster control assemblies are identical to the full length rod cluster control assemblies except for the drive rod assembly and because only the bottom part of each rod contains absorber material. The part length control rod assembly is driven by an electro-mechanical roller-nut mechanism. As was explained in Chapter III, this mechanism is similar to the control rod drive mechanisms used in maritime reactors except for the physical configuration of the nut, and trip behavior. This control rod drive mechanism is non-trip design.

7.1.4 Roller-Nut Type Control Rod Drive Mechanism

Given that control rods are the most important part in the reactor reactivity control system assembly, it is important to describe the differences in the drive mechanism that can be used in both stationary and maritime reactors.

7.1.4.1 Roller-Nut Type Control Rod Drive Mechanism For Maritime Reactor Applications

This drive mechanism consists of: a motor tube assembly, a closure and vent assembly, a lower mechanism assembly, a rotor assembly and stator assembly.

Motor Tube

The motor tube houses the rotor assembly . It is closed on the upper end with a closure and vent assembly and is part of the primary pressure boundary. The motor tube is surrounded by an external motor stator.

Closure & Vent Assembly

The upper end of the motor tube is closed by an insert closure assembly containing a vapor bleed port and vent valve

Lower Mechanism Assembly

This assembly, as a unit, is lowered into the reactor vessel head, and bolted to the reactor head nozzle. Its major components are an assembly housing, a trip spring to assist scram, a thermal barrier and a leadscrew guide housing to reduce the temperature in the roller-nut area and to support the lower end of the leadscrew, and a snubber piston to reduce scram deceleration forces.

Rotor Assembly

The rotor assembly operates completely immersed in primary water within the pressure boundary and is designed to operate without any other lubricant. Its principal parts are the rotor tube, two segment arms, four pivot pins, four roller-nut assemblies, thrust bearing, a radial bearing, four compression springs, a rotational stop and a thrust bearing retainer.

The central support of the rotor assembly is the rotor tube which is hollow to allow axial passage of the leadscrew. It also provides support for the pivots of the rotor segment arms.

The rotor segment arms are attached by pivot pins to the rotor tube. The arm rotates with and pivots on the rotor tube. The upper portion of the two arms are designed to form a four-pole collapsible rotor; the lower portion of the arms each contain two roller nut assemblies which, when the stator is energized, engage with the leadscrew threads. Energizing the stator causes the upper portions of the segment arms to be pulled radially outward. The lower portions of the arms move radially inward, and the four roller-nuts engage the leadscrew.

Four compression springs below the pivots of the arms act to keep the roller-nut disengaged from the leadscrew. To engage the leadscrew, a greater force is applied to the arms above the pivot point by the stator magnetic field.

Stator Assembly

The stator assembly is mounted on the motor tube and together with the rotor nut assembly, form a reluctance-type step motor. When the stator is energized with a continuous direct current, the electromagnetic field attracts the upper ends of the rotor assembly segment arms outward, and causes the roller-nut assemblies in the lower ends of the arms to engage and hold the leadscrew. If an IN or OUT command current is applied to the stator, the rotor assembly remains engaged with the leadscrew and rotates axially about the leadscrew causing it to be raised or lowered. The direction of the rotation depends on the sequence in which the stator winding is energized.

The stator is a six-winding, which permits reverse of magnetic field without reversing priority of any winding. By alternately energizing the opposing windings in proper sequence, a rotating magnetic field is developed within the stator.

Three stator windings are energized at a time to produce a rotating magnetic field. This magnetic field is coupled to a four-pole-rotor, thus forming a reluctance-type drive motor. As the stator coils are sequenced, the separating antifriction roller-nut assemblies are magnetically rotated driving the leadscrew coupled to the control rod assembly. Stator current causes the seaparating roller-nut assembly halves to close and engage the leadscrew. Mechanical springs disengage the roller-nut halves from the screw in the absence of current. For rapid insertion, the nut halves separate and release the lead-screw and control rod. The drive is designed to trip when called for by the reactor protection system or any loss of electric power. The trip signal from the protection system acts to interrupt power to all control rod drive mechanism stators. A mechanical spring, aided by gravity, forces the control rods into the core.

In addition to the normal rod movement and trip features, the control rod drive mechanism and the power supply provide simultaneously powered insertion capability for ship angles outside those encountered during normal operation, including total turnover. The powered insertion mode drives all rods in a speed faster than normal.

7.1.4.2 Roller-Nut Type Control Rod Drive Mechanism for Stationary Reactor Applications

As roller-nut type for stationary reactor, this mechanism also consists of four separate subassemblies. They are the pressure vessel, the stator, the rotor, and the translating leadscrew.

The Pressure Vessel includes an adapter and a motor tube which are connected by a threaded, seal-welded maintenance joint which facilitates replacements of the control rod drive mechanism internals. The adapter is the lower portion of the vessel and contain the lower rotor assembly. The motor tube is the upper portion of the vessel. It contains the upper rotor and provides space for the leadscrew during its upward movement as the control rods are withdrawn from the core.

<u>The Leadscrew Assembly</u> includes a flexible coupling, a leadscrew, a disconnect button. The leadscrew has threads which are engaged with the roller assembly and cause linear travel of the leadscrew when the rotor rotates.

This two subassemblies do not differ significantly from the maritime roller-nut type.

The following two subassemblies differ from the maritime ones, not in their functional design but in their physical conception.

<u>The Rotor Assembly</u> includes a thrust bearing roller-assembly, lower rotor, upper rotor, brake arms, and a radial bearing. Different from the maritime type here the roller-nut is compact.

The internal rotor assembly is the operating center of the mechanism. The rotor assembly is free to rotate in and is held in place within the pressure vessel by bearing assemblies. Five free rotating "rollers" are held captive in the lower cylindrical portion of the rotor and are canted to match the lead angle of the drive rod threads. As internal rotor assembly rotates, the roller-nut turns within the threads of the drive rod, translating vertical motion to it.

The Stator Assembly includes a cooling shroud, electrical conduit and connector, and the basic stator jacket, stack, and windings. Stator energization causes the brake arms on the rotor to swing outward and release the mechanical brake. When the stator windings are energized in a controlled sequence by a power cycler, a rotating magnetic field is produced in the rotor causing it to rotate.

In operation, at least two of the six windings are energized by the sequencing to obtain the required motion. When power to the stator is interrupted, the mechanical brake arms swing in and engage the mechanical brake which stops the leadscrew and the rod control cluster assembly motion. The application of power to any or all of the windings disengage the brakes. The rotational energy is supplied in sequential pulses to the armature which rotates directionally 15 degrees per pulse as controlled by the power supply.

7.1.5 Stability of the Core Against Xenon Induced Spatial Oscillation

In connection with the operational reactivity controls for both stationary and maritime reactors, it is interesting to show the role that the differences of power play in the selection of the reactivity control system.

The part length control rods are provided in that reactors where the problem of xenon oscillations in the axial direction are expected.

Xenon instabilities occur in large reactors, operating at constant power and with very high thermal neutrons flux.

The character of power oscillations that are induced by xenon oscillations can be determined analytically. The regional power can be expressed as an exponential sinusoidal function of the following form:

$$P(t) = P_o\, e^{bt} \sin\,(\,2\Pi\, t/T + \tau_o) + A_o$$

where

$P(t)$ = power at time t

P_0 = amplitude relative to A_o at $t = 0$

b = stability index, h^{-1}

T = period, h

τ_o = phase shift

A_o = average power

The stability index b and the period T describe the behavior of power oscillations in a region of the core after some perturbation has disrupted the normal power level. A positive index denotes a diverging oscillations; a negative index denotes a converging oscillation.

Stability indexes should be calculated for both the axial and azimuthal cases at beginning of life. A sensitivity analysis should be performed to determine the change in stability index as certain parameters are varied.

The maximum expected changes in the parameters must be used to find the maximum changes in the stability index. If the stability index, when all these changes are added together, is positive, power instabilities will occur. Generally, in order that they occur, the reactor needs to be sufficiently large so that two regions can be distinguished, working as independent units, such that the neutrons produced by fission in one of the regions do not promote a large number of fission in the other region. To satisfy this condition, it is enough that one of the reactor dimensions - diameter or height, for example - were several times greater than the neutron mean free path.

In reactors using low enrichment fuel for maritime applications, where the biggest power needed never pass the 400 Mwt, the core is assumed to be stable against xenon oscillations because of its physical size. Due to this condition of stability, reactors used to propel ship do not need to use part length control rods.

7.2 Containment System

7.2.1 Restriction for Containment Design of Maritime Reactors

7.2.1.1 Influence of Safety & Environmental Factors

The containments are designed for all credible conditions of loading, individual normal loads, loads during LOCA, test loads and loads due to adverse environmental conditions.

1°) For maritime reactor containment design, it is necessary to add to the normal loads of a stationary design, the loads derived from ship motion and ship vibration.

2°) Taking into account the differences between the reactor types used in maritime (integral type) and stationary (loop-type) application, the loads during LOCA are lesser in maritime than in stationary reactors.

3°) Due to adverse environmental conditions, such as nautical accident or ship attitude, the loads derived from those conditions are stronger than those supported by stationary reactor containment. Thus, while for stationary reactors the earthquake loads, considered as the critical loads conditions, can have a maximum horizontal ground acceleration of 0.15g and most of the design are made for 0.25g; maritime reactor containment is designed to support vertical acceleration of 1g.

4°) In maritime applications it is necessary to consider the external loads derived from a ship sinking. For that possibility a special anticollapse valve must be built in the maritime containment design.

7.2.1.2 Influence of Economic Factors

Due to the economics resulting from the reduced volume and weight in maritime PWR reactors, the pressure suppression system for containment is adopted in contrast to the dry containment system used in stationary PWRs. The pressure suppression system is used in stationary BWR's but with some different concepts such as the wet well subdivisions. However, the pressure suppression system is used more frequently in some routine BWR operations, whereas in the PWR it used only for emergency blowdown.

7.2.2 Containment Design for Maritime Reactors

7.2.2.1 Pressure Suppression System

The containment that encloses the primary system of a nuclear power plant is the last barrier safeguarding against the release of reactive fission products from nuclear fuel to the environment during a LOCA. A pressure-suppression type of containment system which reduces the equilibrium pressure by steam condensation can be used to reduce the overall size of the envelope around the reactor and minimize weight by keeping the shell design pressure down. These two desirable characteristics, compactness and lightweight, are necessary for a shipboard power system.

The basic concept of pressure suppression is to force the large quantities of steam, which originates from the primary system in the case of a LOCA, through a vent pipe system into a water pool where the steam is condensed. The density, heat capacity, and latent heat of vaporization of water provide a large capacity heat sink and greatly reduce the containment temperature and pressure.

Pressure-suppression containment appears to be very attractive from the safety and economics stand points. It is especially suitable for plants with compact nuclear steam systems such as the integral reactor type for ship power supply where the reduction of volume and weight are the important considerations.

To maintain the integrity of the containment, knowledge of the containment pressure-temperature history during a

LOCA is necessary for the safety analysis and design of the containment. A code must be written to predict the pressure-temperature response to a LOCA within a pressure-suppression containment vessel. Pressure and temperature produced after the double-ended rupture of the largest primary coolant pipe - the pressure surge line, considered the maximum accident credible,are determined by calculations evaluating the combined influence of the energy sources, heat sink and engineered safety features.

Effects of the Marine Environmental on the Containment Design

By virtue of its installation aboard a ship, the containment and the components and machinery within it must be designed to remain in place regardless of orientation if the ship should capsize and eventually sink. The containment vessel must be designed to remain intact without collapse due to external pressure if the ship sinks. This problem is solved using anticollapse automatic pressure valves in the containment vessel such as were described in Chapter V.

The containment must be designed for all credible conditions of loading, including loads due to ship motion and to adverse marine environmental conditions. These loads are:

1. Normal loads include the following:
   - dead loads and their related moments and forces, including any permanent equipment loads and the vertical and lateral pressures of liquids.
   - live loads and their related moments and forces

including movable equipment loads and other loads that vary with intensity and occurrence, like ship motion and pressure differences due to variations in heating and cooling.

2. Severe Environmental Loads : Included in this category are:
   - loads generated by ship collision or grounding
3. Extreme Environmental Loads - Are those which are credible but highly improbable. They include loads generated by capsized or submerged conditions.
4. Abnormal Loads - Are those generated by a postulated high energy pipe break accident. The containment design to avoid the effect of these loads is described below.

7.2.2.2 Description of Pressure-Suppression Containment System

1. Basically the pressure-suppression containment system consists of:
   - the main containment, the dry well, which houses the NSS primary system.
   - up to eight non-identical pressure-suppression chambers in wet wells.
   - the vent pipe systems that vent the two-phase air-water-steam mixture from the dry well to wet wells.
   - a number of engineered safety features, such as dry well spray systems, wet well spray systems and air-circulation fan cooler systems

A schematic drawing of the basic pressure-suppression containment system is shown in Figure 7.3.

2. Main Containment Vessel (Dry Well)

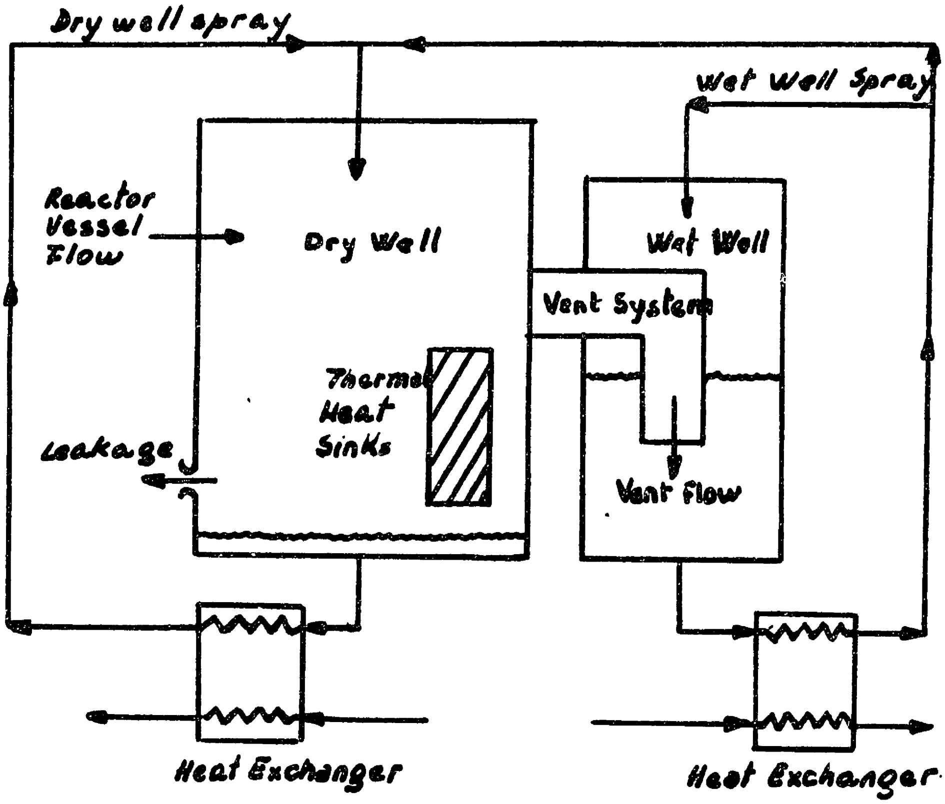
The dry well volume is divided into two regions: a liquid water sump on the bottom and an atmospheric region above, containing a water, steam, and air mixture. Each region is assumed to be well mixed and to have a uniform temperature, but the temperatures of each region may be different. The steam and energy released from the primary system after blowdown are time-dependent.

The engineered safety features for the dry well include a dry well spray system and a fan cooler system for air circulation. The water supplied for the dry well spray system comes from the liquid region of the dry well. A heat exchanger is provided to cool the water before it is sprayed into the dry well atmosphere. The spray water can also be provided from an external source and sprayed directly into the dry well atmosphere.

3. Vent Pipe System

The vent pipe system is a passage composed of a series of flow pipe elements which transport the dry well air-water-steam mixture into the wet well. They may be of two types: (1) straight, constant flow area length of pipe, (2) pipe with abrupt change in flow area or direction. For the pressure drop calculation one needs to know the pipe length, inside diameter and the absolute roughness in the first case and the down stream diameter and pressure drop loss coefficient

FIGURE 7.3 SCHEMATIC DIAGRAM OF PRESSURE-SUPPRESSION SYSTEM MODEL FOR AN INTEGRAL PWR PLANT



in the second case. The exit portion of the vent system may be submerged below the water surface within the wet well.

4. Pressure-Suppression Vessel (Wet Well)

A total of up to eight non-identical wet wells can be used for the pressure-suppression containment system. Each wet well is divided into two regions: a liquid region and an atmosphere region. Thermal non-equilibrium may exist between the two regions. Partial condensation of the steam can also be obtained. A vent-steam condensation coefficient is employed to specify the amount of condensation.

The wet-well spray system is so named because the spray flow taken from the wet-well liquid region provides a means for further utilizing the large thermal energy sink represented by the wet-well water pool to reduce the dry-well pressure. The spray flow passes through a counter-flow type heat exchanger, part spraying into the dry-well and part diverting back into the wet-well.

7.2.2.3 Capability of Pressure-Suppression System for Use in Shipboard Applications

A pressure-suppression test was conducted by B&W in its CNSG to investigate the capability of the concept for use in shipboard applications. Varying vent submergence depths and ship attitudes were studied. That test program utilized a high-pressure (1850 psig) primary system and a simulated CNSG containment.

The response of the marine CNSG pressure-suppression containment has been analyzed for RCS LOCAs from several lines that are connected to the RCS. The lines considered are the pressurizer surge line, the delay heat removal system injection and letdown lines, and an instrument line.

The release of mass and energy from the reactor coolant system through these different breaks are used as input to calculate pressure and temperature supported by the containment shell after accident as a time-dependent function.

Resulting temperature and pressure peaks for the various breaks with wet well 65% full of water are as follows:

Breaks Type	Peak Pressure Psig	Peak Temperature °F
Surge line double-ended rupture	58.6	306
Delay heat removal system injection line	48.1	295
Delay heat removal system letdown line	57.5	304
Instrumentation line	5.2	187

(see Figures 7.4 & 7.5)

Both, a maximum pressure peak of 68.5 psig, and a maximum temperature peak of 314 °F, in the containment was obtained for the pressurizer surge line break with wet-well 70% full of water.

From these peak values, the containment design temperature and pressure have been set. Applying the generally accepted safety pressure factor of 1.5, the containment design pressure was raised to a value of 105 psig, an additional 10% of the temperature increased during transient was added at the maximum temperature expected, leading to a design temperature of 350°F.

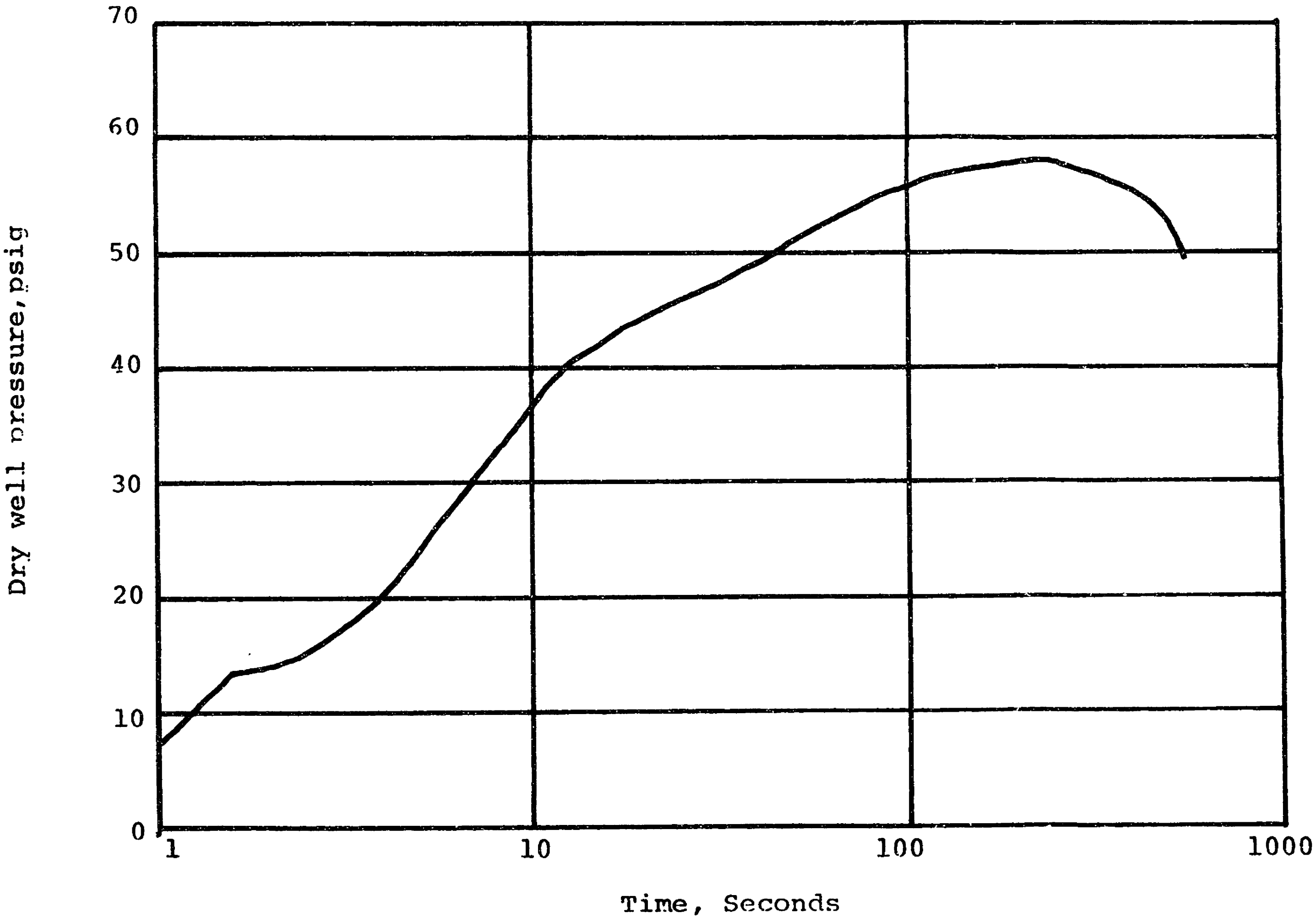


FIGURE 7.4 DRY WELL PRESSURE RESPONSE TO SURGE LINE LOCA WITH WET WELLS 65% FILLED WITH WATER CALCULATED FOR INTEGRAL CNSG.

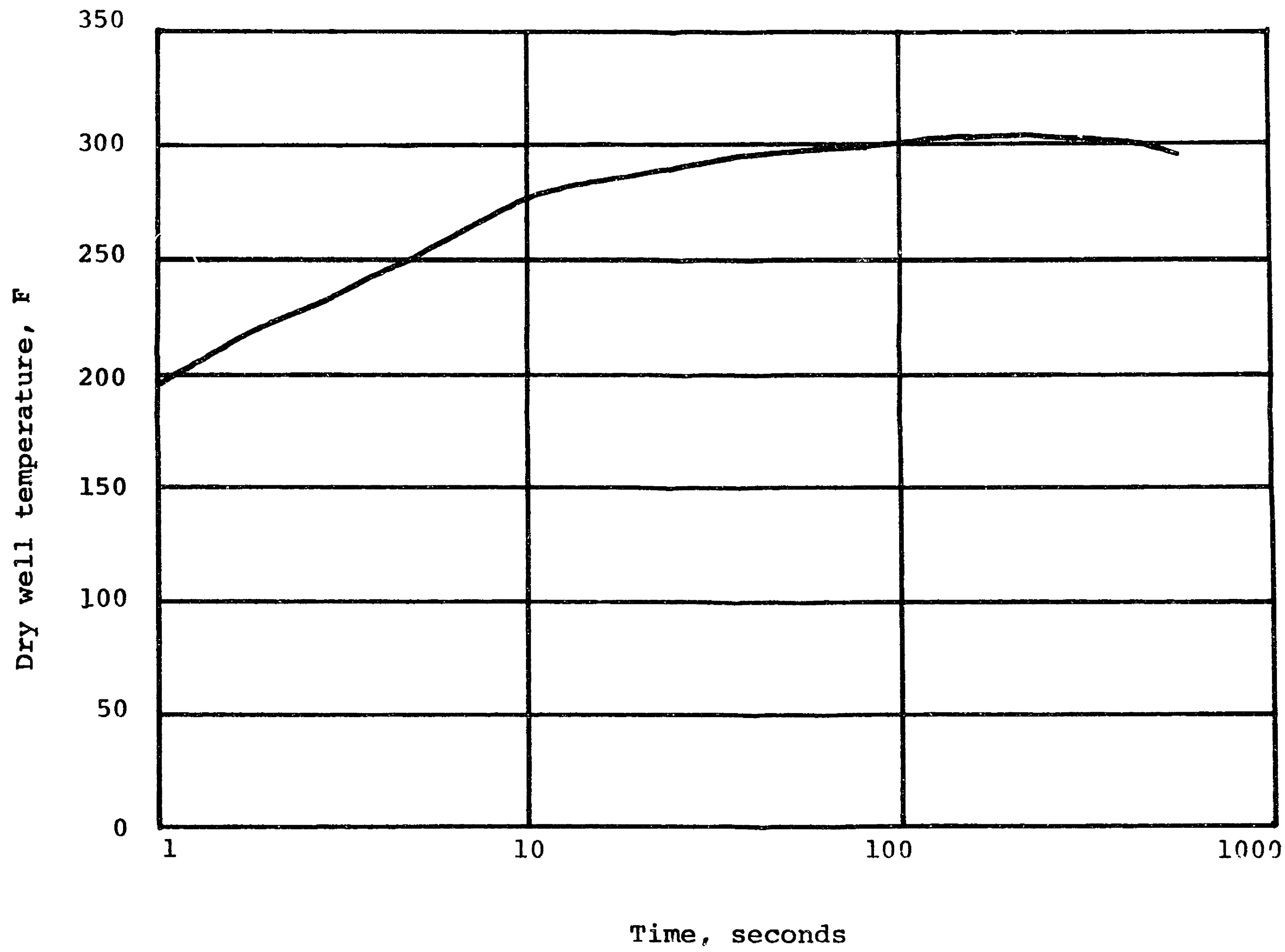


FIGURE 7.5 DRY WELL TEMPERATURE RESPONSE TO SURGE LINE LOCA WITH WET WELLS 65% FILLED WITH WATER CALCULATED FOR INTEGRAL CNSG.

CHAPTER VIII

STRATEGY FOR DEVELOPMENT OF NUCLEAR-POWERED MERCHANT SHIPS

8.1 Introduction

The strategy must be directed toward the development of a more efficient nuclear-propulsion technology. Nuclear-propulsion technology can be improved basically, in two ways.

First, present technology can be made more efficient if production runs will, in fact, reduce unit costs. This means more than simply achieving economics through producing more than one of a kind; it means finding better ways to produce unit number "X" than were used to produce unit number "one". Whether such ways can be found, no one knows.

The second way to achieve more efficient nuclear technology is to encourage research on a diversity of approaches. Whether small reactors become available means of supplying industrial energy depends to a large extent on the number of units per year demanded by the market. If the potential market is only one or two units annually and if each of these is a custom-engineered system, then the cost of small reactors is likely to be prohibitive for most commercial applications. If a number of applications for essentially the same reactor technology could be identified, the prospects for economical small reactors would be substantially improved. Each of these ways will be discussed.

8.2 Opportunities for Technological Improvement in Nuclear Powered Ship

The CNSG for maritime application shown in this work is similar in many respects and in many design details to the conventional utility PWR system. The material for the reactor vessel, the steam generators, the core and the internals are essentially identical to those used in the utility designs. They are chosen to minimize stress and are proven in stationary service. The design detail utilized in CNSG, including fuel assembly support grids, the manufacture of the fuel pins themselves, steam generator tube support sheets, some concepts for control rod drives and many other design aspects, incorporate directly details used in utility reactor design.

However, technological differences still exist in some reactor assemblies depending on their maritime or stationary application which, if we could eliminated them, could make the integral PWR type more economically interesting since they could be used both for shipboard or land-based plant.

8.2.1 Improvement in the Reactivity Control System Design

As was shown before, one of the reactor assemblies that presents more technological differences between stationary and maritime reactors is the reactivity control system.

8.2.1.1 Possibility of Diminished Ship Reactor Maneuver Requirement

Due to maneuvering and reactivity holddown, many control rods are required, just as different types of control rod drive mechanisms are necessary in maritime reactors.

To diminish or avoid these differences, it is possible to seek two alternatives:

First, trying to fulfill the maritime maneuvering requirements with minimum or no stationary reactor assemblies modification. For example, examining if fewer rods can be used and more effective reactivity holdown obtained. That means to adapt the reactor to maritime maneuvering requirements.

Second, adapting ships so that the reactor maneuver requirements do not differ so much from stationary reactor maneuvering, as to be necessary to change the technological design from one to another. For example, same type of control rods drive mechanism could be used in both maritime and stationary land reactors.

If we analyze the types of ships that can be powered economically by nuclear power, nowadays, we can see that these alternatives are feasible. The second alternative can be analyzed more deeply.

In their consideration could be included:

1. Future trends of types of ship more economically feasible to be nuclear powered.
2. Accepting that the more feasible nuclear powered ship can be an Ultra Large Crude Carrier (ULCC) or a high speed container ship with high utilization factor, analyze how the propulsion plants and safety requirements can be designed so that their maneuverability requirements do not differ from that of stationary reactor types. Here, it will be

necessary to consider: trade routes, use of auxiliary machines to power the ship during mooring maneuver with the help of a tug boat, impossibility to avoid collision or grounding only with fast maneuver when the ship is too big; possibility of maneuvering the ship going from full-ahead to full-astern or vice versa without changes in the reactor power.

Trade Routes - With long trade routes, as are necessary for an economical nuclear powered ship, the reactor will operate in steady state and maximum power the major part of the time with high utilization factor. Also the ship will be away from congested areas. Then the maneuver requirements diminish proportionally.

Use of Deepwater Port or Private Terminal

In regard to reduction of nuclear power risks, the use of containers does facilitate centralization of the cargo moving out of the particular geoeconomic region through one or two ports, and thus permits reduction of the number of ports at which the nuclear powered container ship will have to call. That is, the use of container reduces the number of terminals required for the service, allowing the operator to select special terminals at which the nuclear ship will call, and at which proper regulatory safeguards can be established, so that any risks of the nuclear power and the ship maneuver can be reduced to the full satisfaction of all parties concerned.

This will involve aspects such as radiological monitoring of the environment, good location near an isolated anchorage which can be employed if any difficulty should develop with the reactor while the nuclear ship is in port, and especially for the case that we are dealing with now - avoiding congested areas.

Finally, given physical security measures so that they can permit to reduce the ship maneuver requirements in the surrounding of the port just as during mooring maneuver. In the case of nuclear powered ULCC, this aspect is simpler since such vessel generally uses only one loading and one unloading port and many loading/unloading ports are in deep water offshore (single point mooring or similar).

Use of Auxiliary Machines & Boats for Mooring Maneuvers

For safety regulation every nuclear powered ship should be provided with a take-home propulsion system. This system could work alone or in parallel with the nuclear plant. In the last case it could absorb those power loads excess that the reactor, due to its maneuver restrictions, could not be able to accept.

This system could be complemented with bow and stern (active rudders, etc.) thrusters and helped by tugboats and special dock-side mooring facilities.

Relative Efficiency of the Ship Maneuverability to Avoid Nautical Accident

When the ship propelled by nuclear reactor is a ULCC or high speed container ship, the possibility of avoiding nautical accidents only with a reactor high maneuvering is remote. For example, the reference ship for the CNSG IV A designed by B&W, is a nuclear-steam propelled vessel designed and arranged for transporting approximately 591,000 tons of crude oil cargo. To stop this ultra large ship in a short distance as required to avoid collision, is impossible even though the reactor maneuvering response was very quick. Moreover, the highest machinery propulsion maneuvering, i.e., to pass from full ahead to full astern, can be done without affecting the reactor condition.

Full-Ahead & Full-Astern Maneuvering

While the ship is operating at 100% reactor power (full-ahead speed), the master may call for full-astern maneuver as a means of preventing collision. After receiving the command, the turbine control system will begin opening the astern throttle valves while closing the ahead valves. This applies torque in the reverse direction to the turbine shafts, thus stopping and reversing the rotation of the propulsion turbines. The propulsion turbines are directly connected through the reduction gears to the propeller shafts. As the propellers rotate in reverse, the ship slows to stop. Only small perturbations will be seen in the reactor plant secondary load because:

1- The astern and ahead valves will overlap in operation.

2- The rate at which these valves move will be consistent with possible reactor maneuvering rate.

3- Full rated steam flow (as a minimum) will be directed through the astern turbines.

8.2.1.2 Possibility of Use Boric Acid in Coolant as a Reactivity Control Measure

Under U.S. regulations, as was said before, boric acid is presently not a feature of the marine design because the ship sinking accident might result in seawater inleakage and accordingly, in reactivity addition in the core.

The use of boric acid in coolant as a reactivity control measure for long term reactivity effects can be cost effective, since it could permit to remove some control rods and control rod drives mechanisms, just as the total lumped burnable poison rods. The use of boric acid also permits diminishing the control rods reactivity worth. This reduction of control reactivity worth along with less reactor maneuvering requirements could permit the use of jack magnetic type control rod drive mechanisms as in stationary nuclear reactor.

The probability of seawater inleakage in a ship sinking accident and as consequence a dangerous reactivity addition in the core can be very remote since:

1- The sinking accident that could permit a seawater inleakage would arise only from a collision or grounding accident.

2- These accidents would have to be of such magnitude so as to break first the whole collision protection defense, secondly, the reactor containment, and finally the reactor pressure vessel.

3- If the seawater inleakage should occur, the reactivity addition arising from the water coolant dilution will be dangerous only if the reactor cannot be shut down by the action of the control rod assembly.

4- Finally, after a sinking accident, a dilution of the primary coolant with the water inleakaged at the containment could occur due to the breakage of the pressure vessel and not due to the breakage of the containment. In this case a boric acid addition system could be developed in such a way that the quantity of boric acid added could control the coolant dilution and as consequence the reactivity addition derived from that dilution.

8.2.2 Improvement in the Moderation of Ship Motion & Ship Behavior Under Heavy Weather

There are two ways to diminish the design technological requirements of the reactors for shipboard applications due to ship motion.

One, trying to moderate the ship motion; two, going deeper in the calculation of acceleration forces resulted from ship motion to establish rules that do not represent an over design of the reactor assemblies.

8.2.2.1 Seakeeping & Stabilization

All ships should be provided with some STABILIZATION SYSTEM in order to obtain effective stabilization when the ship is underway, as well as when it is still in the water without power or when it is without cargo; to reduce acceleration loading on equipment, and reactor plant.

The ship design could be improved to have excellent seakeeping reliability. Bow thrusters could be provided to permit maintenance of ship orientation when propulsion plants do not work properly (or at very low speed) avoiding the possibility that the ship is oriented perpendicular to the sea wave.

8.2.2.2 Improvement in Calculation of Acceleration Forces

It is important to point out the fact that the behavior of the different kind of ship under the same circumstances can be fairly different. Thus, for example, the acceleration forces detected in NSS "OTTO HAHN" under heavy sea conditions (see Figure 4.2) could not be used directly for the design of an ULCC of 600,000 ton to be powered by the CNSG of Babcock and Wilcox, where due to the dimensions and shape of this kind of ship the acceleration forces could be smaller.

This is a very important point to be considered, especially if the type of reactor to be used was to be the German self-pressurized integral reactor FDR used to power the N.S OTTO HAHN.

In that type of reactor the spatial distribution of vapor voids within the core has to be considered very carefully since it affects not only K_{eff} but also the power distribution. Changes in the void formation are produced by the changes of the vertical acceleration due to the ship's motion on sea. The two main reasons for this are the following:

1. The pressure difference between the vapor dome and the core level decreases when the gravitational force is lowered. When the inlet temperature remains constant, this means a smaller local subcooling and therefore a general rise in bubble formation. This trend can be clearly seen in Figure 8.1.
2. The second effect also to be seen from Figure 8.1 consists in a redistribution of flow rate between various channels. In all channels the sum of the pressure drops, which includes the functional loss terms for density changes in the flow and the hydrostatic pressure, must be equal. Now the changes of hydrostatic pressure drop with g-forces are smaller in the boiling channels and therefore must be compensated for by changes in the functional losses. Thus, a boiling channel will get less coolant flow when gravitation decreases, and this leads to even more boiling.

It should be recalled here that on the OTTO HAHN even under very bad weather conditions, including wind force Beaufort 11 and full power against the sea, not more than 0.2g

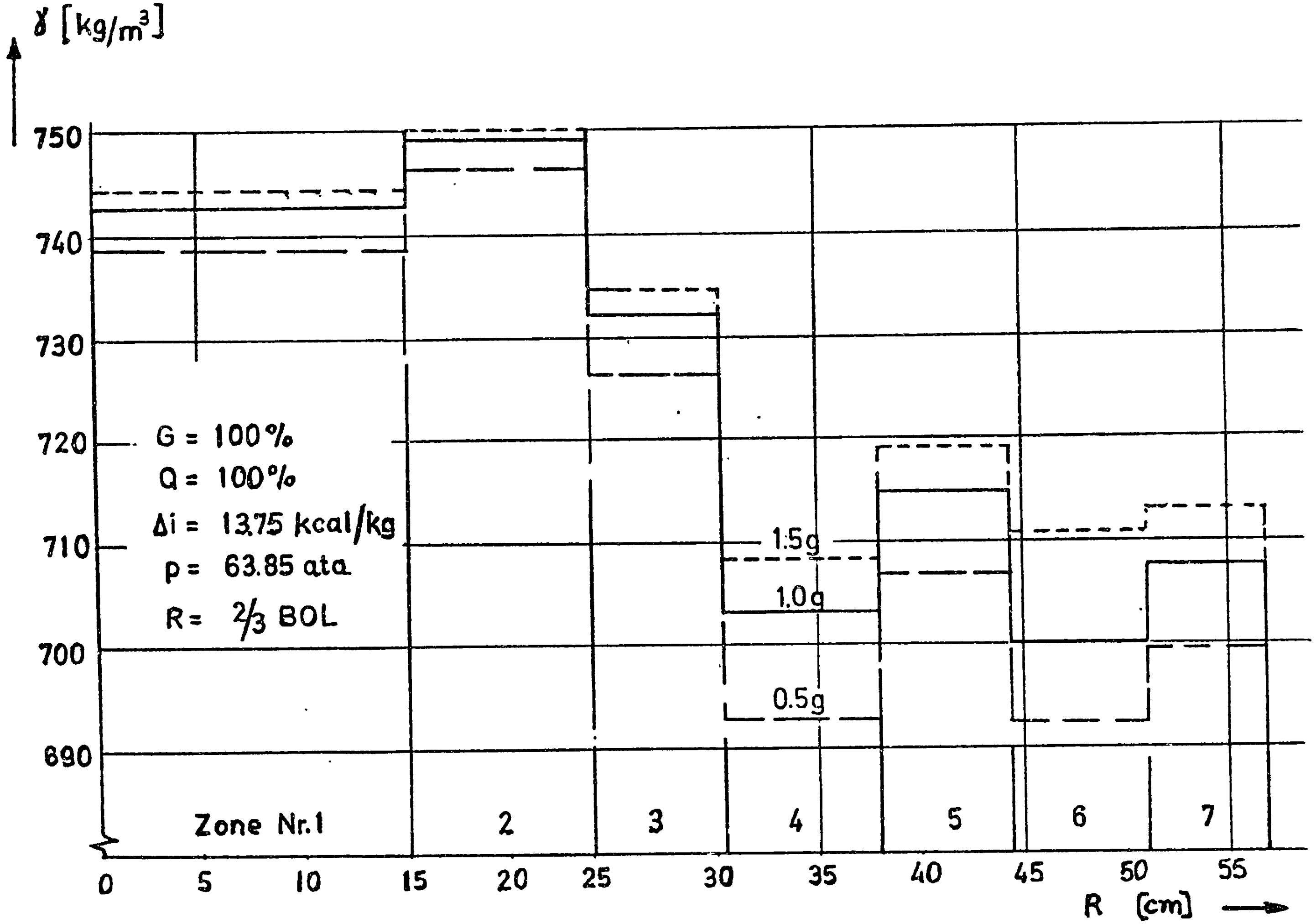


FIGURE 8.1 EFFECT OF VERTICAL ACCELERATION ON THE RADIAL DISTRIBUTION OF MEAN COOLANT DENSITY FOR A SELF-PRESSURIZED INTEGRAL PWR

(see Figure 4.3, Chapter IV) additional accelerations could be measured in the reactor site. This is to be contrasted with the design value of 1g recommended by the USCG and the U.S. Bureau of Shipping. The 1g value seems to provide an unnecessarily large safety margin and certainly adds substantially to the cost.

8.2.3 Improvement of the Ship Reactor Fuel Cycle

Nowadays, whereas long-life cores are needed for ship operation, they are not necessarily required for industrial usage where fuel economics is more important. That is true if the goal is to build research and development nuclear ships. If the intention is to build a competitive nuclear steam generator, the goal must be to balance the higher capital investment for the reactor with savings in fuel cost. Therefore, any new project must be with the incentive of optimizing the fuel cycle. This leads automatically to a high power density and a large mean burnup of the fuel. It is also possible to examine fuel enrichment associated with the use of borated coolant water.

It will be necessary to look at the refueling cycle to determine whether dry refueling should continue to be used, or whether a refueling canal system may be more appropriate, shortening the refueling time and improving safety.

8.3 Improvement on A Diversity of Approaches to Achieve More Efficient Nuclear Technology on Maritime Application

It is necessary to be alert to any opportunities for suitable applications of commercial-type nuclear engineering to ships. Such applications seem to be:

1. Small nuclear reactors for some industrial energy applications
2. Barge-mounted reactors
3. Nuclear-powered icebreakers and nuclear-powered Arctic oil drilling and supply ships.
4. Floating industrial port islands such as gas liquefaction, mineral extraction and other high energy using plants.

8.3.1 Small Nuclear Reactors for Industrial Energy Application

Up to now nuclear fuel has been used for production of electricity. However, in the U.S. electric power production only uses about 25% of the energy from all the fuel consumed. Clearly, if we can use nuclear fuel for other purposes more than electricity production, the role of nuclear-power would be greatly enhanced.

As is shown, the utilization of heat energy depends very much on the temperature at which it is made available, and at present there are three distinct temperatures ranges which are being considered with respect to future non-electrical use of nuclear energy. They may be approximated as follow, (a) low, 70 to 200 °F, (b) intermediate, 200 to 1000 °F, and (c)

(c) high, 1400 to 2000 °F. It is the intermediate range - 200 to 1000 °F that is the more interesting since it is the range that could be supplied by steam at conditions which can be produced with nuclear fuel.

In the present energy situation that is an enormous target of opportunity. But, the fact that modern nuclear plant produce large amount of steam, which greatly exceed even the largest requirements of single industrial plant; an approach to solving the problem of size conflict between nuclear steam supply and industrial demand, could be to build smaller plants such as needed for ship propulsion.

Up to now relatively small nuclear plants have been uneconomic and have received little consideration, however, that as a result of the increasing costs of alternative fuels, small plant such as the CNSG development by Babcock and Wilcox may become economic. This approach could become particular attractive if it proves practical to build factory-assembled, barge-mounted, or easily transportable units. It should be possible to deliver such units in a relatively short time span of perhaps five years, which is more compatible with the time span of industrial planning than the ten-year construction span of the present nuclear plants.

These small reactors could be used in the paper, chemical, petroleum refining, and related industries since this industries operate numerous plants requiring energy sources in the range of one million lb/hr of steam. This demand of individual user plants appear compatible with the power level (300 MWt) of one or perhaps several (for reliability's sake) small reactors. As was said before, the CNSG designed by Babcock and Wilcox - of 312 Mwt - for application in an ULCC is in the limit of size of the existent PWR integral reactor type.

There is a potential market for small reactors in both developed and developing countries although the approach is different in each one. Thus, in terms of potential applications of small reactors in developed countries there does not appear to be a substantial market for small electric power reactor since most of them have electric power grids sufficiently large to make the installation of large unit feasible. But for industrial energy applications especially in developed countries where, uranium, petroleum and coal option are not available, the market potential for small reactors would appear to be as good if not better, than in the United States.

Most of the developing countries have small electric power systems and some times the consumption centers are far away one to another which makes their interconnection expensive and difficult.

In many cases, present-day available commercial power reactors are larger than desirable for developing countries. Thus, there appears to be a potential market for small-to-medium reactors for electric power production in developing countries. A recent market survey of 55 developing countries made by the IAEA indicated a potential market in the 1980s of 140 units of 400 Mw(e). In my opinion, this market may be greater if:

(1) A small power reactor were developed that would be economically superior to other alternatives.

(2) Their technology could be transferred to the developing countries allowing them their own industrial development.

(3) A means of financing could be worked out.

(4) Pre-erected platforms supported reactors were developed.

However, these potential development of small reactors could be incorporated only after they have become a commercial success in developed countries for the following reasons:

(1) Historically, there are bad experiences in the developing countries with prototype development. The development of prototypes in developing countries generally prefer to use that country as experimentation fields resting although on it part of the cost of research and development, without taking care in the real necessity of that country. Thus, the experience up to date is that reactor systems could be sold outside the producing country only after they were sold domestically.

(2) Practical, first units tend to require modifications before they work properly, and this can best be done where more technical experience is available.

(3) Financing, the "proveness" criterion used by international lending institutions would seem to dictate that new technologies be implemented in the developed world first.

Nuclear Heat for Desalination

The integrated pressurized reactor could be also very suitable for generation of nuclear heat in desalination plants. Since the required temperature level for desalinating sea water is low (about 150°C), the primary system pressure of the reactor can be decreased , whenever the steam should be used only for that purpose. In such a case, the pressure decrease should result in a corresponding decrease of the pressure vessel weight and cost.

Mostly, a reactor plant for such application could feed a turboalternator, which supply electric energy to the reactor and desalination plants and surrounding (see Figure 8.2). Thus, the steam generated in the heat exchanger of the reactor should be flowing to the back pressure turbine of a turboalternator. The exhaust steam of the turbine is flowing then to the brine heater where the seawater circulating through a thermal desalination plant is heated up. The heating steam drain is pumped from the brine heater to the reactor feedwater tank. The feedwater is pumped back then to the steam generator of the reactor by means of a normal feedwater pump. The temperature of the brine heater and of the feedwater tank will be controlled by live steam.

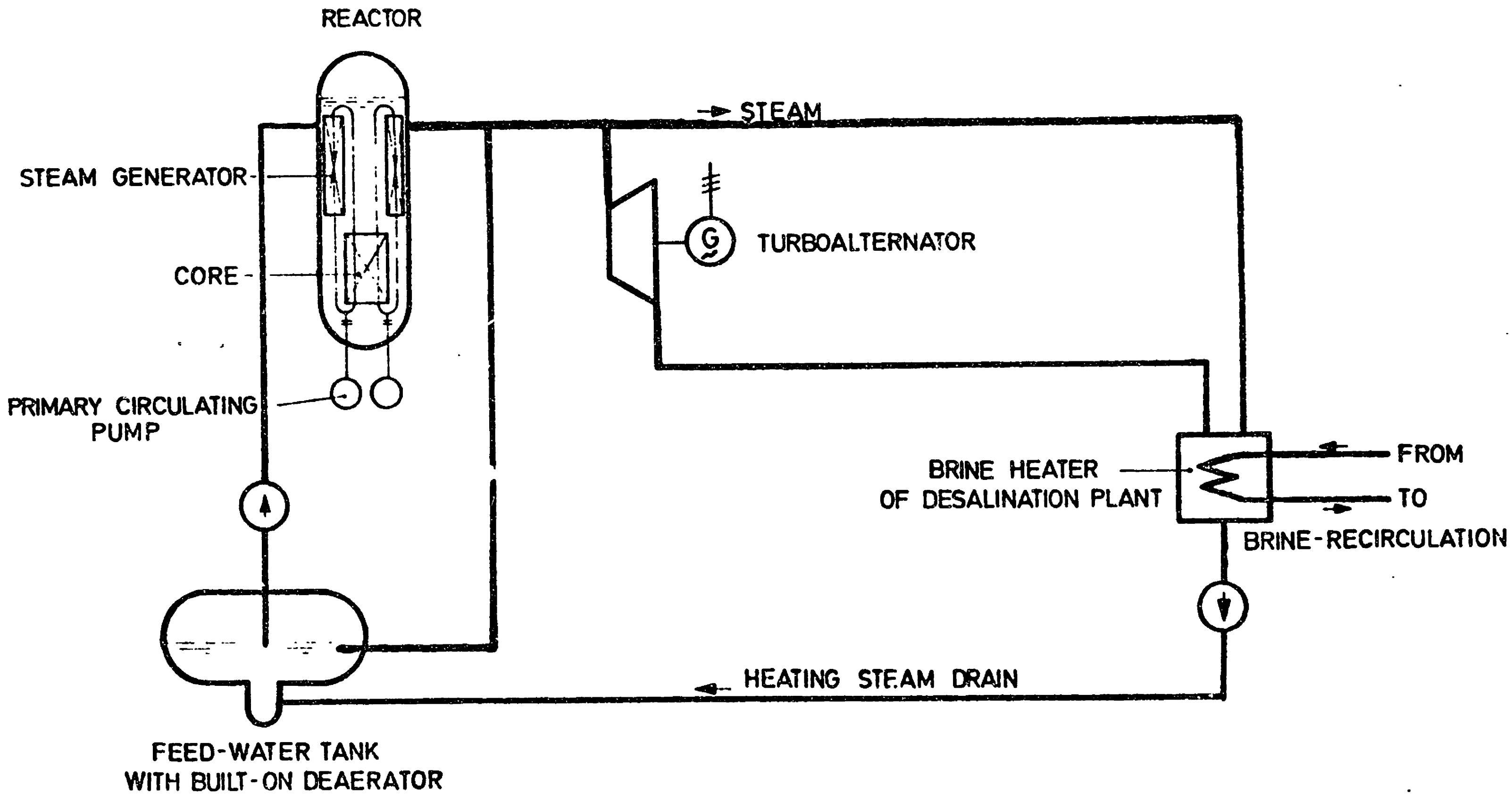


FIGURE 8.2 DIAGRAM OF A NUCLEAR DESALINATION PLANT

Depending on the efficiency of the plant, the specific heat needed by a desalination plant is between 50 and 100 kcal per kg destilled water. Thus, for a desalination rate of 10,000 m^3/day, should be necessary a reactor plant and machinery with a thermal output between 24 to 48 MW. This power can be supply by a reactor plant like that used to power nuclear ships.

As local conditions are very different, a thorough study of all influences and assumptions is necessary to obtain an optimal design and to predict the break-even point for economical application of nuclear heat.

8.3.2 Barge-Mounted Reactors

The offshore nuclear generating concept is a very interesting intermediate step between stationary and shipboard reactor nuclear applications. Its behavior with respect to safety and environmental factors can offer a good experience to the shipboard reactor application. This concept is favored over land-based reactor plant by the following major attributes:

1. Because the plant is offshore, it is surrounded by the sea which will have no resident population dose during the life of the plant. This attribute permits siting the plant nearer areas of high population density than a comperable land-based plant, since because of their large population, these regions have both high demands for electric power or steam

supply and a shortage of viable new power plants sites.

2. The plant can be of a standard design and built by one manufacturer. This gives the manufacturere a maximum control of plant design variables, and minimizes the possibility that as-built systems deviate from the design. To obtain the benefits of standardization the plant design must be made substantially independent of the site characteristics. Therefore, the effort required for review of the design by regulatory agencies should be reduced compared to current land-based plants where the site characteristics play an important role.
3. Because of its offshore siting, the plant is predicted to have significantly less effect on the environment than a comparable land-based plant, and if that should occur, the whole barge could be moved to another site.
4. The barge-mounted plant offers the possibility to be moved to other sites either if along the time for some reason it could affect the environment or if it becomes uneconomic in that site. This aspect could allow the building of small reactors barge mounted and moving them according to the circumstances.

5. Because the plant floats, is standarized, and is manufactured away from the site, the concepts lends itself to separate licensing of the plant and the site. The plant could be decoupled from the earth in the event of earthquakes.

The development of barge-mounted concept should help the future design of shipboard nuclear reactor since:

1. It is a totally integrated nuclear power station mounted on a floating platform. It will be constructed in a shipyard like a ship. Following a compressive testing, at the shipyard like facility, the plant should be ready to be towed to the offshore site for fuel loading and power generation. The only connection on shore should be through underwater transmission cables to the electrical network.
2. The plant should be built on a repetitive basis in a new shipyard-like facility. It is intended to make extensive use of the shipyard technique of assembling the plant and utilizing ship fabrication methods for both structure and equipment, and therefore, can be expected that a savings in plant construction time will result.
3. Because the plant is a standarized plant intended for use at many sites, it is designed to withstand levels of natural phenomena such as sea motion (see Table 8.1) or ship collision. Therefore, the whole

TABLE 8.1

PLANT ACCELERATIONS DUE TO RIGID BODY PLATFORM MOTION
OF A PLATFORM MOUNTED NUCLEAR PLANT

	Towing	Operation	Seismic DBE	Tornado
Maximum Acceleration in plane parallel to platform (1) (3)				
Longitudinal = El 40'	0.20 g	0.02 g	.08 g	.08 g
(pitch) El 140'	0.40 g	0.03 g	.08 g	.10 g
Transverse = El 40'	0.20 g	0.02 g	.08 g	.08 g
(roll) El 140'	0.40 g	0.03 g	.08 g	.10 g
Maximum Acceleration normal to platform (2) (3)				
Heave	0.10	.003 g	0.20 g (4)	.006g
Pitch - Longitudinal distance of 100 ft from transverse center line	0.20 g	.01 g	(4)	.03 g
Roll - Transverse distance of 100 ft from longitudinal center line	0.20 g	.01 g	(4)	.03 g

experience accumulated could be transmitted to the shipboard reactors.

8.3.3 Nuclear-Powered Icebreakers & Nuclear-Powered Arctic Oil Drilling & Supply Ship

The fact that shipowners are unlikely to invest in nuclear ships means that they are not, at present, clearly economically superior or there are still too many uncertainties and risks. However, two applications, nuclear-powered icebreakers and nuclear-powered arctic oil drilling, seems to be promising in the near-term. In both these applications, one governmental and the other commercial, high horsepowers are required, fossil fuel supplies are excessively expensive, and long periods of operation without refueling are desirable.

The oil-drilling application should be a straight forward commercial operation, requiring only from the government the encouragement and help on licensing insurance, etc.

The icebreaker, of course, would be a ship built for government account. If that kind of ship were developed, it will give a lot of experience for further merchant nuclear ship technological design.

8.4 Improvement of the Time Necessary to Deliver a Small Nuclear Plant

In setting up any program for nuclear utilization and supplying energy to industry, it is certainly necessary to take advantage of the experience of the utility industry in utilizing nuclear power. One of the lessons to be learned is the very significant effect of time-related costs on nuclear plant capital cost. These are illustrated in Figure 8.3.

The time related cost shown in 1973 were only about 20% of the total plant cost. For the 1981 date these amount to as much as 40% of the total plant costs. This figure illustrates the necessity for achieving foreshortened schedules for industrial application. It is necessary to use the economy of time to offset the economy of scale which is working against the small reactor economics. For the economics of ship reactor, it is necessary to incorporate this concept . Technologically, it will be possible examining such schedule as utilization of shop labor whenever possible; the use of concurrent construction of ships and barge-mounted plants, at the shipyard; and the use of prefabrication and modularized design for land-based reactors.

Another way is reducing the licensing requirements. It can be possible if a "generic" license to manufacture floating plants - barge-mounted or ship nuclear plant -, and balance of plant licenses for land-based nuclear plants are obtained.

Preordering of material with appropriate insurance or liability commitments to cover losses for such preordering is another obvious way of shortening total plant construction times.

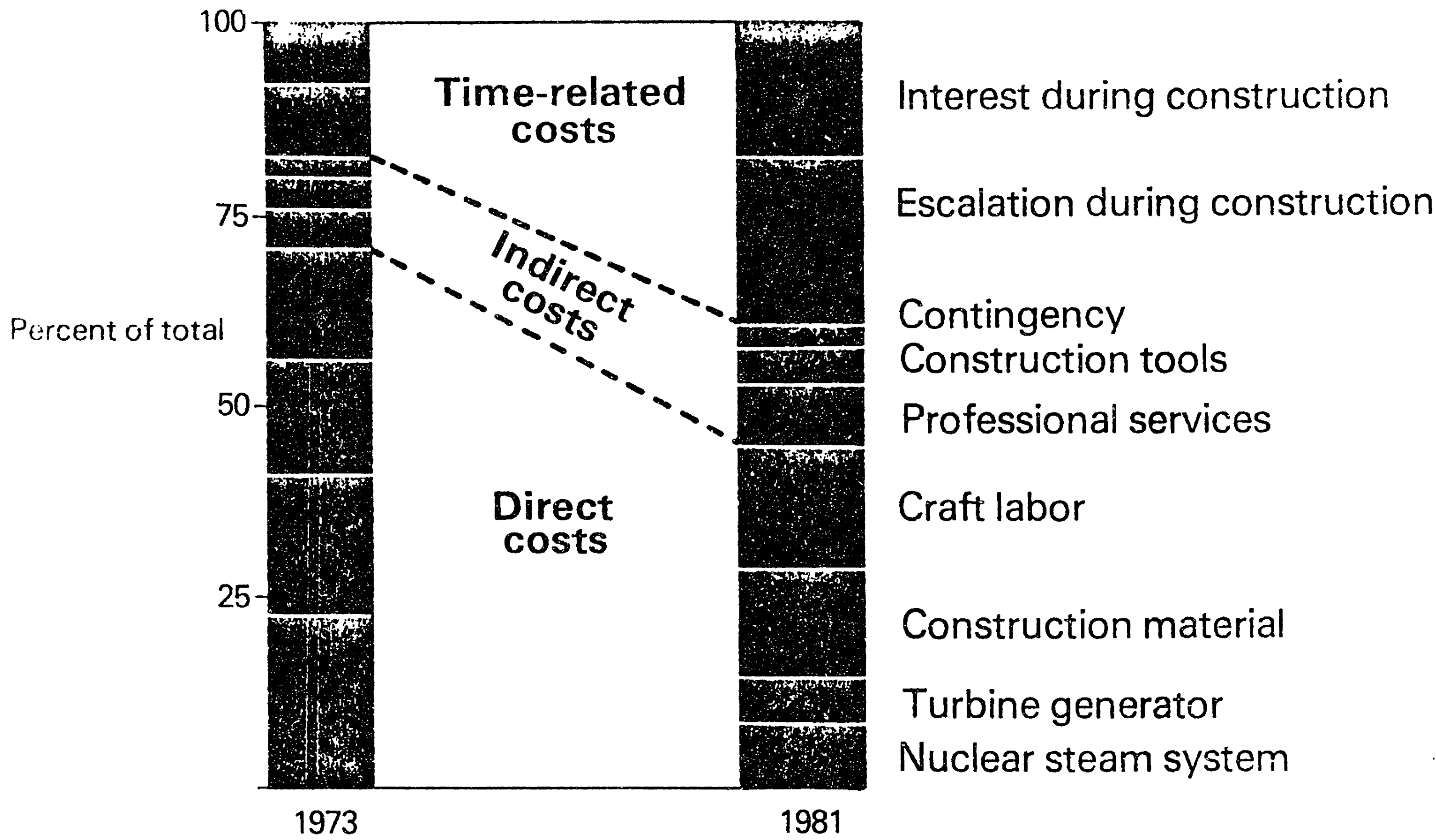


FIGURE 8.3 TIME - RELATED COST IN NUCLEAR PLANT CAPITAL COST

CHAPTER IX

SUMMARY, CONCLUSION & RECOMMENDATIONS

Summary

Although studies have been carried out on many types of reactors for maritime propulsion, the PWR has been chosen for use in all nuclear powered merchant ships and naval so far built or being built.

Since for commercial applications of maritime reactor, it seem likely that any built nuclear propelled ship would use this type of reactor, only characteristics of PWR's type have been considered to establish the technological differences between stationary and maritime reactors.

The present state of technology and experience suggest the integral PWR type as the most suitable for maritime application and the loop PWR type for central power stations. Using this criterion, the most important technological differences between the stationary and the maritime nuclear plant are summarized in the table at the end of the chapter.

Conclusions

1. Although the whole cycle and purpose of a land-based reactor installation does not differ from that of a maritime reactor installation , the use of reactors on shipboard demands additional safeguards and requirements, a fact which makes some of its parts more technically sophisticated. This is

largely because of the following considerations:

a. A ship moves in all directions, degrees of freedom due to sea motions.

b. There is a limited space and weight available for the propulsion plant.

c. A ship must fulfill stability requirements.

d. The unit generated power in a maritime propulsion plant is much smaller than in stationary plant.

e. Reliability.

2. Great technological improvements have been achieved from the first shipboard reactor until the more recently designed. Some of these improvements are:

a. Change from loop to integral reactor type, to obtain more safety and a reduction of volume and weight.

b. The use of zircaloy instead of stainless steel in fuel rod clad and reactor internal increased the specific power density.

c. The improvement of the plant efficiency by an increase of the secondary steam pressure and temperature.

d. The simplification and volume reducing of the auxiliary system.

e. The reduction of design pressure of the safety containment resulting in smaller wall thickness, due to the use of pressure-suppression containment instead of dry-containment system.

Recommendations

1. It is still necessary to obtain a number of essential improvements to achieve a reduction in fuel and capital investment cost as well as in a reduction of weight and space requirements. These improvement could be:

   a. To reduce as much as possible the technological differences between stationary, barge-mounted and shipboard integral reactor type according to their own requirements, so that a standard design for a wide range of applicability would lead to an economical repetitive production. A possible way to achieve these can be the following:

      1. Studying the power necessary in reactors for the three different applications, stationary, barge-mounted or maritime and choose that whose fickleness would be suitable and with the least modifications for each application.

      2. That reactor seems to be an integral PWR of 300 to 350 Mw(t). This reactor can supply the required energy to propel a container ship, or an ultra large crude carrier and to supply both electrical and thermal energy for an industrial plant or a barge-mounted plant.

3. That typical reactor will allow mass production and standardization of steam generators, pressurizers, pressure vessel, primary collant pumps, control rods, control rods drives, etc.

b. Reduction of the ship outage time required for refueling, inspection and dry docking.

Inspection and dry docking schedules should coincide with the ship's fuel cycle. Thus, when the ship is at the facility it should allow that inspection and other scheduled calibrations or repairs can proceed in parallel with fuel refueling. It will be important that while the nuclear merchant ship programs go on, discussions with regulatory agencies should be held to develop requirements for ship and methods to eliminate duplication of inspection and resulting scheduling problems.

Among downtimes for inspections, dry docking and refueling the largest time spent is due to refueling. Therefore, the major attention should be focused on refueling efficiency. Since until this moment, the longer steps in refueling operation are due to: control rod drive disconnection and connection, stud detensioning, fuel transfer, tension reactor vessel stud, couple control rod drive to extension shafts, control area clearup and miscellaneous equipment removal. Then some reduction of refueling time could be obtained as follow:

1. Using methods such as, multiple stud tensioner or breech lock type closure device, which could reduce the time for removal and installation of the reactor pressure vessel head which are always critical-path items.
2. If control rod drive mechanism electrical connection junction boxes and cooling water header connections could be installed at the upper end of the drive structure and identified by mechanism number, the time required to disconnect and connect these mechanisms could be reduced.
3. To reduce the time required to hook-up and release the cask during the fuel transfer operation, fuel transfer and internals handlings casks should be fitted with trunions and lifting beam devices.
4. In dry and shuffle refueling, at least two transfer containers would be used to move and load new fuel from the storage area to the spent fuel pool or the reactor. Thus, while one empty cask removes one spent assembly from the inner region of the core

transferring it to the spent fuel pool, the other cask could remove one outer region assembly and transfer it reloading the vacant slot in the inner core. If such handling can be done, the fuel transfer time (the longest in the refueling time) could be substantially reduced. In N.S. SAVANNAH, refueling with only one cask was used.

5. All equipment to be used over the opened reactor vessel should be attachable by a lanyard to the user or a fixed point to preclude the possibility of dropping the object into the vessel spoiling some reactor part, delaying its recuperation and extending the fuel transfer time.

c. The use of boric acid for long-term reactivity control, and a zonal insertion and reshuffling of fuel elements to achieve a better burnup rate and specific power density in the reactor. It appears that boric acid in the coolant could permit to remove enough control rods and control rods drive mechanism so that borated coolant could be cost effective. By so doing, the smaller diameter control rod drives now in use on ship reactor could be substituted by the more conventional stationary control rod drive structure. The use of borated coolant could lower the fuel enrichment and change the fuel cycle duration. Through the former suggestions a better burnup rate will be achieved.

It is possible that a multi-zone "shuffle" fuel management program of the type widely employed today in land-based power reactor applications will probably be economically superior to the proposed single-zone batch program used for maritime reactor application.

2. The integral PWR plant onboard of N.S."OTTO HAHN" in more than seven years of operation,just as the Babcock and Wilcox study of CNSG,has proven that this reactor is very suited for shipboard use through its excellent operating characteristics, its ease of operation and its high availability. However, although they have exceeded the expectations of designers and operators, shipowners are still unlikely to invest in nuclear ships. This proves that the application of reactors on shipboard presently is not only a matter of technological improvement, but a problem of a combination of assumed uncertainties and risks which are not technological, and the fact that for many applications it is not clearly superior economically. These problems are among others; port entry restrictions, emotional public reactions, licensing time delays,

large and uncertain financing and insurance, labor requirements, time from decision to in service, etc.

In my opinion the most suitable way to overcome the whole problem of the nuclear merchant ship program is by a collaborative program between countries which have the capability to build nuclear ships. Developed or developing countries interested in such program could join This collaborative program could have the following advantages:

a. Several countries making a joint approach to safety and operating procedures are likely to have more influence on the international acceptability of nuclear powered ships and the rules governing their operation than if they acted independently.

b. Commercial small reactors along with commercial lead ship would be available several years earlier.

c. The share of common technology would reduce cost and through standardized design and larger markets could improve sales prospects, performing so successfully the recommendation 1.a.

3. It would be advisable to perform an integrated analysis (such as the Wash-1400 Rasmussen report for stationary nuclear reactors) of nuclear powered maritime accident risks. Up to date these accidents

have only been studied separately. Such a study would yield an appropriate evaluation of the consequences for each of the probable accident occurrences as well as an overall estimate of the nuclear maritime risks.

MAJOR TECHNOLOGICAL DIFFERENCES BETWEEN

STATIONARY & MARITIME REACTORS

System	Subsystem	M A R I T I M E	Ref.	S T A T I O N A R Y	Ref.
Reactivity Control	Control Rod	Only one kind for both purpose regulating & Safety	5.4.1.1 7.1.2	Two kinds: 1- Part length for partial regulation 2- Full length for regulation & safety	5.4.1.2 7.1.3
		High reactivity incremental worth due to no use of soluble poison		Less reactivity incremental worth due to the use of soluble poison	
		Need reactivity excess to override transient xenon	5.4.2		
		No need xenon spatial oscillation control	7.1.5	Need xenon spatial oscillation control	7.1.5
	Fixed Burnable Poison	Burnable poison rod assembly & lumped burnable poison rod for the whole reactor life	5.4.1.1 7.1.2	Use burnable poison rod assembly only in the first cycle	7.1.3
	Chemical Shim	Cannot be used for regular reactivity control	5.4.1.1 7.1.2	Used as long-term reactivity control	7.1.3
	Control Rod Drive Mechanisms	Only separating roller-nut type is used	3.2 3.4 7.1.4.1	Magnetic jack type for full-length control rod	3.3.1
				Roller-nut type for part-length control rod	7.1.4.2 3.3.2

Continuation of Table - MAJOR TECHNOLOGICAL DIFFERENCES BETWEEN STATIONARY & MARITIME REACTORS

System	Subsystem	MARITIME	Ref.	STATIONARY	Ref.
		Need a safety powered insertion system for major ship accident	4.3.2	Do not need that system	
Structure Support	Containment pressure vessel, steam generator, pumps, etc.	Support designed against static & dynamic load (see Fig. 4.9 & 4.10)	4.3.3 4.3.5	Support against static load (see Fig. 4.11, 4.12, 4.13)	4.3.3
		Component & foundation must be non-resonant with propeller-induced frequency	4.4.2	No external-induced vibrations	
	Piping & Valves	Snubbing-type support for isolation from dynamic load & vibration	4.3.6	No special support	
Protection	Reactor Protection	The system must be able to drop control rod at any ship angle.	5.3.1		
	Auxiliary Propulsion	Need a take-home propulsion plant	5.3.2		
	Collision Protection	Need a heavy steel structure between hull & reactor	5.5.1	Containment strength is enough to resist collision	
	Grounding Protection	Need special reactor bottom support	5.5.2		

Continuation of Table - MAJOR TECHNOLOGICAL DIFFERENCES BETWEEN STATIONARY & MARITIME REACTORS

System	Subsystem	MARITIME	Ref.	STATIONARY	Ref.
	Radiation Protection	Must use light material for shielding criteria: optimum shielding with minimum weight	6.5.1 6.5.3	Use preferably concrete criteria: Optimum shielding with minimum cost	6.5.5
Refueling	Facilities	Spent fuel stored outside of the plant could be permanent shore-based or mobile	6.4.1 6.4.1	Spent fuel stored besides of plant Permanent shore-based	
	Technique	Dry-type fuel removal	6.4.2	Wet-type fuel removal	6.4.2
Reactor Control	Basic Control Variable	Use temperature error as regular signal & difference between turbine load request & core generated power as prompt signal	3.2	Use only an average coolant temperature as signal	3.2
		Controlled manually at power level less than 20% total power	3.2	Controlled manually at power level less than 15% total power	3.2
	Instrumentation	Need sensors for ship motion	4.3.8		
Containment		Pressure-suppression system	6.3 7.2.2	Dry containment system	6.3
		Automatic flooding valves	5	Do not need flooding valves	

Continuation of Table - MAJOR TECHNOLOGICAL DIFFERENCES BETWEEN STATIONARY & MARITIME REACTORS

System	Subsystem	M A R I T I M E	Ref.	S T A T I O N A R Y	Ref.
	Heat Removal	Is provided by: 1) Containment cooling 2) Suppression pool & emergency spray system 3) Decay heat removal system	6.3	Is provided by: 1) Fan cooler system 2) Separate spray system 3) Ice condenser system	6.3
Heat Interchange	Steam Generators	Located inside of the pressure vessel 12 modules straight-tube steam generator	4.3.1	Outside of the pressure vessel Two U type-tube steam generator	4.3.1
	Pumps	Primary coolant pumps mounted in pressure vessel. No primary coolant piping		Primary coolant pump mounted separate of pressure vessel & connected by primary coolant pipe	

BIBLIOGRAPHY

1. Analysis of Past, Present & Future Applications of Nuclear Power for Propulsion of Marine Vehicles, M.I.T., Bauman, 1972.

2. Meeting on Small Nuclear Reactors for Industrial Energy, Oak Ridge, Tennessee: July, 1975.

3. International Symposium on Nuclear Ships, Madrid: April, 1972.

4. Proceeding of the Symposium on Nuclear Ships, Hamburg: 1971, Vol. I & II.

5. A Forum Report: The Role of Nuclear Propulsion in Merchant Shipping. New York: 1960.

6. Small & Medium Power Reactors, I.A.E.A, Oslo: October 1970.

7. Report on Nuclear Ship Study. Department of Trade & Industry, London: 1971 & 75.

8. Preliminary Safety Analysis Report of CNSG, Vol, I - IV. National Nuclear Ship Program, U. S. Maritime Administration, 1975.

9. Systems Summary of a Westinghouse PWR Nuclear Power Plant, G.C. Masche, 1971.

10. N.S. Savannah Safety Assessment - Revision II, 1967.

11. Preliminary Design Report of Platform Mounted Nuclear Plant, Vol I & II.

12. Nuclear Merchant Ships. Maritime Transportation Research Board. National Academy of Sciences, 1974.

13. Introduction to Nuclear Merchant Ship Refueling Facilities & Manning Requirements. U.S. Maritime Administration, 1973.

14. Design Data on Motion in Random Seas of Series 60 Ships. L. Vassilopoulos, M.I.T.

15. Ship Design & Construction. A. D'Arcangelo, SNAME.

16. On the Collision Protection of Ships. Norman Jones, Report 75-3, M.I.T.

17. Nuclear Ship Propulsion. Symposium held at TAORMIWA, Italy, 1960.

18. Principles & Standards of Reactor Safety, I.A.E.A. 1973.

19. PSAR, Vol III, PILGRIM, N.P.S.

20. PSAR, Vol III, NUCLEAR UNIT DIABLO CANYON

21. Reference Safety Analysis Report. Westinghouse Nuclear Energy System.

22. Marine Nuclear Steam Propulsion, Executive Briefing.

23. Nuclear Chemical Engineering - Prof. Benedict.

24. The Core Design of the FDR & EFDR by D. Bunemann et al. Symposium of Nuclear Ships on Rio de Janeiro. Brazil: 1972

25. Technical Operation Experience of the N.S.'OTTO HAHN' & Problems of Port Entry," D. Ulken et al. Symposium on Nuclear Ships,1972, Rio de Janeiro, Brazil.

26. "Integrated Pressurized Water Reactors as Energy Sources for Various Purposes" H.T. Hademann et al. Symposium on Nuclear Ships,1972, Rio de Janeiro, Brazil.

27. Performance of the 1st Core of the "OTTO HAHN". M. Koib et al. Symposium on Nuclear Ships,1972, Rio de Janeiro, Brazil.

28. Maritime Carriage of Nuclear Materials. I.A.E.A., Stockholm: June 1972.

29. Some Aspects of Nuclear Safety & Ship Design by G.L. West et al. SNAME Transaction, Vol. 75 , 1967.

30. The Economics of the Containership Subsystem. D.S. Miller, University of Michigan, No. 03, 1968.

31. Measure of Merit for Ship Design. H. Beuford. University of Michigan, No. 007, 1968.

32. Fundamentals of Ship Design Economics. H. Beuford. University of Michigan, No. 086, 1970.

33. Nuclear Merchant Ship Support Facility - Concepts, U.S. Maritime Administration, 1974.

34. Nuclear Merchant Ship Support Facility - Utilization. U.S. Maritime Administration, 1974.

35. Nuclear Merchant Ship Support Facility - Siting U. S. Maritime Administration, 1973.

36. Nuclear Merchant Ship Operations, Experiences & Recommendations. U.S. Maritime Administration, 1973.

37. Technical, Operational & Economic Report on the N.S. Savannah. 4th Period of Experimental Commercial Operation. U. S. Maritime Administration, 1968-1969.

38. Proceedings of the Conference on Nuclear Marine Propulsion held at Virginia Polytechnic Institute: 1964.

39. Nuclear Capital Costs & Cost Trends in Maritime Application. NUS - 278 A, 1966.

40. The Economics & Influences of Nuclear Refueling in Maritime Applications. NUS - 295 A, 1966.

41. The Economics of Nuclear Propulsion for Maritime Applications, NUS - 339, 1967.

42. The Economics of Nuclear Fuel in Maritime Applications, NUS -265 A.

43. Maritime Nuclear Power for High Speed Service. U. S. Maritme Administration, 1967.