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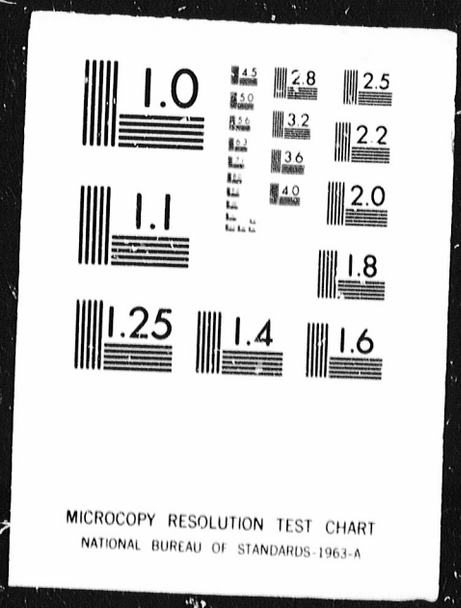

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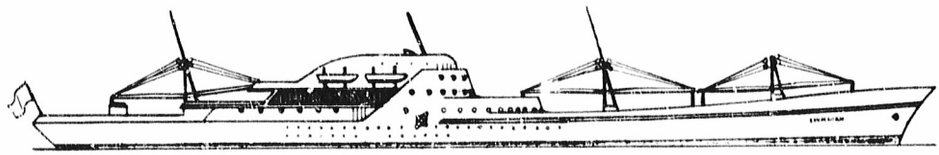
N. S. SAVANNAH SAFETY ASSESSMENT

VOLUME IV

ANALYSES OF HYPOTHETICAL ACCIDENTS



This volume summarizes the analyses of representative hypothetical marine reactor accidents to the Nuclear Ship Savannah in relation to her safety features. The first volume of this assessment described the engineering and construction of the Savannah. Subsequent volumes will include details of operation, maintenance, and environmental factors.

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NOVEMBER 1960

Prepared for
The Atomic Energy Commission
and
The Maritime Administration

by
The Babcock & Wilcox Company
Atomic Energy Division

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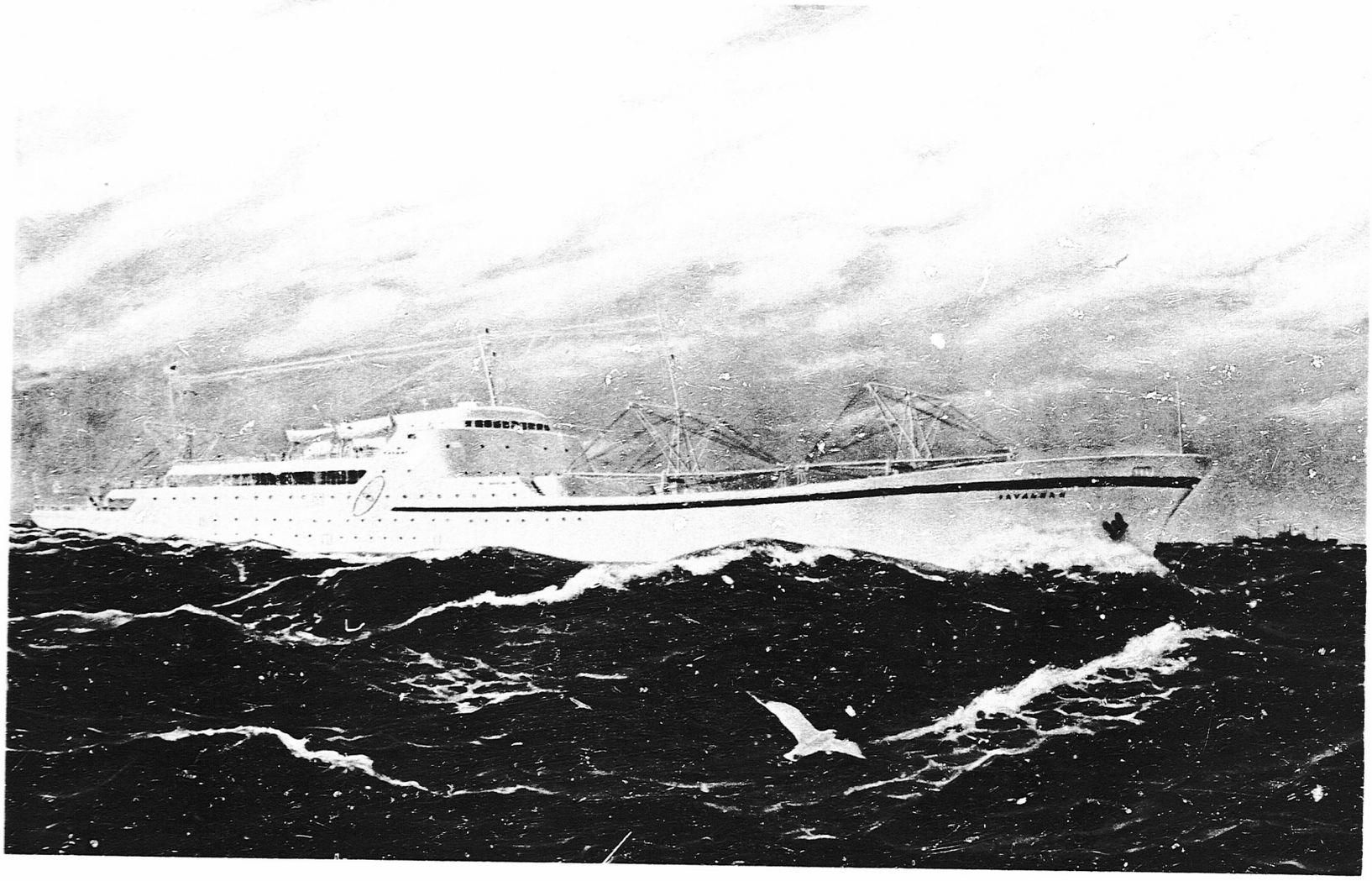

THE NUCLEAR SHIP SAVANNAH

FOREWORD

This volume of the NS SAVANNAH Safety Assessment analyzes representative hypothetical marine and reactor accidents in relation to the safety features of the power plant and the ship. Discussion of these accidents, with their causes and consequences, is presented for careful consideration by responsible persons in the maritime and nuclear field. It is especially directed to those members of regulatory bodies immediately concerned with review of the SAVANNAH's design and its operation in territorial waters and harbor areas.

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TECHNICAL ABBREVIATIONS

Btu/hr-ft-°F	British thermal unit per hour - foot - degree Farenheit
cc/hr	cubic centimeter per hour
cu ft	cubic feet
cu ft/ton	cubic feet per ton
dm	decimeter
dwt	dead weight
ft	foot
g	acceleration of gravity
gpd	gallons per day
gpm	gallons per minute
hp	horsepower
in.	inch
kg	kilogram
kw	kilowatt
lb	pound
lb/hr	pound per hour
μc/cc	microcurie per cubic centimeter
mr	milliroentgen
mr/hr	milliroentgen per hour
mrem	milliroentgen equivalent man
MW	megawatt
psi	pounds per square inch
psia	pounds per square inch (absolute)
psig	pounds per square inch (gage)
r	roentgen
r/hr	roentgen per hour
rem	roentgen equivalent man
shp	shaft horsepower

PART ONE

POWER PLANT ACCIDENTS

Chapter I

REACTIVITY ACCIDENTS

A. Startup Accident
B. Rod Withdrawal in the Power Range
C. Cold Water Accidents
D. Xenon Burnout Transient

I. REACTIVITY ACCIDENTS

A. STARTUP ACCIDENT

1. Nature of the Accident

The purpose of reactor startup is to bring a relatively cold, subcritical reactor to the critical or slightly supercritical condition, thereafter increasing the power level in a controlled manner until the desired power level and system temperature are attained. This is accomplished by the withdrawal of control rods. If, as the result of some malfunction, a rod withdrawal should continue through the prompt-critical stage, the rate of change of power will rapidly exceed the normally desired value. This situation might conceivably develop as a result of operator error, nuclear instrumentation failure, or control rod circuitry malfunction. Operating procedures as well as instrumentation and circuitry are designed to prevent such an occurrence. Nevertheless, it is the purpose of this discussion to examine the consequences of such an accident in the very unlikely event that it should happen.

2. Safety Circuitry

Protection of the core from damage during this transient is provided by three independent safety system features. As the rods are withdrawn, the rate of change of reactor power may increase. When the rate of change exceeds one decade per minute (period = 26 seconds) the safety system causes all control rods that are not fully inserted in the core to be driven into the core at the maximum rate available with the mechanical drive (fast insertion). This automatic fast insertion overrides all withdrawal signals, but the fast insertion action is automatically halted when the rate of change of power is reduced below one-half decade per minute. If the rate of change of reactor power increases to three decades per minute (period = 8.7 seconds), the

safety system initiates a scram, that is, all the control rods are driven into the core via hydraulic pressure. A third safety system circuit scrams the control rods if the reactor neutron level reaches an overpower trip level. With the Start-Run switch in the Start position, the trip level is set at about 25% of maximum power; in the Run position, the overpower trip level is 130% of maximum power.

3. Accident Study Parameters

The startup accident was analyzed on an electronic analog computer at Battelle Memorial Institute, Columbus, Ohio. It was found that the two most important parameters were the rate of reactivity addition and, in the power range, the value of the Doppler coefficient of reactivity.

The rate at which excess reactivity can be added to the reactor by withdrawing control rods is dependent upon the net worth of the rods, the vertical position of the rods in the core, and the withdrawal velocity. During a normal startup the operator may withdraw either a single rod or a group of rods at a time. The design of the circuitry prevents the withdrawal of more than five control rods simultaneously. Based on analytical studies and experimental measurements of control rod worths the total reactivity of the five most reactive control rods is estimated to be less than 5%. Due to the axial variation of thermal neutron flux, the withdrawal of control rods is expected to produce the greatest reactivity addition rate as the bottom of the absorber sections pass near the core midplane. To account for this increased addition rate at the middle of the stroke, the average reactivity insertion rate based upon the over-all stroke was doubled. The analyses of the accident are based upon a specified maximum withdrawal rate of 15 inches per minute limited by the synchronous speed of the drive. This results in a maximum potential reactivity insertion rate of 4×10^{-4} Δk/second for five rods at the position of greatest importance.

The simulation of the accident was divided into two parts: Below the power range the heat output of the reactor is negligible and hence negative reactivity effects due to changes in fuel or moderator

temperature are not appreciable. Consequently, all reactivity additions due to rod withdrawal directly affect reactor neutron level without any compensating reactivity effects. Conversely, in the power range, any increase in neutron level is reflected by a change in [illegible] and moderator temperatures. The negative reactivity effect associated with these increased temperatures tends to offset the positive reactivity inserted as a result of rod withdrawal, thereby resulting in a lower net addition of reactivity and slower reactor power transients.

4. Results of the Study

The results of the analog study are shown in Figures 1 and 2.

Figure 1 shows the result of rod withdrawal in the low power region. It is assumed that the reactor was initially subcritical by 3% ($k_{eff} = 0.97$) and that the neutron flux was 13 decades, a factor of 10^{13} below the neutron flux level at maximum power. As explained above, the maximum rate of continuous reactivity insertion was conservatively estimated to be 4×10^{-4} Δk/second and was assumed to take place throughout the entire transient. It can be seen from Figure 1 that the reactor would reach criticality ($k_{eff} = 1.0$) about 73 seconds after initiation of rod withdrawal. In the analysis it was assumed that the safety system would trip and initiate a fast insertion of all withdrawn control rods when the nuclear instrumentation for the startup range indicated a reactor period of 10 seconds. Figure 1 shows that the reactor power rise would be halted at a power less than one decade above the power at this trip point. If, for some unaccountable reason, a fast insertion should not occur, the reactor period would continue to decrease until it reached a value of 3 seconds, the assumed reactor scram trip level. A delay time of 240 milliseconds between the indication of the 3-second period by the nuclear instrumentation and the initial inward rod motion was incorporated in the reactor simulation. It should be noted that the period trips assumed in the analysis are far shorter than the periods actually used for the SAVANNAH. Therefore, should a startup accident occur on the SAVANNAH, the power excursion would be less than is indicated by this accident analysis. In addition to the delay in the safety system and control rod drive hydraulic scram system,

a time constant of 5 seconds in the period circuitry differentiator is used as a representative value for the low level range. As indicated in Figure 1, the power excursion was terminated with either a 3- or 1-second period trip level, but the power excursion in the latter case would be three decades larger than in the former.

The inherent safety of low enrichment uranium oxide reactors is demonstrated by continuing the power excursion during a startup accident into the power range. This power excursion is presented in Figure 2 for the case of no safety action following a continuous rod withdrawal from 3% subcritical at a reactivity rate of 4×10^{-4} Δk/second. Should the reactor power reach the MW level, the temperature of the oxide fuel would begin to increase with a resultant increase in characteristic U-238 resonance absorption. This negative reactivity effect (Doppler effect) is strong enough to limit the excursion to an instantaneous peak power of less than 400 MW. The negative temperature reactivity effects of the moderator are negligible in this particular case because the duration of the excursion is so short that essentially no heat is transferred out of the fuel rods. Due to the very short duration of the power peak, the total energy added to the core is small enough so that the fuel central temperature in the hot channel after 95 seconds withdrawal is only 900°F. This may be compared with a maximum fuel temperature of about 3800°F at normal full power. The fuel temperature at this time would still be increasing because it is lower than that corresponding to equilibrium conditions at the instantaneous power value. The peak power reached during the excursion is strongly dependent upon the value of the Doppler coefficient. If the value of the Doppler coefficient is -1×10^{-5} $\Delta k/°F$ instead of -2.3×10^{-5} $\Delta k/°F$, the peak power will double, but the integrated heat generation is still small.

It was shown above that the inherent negative reactivity effect of the Doppler coefficient is effective in limiting the initial power excursion. However, continued rod withdrawal will eventually result in a slow power rise. Figure 2 shows that the initial power excursion would reach a maximum value about 150 milliseconds after reaching the overpower scram level. Since the delay time to initiate a rod scram is about 240 milliseconds, rod motion would not be initiated until after this maximum had been passed. Consequently, the principal effect of the

overpower scram is to terminate the rod withdrawal so that a slow power rise does not occur. That situation is examined in Part One, Chapter I, B.

5. Conclusions

The safety features provided to prevent a power excursion caused by an accident of this nature include period fast insertion, period scram, and overpower scram. Three independent safety system failures would have to occur before the reactor excursion would not be terminated by control rod motion. Even if this should happen, inherent reactivity effects would halt the initial power rise approximately 150 milliseconds after the overpower trip is reached. Since the power peak occurs before actual rod motion takes place, small changes in scram delay time or a change in scram velocity would make little difference in the end result and certainly none in peak power.

The initial power level assumed in this study was about four decades (a factor of 10^4) lower than the actual source level. The results are therefore conservative in the sense that a larger reactivity was allowed to accumulate because of rod withdrawal before the power range was reached. This requires more negative reactivity due to heat generation to counteract the added reactivity. Furthermore, the nuclear instrumentation covers 12 decades below maximum power. The operator could not initiate a rod withdrawal from 13 decades below full power without circumventing the interlock which prohibits rod withdrawal at count rates below the instrument capabilities.

The results of the study indicate that no core damage can result and that no danger can exist as a result of a continuous rod withdrawal from source level, if the excursion is terminated by rod insertion shortly after the initial power peak is passed.

B. ROD WITHDRAWAL IN THE POWER RANGE

1. Nature of the Accident

In this accident, it is postulated that from one to five rods are withdrawn from the core at the maximum withdrawal rate beginning with the reactor at the maximum power level. This might occur, for example, through operator error, malfunction of control rod drive circuitry, or malfunction of the automatic control system.

2. Safety Circuitry

Two methods of shutdown protect the core from damage due to a power increase. The control rods will be scrammed when the reactor power measured by the neutron flux detectors reaches 130% of rated level. Secondly, the rods will scram if the primary coolant average temperature at the reactor outlet reaches 540°F. No safety action will be initiated by period measurement since the reactor period scram is not operative above 10% of full power. The same is true for the rapid insertion feature available during the startup accident. The design of the safety system provides for automatic disconnection of these features above 10% of full power because the negative reactivity coefficients in the power range will minimize net reactivity changes during a transient, thus rendering period indication of little value.

The position of the servo group, while in automatic control, is adjusted so that the reactivity held down cannot be more than 0.01 Δk greater than the reactivity to make up for the Doppler deficit (the Δk necessary to compensate for the Doppler coefficient from zero power to full power at operating moderator temperature). This is not enough reactivity to permit net steam formation in the event of an accidental withdrawal of this group of rods.

3. Simulation of the Accident

This accident was studied on the electronic analog computer at Battelle Memorial Institute. Reactivity was added at a constant rate starting at rated power. The rate of reactivity addition, again taken as 4×10^{-4} Δk/second is the maximum anticipated reactivity addition rate occuring when the five strongest rods are being withdrawn through their region of greatest worth at a rate of 15 inches per minute. Also

considered was the possibility of an accident in which there is mechanical failure of the drive carriage and/or the lead screws on one of the control rod drives. In this case, the design of the hydraulic system would limit the motion of the failed rod to 30 inches per minute. Even if the most reactive rod failed the resultant reactivity addition rate would be only half the assumed rate, or 2×10^{-4} Δk/second.

A further assumption — that the coolant temperature at the reactor inlet was constant during the transient — is considered conservative since the inlet temperature would actually begin to increase after one "loop time" (about 20 seconds) thereby causing a corresponding increase in all system temperatures. These increased temperatures would tend to reduce the power excursion because of the negative temperature coefficients of reactivity.

4. Results of the Study

During a transient of the order of one or two loop times in duration, the limiting thermal design criterion in this reactor is central melting of the fuel rod in the hottest channel of the core. Central melting is predicted to occur at a power level well below that at which net steam formation or burnout occurs in the hot channel.

The results of the analog studies for this accident are presented in Figure 3. A scram at 130% of rated power would limit the central fuel temperature in the hot channel to less than 4000°F. If the overpower scram action failed to function, the reactor power and central fuel temperature would continue to rise, and the center of the fuel pin in the hot channel would reach melting temperature (5200°F) approximately 20 seconds after initiation of rod withdrawal. With a continuing rise in power the reactor outlet temperature would reach 540°F in about 30 seconds. A scram due to high reactor outlet temperature would prevent melting of the fuel pin cladding in the hot channel. It was found that the reactor period never fell below 16 seconds, a fact that reinforces the previous statement that it is not necessary to use the period circuitry at this power level.

5. Conclusions

There are two possible methods of safety shutdown following a continuous rod withdrawal in the power range. Either method will

successfully shut down the reactor prior to melting of the cladding in the hottest core channel. However, the occurrence of such an incident is very unlikely. Part I, Chapter II, C presents a more detailed discussion of the dependability of the control rod drive system.

C. COLD WATER ACCIDENTS

Cold water accidents are positive reactivity additions arising from the addition of cold water to the core. Specifically, the causative agent is the inherent negative moderator coefficient of the core in combination with water of a lower temperature than that previously occupying the core. The following analyses examine several actions postulated to add colder water to the core.

1. Idle Loop

a. Mechanism of Cold Water Addition

If a primary coolant loop is inoperative and the isolation valves are closed, the water may lose heat to the containment vessel atmosphere and thus become colder than the water in the loop still in operation. If the loop is then returned to operation, a cold water accident may occur. Assuming such an accident, there are two cases of interest.

In the first, it is assumed that both primary pumps in the idle loop were started before the isolation valves were opened. In the second case, it is assumed that the isolation valves were opened before the pumps were started.

b. Safety Interlocks

There is no normal operating condition in which a primary loop would be isolated. Although a leak or an equipment malfunction might necessitate isolation, the loop would be considered inoperative until the equipment was repaired. Access to the containment vessel is restricted to periods of reactor shutdown, so an idle loop to which repairs were made would be reactivated only when the reactor was subcritical. The postulated accident therefore is one in which an isolated loop has been deliberately activated by an operator while the reactor is critical. To prevent this type of cold water accident special interlocks have been installed to protect the reactor.

Three interlocks limit the temperature and rate of cold water introduction to the core. The reactor inlet valve is a slow-opening valve which takes approximately 4 minutes to fully open. One

interlock controls the sequence of inlet valve and pump operations so that the pumps in the idle loop cannot be started while the reactor inlet valve is open. A second interlock prohibits starting the pump unless the temperature difference of the coolant between the idle loop and the reactor is less than 75° F. In addition to minimizing the power excursion following activation of an idle loop, this interlock protects the primary piping and boilers from severe temperature gradients due to the sudden introduction of hot water to a cold loop. A third interlock prevents idle loop startup unless all 21 control rods are fully inserted in the core. Any cold water addition to the core resulting from an idle loop startup must be preceded by a complete failure or deliberate by-passing of the three interlocks. Should the cooler water be introduced to the reactor by intentionally "defeating" the interlocks, the resulting power excursion would be halted by the combined effects of the negative fuel temperature reactivity coefficient and the overpower scram.

c. Simulation of the Accident

In the following discussion, none of the interlocks are assumed to be functioning. Further, the control rods are assumed to be partially withdrawn making the core critical.

It was further assumed that the reactor was operating at a power level recommended for the condition of the reactor system with one loop shut down, or 54% full power. This was also simulated by analog computer methods at Battelle Memorial Institute. The cold water from the idle loop was assumed to be mixed homogeneously with the hot water returning from the operating loop. The resulting reactor inlet temperature transient was introduced to the simulator which then computed the resultant reactor power excursion.

The assumption that the "hot" and "cold" flow from the two loops in the thermal shield pass mixes perfectly before entering the first pass through the active portion of the core is not an accurate representation of the actual conditions that will take place within the thermal shield area, where the mixing is assumed to occur. However, nuclear calculations show that this approach is conservative. A calculation using slab geometry indicates that perfect mixing of the water would yield a k_{eff} approximately 1.4% higher than if the cold water

(assumed to be 315° F) were to remain separated on one side of the core from the hotter water (500° F) entering the other side. The assumption of perfect mixing is therefore conservative in the sense that it will predict a greater reactivity addition as a result of a cold water accident.

d. Results of the Analysis

As mentioned previously, the idle loop cold water accident was analyzed for both valve limiting and pump limiting cases. Since the isolation valve between the pumps and reactor inlet cannot be opened fully in less than 4 minutes, the valve limiting case was not expected to be as severe as the pump limiting case. Comparison of Figures 4 and 5 bears this out. In the valve limiting case, the combined flow reaches 90% full flow in about 60 seconds. In the pump limiting case, the combined flow reaches 90% full flow in about 0.6 seconds.

The power excursion resulting from the rapid pump limiting flow transient was investigated with and without overpower scram for an idle loop temperature of 130° F. At this temperature the flow transient introduces 2% excess reactivity within the first second of time. The combined effects of the negative Doppler coefficient and the overpower scram limit the power excursion to approximately 500 MW. Since the power rises and falls sharply within 1 second, the integrated heat generation is not appreciable. For the case of no automatic safety action, the increased duration of the excursion results in core damage. The pump-valve interlocks however, will prevent this rapid reactivity introduction by allowing the slow-opening valve to limit the flow rate from the idle loop. In addition, the rod bottom interlock prevents starting of pumps in an idle loop when the reactor is critical.

The accidental start-up of the idle loop for the valve limiting case was investigated for several values of temperature, and for two valve opening time periods. The results are given in Figure 6 for the no-scram cases. With a rapid power increase, the heat generated in the fuel pin has less time to diffuse to the coolant, resulting in larger fuel temperature transients. Increasing the valve opening time from 1 to 4 minutes reduces the peak power by a factor of about two in the case of a 130° F idle loop temperature. With the 4-minute valve, a temperature difference of approximately 100° F can be

tolerated without scramming the reactor. Hence, if the reactor shutdown interlock were intentionally "defeated" and the two remaining interlocks were functioning properly, accidental activation of an idle heat transport loop results in a moderate power rise, however, not sufficient to "scram" the reactor.

e. Conclusions

The reactor is protected by three safety interlocks which prevent the introduction of cold water from an idle loop into the reactor. In addition, the SAVANNAH Operating Manual specifies that the reactor must be shut down and all control rods completely in the core prior to starting the pumps in an isolated loop. The manual also specifies that the temperature of the idle loop must be within 75° F of the temperature of the operating loop prior to starting pumps in the isolated loop. In the unlikely event that the interlocks are not functioning and that the operating procedures are violated, a reactor scram will terminate the excursion.

2. Secondary System Rupture

a. Nature of the Accident

The second mechanism that would reduce the primary coolant temperature below normal value is predicated upon a rupture in the secondary (steam) system. A rupture between the steam generators and the turbines will release additional steam from the steam drums, resulting in rapid reduction of secondary water pressure and temperature. The temperature decrease will tend to cool the primary water passing through the steam generators in the affected loop, and reduce the primary coolant temperature at the reactor inlet.

b. Analysis of the Accident

This analysis utilized the Babcock & Wilcox Datatron computer in a stepwise manner with respect to time. In view of the large changes in fluid properties and temperatures, it was felt that a digital computation would be more accurate than analog simulation.

The accident was analyzed on the basis of the worst possible temperature reduction rate in the primary system. This maximum

cooling rate would result from a large rupture in the secondary system. The break was assumed to occur in the 10-in. steam line, which is common to both boilers and leads to the main propulsion turbine. This is the largest steam line in the system. Actually a larger break — that of complete axial rupture of a steam drum — might be postulated but this accident would not be as severe for these reasons: First, the latter accident occurs in only one drum, not in an area affecting both steam generators. Secondly, although the primary coolant in this drum might be cooled more quickly, the rate of heat transfer is very dependent upon boiler tube and primary coolant film resistances. Third, the cooling effect can occur only while water remains in the heat exchanger. A rapid flashing in the drum would not result in a persistent cooling effect.

c. Results of the Analysis

Figure 7 illustrates the inlet temperature of the primary coolant as a function of time. Actually, there is a time lag between the inception of the accident and the reduction of reactor inlet temperature. However, this time lag is not pertinent to the analysis of this accident.

To establish some frame of reference for comparison, the predicted temperature-time variation of the primary coolant at the reactor inlet is plotted in Figure 7 for the case of cold loop startup. The temperature transient for a cold loop at 392° F (4 minute valve) corresponds closely with the steam system rupture transient. Coolant temperature due to a pipe rupture in the secondary system was calculated only up to one "loop time," after which the transient effect of the reactor would be felt. The dashed line indicates that the additional heating effect of the reactor transient will tend to reduce the rate of cooling of the primary coolant.

The maximum reactor power attained following a steam system rupture can be obtained from Figure 6 if the temperature curve due to a rupture is substituted by a curve due to an idle loop startup with the cold loop at 392° F (4 minute valve). By assuming that the reactor would be operating at 100% of maximum power and making a comparison with the results of the idle loop investigation, this accident would result in a power excursion with a peak power less than 200% of rated power.

An automatic high flux scram is indicated since the 130% flux level would be exceeded. Compounding the accident by assuming a failure to scram on overflux trip might result in central fuel melting of the hottest pins but certainly not exceeding the burnout heat flux conservatively predicted to be in excess of 200% maximum power.

d. Conclusions

A large rupture in the secondary system can produce a rapid cooling of primary cooling water. Normally, the reduction of primary cooling water temperature will result in a reactor power increase that would be terminated by an overpower scram. Even if the safety systems fail, some central melting of the hottest fuel pins may occur, but the burnout heat flux would not be reached. It is concluded that a rupture of the secondary system of the most severe nature would not result in permanent damage to the reactor or in the release of radioactivity from the primary system.

3. Excessive Steam Demand

a. Nature of the Accident

The description of a cold water accident covered structural interlock, and secondary system failures within the system. This section will examine the results of a cold water accident arising from excessive steam demand without a structural failure. Within the limitations of this volume, a rapid change in steam demand can be brought about by a rapid opening of either the main propulsion turbine throttle or the steam by-pass valve.

To examine a rapid steam demand resulting in a reactor excursion beyond normal full power, it was assumed that the reactor was operating at full power when the steam by-pass valve opened accidentally.

b. Steam By-Pass Design

The steam by-pass capacity is specified at 190,000 lb/hr. A pneumatically operated automatic control valve was installed to regulate the flow rate. The flow control valve throttles main steam to the condenser where it enters a sieve-type inlet nozzle which disperses

the steam and entrained water droplets over the condenser tubes and baffles.

High back pressure in the condenser closes the automatic steam by-pass valve, protecting the condenser against over pressure. Manual override control of the opening of the steam by-pass valve protects the ship against malfunctioning of the steam by-pass valve and consequent loss of necessary steam to the main propulsion turbine.

The steam by-pass flow is controlled by a pressure signal from the main steam line, which is designed to maintain constant pressure in this line and constant steam flow from the steam generators. A slight pressure deadband in the control system, and the time lag associated with the throttling valve and its control system results in slight load transients being transmitted to the steam generators. The heat capacity of the steam generators and the primary loop coolant further damps the power transient actually transmitted to the reactor. A slight change in primary coolant temperature is required to accommodate such small load transients transmitted to the primary system.

The control valve in the steam by-pass line is assumed to open, resulting in a steam demand approximately 73% in excess of the maximum power demand.

c. Analysis of the Accident

The transient response of the reactor and heat transport systems was also investigated by analog computer at the Battelle Memorial Institute. For purposes of this study, the influences of the auxiliary systems of the power plant were eliminated.

The accident was analyzed using two values for the moderator temperature coefficient, with and without operation of the reactor control system. The control system was assumed to be capable of withdrawing a group of four control rods simultaneously and introducing reactivity at a rate of 0.01 Δk/minute.

d. Results of the Analysis

Results of the most pertinent analog runs are listed in Table I. A conservative value was used for the Doppler coefficient

(-1.5×10^{-5}/°F) in order to yield largest values for the reactor power excursion.

It can be seen that the lower value of moderator coefficient resulted in lower minimum primary system temperatures. This follows, because the reactor will attempt to meet the new demand only if additional reactivity is due to a lower moderator temperature or the withdrawal of control rods. Naturally, a smaller moderator coefficient would require a greater moderator temperature drop to accomplish the same reactivity addition. Similarly, automatic regulation will result in the addition of reactivity by rod withdrawal, thereby preventing the larger coolant temperature drops associated with no control.

The maximum reactor power reached in any of the cases listed is approximately equal to full power plus the additional demand. The variation in maximum power level is within the accuracy of the calculations and printout.

It was mentioned previously that additional steam demand resulting from opening the by-pass valve would approximate 73% of full power demand. It can be seen that there is little or no overshoot in any of the cases examined. Normally, a reactor scram would occur if the neutron level exceeded 130% of the level at full power. Postulation of a double failure in which both the steam by-pass valve opens during full power operation and the safety system fails to function would not result in damage to the reactor. The power level that is reached is only slightly higher than that at which central melting of the hottest fuel pellet is predicted. No damage to the fuel element cladding would result.

e. Conclusions

Although normal scram is expected in the event of this accident, scram failure would not result in damage to the reactor or any other part of the plant. The effect on reactor power and system temperatures would not differ significantly in either case. Normal operation may be resumed after the cause of the trouble has been remedied.

TABLE I

RESULTS OF SIMULATION OF EXCESSIVE STEAM DEMAND ACCIDENT

Moderator Coef. $\Delta k/°F$	Control Action	Initial Primary System Avg Temp, °F	Minimum Primary System Avg Temp, °F	Final Primary System Avg Temp, °F	Initial Steam Avg Temp, °F	Final Steam Avg Temp, °F	Max. Rod Withdrawal, in.	Max. Reactor Power (% of Full Power)
3.2×10^{-4}	None	508	496	496	458	413	None	168
1.6×10^{-4}	None	508	472	472	458	398	None	168
3.2×10^{-4}	Automatic	508	507	508	458	428	7	176
1.6×10^{-4}	Automatic	508	504	508	458	423	10	172

D. XENON BURNOUT TRANSIENT

1. Nature of the Accident

It is postulated that the reactor has been shut down after an extensive length of time at full power. The xenon-135 concentration will begin to increase upon shutdown because it is being produced by the decay of its precursor, iodine-135, and is no longer being destroyed by neutron absorption. The xenon-135 concentration will reach a maximum value, then decrease as its decay begins to exceed its production from the dwindling iodine-135 "inventory".

If the reactor were to be brought back to full power when the xenon-135 concentration was maximum, xenon would burn out at some maximum rate which would tend to introduce positive reactivity to the reactor. This possibility has been examined to determine the rate of reactivity addition as a function of time and its implications in terms of reactor safety.

2. Reactivity Effects

Figure 8 illustrates the poisoning effect of xenon versus time as the preceding events take place. The A-B portion of the curve represents the steady state poisoning effect of xenon in terms of Δk. At point B the reactor is shut down to zero power. The B-C curve represents the buildup of xenon to maximum value. Thereafter, it would decay according to the dotted line if the reactor had not been brought back to full power at point C, some 4 hours after the shutdown. The C-D curve illustrates the more rapid xenon depletion resulting from this power increase. The initial slope of this part of the curve is shown as C-C' and is found to be about 1.5×10^{-5} Δk/minute.

3. Control of the Reactor

Compared to the control capability of the servo system, this reactivity addition rate is very small. The servo system will normally be able to make reactivity compensations at a maximum rate of 1.5×10^{-2} Δk/minute, a rate 1000 times the xenon burnout rate. No control problem is indicated, therefore, if the servo system is in operation. Indeed, no problem would exist even if manual control were being used since the resulting reactivity addition rate is so small that

an operator could very easily accommodate it with periodic rod movement. Assuming that the system is not being operated automatically, the possibility that the operator ignores the condition is of interest.

4. Reactor Response

In this case, the addition of positive reactivity will result in an increase in reactor power, the rate of which depends upon the rate of reactivity addition. The following analysis assumes that the initial rate of 1.5×10^{-5} Δk/minute continues throughout the period of interest.

Since there has been no additional power demand, any extra power increase can only tend to add energy to the primary and the secondary system coolants and thus increase their temperatures.

By virtue of the inherent negative temperature coefficient of reactivity, this temperature increase will tend to compensate for the reactivity being added by the mechanism of xenon burnout. In fact, the power level will increase to a point where the excess energy addition to the system increases primary water temperature at a rate that corresponds exactly to the reactivity addition rate. This power increase has been calculated to be less than 0.5 MW, so small as to be of little concern regarding burnout. Figure 9 illustrates this power increase as a function of core life for the most probable combination of reactor parameters and also for the worst possible combination.

5. Safety Features

A number of alarms will bring a condition of this nature to the operator's attention. About 55 minutes after the beginning of the temperature increase, the "High Temperature-Reactor Outlet" alarm will be actuated. If no corrective action is taken, "High Temperature - Plant Average" alarm will be actuated 60 minutes from the beginning of the transient. Failure to respond to this alarm will lead to a scram when the reactor outlet temperature reaches 540°F, estimated to occur about 95 minutes from time zero. Even if no scram occurred, "High Boiler Pressure" alarms will actuate 230 minutes from time zero. Failure to give recognition to any of these alarms will result in automatic opening of the boiler pressure relief valves approximately

265 minutes from the start of the incident. Thereafter, the boiler relief valves will function so as to dump the excess heat until the xenon transient has been overcome by subsequent xenon buildup at full power operation and resumption of normal conditions.

6. Conclusion

The most extreme reactivity addition rate due to xenon burnout was postulated in this analysis. The resulting power increase is very small in comparison with normal power. No rapid power transient results. If the operator ignores the immediate condition, the plant will operate normally for approximately 1 hour before alarms bring the condition to the operator's attention. Adequate time is available to accomplish the required minor adjustment in control rod position.

FIG. 1: STARTUP ACCIDENT – LOW POWER RANGE

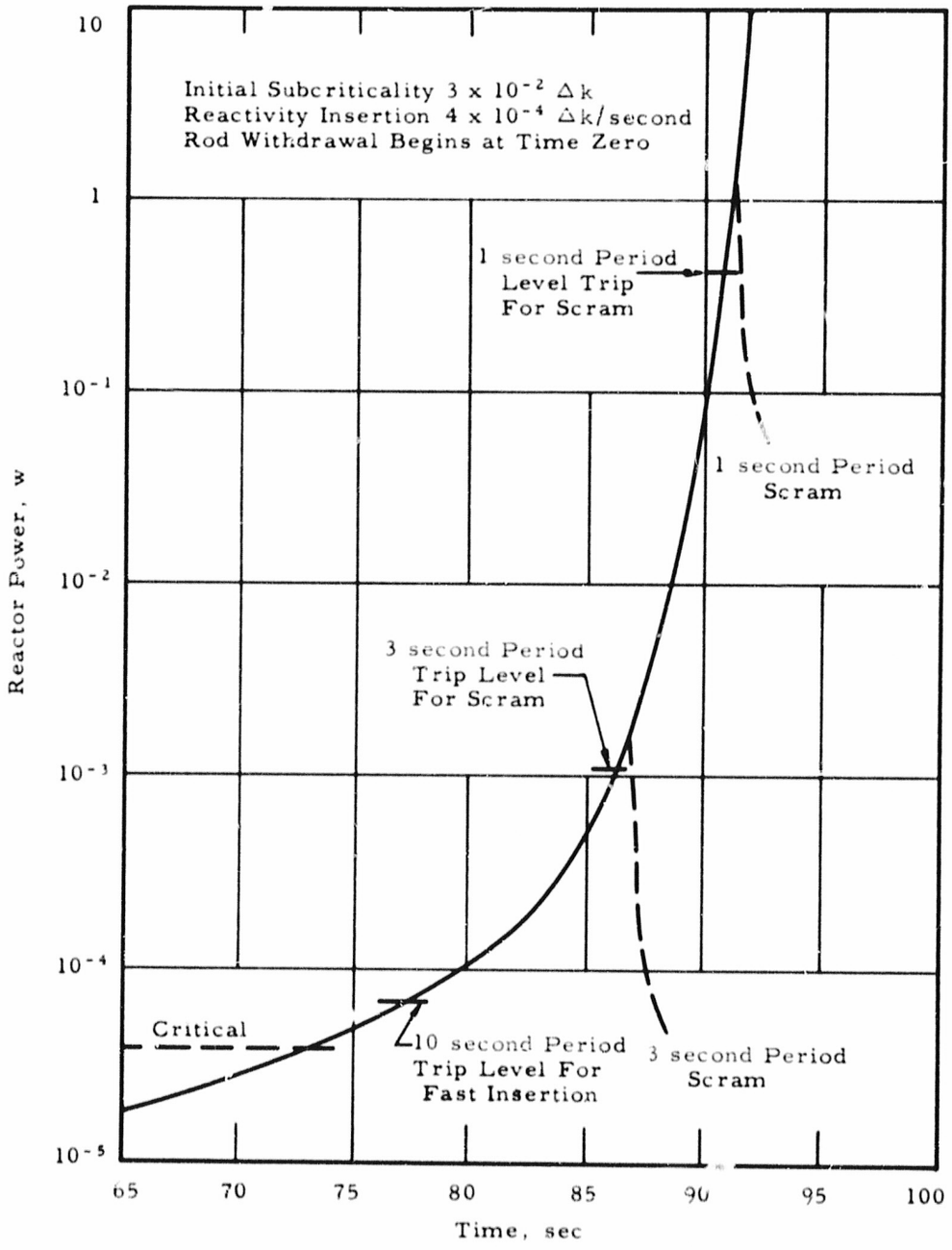

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FIGURE 1

FIG. 2: STARTUP ACCIDENT – HIGH POWER RANGE

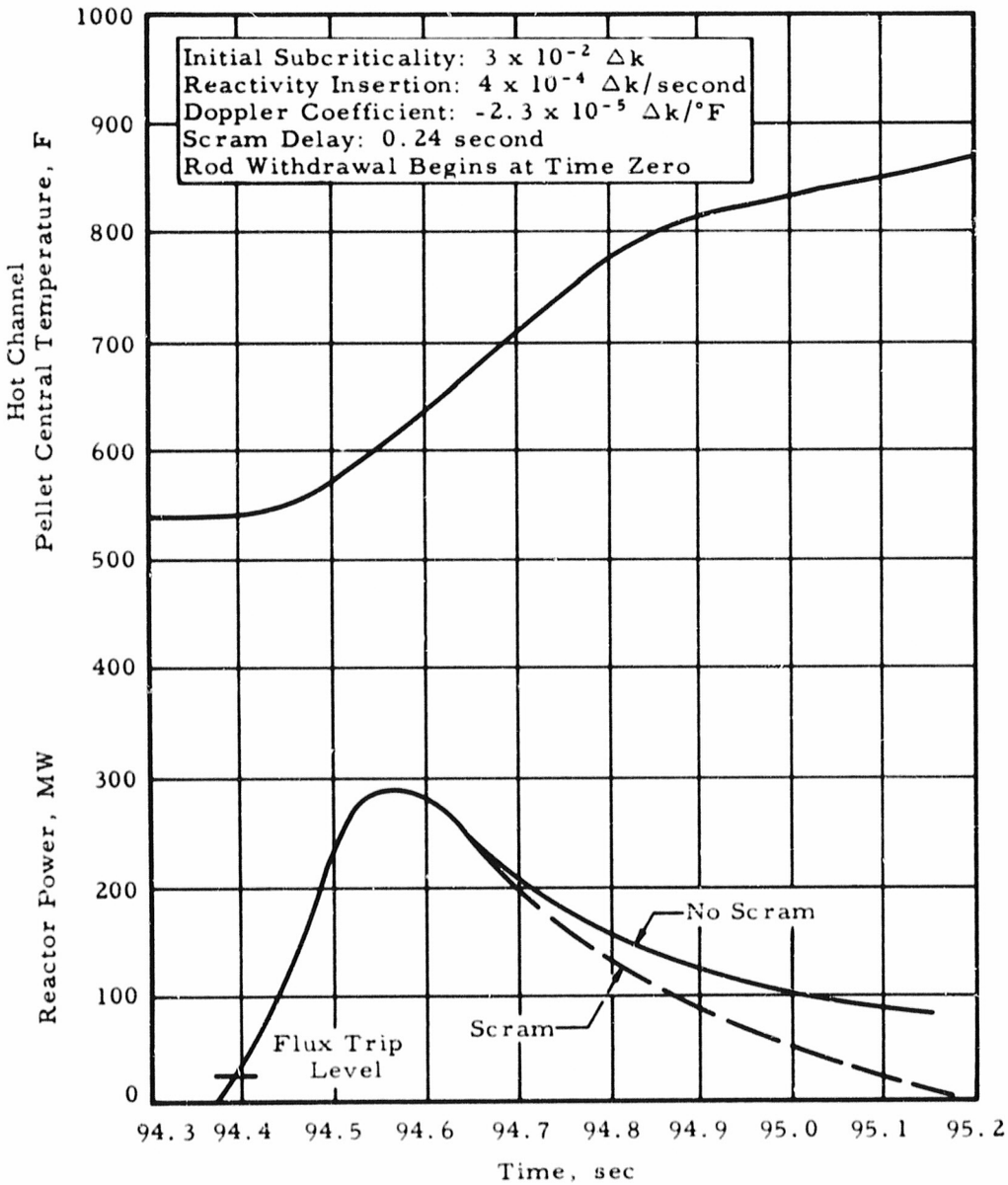

FIGURE 2

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FIG. 3: ROD WITHDRAWAL IN THE POWER RANGE

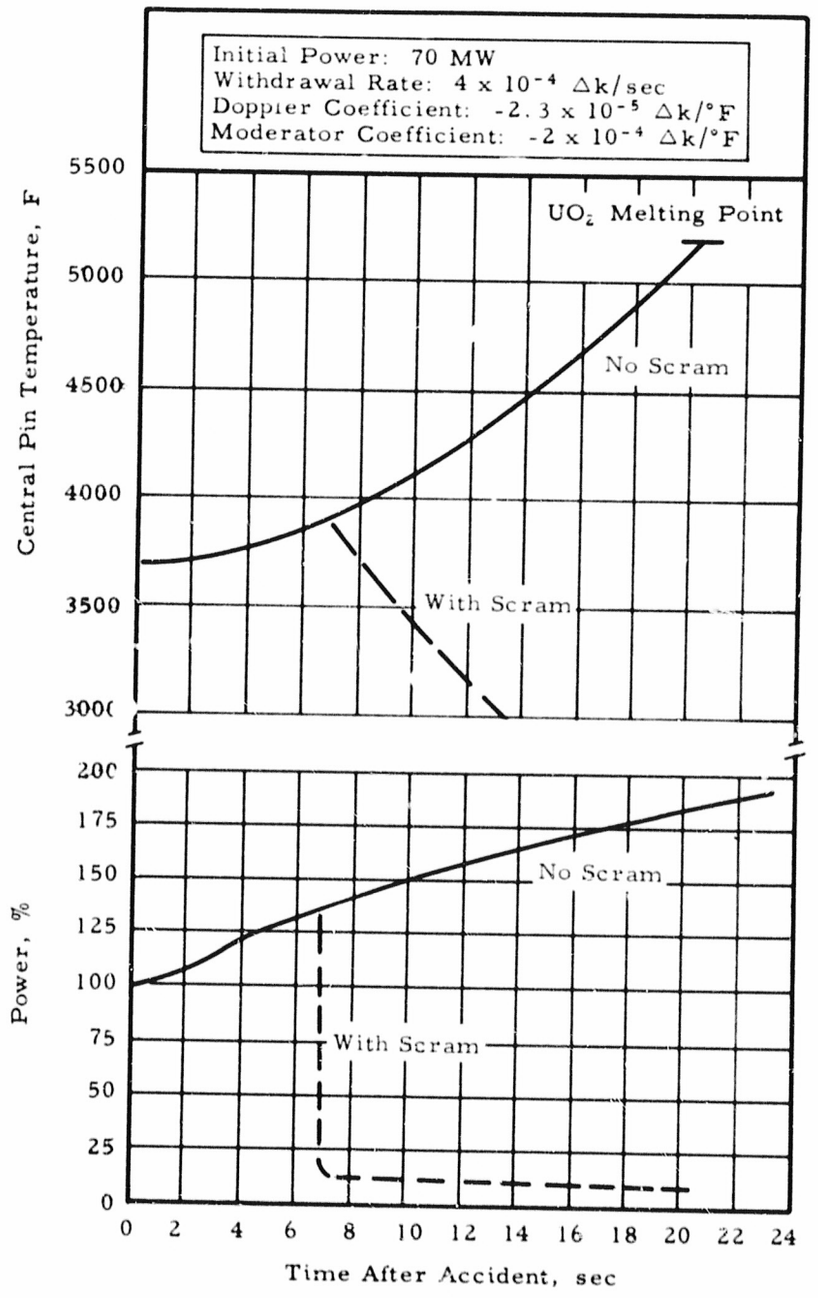

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FIGURE 3

FIG. 4: COLD WATER FLOW TRANSIENT – VALVE LIMITING

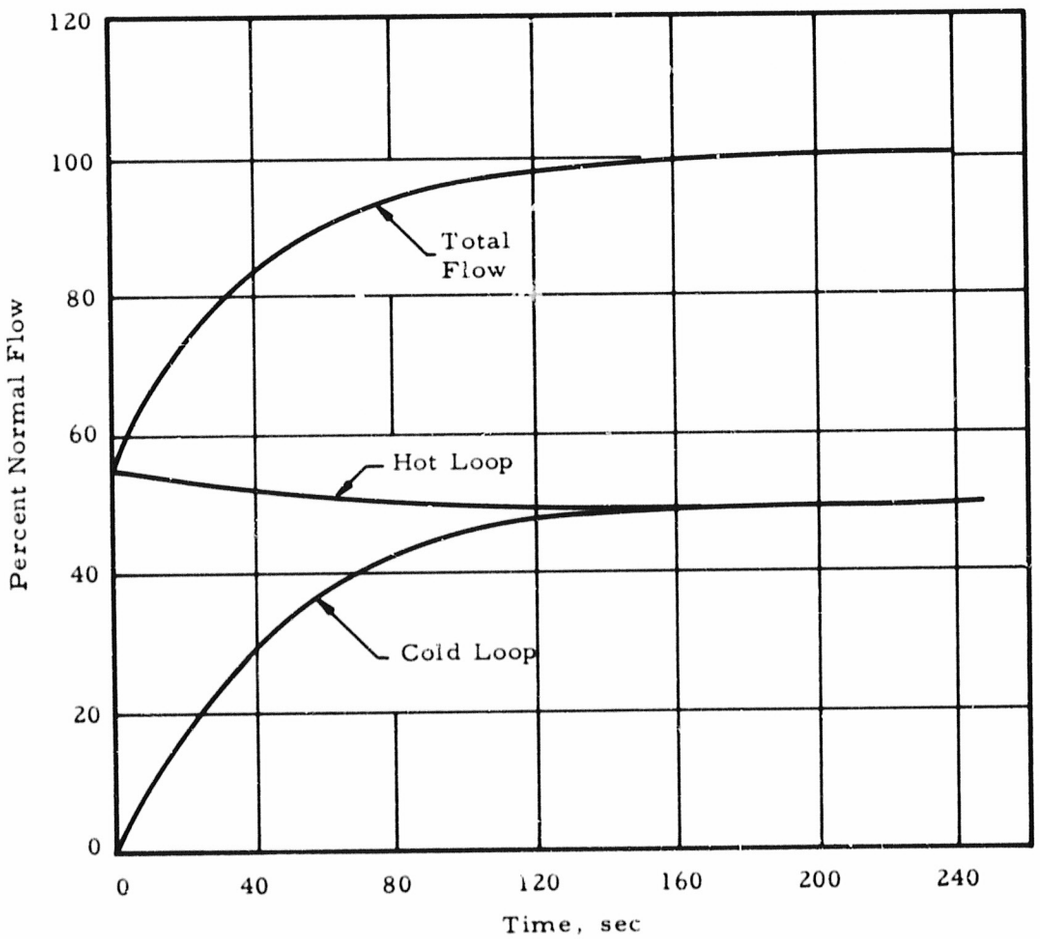

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FIGURE 4

FIG. 5: COLD WATER FLOW TRANSIENT – PUMP LIMITING

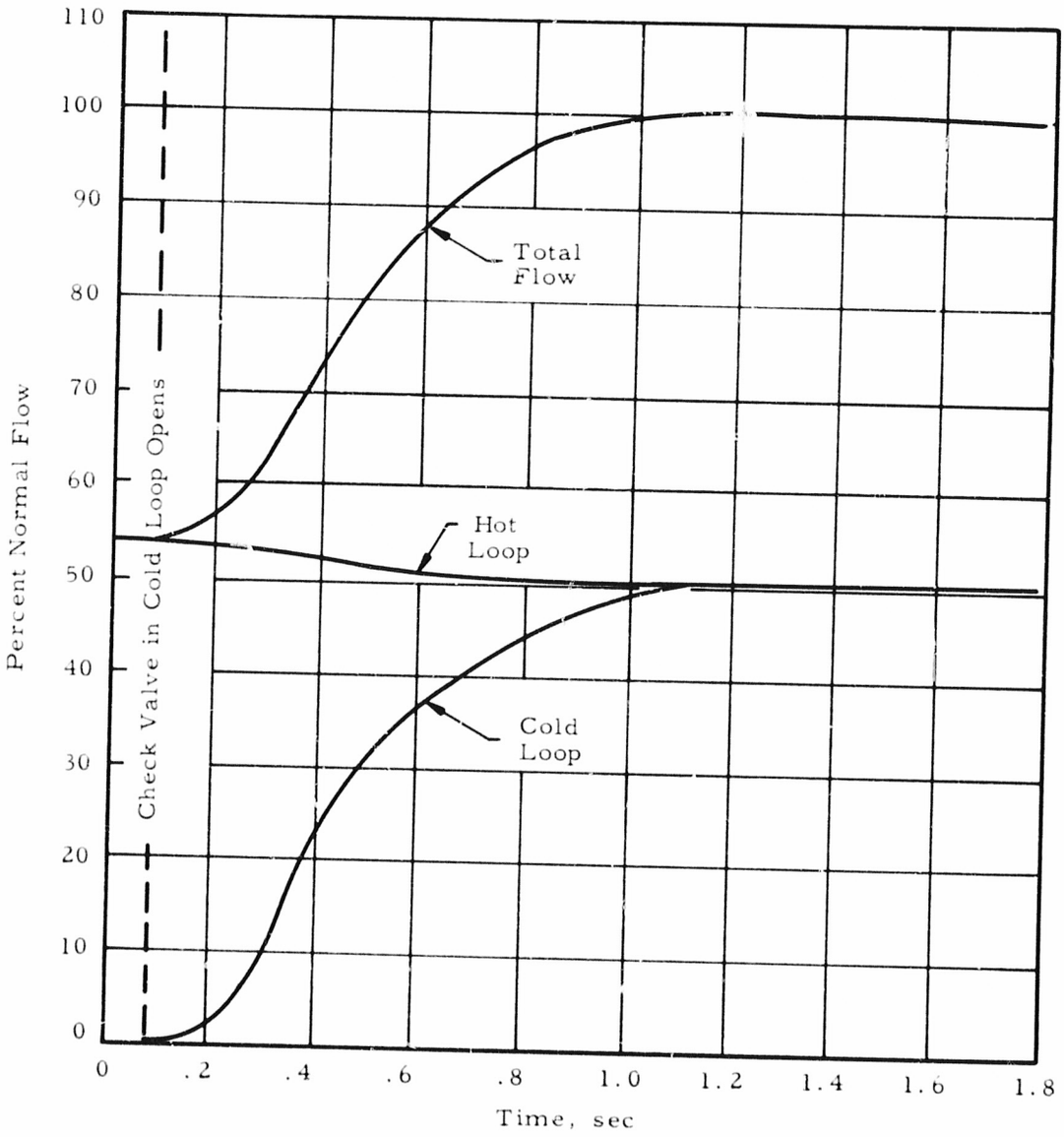

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FIGURE 5

FIG. 6: COLD LOOP STARTUP

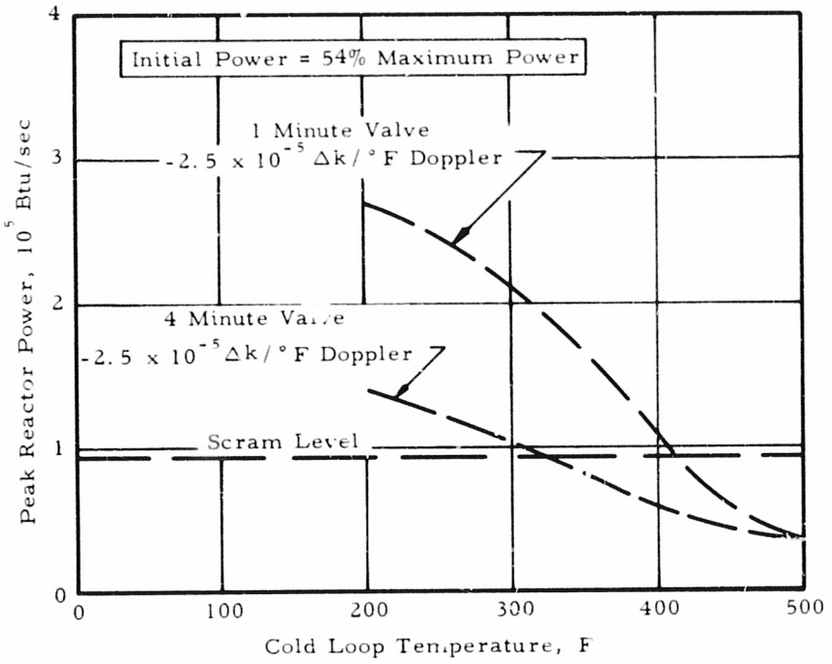

FIGURE 6

FIG. 7: REACTOR INLET TEMPERATURE AFTER SECONDARY SYSTEM RUPTURE

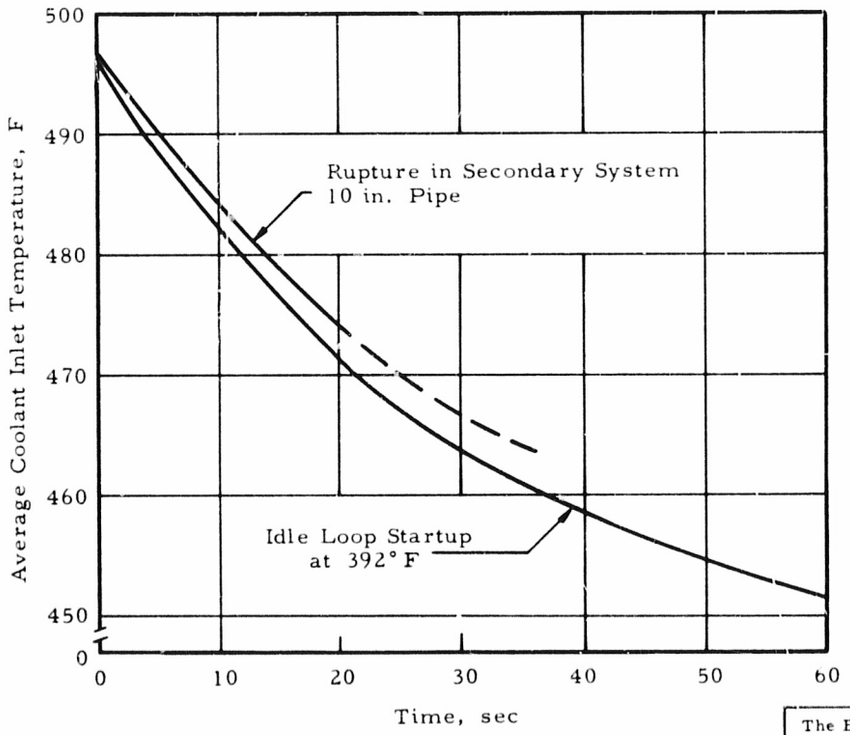

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FIGURE 7

FIG. 8: REACTIVITY VARIATIONS DURING XENON TRANSIENT

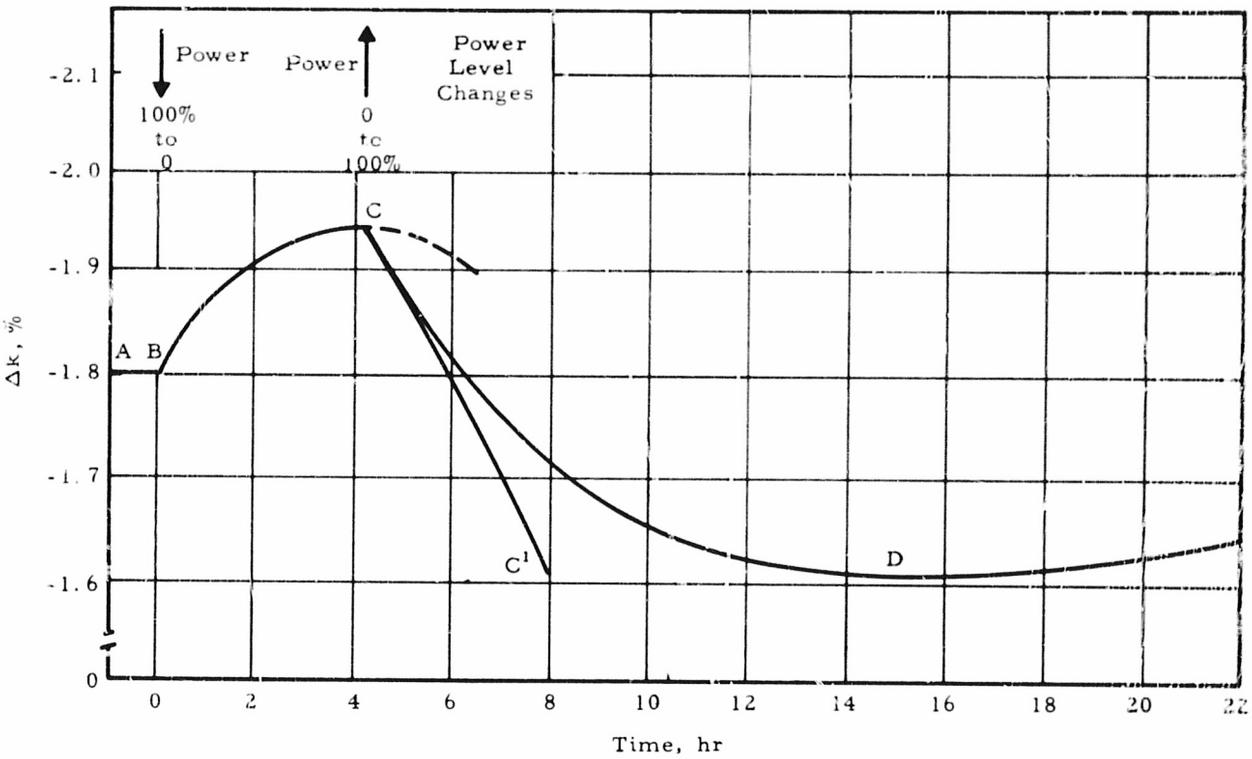

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FIGURE 8

FIG. 9: POWER INCREASE DUE TO XENON BURNOUT -- NO CONTROL ACTION

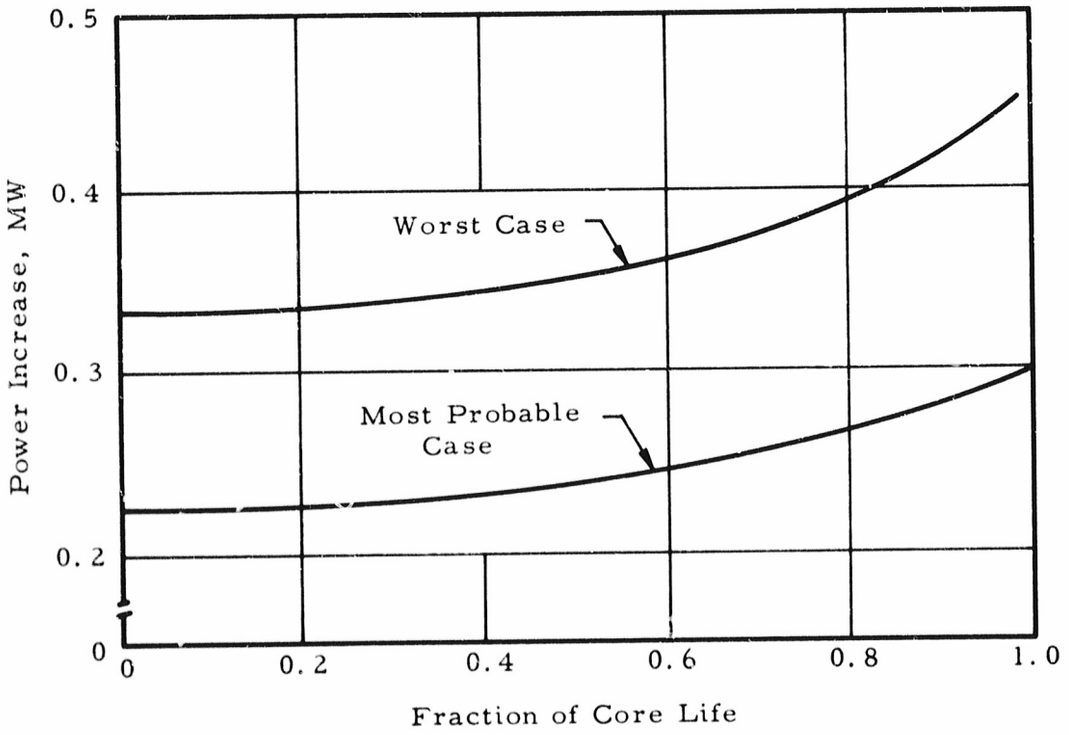

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FIGURE 9

Chapter II

MECHANICAL FAILURES

A. Fuel Element Failure
B. Primary Coolant Leak
C. Control Rod Drive Failures
D. Loss of Instrument and Control Air Supply
E. False Pressure Indication in Containment
F. Loss of Power to Primary System Pumps
G. Loss of Alternating Current Power
H. Loss of Primary Cooling Water
I. Missile Formation

II. MECHANICAL FAILURES

A. FUEL ELEMENT FAILURES

1. Nature of the Problem

Particular attention has been given the failure of one or more fuel pins prior to or during operation of the reactor at appreciable power levels.

Failure of the fuel cladding can result in the dispersion of radioactive fission products and, in the event of gross rupture, some UO_2 fuel throughout the primary and auxiliary systems. Contamination of the primary coolant and associated equipment is the possibility considered in this accident.

One advantage of using oxide fuel is the ability to operate with defective fuel pins. Although the probability of a failure in a single fuel pin may be small, the large number of pins in the core increases the probability that a certain number may become defective. Therefore continued operation of the reactor with several defective fuel pins is desirable over long periods of time. Naturally operation under this condition would result in an accumulation of radioactive fission products in the primary loop, the purification system, the liquid waste collection and storage system, the gaseous waste collection system, and, to a slight degree, in the containment vessel. Most of these fission products would be concentrated in the ion exchangers and gaseous waste collection system.

The following discussion describes the various design features that accommodate this condition and the resultant radiation levels.

2. Radioactivity Release

A program has been developed for predicting, by theoretical means with empirical constants, the composition and quantity of fission products that can be released as a function of the amount of oxide fuel

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exposed to the coolant. The analysis is based on experimental data from the Pressurized Water Reactor Tests run on the Atomic Energy of Canada, Ltd. facilities at Chalk River, Canada.

Although the release of fission products from defective fuel pins will increase the radiation levels outside the piping and components containing primary coolant, the contribution to the total radiation level from the primary system is negligible during full power operation compared with the nitrogen-16 activity. Consequently, the additional activity due to fission products in the primary coolant becomes significant only after the reactor is shut down, and access to the region inside the secondary shielding or the containment vessel is necessary. Figure 10 gives the dose rate 4 in. from the surface of the primary system piping and boiler. The reactor is assumed to have been operating for 300 days prior to the failure of over 250 individual fuel pins, equivalent to the exposure of 363 kg of fuel, or approximately 5% of the total fuel loading. The reactor is assumed to operate an additional 100 days with the fuel exposed prior to shutdown. The radiation level decreases to a value of 200 mrem/hour 4 in. from the piping, 24 hours after the reactor has been shut down. An additional four days would be required to reduce the radiation level to 100 mrem/hour. Since the dose rate following reactor shutdown is directly proportional to the quantity of fuel exposed, the dose for other amounts of exposed fuel can be estimated from Figure 10.

Experience on operating pressurized water plants has shown that, except for isolated components, it is not too difficult to attain a radiation level of 200 mrem/hour a short time after shutdown. Further, it is not unreasonable to permit maintenance activities for short periods at these levels. The above levels therefore seem acceptable for controlled access and a limited amount of direct maintenance operations soon after shutdown.

The shielding on the ion exchangers and gaseous absorption units is conservatively sized to reduce the radiation dose at the outer surface of the units to 200 mr/hour with 7.26 kg of UO_2 exposed to the primary coolant.

Thus, the release of fission products from the defective fuel elements does not significantly increase the direct radiation levels

during reactor operation, but some inconvenience may result after shutdown due to the increased time required before equivalent radiation levels within the secondary shielding and containment vessel are reached.

Although the radioactivity from the fission products is significant, the actual quantity in circulation in the primary system is extremely small. Consequently, the useful life of the ion exchange resins in the primary purification system is not significantly affected. Operation with fission products in the primary coolant does require the use of a gaseous collection system to insure that all fission gases may be accumulated in charcoal beds, as described in Part I, Chapter III, B.

B. PRIMARY COOLANT LEAK

1. Nature of the Problem

The release of all or a large part of the primary system water to the containment vessel is analyzed in Part I, Chapter II, H. Of concern here is the other extreme, the formation of a small leak which may be made up by continual water addition to the primary system. Further, it is assumed that such a leak might occur in an area between the containment vessel and secondary shielding, the leakage products ultimately being discharged through the reactor compartment ventilation system. The analysis of such a condition is therefore pertinent in determining the radiation levels that might be reached in nearby areas adjacent to or on the ship.

2. Description of the Leak

A leak in the primary purification letdown line between the containment vessel and the ion exchangers could release primary system water to the space between the vessel and the concrete shield. The temperature of the water leaving the containment vessel is 110°F, so the leaking water will not flash to steam.

The reactor compartment is provided with natural air supply and mechanical exhaust. There are two independent systems, one serving the lower region between the containment vessel and secondary concrete shielding (Fig. 11) and the other serving the region outside the secondary shielding. The capacity of these systems is adequate to provide a complete change of air every 8 minutes. When desired the containment vessel may be purged with the fan serving the lower region, by opening a remotely operated valve in the bottom of the containment vessel and withdrawing air. Three alternate exhausts to the "weatner" are provided for this ventilation system: one each port and starboard, just above the water line, and one at the top of the fore mast. All air exhausted from the lower region is filtered by a series of three types of filters; (1) "roughing" (2) high efficiency particulate and (3) iodine filters located in the "C" deck fan room ahead of the exhaust blower serving the reactor compartment. Continuous radiation monitoring of exhaust air is provided after the filters. In addition, chemical smoke

tracing is provided in the discharge stream to assure the flow of exhaust away from the ship. The exhausts for the region outside the secondary shielding are located toward the aft end of the top of the superstructure.

3. Radiation Levels

The fission product concentrations in the primary water are computed, based on release coefficients measured for the Shippingport Pressurized Water Reactor, assuming 300 days operation prior to formation of defects in any fuel pins and 100 days operation with the defects, as detailed in Part I, Chapter II, A-2.

Table II summarizes the potential release concentrations, assuming that all of the water vapor that leaks from the system is drawn up the ventilating stack. These data are presented for a stack filter efficiency of 99.9%, and for an assumed leakage rate of 1000 cc/hour. The calculations were carried out for an assumed fuel exposure of 363 kg of UO_2. It is assumed that gaseous adsorption units are operating with an efficiency of 75%, but the reduction in primary water activity due to pressurizer void venting was not considered. The argon-41 activity in the primary water is reported without allowance for removal by the gaseous adsorption unit, since this condition could exist prior to fuel pin leakage.

Table II shows that the activity concentrations of the metals and noble gases of most concern do not result in levels approaching maximum permissible concentrations for continuous occupational exposure. Oak Ridge National Laboratory has advised use of the following criterion: During maximum continuous venting, the activity to the stack discharge may be 10 times the concentration for Maximum Permissible Concentration occupational exposures listed in Table II without exceeding the MPC for continuous public exposure at the point of maximum ground concentration.

The accumulation of fission products equivalent to an exposure of 5% of the fuel to the primary coolant does not require a limit on plant operation in respect to external or inhalation dosage associated with small primary or purification system leaks.

TABLE II

PRIMARY PURIFICATION SYSTEM LEAK

Isotope	Iodine-131	Strontium-90	Krypton-85 m	Krypton-87	Xenon-133	Xenon-135	Argon-41
Concentration in primary water (μc/cc)	8.5×10^{-1}	3.3×10^{-5}	1.7×10^{-1}	1.3×10^{-1}	1.8	1.2	3.1×10^{-2}
Leak rate (cc/hr)	1000	1000	1000	1000	1000	1000	1000
Ventilation rate (cc/hr)	6.8×10^{9}	6.8×10^{9}	6.8×10^{9}	6.8×10^{9}	6.8×10^{9}	6.8×10^{9}	6.8×10^{9}
Assumed filter attenuation	10^{-3}	10^{-3}	None	None	None	None	None
Concentration at top of stack (μc/cc)	1.25×10^{-10}	4.8×10^{-15}	2.5×10^{-8}	1.9×10^{-8}	2.7×10^{-7}	1.8×10^{-7}	4.6×10^{-9}
MPC (μc/cc) for continuous occupational exposure*	3×10^{-9}	1×10^{-10}	1×10^{-6}	2×10^{-7}	3×10^{-6}	1×10^{-6}	4×10^{-7}
Fraction of MPC continuous occupational exposure	0.031	0.000048	0.025	0.095	0.09	0.10	0.011
Half Life	8.05 days	19.9 yr	4.4 hr	78 min	5.3 days	9.2 hr	109 min

* National Bureau of Standards Handbook No. 69, June 5, 1959.

C. CONTROL ROD DRIVE FAILURES

1. Nature of the Problem

The malfunction or disability of several reactor control rods is of concern because of the potential hazards involved in the inability to shut down the reactor. This problem has been studied to determine the probability of rod malfunctions of various degrees of severity.

The basic approach to this analysis was to determine the most probable design parameters for the SAVANNAH core and to assign tolerances or confidence limits to each value according to the best available estimates of the possible range of values for each parameter. These parameters were used in their most probable and worst case combinations to describe the degree of control necessary to accomplish the following operations:

1. Shutdown

   At full power the average fuel temperature is considerably higher than the moderator temperature. The high fuel temperature results in a broadening of the fuel resonances and a reduction in reactivity. A loss of power (100% to zero power) would result in a cooling down of the fuel and an associated addition of positive reactivity. A certain degree of control is necessary to compensate for this reactivity addition. The reactivity associated with a reduction in power from 100% to zero power is defined as the power (Doppler) deficit.

2. Cooldown

   The reactor coolant exhibits a negative temperature coefficient of reactivity. The loss of reactivity associated with the heating up of the moderator from room temperature to operating temperature (508°F) is defined as the temperature deficit. As the reactor temperature is increased, control rods may be withdrawn to compensate for this reactivity loss. Likewise, control rods equivalent in reactivity to the temperature deficit must be inserted before the reactor may be cooled down.

3. Holddown

   If the reactor is shut down for several days after a power run, the xenon-135 will decay. Since the xenon thermal neutron absorption cross section is quite high, the xenon-135 decay results in the need for additional control compensation.

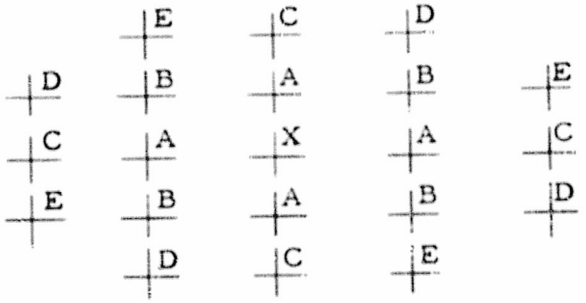
Figure 13 is a plot of the most probable values of temperature coefficient, Doppler deficit, and temperature deficit versus core life.

2. Rod Disablement

The possibility of a portion of the available rod worth being disabled was investigated to determine the maximum permissible number of disabled rods which would still permit accomplishment of shutdown, cooldown, and holddown.

Calculations and experiments have been performed to determine the worth of control rods individually and in groups.

The control rod groups are arranged in the reactor core as shown below.



The control rods are arranged into groups as follows:

Group	Rods	Operation
I	A - rods	manual when not connected through S_1 and S_2
II	B - rods	
III	C - rods	manual
IV	D - rods	manual
V	E - rods	manual
	X rod is not assigned to a group	manual
S_1	A - rods	manual or automatic
S_2	B - rods	manual or automatic

The servo group, S_1, is utilized to accomplish power changes from zero to 100% maximum power while maintaining an average primary system temperature of 508°F. From zero to approximately 15% maximum power Group S_1 is positioned manually; above 15% maximum power, Group S_1 may be on automatic control.

As the xenon poison accumulates in the core during power operation, the servo group will be withdrawn automatically. To maintain the Group S_1 in the "full power" position, the remaining manual groups and finally the central rod will be withdrawn. At the end of core lifetime all the rods except the Group S_1 will be fully withdrawn after the reactor has been at maximum power for sufficient time for xenon to reach equilibrium.

For operation at power levels less than 100%, the operator will base the desired Servo Group position on a linear interpolation between the zero and 100% power pointers.

The worth of any control rod or group of rods is dependent on the number and location of other control rods in the core. Table III lists the approximate rod worth for the case where only that group of rods is withdrawn.

TABLE III

ROD WORTHS, INDIVIDUALLY WITHDRAWN

Group	Worth per rod: Full Out - Reactor Hot	Worth per rod: Full Out - Reactor Cold
A	0.0075	0.0063
B	0.0075	0.0063
C	0.0072	0.0062
D	0.0077	0.0065
E	0.0077	0.0065
X	0.006	0.0051

The allowable number of disabled rods for the worst cases, based on the above data, are summarized in Table IV.

TABLE IV

PERMISSIBLE NUMBER OF DISABLED RODS

Action (Previously Defined)	Number of Rods Withdrawn at Critical Period of Core Life	Number of Withdrawn Rods Needed For Action	Number of Withdrawn Rods Which Can be Disabled
Shutdown	21	4	17
Cooldown	16	10	6
Holddown	16	13	3

3. Control Rod Drive Design Features

The control and safety actions, provided by 21 control rods, are initiated by two systems which are basically separate up to the point of their action on the rod drive line. The mechanism has been deliberately designed so that malfunction of the control system cannot affect the ability of the safety (scram) system to drive the rods into the core.

The basic features of the control rod drive are illustrated in Figure 14. Reactor pressure tends to force the control rod extension shaft upward until the collar on the shaft bears against the drive carriage. Rotation of a two-phase motor moves the carriage up or down on Acme lead screws and thus varies rod position. The maximum speed of the carriage is set by the synchronous speed of the motor. The lead screws cannot be turned by the collar on the control rod extension shaft pressing against the drive carriage.

The basic operation of the scram system (Fig. 15) is a straight forward hydraulic system powered by energy stored in compressed gas accumulators. Normally, the accumulator is charged to approximately 3000 psi. Upon release of the scram valve, hydraulic pressure is applied to both sides of the piston by forces as shown in the figure. As the rod begins to move downward, hydraulic fluid flows from the lower part of the cylinder into the upper part and the net force on the rod falls to that required to overcome friction and reactor pressure

only. The control rod extension shaft is not connected to the drive carriage, so the rod is free to move downward, away from the drive carriage, as soon as accumulator pressure is applied.

Since each rod is supplied by an individual hydraulic accumulator, an accumulator failure would result in the loss of scram ability of only one rod. Only a rupture in the main hydraulic supply header would result in loss of hydraulic supply to all drives. However, in this case the individual pressure accumulators, which are connected to the main header through check valves, would accomplish a successful rod scram by virtue of their stored potential energy. All rods are automatically scrammed on loss of pressure to the main hydraulic supply header.

4. Drive Failures

A careful study of 65 possible component failures within the drive system was made by the manufacturer, to determine the effect of each potential failure.

Failures of interest in this study are those which would result in the loss of ability to scram one or more rods.

1. In all cases a drive system failure will not result in the disability of more than one rod.
2. During normal operation, the loss of ability to scram any rod hydraulically does not preclude the ability to accomplish a fast insertion using the electric motor drive.

In similar fashion, the electronic scram circuitry is designed so that a failure of any part of it will not prevent a rod scram. Each individual monitoring channel has its own power supply, and duplication or multiplicity of each of the important safety channels is used. The scram signal bus is fed by all these voltage signals, and the bus in turn feeds the safety amplifier circuit. Interruption of proper operation of the bus or a loss of power will automatically initiate a scram.

5. Misalignment Study

In addition to determining the dependability of the control rod drive, assurance of a rod scram depends upon proper alignment of the control rod. The reactor internals are designed and supported to prevent any binding of the control rod when the reactor experiences the

worst gravitational deflective forces — that is, when the ship is heeled over and the reactor in a horizontal position. Table V summarizes the minimum control rod side clearance at each section of the reactor assembly with the net estimated maximum misalignment and displacement at that point for the worst condition.

TABLE V

MAXIMUM CONTROL ROD MISALIGNMENT

Location	Rod Side Minimum Clearance, in.	Reactor Internals Maximum Misalignment Plus Displacement, in.
Upper Flow Baffle		
Top	0.247	+0.032
Bottom	0.247	+0.017
Upper Grid Plate		
Top	0.370	+0.007
Bottom	0.370	+0.009
Fuel Container Assembly		
Top	0.179	-0.275
Bottom	0.179	-0.067
Lower Flow Baffle		
Top	0.307	-0.043
Bottom	0.425	-0.085

NOTE: + upward direction with ship on its side
- downward direction with ship on its side

The control rod is pivoted at its connection with the drive line extension shaft. When the reactor is in a horizontal position, the control rod will rest on some surface or surfaces of the control rod channel. The combination of misalignment and displacement for the Fuel Container Assembly and the Lower Flow Baffle is (in the worst case of misalignment and the reactor in a horizontal position) in the

downward direction while the Upper Flow Baffle and Upper Grid Plate could be in the upward direction. Because of the rod channel clearances, specified in Table V, the control rod will, under these conditions, rest on the bottom of the Fuel Container Assembly and also some point on the Upper Flow Baffle. For binding to occur, the control rod would have to bear against the upper surface of the control rod channel at the top of the Fuel Container Assembly and against the bottom surface of the channel at the bottom of the Fuel Container Assembly. Since in the horizontal position the control rod rests on the Upper Flow Baffle Assembly, the total downward displacement from normal rod position will be the difference between the minimum rod side clearance (0.247 in.) and the upward misalignment of the Upper Flow Baffle (0.032 in.). This displacement of 0.215 in., plus the minimum Fuel Container Assembly clearance of 0.179 in. (a total of 0.394 in.) is larger than the downward misalignment plus displacement (0.275 in.) at the top of the Fuel Container Assembly. Therefore no binding of the rod will result from the worst hypothetical condition of maximum misalignment and the reactor in a horizontal position.

The control rod drive supplier has indicated that the drives will function properly in the horizontal position. This statement is based on drive tests at angles up to 30 degrees from vertical, drive tests under imposed misalignment, drive deflection measurements and calculated support structure deflections. There is no reason to believe that the drive would not properly scram, even if the ship should capsize completely. The lateral deflective forces on the core assembly would again be less than in the horizontal position. Further, the spring-loaded safety latch mechanisms do not rely on gravity.

As far as can be ascertained the mechanical and structural integrity as well as the alignment features of the drive system and core assembly will permit a successful scram when the ship is in a horizontal position, the worst possible attitude.

6. Conclusions

Under the worst combination of conditions and with the reactor having operated at full power for a substantial period of time, only the weakest 13 of the 16 withdrawn rods must be inserted to accomplish a

complete cooldown and holddown. Consequently, complete holddown could be accomplished if three rods could not be inserted in any manner.

The foregoing investigation indicates that the simultaneous loss of ability to insert four rods, either by scram or electric drive, is an extremely improbable event. However the occurrence of this event would not necessarily jeopardize the reactor since it can be shut down hot (power reduced to zero) by the insertion of any four rods that are completely withdrawn when the scram signal is initiated.

D. LOSS OF INSTRUMENT AND CONTROL AIR SUPPLY

1. Nature of the Accident

Loss of air from the instrumentation and control air supply system will cause the pneumatically operated valves in the auxiliary systems to close. As an additional consequence the air-operated indicators and instruments will not function properly. As a result, there will be a number of false annunciated conditions. For example, some important annunciators such as "Low Pressure Cooling Water System", "Low Pressure Pressurizer", "Low Level Pressurizer" and "Low Flow Total Steam" will alarm

2. Probability of Failure

A failure of this type within the air supply is very unlikely. Further, the failure of an air compressor would not disable the system since either of the two compressors available can supply the entire air demand. (See Fig. 16.) Isolation valves permit continued operation of the air system in the event of a leak in a critical area.

3. Immediate Results

Immediately upon loss of air the reactor control system will switch from "automatic" to "manual" (whenever the steam flow falls below 10% of full power). As pointed out previously, loss of air creates a false indication of loss of steam flow, resulting in a switch to manual operation.

Since most of the indicator gages are not functioning in this case, the operator, who cannot determine actual plant conditions, is forced to shut down the reactor by scramming the control rods. This action will automatically close the main propulsion turbine throttle and cause a startup of the two 750 kw emergency diesel generators. About 15% of the normal steam flow will continue to be used by the turbine generators until the operator takes them off the line and the electrical load is picked up by the diesel generators.

If the operator does not take any action and if the makeup valve supplying water from the buffer seal system to the primary system is closed, sufficient steam supply is available for 7 minutes of turbine generator operation. If control air is restored within the 7-minute

period so that the makeup valve functions and no further action is taken by the operator, sufficient steam supply is available to operate the turbine generators for an additional 12 minutes.

4. Operator Action

After scramming the reactor on loss of air supply, the operator is instructed to remove decay heat by using emergency cooling system or, cool the reactor by dumping the heat to the steam generators rather than by using the emergency cooling system. The feedwater valve to the steam generators will have closed automatically upon loss of air. The operator may have this valve opened manually and start the port feed pump to maintain a proper drum water level. The steam formed in the steam generators due to reactor decay heat may then be dumped to one of the auxiliary condensers.

Failure of the operator to take any action will result in depletion of the available water in the boilers and subsequent loss of steam supply to the turbine generators causing a loss of electrical power to the main bus. The automatic scram resulting from this loss of power will shut down the reactor. The primary pumps will lose power and be shut off. The emergency cooling system will also be activated automatically on loss of power to the main bus.

5. Summary

The air supply system is designed to provide reliable service and is not likely to fail. If control air is lost the plant must be shut down until the leak can be repaired. The primary system will not be damaged and the operator can take appropriate action to prevent damage to the buffer seal booster and charge pumps.

E. FALSE PRESSURE INDICATION IN CONTAINMENT

1. Nature of the Accident

Two pressure sensors are provided to measure and indicate pressure within the containment vessel. The high range sensor indicates pressures from zero to 200 psig, the low range sensor from zero to 15 psig. The low range pressure transmitter will actuate an annunciator when the containment pressure exceeds 3 psig. In addition, when the containment pressure reaches 5 psig, a signal will actuate a pressure controller which will, in turn, isolate the air supply to a number of air controlled valves around the containment vessel. Since many of these valves require a constant supply of air to remain open, they will automatically close on the high pressure signal. Solenoid operated valves on air lines and hydraulic lines penetrating the containment are also closed automatically. This feature ensures the closing of all containment vessel penetrations in case of a pressure build up of 5 psig within the containment vessel.

Consequently, the problem to be examined in this section is a false high pressure indication which results in the automatic closing of these valves.

2. Immediate Effects

The immediate effects may be understood best by examining the systems that are immediately affected by automatic closing of the isolation valves.

The primary system is serviced by several auxiliary systems which are rendered inoperable immediately. The valves in the following locations will close; the letdown line to the purification system, the intermediate cooling system, the continuous primary liquid sampling system and the gaseous waste system. Flow of water to the buffer seal surge tank through the buffer seal return will also be discontinued. This results in isolation of the primary system from its associated auxiliary systems with the exception of water being pumped from the buffer seal surge tank through the buffer seals and makeup line into the reactor vessel. Since there would be no water flowing into the buffer seal surge tank, its contents will be emptied into the primary

system within 6 to 12 minutes.

The valves that control flow in the feedwater lines leading to the steam generators will also close, thereby shutting off all feedwater flow to the steam generators. The quick-closing steam stop valves will also close, shutting off flow of steam from the steam generators.

Closing of the valves in the intermediate cooling water system will result in loss of cooling water flow to the pumps. Containment cooling and shield water cooling are likewise lost. The ability of the control rod hydraulic system to accomplish a scram is not impeded.

3. Operator Action

Upon acknowledging the automatic alarms for this condition, the operator will immediately check both pressure gauges to determine whether abnormal pressure actually exists in the containment vessel. He can do this by checking both pressure gauges mentioned previously, and in addition, the operator can check the pressure within the primary and auxiliary systems to see if a rupture has occurred. If a false pressure indication is found to be the cause of the situation, the normal functions of the system may be reestablished from the main console by overriding the pressure controller. This will open the valves that had been shut and permit the systems to resume their normal operation.

4. Operator Error

Although the normal procedure for the operator has been outlined, it is of interest to examine the possibility that the operator does not or cannot override the pressure controller. In this case, the immediate condition is assumed to be the discontinuance of flow in the auxiliary systems — a situation somewhat analogous to the accident previously described, "Loss of Instrument and Control Air Supply."

If the operator cannot override the air pressure controller, several steps should be taken to prevent damage to equipment. He should scram the reactor.

The emergency cooling system is not isolated by the false pressure indication and may be used to cool the reactor at the operator's discretion. If the situation cannot be corrected in a few minutes, the buffer seal charge pumps should be shut off to prevent damage in the event the surge tank is drained.

If the operator fails to take any action the steam to the turbine generators will be lost thereby causing a loss of electrical power to the main bus. The automatic scram resulting from this loss of power will shut down the reactor immediately. The primary pumps will lose power and be shut off. The emergency cooling system will also be activated automatically on loss of power to the main bus.

5. Conclusions

Improper functioning of the low range containment pressure instrument could result in automatic shutdown of the auxiliary systems. Normal operator action will restore these systems to original operating conditions. Failure of the operator to take any action will result in minor damage to the plant, but will not present a hazard to shipboard personnel. Malfunctioning of the high range containment pressure instrumentation presents no hazard since it is used for indication only.

F. LOSS OF POWER TO PRIMARY SYSTEM PUMPS

1. Nature of the Problem

The accident examined here results from a sudden partial reduction or a complete stoppage of coolant flow through the reactor core when the reactor is operating at an appreciable power level. The reduction of flow below that required to cool the reactor at the particular power level in question will naturally increase the temperature of the coolant as well as the reactor internals. Furthermore, there is danger of a combination of low coolant flow and excessive reactor power for that flow – a condition that might result in burnout of some of the hotter fuel pins.

A reduction of flow within the primary system may be due to an electrical failure of the power supply to the pumps, or it may be due to mechanical failure of a single pump.

2. Regions of Safe Operation

The potential consequences of single or multiple pump failures were determined by analyzing the reactor power at which fuel pin burnout or pellet center melting occurs in the hot channel, under various combinations of pump operation (Fig. 17). The burnout and pellet-center melting curves are based on steady-state operation at the indicated reactor power. Figure 17 also indicates the safe operating power for each pump combination. Starting at the operating power for the initial pump configuration, the loss of flow due to a pump failure can be traced as a horizontal line at constant power with decreasing flow. This approximation to the actual transient is conservative since the reactor power would decrease slightly following the flow decrease. This power reduction would be caused by the negative reactivity associated with an increase in average coolant temperature in the reactor core.

Starting at the safe operating power for each pump combination, horizontal lines can be drawn to indicate the path of the loss of flow transient. If the initial maximum operating power is less than the steady-state burnout power for the flow rate of the final pump configuration, no damage to the fuel elements is incurred. Table VI summarizes

esults of the various combinations of pump malfunctions based on examination of Figure 17. This table indicates that none of the pump failure combinations, with the exception of a complete loss of all pumping power, result in damage to the core either as a result of burnout or fuel melting. It appears, therefore, that complete loss of power to all pumps is the only case of concern.

3. Analysis of Complete Loss of Pumping Power

The loss of power to all pumps was studied for the normal case in which the reactor is scrammed by the pump failure monitor and for the extreme case in which no automatic safety action is initiated. A transient flow analysis of the primary system was conducted for the case of total loss of forced coolant flow.

The flow coastdown was simulated on an electronic analog computer at Battelle Memorial Institute.

4. Results of the Analysis

The flow transient used to describe coastdown of the coolant after complete loss of electrical power to the pumps is shown in Figure 18. Reactor power following initiation of the accident is plotted in Figure 19.

Comparison of the two reactor power curves shows that the initial drop in reactor power is accelerated by scram action. With no scram the decrease in power is slower, some time being involved in heating the primary coolant. In addition to supplying the negative reactivity to reduce reactor power, the coolant temperature increase must override the positive reactivity component resulting from the cooling of the fuel. For the worst case, the core outlet temperature reaches saturation approximately 3 seconds after the loss of pump power. Since the effect of void formation was not simulated on the Battelle analog, the indicated reduction in power after the first 3 seconds is slower than anticipated for actual reactor operation.

The average reactor power transient generated in the Battelle Memorial Institute analog then served as the basis for analyzing the effect of the total loss of coolant flow on the "hot channel" fuel pin. The variation of clad temperatures after the loss of flow is very sensitive to the assumption of the value of the heat transfer coefficient at the clad surface.

TABLE VI

RESULTS OF PUMP MALFUNCTIONS

Initial Condition		Final Condition		
Pumps Operating	Safe Operating Power, %	Pumps Operating	Power for Burnout or Fuel Melting, % of full power	Damage to Hot Channel Element
4	100	3	160	None
4	100	2 (2 loops)	160	None
4	100	2 (1 loop)	160	None
4	100	1	160	None
3	86	2 (2 loops)	160	None
3	86	1	160	None
2 (2 loops)	65	1	160	None
2 (1 loop)	59	1	160	None
Any Combination	---	0	160	Automatic Scram

Figure 20 gives the fuel pin surface and pellet-center temperatures for the case in which the heat transfer coefficient accounted for local and net boiling at the surface of the clad. The thermal conductivity of the mixed helium and fission products in the gap was assumed to be 0.03 Btu/hr-ft-°F. Because of substantial resistance to heat transfer in the gap between the pellet and the clad, the clad temperature seeks the temperature of the coolant for any reasonable value of surface heat transfer coefficient. In this case, the maximum clad surface temperature was 625°F. Figure 20 represents the conditions actually anticipated in the event of a complete loss of coolant flow.

To demonstrate the effect of surface heat transfer coefficient on the clad temperature, several conservative assumptions were applied to the transient loss of coolant flow analysis for the "hot channel". For the normal case in which the reactor is scrammed, the coolant temperature in the hot channel increased to the saturation temperature (617°F) during the first 10 seconds of the flow coastdown. Concurrently, the heat transfer coefficient is assumed to be reduced from 2670 to 20 Btu/hr-ft^2-°F, to reflect the combined effects of the flow reduction and the formation of a steam film around the fuel pin. The heat transfer coefficient of 20 Btu/hr-ft^2-°F represents the minimum value for complete film boiling. For the case in which no safety action is initiated, the time to saturation and minimum heat transfer coefficient in the hot channel is reduced to 3 seconds.

For the scram case, (there is a 0.5-second delay until the rods move inward) a maximum clad surface temperature of 1800°F is reached approximately 35 seconds after the loss of flow. Scram delay time is not a critical parameter in this accident since elimination of scram delay would simply reduce the peak clad temperature by 100°F. For the extreme case in which no safety action is initiated, the clad surface temperature reaches 2400°F in 30 seconds. Even in this case, the heat generation rate is so low at the time the clad reaches its peak temperature that film boiling heat transfer is sufficient to remove the heat. In these extremely conservative assumptions, any damage to the core would be minor and limited to the midplane of a few isolated hot channels. The resultant release of fission products and the corresponding radiation doses would be similar to those described in the fuel element failure section.

5. Conclusions

Although these results indicate that, with appropriate safety action, the fuel element cladding in the hot channel does not melt following the loss of coolant flow, cooling might be required to prohibit the formation and collection of steam at the top of the reactor vessel. It should be emphasized that the calculations of cladding temperatures in the hot channel were based on very conservative assumptions as evidenced by the foregoing discussion. The temperature of the hottest fuel pins will very likely fall considerably below the values reported for film boiling (see Fig. 20).

With either diesel generator operating, the primary coolant pumps can operate on the low speed windings to circulate primary coolant through the reactor, cooling it over a long period following the accident. Thus decay heat can be removed through the boilers and auxiliary condenser. If primary pumps cannot be used to circulate the coolant, two additional methods are available, depending upon the source of emergency electrical power. With either of the 750 kw diesels operating, decay heat can be removed through the letdown coolers and the intermediate coolers by operation of the buffer charge pump and one set of intermediate circulating pumps. With the 300 kw diesel generator operating, decay heat can be removed by the emergency cooling system. Thus, at least two methods are available for removing decay heat in case of a loss of turbine generator power. Since the auxiliary diesel generators and the emergency diesel generator are independent sources of electric power, a simultaneous failure of both sources is not likely to occur.

G. LOSS OF ALTERNATING CURRENT POWER

1. Nature of the Problem

Much of the normally functioning power plant equipment depends on electric power. This equipment includes pumps, compressors, pressurizer heaters, electrical control rod drive motors, and instrumentation. The loss of electric power to some of the items could result in the abnormal condition described here.

2. Electrical System Description

The generating equipment includes two 1500-kw turbogenerators, two 750-kw auxiliary diesel generators and a 300-kw emergency diesel generator. These generators, rated at 450 volts, 3 phase, 60 cycles, can deliver a 25% overload for 2 hours. The 300-kw emergency diesel generator, located on the Navigation Bridge Deck, will supply power to the emergency switchboard during emergencies in which both the main turbine generators and auxiliary diesel generators are inoperable. The turbine and auxiliary diesel generators, located in the machinery space, supply power to the main switchboard. For increased reliability, the main switchboard is divided into two sections, using a double bus arrangement. One turbine generator and one diesel generator are connected to each bus section of the main switchboard. The two bus sections are tied together by a normally closed circuit breaker. Figure 21 is a schematic diagram of the electrical system.

Most of the motor controllers for the electrical loads associated with the nuclear propulsion plant are grouped in Marine Group Control Centers which are further divided into two sections, following the double bus arrangement of the main switchboard. To maintain continuity of power to each section of the Group Control Centers, even in the event of a main switchboard bus fault, automatic bus transfer equipment is provided for Groups A, C, and D. The transfer arrangement for Control Group D is unique in that the automatic bus transfer switch transfers these loads from a main switchboard bus to the emergency switchboard bus. Groups A and C transfer between the two bus sections of the main switchboard. No automatic bus transfer equipment is provided for the primary coolant pumps, Group B.

A battery protected source is provided to supply power to those loads requiring an especially dependable power source, free of interruptions and surges. The battery protected source consists of two 40-kw, ac-dc motor generator sets fed from the 450 volt main switchboard; the d-c output is connected to a 125 volt d-c bus. The bus is also supplied by storage batteries. During normal operation, the batteries float on the line and are maintained fully charged. Two dc-ac motor generator sets are connected to the 125 volt bus; the a-c output provides the 120 volt, single phase, 60 cycle power for critical loads. Each motor generator set can carry the full demand load of the system that it serves. Non-critical 120 volt power is supplied by 450/120 volt transformers.

3. Normal Operation of the Electrical System

Operation of the two 1500-kw turbine generators is possible during periods of nuclear reactor operation and several minutes after reactor shutdown. Emergency diesel generator operation normally will be limited to emergency conditions when neither the turbine generators nor the auxiliary diesel generators are available. The 750-kw auxiliary diesel generators provide power for the following:

1. Reactor system warmup and reactor startup.
2. Normal operation (spare).
3. The loads required for decay heat removal after a scram or normal shutdown.
4. Emergency "take-home" power in case steam cannot be generated.

Power for the electrical system will be supplied by the two 1500-kw turbine generators, the two 750-kw auxiliary diesel generators serving as standby. The emergency switchboard in turn is normally supplied from the main switchboard by bus ties. During this period, any one of the following conditions will result in automatic starting of the two auxiliary diesel generators and automatic tripping of the ship's non-vital loads:

1. Overcurrent of either turbine generator
2. Tripping of either turbine generator breaker
3. Reactor scram
4. Reverse power relay indication on either turbine generator.

5. Turbine generator low lubricating oil pressure — either turbine
6. Turbine generator over speed trip — eith turbine
7. Bus undervoltage — Main Bus Section No. 1 or Section No. 2.

The auxiliary diesel generators have a 15-second startup capability, and the first unit to reach rated voltage is automatically synchronized with the turbine generators already in operation. Automatic paralleling of the second diesel then follows. The loss of all power from the main board will result in automatic startup of the Emergency Diesel Generator and the Emergency Cooling System.

The reliability of the Electrical System is essential to reactor safety, and therefore several unusual operating conditions that might possibly affect reactor safety are considered.

4. Loss of Turbine Generators

Considered first is the loss during normal operation of one 1500-kw turbine generator. This loss results in an automatic startup of the two 750-kw auxiliary diesel generators as well as an automatic tripping of the ship's non-vital loads. During this period, prior to auxiliary diesel generator startup, the remaining turbine generator load requirement is 1833 kw. One turbine generator can deliver a 25% overload (1875 kw) for 2 hours. Although the turbine generator is momentarily overloaded, the first diesel generator would normally parallel with the turbine generator in less than 15 seconds, and in turn would soon be followed by the second auxiliary diesel generator. The generating capacity of the two auxiliary diesels equals that of the turbine generator lost; thus continued nuclear plant operation is possible.

A rapid loss of power from both turbine generators could result in a momentary loss of power before the auxiliary diesel generators could be started. This loss of power to the main bus would result in an automatic scram of the reactor and automatic startup of the emergency diesel generator and the emergency cooling system. Although the loss of one or both turbine generators normally results in a startup of the diesel generators, the scram signal would result in an independent signal for automatic startup of two auxiliary diesel generators, and would also automatically trip the ship's non-vital loads. The main

propulsion turbine throttle also closes automatically upon scram. One diesel generator will be on line in less than 15 seconds after the scram signal was initiated, and the other, almost immediately after the first.

5. Loss of Auxiliary Generators

A very improbable consideration is that neither diesel generator would start automatically following the loss of one turbine generator. As mentioned, this would result in a period of overload on the single turbine generator — a condition easily corrected by reducing reactor power to the point where only two primary coolant pumps are required. This would reduce the total load on the turbine generator to 1373 kw, well within its normal rating. The diesels would then be started manually.

An even more remote possibility is that neither diesel generator would start upon loss of both turbine generators. Periodic startup of the diesels are scheduled to check their availability. Both units are not likely to fail unless the main engine room becomes flooded with water. In this case, the 300-kw emergency diesel generator and the emergency cooling system would start automatically to remove decay heat and provide emergency power for the ship's emergency loads.

Loss of forced coolant flow to the reactor as a result of the loss of electric power to the pumps is covered in Part I, Chapter II, F.

6. Conclusions

Since there are two independent turbine generators and two independent auxiliary diesel generators, the probability of loss of electric power to the main bus is very small. Even in the very unlikely event that both the normal and auxiliary sources of electric power become inoperable, the emergency diesel generator will be available to provide the necessary power for reactor cooling.

H. LOSS OF PRIMARY COOLING WATER

1. Nature of the Accident

The loss of coolant accident is postulated to occur as a result of the failure of primary piping or other associated high pressure piping components. The failure is considered to be ductile rather than brittle, as explained in Part I, Chapter II, I. The rapid loss of primary coolant will result from flashing (at 500°F and 1750 psi) through the breach in the high pressure system. Subsequent events are discussed in detail after the following discussion concerning some of the major assumptions made during analysis of this accident.

2. Assumptions

The following assumptions, resulting in the most pessimistic chain of events, describe the effect of the accident on the reactor and on the integrity of the containment vessel.

1. The maximum mechanical failure is a rupture in the primary system equivalent in size to the internal diameter of the largest primary pipe (12 9/16 in. ID or 125 $in.^2$)
2. The rupture occurs in a cold leg of the primary system, at a temperature of approximately 495°F.
3. The steam and water mixture from one boiler is released to the containment via the primary system rupture.
4. The reactor is at full power (69 MW) at the time of rupture.
5. The primary piping leak cannot be isolated from the rest of the primary system.
6. The reactor produces full power for 5 seconds after the rupture occurs. (This assumption was applied only to the analysis of containment integrity. To determine the thermal history of the reactor, a normal scram was assumed.)
7. The air-conditioning system ceases to remove heat from the air-steam mixture within the containment.
8. The steam generators cease to remove heat from the primary water.

3. Discussion of Assumptions

Reasons are given for the foregoing assumptions and the degree of pessimism used in the description of the incident.

1. The assumption of such a large break in the primary system piping results in the prediction of a very rapid rate of coolant loss from the primary system. Thus a large leak tends to magnify the rate of core heatup as well as the rate at which pressure builds up in the containment vessel. It is generally conceded that it is impossible to design and fabricate a high pressure system that will be 100% safe from failures. However, the probability of failure generally can be minimized through proper inspection and testing techniques, and through judicious care in establishing the number and location of penetrations to a safe level in areas where isolation of a leak is difficult or impossible.

2. The rupture was assumed to occur in the cold leg of the primary system because it is believed that the primary fluid would be released faster in this case. The cold leg would not flash as rapidly as the hot leg, thus permitting a more rapid flow of fluid through the ruptured areas. The more rapid flow of fluid permits a higher rate of increase of pressure within the containment vessel.

3. The assumption that steam and water from one boiler are released to the containment via the rupture adds energy to the containment vessel, resulting in a higher containment pressure.

4. The assumption that the reactor is at full power at the time of the incident is made because the decay heat and stored thermal energy in the fuel are larger than at reduced power levels. The results are (1) maximum addition of energy to the coolant, and (2) the maximum fuel temperature rise upon loss of coolant.

5. The inability to isolate the leak prevents minimizing the amount of leakage from the core, resulting in the most pessimistic core temperature history and the largest containment pressure buildup.

6. The assumption that the reactor power is not decreased for 5 seconds results in an additional supply of energy to the steam within the containment vessel. Actually, the reactor power would decrease rapidly because of void formation.

7. If the air-conditioning system is assumed to be inoperable, the only available cooling results from condensation of steam on the inside surface of the containment vessel and subsequent transfer of heat through the vessel and adjacent shielding. This assumption therefore tends to maximize the pressure buildup within the containment vessel.

8. If the steam generators are assumed to remove no more heat from the primary system, an additional heat sink has been ignored, again resulting in maximum pressure buildup.

4. Sequence of Events

Table VII presents a chronological sequence of events, predicated on the above assumptions. The time stated for each event is only approximate within the accuracy of the assumptions incorporated in the analysis but is intended to give some appreciation for the period of time involved during the incident.

5. Blowdown Analysis

The results of the blowdown analysis are presented in Figures 22 and 23. The pressure drops to the saturation pressure corresponding to the primary coolant temperature in a fraction of a second. During this short interval, liquid water discharges into the containment vessel through the failure. The flow is assumed to be similar to flow through an orifice. After the pressure reaches saturation, two-phase flow occurs and the flow rate through the rupture is computed by a quasi-steady-state method, based on a method reported by the American Society of Mechanical Engineers. The mass of coolant remaining in the primary system steadily decreases to approximately 5% of the initial mass. Unless the break occurs near the reactor inlet nozzle, the mass of coolant remaining is independent of the location of the rupture. The inertia forces resulting from the agitation of flashing and boiling of the coolant during the blowdown period tends to overcome any tendency toward separation of the steam and water. If the break occurred at the reactor inlet nozzle, the mass of water remaining would be less because some of the water remaining after the blowdown would drain to the containment vessel by gravity.

TABLE VII

SEQUENCE OF EVENTS DURING ACCIDENT

Time	Event
0	Longitudinal split occurs in largest primary coolant pipe, equivalent in area to the cross section of the pipe. (12 9/16 in. ID)
< 1 second	Primary system pressure drops to saturation (approximately 700 psi).
1 second	Reactor scram based on low primary system pressure. Dynamic forces on the control rods following a major leak do not hinder the scram action of the control rods since ample hydraulic pressure is always available to force the control rods into the core. In fact, the net force available to drive the rods into the core is greater because of the decrease in primary system pressure.
4 seconds	Approximately 20% of the primary system volume is filled with steam. Negative void coefficient due to flashing of water in the core assists control rods in reducing reactor power.
30 seconds	Pressure equilibrium reached between primary system and containment vessel. Water level in the reactor vessel is below lower grid plate so core is exposed to steam only.
500 seconds	Decay heat generated in the fuel elements heats the cladding to the melting temperature (2550°F) in the hottest region of the reactor core. Released fission products are assumed to be instantly dispersed uniformly throughout the containment vessel.
13 minutes	Radiation dose rate outside the secondary shielding reaches a maximum of approximately 20 r/hr.
2 hours	Assuming total gas leakage from the containment vessel at the rate of 0.2% of the containment free volume in 24 hours, the maximum downwind dose rate under worst atmospheric conditions is less than 10 mr/hr.

Figure 23 shows the increase in steam volume during the blowdown. It is interesting to note that after 10 seconds only 3% of the system mass has flashed to steam, but this mass occupies 50% of the primary system volume. As mentioned previously, this rapid increase in steam volume provides an inherent shutdown mechanism as a result of the negative void coefficient of reactivity.

6. Fuel Melting

Calculations have been made to determine the rate at which the fuel pins melt following the loss of coolant. During the first 10 seconds of the blowdown, heat is transferred from the fuel pin at a rate based on the heat transfer coefficient calculated from the Jens-Lottes correlation for ΔT_{sat}. For the remainder of the blowdown, during which time more than 50% of the primary system volume is filled with steam, heat transfer is based on a coefficient for low velocity steam. After 30 seconds the fuel pins were assumed to be insulated. The variations of clad surface and fuel temperature in the center average fuel pin following the primary system rupture are presented in Figure 24. The shutdown of reactor power results in a flattening of the temperature profile in the fuel pin. The average pellet central temperature decreases from its initial steady-state value of 1630°F until temperature equilibrium is established in the insulated fuel pin requiring approximately 2 minutes. Due to the decreased heat transfer coefficient, the clad surface temperature rises to meet the pellet central temperature. The fuel pin temperature increase following establishment of temperature equilibrium represents the storage of decay heat in the UO_2 pellet and stainless steel cladding. Since the melting point of the stainless steel cladding (approximately 2550°F) is significantly below the melting point of the UO_2 fuel pellet (approximately 5200°F) the temperature of the fuel pin will increase until the cladding melts. When a sufficient portion of the cladding melts, splinters and fragments of the UO_2 pellets will fall to the bottom of the reactor vessel.

Similar curves of fuel temperatures were computed for the range of initial power densities expected in the reactor. Based on the radial power distribution with no control rods in the core, corresponding to the end of the core lifetime, the time elapsed until pins at various radial positions reach 2550°F was computed (Fig. 25). The fuel pins

in the high flux regions of the reactor begin to melt 8 minutes after the system rupture. Theoretically, 50% of the pins will have failed approximately 40 minutes after the system rupture.

7. Post-Meltdown Criticality

As the fuel cladding continues to melt, the pins lose their structural integrity and the UO_2 drops out of the cladding, ultimately collecting in the lower portion of the pressure vessel. The possibility of a post meltdown fast reactor criticality was investigated for the most reactive configuration of a UO_2 sphere with an infinite reflector of stainless steel. The k_{eff} for this geometry was computed to be 0.657, using fast reactor calculations based on 10-group cross sections. There appears, therefore, to be no possibility of fast reactor criticality following the core meltdown.

In a similar fashion the possibility of a critical configuration at thermal, or epithermal, energies was analyzed. Figure 26 illustrates the dependence of the infinite multiplication factor on the ratio of water volume to metal volume in the UO_2 and melted cladding as it settles to the bottom of the vessel. It was assumed that the "egg crate" assembly did not melt but stayed in the core region. Assuming that all the fuel particles were spheres of the same size, with a water-to-metal ratio of 0.35, results in a value of about 0.95 for k_∞. There may be some random distribution of small fuel particles, which would result in a more highly compact mass, with a water-to-metal ratio below the value for uniform spheres. Figure 26 indicates that this would result in an even smaller value for k_∞. It therefore seems unlikely that a fast or thermal chain reaction could result from a core meltdown.

8. Post-Meltdown Heat Transfer

Decay heat continues to be generated after the fuel-clad mixture has melted and dropped to the bottom of the vessel. The next question of obvious concern is whether the fuel can attain a temperature high enough to melt through the reactor vessel. If it can the integrity of the containment vessel is jeopardized. Although it is recognized that the integrity of the containment vessel is a much more fundamental consideration than the reactor vessel, the best way to protect the containment

vessel from the effects of molten fuel in the event of this incident is to ensure that the fuel cannot melt through the reactor vessel.

Figure 27 illustrates the method used to permit adequate cooling of the lower head of the vessel. A gap of approximately 3/4 in. is provided between the insulation of the lower head of the vessel to permit cooling by natural circulation of water after loss of coolant from the primary system. The cooling water that becomes available for this purpose after the incident results from the primary water collecting to a depth of approximately 30 in. in the bottom of the containment vessel. Water enters at the bottom, center, of the insulation and flows upward in the gap between insulation and vessel to outlet holes placed 18 in. above the bottom of the insulation. The following criteria were established for determining the required flow areas:

1. Less than 30°F ΔT across the reactor vessel lower head during normal operation.
2. Sufficient cooling of the reactor vessel lower head after core meltdown to prevent failure of the reactor vessel at this point.

The required areas for flow were found to be:

1. From 5 to 10 $in.^2$ at the bottom of the lower head to provide an inlet passage for natural circulation of cooling water.
2. No less than 60 $in.^2$ exit area arranged in four or more equally spaced holes near the top of the lower head based upon a vertical distance of 18 in. from the bottom holes to the outlet holes.

Heat transfer calculations were based on the following assumptions:

1. Fluid enters at saturation temperature.
2. A heat rate equivalent to the maximum heat generation rate of the molten metal was used for steady-state calculations (see Fig. 29).

Figure 28 plots the available natural circulation head versus flow for a vertical hot leg of 18 in. Also shown on the same figure is the pressure drop versus flow for two cases fitting the above flow area requirements. It can be seen that 37,000 lb/hr of flow can be maintained with a 5 $in.^2$ entrance area and a 60 $in.^2$ exit area.

Figure 29 plots the heat generation rate of the molten metal at the bottom of the reactor head, and can be seen to exhibit a maximum point. This is a result of two opposing effects: the mass of molten metal increasing with time, and the rate of decay of heat generation within the metal decreasing.

It is reasonable to believe that this method can maintain adequate cooling, and prevent the reactor vessel from reaching temperatures that would result in damage to the vessel.

9. Pressure Within the Containment

Before analyzing the radiation hazard resulting from core meltdown, it is of interest to examine the degree of integrity and adequacy of the containment vessel. The reader is referred to the previously mentioned assumptions upon which the analysis of this accident was made.

The heat content of the water in various primary and secondary components is presented in Table VIII.

If an instantaneous release of primary and secondary fluid in the containment vessel is assumed, a hypothetical maximum pressure rise can be computed. However, during the finite time in which this release takes place, heat will be absorbed by the walls of the containment vessel and the actual pressure will rise to a value lower than the hypothetical maximum pressure. The principal materials that act as heat sinks are given in Table IX. During the first seconds following the rupture, heat absorption in the relatively cold (130°F avg.), uninsulated steel will be very rapid because of the high condensing film coefficients and the relatively high thermal diffusivity of steel. After longer periods of time, approximately 15 minutes, heat absorption in the lead and polyethylene and heat transfer through the walls of the vessel become important. The weight of the materials used as heat sinks in Table IX are in most cases lower than the actual weight of the components presented in Volume I. This leads to a more conservative pressure calculation.

The heat released and heat absorbed by various components during the 30-second blowdown period are presented in Table X.

The calculated pressure and temperature within the containment vessel following the hypothetical maximum incident is presented in Figure 30. The maximum pressure of 173 psig is reached when all of the fluid is released.

The contribution of other potential sources of energy and their effects are negligible or the probability of their occurring is very small. The potential effect of these energy sources on the containment pressure is dependent upon the rate at which the energy is released. Table XI presents the potential energy sources and their individual effect on containment pressure. The energy associated with the nuclear burst is a hypothetical maximum value based on the instantaneous introduction of all the excess reactivity available at operating temperature and pressure. Although the energy is shown in Table XI for a "Nuclear Burst", this is included solely for illustrative purposes, since there is no physical means of accomplishing such a reactivity introduction.

The design of the containment shell and all its penetrations comply with the American Society of Mechanical Engineers code for Unfired Pressure Vessels. Vessels designed to this code have been tested to demonstrate adequate strength at pressures 50% above the design pressure. Therefore, there is good reason to believe that this vessel would contain the maximum pressure. Additional potential sources of energy will not produce pressure increments greater than 30%. Since these pressures are well within the anticipated failure pressure for the vessel, no adverse effects are anticipated from the additional potential sources of energy.

TABLE VIII

HEAT CONTENT AND WEIGHT OF PRIMARY AND SECONDARY FLUIDS

Primary Fluids	Weight, lb	Heat Content, Btu
Steam Generator	8,835	4,396,000
Cold Leg	4,147	2,010,000
Reactor	46,746	23,256,000
Hot Leg	3,228	1,647,000
Pressurizer	2,989	2,127,000
	65,945	33,436,000
Secondary Fluid	**Weight, lb**	**Heat Content, Btu**
Steam Generator	8,000	3,560,000

TABLE IX

WEIGHT OF MATERIALS ACTING AS HEAT SINKS

Material	Weight, lb
Containment Vessel	590,000
Structural Steel	79,700
Lead Shielding	1,036,000
Polyethylene Shielding	82,200
Neutron Shield Tank Lead	49,500

TABLE X

TOTAL HEAT RELEASED AND ABSORBED DURING BLOWDOWN PERIOD

	Released, Btu	Absorbed, Btu
Primary System	33,436,000	
Secondary System	3,560,000	
Decay Heat	189,000	
Full Power for 5 seconds	327,000	
Stored Heat in Fuel Pins	426,000	
Containment Vessel Walls		2,150,000
Structural Steel		1,148,000
Neutron Shield Tank		127,000
Air Within Vessel		133,000
Total	37,938,000	3,558,000

TABLE XI

SOURCES OF ADDITIONAL ENERGY DURING MAXIMUM HYPOTHETICAL ACCIDENT

	Energy, Btu	Resultant Total Containment Pressure, psig
1. Burning of all hydrogen in primary coolant	50,300	173
2. Secondary side of second boiler	3,560,000	190
3. Heat from hot components in primary system	Negligible until after maximum pressure is reached	
4. Higher initial primary system temperature (538°F)	2,440,000	185
5. Oxidation of molten fuel element cladding after meltdown	5,250,000	Slow process with no contribution for initial 500 seconds
6. Nuclear burst (7% step increase in excess reactivity)	9,200,000	220

10 Containment Vessel Leak Test

A containment vessel leak test will be performed prior to initial loading of fuel into the reactor. The test involves a number of complications because of the high pressures of interest and the required accuracy of results. The test will be performed with an internal pressure of 30 psig and a sensitivity sufficient to demonstrate that the leakage is less than 0.2% of the contained air per day. Although the calculated average pressure over an 8-hour period following the hypothetical maximum accident is 60 psig, the leak test will be performed at a lower pressure to insure that equipment within the containment vessel is not damaged during testing.

The sensitivity of leak rate data is affected by the variations in atmospheric temperature and pressure conditions. To minimize these effects, an internal pressure reference system will be used for comparison with containment vessel pressure. The reference system will be comprised of pressure tubing inside the containment vessel which will be subject to the same temperature variations as the containment. The pressure difference between the reference tubing and containment air will be insensitive to changes in barometric pressure.

Since certain pneumatic valve operators and instrumentation will have to be disconnected, it does not appear feasible to repeat this leak test at 30 psig with systems at normal operating conditions. Similar full-scale leak tests could be performed only at intervals coinciding with complete refueling. However, it is intended to check penetration integrity with periodic leak tests and to check the personnel airlock after using it for entry.

For additional information regarding vessel design and penetrations, the reader is referred to Volume I of the Safety Assessment.

11. Analysis of Radiation Hazards

Following the melting of fuel elements, the gaseous fission products and smaller percentages of the non-volatile fission products,

which have been generated and accumulated within the fuel elements, are released. The exact quantity and rate of release are difficult to evaluate. Experimental work reported by the Bettis Laboratory indicates that the percentage of fission products released is very dependent upon the UO_2 pellet density and temperature.

Pressure equilibrium between the primary system and containment vessel is established before the release of fission products that would result in a time delay between failure of the fuel element and dispersal of fission products throughout the containment vessel. However, in these analyses the fission products are assumed to be uniformly distributed within the containment vessel at the moment of release from the fuel rods.

It was assumed that the fission products were released when the cladding temperature reached the melting point, 2550°F. Figure 31 presents the percentage of the total core activity that is released, which is related to the number of rods that have failed, as a function of time after the onset of the incident for different periods of operation at 100% power. The fission-product inventory was taken to be that for 600 days of operation at 69 MW.

Calculations of the percentage of the total activity that would be released were based on 100% of the fuel rods having failed within 2 hours after the incident. It was assumed that failed fuel elements would release fission product percentages shown in Table XII.

These assumed release values for both the volatile and the nonvolatile fission products are conservative when compared with experimental results reported by the Oak Ridge National Laboratory from tests on the release of fission products from molten reactor fuels.

The dose rate from activity released to the containment vessel is shown in Figure 32. Further details of shipboard radiation levels are given in Part II, Chapter X, of this volume.

TABLE XII

FISSION PRODUCT RELEASE

Element	% Released
xenon, krypton rubidium, iodine bromine, cesium	100
strontium, barium	10
tellurium selenium, antimony, europium, molybdenum, technetium, samarium	1
yttrium, lanthanum, cerium, praseodymium, neodymium, promethium, zirconium, niobium, ruthenium, rhodium	0.1

The leakage rate of the containment vessel was taken as 0.2% of the free volume per day, the maximum acceptable leakage rate. It was assumed that all leakage would be into the reactor compartment and that the containment would be continuously ventilated to the top of the Fore Top Mast (height above water level about 90 ft with ship loaded). It was further assumed that the reactor compartment discharge would be filtered to 99.9% efficiency for all activity except the noble gases, xenon and krypton.

I. MISSILE FORMATION

1. Nature of the Problem

The presence of a high pressure coolant within the containment vessel presents possibility of missile formation. Therefore this probability and other modes of missile formation are considered here.

2. Pipe Failures

The following conclusions regarding industrial experience with pipe and equipment failures in stationary and marine plants are based on a survey of marine experience during the past 10 years:

1. With the exception of two cases involving the use of carbon molybdenum steel, there have been no complete circumferential failures in large diameter pipes. In no case was there any record of high velocity fragments resulting from pipe failures.

2. The only machinery or equipment failures that have generated missiles of any consequence were main propulsion and turbine generator rotor failures.

As to brittle fracture of high pressure piping, test performed at the Naval Research Laboratory showed the following:

1. A stress raiser of some kind, such as a crack or sharp notch must be present.

2. A stress must result in the vicinity of the stress raiser and be sufficient to cause localized yielding in that area.

3. A service temperature below the nil ductility, or transition temperature, of the material is necessary.

If all three of these conditions are present, a brittle fracture is possible; if one or more is absent, a brittle fracture will not occur.

Stress raisers like sharp edges and notches have been eliminated in design of all high pressure equipment in the containment vessel. Cracks or sharp notches encountered during manufacturing are eliminated through rigid inspection and the use of high quality assembly and welding methods. During high pressure operation, system temperatures will be well above the nil ductility temperature. Even during startup, the primary system temperature will be brought

up to 200° F before the system pressure is increased above 100 psi. The maximum nil ductility temperatures are expected to range from 50° F in the unirradiated condition to possibly 125° F in the part of the reactor vessel surrounding the core after long irradiation time. Therefore, there is adequate temperature margin to prevent operation below the nil ductility point.

All primary and secondary system equipment is designed to existing standards of the United States Coast Guard and the American Bureau of Shipping. In view of the maritime experience with systems designed to these standards, any pipe failures could be expected to occur as tears and splits with no fragments formed.

3. Other Failures

The largest rotating items in the containment vessel are the primary loop coolant pumps. The moderate size of these pumps, and the strength of their enclosing steel casings precludes the possibility of any part piercing the steel shell of the pump. The rotating mass of the containment cooling fans is small and the cross-sectional area is large, so their failure could not rupture the containment vessel plates.

The only likely source of missiles within the containment vessel are the small accessory components such as sampling valves, instrument wells, and gauges, any of which might break loose and be propelled by the high pressure primary coolant. The effects of such missiles are amenable to analysis using empirical formulas developed from experiments reported by the U. S. Army Ballistics Research Laboratory and the Stanford Research Institute, Palo Alto, California.

In this analysis, a resistance thermometer in the primary system was assumed to be released by instantaneous failure around the full circumference of the weld to the primary piping and accelerated to a maximum velocity because of impingement of the primary coolant. The following data were computed to examine the possibility of penetration of the containment vessel.

Weight of the missile, lb	11
Projected area of the missile, ft^2	0.0534
Velocity of coolant at the orifice, ft/sec	346
Attained velocity of the missile, ft/sec	
(12 degree jet angle)	177
(60 degree jet angle)	106

Figure 33 compares the results of the empirical formulas developed at Ballistics Research Laboratory and Stanford Research Institute. The specific energy of the missile (kinetic energy $\frac{1}{d}$ diameter) for the conditions given is 1900 ft-lb/in. This figure shows that the minimum plate thickness required to resist penetration by the reference missile is predicted by Ballistics Research Laboratory to be about 0.15 in. and by Stanford Research Institute to be approximately 0.25 in. In view of the inexact nature of the problem, this agreement appears to be reasonably good. The minimum wall thickness of the SAVANNAH containment vessel is at the hemispherical end, consisting of 1 1/4-in. plate. Toward the cylindrical portion of the vessel, the plate is gradually increased in thickness to 3 3/4 in. Penetration by this or a similar missile is not likely in view of the large thickness factor between predicted minimum value for penetration resistance and the actual plate thickness.

4. Summary

The possibility of a brittle failure of high pressure piping resulting in missile formation is very improbable based on past industrial experience as well as the design, procurement, manufacturing, testing and inspection methods and the conditions of operation. High speed rotating machinery presents no hazard to the containment vessel. Smaller missile formation, although conceivable, will not penetrate the vessel according to experiments performed at two laboratories specializing in these types of studies.

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FIG. 10: DOSE RATE WITHIN CONTAINMENT

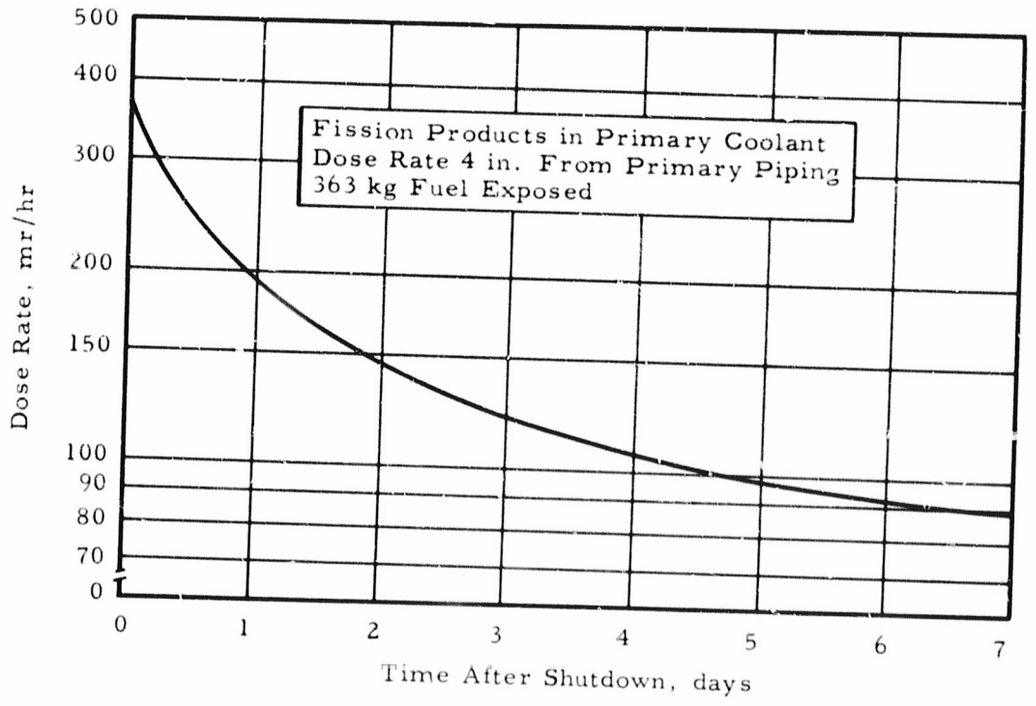

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FIGURE 10

FIG. 11: REACTOR COMPARTMENT VENTILATION SYSTEM

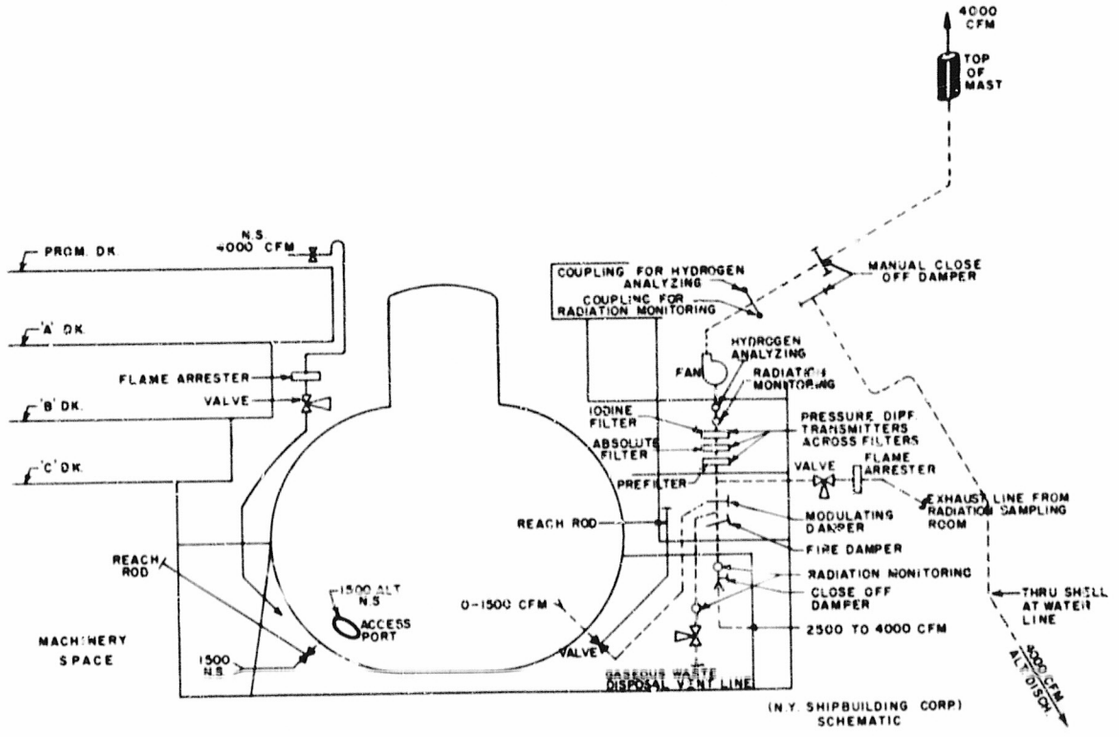

FIGURE 11

N. Y. Shipbuilding Corp.

FIG. 12: EFFECTIVE MULTIPLICATION FACTOR Vs MODERATOR TEMPERATURE

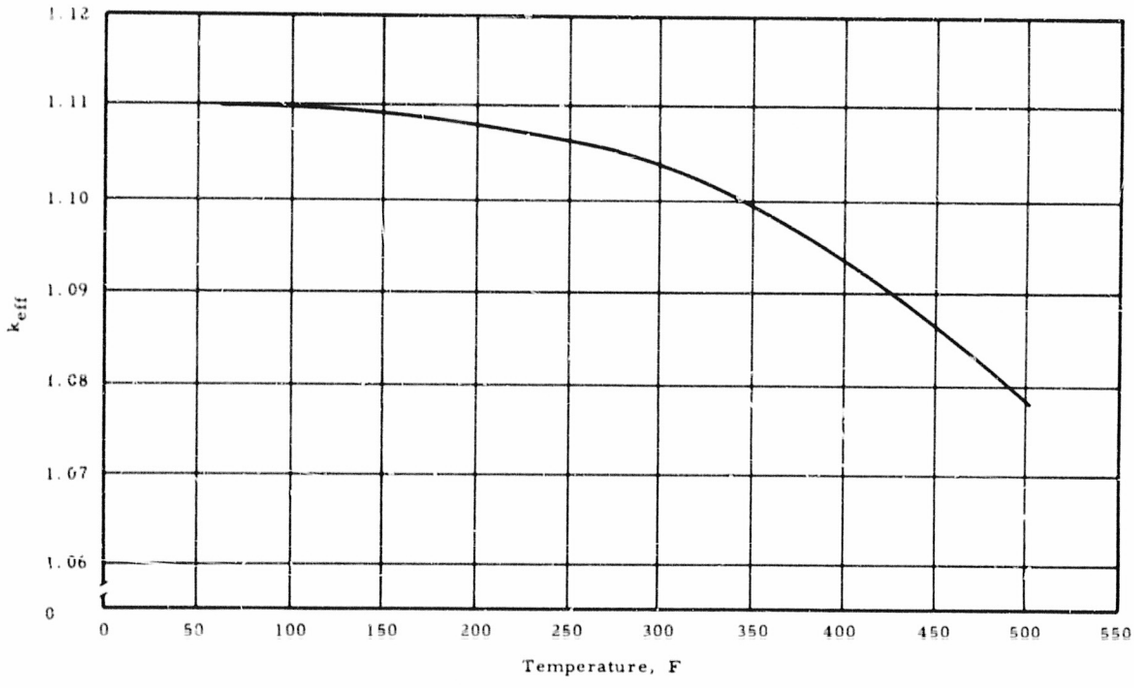

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FIGURE 12

FIG. 13: TEMPERATURE-REACTIVITY EFFECTS Vs CORE LIFE

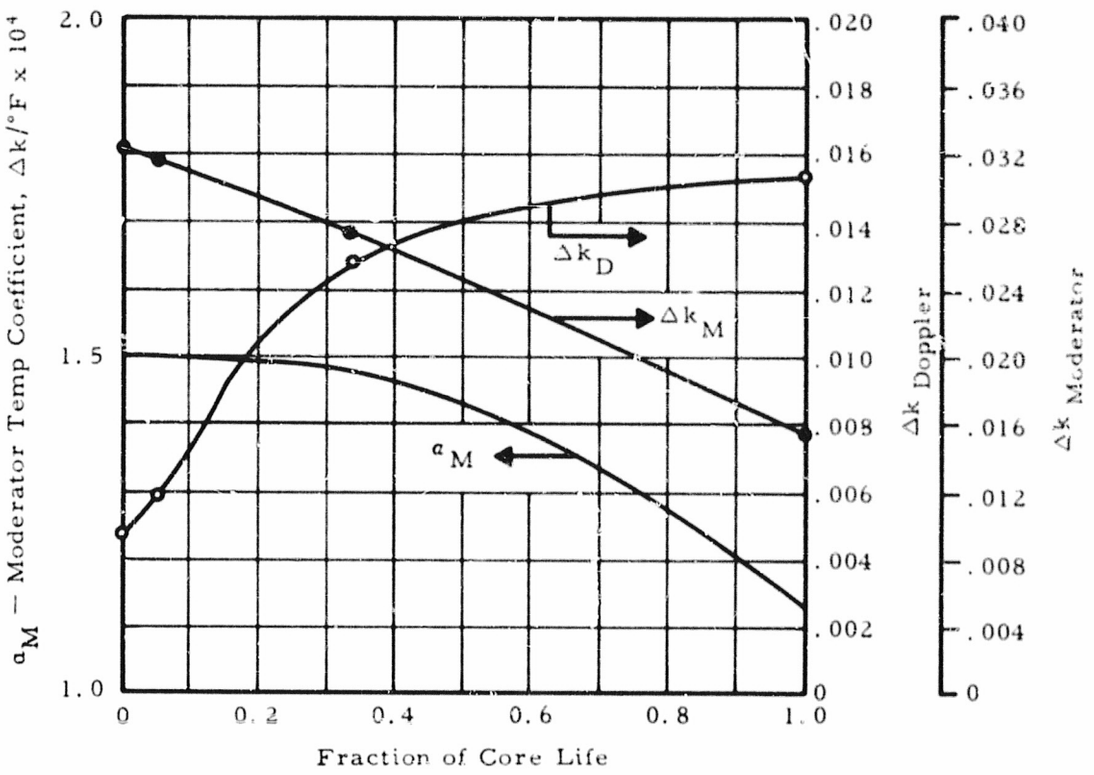

FIGURE 13

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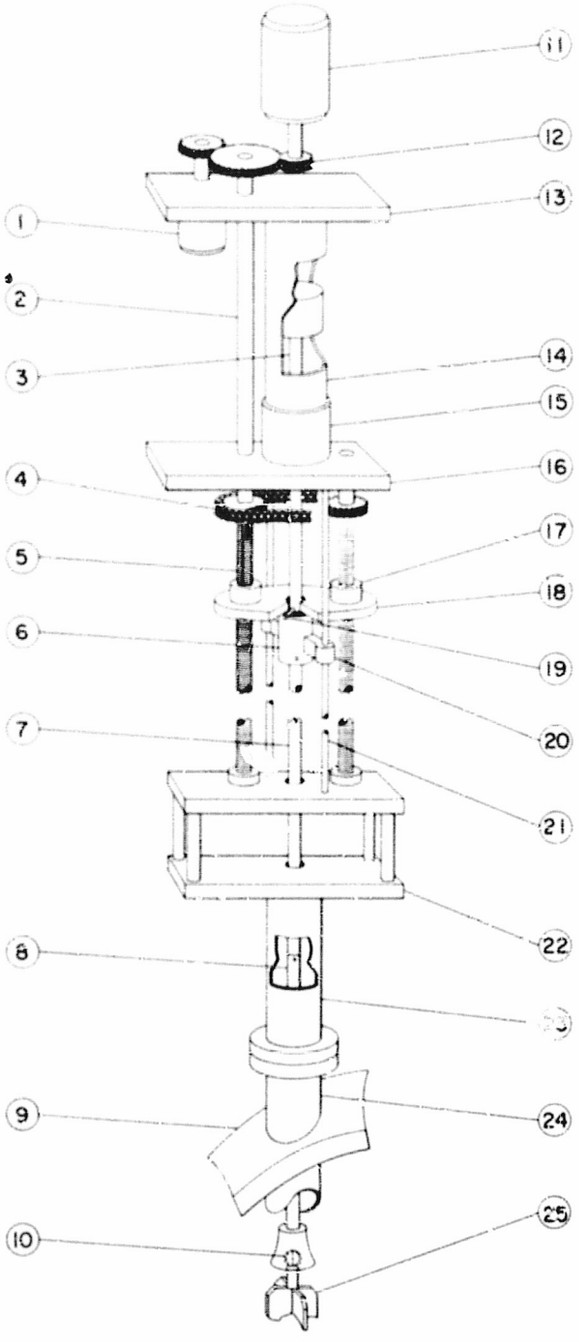
FIG. 14: CONTROL ROD DRIVE



(1) Position Indicator
(2) Lead Screw Jack
(3) Hydraulic Piston Shaft
(4) Chain and Sprocket Drive Assembly
(5) Lead Screw
(6) Coupling Piece
(7) Buffer Seal Shaft
(8) Buffer Seal Extension Shaft
(9) Reactor Closure Head
(10) Handling Knob
(11) Rod Drive Motor
(12) Reduction Gear
(13) Upper Mechanism Flange
(14) Hydraulic Cylinder
(15) Dash-Pot
(16) Alignment Plate
(17) Drive Nut
(18) Nut Carriage
(19) Drive (Actuator) Pin
(20) Latch Arm
(21) Tie Rod
(22) Thimble Platform
(23) Buffer Seal
(24) Control Rod Nozzle
(25) Control Rod

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FIGURE 14

FIG. 15: CONTROL ROD SCRAM SYSTEM

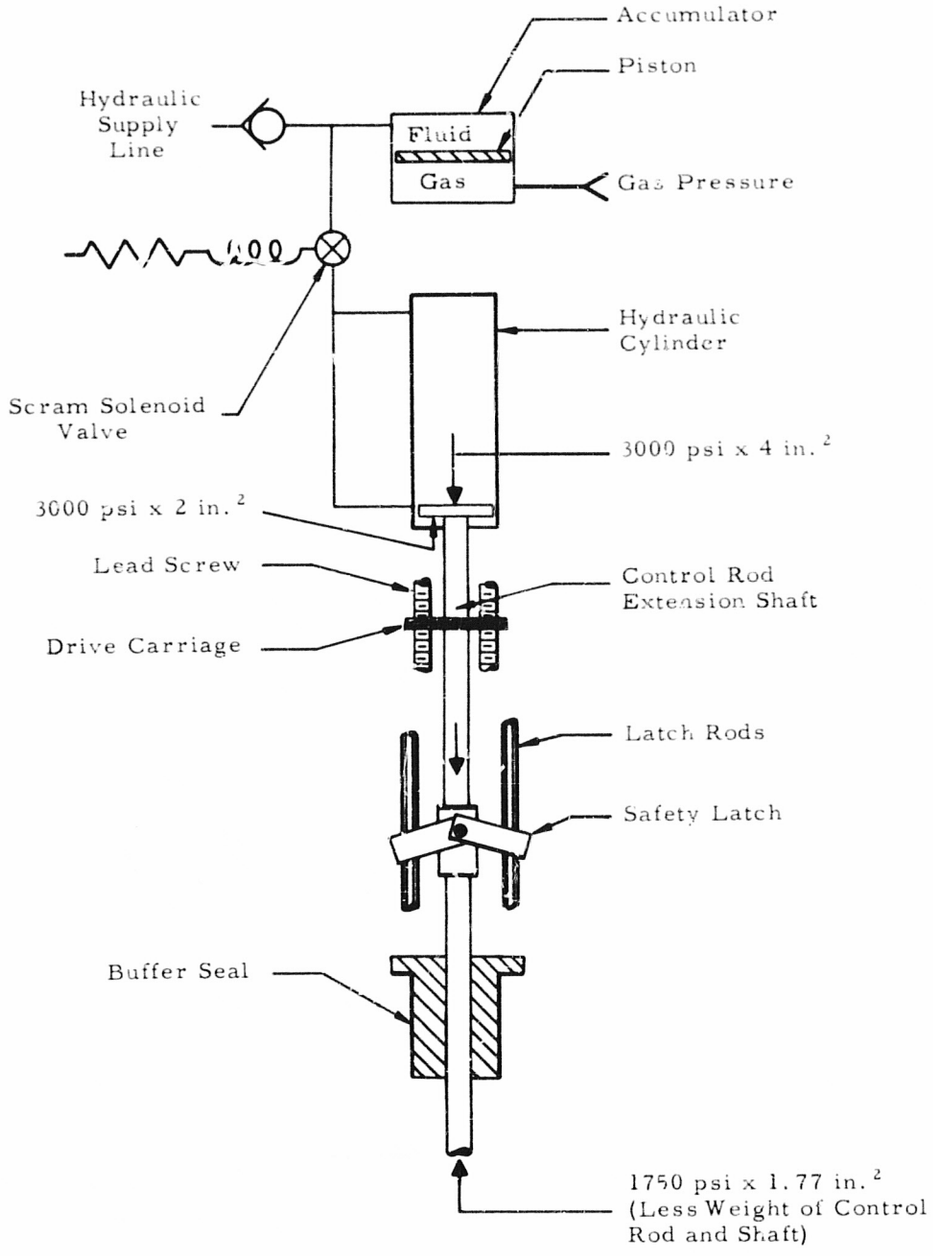

FIGURE 15

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FIG. 16: INSTRUMENTATION AND CONTROL AIR SUPPLY

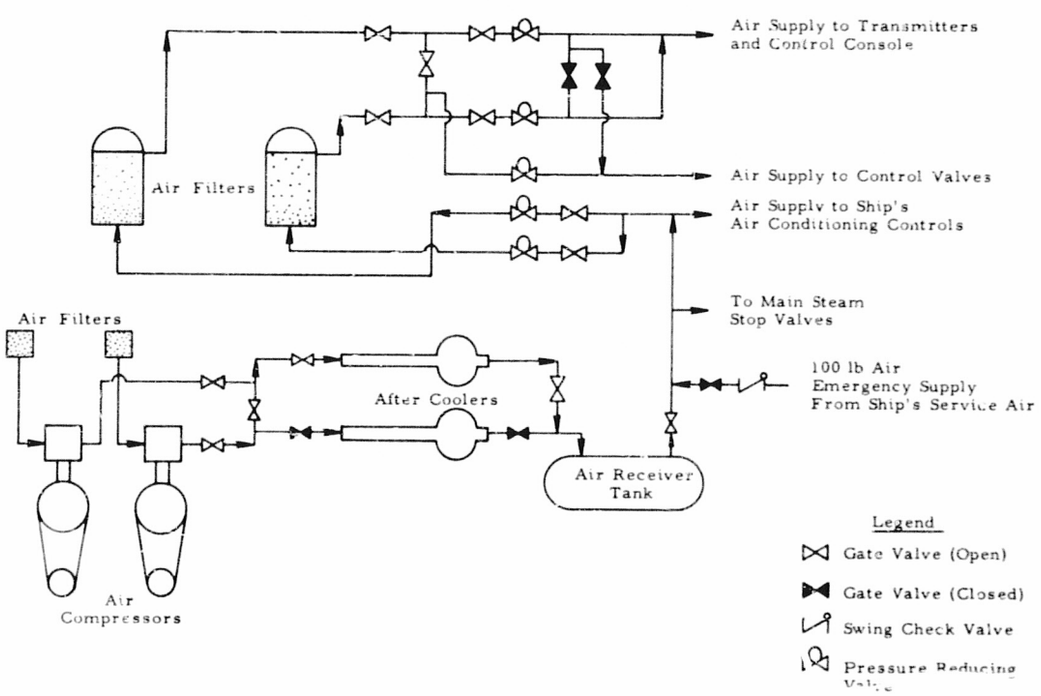

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FIGURE 16

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4

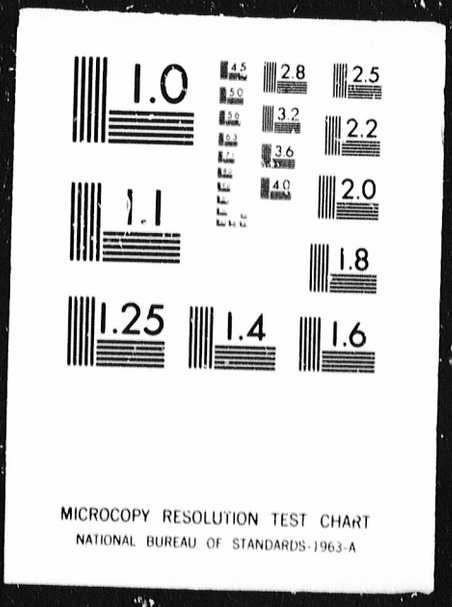

FIG. 17: LIMITS ON REACTOR OPERATION

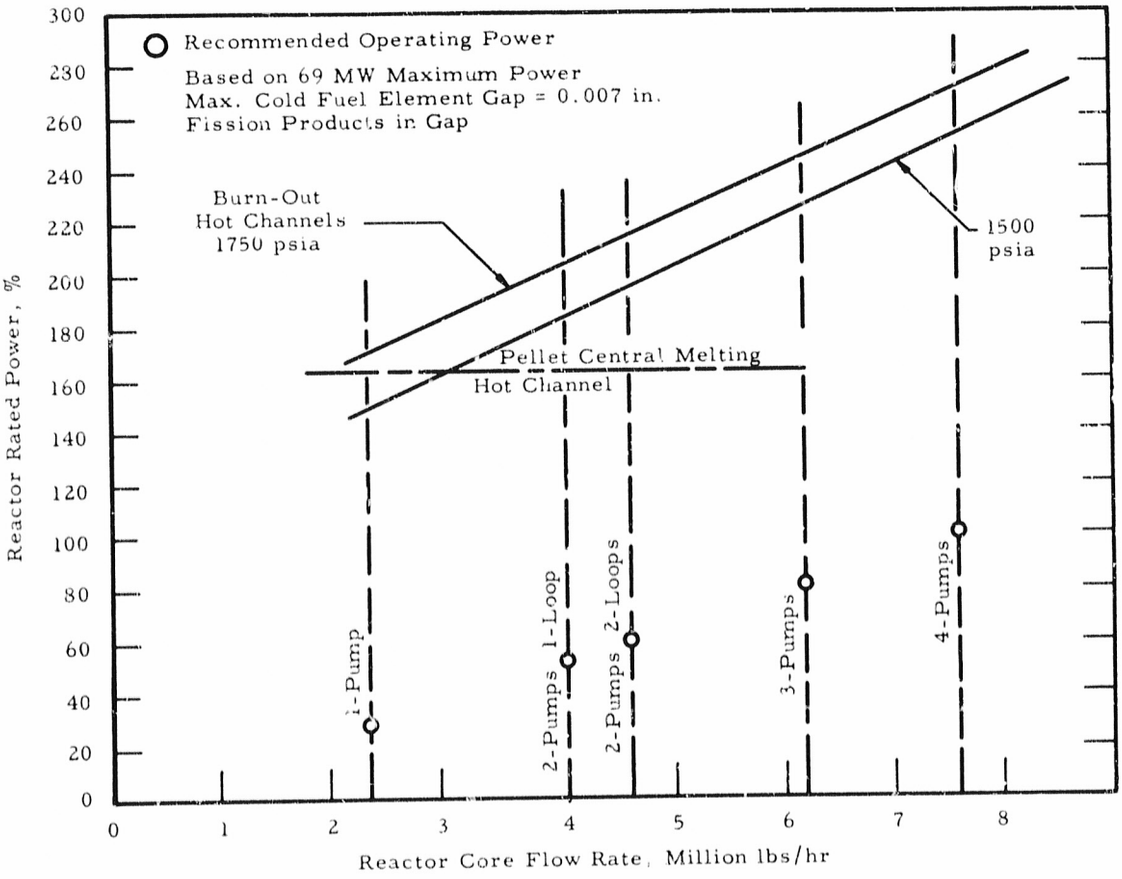

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FIGURE 17

FIG. 18: FLOW COASTDOWN DUE TO LOSS OF PUMPING POWER

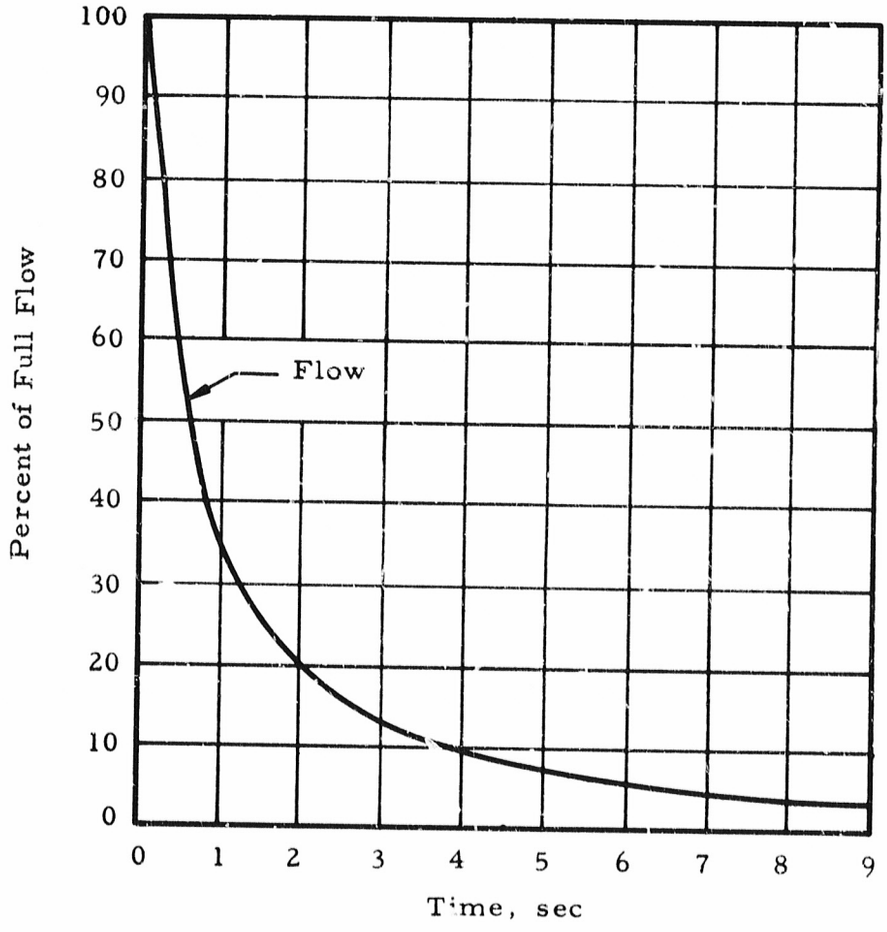

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FIGURE 18

FIG. 19: REACTOR POWER REDUCTION DUE TO LOSS OF PUMPING POWER

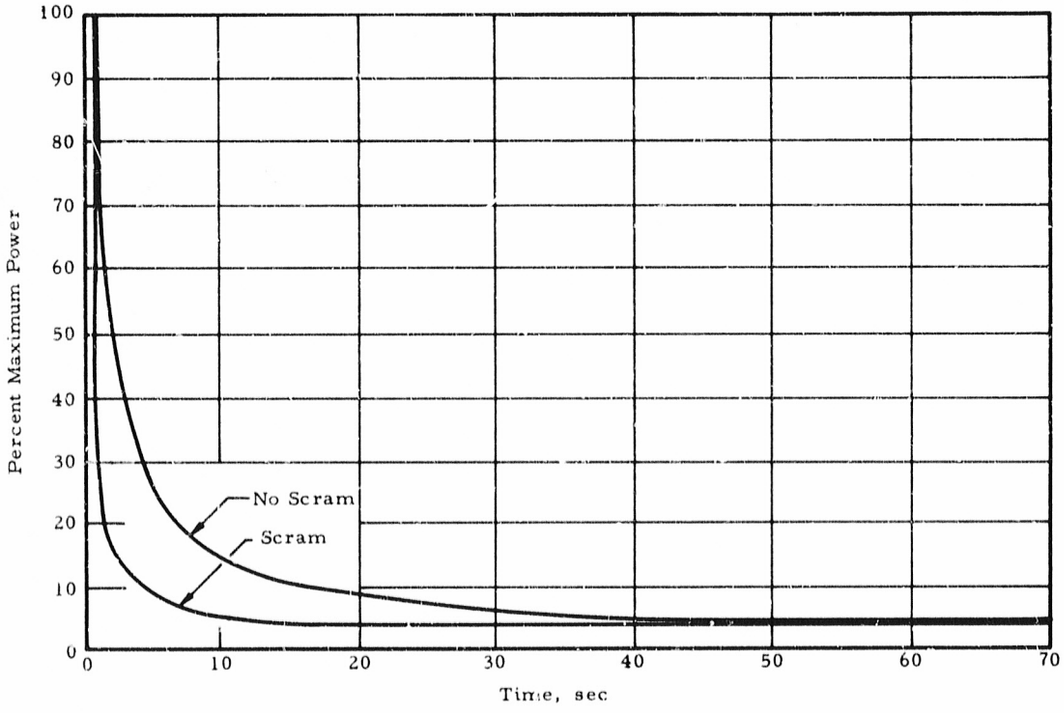

FIGURE 19

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FIG. 20: FUEL AND CLAD TEMPERATURES RESULTING FROM COMPLETE LOSS OF COOLANT FLOW

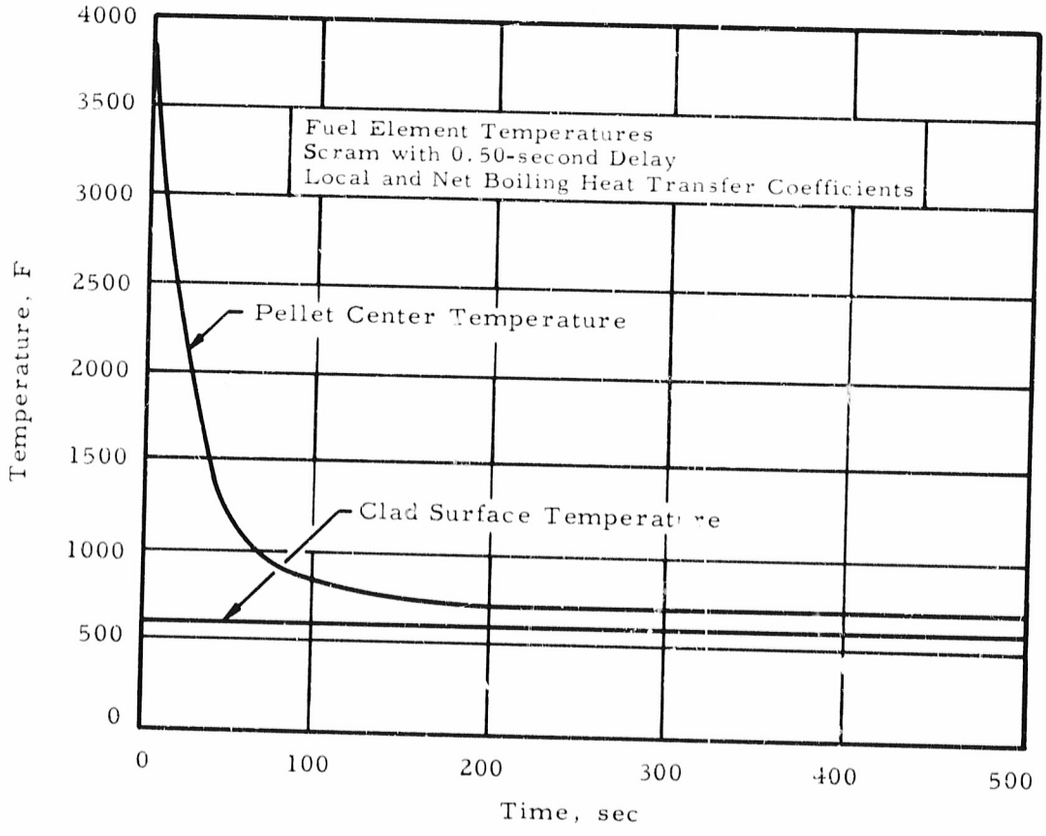

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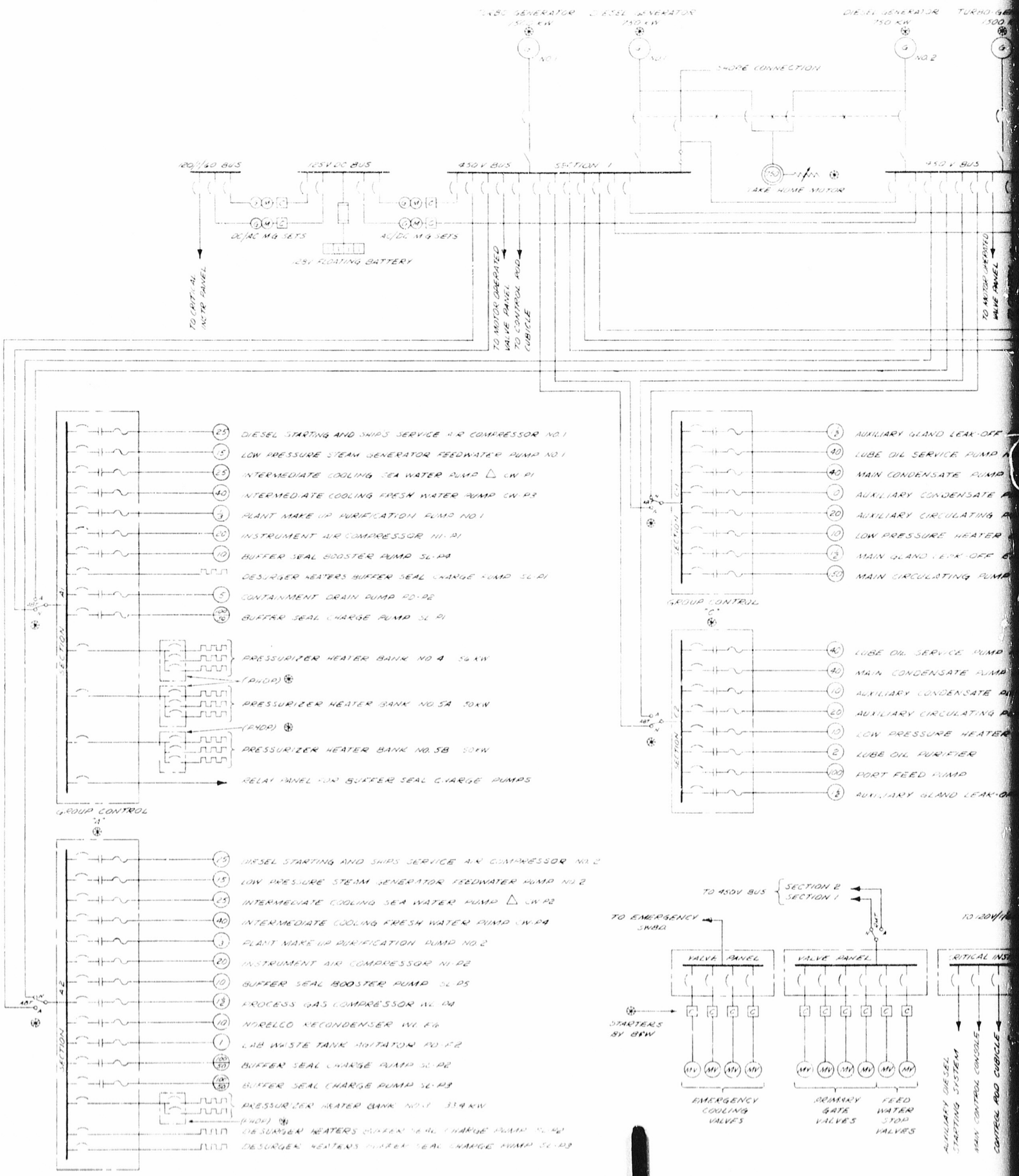
FIGURE 20



TURBO GENERATOR 1500 KW
DIESEL GENERATOR 750 KW
DIESEL GENERATOR 750 KW
TURBO-GE 1500
NO. 1
NO. 2
SHORE CONNECTION
120/1/60 BUS
125V DC BUS
450 V BUS
SECTION 1
450 V BUS
TAKE HOME MOTOR
DC/AC MG SETS
AC/DC MG SETS
128V FLOATING BATTERY
TO CRITICAL INSTR PANEL
TO MOTOR OPERATED VALVE PANEL
TO CONTROL ROD CUBICLE
TO MOTOR OPERATED VALVE PANEL
DIESEL STARTING AND SHIPS SERVICE AIR COMPRESSOR NO. 1
LOW PRESSURE STEAM GENERATOR FEEDWATER PUMP NO. 1
INTERMEDIATE COOLING SEA WATER PUMP △ CW-P1
INTERMEDIATE COOLING FRESH WATER PUMP CW-P3
PLANT MAKE UP PURIFICATION PUMP NO. 1
INSTRUMENT AIR COMPRESSOR NI-P1
BUFFER SEAL BOOSTER PUMP SL-P4
DESURGER HEATERS BUFFER SEAL CHARGE PUMP SL-P1
CONTAINMENT DRAIN PUMP PD-P2
BUFFER SEAL CHARGE PUMP SL-P1
PRESSURIZER HEATER BANK NO. 4 56 KW
(PHOP)
PRESSURIZER HEATER BANK NO. 5A 50 KW
(PHOP)
PRESSURIZER HEATER BANK NO. 5B 50 KW
RELAY PANEL FOR BUFFER SEAL CHARGE PUMPS
SECTION A1
GROUP CONTROL "A"
AUXILIARY GLAND LEAK-OFF
LUBE OIL SERVICE PUMP
MAIN CONDENSATE PUMP
AUXILIARY CONDENSATE
AUXILIARY CIRCULATING
LOW PRESSURE HEATER
MAIN GLAND LEAK-OFF
MAIN CIRCULATING PUMP
SECTION C1
GROUP CONTROL "C"
LUBE OIL SERVICE PUMP
MAIN CONDENSATE PUMP
AUXILIARY CONDENSATE
AUXILIARY CIRCULATING
LOW PRESSURE HEATER
LUBE OIL PURIFIER
PORT FEED PUMP
AUXILIARY GLAND LEAK-OF
SECTION C2
DIESEL STARTING AND SHIPS SERVICE AIR COMPRESSOR NO. 2
LOW PRESSURE STEAM GENERATOR FEEDWATER PUMP NO. 2
INTERMEDIATE COOLING SEA WATER PUMP △ CW-P2
INTERMEDIATE COOLING FRESH WATER PUMP CW-P4
PLANT MAKE UP PURIFICATION PUMP NO. 2
INSTRUMENT AIR COMPRESSOR NI-P2
BUFFER SEAL BOOSTER PUMP SL-P5
PROCESS GAS COMPRESSOR WL-P4
NORELCO RECONDENSER WL-F6
LAB WASTE TANK AGITATOR PD-F2
BUFFER SEAL CHARGE PUMP SL-P2
BUFFER SEAL CHARGE PUMP SL-P3
PRESSURIZER HEATER BANK NO. 1 33.4 KW
(PHOP)
DESURGER HEATERS BUFFER SEAL CHARGE PUMP SL-P2
DESURGER HEATERS BUFFER SEAL CHARGE PUMP SL-P3
SECTION A2
TO 450V BUS
SECTION 2
SECTION 1
TO EMERGENCY SWBD
TO 120V/1/
VALVE PANEL
VALVE PANEL
CRITICAL INS
STARTERS BY B&W
EMERGENCY COOLING VALVES
PRIMARY GATE VALVES
FEED WATER STOP VALVES
AUXILIARY DIESEL STARTING SYSTEM
MAIN CONTROL CONSOLE
CONTROL ROD CUBICLE

L SYSTEM

VOICATES H.P. SIZE)

OTOR OPERATED VAL

MOTOR

ERATOR SET

ETER

TRANSFER SWITCH

NUS TRANSFER

NUAL TRANSFER SWI

EAKER

ONNECT

ROLLER WITH OVERL

ER

INTERLOCK

INTERLOCK

LINKS

MOTOR

ER HEATER DISTRIB

BLANK PAGE

FIG. 21: ELECTRI

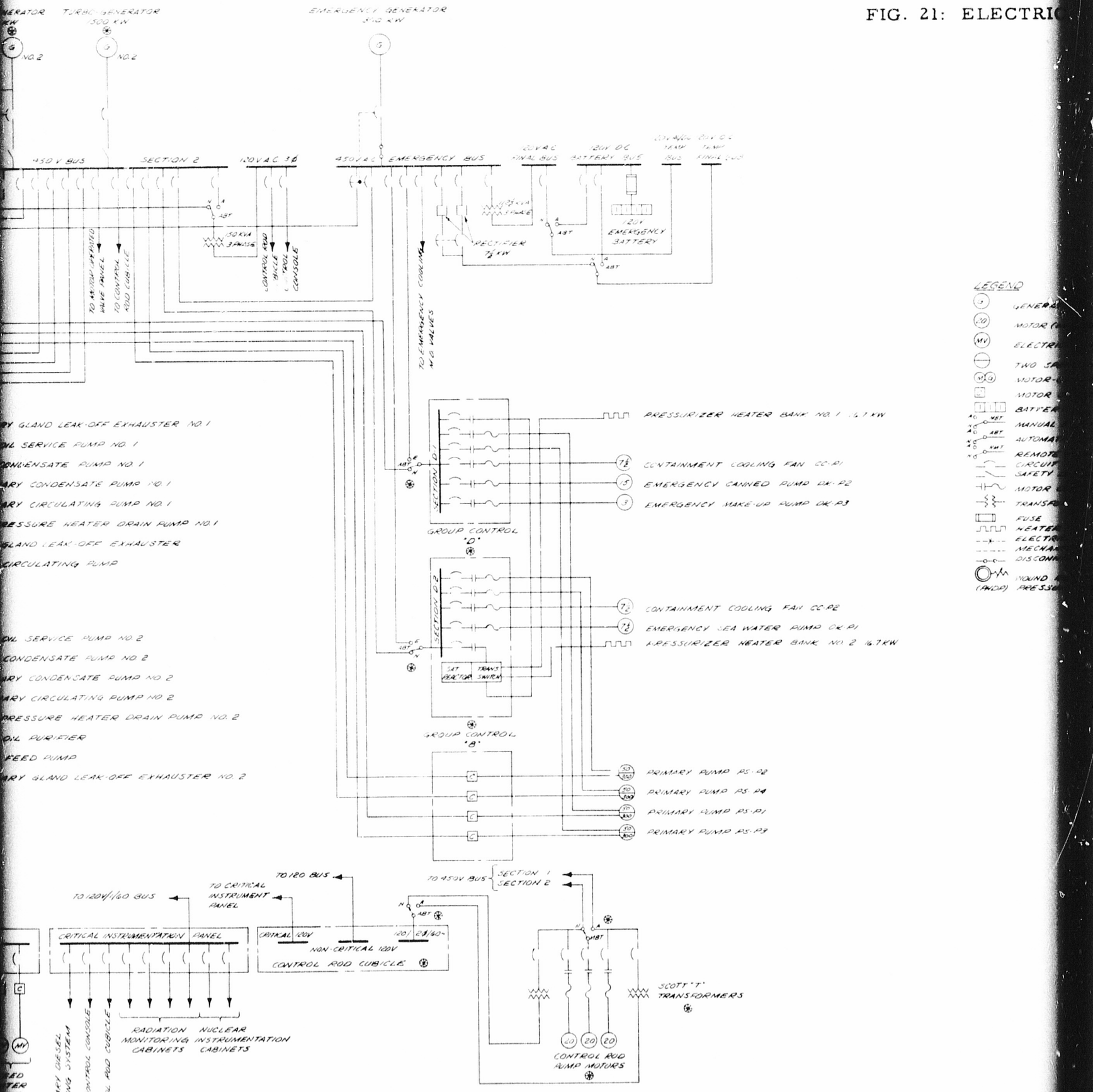


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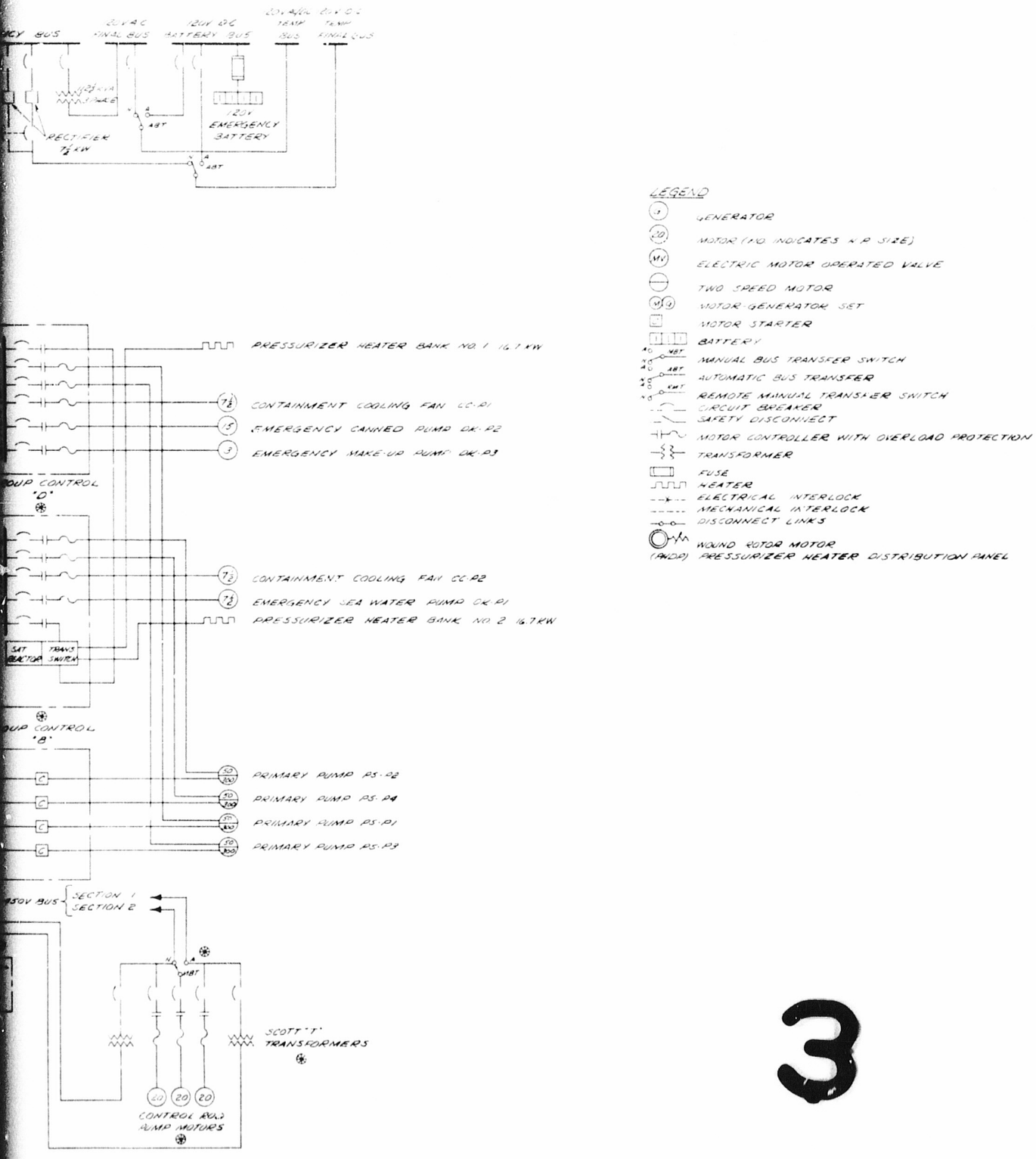
FIG. 21: ELECTRICAL SYSTEM DIAGRAM



LEGEND

- GENERATOR
- MOTOR (NO. INDICATES H.P. SIZE)
- ELECTRIC MOTOR OPERATED VALVE
- TWO SPEED MOTOR
- MOTOR-GENERATOR SET
- MOTOR STARTER
- BATTERY
- MANUAL BUS TRANSFER SWITCH
- AUTOMATIC BUS TRANSFER
- REMOTE MANUAL TRANSFER SWITCH
- CIRCUIT BREAKER
- SAFETY DISCONNECT
- MOTOR CONTROLLER WITH OVERLOAD PROTECTION
- TRANSFORMER
- FUSE
- HEATER
- ELECTRICAL INTERLOCK
- MECHANICAL INTERLOCK
- DISCONNECT LINKS
- WOUND ROTOR MOTOR
- (PWDP) PRESSURIZER HEATER DISTRIBUTION PANEL

3

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FIGURE 21

FIG. 22: PRIMARY SYSTEM BLOWDOWN

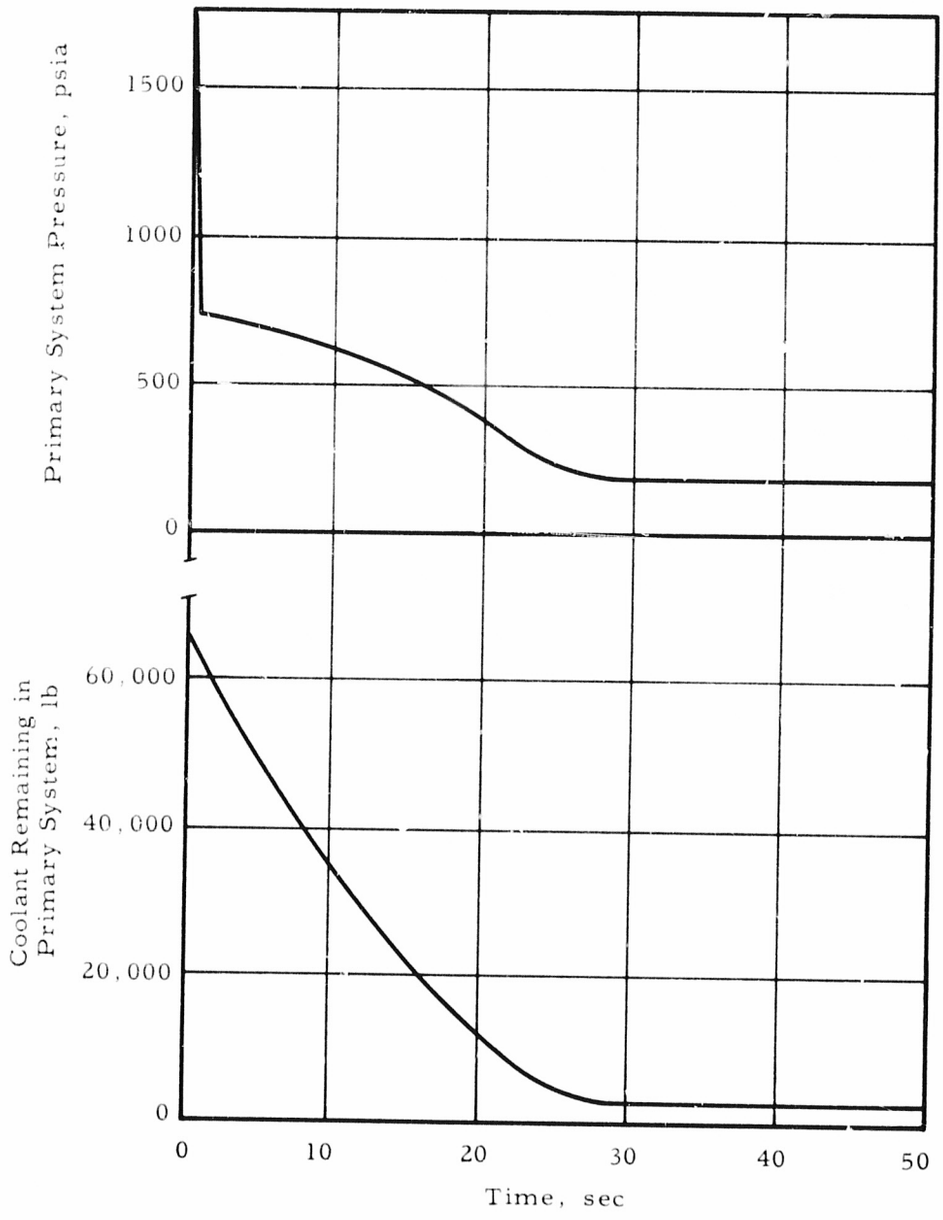

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FIGURE 22

FIG. 23: VAPOR FORMATION IN PRIMARY SYSTEM

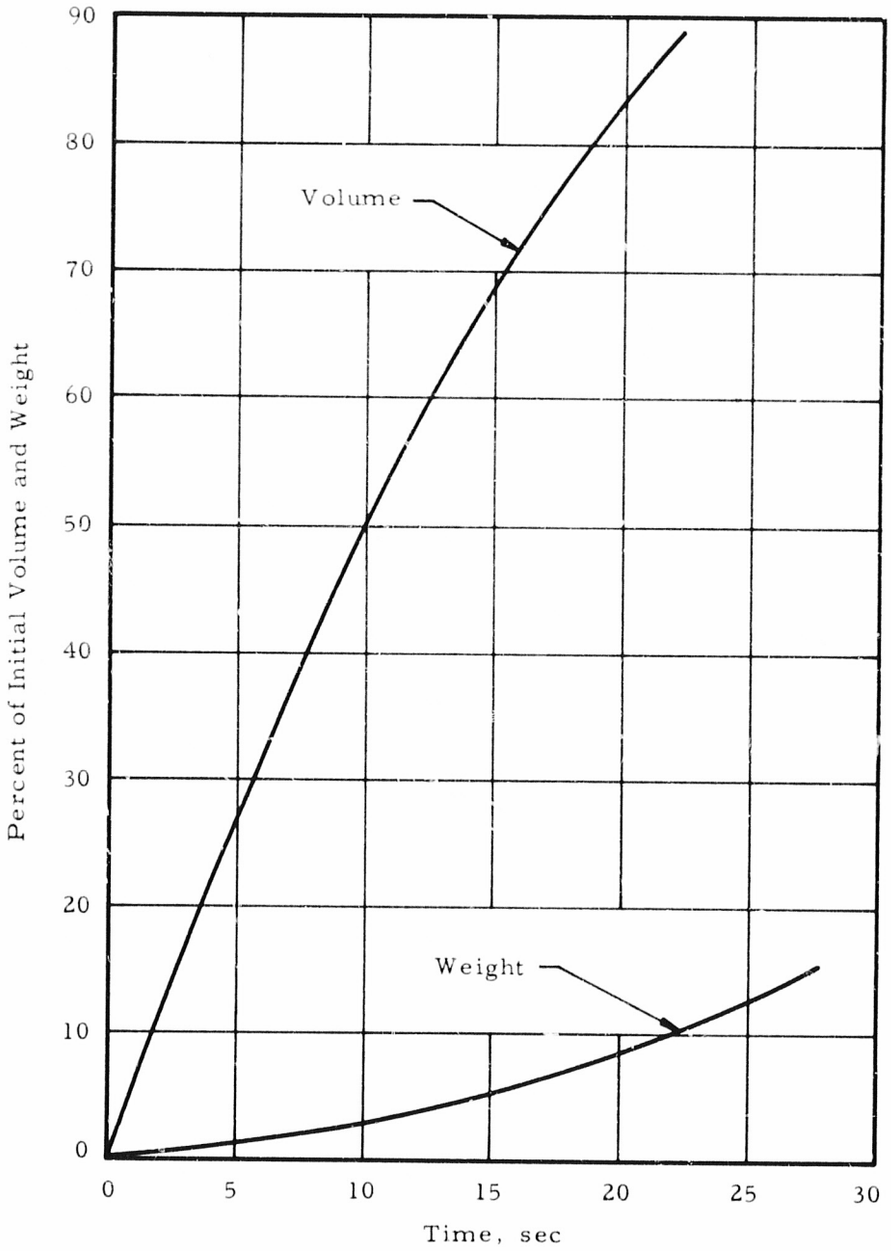

FIGURE 23

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FIG. 24: AVERAGE FUEL ELEMENT TEMPERATURE AFTER LOSS OF COOLANT

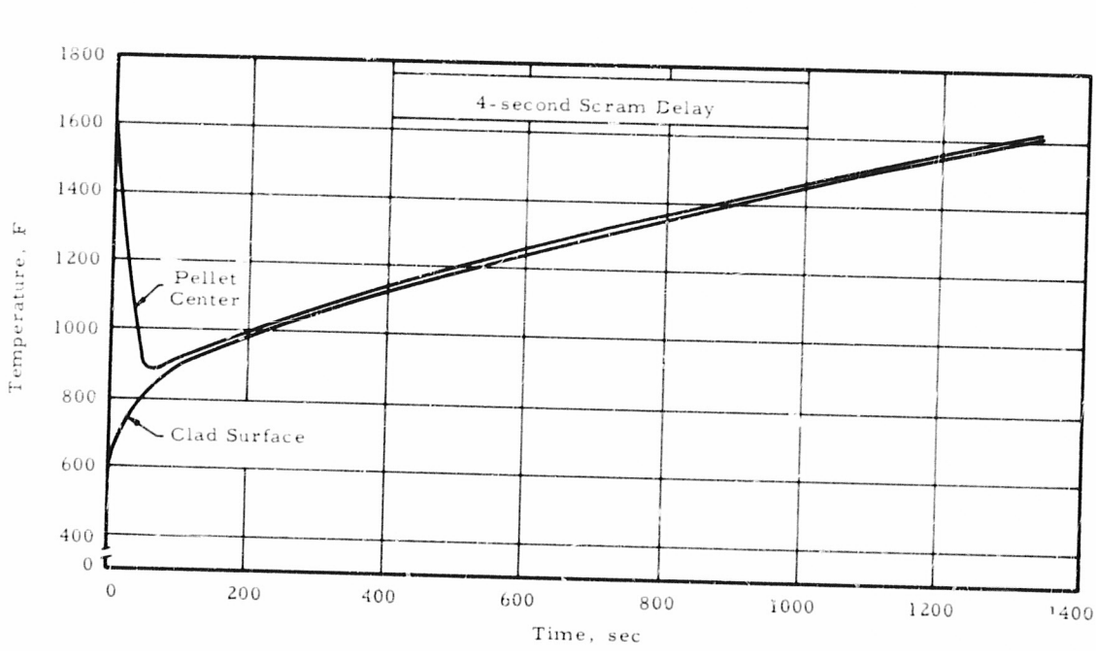

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FIGURE 24

FIG. 25: TIME VARIATION OF RADIAL TEMPERATURE DISTRIBUTION

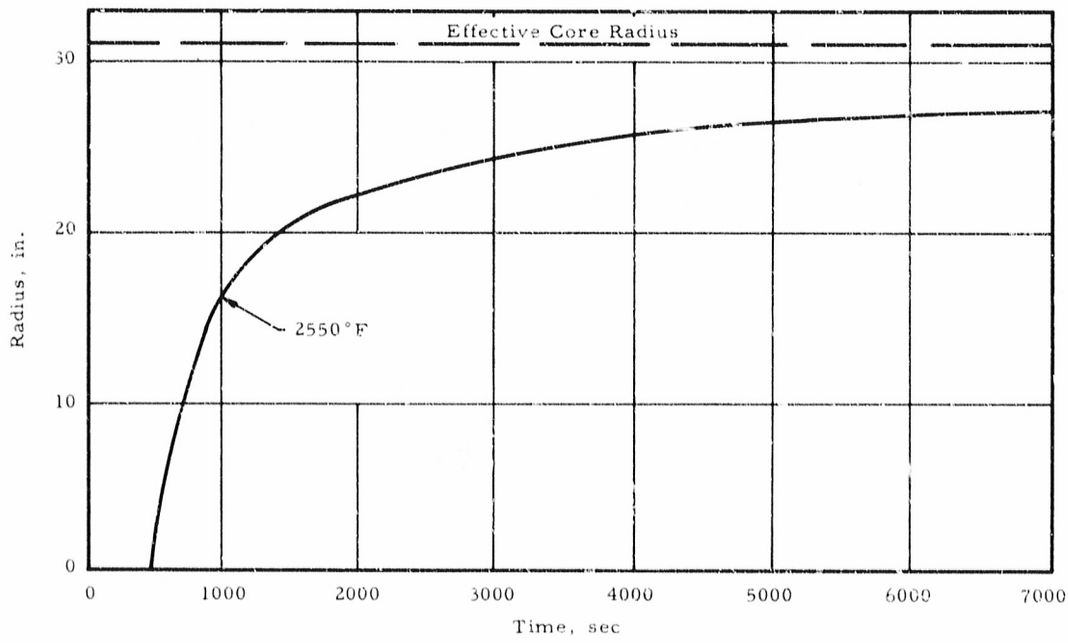

FIGURE 25

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FIG. 26: k_∞ Vs WATER-METAL VOLUME FRACTION FOR MELTED DOWN CORE

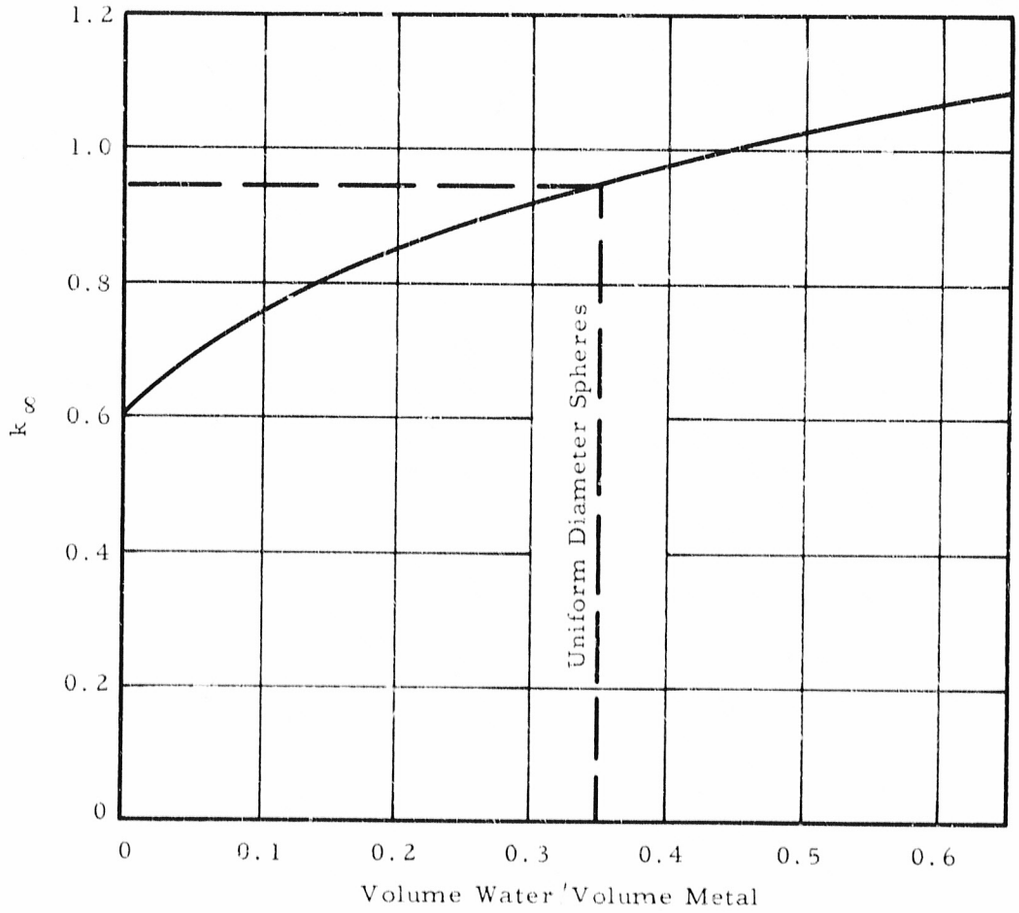

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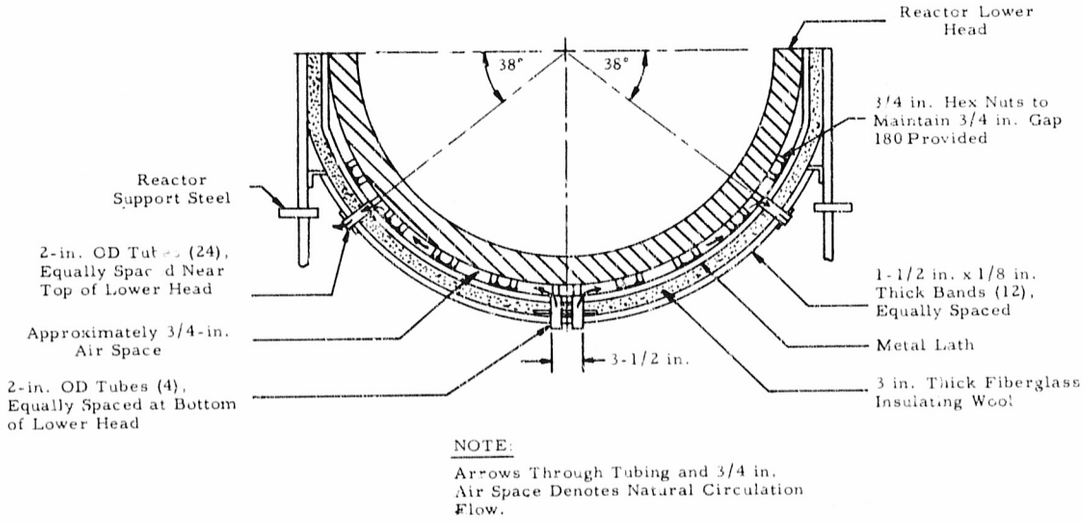
FIGURE 2

FIG. 27: ARRANGEMENT OF INSULATION ON LOWER HEAD OF REACTOR VESSEL



NOTE:

Arrows Through Tubing and 3/4 in. Air Space Denotes Natural Circulation Flow.

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FIGURE 27

FIG. 28: PRESSURE DROP THROUGH LOWER VESSEL HEAD

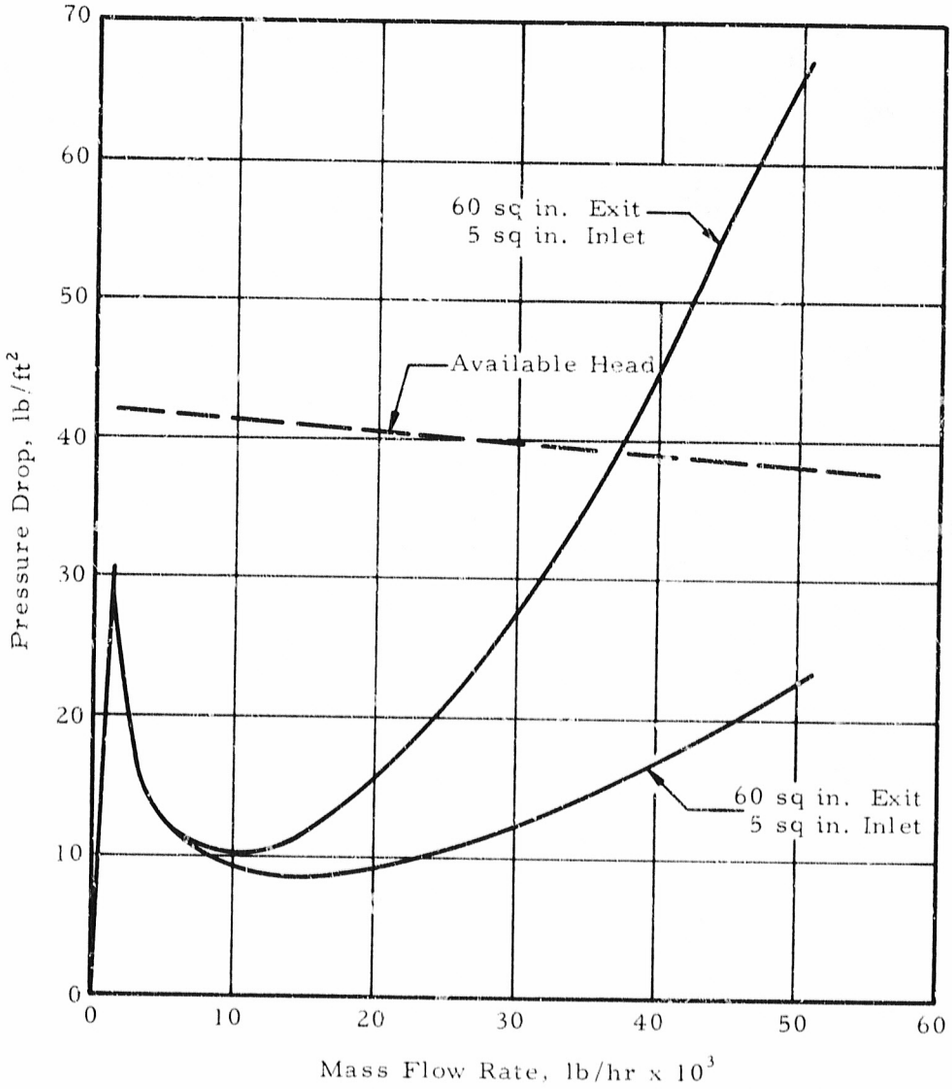

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FIGURE 28

FIG. 29: HEAT GENERATION RATE IN MELTED DOWN FUEL

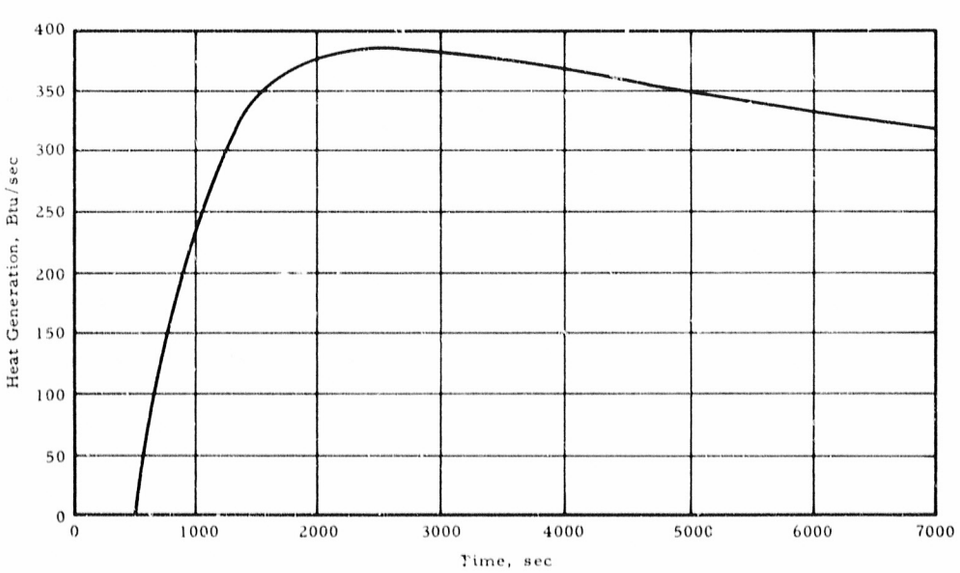

FIGURE 29

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FIG. 30: CONTAINMENT TEMPERATURE AND PRESSURE AFTER PRIMARY SYSTEM RUPTURE

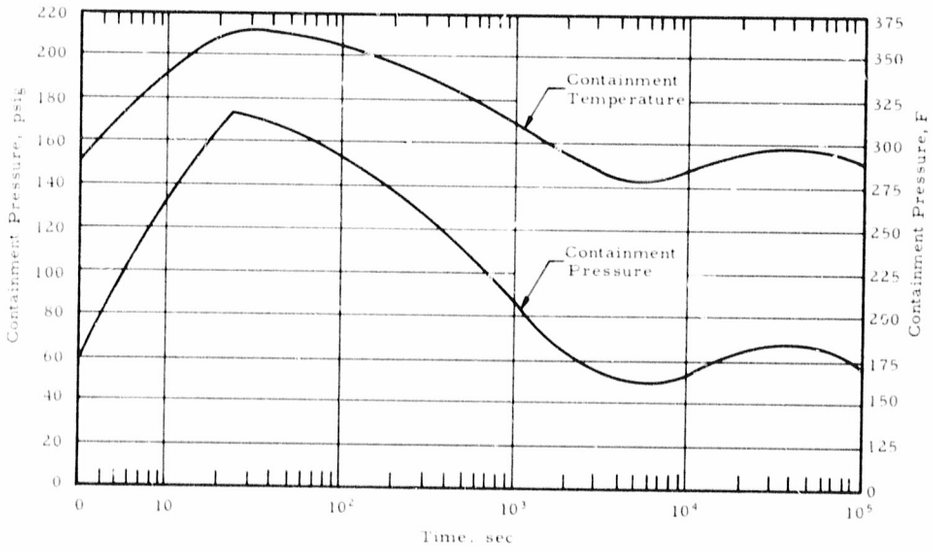

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FIGURE 30

FIG. 31: RATE OF FISSION GAS RELEASE INTO CORE REGION FROM MELTED FUEL ELEMENTS

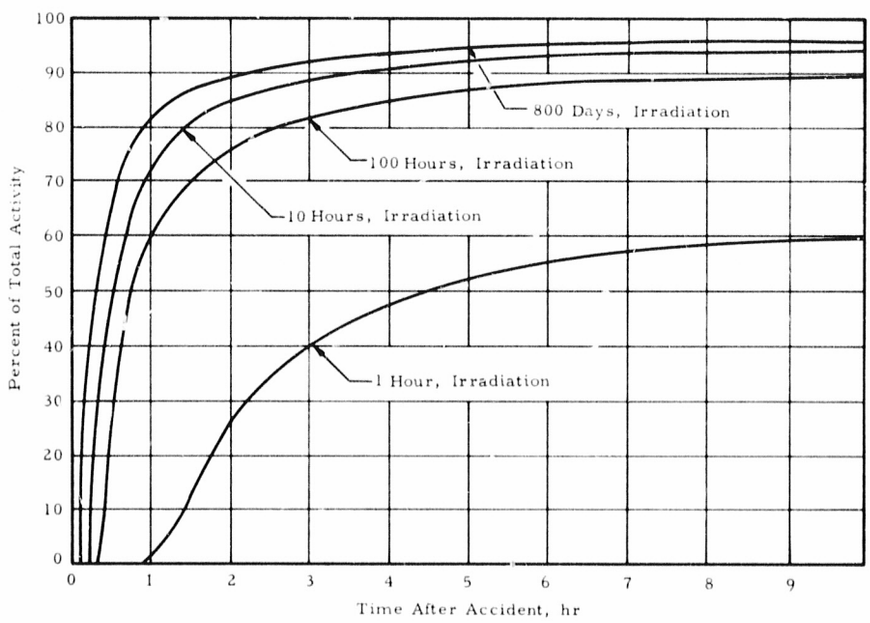

FIGURE 31

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FIG. 32: DOSE RATE ADJACENT TO CONTAINMENT VESSEL FOLLOWING MAXIMUM HYPOTHETICAL ACCIDENT

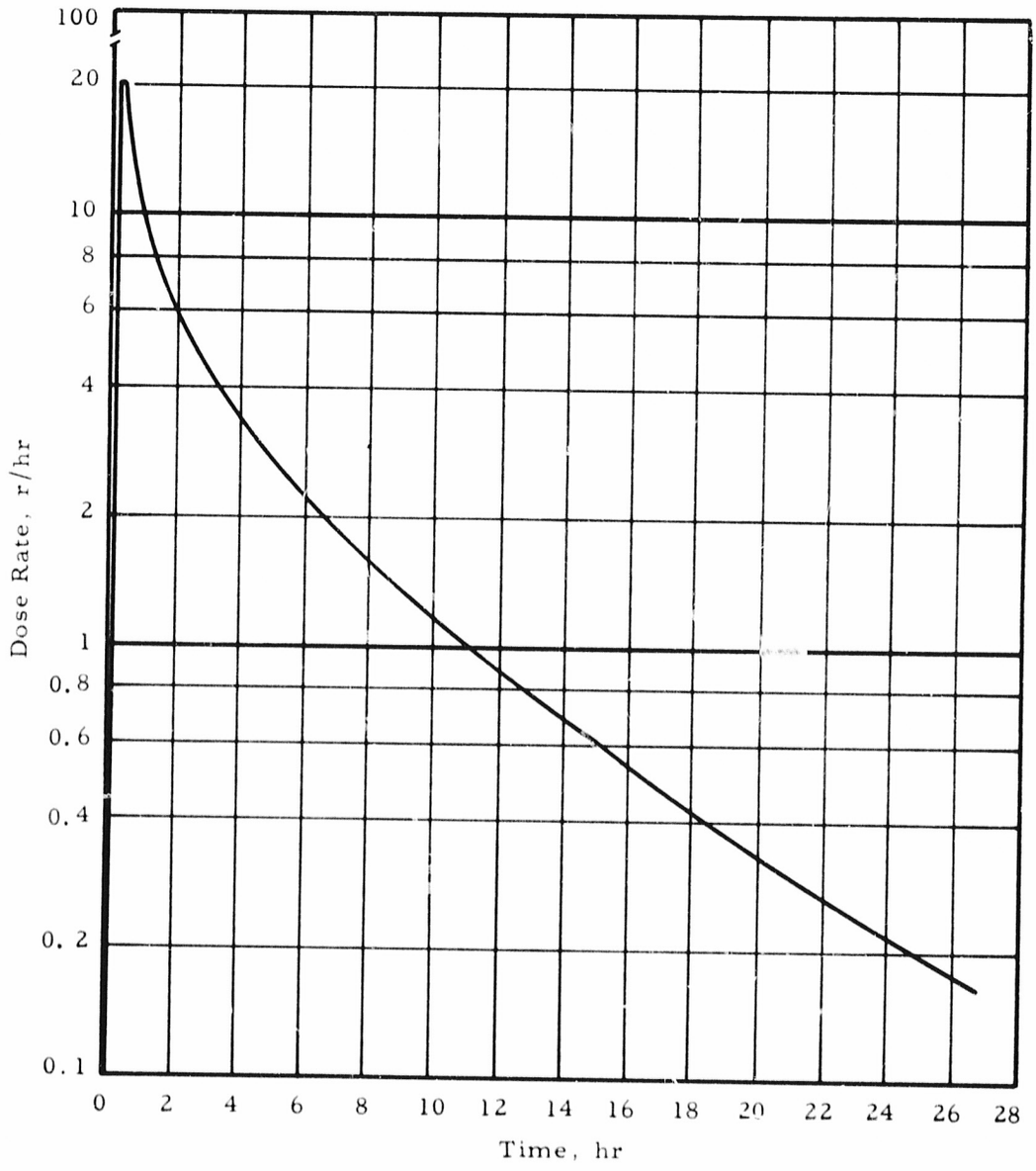

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FIGURE 32

FIG. 33: SPECIFIC ENERGY FOR PENETRATION
(Normal Incidence)

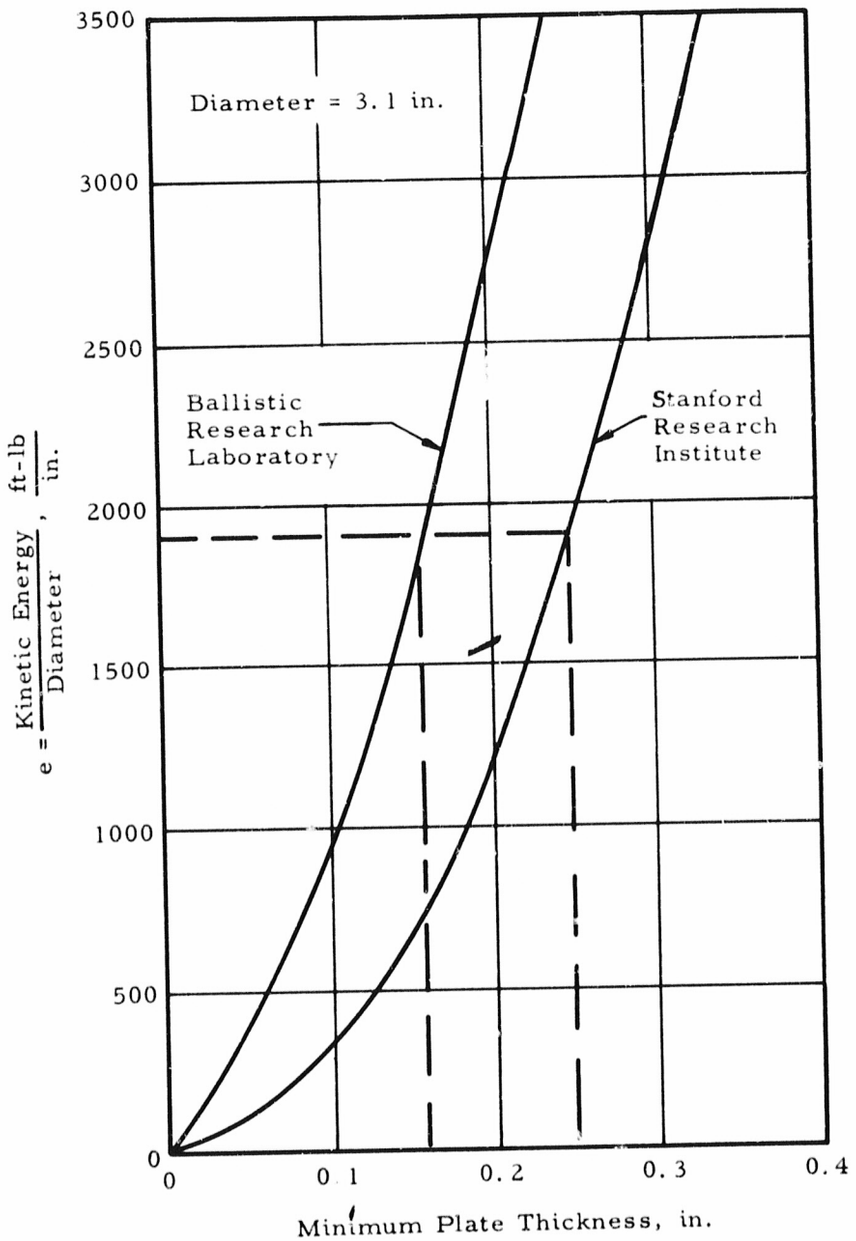

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FIGURE 33

Chapter III

MISCELLANEOUS ACCIDENTS

A. Fire Within the Containment Vessel

B. Failure of the Waste Management System

III. MISCELLANEOUS ACCIDENTS

A. FIRE WITHIN THE CONTAINMENT VESSEL

1. Nature of the Accident

A major fire within the containment vessel could damage equipment and materials in the vicinity, and cause dangerous or costly electrical short circuits. In addition, the increase in pressure and temperature associated with heating of the air confined within the vessel would be of particular concern. Several materials warrant discussion.

2. Materials of Interest

Non-flammable insulation has been used on the primary system and the reactor vessel. The primary system and auxiliary systems are insulated with Johns-Manville Thermobestos Insulation covered with an asbestos cloth. This, in turn, is covered with a fire-proof sealing compound of asbestos fibre and a special fire-proof cement. Insulation, asbestos covering and sealer are all fire proof and cannot be ignited at any temperature to which they might be subjected. The reactor pressure vessel is insulated with fiberglass insulating wool and covered with a stainless steel expanded metal lath. Again, the insulating material is completely fire proof. Although thermal insulating material may be found in large quantity within the containment, it will neither support combustion nor contribute to a fire of any kind.

The hydrogen addition system raises some question about a hydrogen fire or explosion. Although this system is located on the weather deck, it functions as part of the over-all power plant system. The hydrogen is supplied from a single line connected to a supply header which, in turn, is connected to two hydrogen supply cylinders. This single line adds hydrogen to the buffer seal surge tank, located in the lower void space outside the containment vessel. At this point hydrogen

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is dissolved into the primary cooling water to recombine radiolytic oxygen and thus reduce the corrosion rate in the primary system. (For a more complete description of the system, see Vol. I of the Safety Assessment.)

A relief valve in the hydrogen supply line prevents excessive pressure in case the pressure regulator failed. In the event of a fire in the vicinity of this system, flame arrester valves on the discharge side of the hydrogen supply would prevent any backflash from reaching the hydrogen supply connected to the manifold. An excessive flow shut-off valve would isolate the supply header if the hydrogen supply line were accidentally severed. The piping from the hydrogen cylinders to the buffer seal surge tank is completely enclosed within a special duct, and the air in the surrounding annulus is monitored for the presence of excess hydrogen. As an additional precaution, hydrogen detectors are so located that any leakage from the system may be detected. Detectors also monitor the air inside the containment vessel and secondary shielding. In view of the location of the system and the precautions taken, damage to the containment vessel or the systems within it is considered improbable.

The only other material of interest in the containment vessel with a potential fire hazard is the 150-gallons of fluid used in the hydraulic system of the control rod drives. None of the hydraulic fluids available are truly non-flammable, therefore, the characteristics of these materials are examined further in the following discussion.

3. Examination of the Fire Hazard

The hydraulic fluid for this application was chosen after a comparison of three major kinds — petroleum base, water base, and phosphate ester fluids — with reference to fire resistance, radiation damage effects, sealing ability, corrosion effects, maintenance, wear effects, sludge formation, viscosity, and field experience.

The petroleum base fluid selected for the hydraulic system proved superior to the other types in all respects except fire resistance. This is to be expected since some of the other fluids are manufactured specifically to resist fire. Nevertheless, the relative difference in fire hazard is very small for these reasons:

1. The ambient and metal temperatures are relatively low, the maximum metal temperature being of the order of 550° F.

2. A fire of sufficient magnitude and/or intensity must exist to ignite the fluid. Since the fluid is the only flammable material within the containment vessel, there are no other potential sources of fire.

3. A limited oxygen supply exists.

Normally, flash point and fire point are used as guides to describe the degree of flammability of a fluid. However, these tests are designed to determine the relative evolution of flammable vapors. The auto ignition temperature, that is, the temperature at which a fluid will ignite spontaneously and support combustion as long as there is a sufficient supply of oxygen, is a much better guide when sources of heat other than flame are involved.

The auto ignition temperatures of the three available fire resistant hydraulic fluids compare with oil as follows:

Fluid	Flash Point, °F	Fire Point, °F	Auto Ignition Temp, °F
Oil-water emulsion	None	---	755
Glycol water	None	---	725
Phosphate ester	455	655	1150
Petroleum base hydraulic oil	420	470	755

The auto ignition temperature of the petroleum base fluid is some 200° F higher than the hottest metal temperature within the containment. Furthermore, these auto ignition temperatures are determined by production of an intimate mixture of fluid and oxygen to encourage combustion. It is not likely that possible failures in the hydraulic system will produce these mixtures. This type of hydraulic fluid has been used under much higher temperature conditions for many turbine applications and experience has shown that no fire hazard exists.

4. Containment Pressure Rise in Event of Fire

Although no reason has been found for believing a fire can result from a leak in the hydraulic system, a calculation was made to determine the pressure rise within the containment vessel, assuming

that such a reaction had occurred. The following data were used to determine resulting conditions:

Containment volume, ft^3	= 32,300
Containment air mass, lb	= 2,420
Available oxygen, lb	= 484
Weight of available oil, lb	= 1,090

Approximately 3.4 lb of oxygen is needed to burn each pound of oil. Therefore, the amount of oxygen within the containment vessel is not sufficient to burn all the oil completely. Additional air cannot be added from the ship's air supply because these air lines will be isolated upon a pressure rise within the containment. The heating value for this oil is 18,220 Btu/lb. The pressure within the containment cannot exceed 113 psi even if all the available oxygen were used in combustion. Since steady-state design pressure is 173 psi, the pressures reached within the vessel are less than design pressure.

Conduction of all the heat formed in burning the oil to the containment vessel would not raise the vessel's temperature more than 50°F. The design strength is, therefore, unaffected.

5. Conclusion

Since no reason has been found to believe a fire can start within the containment, and since the result of the worst conceivable fire would not endanger personnel or damage the containment vessel, no hazard to the crew or general public would result from a fire within the containment vessel.

B. FAILURE OF THE WASTE MANAGEMENT SYSTEM

1. Nature of the Problem

Failure of one or more fuel pins and the effect on reactor shutdown dose levels within the containment vessel were discussed in Part One, Chapter II, A. The radiation levels existing at the top of the foremast were discussed in Part One, Chapter II, B for the case of a small leak of primary coolant water into the reactor compartment. Therefore, this analysis is intended to examine the relative hazards resulting from a release of radioactivity from the gaseous waste handling system.

2. Waste Management Philosophy

The Waste Management System is designed to contain all radioactive liquid and solid wastes. During normal operations no liquid or solid radioactive matter need be discharged to the ocean or to harbors and estuaries. This system is also designed to contain all gaseous radioactive wastes for short periods of normal operation in confined waters or during unfavorable meteorological conditions. Under favorable weather conditions small quantities of radioactive gases can be released at a controlled rate. The activity level of these gases is monitored after passing through high efficiency particulate and iodine filters, and provisions are made for dilution of gaseous activity before discharge as required. Provision for collection of gaseous radioactivity from the primary system permits continued operation with fission products in the primary coolant, these fission products resulting from defects in the fuel elements.

3. Waste Collection Equipment

The waste collection equipment is comprised of three auxiliary systems, each serving a particular function.

1. The primary purification system removes and collects dissolved and particulate corrosion and fission products from the primary system.

2. The gaseous waste collection system and absorption process equipment collects fission product gases and retains high activity gaseous waste on charcoal beds.

3. The equipment drain and waste collection system stores all liquid drainage from the reactor system and supporting laboratories.

These systems and their functions are described in detail in Volume I of the Safety Assessment. The design basis for the gaseous absorption process equipment is briefly summarized to augment discussion of radioactive wastes in this system.

4. Design Basis

The gaseous absorption process equipment is designed for operation only after fission products have been detected in the primary system. The letdown water is monitored continuously for such activity. The charcoal absorption units can collect the gaseous fission products released from 726 kg of fuel (approximately 10% of the total fuel loading) exposed to the primary coolant for 100 days of operation late in the life of the core.

The distribution of volatile radioactivity within the various reactor system components is presented in Table XIII, based on the assumption that 363 kg of UO_2 (approximately 5% of the total fuel inventory) are completely exposed to the primary coolant. Following detection of fission products in the primary coolant the gas removal system will be operated and the pressurizer void will be vented to this system. Therefore, the columns represent the expected gaseous fission product activity distribution after 100 days of operation and with 5% of the fuel exposed to primary coolant.

5. Gaseous Activity Accumulation

During normal plant operation, the principal sources of volatile activity are operational drainage and minor leaks. The potential activity release outside the containment was discussed in Part One, Chapter II, B. Leaks within the containment vessel present no hazard since the volatile activity is contained and can be purged under favorable weather conditions. Operational drainage from the primary system is collected in the containment drain tank and subsequently transferred to the waste storage tanks. In confined waters or during adverse wind and weather conditions, the vent gases from

these tanks are accumulated in the containment vessel for subsequent release at sea.

The principal volatile activity with fission products in the primary coolant is radioactive xenon and krypton. Any primary system leakage into the containment vessel, containment drain tank or effluent condensing tank will contain these radioactive gases.

6. Normal Venting of Stored Gases at Sea

Following operation in confined waters, the normally vented gaseous activity will have accumulated in the containment vessel. This activity will be comprised of xenon, krypton, and argon isotopes. The activity of each isotope estimated to accumulate each day and its respective half-life are presented in Table XIV. These activities are based on 300 days of operation without defected fuel pins, followed by 100 days of operation with 363 kg of UO_2, 5% of total fuel loading, exposed to the primary coolant. The effect of radioactive decay during holdup in the containment vessel is indicated by the activity 5 days after leaking from the primary system. The effect of the gaseous absorption process equipment is shown in the last column. The leak rate assumed for this example is 30 gallons of water per day.

With 363 kg of UO_2 fuel exposed to the primary coolant, the release of the Xe-133 is about 0.006 curies at a primary system leakage of 30 gallons per day. When diluted by the 4000 cfm fan, this gaseous activity is discharged at the top of the forward mast at an average concentration of 3.8×10^{-8} $\mu c/ml$, one tenth the allowable concentration for unrestricted areas. Further atmospheric dilution outside of the stack by a factor of 100 or more will occur. Consequently, the gaseous activity released at sea during continuous venting of the waste storage tanks or the containment vessel does not present a hazard.

TABLE XIII

DISTRIBUTION OF GASEOUS RADIOACTIVITY

Location	Argon Activity curies (No Fission Products)	Fission Product Activity, * curies: No Gas Removal System	Gas System (No Void Venting)	Gas System (Void Venting)
Pressurizer Void	0.38	237	19	7.6
Buffer Seal Surge Tank Void	1.54	976	79	31
Effluent Condensing Tank Void	0.23	144	11.7	4.5
Effluent Condensing Tank Liquid	0.144	91	7.4	2.9
Primary Water	1.23	773	62	24
Gaseous Absorption Units	--	--	1840	1930

* After 300 days of full power operation before fuel element failure followed by 100 days of operation with failed elements, (363 kg of UO_2 exposed to primary coolant.)

TABLE XIV

GASEOUS RADIOACTIVITY ACCUMULATION

(Operational Drainage 30 gallons for 1 day)
363 kg UO_2 exposed

		Without Gas Absorption System		With Gas Absorption System
Isotope*	Half-Life	Initial Activity, curies	Activity After 5 days, curies	Initial Activity, curies
A-41	1.82h	0.0036	-	0.00011
$Kr-83^m$	1.9 h	0.0067	-	0.00021
$Kr-85^m$	4.4 h	0.019	-	0.0006
Kr-85	9.4 y	0.00061	0.0006	0.000019
Kr-87	1.3 h	0.015	-	0.00047
Kr-88	2.8 h	0.036	-	0.00047
$Xe-131^m$	12 d	0.00096	0.00096	0.000096
Xe-133	5.3 d	0.20	0.11	0.0062
Xe-135	9.1 h	0.13	-	0.0041

* Isotopes with half-lives of less than 1 hour have been omitted.

7. Failure Within the Gas System

The foregoing analyses dealt with normal operation and venting after fission products were found in the primary stream. It now becomes pertinent to consider the rupture of a charcoal bed, which results in complete release of the activity held up in the bed.

It is assumed that 363 kg of fuel are exposed to the primary coolant over a period of 100 days, and that the leak occurs after 300 days of full power operation. Normally only one charcoal bed would be operating at any one time. If the bed had been operating for 50 days with leakage from the primary system, it would contain most of the gaseous activity collected from the system. It is assumed in this example that this is the ruptured bed. Table XV lists the activity calculated to exist in this bed at the time of the rupture.

8. Environmental Hazards

To estimate the hazard from this rupture, it was assumed that the gas was discharged in an instantaneous puff and that there was no attenuation by condensation on the ducting, surroundings, or

filter. It was further assumed that a nighttime inversion was prevailing, under the following conditions as applied in the Sutton diffusion calculation:

N = 0.5 (stability parameter)

C = 0.13 (diffusion coefficient)

U = 2.2 m/second (wind velocity)

To obtain a conservative value for the dose level, it was assumed that the ship was not in motion. The wind speed was therefore taken as the true wind speed. The release was assumed to be at ground level. The maximum external radiation exposure under these conditions was approximately 800 mrem gamma and 680 mrem beta at 100 meters from the ship.

TABLE XV

GASEOUS RADIOACTIVITY IN OPERATING CHARCOAL BED *

Isotope	Activity, curies
Kr 85 (m)	4.93
Kr 85	26.0
Kr 87	0.93
Kr 88	5.7
Xe 131 (m)	28.1
Xe 133	1660
Xe 133 (m)	19.2
Xe 135 (m)	15.9
Xe 135	83.0
Xe 137	0.025
Xe 138	0.23

* Assuming 363 kg fuel exposure to primary coolant.

PART TWO

SHIP ACCIDENTS

Chapter I

SAFETY ASSESSMENT

A. Conventional Ship Safety
B. Collisions
C. Grounding
D. Heavy Weather
E. Fire and Explosion
F. Flooding and Sinking
G. Salvage

I. SAFETY ASSESSMENT

The NS SAVANNAH was designed to comply with the highest standards of safety in respect to conventional hazards as well as the potential hazards associated with the operation of a nuclear propulsion plant. The following basic safety criteria were established early in the program.

1. The SAVANNAH shall be as safe as any other vessel in her class with respect to the usual hazards at sea.
2. In no credible accident shall there be any hazardous release of radioactivity to the environment.

It is believed that these criteria have been satisfied.

The accidents to which a ship is exposed in normal operation include collision, grounding, flooding, sinking, heavy weather, fire, and explosion -- all of which have been analyzed in detail. It should be noted that these hazards are considered principally from the standpoint of release of radioactive material rather than from the conventional aspects of safety of life at sea. The safety of passengers and crew from the normal hazards of ocean travel is considered to be adequately handled by compliance with existing maritime regulations. Normal ship safety requirements are treated only insofar as they affect the SAVANNAH's ability to avoid or withstand severe damage. The special hazards associated with war service, such as operations with naval vessels, ballistic and mine damage, etc., are not considered. Detailed analyses of various ship accidents are given in succeeding chapters.

A. CONVENTIONAL SHIP SAFETY

The safety of any ship depends primarily on three factors:

1. care and judgment of responsible personnel,

2. complete and immediate knowledge of potential hazards by responsible personnel, and
3. complete control of vessel and equipment at all times.

Only factor 3, and factor 2 to some extent, can be ensured through component and system design.

The principal requirements for complete control of a ship are: (1) the maintenance of power to the shaft, and (2) adequate stopping power. On the SAVANNAH, means for fulfilling these requirements have been provided to an unusual degree.

Electrical power is assured by the installation of three independent sources in addition to the two normally used 1500-kw ship service turbogenerator sets. Two automatic-starting (< 15 seconds), 750-kw auxiliary diesel generators were installed. In case of reactor scram or shutdown, the stored heat in the system can maintain power through the ship's service turbogenerators until the auxiliary units can be brought into service. If for any reason the two auxiliary diesel generators are not available, or fail to start, the 300-kw emergency diesel generator can supply sufficient power to operate vital systems such as the decay heat removal units. The emergency generator, located on the navigation bridge deck, is provided with independent switchgear and is started automatically from a continually charged storage battery.

A 750-hp electric "take-home" motor was installed for propulsion in case of main propulsion unit failure. With this motor, the SAVANNAH can develop almost 7 knots in a mild sea. Power for this motor is taken from the auxiliary diesel generators.

Head reach, the distance required to stop from design speed, and backing power on the SAVANNAH are comparable to those of other ships in this class. For example, the SAVANNAH will stop in 3900 ft from 20 knots using the design backing torque, compared with 3780 ft for the MARINER. Thus, the maneuvering capability of the SAVANNAH compares favorably with conventional steam vessels.

There are two conning stations on the SAVANNAH, one on the bridge, the other at the after steering station on the docking bridge. There is also a handwheel on the steering gear.

The steering gear has two independent steering plants, each capable

of handling the rudder under design conditions. Transfer from one plant to another is effected by a crossover in the steering engine room. There are two electric, bridge activated, steering control systems to operate either steering plant. In addition, a hand pump is supplied to allow positioning of the rudder in case of main hydraulic plant failure. It is concluded that the steering gear has more than adequate backup.

The only vital units on the SAVANNAH without backup are the rudder, propeller, and shafting, a characteristic of all single-screw ships. Loss of propeller and shafting failure are infrequent occurrences; loss of rudder occurs even less frequently. While such a situation is dangerous in that the ship is more or less out of control, countermeasures are well defined. Anchoring in restricted waters and the use of tug services in open sea constitute the normal procedures. As a rule, these failures are not dangerous in themselves; at worst, they lead only to those hazards described hereinafter.

Avoiding potential ship hazards is primarily a function of navigation and communication. The wheel house and the navigation bridge on the SAVANNAH are designed for maximum visibility, and the internal communication system is of the highest order.

With the navigational equipment provided, the responsible personnel aboard the SAVANNAH will be assured of accurate information of potential hazards.

The most important factors in safety at sea – care and judgment – naturally do not result from the design of a ship. They are the results of crew training, experience, and vigilance. The SAVANNAH's crew was selected and trained with great care, as detailed in Volume II of the Safety Assessment.

B. COLLISIONS

Almost all collisions between ships at sea can be traced to errors in the judgment of responsible personnel. Collisions attributable directly to equipment failure occur only rarely. Since collisions cannot be eliminated through design, provisions must be made in a nuclear-powered ship to protect the reactor and containment against the effects of collisions.

A thorough study of collision records was undertaken to determine the mechanisms of collision and to define the magnitute of damage to be expected. Details of this study are given in Part Two, Chapters III, IV and V. In the course of the investigation, the available data on some 60 major collisions were reviewed, including reports from the United States Coast Guard on collision circumstances, American Bureau of Shipping and United States Salvage Association damage surveys, testimony of passengers and crew members who had experienced collisions, and U. S. Navy damage reports. In addition, statistical data on ship collisions were obtained from U. S. Salvage, Lloyds, and other underwriting organizations. [2]

The data were first reviewed for evidence and magnitude of shock damage. The acceleration or deceleration of large masses could conceivably damage vital reactor components, rupture pipelines, etc. Specially strengthened and armored vessels as well as conventional merchant ships were included in the study.

No evidence whatever of shock damage was found in any collision. Hence, it can be concluded that shock values in collision are less than those to which merchant ship foundations and mountings are normally designed (≈ 2 g's). In addition, testimony of personnel involved in collisions of the most rigid ships (battleships) revealed that no shock could be felt as the vessels collided. Since personnel are often thrown off their feet in a seaway, this testimony indicates that the upsetting forces are even less than those encountered in severe roll, pitch, and heave (≈ 0.6 g). Further substantiation obtained by analytical methods indicated even lower accelerations. Since all nuclear components on the SAVANNAH are estimated to be capable of withstanding shock loadings of approximately 1 g without primary system rupture, it

is concluded that in no conceivable collision will the SAVANNAH's power plant sustain shock damage. This collision further eliminates any possibility of damage when the SAVANNAH is the striking vessel since penetration is not a problem in this case.

Having eliminated shock as an important effect, an attempt was made to define the mechanisms of penetration in collision. It was found that a good correlation exists between the kinetic energies of the colliding vessels and the structural damage incurred. The principal components contributing to collision resistance were identified as decks and transverse bulkheads on the struck ship, and the decks, longitudinal bulkheads, and some angular component of the shell on the striking ship. Therefore, on a given ship, the decks in way of the area to be protected are the main collision barrier. Since the collisions were assumed to be inelastic, the correlation is further evidence of negligible shock in collision.

From the curve of kinetic energy and collision damage (Fig. 43) it is possible to predict, with a considerable degree of accuracy, the depth of penetration resulting from a collision of any two vessels. It follows that the depth of penetration that the SAVANNAH would experience in a hypothetical collision with another vessel can be predicted and, within limits, the ship's structure can be designed to withstand effects of a collision of any given magnitude.

The limit of penetration without reactor system damage was assumed to be the reactor compartment longitudinal bulkhead. The deck scantlings (Table XIX) of the SAVANNAH were increased above those of conventional ships to the extent that only about 0.7% of the world's merchant fleet (approximately 300 ships) could penetrate to this bulkhead.

The probability of a collision in which penetration beyond the reactor compartment longitudinal bulkhead would occur was then evaluated for the SAVANNAH over a 20 year ship life. The calculated value of this probability is about 7×10^{-5}, or, within the limits of accuracy of the data, between 10^{-4} and 10^{-5}. This figure, as detailed in Part Two, Chapter V, is believed to be conservative and is actually closer to 10^{-5}. A probability of one in 100,000 over a 20 year ship life compares favorably with the high range of values given in a study by the Atomic Energy Commission for the release of substantial amounts of fission

products outside containment.[3] The highest probability for such an accident quoted in this report was 10^{-5} /reactor/year, which over a 20 year period would be 2×10^{-4}, or essentially a factor of 10 higher than that which is considered the best value for a similar accident involving the SAVANNAH.

In estimating the probability of a dangerous collision, it was found that accidents involving passenger ships take place most frequently within "approach" areas. An approach area, defined as extending from the principal sea mark at the harbor entrance to the point of departure 100 miles from the nearest land, is characterized by the merging of sea lanes at harbor approaches, heavy traffic, and relatively high ship speeds. Therefore, since the area of highest collision probability extends out to 100 miles at sea, the exposure risk of the general population is, on the average, probably less than that for most land-based reactors.

Very few collisions involving passenger ships in harbors are recorded (none in the statistical sample used to develop the probability data); in addition, the very low speeds preclude penetration of the reactor compartment.

The findings of the collision studies are summarized as follows:

1. Collision offers no danger of shock damage to reactor components.
2. Containment rupture in a harbor is practically impossible.
3. Collisions involving containment rupture are limited to areas outside harbors. The probability of such an event over the life of the ship is very low, of the order of 10^{-5}. This figure compares favorably with the "classical" values given by the AEC for an uncontained release from shore-based plants. In addition, since the area of greatest danger extends to 100 miles at sea, the degree of probable isolation is probably greater than that for any shore-based power reactor.

It is therefore concluded that the SAVANNAH can be operated anywhere in the world with little fear of a dangerous release of radioactive material as a result of collision.

C. GROUNDING

The effects of grounding are similar to those of collision — crushing of structure and shock damage may be expected. However, an investigation of grounding accidents reveals shock damage to be negligible. In all severe cases of grounding, extensive crushing of structure held accelerations to very low levels, indicating that shock in grounding is of little concern from the standpoint of reactor safety.

Grounding, particularly by passenger ships, occurs much less frequently than collision. Since passenger ships usually call only in well-charted ports, their officers and pilots are generally well aware of local hazards. Being a passenger ship, the SAVANNAH is certainly less likely to ground than vessels of other types.

The damage sustained in grounding is strongly dependent upon the state of the sea. In harbors or sheltered waters, structural damage is generally limited to that incurred in initial contact and the ship has little tendency to work. The majority of these groundings are bow-on or stern-on, and would not normally involve the reactor compartment. Damage to the inner bottom in way of the reactor compartment could result from certain harbor groundings, but these are rather infrequent. In such cases, the low-level liquid wastes stored in the inner bottom tanks would be jeopardized, however, purification of these wastes will limit the amount of radioactivity involved. The probability of even a very limited release of low-level waste is considered negligible.

On very rare occasions ships have been known to lie over on one side upon grounding or sinking in shallow water. Should this occur the SAVANNAH's reactor, primary system, and other vital auxiliaries are sufficiently braced to prevent dislocation of components. No failure of primary system or containment can result from the ship resting on its side. (See Part One, Chapter II, C, 5)

Grounding in unprotected waters results in more extensive damage because of the working of the ship under the action of the sea. In the worst case, the ship may pivot about the point of contact and eventually break up. Because of the discontinuities in the hull girder forward and aft of the reactor compartment, such breaks will not occur in way of the containment. Once the reactor compartment section has broken

free of the water-borne sections, or after the water-borne sections have sunk the relatively short containment vessel, with its great strength and weight, will have little tendency to move under wave action, and structural damage will be greatly dimished. Under these conditions, release of low-level wastes and loss of demineralizers and other equipment outside containment must be assumed. The demineralizers are small, heavy, and well protected, and therefore are unlikely to release their resins to the sea. The small radioactive inventory in the waste tanks is not considered a source of danger to the environment because areas in which such groundings can occur are fairly remote.

Breaking up is a very slow process and ample time ordinarily will be available for removal or immobilization of the containment shell and reactor systems. Appropriate procedures are given in Part Two, Chapter X.

In conclusion, it is believed that any grounding of the SAVANNAH that would cause extensive structural damage is unlikely. In addition the low activity of any consequent release and the existence of effective countermeasures minimize the environmental hazard that such an accident might create.

D. HEAVY WEATHER

Under heavy weather conditions, the SAVANNAH is exposed to the most severe loadings on equipment foundations and structure. Since accelerations in collision and grounding are much less than those imposed by roll, pitch, and heave, the design of the SAVANNAH was dictated by the latter.

Although considerable data have been compiled on roll, pitch, and heave in moderate-to-very rough seas, only spot reports are available for extreme conditions. Since extreme conditions occur at rather rare intervals, long periods of careful measurement at sea are required to obtain an adequate statistical sample. Hence, maximum accelerations under extreme conditions cannot be defined at present.

For many years, it has been the practice to design merchant ships to severe conditions (less than extreme), with relatively high safety factors. The soundness of this approach is evidenced by years of experience in which ordinary merchant ships have survived hurricane seas without damage.

The SAVANNAH was designed with the following specifications for roll, pitch, and heave.[4]

1. Maximum roll 30 degrees from the vertical, with periods of 13 and 23 seconds (center of roll 20 to 30 ft above the base line) in light and loaded conditions.
2. Pitch and heave — maximum pitch amplitude of 7 degrees, combined pitch and heave accelerations, 0.25-0.50 g at the reactor.

These criteria represent very severe conditions, and in no case will they be exceeded by a factor as great as two, even under the worst conditions. Ship motion, an important factor in passenger comfort, can be sharply reduced by changing course and speed in heavy seas. It is unlikely, therefore, that the SAVANNAH will be exposed more than momentarily to motions as severe as the design criteria. The safety factors employed in the design of the SAVANNAH are considered adequate, and it is estimated that all her components can withstand accelerations of at least 1 g, and in most cases 2 g. Since the maximum loading (static plus dynamic) defined by the design criteria

is about 0.6 g (roll), the margin is fully adequate for maximum sea conditions. No damage to machinery components from ship motion will result under any sea condition.

E. FIRE AND EXPLOSION

The SAVANNAH has been designed to the highest standards of fire protection. The fire protection systems (described in Part Two, Chapter VIII) are believed capable of protecting the ship from the most severe fire hazards.

Three fire pumps are installed, two in the machinery space and one in the shaft alley. The shaft alley fire pump is connected to the emergency switchboard and hence power is assured, even with loss of the machinery area. The carbon dioxide smothering system has sufficient capacity to protect the largest compartment on the ship. In addition to the manual detection and alarm systems, there are three independent fire detection systems.

Fire from an external source is not considered a threat — neither are internal explosions, since the SAVANNAH, as a passenger ship, is forbidden by Coast Guard regulations to carry "dangerous cargo" such as explosives in bulk. External explosions present only a missile hazard, and the reactor and containment are well protected by the ship's structure and shielding.

The chief threats posed by internal fires are loss of power and damage to shielding, containment, and vital reactor equipment. The possibility of fire within the containment vessel has been eliminated because of the scarcity of combustible material and the limited oxygen available in the closed vessel. However, a carbon dioxide extinguishing system is installed in both the containment and the reactor compartment to limit damage should a fire develop. The effect of fire within the containment vessel is discussed in greater detail in Part I, Chapter III of this report. Although a fire in the reactor compartment could cause shielding damage, partial loss of shielding is not crippling since reduced power and/or establishment of exclusion areas would permit continued vessel operation. The major source of ship's machinery space fires has been eliminated aboard the SAVANNAH with the elimination of the main propulsion boiler and associated fuel oil systems.

The SAVANNAH's fire-fighting systems are believed capable of extinguishing any fire that might break out. Damage to the reactor system should be negligible and, in general, limited to certain shielding. Hence fire is not considered a threat to nuclear safety.

F. FLOODING AND SINKING

Complete loss of power, and the effect on decay heat removal, is the major problem in flooding and sinking. Flooding without sinking is of little concern since power for decay heat removal is available from the emergency generator on the navigation bridge deck. In sinking, it is assumed that decay heat removal will be interrupted, resulting in at least partial core meltdown and fission product contamination of the containment vessel.

Preservation of containment in sinking is assured by flooding valves (located in the shell) which open automatically when the ship submerges in deep water. When the pressures, inside and outside the containment vessel, are equalized, the valves close, resealing the vessel. This protects the containment shell against external pressures in deep sinking. Post-sinking problems are covered in Part Two, Chapter X.

G. SALVAGE

Salvage methods depend on the depth of the water since operational difficulties increase with depth. In less than 300 ft of water, almost complete control of the containment is possible. Salvage connections, sized to take standard Navy diver hose, are located on the containment shell and permit sampling of the containment contents and purging as necessary. If permanent immobilization is required, the same connections can be used to fill the vessel with concrete.

If sinking occurs in water less than 100 ft deep, the ship and the containment can be raised and salvaged. At these depths the containment would not be flooded with salt water. Under extreme diving conditions, which limit underwater labor, the ship's structure could be blown away from the containment vessel and the containment raised by tidal lift.

Below 300 ft, the ship will be inaccessible but the flooded containment vessel is a very effective waste disposal package. With the low corrosion rates in deep water, it would take years for the s . water to penetrate the containment shell. Most of the fission products would be contained within the massive primary system, and the probability of environmental contamination would be negligible.

The methods of immobilization and recovery used to salvage the sunken vessel could also be used to salvage the grounded vessel. The accessibility of the grounded vessel, however, would generally simplify the operation.

In conclusion, no hazard to the environment is anticipated from either the sunken or the grounded ship.

Chapter II

CONVENTIONAL SHIP SAFETY

A. Design Standards

B. Machinery

C. Hull

II. CONVENTIONAL SHIP SAFETY

From a conventional standpoint, the NS SAVANNAH was designed to be as safe as any vessel in her class. The most rigid interpretations of the rules of the various regulatory bodies were applied in all cases.

A. DESIGN STANDARDS

SAVANNAH is designed to a two-compartment standard of subdivision (the ship will remain afloat with two compartments totally flooded) at a draft of 29 ft - 6 in., with the margin line 3 in. below "B" deck. This exceeds the requirements of the U. S. Coast Guard and Senate Report 184. The ship complies with all applicable laws of the United States and the requirements at the time of delivery of the following regulatory bodies and rules:[4]

1. American Bureau of Shipping
2. United States Coast Guard
3. United States Public Health Service
4. International Convention for Safety of Life at Sea, 1948
5. International Load Line Convention, 1930
6. Report of the Technical Committee on Safety at Sea of the Subcommittee on Department of Commerce and Merchant Marine of the Senate Committee on Commerce, contained in Senate Report 184 of the 75th Congress, First Session, except as modified by Coast Guard Rules.
7. U. S. Customs Admeasurement
8. Panama Canal Regulations
9. Suez Canal Regulations
10. A. I. E. E. Standards No. 45, current Revision
11. Federal Communications Commission
12. Standards for Protection Against Radiation, Atomic Energy Act, as amended, Title 10, Part 20 of the Federal Register.

SAVANNAH's design was also influenced by the recommendations of the Hazards Evaluation Branch and the Advisory Committee for Reactor Safeguards of the U. S. Atomic Energy Commission.

Of these rules, several do not have the force of law, and are in fact more restrictive than the legal requirements. In particular, Senate Report 184, which followed the investigation of the MORRO CASTLE disaster, has never been adopted as law. Its recommendations, considerably more restrictive than the requirements of the Coast Guard, American Bureau of Shipping, and the London Conferences, have been generally applied to the SAVANNAH. Hence the SAVANNAH has been designed to the most restrictive of existing laws and recommendations.

B. MACHINERY

From the standpoint of ship safety, the maintenance of sufficient power to provide steerage and maneuverability is the principal requirement of the power plant. To meet this objective, all principal components aboard the SAVANNAH have been provided with backup except the rudder, propeller, and shafting.

A 750-hp (nominal) take-home motor, operating from the auxiliary diesel generators, supplies emergency propulsion power. A quick-connect coupling allows the motor to be coupled to one of the high speed pinions of the reduction gear in less than 2 minutes. With this motor in use and the main propulsion plant secured, SAVANNAH can develop almost 7 knots in a mild sea. When heading into a 50-knot wind, a speed of about 1 knot can be maintained with no sea current. This speed allows the ship to turn and run with the wind and sea if desired.

Loss of propeller is generally due to the striking of submerged objects which bend or break the blades. Failure of shafting can follow through propagation of the propeller damage, usually after extended operation at high speeds. The tailshaft failures of war-built ships, which were caused by a "vibration critical" close to operating speeds, have been essentially eliminated on recently built vessels. Although loss of propeller or tailshaft is dangerous in that the ship would be without propulsion, it is a fairly common hazard and countermeasures are well known. Ordinarily the vessel is controlled by sea anchors, or allowed to drift until a tug arrives to tow her in. Loss of propeller on a lee shore can lead to grounding, but in most cases the anchors can hold the ship off shore. Each of the SAVANNAH's anchors has 990 ft of chain, and unless the bottom shoals very steeply, the anchors can be dropped far enough off shore to hold the ship in fairly deep water. In a restricted waterway, a ship having lost a propeller could drift out of the channel and possibly run aground. In general, loss of propeller or shafting leads only to the hazards detailed in succeeding sections on grounding and collisions.

Although it can lead to other accidents, loss of steerage alone is not dangerous.

There are two conning stations on the SAVANNAH, one on the bridge, the other at the after steering station on the docking bridge. There is

also a handwheel on the steering gear. The steering gear has two independent power plants, each capable of handling the rudder under design conditions. Transfer from one plant to another is effected by a crossover in the steering engine room. In addition, a hand pump which develops a ram cylinder pressure equivalent to one-fourth the maximum ahead rudder torque is provided for rudder positioning in case of main hydraulic plant failure.

Loss of rudder occurs less often than loss of propeller and is thus not as likely to lead to an accident. Submerged objects are also the principal cause of rudder loss although a few failures are known to have been caused by structural fatigue. Emergency measures are similar to those used for loss of propeller, with the possible addition of jury-rig steering.

In conclusion, the loss of a non-duplicated component (propeller, rudder, or shafting) can lead only to accidents that are analyzed in other sections of this report. The fact that over 95% of the world's merchant ships are single-screw vessels is evidence enough that these non-duplicated components are seldom lost. Hence, the risk of loss must be an acceptable one.

Electrical power for vital machinery is supplied by three independent sources. In addition to the ship's service turbine generators, two automatic-starting (< 15 seconds) auxiliary diesel generators are installed. These 750-kw, 450-volt, 3-phase, 60-cycle units serve as standby for the ship's electric plant and operate off the main switchboard. Installed in the navigating bridge deck is a 300-kw emergency diesel generator (with its own switchboard) which starts automatically upon loss of the main switchboard. Power to vital auxiliaries is supplied by batteries floating on the line while the emergency generator builds up to the proper voltage. The emergency generator is sized for normal emergency load (lighting, fire pumps, radio, etc.) plus decay heat removal. Hence electric power to vital components is assured at all times.

C. HULL

The SAVANNAH's reactor and containment foundations are much heavier than the foundations for a conventional passenger-cargo ship's machinery. The heavy longitudinal members are carried well beyond the reactor space transverse bulkheads to tie with a smooth transition into the hold double-bottom structure. Hence SAVANNAH would not be penalized in situations (such as grounding on a ledge) in which discontinuities prejudice longitudinal strength.

Stability of the SAVANNAH is equivalent to that of a normal passenger-cargo ship, and, since there is no fuel oil to be consumed in passage, there will be less variation in GM (the measure of stability) during a long voyage.

SAVANNAH's wheelhouse and navigation bridge were designed to give maximum visibility through the widest possible angle — almost 180 degrees, which is equal to or greater than that of any other passenger-cargo ship. Navigation and communication equipment are of the highest order.

As the following tables show, SAVANNAH's maneuvering capabilities compare favorably with conventional steam vessels. Values in the tables were derived from SNAME Technical Research Bulletin No. 3-5.[5] The figure "55 percent ahead propeller speed" corresponds to the average "full astern" rpm at dead in the water. "Design backing torque" is specified as 80% of ahead torque, at 50% of ahead rpm, and is developed as soon as turbine chest pressure reaches design values, with considerable headway still on the ship. The time to this point is essentially the time to open the astern throttle and close the ahead throttle, or about 20 seconds. The thrust then remains approximately constant until the ship is dead in the water.

TABLE XVI

SHIP CHARACTERISTICS

	Displacements, tons	Number of Screws	Design Speed, knots	SHP, Normal
P2-N-MA 40a (NS SAVANNAH)	21,850	1	20.25	20,000
P3-S2-LS2 (INDEPENDENCE)	30,100	2	22.5	37,000
P2-S1-DN3 (USNS BARRETT)	17,700	1	19.2	12,500
C4-S-1a (MARINER)	21,000	1	20	17,500
P2-S2-R2 (GENERAL BRECKINRIDGE)	20,700	2	19	17,000

TABLE XVII

HEAD REACH UNDER VARIOUS CONDITIONS

	Head Reach in Feet			
	A	B	C	D
NS SAVANNAH	3900	1735	4770	2260
S. S. INDEPENDENCE	4090	1580	4940	2000
USNS BARRETT	3790	1890	4640	2405
MARINER	3780	1720	4530	2170
USNS GENERAL BRECKINRIDGE	3750	1860	4110	2085

A. Stopping from design speed using design backing torque

B. Stopping from 12 knots using design backing torque

C. Stopping from design speed using backing torque at dead in the water corresponding to 55% of normal ahead propeller speed

D. Stopping from 12 knots using backing torque at dead in the water corresponding to 55% of normal ahead propeller speed

TABLE XVIII

COMPARISON OF STOPPING TIME
(From Same Initial Speed)

Using Design Backing Torque

	NS SAVANNAH		MARINER	
Speed of Approach, knots	20	12	20	12
Head Reach, ft	3900	1725	3780	1720
Stopping Time, seconds	247	172	241	175

Using backing torque at dead in the water corresponding to 55% normal ahead propeller speed

	NS SAVANNAH		MARINER	
Speed of Approach, knots	20	12	20	12
Head Reach, ft	4770	2260	4530	2170
Stopping Time, seconds	311	223	300	213

Chapter III

COLLISION AT SEA

A. Penetration and Structural Damage

B. Shock in Collision

III. COLLISION AT SEA

Although the frequency of collisions between ships has declined steadily during the past century, collisions still occur often enough to constitute the greatest single threat to the integrity of the SAVANNAH's containment. That collisions still occur in spite of the revolutionary advances in navigation and communication of the past few years seems to indicate that the problem will never be completely eliminated as long as judgment is an essential part of ship operation. Thus, no matter how well a ship is equipped and manned, there is still the possibility of a damaging collision with another vessel. In this section, the mechanisms and effects of collision are analyzed to show the basis used in the SAVANNAH design to obtain collision protection.

The two potential sources of nuclear hazards in collisions are shock and penetration. The acceleration or deceleration in collision can result in the dislocation of machinery components, rupture of pipe lines, loss of shielding, etc. Penetration in an inelastic collision poses danger of crushing the ship's structure, opening machinery and reactor spaces to the sea, rupture of containment, and dislocation of components.

This analysis is limited to collisions between merchant ships – a limitation that does not severely restrict the analysis. Merchant ship operation differs significantly from naval practice, particularly with regard to formation steaming and close approach maneuvers practiced by naval vessels. In general, merchant ships avoid sailing in close proximity to other ships and hence warship collision reports would not constitute a fair statistical representation of merchant ship collision data. Other special-purpose ships such as ice breakers and patrol vessels are included in the analysis.

The hazards of war service are not considered. The SAVANNAH is not intended to be subjected to gunfire, torpedo attack or other explosive damage. This limits shock effects to those resulting from collision, grounding, and heavy weather.

The two principal effects of collision – shock and penetration – are considered in detail in this chapter.

A. PENETRATION AND STRUCTURAL DAMAGE

The prediction of the effects of ship collisions does not lend itself well to analytical procedures. The structure of both ships is subject to complex progressive failures resulting from the buckling of panels and the shearing, tearing, crushing, and bending of plates and shapes. Most of the work done by the forces on the structure takes place beyond the elastic limit of steel, rendering the normal analytical procedures of structural design completely inapplicable. Therefore, a semi-analytical approach based on over-all damage and kinetic energies lost in collision was developed.

Some 60 serious collisions of recent years were thoroughly examined in an effort to obtain some correlation between the damage sustained in collision, the characteristics of the ships involved, and the circumstances of the accident. Initial data on collision circumstances were supplied by the United States Coast Guard; damages were assessed from American Bureau of Shipping surveys. Additional data were obtained by inspecting the ships involved in recent collisions, and from U. S. Coast Guard hearings.

With the assumption that ship collisions are almost inelastic, the energy lost in a collision may be calculated as follows:

Defining

M_A = the mass of the struck vessel

V_A = the velocity of the struck vessel

M_B = mass of the striking vessel

V_B = velocity of the striking vessel

U = final common velocity component along the course of the striking vessel

dm = virtual increase in mass of the struck vessel stemming from the mass of water which moves with it in collision,

we have

$$U(M_B + M_A + dm) = M_B V_B + M_A V_A,$$

$$U = \frac{M_A V_A + M_B V_B}{M_B + M_A + dm} \qquad (1)$$

and the loss in kinetic energy in collision becomes

$$\left[\frac{M_B\left(M_A + dm\right)}{2\left(M_A + M_B + dm\right)}\right]\left(\overline{V}_B - \overline{V}_A\right)^2 \qquad (2)$$

The effect of the increase in virtual mass (dm) on the loss in kinetic energy is shown in Figure 34. Evidently, the results are not very sensitive to wide variations in dm. From the work of Koch and Dieudonne (on the virtual mass of liquid in the case of a hull vibrating transversely in deep water), it is estimated that the added liquid mass dm should be approximately $0.4\ M_A$.[6,7] Figure 34 shows that for a large variation in dm, the lost energy varies only a few percent.

Substituting this value for dm in Equation (2) and using displacement instead of mass,

$$\text{Loss in KE} \propto \left[\frac{\Delta_B\left(\Delta_A + 0.4\,\Delta_A\right)}{\Delta_B + \Delta_A + 0.4\,\Delta_A}\right]\left(\overline{V}_B - \overline{V}_A\right)^2 \qquad (3)$$

or:

$$\text{Loss in KE} \propto \left[\frac{\Delta_A\,\Delta_B}{1.43\,\Delta_B + 2\,\Delta_A}\right]\left(\overline{V}_B - \overline{V}_A\right)^2 \qquad (4)$$

Evaluating the resistance of ship structure to collision was rather difficult. In particular, the principal contributors to the energy absorption process can only be tentatively identified. After careful survey of damage data, it was found that only the structural members with principal dimensions in the direction of penetrations absorb substantial energies. Of particular importance are the decks and flats of the struck vessel, which serve a dual purpose in resisting collision. They absorb energy in crushing and are also responsible for initiating the collapse of the striking ship's bow. The importance of the latter effect in limiting penetration can be seen by comparing Figures 35, 36, and 37. In striking the tanker in Figure 35, the tanker in Figure 36 was resisted only by the main deck. With her bow virtually intact, she ploughed through the shell and two longitudinal bulkheads of the struck ship, nearly cutting it in two. The STOCKHOLM (Fig. 37) striking the ANDREA DORIA with much the same kinetic energy loss, was resisted by an array of decks, including the inner bottom structure, and penetrated only about 30 ft into the ANDREA DORIA.

This pattern of structural behavior was observed repeatedly in the collision analysis, explaining the complete lack of correlation between kinetic energy loss and depth of penetration. Evidently, a single deck is not sufficient to initiate crushing of the striking ship's bow; hence very deep penetrations can be expected in tankers. With an array of decks, the bow of the striking ship absorbs energy and significantly reduces penetration. No measurable difference in the behavior of longitudinally framed vessels and transversely framed vessels was found, primarily because the entire shell-frame complex on the struck ship contributes little, or nothing, to collision resistance.

Based on the collision studies, a Resistance Factor (R_T) was developed. This is simply a measure of the damaged volume of structural members effective in collision. Members included were

1. decks and flats on both struck and striking vessel,
2. transverse bulkheads in struck vessel when squarely hit,
3. longitudinal bulkheads in the striking vessel, and
4. the component of the shell of the striking ship in the direction of collision.

R_T was then defined as

$$R_T = \sum P_N L_N t_N + \sum P_n L_n t_n,$$

where

P_n, L_n = length and depth of penetration of the nth member of the struck vessel,

t_n = thickness of the nth member of the struck vessel, and

P_N, L_N = length and depth of penetration of the nth member of the striking vessel.

Figure 42 illustrates the methods used in calculating resistance factors and energy loss in collision. Except at low speeds where points are rather widely scattered, a fairly consistent correlation exists between resistance factor and energy loss. It is evident from testimony at inquiries that the masters of striking vessels underestimated the speeds of their ships in collision. Since the striking vessel usually

reverses its engines some time before collision and coasts into the struck vessel, the speed of the striking vessel at impact cannot be determined accurately. When traveling at 5 knots, a 2-knot underestimation of speed results in a 300% underestimation of kinetic energy; at 15 knots the error is only 30%. Hence, for this study, the low-energy collisions ($< 300,000$ ton-knots^2) supply little information.

Several collisions were eliminated because of the extreme angles of encounter. When the bow of the striking ship penetrates at a sharp angle, the problem becomes much more complicated, with the possibility of glancing blows and irregular damage to both striking and struck ships. In addition, penetrations are usually more shallow. As a result, only collisions with angles of encounter approaching 90 degrees were considered. This eliminated only one or two collisions in the high-energy range. Finally, it was assumed that the ship being struck was at rest prior to the collision. This assumption leads to a conservative result and an excellent correlation between lost energy and resulting damage.

A simplification of the equations is made possible by these restrictions. The final common velocity now lies in a direction perpendicular to the axis of the struck vessel and Equation (4) becomes

$$\text{Loss in Ke} \propto \left[\frac{\Delta_B \cdot \Delta_A}{1.43\,\Delta_B + 2\,\Delta_A}\right] \left[V_B \sin\theta\right]^2, \qquad (5)$$

where θ is the angle of encounter.

Further refinements in estimating kinetic energy loss and resistance factors were attempted, but the correlation with the collision data was no better. Specifically, the use of Equation (2) (the exact expression for loss in kinetic energy) to estimate damage to the SAVANNAH gives almost the same results as Equation (5), but with a wider spread of plotted data. In reality, the principal interest lies in the component of energy perpendicular to the axis of the struck vessel, and, since the most dangerous collisions will take place along this path, is not unduly restrictive to consider only the collisions that approach this condition.

The results of calculations are plotted in Figure 43. The degree of correlation is remarkable, particularly in the region of higher energy losses. The shaded portion represents the area of 18 low-energy collisions. The upper range of energies includes all major ship collisions

of recent years, and the validity of the curve seems borne out by the position of the recent collisions of the CONSTITUTION - JALANTA (point C) and the SANTA ROSA - VALCHEM (point B). The latter, from the standpoint of absorbed energy, is the worst accident on record involving merchant ships, and, although collisions involving higher kinetic energies are recorded, they are not sufficiently well-documented to be included. Hence, extrapolation of the curve to very much higher displacements and speeds may not be valid. In particular, it is felt that the collision mechanism may be entirely different in the case of a very large, high-speed ship striking a very small vessel. In general, however, as long as the ships involved are of conventional structure and reasonable relative size, it is believed that the curve can be extrapolated with a fair degree of confidence to include the full range of energy potential in the world's merchant fleet.

SHOCK IN COLLISION

Attempts to define quantitatively the effects of shock on components of colliding ships were not entirely successful. The complexity of ship structure introduces unknown damping forces which make it difficult, if not impossible, to define the problem. A wide range of accelerations can be obtained, depending on what assumptions are made for hardness of ship structure and velocity changes at impact. Theoretical approaches to the problem considered the two extremes: fully elastic and perfectly inelastic impact. These calculations gave values ranging from 0.39 g for the inelastic to 40 g for the elastic.

Figure 44 indicates the possible modes of behavior between the two extremes.

Only three variables can be assumed with any degree of certainty: (1) the initial velocity OA, (2) the final velocity OT, and (3) the fact that the velocity curve should be tangent at M to the TM of the final common velocity. A velocity curve like ACM implies striking in a soft spot, in which case the initial penetration takes place with little change in velocity. This indicates a highly inelastic collision. A curve like ADM implies striking in a hard spot, in which case the initial rate of velocity loss is greatest. The extreme case of the elastic collision is shown in AEM where the initial velocity loss is instantaneous.

The total duration of impact, t, can be approximated by assuming the total depth of penetration and solving for increments of time, t/4 for example. In the case of curve ACM, it can be assumed that total penetration will be large, say 40 ft. Then

$$t/4\,[(26-2) + (20-4) + (12-5) + (7-6)] = 40$$

$$\therefore\ t = 3.33 \text{ seconds}$$

The corresponding greatest deceleration would be about 13.2 ft/seconds^2 or 0.41 g. The curve ADM results in a deceleration of 1.33 g, assuming a 20 ft penetration. Using the methods of elastic impact, a deceleration of 16.6 g is obtained for the curve AEM.

In an effort to narrow the range of g values indicated above, the U. S. Coast Guard and American Bureau of Shipping collision damage data was resurveyed. The following general conclusions were drawn.

1. Ship collisions are highly inelastic. Extensive crushing of structure was noted in all cases, even in a collision involving two battleships, the most rigid structures afloat.
2. There was no evidence whatever of shock damage to machinery components in any collision.

Inspection of the photographs (Figs. 35 through 41) affirms these conclusions. Figures 35 and 36 show the participants in a high speed tanker collision. Penetration was approximately 80% of the beam. According to testimony of the crew, no shock was felt in any part of the ship. Figures 38 and 39 show the results of cargo ship collisions. The superintendent engineer of the line operating the struck vessel shown in Figure 38 was aboard at the time of collision, and reported that no shock was felt.

Figure 39 shows a cargo ship that was struck head on. Again, no shock effects were reported. In all these cases, review of American Bureau of Shipping and owners' damage reports showed no damage to machinery attributable to shock.

From the standpoint of shock, collisions involving specially strengthened vessels are of more interest because penetrations are much less and decelerations proportionately higher. One such collision is illustrated in Figures 40 and 41. The Coast Guard cutter EASTWIND (Fig. 40) was struck by a tanker (Fig. 41) traveling at 15 knots. The EASTWIND's heavy icebreaker construction, which includes an icebelt of 1-1/2 in. high tensile steel, limited the penetration to 12 ft in the strengthened portion of the hull. Despite the very rapid deceleration (which this depth implies), no shock damage resulted.

Perhaps the ultimate in the collision of rigid structures was the wartime collision of the battleships INDIANA and WASHINGTON. The WASHINGTON, making about 18 knots, struck the INDIANA near the No. 5 turret. Penetration extended only to the first torpedo bulkhead, a distance of less than 6 ft. Extensive crushing of the WASHINGTON's bow structure resulted, but a careful survey of machinery revealed no shock effects. Testimony of the WASHINGTON's Officer of the Deck further indicated that no shock was felt on the bridge during the collision.

Inspection of American Bureau of Shipping and Lloyd's damage reports for 28 other major collisions further confirmed the theory of low accelerations during collisions. In the collision of the STOCKHOLM and the ANDREA DORIA, particular attention was paid to possible shock damage to the STOCKHOLM's main engine foundations. In spite of the size of the main diesel and the speed of impact (about 18 knots), no damage was discovered. Testimony of the crew of the ANDREA DORIA revealed that personnel in the vicinity of the point of impact did not even know the collision had occurred.

The repeated testimony that no shock was felt in collision indicates that the upsetting force is even less than that encountered in extreme roll and pitch. Personnel are quite often thrown off their feet in a heavy seaway, yet in no collision was shock of such magnitude experienced. Since estimated g values in several pitch and roll are of the order of 0.6, it is evident that accelerations to be expected in collisions will be very low.

From these findings, no absolute g value can be predicted; however it is evident that the velocity-time curve will be much closer to ACM in Figure 44 than to any of the more extreme cases. The time-dependent function of the velocities for one collision, as derived from the energy absorption curve of Figure 43, is shown in Figure 45. This function is, of course, based on the assumption that there is no instantaneous velocity loss, but after penetration as indicated, decelerations are about 1/3 g. This gives further evidence of negligible shock in collision.

From the above, it is concluded that the accelerations obtained in ship collisions are very low and should not be expected to exceed about 0.6 g.

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FIG. 34: EFFECT OF ADDED MASS OF WATER ON ENERGY ABSORBED

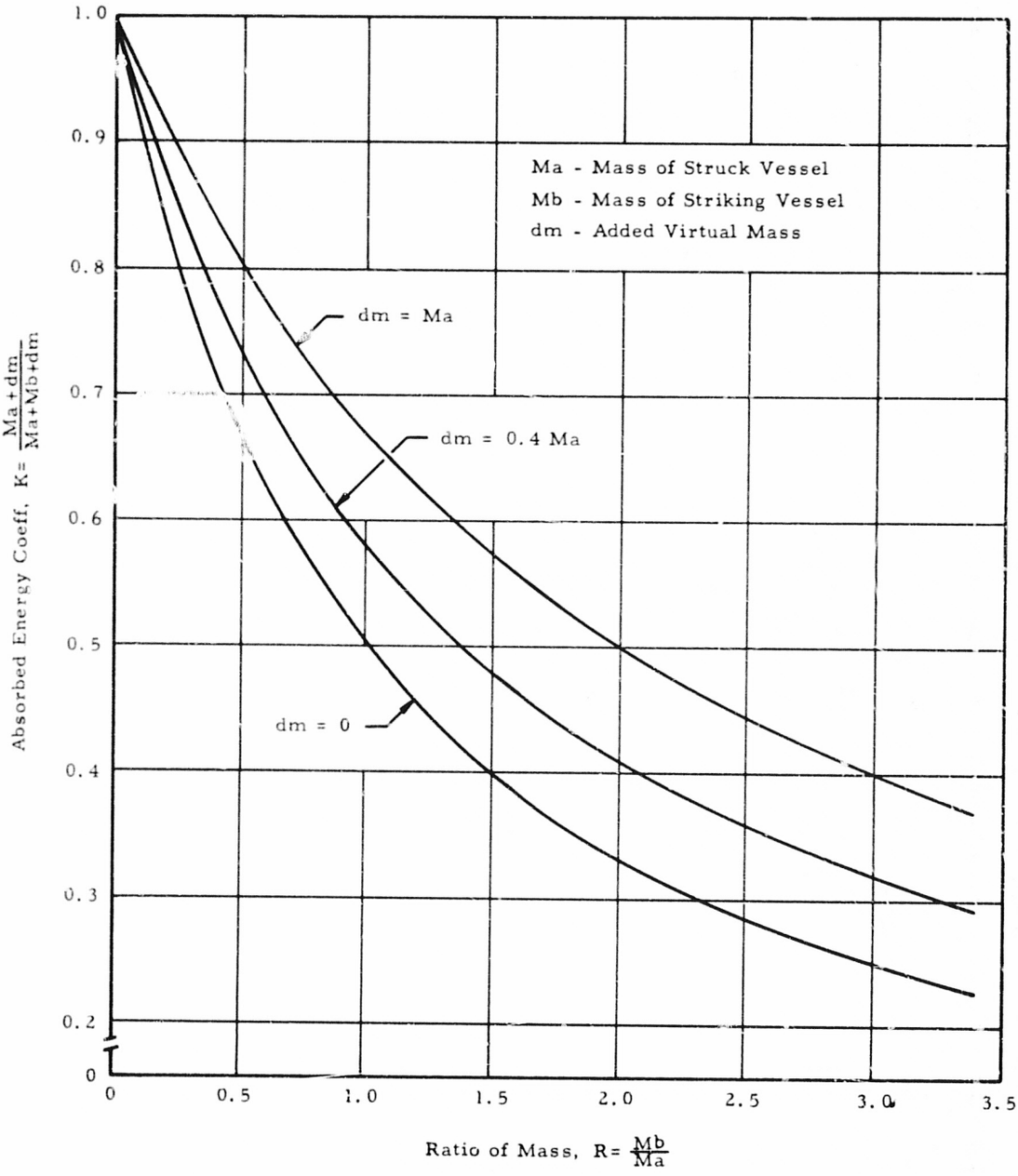

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FIGURE 34

FIG. 35: DAMAGED BOW OF VESSEL STRIKING THE TANKER SHOWN IN FIGURE 36

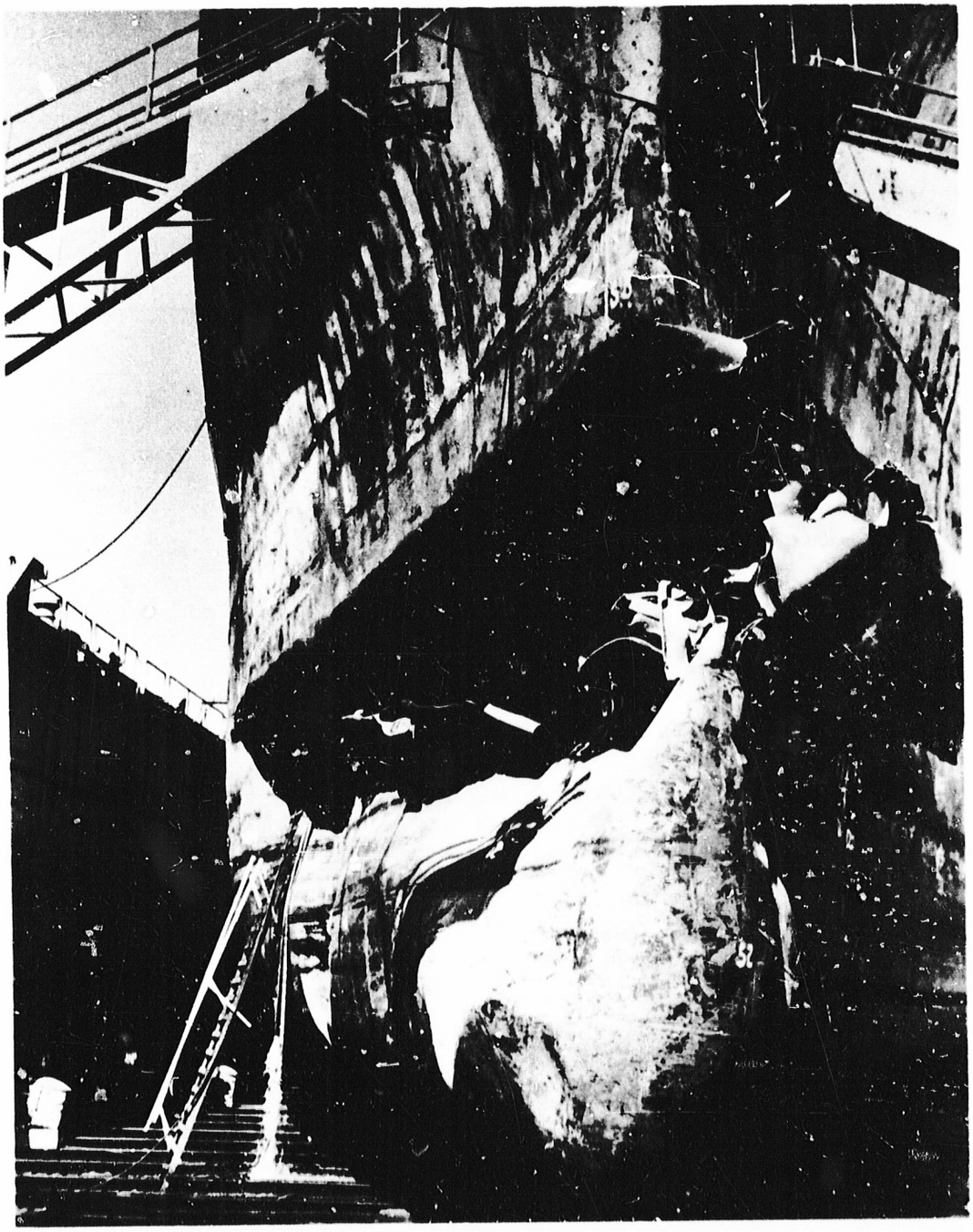


FIGURE 35

FIG. 36: DAMAGED AREA OF STRUCK TANKER
(About 80% of the Beam was Penetrated)

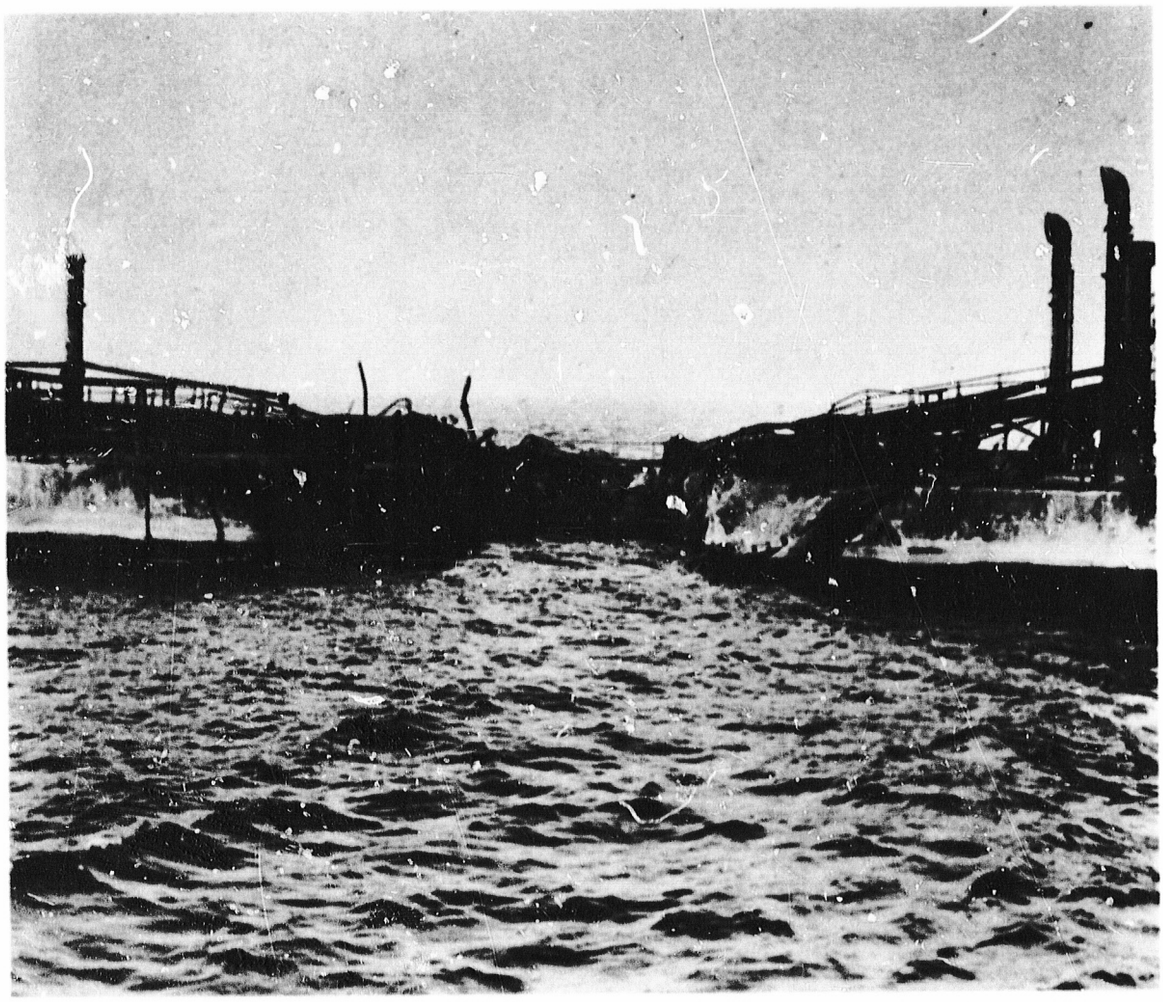


FIGURE 36

FIG. 37: BOW OF MV STOCKHOLM SHOWING DAMAGE SUSTAINED IN STRIKING THE ANDREA DORIA

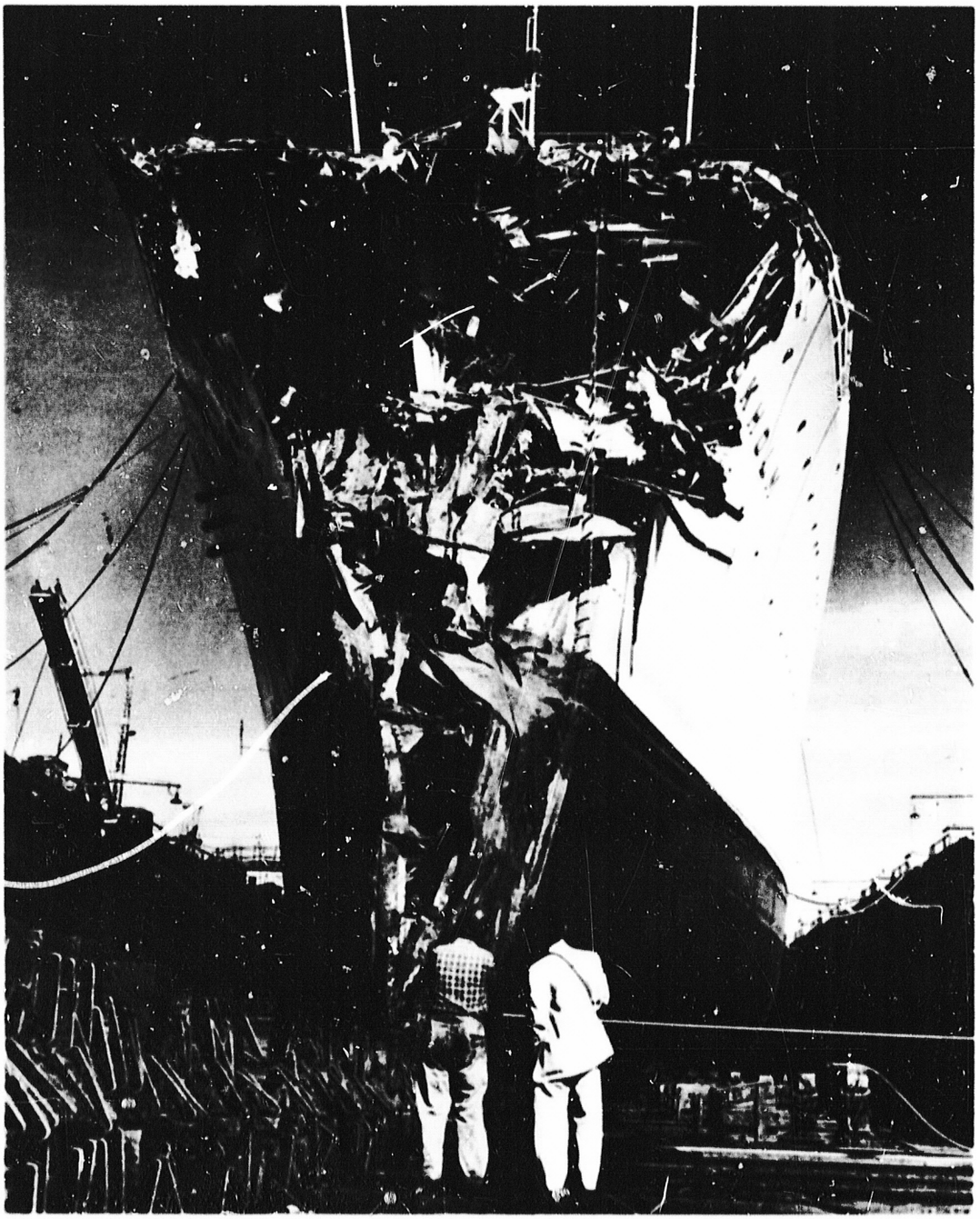


FIGURE 37

FIG. 38: VIEW OF CARGO SHIP STRUCK BROADSIDE

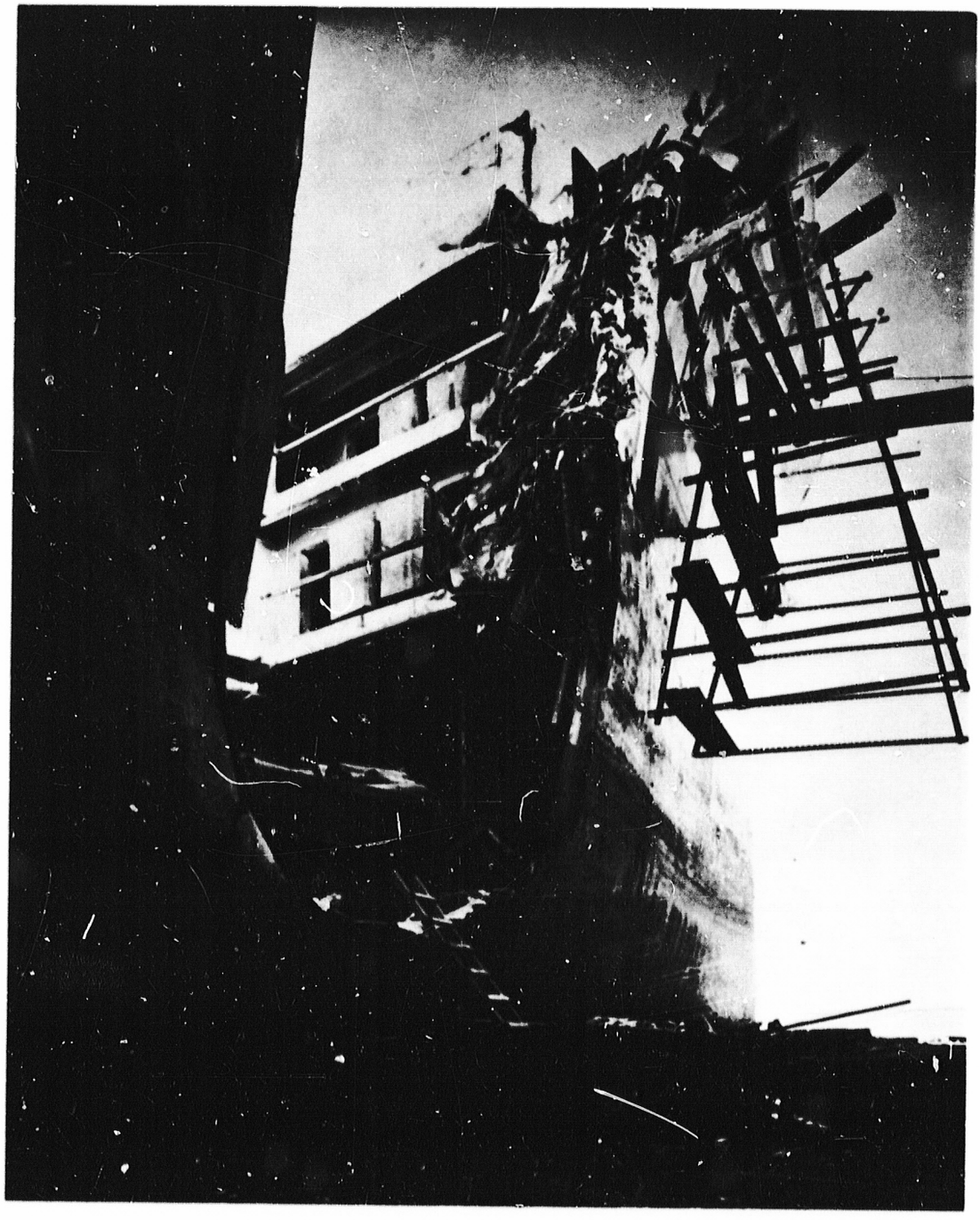


FIGURE 38

FIG. 39: BOW OF CARGO SHIP STRUCK HEAD ON

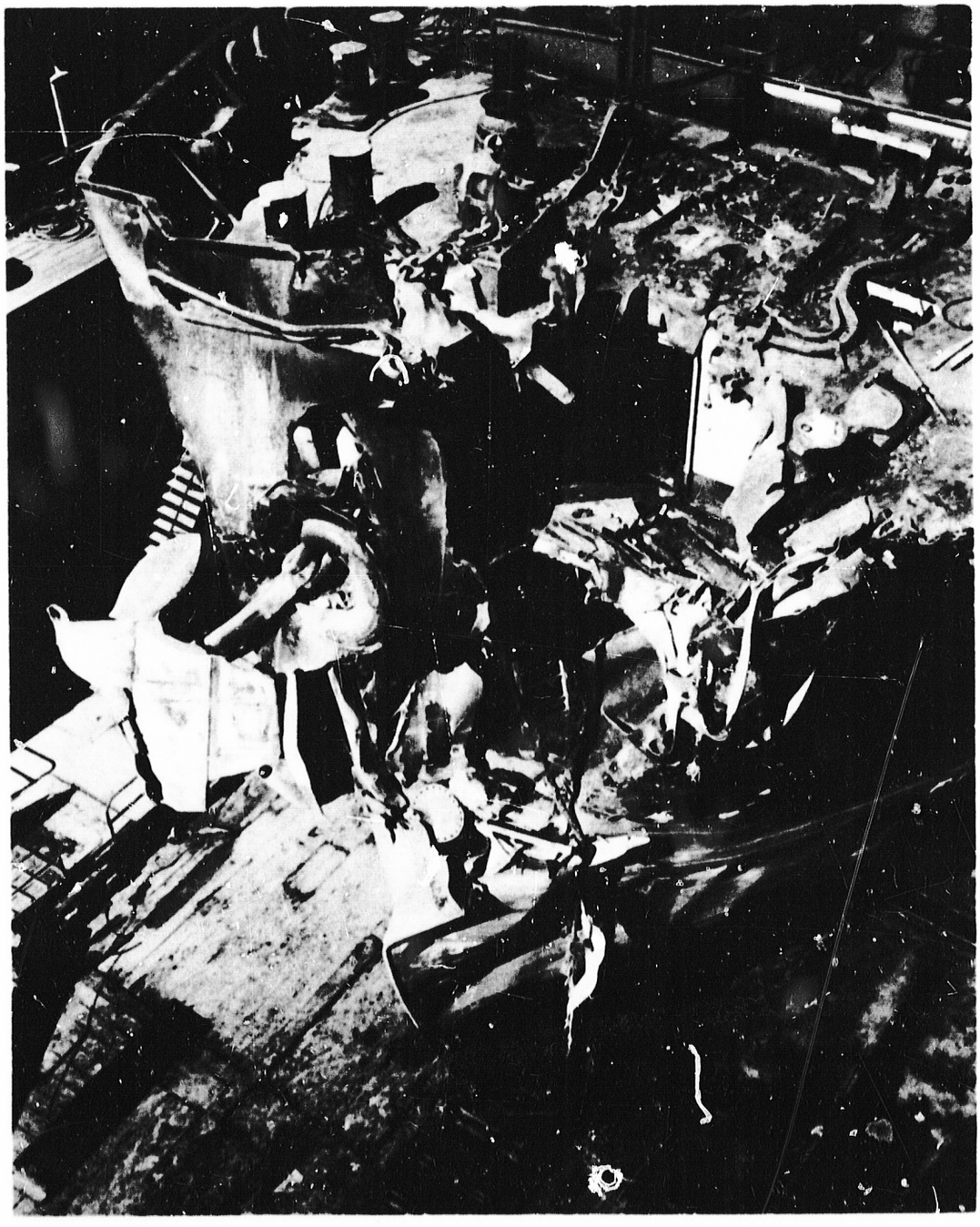
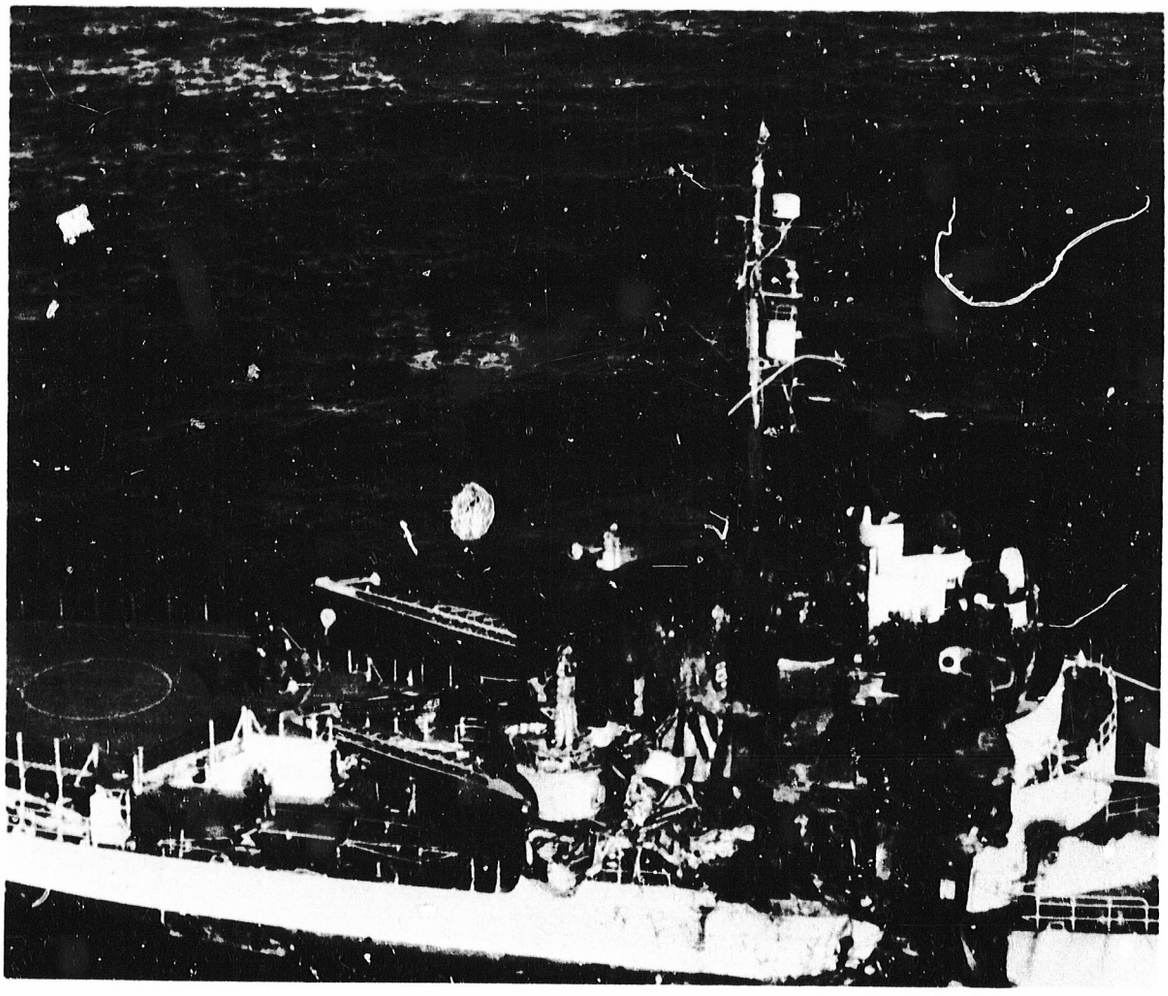


FIGURE 39

FIG. 40: VIEW OF U. S. COAST GUARD CUTTER EASTWIND
(Struck by the Tanker GULFSTREAM Traveling at 15 knots)



U. S. Coast Guard
Photo

FIGURE 40

FIG. 41: DAMAGED BOW OF THE GULFSTREAM

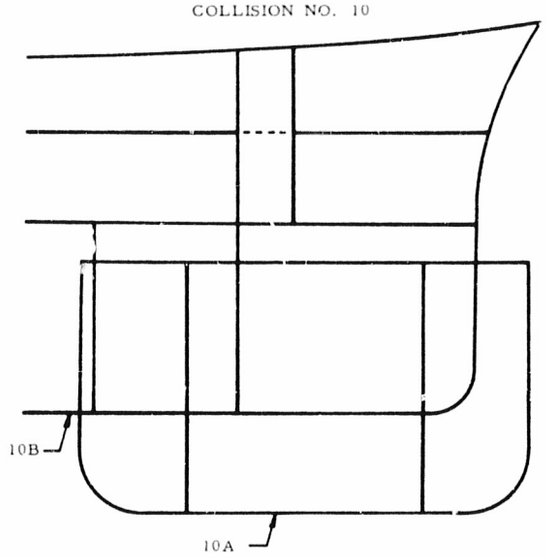


FIGURE 41

U. S. Coast Guard
Photo

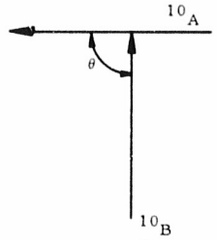
FIG. 42: METHOD USED TO CALCULATE RESISTANCE FACTOR AND ENERGY LOSS

COLLISION NO. 10



COLLISION REPORT DATA:

Δ_A	21800	Tons
Δ_B	19500	Tons
V_A	15	Knots
V_B	15	Knots
L	60	Feet
P	60	Feet
H	39	Feet
θ	90°	



Energy Absorbed in Collision $E_T = \frac{\Delta_A \Delta_B V_B^2}{1.43\Delta_B + 2\Delta_A}$

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RESISTANCE FACTOR CALCULATION NO. 10

STRUCK VESSEL 10_A					STRIKING VESSEL 10_B				
DK	P	L	t	R.F.	DK	P	L	t	R.F.
Main	60	60	.83	2988	NO DECKS INVOLVED				
					SHELL DAMAGE				
					H	P	t	x1.4	R.F.
					5	60	.60	1.4	252
				2988					252

Resistance Factor, Struck Vessel A — 2988
Resistance Factor, Striking Vessel B — 252
Total Resistance Factor, R_T ——— 3240

ABSORBED ENERGY CALCULATION NO. 10

$V_B = 15 \times \text{Sine } 90° = 15$

$E_T = \frac{21800 \times 19500 \times (15)^2}{1.43 \times 19500 + 2 \times 21800}$

$E_T = 1,340 \times 10^3$ TON-KNOT^2

Resistance Factor - R_T	3240
Absorbed Energy - E_T	1340×10^3

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FIGURE 42

FIG. 43: RESISTANCE FACTOR Vs ENERGY LOSS
(Showing Results of Collision Data Analysis)

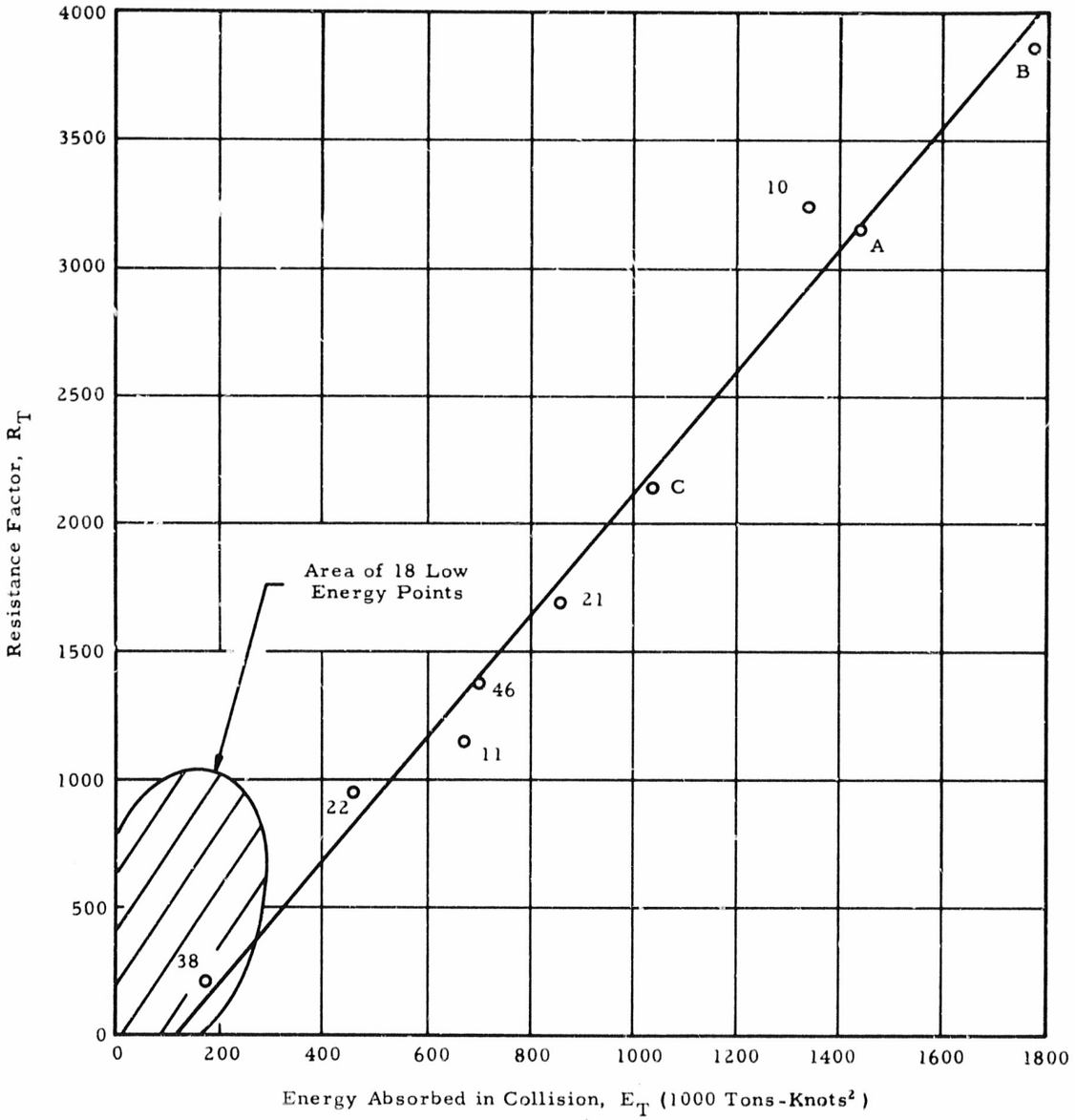

FIGURE 43

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FIG. 44: SHIP VELOCITY DURING COLLISION
(Showing Effect of Various Ship Construction)

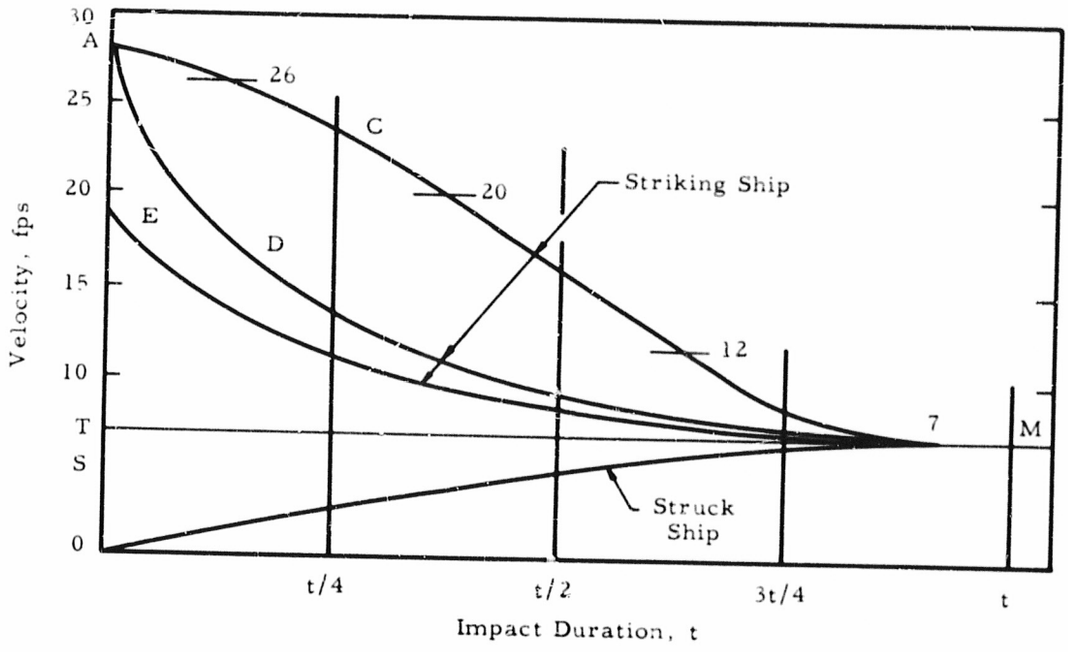

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FIGURE 44

FIG. 45: COLLISION DATA – CARGO-PASSENGER SHIP AND NS SAVANNAH

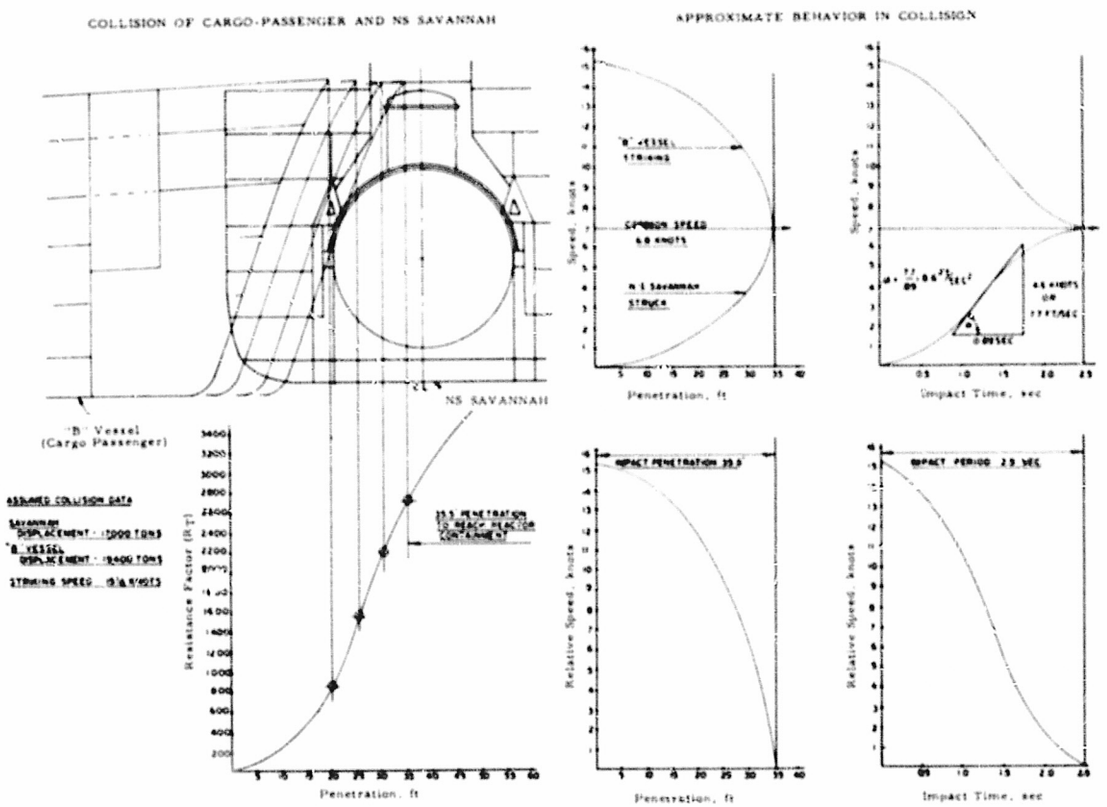

FIGURE 45

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Chapter IV

RESISTANCE OF NS SAVANNAH TO COLLISION

A. Design Criteria for Collision Protection

B. Critical Energy and Critical Speed

C. Kinetic Energy Potential of the World's Merchant Fleet

D. Collision Barrier Design

E. Ships of Greater Kinetic Energy

F. Accuracy Evaluation

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IV. RESISTANCE OF NS SAVANNAH TO COLLISION

If the validity of the energy absorption curve over the full range of the world's merchant fleet is accepted, theoretically, the SAVANNAH could be designed to withstand any conceivable collision. This is not feasible, however, because the SAVANNAH — with a maximum displacement of 21,840 tons — is in the range of small-to-medium size ships with respect to mass and speed. Predicting the effects of a collision with the QUEEN MARY, for example, at 30 knots and about 80,000 tons displacement, extends the analysis into the area of ships of widely different masses and energies. Extrapolation of the energy absorption curve in this area is not well grounded. In addition, the structure required for protection in such a collision would not be practical from a weight and volume standpoint, and by virtue of strength alone, might not conform to the predicted energy absorption curve.

In order to predict penetration and to absorb the maximum kinetic energy, both striking and struck ships must crush and absorb energy. The decks of the struck ship can be expected to crush only as long as their thickness is in approximately the same range as normal ship's structure. When the thickness increases to 3 or 4 in. the deck may fail to crush the striking vessel and perhaps allow penetration to the reactor compartment in the spaces between.

Thus, designing to withstand the maximum collision involves a weight and cost penalty that might radically limit the usefulness of the ship. It was apparent, then, that the SAVANNAH had to be designed to withstand a somewhat less severe collision. At the same time, however, it had to be shown that a negligible probability of a more severe collision exists. It is believed that this has been accomplished in the design of the SAVANNAH.

As mentioned previously, shock loadings in collision are extremely low, evidentally not as great as those encountered in maximum roll, pitch, and heave. Therefore, the machinery foundations and all critical components of the SAVANNAH are designed to normal commercial standards of roll, pitch and heave, with ample safety factors provided. Since it is estimated that shock loadings of approximately 1 g can be tolerated without primary system rupture, it is concluded that shock effects in collision do not constitute a hazard to ship safety.

This conclusion also suggests that the possibility of damage to the reactor plant would be eliminated in a collision in which the SAVANNAH were the striking vessel. The reactor compartment and machinery spaces can be subjected only to accelerations transmitted through all the structure forward of the reactor; these accelerations would obviously be significantly less than those experienced when the SAVANNAH is struck. Since penetration is not a problem when the SAVANNAH is the striking vessel, the reactor and machinery spaces would not be damaged in any such collision.

A. DESIGN CRITERIA FOR COLLISION PROTECTION

Behavior in collision can be accurately assessed only for structures similar to that of conventional ships. Therefore, only the structure outboard of the reactor compartment longitudinal bulkhead is included as "collision resistance". The additional resistance afforded by the collision mat, shielding, and containment vessel is considered separately. A "serious collision" is defined in this analysis as one in which penetration to the reactor compartment longitudinal bulkhead occurs; the energy absorption potential of the structure in this area then determines the maximum accident that can be withstood.

Therefore, by virtue of the designed resistance of the SAVANNAH, the probability of "serious collision" is reduced to a minimum because the magnitude of such "serious collision" can only be achieved by a limited number of ships. This probability, a function of several factors, is fully covered in Part Two, Chapter V. One of these factors is the probability that ship in collision with the SAVANNAH possesses sufficient size and speed to develop enough kinetic energy to penetrate the reactor compartment. This factor of course must be kept to a minimum since it determines the magnitude of the collision that the SAVANNAH can withstand. It is the only important factor whose absolute value can be fixed in the design of the SAVANNAH.

B. CRITICAL ENERGY AND CRITICAL SPEED

Rearranging Equation (5), and assuming the worst case of a 90-degree collision, gives

$$\text{Loss of KE} \propto \Delta_B V_B^2 \left[\frac{\Delta_A}{1.43\,\Delta_B + 2\Delta_A} \right] \quad (6)$$

From this

$$\Delta_B \cdot V_B^2 \propto (\text{loss in KE}) \frac{1.43\Delta_B + 2\Delta_A}{\Delta_A}, \quad (7)$$

and

$$V_B \propto \sqrt{(\text{loss of KE}) \frac{(1.43\Delta_B + 2\Delta_A)}{\Delta_A \cdot \Delta_B}} \quad (8)$$

In Equations (7) and (8), the quantities $\Delta_B V_B^2/2$ and V_B are designated critical energy (E_c) and critical speed (V_c). These are the energies and the speeds required by any given vessel to reach the SAVANNAH's reactor compartment longitudinal bulkhead. The loss in kinetic energy is obtained from the energy absorption curve of Figure 43 by estimating the resistance factor of the striking ship and the SAVANNAH. The critical energy is thus a function of the bow structure of the striking ship, the SAVANNAH's collision protection structure, and the draft at which each is operating (which determines the interacting structural areas). Obviously then, the displacement and the design speed of the striking ship will determine the maximum kinetic energy available at any particular load condition. If the kinetic energy is less than E_c, for the vessel type, the striking ship does not constitute a threat to the SAVANNAH.

Although the preliminary estimates of potential danger to the SAVANNAH were based on critical energy, critical speed is a more convenient criterion once the type of ship involved has been determined. The calculation of both criteria is illustrated in Figure 46 which represents a hypothetical collision between the SAVANNAH (at mean draft of 24 ft-0 in.) and a loaded T2 tanker.

The calculated resistance factor is based on a 90-degree collision, the worst case, and the depth of penetration is assumed to extend to the reactor compartment bulkhead. The length of the damaged area on

SAVANNAH is assumed to be the beam of the striking vessel modified by a factor of 1.33. This factor was derived from the data of the collision studies since the speed of the SAVANNAH itself in collision cannot be predicted. It represents an average value for 50 collisions, in most of which the struck vessel was traveling at a relatively low speed. If SAVANNAH were motionless at the time of collision it would be inapplicable, but this situation is highly improbable and occurs generally in harbors. In Part Two, Chapter V, it is shown that harbor collisions involving passenger vessels are virtually non-existent.

The load conditions of the two vessels in collision has a marked effect on the structural members involved. No general rule is universally applicable, but in most cases the worst condition, with respect to the structure involved, exists when SAVANNAH is loaded and the striking ship is light. In particular the T2 tanker in Figure 46 would strike SAVANNAH well above the inner bottom. This is not necessarily the worst condition with respect to penetration because of the greatly reduced mass of the striking vessel but the drafts of the vessels must be considered throughout the analysis.

C. KINETIC ENERGY POTENTIAL OF THE WORLD'S MERCHANT FLEET

A detailed survey of the world's merchant fleet was undertaken to determine the distribution of maximum kinetic energy of each vessel. The maximum kinetic energy, derived from the full-load displacement and the designed sea speed — the maximum continuous sea speed — of the ship, represents the energy that must be absorbed in collision with the SAVANNAH.

The principal sources of information on the world's merchant fleet are the registers of the various classification societies such as Lloyd's, American Bureau of Shipping, and Bureau Veritas. In addition, the Statistics and Special Studies Office of the Maritime Administration has published statistical analyses of the world fleet. The Maritime Administration data however were based on criteria not directly applicable to this study; hence it could be used only as a guide. Lloyd's Register was chosen as the basic source of information because

1. it has the widest coverage of all the classification societies, listing some 35,000 entries of all types and flags,
2. it presents more information on each vessel, and
3. it lists many vessels registered with other classification societies.

From a random sample of every tenth ship listed in Lloyd's, a cumulative grouped distribution was plotted as a function of maximum kinetic energy (Fig. 47). All non-passenger vessels constructed prior to 1937 were excluded, their smaller size and slower speed in general not being representative of ships the SAVANNAH would encounter over the next 20 years. Non-self propelled vessels and non-seagoing vessels were also excluded since they obviously could not harm the SAVANNAH and would serve to bias the distribution in the low energy groups. Further, since many hundreds of these vessels are not registered anywhere it was believed that complete exclusion would provide a more realistic analysis.

The high energy section of the cumulative distribution curve, the area of greatest interest, was checked by actual count of the vessels in this category. This group consists almost entirely of large tankers

and passenger ships. Separate distributions for these classes are plotted in Figure 48.

Critical energies and speeds were then calculated for different types and classes of vessels that might collide with the SAVANNAH under various load conditions. It was immediately evident that, considering normal ship's structure, a large percentage of the world's merchant fleet could be eliminated, including most dry cargo vessels whose size and speed would not imperil the SAVANNAH. However, since a large number of potentially dangerous ships remained, it was necessary to increase the SAVANNAH's collision protection structure. By increasing the resistance factor, critical energy and speed were increased for the entire fleet.

D. COLLISION BARRIER DESIGN

The degree of increase of collision resistance was determined by the desire to eliminate the maximum number of ships without exceeding the range of existing collision data. From the histograms of Figure 49, the optimum design point is for tankers of kinetic energy of 2.5×10^6 ton-knots^2. Beyond 2.5×10^6 ton-$\overline{\text{knots}}^2$, the histogram for passenger ships does not fall off as rapidly as that for tankers; the total number of ships involved would require a large increase in structural strength to eliminate a very few ships. The peak, at 2.5×10^6 ton-$\overline{\text{knots}}^2$, is primarily due to the war-built T2 tankers, many of which are still operating in their original form. Therefore, the SAVANNAH's collision protection was designed to withstand a collision with a T2 tanker at full load and sea speed.

In Table XIX, the scantlings of the SAVANNAH's major structural units are compared with those of conventional ship design.

TABLE XIX
COMPARISON OF DECK SCANTLINGS

Area	Thickness, in. Conventional Design	SAVANNAH
Promenade Deck	0.39	0.45
"A" Deck	0.625	0.625
"B" Deck	0.49	0.625
"C" Deck	0.56	0.625
"D" Deck	0.28	0.625
14'-0" Flat	0.28	0.625
Tank Top	0.60	0.625

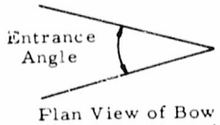
In addition the outboard strakes in B, C and D decks were increased from an average of 0.50 in. to 0.875 in.

TABLE XX

ESTIMATED BEHAVIOR OF NS SAVANNAH IN COLLISION WITH STANDARD VESSELS

Vessel	Displacement, tons	Critical Kinetic Energy, tons-knots^2	Entrance Angle, degrees	Decks, number	Critical Speed, knots	Design Speed, knots
Passenger and Cargo Vessels						
VC2-S-AP2 (VICTORY)	14832	1.67×10^6	42.5	8	15	16.5
C3-S1-BR1	16700	2.41×10^6	40.5	11	17	17.3
SANTA ROSA	19364	2.48×10^6	35.5	10	16	20.0
MARINER	21000	2.69×10^6	41.2	11	16	20.0
BRAZIL	22700	2.91×10^6	36.6	10	16	23.0
INDEPENDENCE	30090	2.95×10^6	35.8	11	14	22.5
CARONIA	31240	3.06×10^6	35.7	12	14	22.0
MAURETANIA	34000	2.87×10^6	33.5	11	13	22.0
AMERICA	35440	3.00×10^6	24.6	14	13	22.0
NIEUW AMSTERDAM	36235	3.55×10^6	41.2	10	14	20.5
QUEEN MARY	77400	4.68×10^6	25.2	13	11	29.0
Tankers						
T2*	13700	2.22×10^6	51.0	9	16	15
BALTIMORE CITY SERVICE*	19200	2.61×10^6	58.6	8	16.5	16.5
ESSO SUEZ*	19500	2.82×10^6	63.0	8	17	16
W. ALTON JONES*	22600	2.71×10^6	52.8	8	15.5	17.5
T2▽	23000	2.59×10^6	57.2	8	15	15
PHOENIX*	25700	2.89×10^6	82.4	6	15	15.1
SINCLAIR PETROLORE*	34000	4.35×10^6	82.8	6	16	14.2
BALTIMORE CITY SERVICE▽	42700	5.80×10^6	70.5	9	16.5	16.5
UNIVERSE LEADER*	48000	6.13×10^6	84.2	7	16	15.3
W. ALTON JONES▽	49636	5.58×10^6	94.6	10	15	17.5
PHOENIX▽	56658	6.37×10^6	88.8	9	15	15.1
UNIVERSE APOLLO*	63000	8.05×10^6	90.0	7	16	15.0
SINCLAIR PETROLORE▽	75630	12.24×10^6	98.4	10	16	14.2
UNIVERSE LEADER▽	106690	12.00×10^6	94.5	9	15	15.3
UNIVERSE APOLLO▽	137150	21.00×10^6	96.8	9	17.5	15.0

* In ballast
▽ Full load



Plan View of Bow of Striking Vessel

NOTES

1. NS SAVANNAH assumed in operating condition:

Draft	24 ft - 0 in.
Displacement	17000 tons
Speed	20 knots

2. Critical speed – speed at impact of striking vessel to reach SAVANNAH longitudinal bulkhead (90° Collision)
3. Entrance angle – measured at design draft of striking vessel
4. The number of decks refers to the total resistive decks of the striking vessel and the SAVANNAH in collision.

E. SHIPS OF GREATER KINETIC ENERGY

Still to be evaluated are the number of ships in the world's merchant fleet that possess sufficient size and speed to penetrate the SAVANNAH's reactor compartment longitudinal bulkhead.

The critical energies and speeds for the ship types of greater kinetic energy were evaluated, borderline classes in particular being considered at all load conditions. Table XX lists the characteristics of the various classes and types.

A particularly interesting characteristic of tankers is illustrated in Figure 50. Plotted against displacement, the critical speed reaches minimum at a displacement of about 35,000 tons. Thus the huge new tankers of 65,000 - 105,000 tons displacement have a higher critical speed than the so-called "Supertankers". This behavior is explained by the ships' extremely bluff bows (large entrance angles) which result in much larger areas of crushable material in the decks and flats forward. Since these vessels are designed with relatively low sea speeds, they apparently will not be of concern to the SAVANNAH. The trend toward larger and larger tankers therefore need not affect the analysis here.

The effect of the collision mat and the reactor compartment bulkhead complex itself cannot be evaluated empirically because no data on the behavior of such a structure in collision are available. However, the effect in resisting collision can be estimated from the curves of deceleration in collision (Fig. 45), and methods of impact stress analysis.[8] A conservative estimate of the resistance of the stiffened bulkhead complex is an effective decrease of 1 knot in the speed of the striking ship. One of the principal effects of the collision mat, that of preventing local punching, does not affect the collision analysis.

Considering the 1 knot decrease in speed, only 2.5% of the world's merchant fleet is capable of damaging the SAVANNAH's reactor compartment, assuming that the entire fleet always operates at full speed and full load displacement. This last assumption is obviously incorrect. Tankers, for example, operate in ballast about 50% of the time, and dry cargo ships operate at full displacement much less than half the time. Therefore, the operating characteristics of the

world's fleet were surveyed in an attempt to evaluate the effects of part load operation on the size of the potentially dangerous group of ships.

Table XXI lists typical draft conditions for various ships, considering, of course, possible increase in speed under reduced loads. Applying this data, the potentially dangerous group is reduced to about 2% of the total at any given time.

TABLE XXI

TYPICAL OPERATING CONDITIONS
(Assumed Average Condition of World Fleet)

Vessel	% of Fleet	% Time	% Draft
Tanker	10	50	55
Passenger	1	50	80
Passenger-Cargo	5	25	60
		50	80
		25	100
Cargo	74	25	60
		50	80
		25	100
Others	10	50	60

NOTE: Weighted average, 50% of time world fleet is at 60% displacement.

Collisions occurring at angles of less than 90 degrees were not considered in developing the energy absorption curve owing to the greater complexity of the structural failure and the difficulty of identifying the collision-resistant members. Although most serious collisions occur at angles around 90 degrees, the average value is about 70 degrees. Thus credit can be taken for the resulting increase in resistance factor — found to be equivalent to a 1-knot decrease in

design speed with a corresponding decrease of the dangerous group of ships to 0.9% of the total.

A further 1-knot decrease in design speed was assumed in order to account for efforts to stop the striking ship immediately before collision. In all cases of collision on record there is no instance of the striking ship not making some effort to stop. Occasionally the full astern signal came too late to reverse the screws, but invariably the striking ship cuts its ahead power substantially and coasts into the struck vessel. In most cases speed is substantially reduced (more than 2 or 3 knots) before the instant of impact. The 1-knot decrease is therefore considered reasonable, reducing the dangerous group to 0.7% of the total. This figure then is a reasonable final estimate of the number of ships that at any time possess sufficient kinetic energy to reach the SAVANNAH's reactor compartment longitudinal bulkhead.

F. ACCURACY EVALUATION

The accuracy of the preceding analysis is completely dominated by the validity of the absorbed energy curve (Fig. 43) which is believed to be a fair representation of the collision mechanism within limits of available data. Thus far each major collision has fitted the curve remarkably well.

Extrapolation of the curve to much larger vessels may not be justified. However, in determining the number of ships that could seriously damage the SAVANNAH, the borderline cases eliminated in Part Two, Chapter V lie in the range between the ANDREA DORIA and the SANTA ROSA collisions. Since minor reductions in speed do not affect the larger ships, whose design speed may exceed the critical by as much as 20 knots, the analysis is not actually based on an extrapolation.

Further, it is believed that the results of the analysis are very conservative. Most collisions involving passenger ships occur under conditions of poor visibility. Although it has been established that ship captains do not significantly reduce speed during a fog, it is customary to reduce speed by a few knots. Just prior to a recent collision between the CONSTITUTION and the JALANTA, the CONSTITUTION had reduced speed from 22 to 18 knots. At the actual time of collision, the CONSTITUTION had succeeded in further reducing speed to 14 knots. Although the CONSTITUTION, whose design speed is 7.5 knots in excess of her critical speed, is very definitely in the group of potentially dangerous ships, the JALANTA collision would not have penetrated the SAVANNAH's reactor compartment. Figure 51 shows that minor speed reductions significantly reduce the size of the dangerous group. A speed reduction of only 2 knots will reduce the size of the group by a factor of seven. Thus it is felt that the value 0.7% is highly conservative.

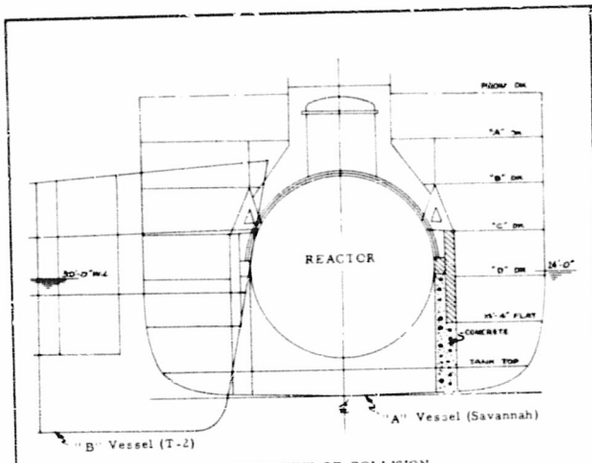
No great change in the number and distribution of the world's merchant fleet, and hence no great change in the percentage of vessels capable of damaging SAVANNAH, is expected over the next 20 years. Of the two major classes of vessels which are capable of damaging SAVANNAH, passenger ships and tankers, the latter have been shown to reach a maximum of damage potential in the intermediate range of

size. The huge new supertankers (85,000 tons dwt) are generally being designed with low sea speeds and very bluff entrance angles, so that in most cases they have critical speeds of 1 to 2 knots greater than their design speeds. Since expansion of the tanker fleet will most probably lie in this size and speed range, the tanker fleet of 10 or 20 years from now may have proportionately fewer vessels capable of damaging SAVANNAH.

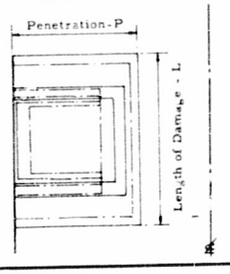
A somewhat different situation exists with passenger ships. Those in the group capable of damaging SAVANNAH usually have design speeds greatly in excess of critical speeds. Hence a general speed increase in new passenger ships will have no effect, and only the number at sea can change the proportion of the fleet capable of damaging SAVANNAH. It is felt that the total number of such vessels is unlikely to change substantially. Following the wave of postwar building to replace lost tonnage, most passenger ships have been built as replacements for existing vessels. Since both the old and the new ships are capable of penetrating to SAVANNAH's reactor compartment, increases in size and speed will have no effect on the number of ships in group.

A slight increase in the number of cargo ships in the dangerous group might be expected if size and speed were increased. Most of the new cargo vessels in the world's merchant fleet are much too small and too slow to cause damage to the SAVANNAH's reactor compartment. Thus the percentage of dangerous vessels in the world's merchant fleet is not likely to change significantly during the next 20 years.

FIG. 46: CALCULATION OF RESISTANCE FACTOR AND CRITICAL SPEED (NS SAVANNAH Struck by Loaded T-2 Tanker)



ELEVATION VIEW OF COLLISION



PLAN VIEW OF DAMAGED AREAS

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30 CHURCH ST. NEW YORK, N. Y

RESISTANCE FACTOR CALCULATION

STRUCK VESSEL (SAVANNAH)					STRIKING VESSEL (T-2)				
DK	P	L	t	R.F.	DK	P	L	t	R.F.
B	23.0	30.2	0.70	485	Foc'le	24.5	33.8	0.40	331
C	17.3	20.7	0.70	250	Main	22.0	21.5	0.50	236
D	17.3	17.5	0.70	212	2nd	20.0	16.7	0.50	167
14'-4	16.8	15.0	0.63	159					
T.T.	14.0	13.4	0.63	118	H	R.P. OF STRUCK VESSEL	t	x1.4	
			Σ R.F	1224	5.0	88.4	0.50	1.4	310
INCREASE IN DAMAGE DUE TO FORWARD MOTION OF SAVANNAH			1.33 x Σ R.F	1632					1044

Resistance Factor, Struck Vessel	1632
Resistance Factor, Striking Vessel	1044
Total Resistance Factor, R_T	2676
Absorbed Energy, E_T	1229×10^3
Displacement, Struck Vessel Δ_A	17000
Displacement, Striking Vessel Δ_B	23000

CALCULATION OF SPEED TO REACH REACTOR COMP'T

$$V_B = \sqrt{\frac{E\,(1.43\,\Delta_B + 2\,\Delta_A)}{\Delta_A\,\Delta_B}} = \sqrt{\frac{1229 \times 10^3\,(1.43 \times 23000 + 2 \times 17000)}{17600 \times 23000}}$$

Speed Required V_B = 15 Knots

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FIGURE 46

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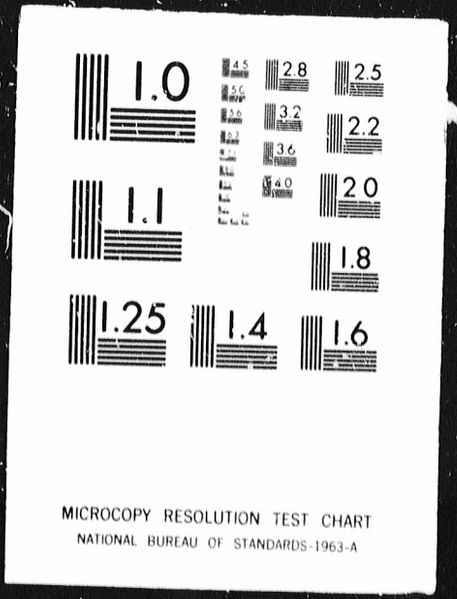
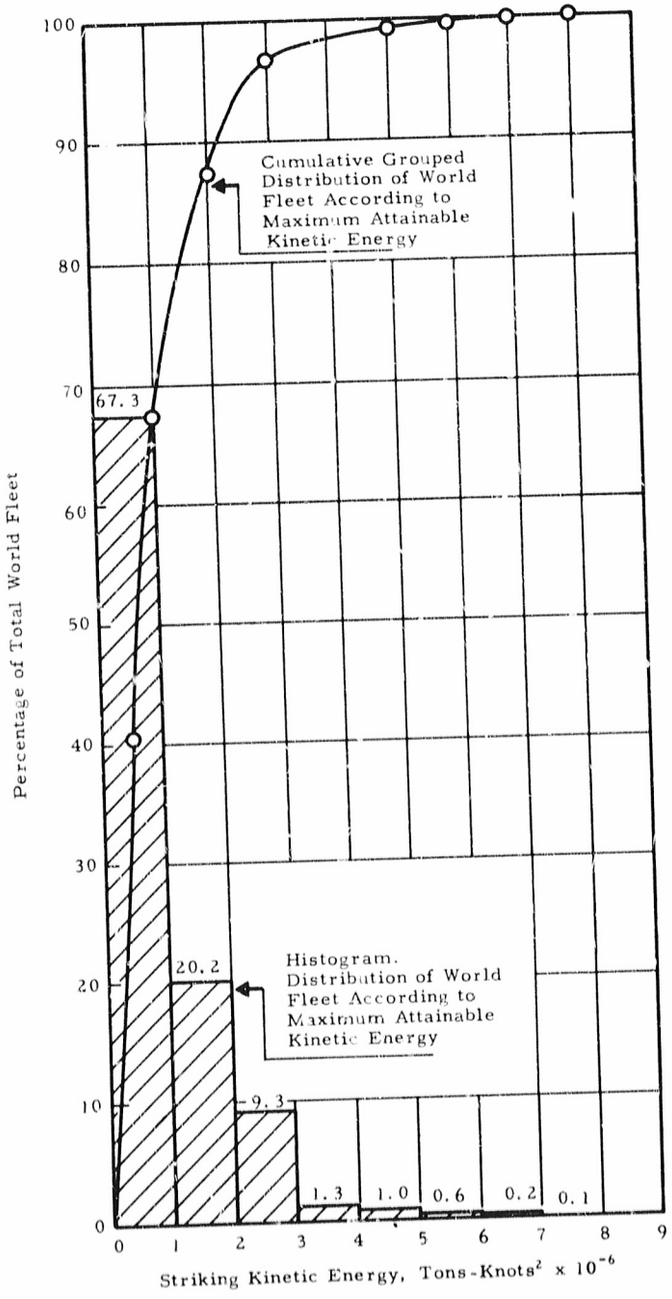


FIG 47: CUMULATIVE DISTRIBUTION OF WORLD'S MERCHANT 'LEET AS A FUNCTION OF MAXIMUM KINETIC ENERGY



NOTES

1. Per cent frequencies are based on every tenth listing in Lloyd's Register 1958-1959 for ships constructed 1937 or later. Barges, tugs, and wooden vessels excluded.
2. All passenger ships of 10,000 gross tons or more are included regardless of date of construction.
3. Kinetic energy calculated assumed all ships at maximum displacement and design speed.

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FIGURE 47

FIG. 48: CUMULATIVE DISTRIBUTION OF TANKER AND PASSENGER FLEETS AS A FUNCTION OF MAXIMUM KINETIC ENERGY

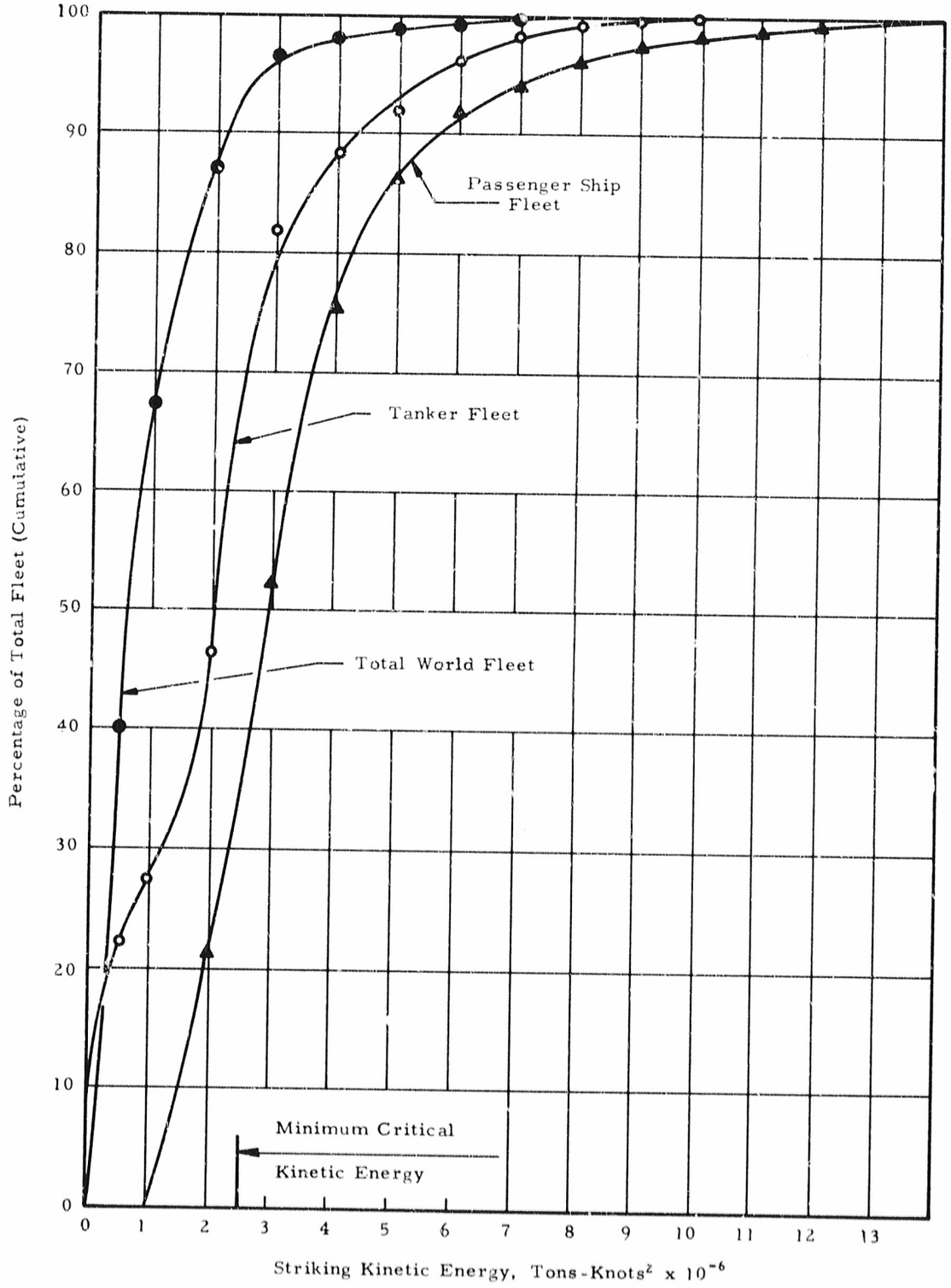

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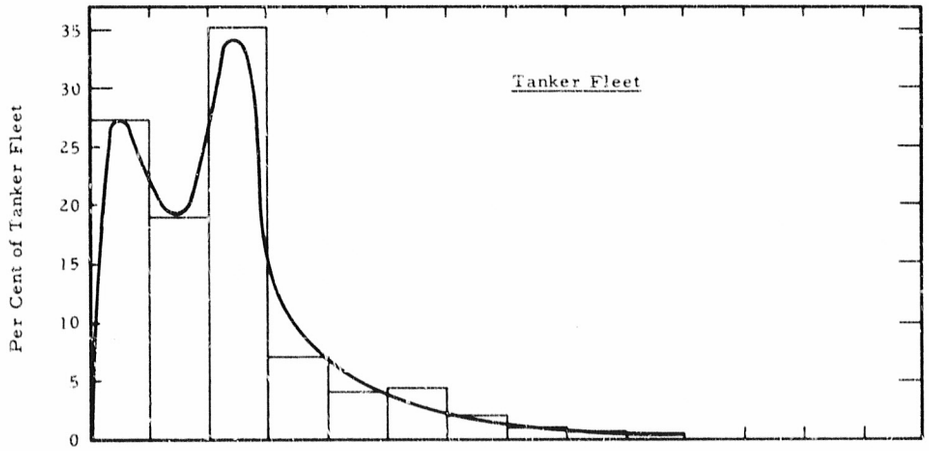
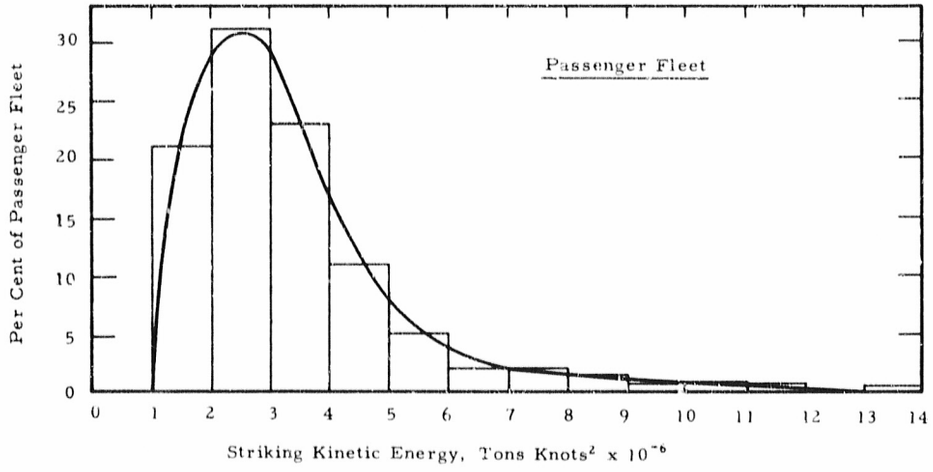
FIGURE 48

FIG. 49: PERCENTAGE OF PASSENGER AND CARGO FLEETS Vs MAXIMUM STRIKING KINETIC ENERGY



NOTES

1. Per cent frequencies are based on all Listings in Lloyd's Register 1958-1959 for tankers constructed 1937 or later.
2. Kinetic Energy calculated all tanker at maximum displacement and design speed.



NOTES

1. Per cent frequencies are based on all passenger ships of 10, 000 gross tons or more regardless of date of construction.
2. Kinetic Energy calculated assumed all passenger ships at maximum displacement and design speed.

FIGURE 49

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FIG. 50: DISPLACEMENT OF STRIKING VESSELS Vs CRITICAL SPEED

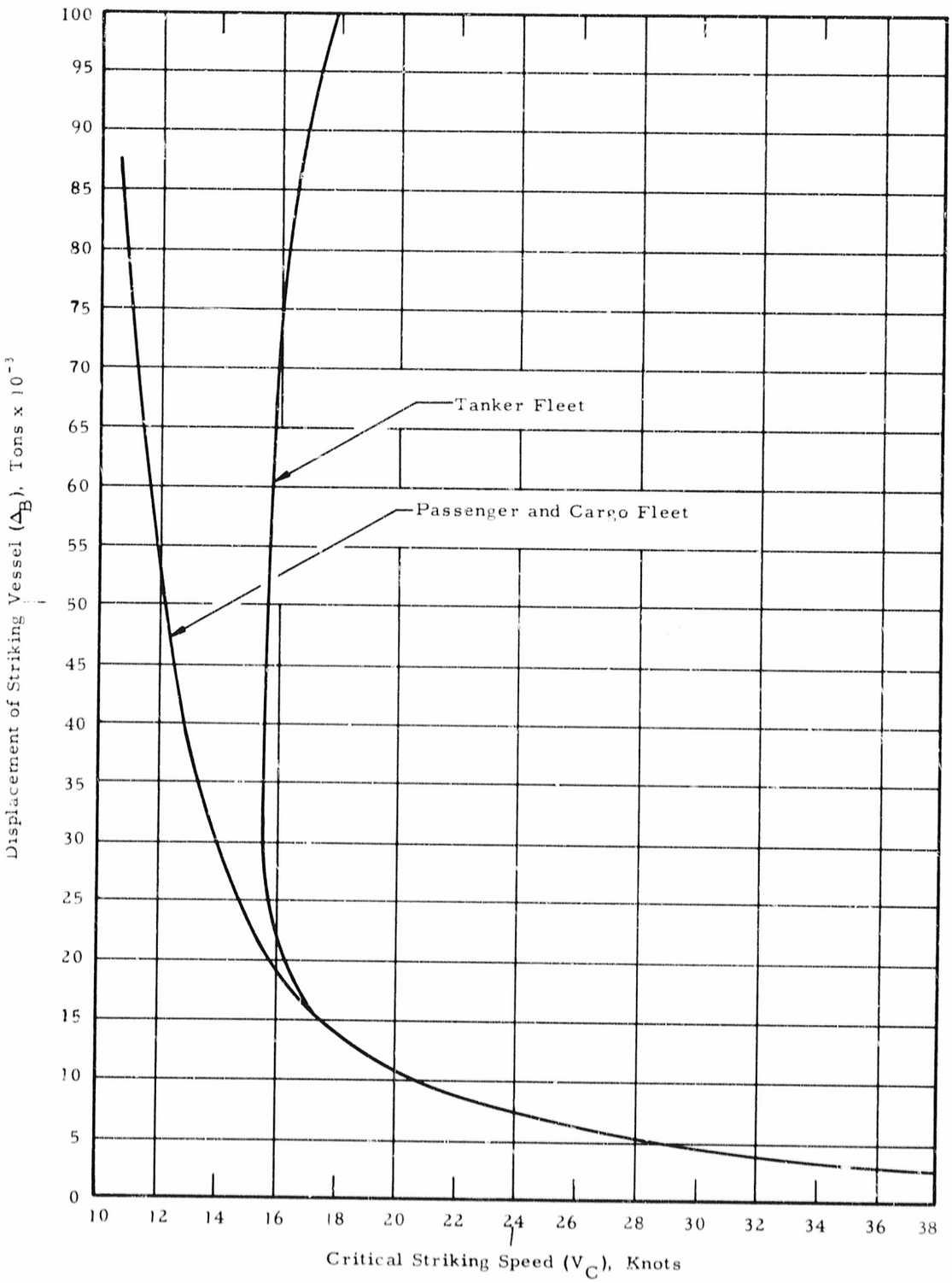

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FIGURE 50

FIG. 51: PERCENTAGE OF WORLD FLEET CAPABLE OF PENETRATING THE REACTOR COMPARTMENT

Condition Description		% Capable	Remarks
Total world fleet operating at full load displacement and design speed.		2.5	Maximum percent capability of world fleet to penetrate to reactor compartment.
Total world fleet in average operating condition (see below) and at full design speed.		2.0	Realistic percent capability at any given time. No consideration to speed reduction.
Total world fleet in average operating condition (see below) but with striking speed reduced for various collision considerations.	Speed Red. in Knots		
	1	0.9	Capability considering average encounter angle of 70 degrees reducing striking speed vertical component by 1 knot.
	2	0.7	Capability assumes 1 knot of speed reduction before collision (considered most probable %).
	3	0.2	Capability assumes 2 knots of speed reduction before collision.
	4	0.1	Approximate approach speed reduction into harbors.

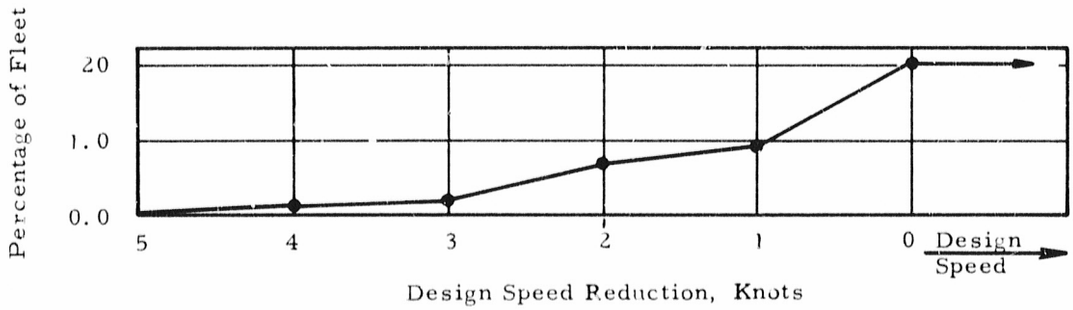

FIGURE 51

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Chapter V

PROBABILITY OF REACTOR SYSTEM DAMAGE IN COLLISION

A. Probability of a Collision Occurring

B. Probability of SAVANNAH Being the Struck Vessel

C. Probability of Collision Occuring in Way of the Reactor Space

D. Probability of Collision Breaching the Reactor Compartment

E. Accuracy

V. PROBABILITY OF REACTOR SYSTEM DAMAGE IN COLLISION

Having established the limits of size and speed necessary for a ship to penetrate the reactor compartment, the probability of a collision of greater magnitude must be determined.

Consider the following separate events, all of which are necessary for a collision to damage the reactor compartment.

$$P[A] = P[\text{A collision involving SAVANNAH occurs}] \tag{9}$$

$$P[B] = P[\text{Given A, SAVANNAH is struck}] \tag{10}$$

$$P[C] = P\left[\begin{array}{l}\text{Given A and B, collision occurs in}\\ \text{way of the reactor compartment}\end{array}\right] \tag{11}$$

$$P[D] = P\left[\begin{array}{l}\text{Given A, B, C, the striking ship has}\\ \text{sufficient size and speed to reach the}\\ \text{reactor compartment}\end{array}\right] \tag{12}$$

$$P[E] = P\left[\begin{array}{l}\text{Given A, B, C, D, the striking ship}\\ \text{is actually operating at above critical}\\ \text{speed}\end{array}\right] \tag{13}$$

Each of these probabilities is conditional upon the others, and all are necessary to cause serious collision. Thus, from the multiplication theorem of probability, it follows that

$$P[A, B, C, D, E] = P\left[\begin{array}{l}\text{collision of sufficient magnitude}\\ \text{to breach the SAVANNAH's}\\ \text{reactor compartment occurs}\end{array}\right], \tag{14}$$

where

$$P[A, B, C, D, E] = P[A] \cdot P[B] \cdot P[C] \cdot P[D] \cdot P[E]. \tag{15}$$

The probability that the striking ship is large enough and traveling fast enough to reach the reactor compartment, $P[D]$, was evaluated in Part Two, Chapter IV. The numerical values of the remaining components of the total probability are evaluated below. Because of difficulty in acquiring data, $P[E]$ is evaluated simultaneously with $P[A]$.

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A. PROBABILITY OF A COLLISION OCCURRING

It is proposed to regard the ship's career with respect to collisions as a Bernoullian sequence, which must satisfy these conditions:

1. In each trial the possible results must be described as a "success" or a "failure".
2. The probability of success is the same for each experiment.
3. Each result is independent of the others.

Condition 1 may be satisfied by defining a collision as a "success" in a given trial. The trial can be a single trip or a year's operation. The latter is used here, being somewhat less dependent on run, weather, and sea conditions. In addition, with a passenger ship in world-wide service, trip and years service are nearly the same. A "trip" basis requires detailed data on length and frequency of run, most of which is unavailable.

Because the standards of equipment and personnel aboard ship are maintained at a relatively constant level, Condition 2 is believed satisfied. Regular inspections and surveys prevent radical deterioration of equipment during ship lifetime.

It seems realistic to assume that each collision is an independent event. No cases are known of multiple accidents where one collision causes another. Following each collision, classification societies require resurvey and the same minimum standards of equipment as before. There is no reason to believe that command and operation will deteriorate following a collision. In fact, both will probably improve. Hence it is believed that Condition 3 is satisfied, and the problem can be regarded as a Bernoullian sequence.

Taking p as the probability of a collision in any one trial, and $q = 1-p$ as the probability of not having a collision in any one trial, the probability of having at least one collision in "n" trials is then

$$P[A] = 1-q^n = 1-(1-p)^n. \qquad (16)$$

The number of trials, "n", is the postulated life of the ship, taken at 20 years as is customary.

The probability, p, of collision in any one trial must be evaluated from actual collision data, the principal sources of which are the various underwriting agencies and the U. S. Coast Guard. Unfortunately, no two sources use the same basis in accumulating data so there is little or no consistency between sources.

Lloyd's and Liverpool Underwriters publish lists of total and partial losses from all causes at sea and reference 9 gives similar data over a period of years. This information, some of which is plotted in Figure 52, is valuable in establishing trends but is not too useful in determining probabilities. A complete operating history of a precisely defined and controlled sample is actually required.

Such a history was developed by the United States Salvage Association which, as surveyor for a large group of marine underwriters, is called upon for technical services in all accidents involving ships insured by member firms. Approximately 60-80% of American Flag vessels and 20-40% of all seagoing ships are included in the insured group so considerable information is available on a large and fairly consistent sample of the world's merchant fleet. The U. S. Salvage data include all accidents occurring to seagoing vessels (of more than 5000 tons gross) that met these conditions:[19]

1. Cost of repair exceeds $30,000.
2. Cost of repair to propeller, tailshaft, and rudder, or any combination exceeds $10,000.
3. All cases involving propulsion or auxiliary machinery failures or damage, regardless of cost.

Table XXII was developed from this data. An added restriction is the elimination of all collisions with barges, tugs and non-seagoing vessels. Since it is obvious that no collision of this kind could damage SAVANNAH's reactor compartment their inclusion would needlessly increase the collision frequency. Consistency has been maintained by excluding such collisions from the calculation of P [D] in Part Two, Chapter IV.

The value of p, the probability of collision in any one trial, is calculated separately for each of three areas in which operating conditions are unique: harbors, harbor approaches, and the high seas. Each area probability is then weighted by factors that express the probability of operation at greater than critical speed, β, and the percentage of the ship's life spent in the given area, γ. Thus p is given by the sum of the weighted area probabilities,

$$p = P_H \beta_H \gamma_H + P_A \beta_A \gamma_A + P_S \beta_S \gamma_S, \qquad (17)$$

where the subscripts refer to harbors, approaches, and at-sea conditions.

TABLE XXII

SUMMARY OF COLLISION DATA
(U. S. Salvage Association)

PASSENGER SHIPS

	1953	1954	1955	1956	Average
Number in sample	49	64	61	62	59.0
Number of vessels in collision					
in harbors	0	0	0	0	0
in approaches	2	0	0	4	1.50
at sea	0	0	0	0	0

TANKERS

Number in sample	347	360	427	407	385.3
Number of vessels in collision					
in harbors	11	5	4	7	6.75
in approaches	5	2	6	7	5.00
at sea	1	0	0	0	0.25

CARGO SHIPS

Number in sample	1112	1028	1149	1181	1117.5
Number of vessels in collision					
in harbors	14	3	12	11	10.00
in approaches	11	7	8	12	9.50
at sea	1	0	0	0	0.25

MISCELLANEOUS
(Military, Bulk Carriers Other Than Oil, etc.)

Number in sample	70	77	96	119	90.5
Number of vessels in collision					
in harbors	1	0	3	0	1
in approaches	1	3	3	1	2
at sea	0	0	0	0	0

The three areas of operation are defined below.

1. Harbor — operation within the limits of a sheltered port. The harbor is assumed to extend out to the principal sea mark at the harbor entrance. New York Harbor for example would be bounded by Ambrose Lightship. Harbors are generally identified with dense populations, heavy traffic, and relatively low ship speeds.

2. Harbor Approaches — operation between the principal sea mark at the harbor entrance and the last point 100 miles from the nearest land. The 100-mile limit is predicated on an assumed minimum safe distance for the general population for an uncontained major nuclear accident under inversion conditions. As defined, this area approximates but does not, under all circumstances, parallel the area of merging sea lanes with fairly high traffic densities. The approach area is characterized by essentially full sea speed and heavy traffic, generally well within the 100 mile line.

3. At Sea — operation in areas 100 miles or more from land. Normal full power is assumed, and traffic densities are generally light.

Obviously, these areas will change radically with destination and shipping route. For example, on the North Atlantic run, departure for Europe is usually taken as Nantucket Light, more than 200 miles from Ambrose Light and New York Harbor. Thus about 250 miles or almost 10% of the total voyage lies within 100 miles of land. On the Southern European run, for which departure is taken from Ambrose itself, only the first 100 miles lie in the approach zone. Although these areas can be defined fairly accurately for a ship on a designated run, they can only be estimated for the SAVANNAH whose range is world-wide.

Since ship operating conditions are assumed to be constant within a given area, the probability that the striking ship is actually operating at a speed greater than critical, $P[E]$, is evaluated as β and included in the probability of collision itself. Precise definition of β is very difficult but by assuming the worst conditions a reasonable value can be assigned to each area. At sea, β is taken as unity because merchant ships normally proceed at full power. Though vessels might be expected to reduce speed in approach areas, recent accidents indicate that this is not always done. Hence the probability of operation above critical speed is again taken as unity.

Widely different operating conditions prevail in harbors. Passenger ships, for instance, proceed at very low speeds. The QUEENS, with a critical speed of 11 knots, normally makes less than 10 knots in New York Harbor. Tankers and dry cargo ships do not reduce speed nearly

as much, but their critical speeds are very close to their design speeds and a slight reduction will drop the speed below the critical. Hence β will approach zero in harbors.

The percentage of the ship's lifetime spent in each operating area is again a function of service and route; lacking exact information on the SAVANNAH's itinerary, a value for γ is difficult to obtain. A fair estimate of port time can be obtained from the well established "use factor", which defines the percentage of time the ship operates at sea. The accepted factor for passenger ships is about 0.75, indicating that a ship spends 25% of its lifetime in port. Since in its initial voyage the SAVANNAH will probably spend even a greater percentage of time in port, 25% is considered conservative. The use factor is 0.85 for tankers, 0.65 for cargo ships.

The time spent in approaches varies sharply with route and service. An investigation of several trade routes indicates that with a few exceptions (most of which are tankers) American flag vessels spend up to about 10% of their life in waters defined as approaches. The special nature of the SAVANNAH's intended service could easily increase this time, but lacking precise information, the commercial service requirements are assumed to govern. Therefore a value of 0.10 is assigned to γ_A, and the remaining 65% is considered as sea time.

The coefficients of the area's probabilities are then:

$\beta_H \longrightarrow 0$ $\qquad$ $\gamma_H = 0.25$

$\beta_A = 1.00$ $\qquad$ $\gamma_A = 0.10$

$\beta_S = 1.00$ $\qquad$ $\gamma_S = 0.65$

The probabilities for passenger ships are calculated from the data on Table XXII. The passenger ship data are limited and a better value undoubtedly could be obtained from all types of ships, but the difference in service requirements, particularly the need for passenger ships to maintain a schedule, justifies separate treatment. Since passenger ships seem to be involved in proportionately more collisions, this treatment further leads to a more conservative result. The area probabilities are

N = total number of ships operating, averaged over four years,

n_x = total number of ships involved in collision in area x, averaged over four years, and

$P_x = \frac{n_x}{N}$ = probability of collision in area x.

Details of the calculation for passenger ships, tankers and cargo vessels are given in Table XXIII which shows that the total probability of collision is considerably higher for passenger ships than for the others. It is believed this reflects the need to maintain schedule as mentioned previously.

Before considering the absolute accuracy of the area probabilities, several conclusions can be drawn from Table XXIII. First, the probability of a harbor collision in which a ship is operating at greater than critical speed is negligible for the classes considered. In particular, with both β and γ approaching zero, a passenger ship collision of this magnitude in a harbor appears practically incredible. Collisions involving ships at sea are also relatively rare. The approach area is by far the most dangerous for all types of vessels.

It should be emphasized that of the three factors evaluated in Table IX, the probability of collision, p, is a characteristic of the area; the length of time in the area, γ, is a characteristic of the particular ship under discussion; and the probability of the collision occurring at speed above critical, β, is a characteristic of the total merchant fleet.

The probability of a collision with a ship of greater than critical speed, in any one trial, given by Equation (17), is then equal to 0.00254 for the SAVANNAH. The probability of collision with a ship at greater than critical speed over the assumed life of the SAVANNAH is then given by Equation (16)

$$P[A] \cdot P[E] = 1-(' p)^n = 0.04955.$$

TABLE XXIII

CALCULATION OF AREA PROBABILITIES
(Averaged Over Four Years)

	Passenger	Tanker	Cargo
Number in sample (1)	59.0	385.3	1117.5
Number in collision			
in harbors (2)	0	6.75	10.00
in approaches (3)	1.50	5.00	9.50
at sea (4)	0	0.25	0.25
In Harbors			
$P_H = (2)/(1)$	0	0.0175	0.00895
B_H	→ 0	0	0
γ_H	0.25	0.15	0.35
$P_H \beta_H \gamma_H$	0	0	0
In Approaches			
$P_A = (3)/(1)$	0.0254	0.0130	0.00852
β_A	1.00	1.00	1.00
γ_A	0.10	0.10	0.10
$P_A \beta_A \gamma_A$	0.00254	0.00130	0.000852
At Sea			
$P_S = (4)/(1)$	0	0.00065	0.00022
β_S	1.00	1.00	1.00
γ_S	0.65	0.75	0.45
$P_S \beta_S \gamma_S$	0	0.00049	0.000099
$\Sigma P \beta \gamma$	0.00254	0.00179	0.000941

B. PROBABILITY OF SAVANNAH BEING THE STRUCK VESSEL

The SAVANNAH, being a passenger ship as well as the first nuclear powered merchant ship, will undoubtedly be operated more carefully than most ships. Hence in any accident she might be expected to be the vessel struck. However, in ship collisions the striking vessel often is not at fault or at any rate cannot be proved at fault. In fact, assignment of blame for any ship collision is never an easy task. Therefore, regardless of careful operation the probability that SAVANNAH is the struck vessel is taken as 0.50.

C. PROBABILITY OF COLLISION OCCURRING IN WAY OF THE REACTOR SPACE

"In way of the reactor space" refers to a damaged length on the ship equal to the length of the reactor space plus a length on either end equal to one-half the maximum beam of a possible striking ship. Logically, the probability density function over the ship's length would be constant, and the probability of collision in way of the reactor space would be simply the damaged length divided by the total length of the ship. However a survey of accident data indicates that this is not the case.

The density function of the location of collisions along the ship's length derived from 60 major collisions investigated by the U. S. Coast Guard is shown in Figure 53. The forward peak apparently stems from the relative ease of avoiding impending collisions aft of midships. The assumed beam of the striking vessel, 100 ft, is representative of very large vessels. The damaged length of the SAVANNAH is then the portion of the ship between a point (a), 50 ft forward of the forward reactor compartment bulkhead, and a point (b), 50 ft aft of the reactor compartment bulkhead. The probability of collision in this region is given by

$$P[C] \frac{\int_a^b N(l)\, dl}{\int_L N(l)\, dl} = 0.439$$

D. PROBABILITY OF COLLISION BREACHING THE REACTOR COMPARTMENT

Having evaluated the component probabilities, the total probability of a collision breaching the reactor compartment can be defined. From the component probabilities

$$P[A] \cdot P[E] = 4.955 \times 10^{-2},$$
$$P[B] = 5.0 \times 10^{-1},$$
$$P[C] = 4.39 \times 10^{-1},$$
$$P[D] = 7.00 \times 10^{-3},$$

and from Equation (15),

$$P[A, B, C, D, E] = 7.56 \times 10^{-5}.$$

Thus the probability of a breach in collision is about 7 in 100,000.

E. ACCURACY

The validity of this figure is, like the results of most probability studies, open to question. The accuracy of P [D] was discussed in Part Two, Chapter IV. The probability of striking in way of the reactor compartment is fixed by the SAVANNAH's physical characteristics and by the distribution given in Figure 53. Since the distribution is based on the best established of the available data, P [C] is believed valid. In any event neither P [C] nor P [B] should vary significantly if a new statistical sample were chosen.

The probability of collision is dependent upon rather limited statistical data, and the probability that the striking ship is operating at a speed above critical is almost without basis. In the latter case, the assigning of a value of unity to β_A and β_S is certainly the most conservative procedure. The statement that β_H approaches zero in harbors is based on studies of harbor speeds and the knowledge that a very slight decrease in speed eliminates many ships from the dangerous class.

In harbors there is a further factor of safety in the usual segregation of tankers from other ships. Normally the tanker docks are located in outlying areas far from the passenger and cargo docks and hence tankers are not likely to be operating in the vicinity of passenger ships within the harbor. Since the statistical data show no passenger ship collisions,

a finding that seems well verified by the care with which such vessels operate in harbors, the position that a collision involving rupture of SAVANNAH's reactor compartment in harbors is incredible seems fully justified.

Designation of the approach as the principal area of danger is quite reasonable. The absolute value of the probability of collision in that area is not as easy to establish, but it is generally considered highly conservative. In addition to the conservativeness of P [D], the assumption of full sea speed in the approach area becomes less and less representative as the harbor entrance is approached.

As to the accuracy of the calculated components, the over-all probability may be subject to error, but the conservative absolute value is actually much less than 7.56×10^{-5}, probably somewhere between 10^{-4} and 10^{-5}.

The most important intangible factor is the SAVANNAH's position as the first nuclear powered merchant vessel. Accordingly, great care and good judgment will be exercised in her operation, which will tend to minimize the possibility of collision or other casualty.

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FIG. 52: COLLISION FREQUENCY IN WORLD'S FLEET

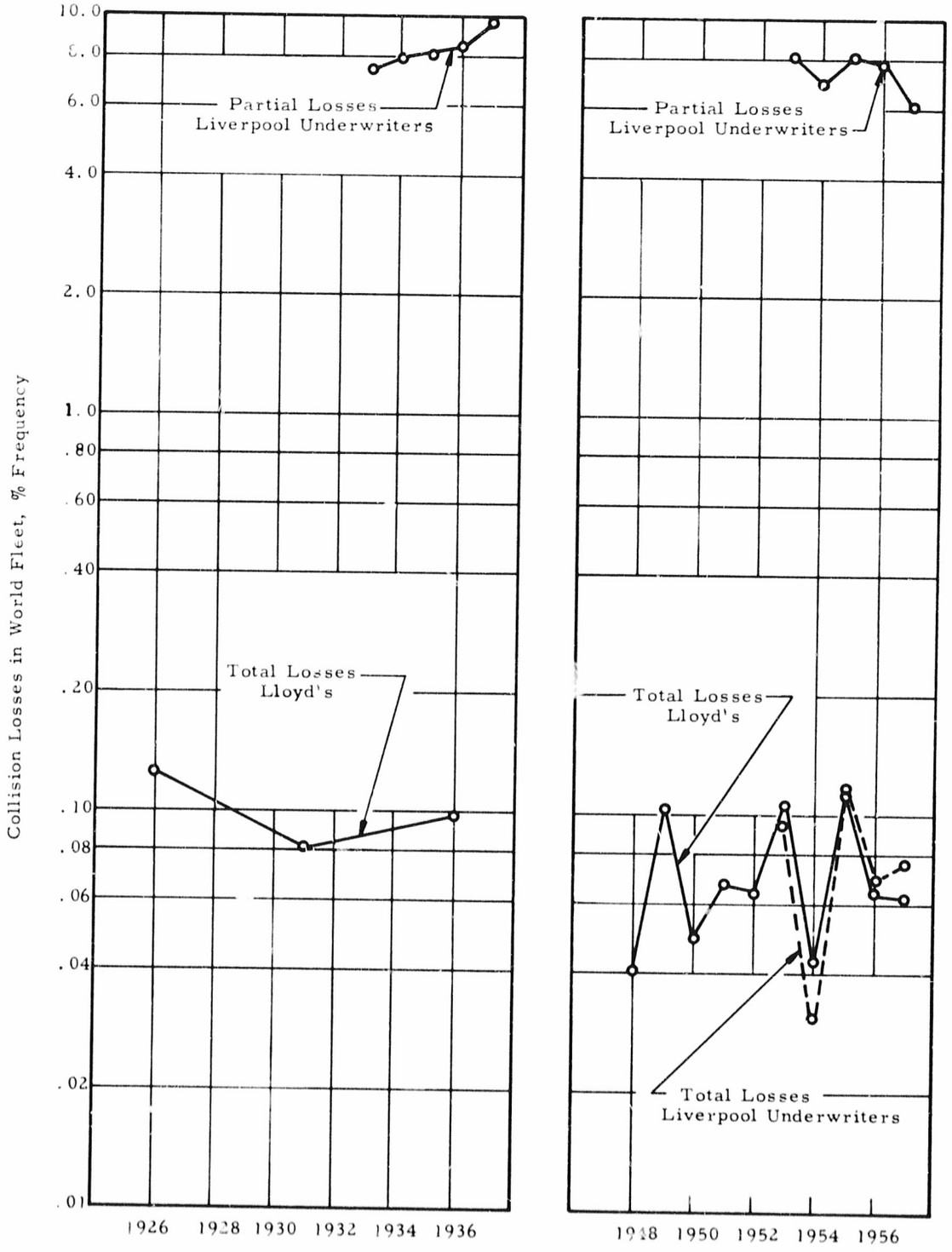

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FIGURE 52

FIG. 53: LOCATION OF COLLISION ALONG SHIP'S LENGTH

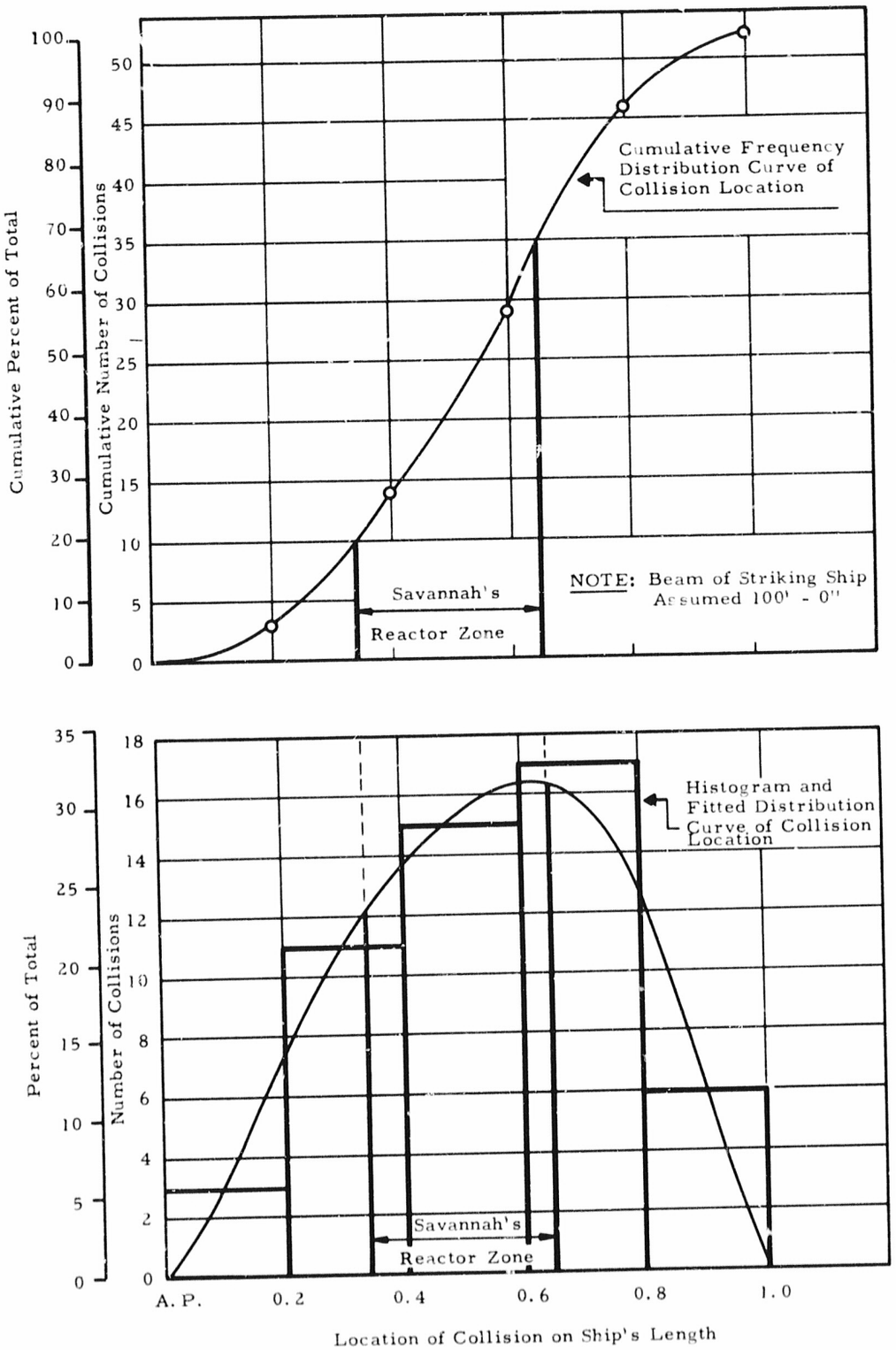

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FIGURE 53

Chapter VI

GROUNDING

A. Effects of Grounding
B. Grounding in Channels and Harbors
C. Grounding in Open Water Areas
D. Breakup

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VI. GROUNDING

The effects of grounding are similar to but of a more local nature than the effects of collision. Shock damage, penetration of shell, distortion of double bottom structure, penetration of inner bottom plating, and rupture of containment are all conceivable. All of these effects are surveyed here to establish that SAVANNAH's construction will maintain containment integrity in a grounding accident.

Surprisingly, grounding is less common than collision. As with collision, a distinct difference in behavior exists between various types of ships, passenger and passenger-cargo ships being involved in fewer groundings than tankers and cargo ships. A partial explanation may be that passenger ships usually visit only well charted ports, and at fairly regular intervals. Thus ship officers are usually well aware of the hazards associated with any approach to the ground.

Grounding accidents, like collisions, are definitely a function of the area in which they occur. Most groundings occur at low speed in channels and harbors. In the worst case, grounding occurs as a result of loss of power or steerage on a rocky lee shore and is accompanied by partial or total breakup of the ship. However, accidents of this kind rarely occur.

Another very important factor in grounding is the condition of the sea. Excluding high-speed grounding on rocks, most of the structural damage is caused by the vessel working under the action of the sea after initial contact with the ground. In sheltered waters, therefore, even a severe grounding on a hard bottom may not affect reactor safety, but relatively slight contact in heavy weather may eventually break up the ship. However, it takes a long period of working for severe damage to result.

In most cases of grounding, the emergency diesel generator and cooling water will be available for decay heat removal. Therefore,

there is little danger that fission products from core meltdown will be released to the containment interior. This considerably reduces salvage problems.

A. EFFECTS OF GROUNDING

1. Shock

Grounding shock is invariably less than that encountered in maximum roll, pitch, and heave. As with collision, extensive crushing of ship's structure may be experienced. Only occasionally have personnel reported severe jolts in a grounding accident. The most dangerous shock results from a high speed collision with a rock or underwater object. The collision of the battleships WASHINGTON and INDIANA could be considered a case in point. The bow of the WASHINGTON, of much heavier construction than a merchant ship, was crushed to the degree that no shock was felt on the bridge. Such a collision is practically equivalent to striking a sea wall.

The case of a ship without power drifting on the rocks in a rough sea is considerably less dangerous than a bow-on grounding since the ship could drift on at any angle and the bow and stern structures are the hardest parts of the ship. Hence shock effects in groundings are not considered important in the SAVANNAH's design.

2. Penetration Effects

Structural damage in grounding is usually limited to the shell and double bottom; although holing of the inner bottom is not uncommon, very little structural damage results above it. In way of the SAVANNAH's reactor is a 5-ft-deep double bottom with a heavy egg-crate construction and massive containment foundation, which offers adequate protection for any conceivable grounding accident in that area of the ship.

Rupture of the containment, very unlikely in view of the degree of protection, could only occur if the ship were pivoted on rock in the vicinity of the reactor compartment, grinding back and forth for a long while. In addition to the low probability of the ship grounding in such a position, the length of time required to break up the containment will permit its salvage or removal of the ship to deep water. (See Part Two, Chapter X.)

The integrity of radioactive sources outside the containment shell (e. g., waste storage tank, demineralizers) could be jeopardized in grounding. Although outside the containment, these sources are well protected by the concrete shielding, the reactor compartment boundaries, and the heavy inner bottom structure. Low level waste, which may be stored in the inner bottom, is monitored for activity level prior to diversion to the inner bottom, and, if required, is passed through a demineralizer before storage to reduce the activity level. The demineralizer and other plant components can be isolated from the inner bottom tanks by closing valves.

B. GROUNDING IN CHANNELS AND HARBORS

Most groundings occur in harbors and harbor approaches — areas characterized by shallow water, lack of maneuvering room, and high traffic density. Slight errors in navigation, loss of power or steering, misjudgment of traffic situations, shifting channels, etc. can cause a ship to run aground.

The most important characteristic of harbors and channels is sheltered water. Because of this, a ship grounded in a harbor generally will not work to a significant extent. Therefore, resulting structural damage is limited. Since most groundings are bow-on or stern-on, the SAVANNAH's reactor compartment would be subject to little or no harm. One exception is a sideways slip, which can occur if a turn in a narrow channel is misjudged. This situation is fairly common in river channels, but the soft mud bottom and sheltered water prevents major damage. Another grounding situation involving the midships area has occurred occasionally in small, unsheltered ports on the west coast of South America. The combination of harbor swell and shallow water can actually drop a moored ship to the bottom of the slip, causing extensive denting of plates and twisting of floors. However, no penetration of shell or inner bottom has ever been reported.

Certain "freak groundings" involving high speed contact with rocks and reefs have been reported in harbors and approaches. Although structural damage may be extensive, the SAVANNAH's containment is believed to be adequately protected. Again, sheltered waters limit damage to the initial contact. In most of the world's harbors and approaches the location of rocks and reefs is well known and clearly marked; thus, accidents due to these obstructions generally suggest improper ship-handling. The SAVANNAH's greatest protection against such accidents lies in the ability and care with which she will be operated.

Another very rare type of grounding occurs when the ship lies over on its side after contact with the ground. This also occurs occasionally in shallow-water-sinking as in the case of the NORMANDIE. The SAVANNAH's primary system and vital components are thoroughly braced to prevent major dislocation in such an incident. Dislocation of

the reactor itself is prevented by cross braces on the containment shell. The boiler drums and the pressurizer are equipped with snubbers to provide support from the containment at a ship attitude of 90 degrees. The primary loop piping is cross-braced within itself and to the containment. When the ship is lying on its side, substantial deflection would occur only in the primary shield tank which would rest upon the reactor vessel. There may be leakage from the tank, but no substantial structural failure. Some local yielding, but no undue stresses, may be expected in the containment vessel and component foundations. Since both containment and primary system remain intact, the attitude of the ship after grounding (or sinking) does not constitute a safety problem.

Under storm conditions, some harbors are not "sheltered". In this case, structural damage may more nearly approximate that of the open water groundings discussed later . However, working of the ship during the limited period of heavy weather should not cause major damage.

C. GROUNDING IN OPEN WATER AREAS

Working can be presumed to cause considerable structural damage to the ship grounded in open water. The sea can also impede or delay efforts to get the ship off the ground and, as discussed in Part Two, Chapter X, make the salvage or immobilization problem much more difficult. However, open-sea grounding of ocean going vessels is very rare, particularly with passenger ships. The principal causes of open water grounding are the loss of power or steering, navigational error, or uncharted hazards.

Loss of power or steering, as discussed in detail in Part Two, Chapter II, occurs infrequently and almost never in conjunction with the ship's position off a lee shore. If the shore upon which the ship is drifting does not shoal too steeply, the anchors quite often can hold the ship off the ground until a tug is obtained.

Navigational errors probably cause most open water groundings, particularly those occurring on isolated reefs and rocks. On most of the world's sea lanes, reefs and rocks are well charted. Passenger ships are practically never involved in accidents of this kind; further, the failure or misuse of navigational aids, the cause of some spectacular groundings, has been practically eliminated in recent years.

Navigational errors probably cause most of the groundings involving uncharted hazards — principal among which is shifting sand bars. In open water, sand bars are not too common, and passenger ships generally avoid such areas by as wide a margin as possible.

D. BREAKUP

Under the worst conditions the SAVANNAH might ground on an inaccessible shore where the action of the sea and rock could combine to work the ship continually without freeing her. Eventual breakup of the ship can be expected in such a case. However, this would probably require considerable time, allowing opportunity for appropriate remedial action.

If grounded under the conditions described, the ship would pivot under the action of the waves and eventually break at the point of lowest strength on her hull girder. The massive inner bottom structures amidships, which gradually taper off in the holds forward and aft, will cause such a break to occur well forward or aft of the reactor space. Since the containment is connected to the ship's structure at only one point, it will not be affected by the wracking of the hull girder. Therefore, the break will not damage the containment.

Once the ship is broken, the water-borne buoyant segment may break free and sink or re-ground, and the pivoting process will be repeated. Essential to a second breaking, however, are a fairly long lever arm and another discontinuity in the remaining girder. However, the structure in way of the reactor has no discontinuity and is the strongest part of the ship, so again the break will take place elsewhere. Eventually the midships portion of the ship will be essentially free to sink or re-ground. Since the short section of uniform strength has no buoyancy it will have little tendency to move under wave action and thus will not grind on the ground. Should the containment itself break clear of the ship's hull the same situation would prevail with even less likelihood of motion.

After the final stage of breakup is reached, when all sections of the ship have sunk, years may pass before the action of the water produces any effect. Therefore further immobilization of the containment is possible. For instance the shell can be filled with concrete ($\approx$ 1200 tons) and the entire reactor system will be completely contained against any release of radioactivity.

In conclusion, no hazard to the environment is anticipated from even the most serious grounding.

Chapter VII

HEAVY WEATHER

A. Motions of Ships in a Seaway

B. SAVANNAH Design Criteria

C. Storm Damage

VII. HEAVY WEATHER

The most familiar aspect of ship operation is the motion of the ship in a seaway. Unlike shore stations, a shipboard nuclear power plant must be designed to withstand the static and dynamic loadings imposed by rolling, pitching, and heaving of the ship itself. Further, since wracking of the hull beam under these motions may open seams and lead to flooding, the design of the hull girder is also affected.

As shown in Part Two, Chapters III and IV, the accelerations imposed by the abnormal forces of collision and grounding were always less than those experienced in ship motion in a seaway. Hence the governing criteria for dynamic loads on vital components are those imposed by the ship's motion. These motions are briefly described here, and the design criteria of the SAVANNAH are established.

A. MOTIONS OF SHIPS IN A SEAWAY

The motions of a ship in a seaway can be resolved into six components of rotation and translation, all of which are completely interdependent but not necessarily additive. The rotations are rolling, pitching, and yaw; the translations are fore-and-aft drift, lateral drift, and heave.

Yaw is the rotation of the ship about the vertical axis. Together with lateral and fore-and-aft drift, yaw is characterized by low velocities and negligible accelerations. These motions thus do not affect nuclear safety and can be neglected.

Heave is the movement of the ship parallel to the vertical axis. When a ship lies with its ends on successive crests of a wave, the finer form of the ends causes the ship to sink deeper than she would in still water. As the wave crest moves along the ship, the fuller midship sections cause the ship to rise above the still-water condition.

The two remaining motions are roll — the rotation of the vessel about the longitudinal axis, and pitch — the rotation about the transverse

axis. Both motions impose a static load on components from the inclined position and the dynamic forces of the ship's motion. In roll, the static load is very much higher than the dynamic, but in pitch the opposite is usually true.

Maximum values of roll, pitch, and heave accelerations are difficult to obtain since these motions are highly dependent upon sea conditions. Recently, however, considerable experimental data have been amassed on several types of vessels under actual sea conditions. The accelerations and amplitudes of roll, pitch, and heave — measured over long periods of time — can, within limits, be fitted with a log-normal distribution over wide ranges of sea and operating conditions and with a Rayleigh distribution for given states of sea, ship speed, and course.[10] From these distributions, the probability of exceeding given values of accelerations and roll and pitch amplitudes can be defined for a particular ship.

A limited amount of data is available on the behavior of vessels similar to the SAVANNAH. Heave and pitch accelerations and roll and pitch amplitudes for a Mariner class vessel were carefully measured during a North Atlantic crossing.[11] The tests were run at various courses and headings in a well developed state-5 sea. The state of sea code is taken from the U. S. Navy Hydrographic Office, which defines a state-5 sea as very rough.[12] During the tests, wind velocities were Force 6-7 Beaufort (strong breeze or moderate gale) and most probable maximum wave height was about 15 ft. Seas of not more than state-5 are encountered in the North Atlantic about 80% of the time.

Figures 54 through 57 show the probability of exceeding given values of heave and pitch accelerations and roll and pitch angles for various courses and speed.[11] In Figures 56 and 57 the roll and pitch angles are given as the "double amplitude"; i.e., the roll angle from the vertical is half the roll angle given. All other roll angles quoted here are measured from the vertical.

From these curves, it is evident that accelerations and amplitudes are extremely low even in a very rough sea. Tests in seas greater than state-5 are limited to spot observations, primarily because such seas occur only rarely and are of relatively short duration. Thus, extremes

of roll may occur at unpredictable intervals and are seldom logged in the operation of merchant ships.

The maximum well-established roll angle reported occurred on the U.S.S. COMPASS ISLAND, a Mariner class ship. In a storm off Delaware, with wind velocities of 55-63 knots (Force 11 Beaufort, just short of a hurricane) on her quarter, the COMPASS ISLAND lost all power and reported rolls of 35-40 (single amplitude) degrees. Using the established methods of estimating ocean waves from an assumed duration and fetch, the significant wave heights were estimated to be on the order of 28 ft with a maximum of about 35 ft.[13] This is an extremely rough sea, probably approaching the limits to be encountered except in hurricanes and typhoons and in the great storm areas of the Drake Strait and the Southern Ocean. The COMPASS ISLAND normally operates at light draft and is thus more prone to roll than a deep-loaded vessel.

Of particular interest in the COMPASS ISLAND incident was the performance of her stabilizers. Prior to loss of power and way, the stabilizers held maximum rolls to less than two degrees. Although such a radical reduction in roll amplitudes cannot be expected under all conditions, the stabilizers are evidently very effective in damping roll.

Further information on several types of ships tends to confirm the foregoing.[10] In general the following conclusions can be drawn:

1. Under most conditions roll amplitudes will be significantly less than 30 degrees. Extreme rolls will be quite uncommon, not exceeding 35 to 40 degrees in the light condition. In the loaded condition, extreme rolls probably will not exceed 30 degrees.
2. Pitch amplitudes and accelerations are extremely low at the reactor proper.
3. Under most conditions the greatest dynamic loadings will be obtained in heave.

Referring again to Figures 54 through 57, changes in course and speed produce striking effects. Excluding roll, turning and running with the sea results in marked reductions in pitch and heave accelerations and pitch angles. In head seas, decreases in speed also cause significant reduction. Thus the behavior of the ship in storm seas depends to a great extent on course and speed.

B. SAVANNAH DESIGN CRITERIA

The data in Part Two, Chapter VII, A, do not, of course, establish absolute magnitudes of maximum roll, pitch, and heave accelerations. Further tests are now under way but, again, the infrequent and unpredictable occurrences of storm seas require long periods of time at sea and semi-permanent instrumentation. Since few merchant ships are so equipped, accumulation of sufficient data for an adequate statistical sample is a difficult, time consuming process. Although several series of long term tests are now under way, it is impossible to define precisely maximum accelerations and amplitudes of roll, pitch, and heave.

In the design of conventional merchant ships, very little consideration is given to maximum ship motions. The practice for many years has been to design to reasonable values with relatively high safety factors; experience has borne out the validity of this approach.

The motions recorded in Part Two, Chapter VII, A, although not maximum values, are still representative of motions that would occur in very rough seas and are unlikely to be exceeded. The maximum values are definitely well within a factor of two greater than those recorded. From considerations of available freeboard, the roll amplitudes are somewhat limited by the submergence of the deck edge. In the loaded condition in smooth water, the SAVANNAH's deck edge will go under with a roll of about 27 degrees. In the light condition the amplitude for deck edge submergence is a little more than 35 degrees. This is not to say that these amplitudes cannot be exceeded. The deck edge can and probably will go under occasionally, and in certain seas these amplitudes can be exceeded without submerging the deck edge. However the ship will not roll very much beyond these values, which can be considered a limit for practical purposes.

Since the SAVANNAH is a passenger as well as a cargo vessel, passenger comfort is an important consideration. Research on passenger comfort has indicated that slight accelerations have a marked effect on sea sickness. Apparently, discomfort begins at about 0.1 g and passenger ranks begin to thin when accelerations reach 0.2 g. The marked reductions in accelerations achieved by alterations in course or speed are well known to ship's officers, and passenger ships generally handle

the problems associated with very rough seas by one or both of these methods. Hence, if the SAVANNAH is subjected to extremes of roll, pitch, and heave, her officers will undoubtedly initiate remedial measures.

The following design criteria were therefore established for the motions of the SAVANNAH in a seaway:

Roll

Axis of roll, 20 and 30 ft above the base line for light and full-load conditions respectively.

Period of roll, 14 and 23 seconds for light and full-load conditions respectively.

Amplitude of roll from upright, 30 degrees.

These values closely approach reasonable maximum conditions, e.g., at the control rod drives the maximum loadings (static plus dynamic) are about 0.6 g.

Pitch and Heave

Combined heave and pitch accelerations of 0.25 to 0.30 g at the reactor.

Pitch amplitude of 7 degrees.

Since the reactor compartment, containing most of the ship's vital components, is located almost on the center of pitch, pitch accelerations are negligible. At the control rod drives, pitch accelerations of 7 degrees are less than 0.05 g. Heave and pitch are additive under certain conditions so they are lumped together. Although not maximum the values are again in the high range.

The SAVANNAH's vital components were designed to withstand accelerations of at least 1 g (and in most cases 2 g) and are therefore fully adequate to withstand maximum sea conditions.

Since abnormal shock loadings in collision and grounding are much less severe than those of maximum roll, pitch, and heave, it is concluded that neither these nor the normal motions of the SAVANNAH in a seaway will cause damage to machinery.

C. STORM DAMAGE

The principal source of danger in heavy weather is the working of a ship as a beam in the sea. In addition, the forces of wind and sea can cause considerable damage to equipment during rough weather. As a result, seams may be opened in the hull, spaces may be flooded, and the ship may sink. These conditions, of little direct interest to nuclear safety, are covered in Part Two, Chapters II and IX.

FIG. 54: HEAVE ACCELERATION PROBABILITIES FOR A MARINER CLASS VESSEL ON A NORTH ATLANTIC RUN

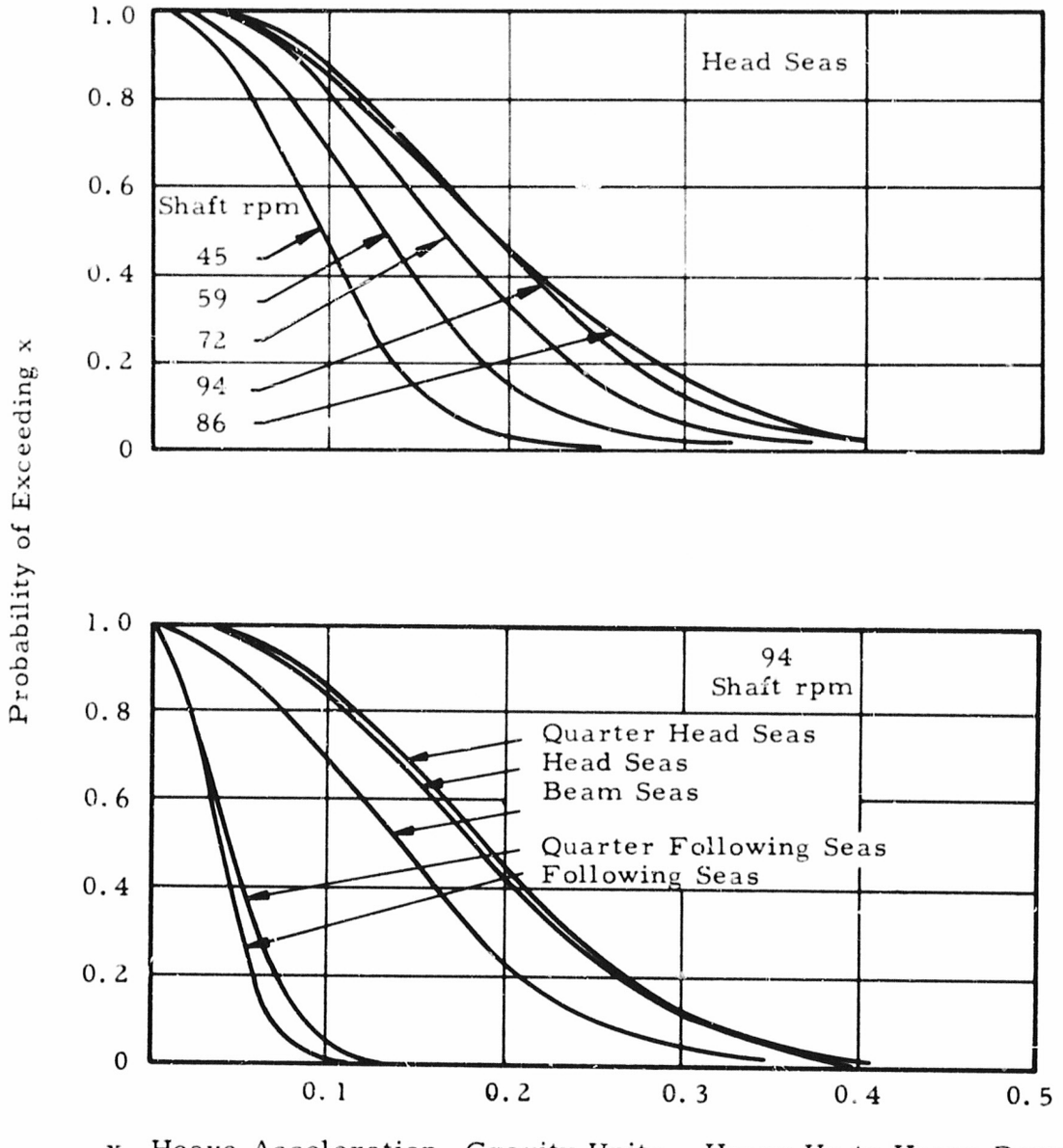

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FIGURE 54

FIG. 55: PITCH ACCELERATION PROBABILITY FOR A MARINER CLASS VESSEL ON A NORTH ATLANTIC RUN

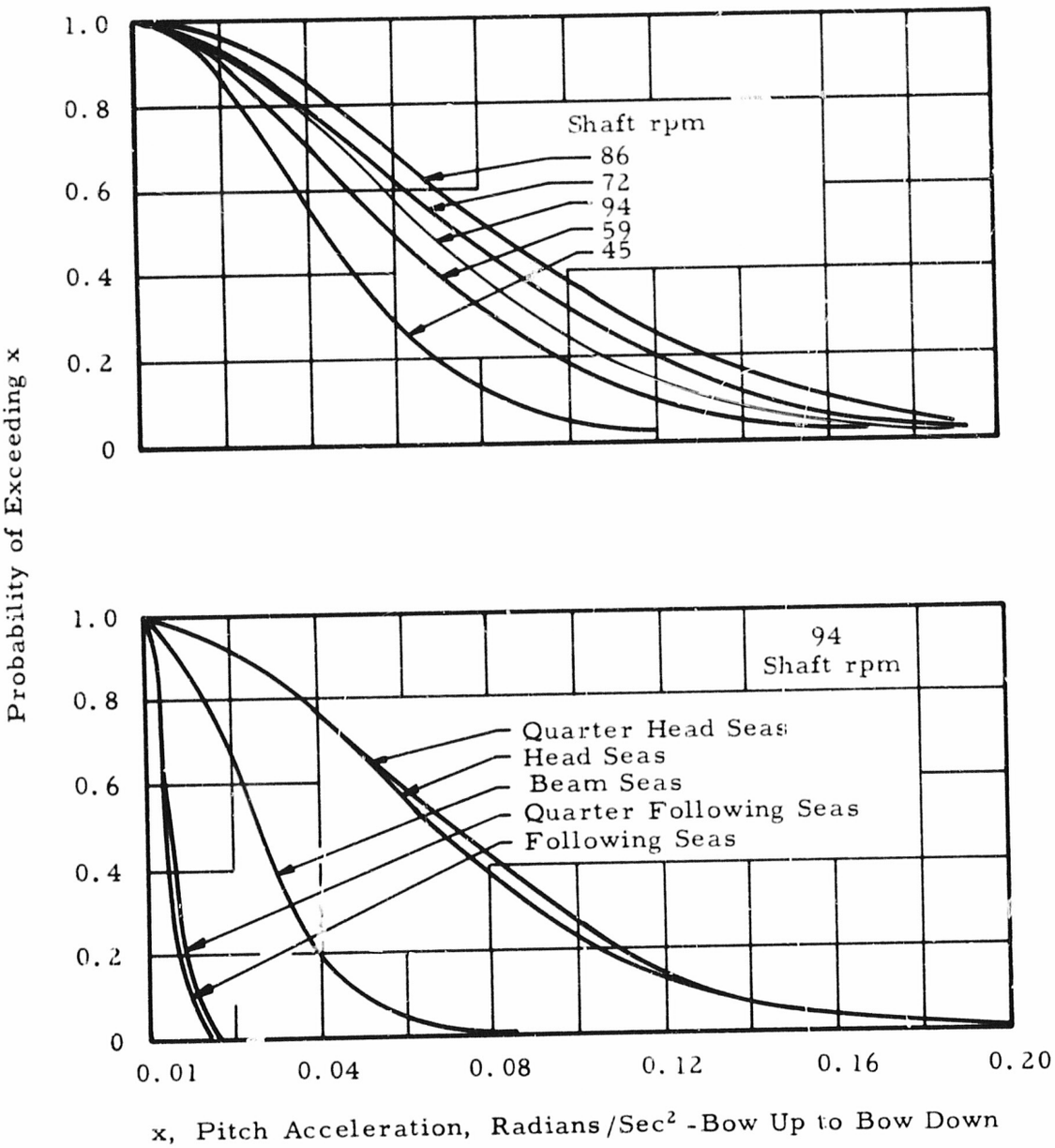

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FIGURE 55

FIG. 56: PITCH ANGLE PROBABILITIES FOR A MARINER CLASS VESSEL ON A NORTH ATLANTIC RUN

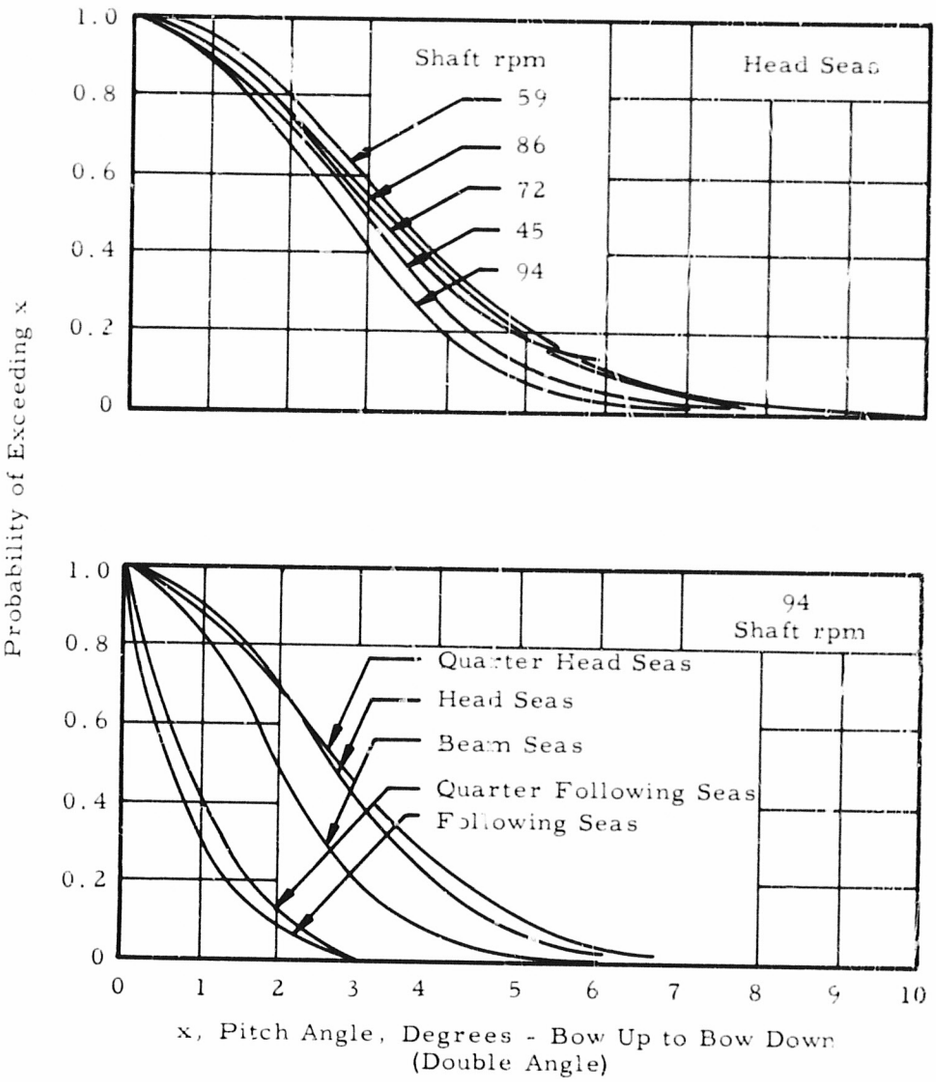

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FIGURE 56

FIG. 57: ROLL ANGLE PROBABILITIES FOR A MARINER CLASS VESSEL ON A NORTH ATLANTIC RUN

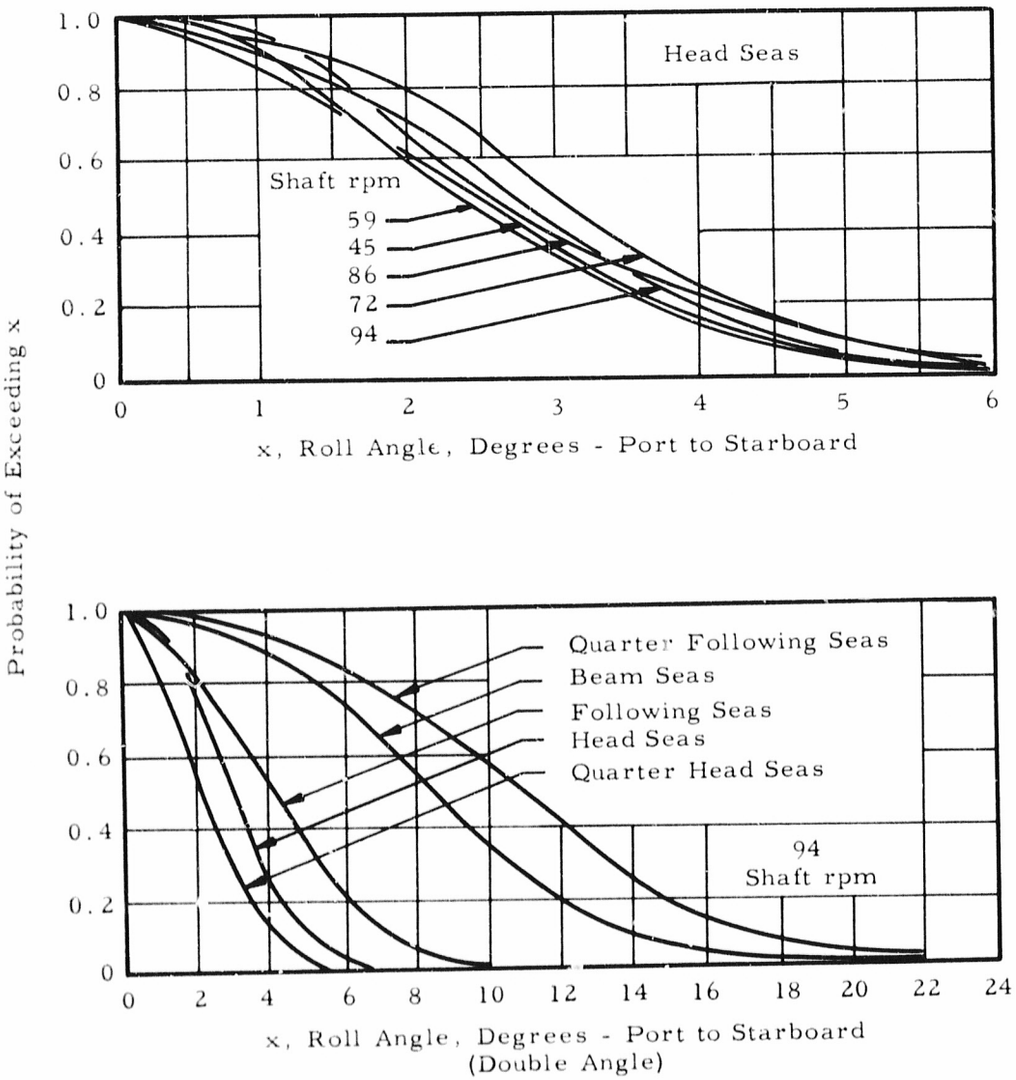

FIGURE 57

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Chapter VIII

FIRE AND EXPLOSION

A. Fire Aboard the SAVANNAH

B. Fire Detection Systems

C. Fire Extinguishing Systems

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VIII. FIRE AND EXPLOSION

The SAVANNAH is designed for maximum fire protection as specified by the U. S. Coast Guard, American Bureau of Shipping, and other regulatory agencies. The fire detection and extinguishing systems are believed to provide fully adequate protection from the maximum fire hazard.

A fire at sea would threaten reactor safety only if the ship or its cargo were the source. Fire resulting from an external source, say a collision with a vessel transporting combustibles, could be controlled with standard ship fire-fighting equipment. In this case, any missiles from an explosion aboard the second vessel would be obstructed by SAVANNAH's structure and concrete shielding, thus preventing penetration of the containment shell. A "double accident" of this kind then does not appear to be of concern to the SAVANNAH's nuclear safety.

The hazard of internal explosion is of little concern since the conveyance of explosives and other dangerous cargo is forbidden on passenger ships. Although some potentially dangerous liquids might be carried, the limited quantity involved would present more of a fire hazard than an explosion hazard. The hydrogen addition system is, of course, an exception. However this system's storage tanks are located outside on "A" deck where the effects of an explosion would be minimized. In addition, the storage tanks could be easily jettisoned if necessary. The system's piping — all protected by heavy structure — is not likely to fail. All lines are completely enclosed, and the surrounding annulus is monitored for escaping hydrogen. The lines are run in well-ventilated spaces to minimize the possibility of explosion further in event of a leak.

A. FIRE ABOARD THE SAVANNAH

As it affects reactor safety, the principal danger of fire at sea is loss of power; also of concern is loss of shielding and containment.

The reactor and power plant would be most prone to damage should a fire reach the areas adjacent to the reactor and machinery spaces. Most likely to occur in machinery spaces are electrical and oil fires, the latter originating from combustibles such as rags, paint, cleaning compounds, and engine and lubricating oils. With the elimination of main propulsion boilers and fuel oil systems on a nuclear powered ship, the major source of oil fires is eliminated. The storage of diesel fuel for the take-home and hotel load boilers is by no means as hazardous as the combustion systems of a conventional ship.

If the main switchboard is damaged, the emergency unit is energized automatically, and the emergency diesel generator is started. Throwover of the bus transfer equipment occurs when the generator reaches the required voltage. Although loss of both main and emergency switchboards within a short time is very unlikely, it would lead to the loss of power incidents previously described. Should fire render the machinery space untenable, the emergency diesel generator capacity suffices to supply power to the low-speed windings of the coolant pumps, the emergency cooling system, and the third fire pump in the shaft alley. A fire in the machinery space probably wouldn't damage shielding since the concrete, lead, and polyethylene are well removed from the machinery space bulkhead.

Fire in the storerooms or stabilizer rooms outside the reactor space would produce the same effects as fire in the engine room, except that no vital machinery would be involved.

Fire in the reactor compartment outboard of the containment shell is unlikely because the area contains very little combustible material. However, should a fire occur, part of the secondary shielding might be destroyed. Particularly vulnerable is the polyethylene, which could melt and flow down the outside of the containment vessel. Although polyethylene is nontoxic, the partial loss of secondary shielding might necessitate a power reduction. Lead melts at about 620°F and loses much of its structural strength at significantly lower temperatures. However, the reactor undoubtedly would be shut down under such circumstances so the condition of secondary shielding is not critical.

A similar problem would be encountered if fire broke out in the hold forward of the reactor space. Here the hold bulkhead directly supports l ..d and polyethylene shielding, which might be damaged. With the lead bonded to the bulkhead a severe fire in the hold might destroy the bond. However, loss of shielding in such an unmanned area is not critical in itself and the plant could still function with limited access in Hold No. 4.

Fire within the containment shell is considered unlikely owing to the closed-circuit ventilation system and the small amount of combustible material present. As to electrical fires, they would not spread after power to the particular unit or cable was turned off.

B. FIRE DETECTION SYSTEMS

There are smoke detection systems in all cargo holds, hatch trunks and cargo tween-deck spaces, paint and lamp room, boatswain's stores, carpenter shop, baggage rooms, and similar spaces generally inaccessible during a voyage. Audible, visible, and olfactory smoke detectors are provided in the chart room. A fan continuously pulls air from each hold in turn — and in the presence of smoke, a photocell activates the audible and visible alarms.

Also provided is an electrical fire alarm system which utilizes fully enclosed thermostats to register visual and audible signals on an annunciator panel in the wheel house. The system is divided into various zones and the alarm indicates the zone in which the fire occurs.

In addition there is a manual detection and alarm system consisting of "break glass" alarm boxes. A watchman's mechanical clock system provides a positive check on the fire watch patrol.

C. FIRE EXTINGUISHING SYSTEMS

1. Water

Two motor-driven, 400-gpm centrifugal fire pumps are located in the machinery space; a third, in the shaft alley is connected to the emergency diesel generator. The machinery-space pumps take suction from a common sea chest via individual pump suction gate valves and discharge through a stop valve to the engine room fire main which services the following:

1. one fire main riser leading to the deck fire main system,
2. machinery space fire station, port side,
3. machinery space fire station, starboard side, and
4. sanitary and swimming pool filling station.

The third fire pump takes suction from its own fire sea chest and discharges to an individual riser connected to the deck fire main. This pump is connected to the emergency generator, ensuring that power and salt water are available for fire fighting as long as the ship is afloat.

Hydrants and hose points at fire stations are located as indicated on the Fire Control Diagram shown in Volume I of the Safety Assessment. A fire main pressure gage located near the fire alarm panel in the chartroom allows the fire watchman to determine whether prope. pressure is being maintained at all times. All nozzles are variable-pressure with fog spray features.

2. Smothering

Fixed carbon dioxide systems protect machinery and reactor spaces, paint and lamp room, carpenter shop, boatswain's stores, general cargo spaces trunks, baggage rooms, and similar spaces. These systems are combined with the smoke detecting systems except in the carpenter shop. In addition, a fixed carbon dioxide system is provided within the containment shell.

Sufficient 100 lb cylinders of carbon dioxide are stored in various areas to provide protection for the largest compartment. The machinery space cylinders are arranged for simultaneous discharge to bilges and all machinery spaces up to "A" deck. Cross-connections between each group permit these cylinders to be used for cargo protection

and vice-versa. When carbon dioxide is released to the spaces a pressure switch shuts down the ventilation system.

Fixed, local carbon dioxide systems installed in the engineers paint locker and in the emergency generator room are arranged for the simultaneous release of all bottles in any one system. Portable fire extinguishers, sand, fire axes and related equipment are located in accordance with U. S. Coast Guard regulations.

Chapter IX

FLOODING AND SINKING

IX. FLOODING AND SINKING

Flooding of the SAVANNAH without sinking is of concern only if the machinery and reactor spaces are involved. Eventual loss of main and auxiliary propulsion units, turbine generator sets, and switchboards may be presumed.

Two methods of decay heat removal are available during the process. Initially, heat can be transferred from the reactor core to the steam generator by a primary coolant pump running at low speed. The steam generated on the secondary side can be dumped to the auxiliary condenser as long as that unit is functioning. The intermediate cooling system — the normal decay heat removal mechanism — will operate only until the pumps are submerged.

The emergency sea water cooling system within the containment shell will continue the decay heat removal process when the machinery space is lost. Power for this system is supplied from the emergency generator. The loss of all propulsive power is a typical hazard of the sea and is not considered since the vessel may be rescued by tugs, or, in extreme cases, may incur damage of the kinds previously analyzed.

The basic problem involved in sinking of the SAVANNAH is loss of power and the resultant effects on reactor decay heat removal. These effects depend primarily on accident circumstances such as rate of sinking, depth of water, and condition of machinery.

By far the most important of these is the time of initial submergence; i.e., the time lapse between recognition of the emergency and loss of the last source of power. This period has been known to vary from days to a matter of minutes.

Actually there will be no progressive loss of machinery. When the machinery space begins to flood, all auxiliaries, being nonsubmersible, will be secured immediately, rendering the entire propulsion plant inoperative and leaving only the emergency diesel generator as a power source. The emergency unit then supplies power to the

emergency sea water cooling system which is operated by relays from the console and the emergency switchboard, and decay heat removal can begin. The extent of decay heat removal depends upon the length of time before the emergency generator is submerged.

After the upper deck submerges, the containment shell flooding valves open and the containment vessel floods. The salt water serves as a heat sink to aid in decay heat removal and eliminates the possibility of a molten core melting through the pressure vessel. When the pressure inside and outside the containment vessel are equalized, the valves close. Fuel and fission products are triple-contained in the fuel element cladding, the primary system, and the flooded containment shell. In shallow water, the ship can be salvaged and the radioactive components recovered. In deep water, where salvage is impossible, corrosion through the triple barrier will require many years, thus delaying fission product release until the activity has decayed to a very low level.

If the core is not completely cooled before the ship submerges, partial meltdown will occur after sinking and fission products may be released to the flooded containment shell via the primary system relief valves.

If the ship breaks up, freeing the containment shell, the containment vessel with about 700 tons of steel and 550 tons of lead shielding would sink — and the previous analysis would still apply.

Chapter X

SALVAGE

A. Hypothetical Maximum Accident

B. Salvage of the Sunken Vessel

C. Salvage of the Grounded Vessel

X. SALVAGE

After a serious accident, correct predetermined procedures must be initiated for immobilization, recovery, or decontamination of SAVANNAH's reactor system. These procedures for either nuclear or marine accidents include definition of exclusion areas following the accident and procedures for salvage of the sunken or grounded vessel. Two general areas of safety are involved: (1) safety of passengers and crew, and (2) assurance that the abandoned nuclear plant does not constitute a hazard to the environment.

The most serious hypothetical accident (described in part one of this Volume) involves the ductile failure of a main coolant loop and the rapid loss of primary coolant to the containment vessel. The loss of primary coolant from the reactor removes the method of transferring decay heat from the reactor core with the result that the core temperature will eventually reach the melting point of the cladding material (stainless steel). When this point is reached, the fission products, which have accumulated in the fuel pins, are released to the reactor and eventually to the containment vessel through the failure in the primary system.

As described in Part I of this volume and in Volume I of the Safety Assessment every effort was made to insure that the containment would be completely isolated in the event of such an accident. Automatic valves (which close on high containment pressure) and check valves will seal off any possible escape route for the containment gas through any penetrations which may fail inside the containment vessel. However, for this analysis, a finite leak rate to the reactor compartment lower void of 0.2% of the containment free volume per 24 hours was assumed.

The question of environmental hazard was discussed in part one of this volume. The following section discusses the potential radiation levels which may exist at various locations on the ship following the

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maximum hypothetical accident

A. HYPOTHETICAL MAXIMUM ACCIDENT

The maximum dose rate is based on 100% of the gaseous fission products and smaller percentages of particulate fission products being released from the molten core. Both the magnitude of the dose and the time required to melt the core are difficult to predict, but the values given are believed to represent the worst conditions possible.

The factors of conservatism in this and other calculations should be continually borne in mind particularly in the determination of exclusion areas. The calculations that follow, also based on the worst possible conditions, are intended only to show the feasibility of certain operations. Since the actual radiation levels will generally be much lower, radiation surveys will be conducted at the time of the accident to determine exclusion areas.

Figure 58 shows the radiation level fall off with distance aft along the ship's centerline at the 23-ft level. In this case it is assumed that all hold spaces are empty and that the only attenuating medium is ship structure. The machinery spaces and cargo holds are untenable at the maximum radiation level. The radiation level falls off quite sharply from the maximum value (a factor of about 100 in 24 hours) as shown in Figure 59. Hence the integrated doses at various points on the ship are of more interest.

In Figure 60, the integrated dose on the main deck aft of hold No. 5 during the first 24 hours is plotted against position. This area obviously can be continuously manned. If there is cargo in the after holds the radiation level in most of the ship's after part are negligible. Thus, if there is power to the shaft, the ship can be steered from the after steering station for an indefinite period.

The leakage of radioactive gases through the containment can be controlled by maneuvering the ship to clear inhabited areas and other vessels of the masthead discharge.

Access to the midship area is limited. However, if the ship must be abandoned the integrated dose on the boat and promenade decks will be much lower than the occupational emergency exposure of 25 rem set by the National Committee on Radiation Protection (NCRP).*

* Addendum to National Bureau of Standards Handbook 59 "Permissible Dose from External Sources of Ionizing Radiations", April 15, 1958

The integrated dose during the 30 minutes required to muster and disembark passengers are plotted in Figure 61.

The machinery space also must be temporarily abandoned. Integrated doses in the control room will be about 1 r during the first 30 minutes and 5 r during the 24-hour period immediately following. However, a period of about 13 minutes is available before maximum shipboard radiation levels are obtained. If the rupture is detected immediately, the plant can be secured and the take-home motor brought on the line before maximum radiation levels prevail. Attenuation by machinery and components was not considered in calculating maximum radiation levels for the engine room. Many areas of the space should be fully accessible at all times in view of the local shielding. In particular, the lower level, to which access is available through the shaft alley, might be completely accessible for limited periods under all conditions.

After the first 24 hours, all areas of the ship will be accessible for various periods of time, so the passenger quarters and midship areas in general may be reoccupied, but subject to local monitoring. Note again that the conditions and radiation levels represent the worst possible conditions, and that exclusion areas and operations may be substantially modified if local radiation surveys so warrant.

In conclusion, under the most pessimistic assumptions, it appears that this accident would not expose passengers and crew to radiation levels much in excess of emergency exposure recommendations. * The National Committee for Radiation Protection recommended allowable emergency dose for radiation workers — 25 rem, allowed only once in an individual's lifetime — could be obtained only if the worker spent several hours in areas of the highest radiation level.

Even under the worst condition, power is available for navigation, and passengers can be evacuated to the ship's after part or they can abandon ship, depending on wind and sea conditions and the magnitude of the accident.

* Cottrell, W. B., et al., Environmental Analysis of NS SAVANNAH Operation at Camden, ORNL-2867, Oak Ridge National Laboratory, May 5, 1960.

The rapid fall-off in radiation levels will permit almost unlimited access to any part of the vessel after 24 hours. Therefore, subject to wind and sea conditions, and assuming the ship proper is undamaged, no particular difficulty will be encountered in bringing the ship into a safe anchorage either with the take-home motor or with tugs. Containment leakage can be controlled through the ventilation system since the reactor compartment is essentially gastight. Once the ship is safely anchored, normal cleanup and decontamination procedures will apply.

B. SALVAGE OF THE SUNKEN VESSEL

In all cases of sinking it is assumed that the reactor is shut down and that decay heat removal has been initiated, since these operations can be carried out before the ship is abandoned.

After interruption of decay heat removal, when the ship sinks and the emergency diesel generator stops, there is not enough heat loss through the pressure vessel walls, containment air, and containment vessel to dissipate the decay heat. Ultimately then, the release of primary coolant through the pressurizer relief valves results in some melting of the core. Fission gases from the core meltdown may or may not be released to the containment. In particular, containment pressures are very low, and leakage of fission product gases through the containment is negligible.

The feasibility of salvaging a sunken vessel depends almost entirely on the depth of water.

1. Salvage at Depths of Less Than 100 Feet

As described in Part Two, Chapter IX, the SAVANNAH containment shell will flood at depths below 100 ft. In shallower water some parts of the ship's superstructure will remain above the water. If sunk on an even keel, the emergency diesel generator might remain in operation to cool the core fully and achieve normal plant shutdown. However, if the ship sinks on her side, submerging the emergency generator, the core might melt down — in which case the containment would be contaminated but unflooded.

Depths of less than 100 ft are usually found in harbors and shipping channels. If a vessel sinks in one of these areas, the port authorities may require its removal as a navigational hazard. A conventional vessel can be raised as a unit or cut up and raised in sections. Under extreme conditions — say strong currents which interfere with diving operations, the ship's structure can be blown apart and spread over the bottom. All three of these methods are partially applicable to the SAVANNAH except that the reactor containment must not be ruptured.

The containment shell has four 1 1/2-in. salvage connections, internally piped to four points 90 degrees apart on the internal circumference. These fittings — sized to accommodate standard Navy diver's

hose – are located on the top of the containment shell, providing relatively simple access to a diver.

Liquid or gas inside the sunken containment vessel can be sampled through the salvage connections, making it possible to determine the condition of the containment any time after sinking. If a sample is radioactive, several measures can be taken depending on the amount of damage to the ship, the urgency of demands for removal of the ship, and the diving conditions.

Since a vessel is likely to sustain only slight damage in a harbor sinking, it may be possible to salvage it intact using standard techniques (constructing cofferdams, sealing openings and pumping, etc.). Experts in the salvage field have indicated that any vessel can be raised from waters of less than 100 ft depth if cost is no object. This is particularly true in harbors where weather and sea conditions are unlikely to interfere with diving operations. Throughout the refloating operation, as a prelude to cleanup, the containment can be continuously purged with uncontaminated water via the salvage connections.

If diving conditions or the condition of the ship make the expense of refloating prohibitive, the ship could be blown apart with explosives. Although care would be necessary in the vicinity of the containment, wartime experience with explosives, particularly the cord type, clearly indicates that the containment could be cleared from the ship structure without damage, then raised and brought to shore by tidal lift.

The water shielding, as well as the rapid decay of fission product activity (200 mr/hr 24 hours after shutdown) will protect a diver from an overdose of radiation at all times. Even under optimum salvage conditions, a diver would not reach the containment area for several days after shutdown.

2. Salvage at Depths of 100 - 300 Feet

At a depth of 100 ft, the containment becomes flooded, but this in itself would not affect the salvage operation. Below 100 ft, depth and sea conditions are the principal factors. If the ship is lying in protected waters, it can be raised if absolutely necessary, or the containment can be freed from the ship's structure and raised separately.

Unless raising of the vessel were imperative for navigational or other purposes, the containment shell would be filled with concrete and left on the bottom. As the depth of water approaches 300 ft, the labor involved in removing the reactor hatch and attaching the hose to the salvage connections essentially represents the limit of accomplishment for divers. Even this program would probably require several divers and considerable time. With the fission products encased in concrete, and the whole mass surrounded by the containment vessel, no problem of contamination is expected.

3. Salvage at Depths Exceeding 300 Feet

Below 300 ft, essentially no work can be performed by a diver. The extreme working depth is very close to 300 ft, at which point a diver can remain on the bottom only a few minutes at a time. Therefore, nothing could be done to recover or immobilize the reactor system at depths below 300 ft.

This would not necessarily constitute a hazard as the containment shell will be flooded and the flooding valves closed. Corrosion in deep waters takes place much more slowly than in shallow water and, protected by the ship's structure, circulation rates around the containment are low. Many years might pass before corrosion penetrates the containment vessel. The massive primary system would still contain most of the long-lived fission products and many more years would pass before they were released. Circulation within the containment would be practically nonexistent. In fact, the containment-primary system complex is probably the most effective waste disposal package ever devised.

In conclusion, the sunken ship is not expected to present a hazard to the environment. Above 300 ft the fission products can be controlled or immobilized. Below 300 ft the containment and primary system provide adequate protection over an extremely long period of time.

C. SALVAGE OF THE GROUNDED VESSEL

Salvage in this case is often much easier than salvage of a sunken vessel, although the feasibility of recovery is again an economic problem. With no problems of underwater labor, almost any grounded ship can be salvaged, if only in sections. Again, if the vessel is filled with concrete the fission products will be permanently fixed. On a rough, rocky shore the ship may break up over a long period of time during which, however, the containment could be moved to deeper water and sunk. Since the concrete-filled containment will weigh more than 3000 tons, it is not likely to be worked or ground on the rocks even by the roughest seas.

If grounding occurs in an inhabited area, containment leakage will not present a serious hazard unless the emergency generator is lost and part of the containment remains above water. In this case, the containment can be purged with water to remove the soluble gaseous fission products. In addition, the low pressures resulting from loss of coolant through relief valves will significantly diminish the magnitude of leakage through the containment. In conclusion, the grounded vessel will not constitute a hazard to the environment.

FIG. 58: DOSE RATE AFT OF THE CONTAINMENT VESSEL FOLLOWING MAXIMUM HYPOTHETICAL ACCIDENT
(Along Ship's Centerline at 23 ft Level 13 min After the Accident)

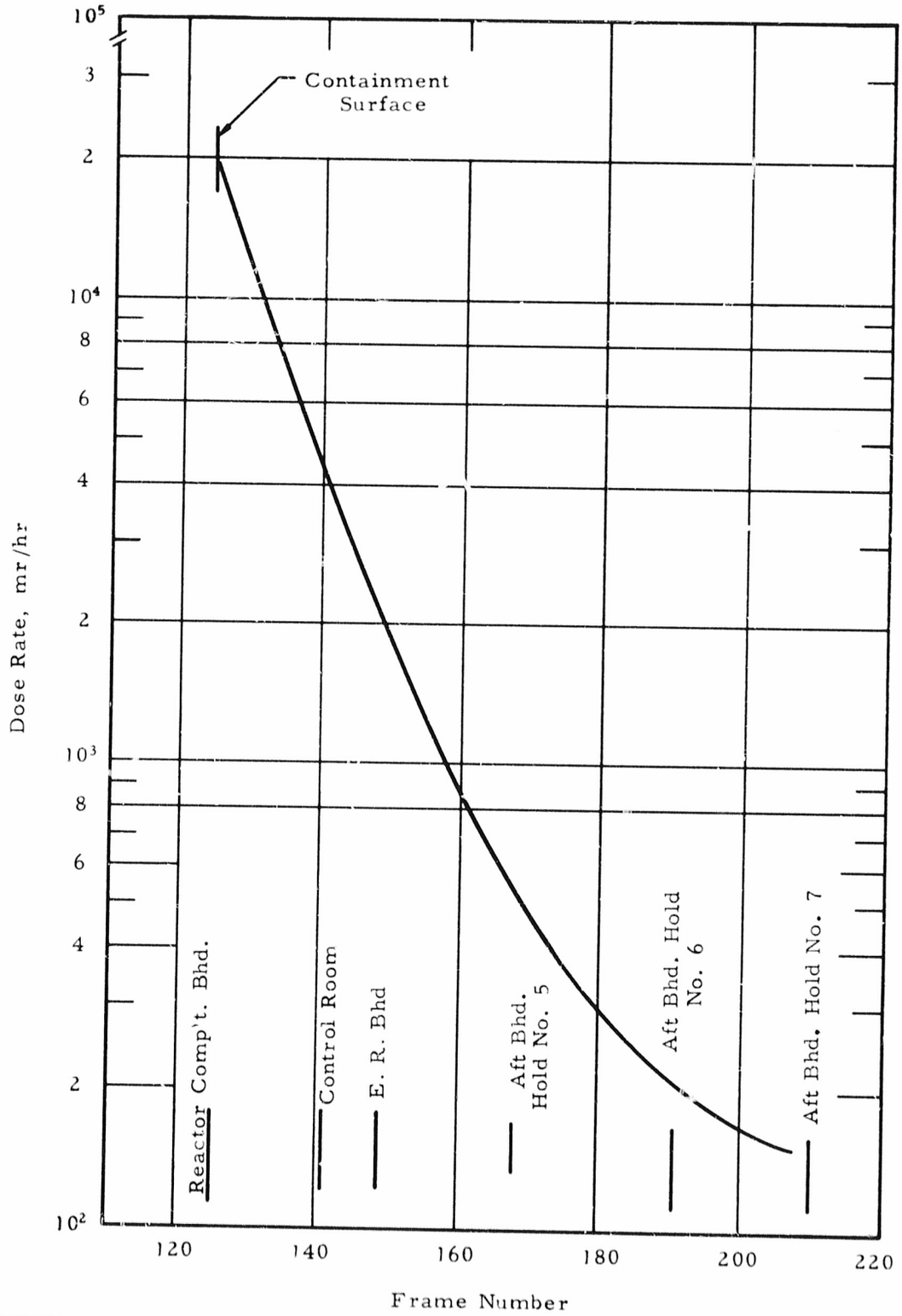

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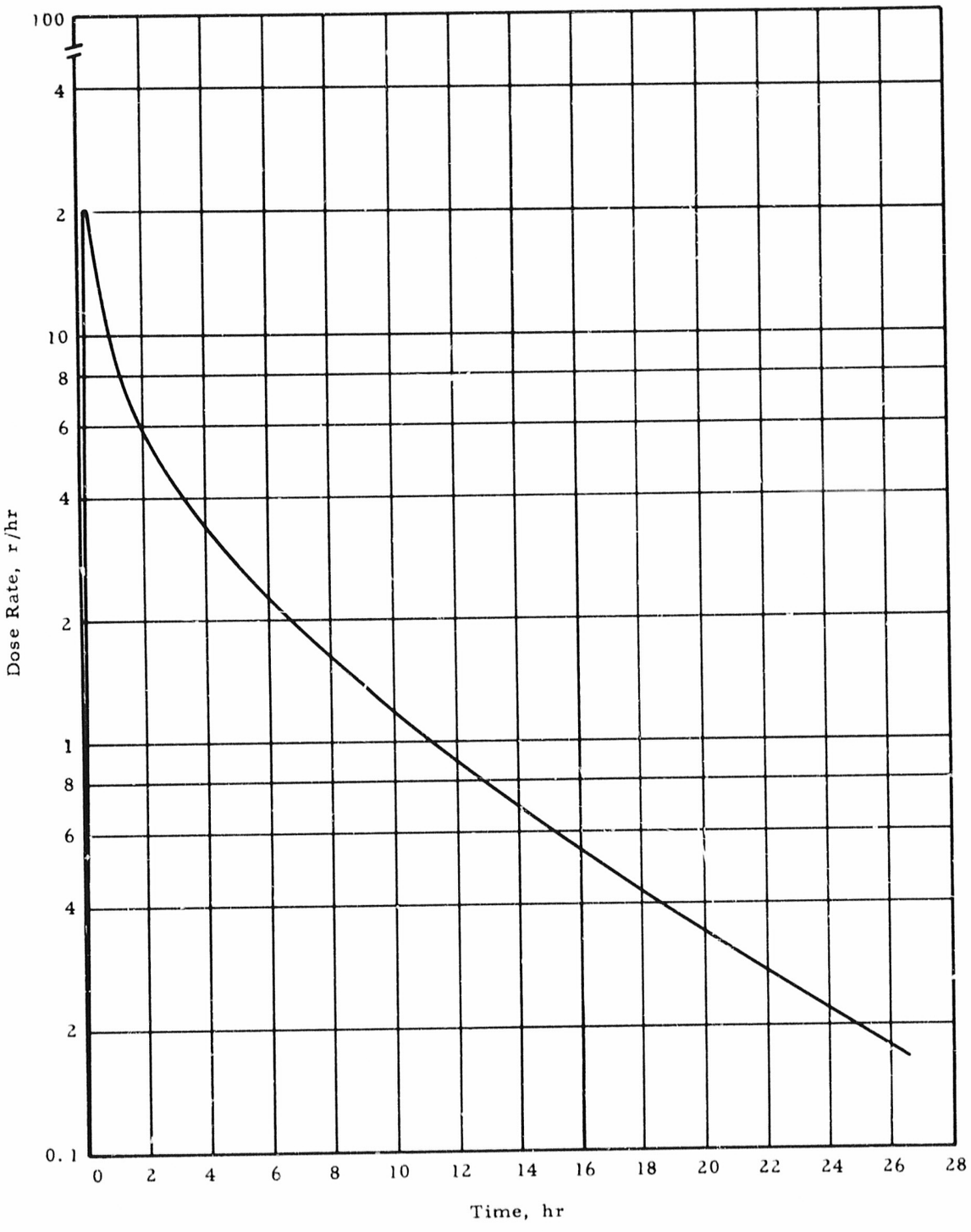
FIGURE 58

FIG. 59: DOSE RATE AT REACTOR COMPARTMENT BULKHEAD Vs TIME AFTER MAXIMUM HYPOTHETICAL ACCIDENT



The Babcock & Wilcox Co.
Atomic Energy Division

FIGURE 59

FIG. 60: TOTAL INTEGRATED DOSE ON "A" DECK AFT OF SHIP'S CENTERLINE
(First 24 hours Following Maximum Hypothetical Accident)

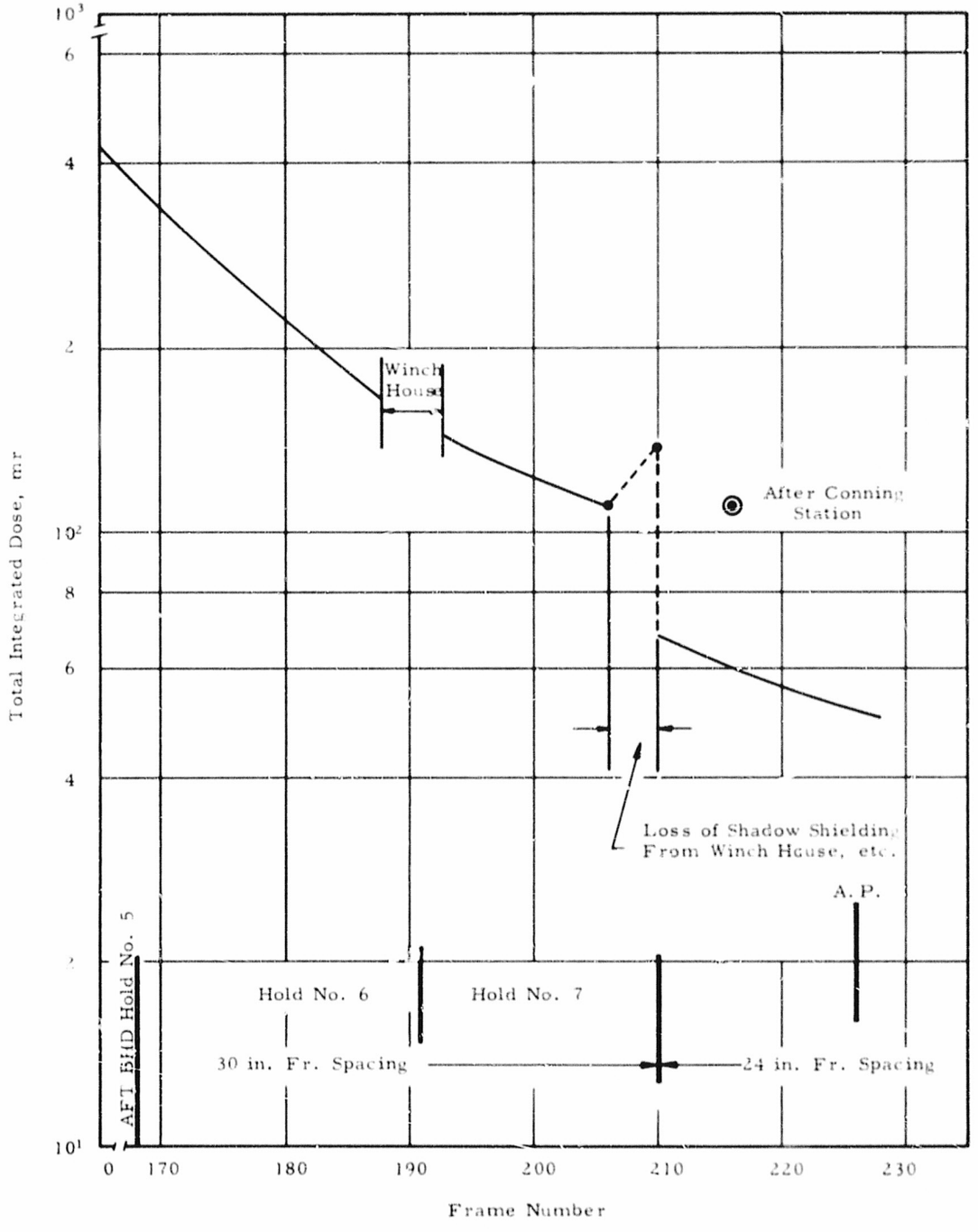


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FIGURE 60

FIG. 61: TOTAL INTEGRATED DOSE AT VARIOUS SHIP LOCATIONS
(First 30 minutes Following Maximum Hypothetical Accident)

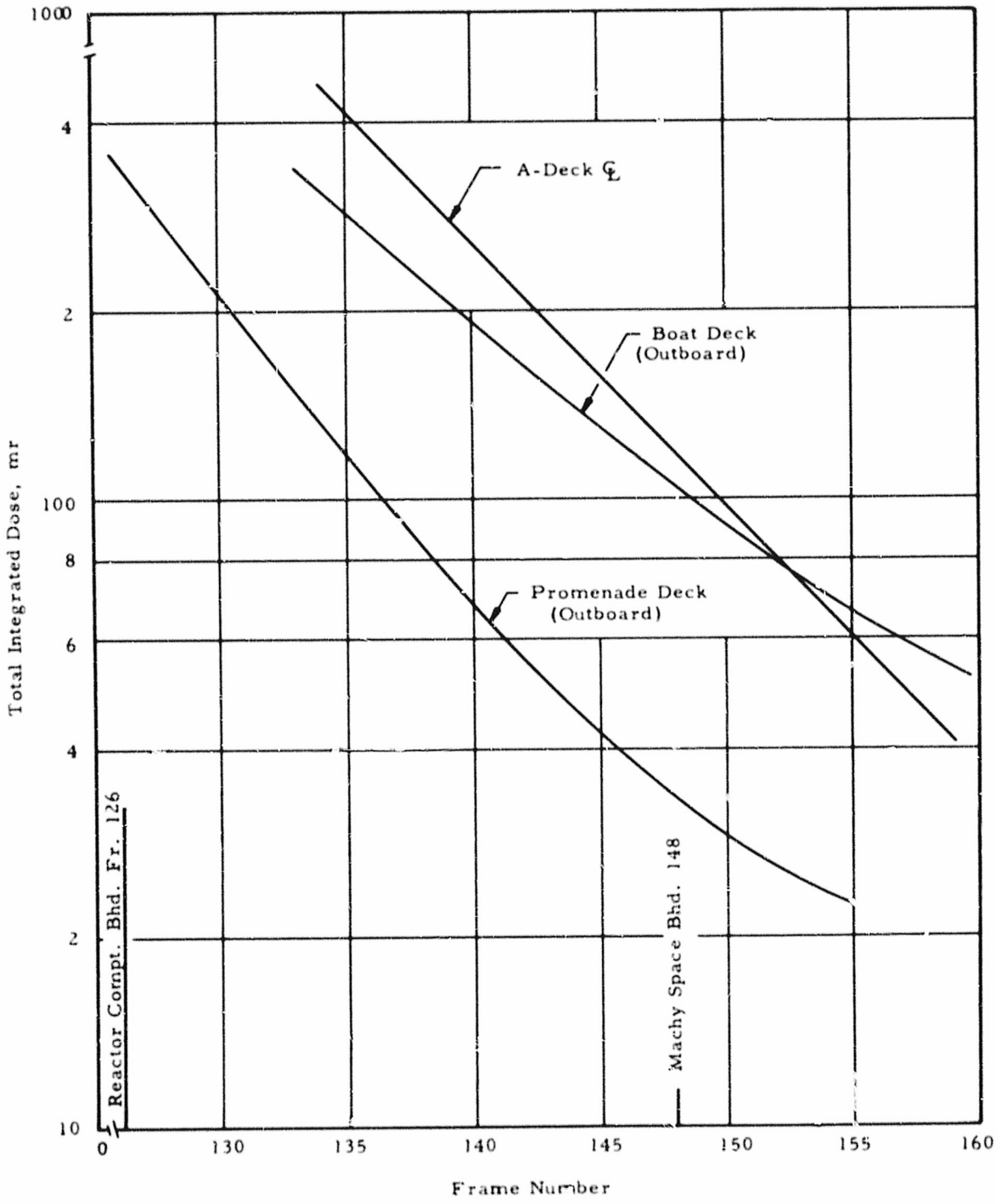

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FIGURE 61

Chapter XI

DESCRIPTION OF THE SAVANNAH

XI. DESCRIPTION OF THE SAVANNAH

The following brief description of the SAVANNAH and her reactor system is included for the convenience of the reader. Further details of SAVANNAH's system and components are given in Volume I, NS SAVANNAH Safety Assessment, Design and Engineering.

The NS SAVANNAH is a single-screw, shelter-deck, passenger-cargo ship with raked stem and modified cruiser stern. It has three complete decks, "A", "B", and "C", with platform decks at various levels. A promenade deck extends over "A" deck for about one-third of the ship's length, the forward end slightly forward of amidships. Passenger staterooms are on "A" deck, public rooms and the swimming pool on the promenade deck, and the passenger dining room on the "B" deck.

Ten watertight transverse bulkheads divide the hull into peak spaces, seven holds, a machinery space, and a reactor space. From the bow these components are as given below.

Hold Number	Length, ft-in.	Assignment
Fore Peak	28-0	Tank and Stores
1	50-0	General Cargo and Deep Tank
2	55-0	General Cargo
3	55-0	General Cargo
4	55-0	General Cargo
Reactor	60-0	Reactor, Stores, Utility Spaces and Stabilizers
Machinery	55-0	Machinery
5	50-0	General Cargo
6	57-6	General Cargo
7	47-6	General Cargo, Supplementary Startup Boilers
Aft Peak	32-0	Tank, Steering Gear, Stores

At load draft of 29 ft-6 in. in salt water, the ship is designed for a total deadweight of 10,000 tons, distributed as follows

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Item	Weight, Long Tons
Passengers, Crew, Effects, and Stores	200
Diesel Oil, Lubricating Oil, Fresh Water	550
General Cargo	9250
Total Deadweight	10,000

Preliminary light ship weights are as follows.

Item	Weight, Long Tons
Containment and Shielding	1930
Hull Steel	7300
Ship Outfit and Equipment	870
Propulsion Equipment (wet)	1750
Total Light Ship	11,850

The principal characteristics of the ship are given below.

Dimensions	
Length, over-all (approx.)	596 ft 6 in.
Length, Forepost to Rudder Post	545 ft 0 in.
Beam, molded	78 ft 0 in.
Depth, Baseline to "A" deck	50 ft 0 in.
Draft, design	29 ft 6 in.
Draft, light (approx.)	18 ft 6 in.

Coefficients (Based on Effective Form Length)	
Block	0.621
Prismatic	0.640
Midship Section	0.971
Waterplane	0.754
Speed-Length Ratio (sea speed)	0.88
Speed-Length Ratio (trial speed)	0.91

Displacement and Tonnage	
Full Load Displacement	21,850 Tons
Light Ship (approx.)	11,850 Tons
Cargo Deadweight (approx.)	9,250 Tons
Gross Tonnage (approx.)	12,220 Tons
Net Tonnage (approx.)	6,030 Tons

Complement	
Passengers (one class)	60
Officers, Crew and others	124
Total	184

Speed and Power

Trial Speed at Normal Shaft Horsepower, kno	21
Speed with Emergency Propulsion Motor	7
Normal Shaft Horsepower	20,000
Maximum Continuous Shaft Horsepower	22,000

Machinery

Main Unit — pressurized water nuclear reactor; geared steam turbine

Emergency Propulsion Motor, Effective Shaft Horsepower		640
Number Propellers		1
Turbine Generators, kw	2 at	1,500
Diesel Generators, kw	2 at	750
Emergency Diesel Generators, kw	1 at	300
Evaporators, gallons per day	2 at	16,000

Cargo Capacity

General Cargo, cu ft	746,000
Refrigerated Cargo	None
Cargo Oil	None

Arrangements of "A", "B", and "C" decks, and an inboard and outboard profile are shown in Figures 62, 63, and 64. Arrangement of shielding, containment, and spaces adjacent to the reactor compartment are shown in Figure 37.

The SAVANNAH is designed to a two-compartment standard of subdivision at a draft of 29 ft-6 in. and with margin line 3 in. below "B" deck.* This is in excess of the subdivision requirements of the U. S. Coast Guard and Senate Report 184. The ship will comply with all applicable laws of the United States, the requirements of the various regulatory bodies, and the rules, as listed in Part Two, Chapter II, A, in force at the time of delivery.

Stability of the SAVANNAH is equivalent to that of a normal passenger-cargo ship. In addition, since no fuel oil will be consumed, the measure of stability (GM) will vary less during a long voyage.

The wheelhouse and navigation bridge are designed to give maximum visibility through the widest possible angle, allowing almost 180 degrees

* The ship will remain afloat when two compartments are completely flooded.

vision, which is equal to or greater than that of any other passenger-cargo ship. Navigation and communication equipment are the best available.

The machinery space is located just aft of amidships. The control room is on the "20 ft Flat" between frames 141 and 149. In general, the propulsion plant conforms to conventional marine practice except for particular features dictated by the use of saturated steam.

The main propulsion unit consists of a cross-compound geared turbine operating at 430 psig on steam saturated at the throttle to deliver 20,000 shp normal and 22,000 shp maximum. A wound rotor electric take-home motor is provided as a standby for the main propulsion unit. Developing 750 hp (nominal), the take-home motor is coupled to one of the high speed pinions in the reduction gear, and a quick-connect coupling permits engagement in less than two minutes. With the main propulsion unit secured and the take-home motor in use, SAVANNAH can develop about 7 knots in a mild sea. In case of reactor failure, the stored heat in the primary system will be available for a short period, so the SAVANNAH will never be without power to the shaft.

Fresh water is supplied from two 16,000-gpd salt water distilling plants of the low-pressure, two-stage flash type. Operation and performance requirements are listed below.

Capacity, gallons per day (after 90 days at 7.5 psia bled steam 85° F cooling water)	10,400
Capacity, gallons per day (after 90 days operation with auxiliary steam at 5 psig and 85° F cooling water)	16,000
Feed water density	1/32nd
Brine density (maximum)	1.5/32nd
Shell vacuum, in. Hg (second stage)	24 to 27
Sea salt distillate, grains/gallon (maximum)	0.25
Tube material	90-10 Cu-Ni
Tube diameter, in.	5/8
Tube wall thickness, in.	.049
Evaporator feed temperature, °F	85

The distilling plants furnish all primary and secondary makeup water for the power plant. Although a separate supply of water is available for drinking, washing, and cooling, the evaporators will normally provide for these functions.

The SAVANNAH's reactor compartment, located forward of the machinery room between frames 99 and 126, is enclosed by transverse bulkheads at these frames and by longitudinal bulkheads located 21 ft-9 in. off the centerline, port and starboard. Between these bulkheads and the shell of the ship are located storage and stabilizer spaces, baggage rooms, and other miscellaneous spaces.

The specially strengthened decks outboard of the reactor compartment form the principal barrier to penetration in event of collision. At the longitudinal bulkhead, additional collision resistance is obtained from the complex formed by the concrete secondary shield, the redwood and steel collision mat, and the longitudinal bulkhead. (See Fig. 65.) The collision mat itself is composed of alternate layers of 1-in. steel plates and 3-in. slabs of redwood.

The reactor compartment bulkheads support the secondary biological shielding in the area from "C" deck to the inner bottom. In most areas, this shield is 4-ft-thick reinforced ordinary concrete. From "C" deck to the 14 ft-0 in. level and extending 15 ft fore and aft from the containment center, the outboard 2 ft of concrete is replaced by the collision mat. Above "C" deck the secondary shield is composed of lead and polyethylene laid directly on the upper half of the containment shell and cupola.

The reactor's heavy steel containment shell, designed to ASME standards for unfired pressure vessels, is a gas-tight structure designed to withstand 173 psi — the pressure surge anticipated from the assumed hypothetical maximum accident. The dual-purpose containment vessel will contain gases released in the event of a severe accident, and also supports the secondary shielding.

There are no generally accepted codes or specifications covering such a dual-purpose structure, so both the shell and shielding support systems were designed separately. The shell and all penetrations comply with the ASME Code for Unfired Pressure Vessels, and the supporting rings have been analyzed under the first principles of mechanics. The

members provide a safety factor of four for all loads except that of the hypothetical maximum accident, and a factor of three for all loads including the hypothetical maximum accident. These safety factors are obtained for support of secondary shielding and for 30 degree roll of the ship. The calculations have been thoroughly checked by the U. S. Coast Guard and the American Bureau of Shipping.

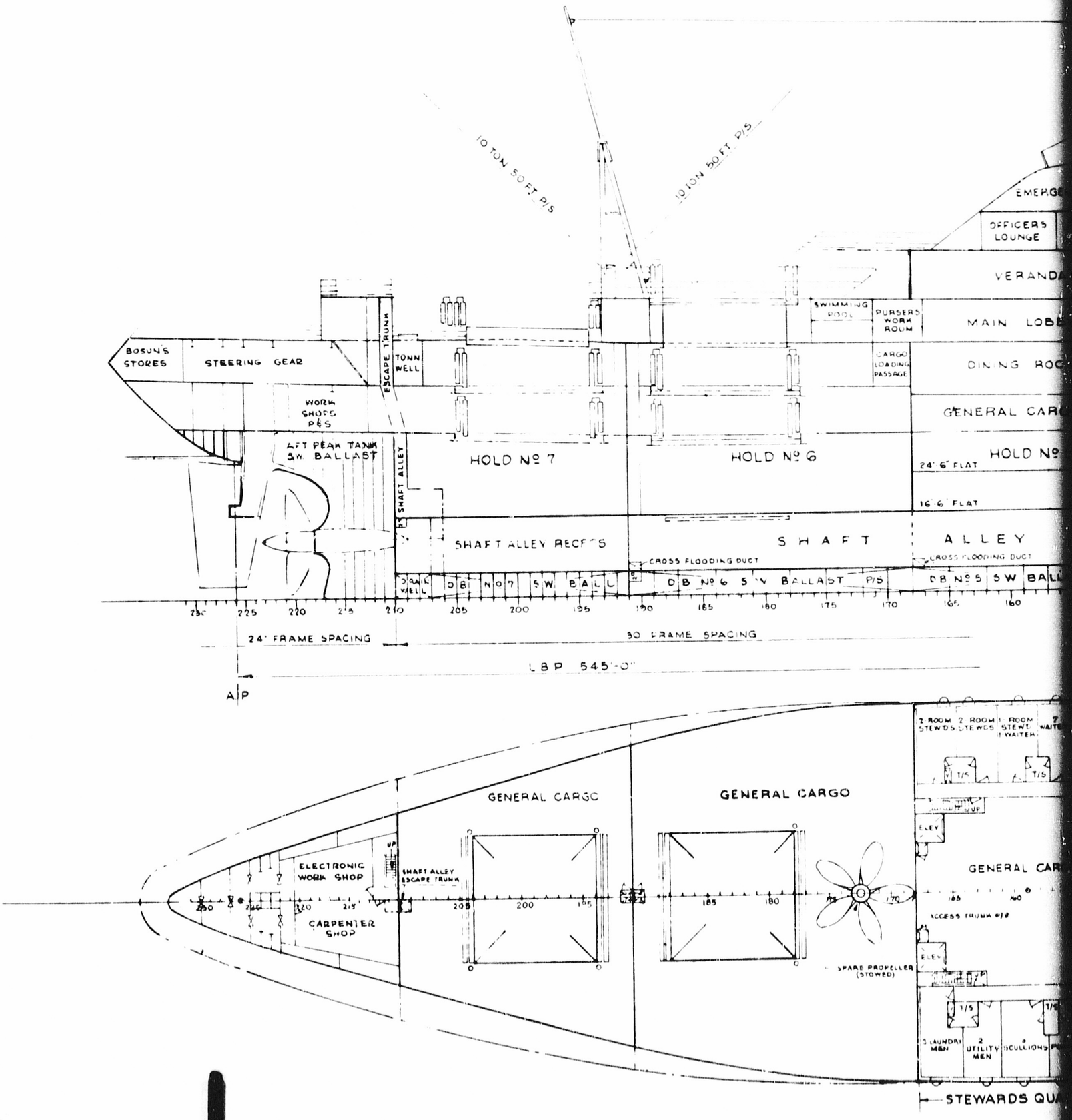
The containment shell foundation structure, tied into the ship's double-bottom structure, can withstand any anticipated load under normal or accident conditions. Above the tank top, three deep continuous longitudinal girders on each side, port and starboard, are welded to the tank top to support the containment vessel in the cylindrical portion only (the hemispherical ends overhang). These girders, set normal to the containment vessel, are integrated transversely with deep saddles which fall in line with the inner bottom floors. Only the after-end of the containment vessel is bolted to a transverse saddle of very heavy construction, leaving the vessel free to expand. Chocks following the contour of the hemispherical ends are fitted to the top of two longitudinal girders at both ends, the after chocks being hard up against the containment shell and the forward chocks approximately 5/8 in. clear to allow for expansion.

Additional longitudinal support, port and starboard, is supplied by the intersection of "C" deck and the reactor compartment longitudinal bulkhead. A wide heavy faceplate at the intersection provides positive support in the event the ship is on her beam ends.

The space below the containment shell and within the concrete shielding contains the primary coolant purification system, waste collection and storage systems, and other heat transport and supporting systems. The reactor compartment bulkhead and deck scantlings are effectively gastight and are designed to an internal collapse pressure of 5 psi.

The space surrounding the containment vessel is provided with natural air supply and a mechanical exhaust. (See Fig. 11.) During normal operation, there will be a complete air change every 8 minutes. When desired, the containment vessel may be purged by opening a remotely operated valve in the bottom of the containment vessel and exhausting air to the lower void space ventilation system. Discharge

to the weather may be made alternatively at three points: one each port and starboard just above the water line, and the third at the top of the forward mast approximately 90 ft above the water line. All air exhausted from the reactor space to the outside is filtered by a series of high efficiency filters (roughing, absolute, and iodine) located downstream of the exhaust blower. In addition, an auxiliary reactor compartment lower void ventilating system containing a five stage filter will be provided to assure that this space can always be vented through particulate and iodine filters. The effluent from the auxiliary system will be discharged from the top of the mast or the top of the superstructure. Continuous radiation monitoring of exhaust air is provided after filtering. Chemical smoke tracing provided in the exhaust system indicates the trajectory of discharge away from the ship.


10 TON 50 FT P/S
10 TON 50 FT P/S
EMERGE
OFFICERS LOUNGE
VERANDA
MAIN LOBE
DINING ROO
GENERAL CARG
HOLD Nº
24' 6" FLAT
16' 6" FLAT
SWIMMING POOL
PURSERS WORK ROOM
CARGO LOADING PASSAGE
BOSUN'S STORES
STEERING GEAR
ESCAPE TRUNK
TONN WELL
WORK SHOPS P&S
AFT PEAK TANK S.W. BALLAST
SHAFT ALLEY
HOLD Nº 7
HOLD Nº 6
SHAFT ALLEY RECESS
SHAFT ALLEY
CROSS FLOODING DUCT
CROSS FLOODING DUCT
DRAIN WELL
D B Nº 7 S W BALL
D B Nº 6 S W BALLAST P/S
D B Nº 5 S W BALL
230 225 220 215 210 205 200 195 190 185 180 175 170 165 160
24" FRAME SPACING
30 FRAME SPACING
L B P 545'-0"
A P
GENERAL CARGO
GENERAL CARGO
ELECTRONIC WORK SHOP
SHAFT ALLEY ESCAPE TRUNK
CARPENTER SHOP
UP
230 225 220 215 205 200 195 185 180 170 165 160
SPARE PROPELLER (STOWED)
2 ROOM STEWDS
2 ROOM STEWDS
1 ROOM STEWD 1 WAITER
T/S
T/S
ELEV
GENERAL CAR
ACCESS TRUNK
ELEV
T/S
T/S
3 LAUNDRY MEN
2 UTILITY MEN
3 SCULLIONS
STEWARDS QU

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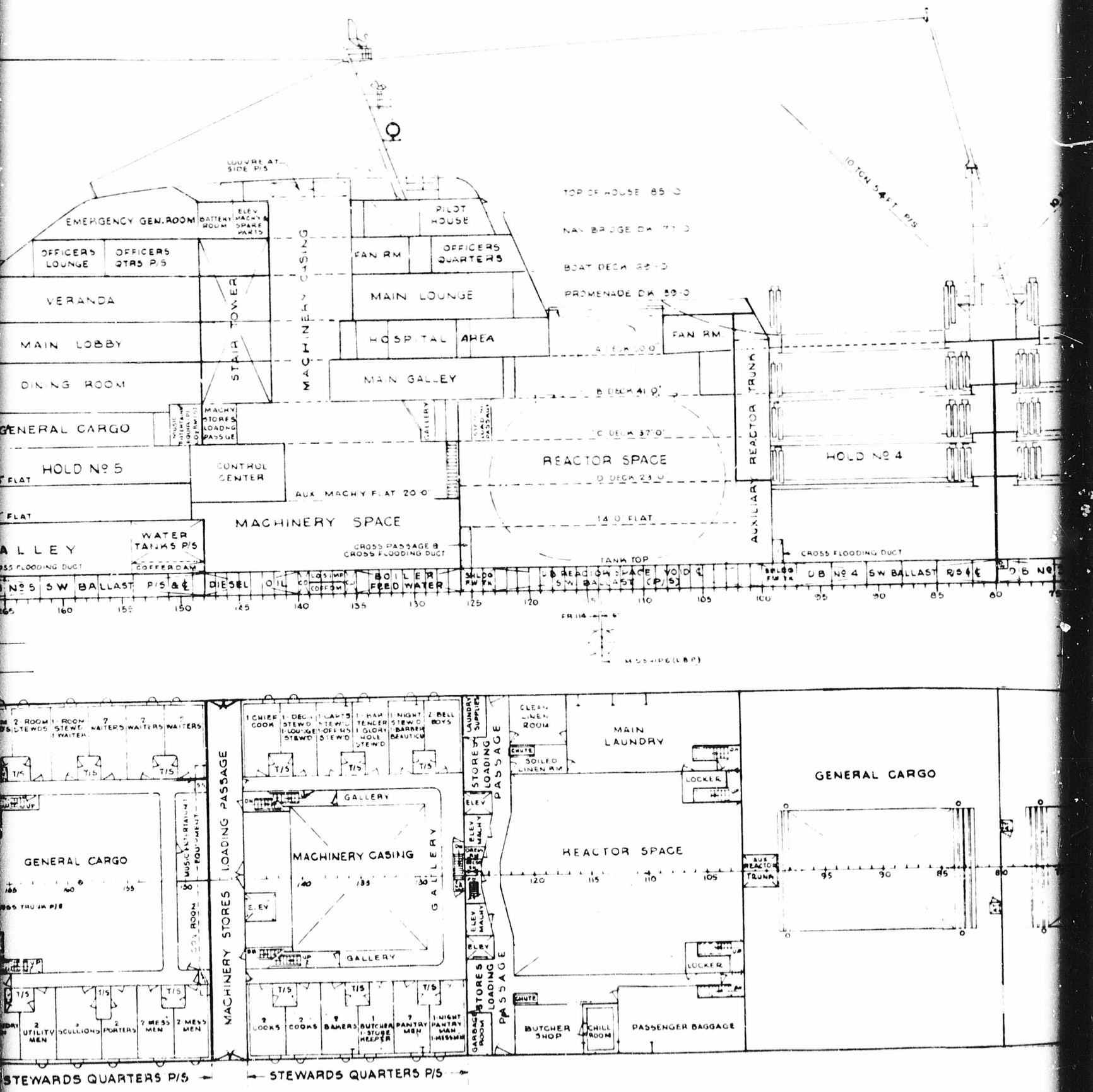

LOUVRE AT SIDE P/S
EMERGENCY GEN. ROOM
BATTERY ROOM
ELEV MACHY & SPARE PARTS
OFFICERS LOUNGE
OFFICERS QTRS P/S
VERANDA
MAIN LOBBY
DINING ROOM
GENERAL CARGO
HOLD Nº 5
STAIR TOWER
MACHINERY CASING
PILOT HOUSE
FAN RM
OFFICERS QUARTERS
MAIN LOUNGE
HOSPITAL AREA
MAIN GALLEY
MACHY STORES LOADING PASSGE
CONTROL CENTER
AUX MACHY FLAT 20'-0"
MACHINERY SPACE
WATER TANKS P/S
COFFERDAM
CROSS PASSAGE & CROSS FLOODING DUCT
TOP OF HOUSE 85'-0"
NAV BRIDGE DK 77'-0"
BOAT DECK 69'-0"
PROMENADE DK 59'-0"
FAN RM
A DECK 50'-0"
B DECK 41'-0"
C DECK 32'-0"
D DECK 23'-0"
14'-0" FLAT
REACTOR SPACE
TANK TOP
AUXILIARY REACTOR TRUNK
HOLD Nº 4
CROSS FLOODING DUCT
10 TON 54 FT P/S
SW BALLAST P/S & ℄
DIESEL OIL
BOILER FEED WATER
SHLDG FW TK
DB REACTOR SPACE VOID ℄ SW BALLAST (P/S)
DB Nº 4 SW BALLAST P/S & ℄
FR 114
MIDSHIPS (LBP)
2-ROOM STEWDS
1-ROOM STEWD 1 WAITER
2 WAITERS
1 CHIEF COOK
1-DECK STEWD 1-LOUNGE STEWD
1-CAPTS STEWD 1-OFFRS STEWD
1-BAR TENDER 1 GLORY HOLE STEWD
1-NIGHT STEWD 1-BARBER BEAUTICN
2-BELL BOYS
LAUNDRY SUPPLIES
CLEAN LINEN ROOM
MAIN LAUNDRY
SOILED LINEN RM
LOCKER
GENERAL CARGO
MUSIC ENTERTAINMENT EQUIPMENT
MACHINERY STORES LOADING PASSAGE
GALLERY
MACHINERY CASING
ELEV
ELEV MACHY
REACTOR SPACE
AUX REACTOR TRUNK
STORES LOADING PASSAGE
2 UTILITY MEN
2 SCULLIONS
2 PORTERS
2 MESS MEN
2 COOKS
2 BAKERS
1 BUTCHER 1-STORE KEEPER
2 PANTRY MEN
1-NIGHT PANTRY MAN 1-MESSMAN
GARBAGE ROOM
BUTCHER SHOP
CHILL ROOM
PASSENGER BAGGAGE
STEWARDS QUARTERS P/S
STEWARDS QUARTERS P/S

2

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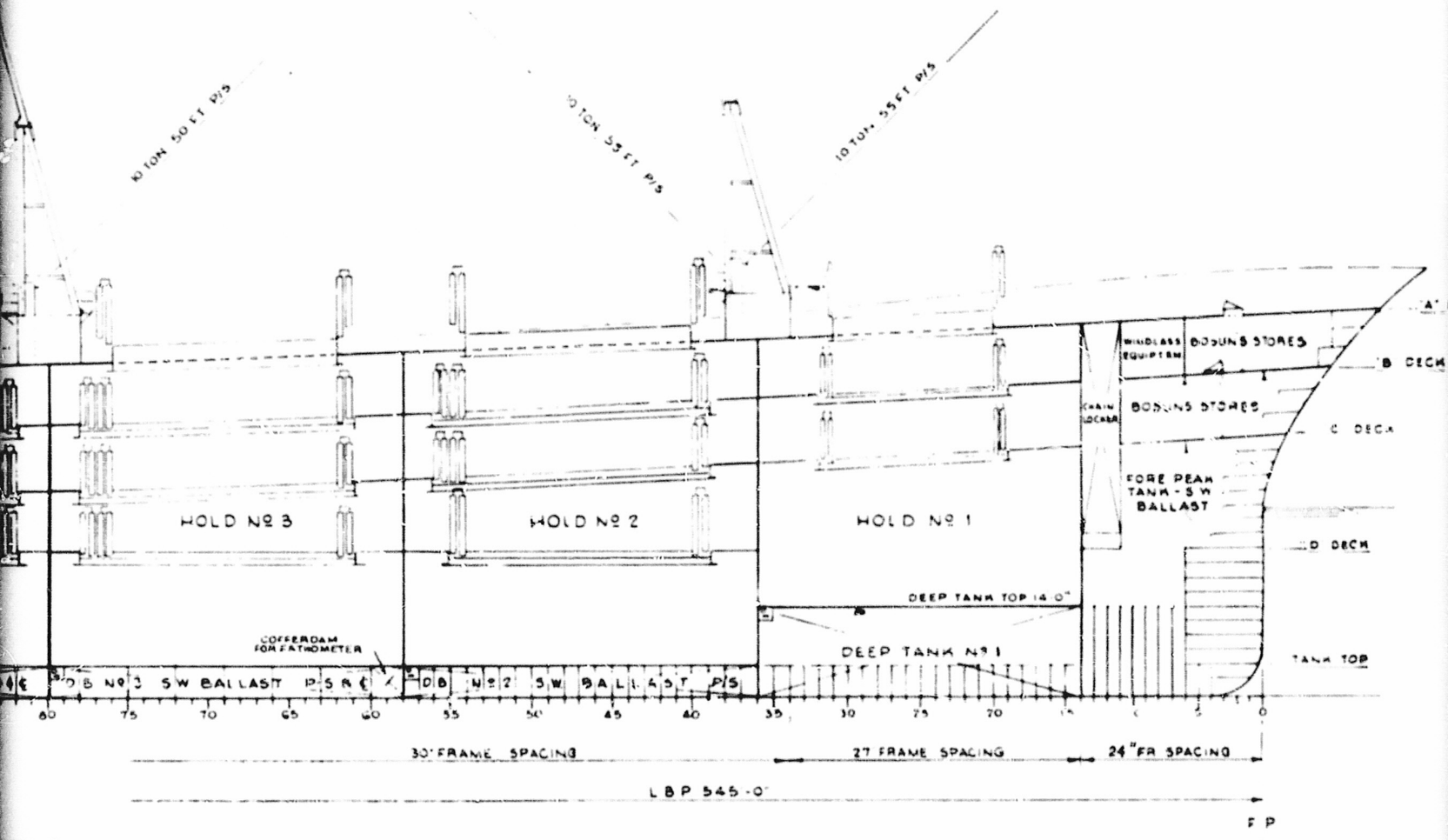
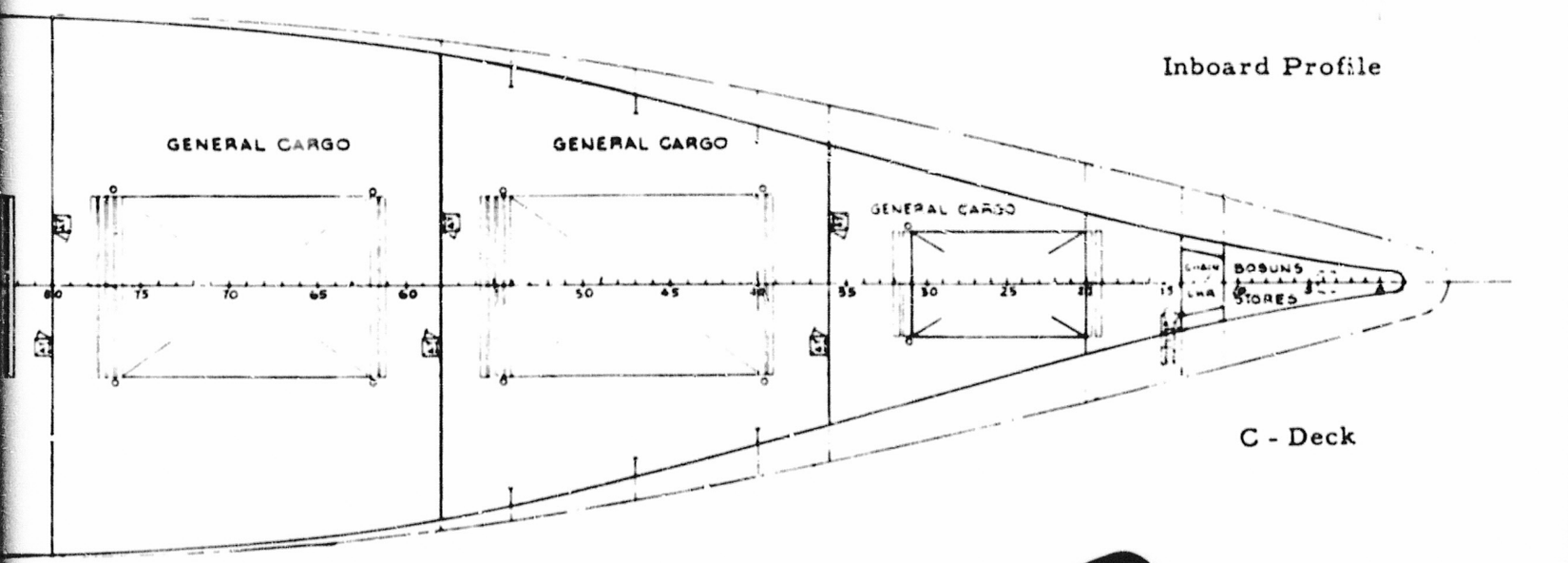
FIG. 62: INBOARD PROFILE AND "C" DECK PLAN



Inboard Profile



C - Deck

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FIGU

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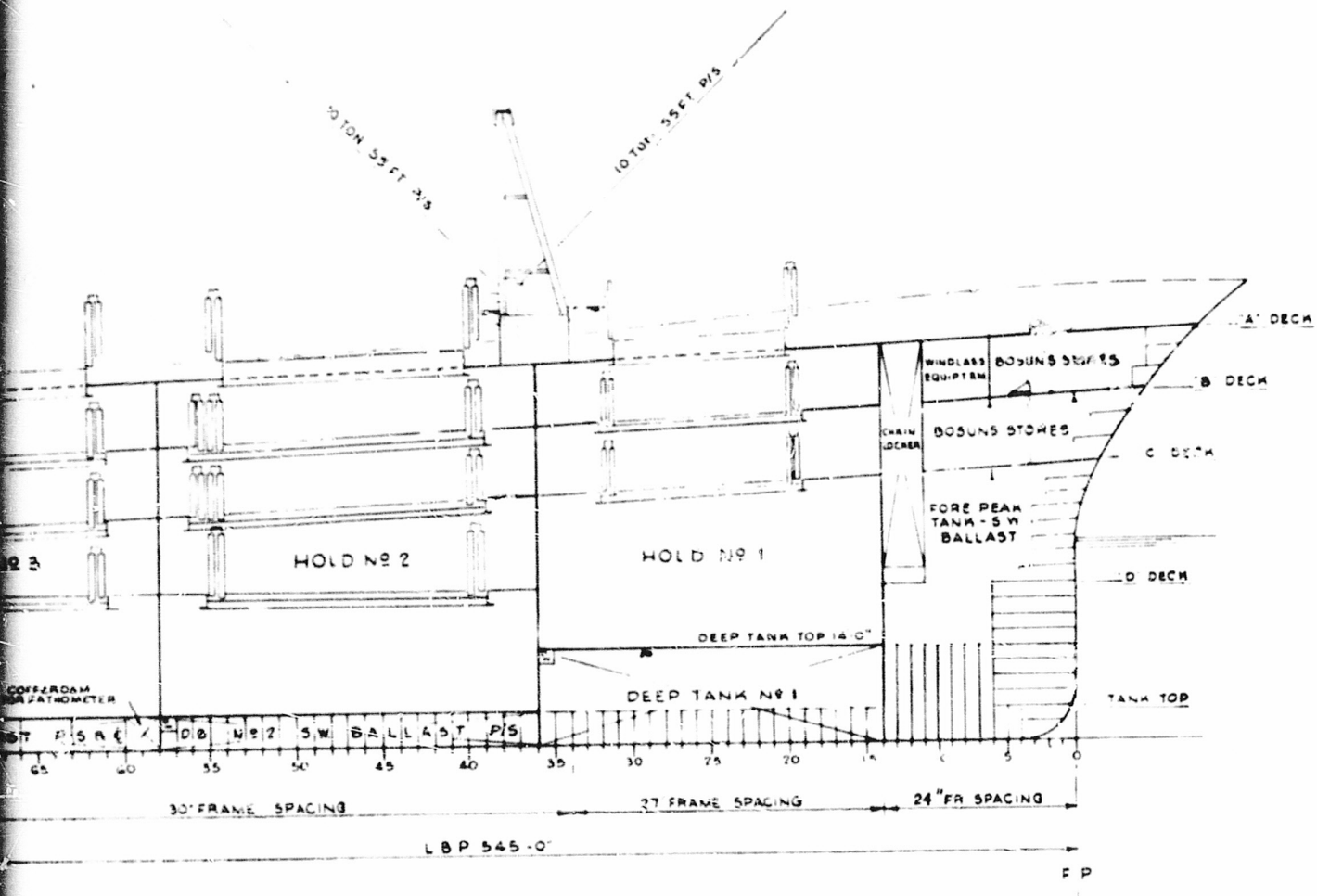
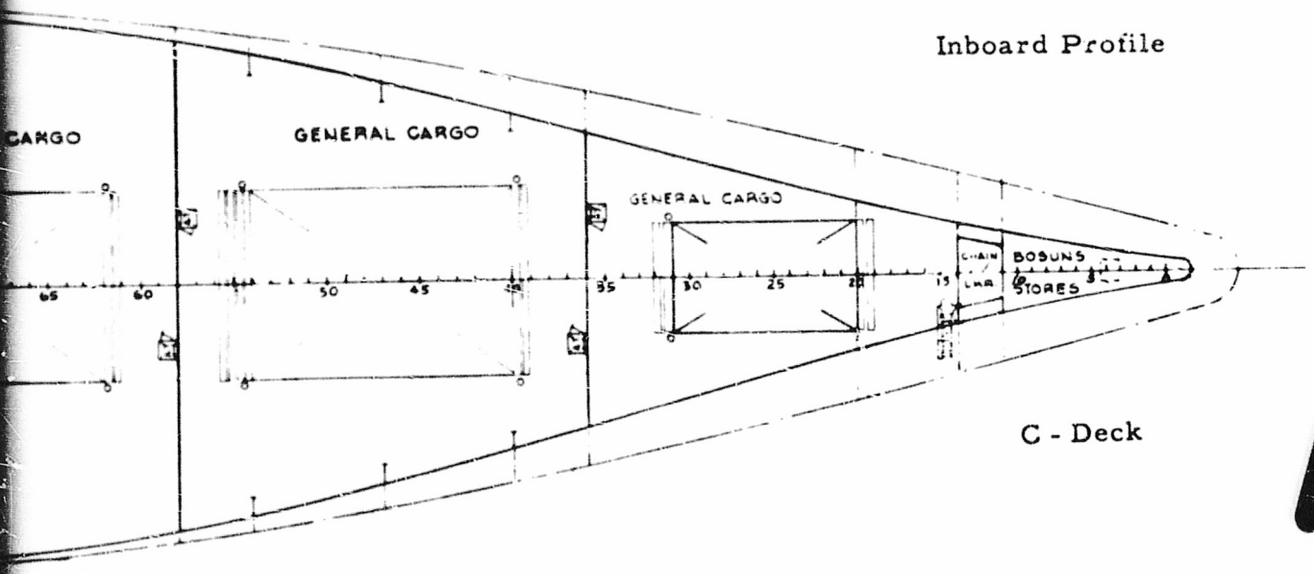
FIG. 62: INBOARD PROFILE AND "C" DECK PLAN



Inboard Profile



C - Deck

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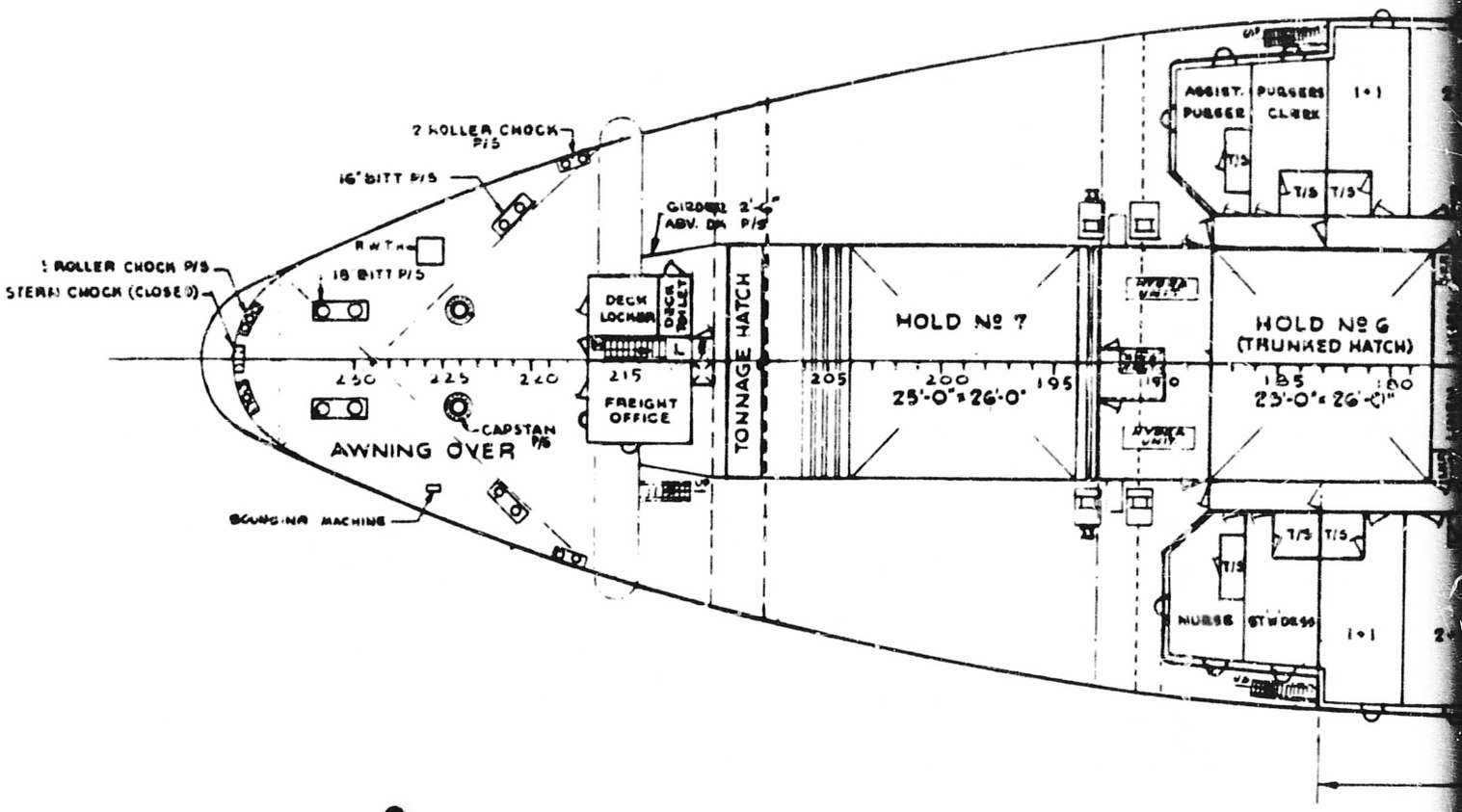
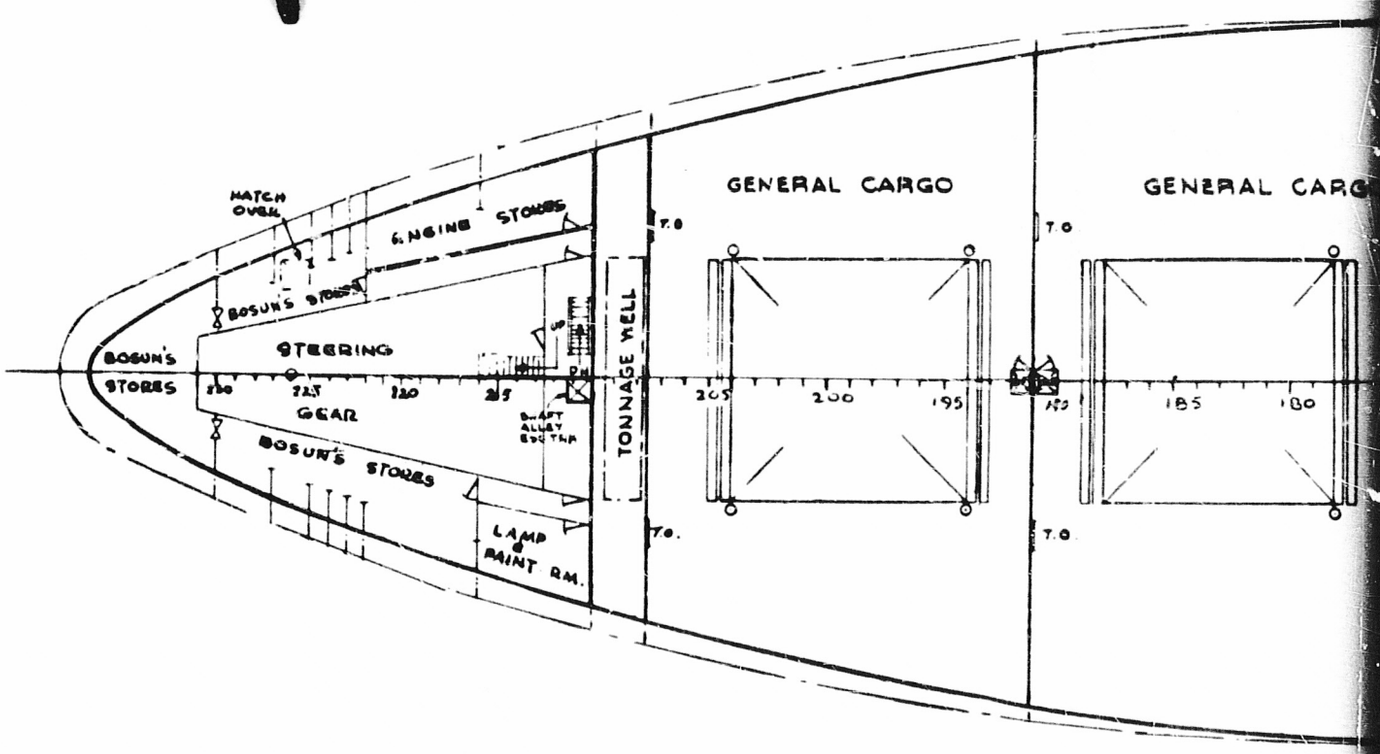
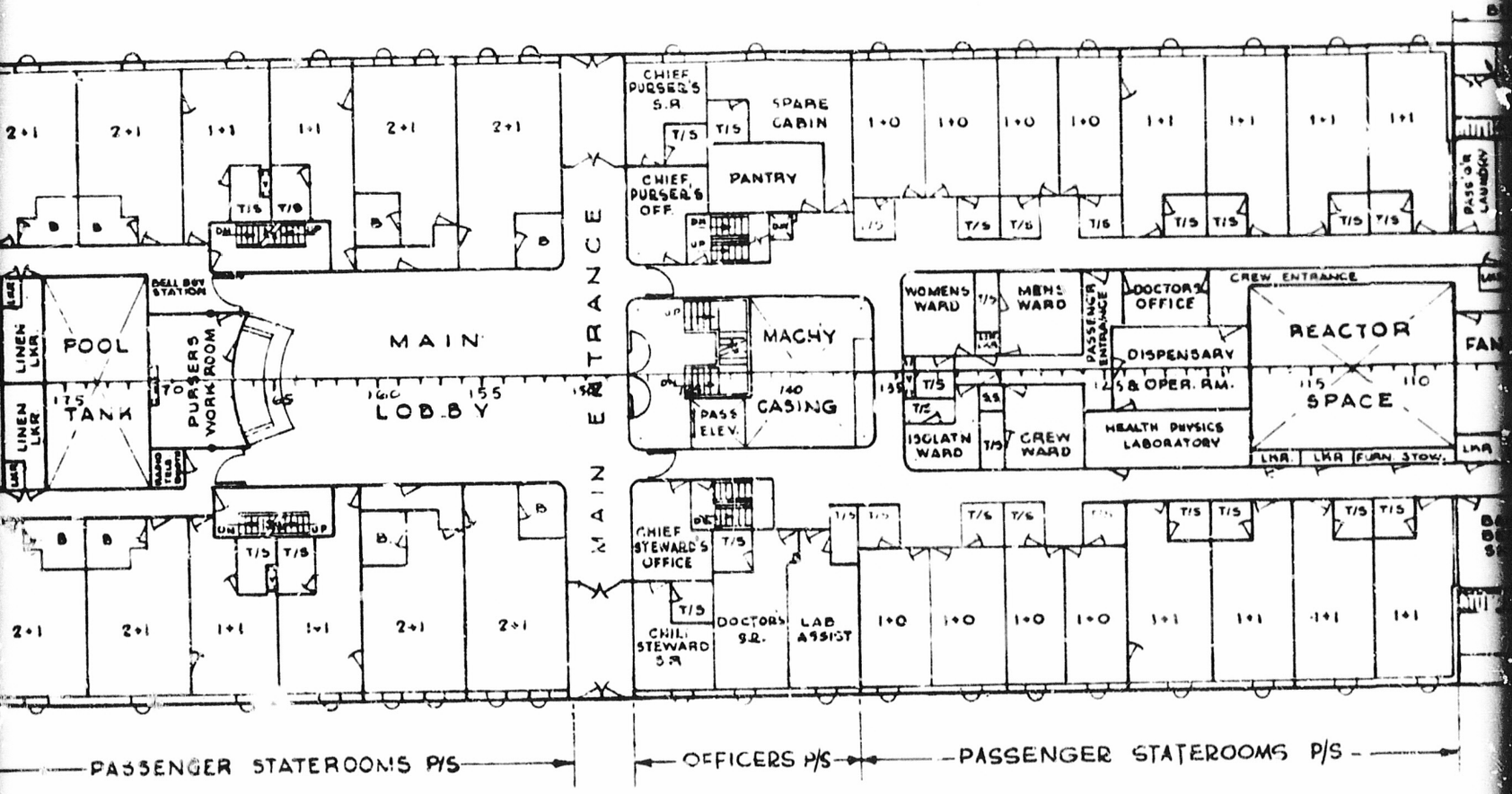
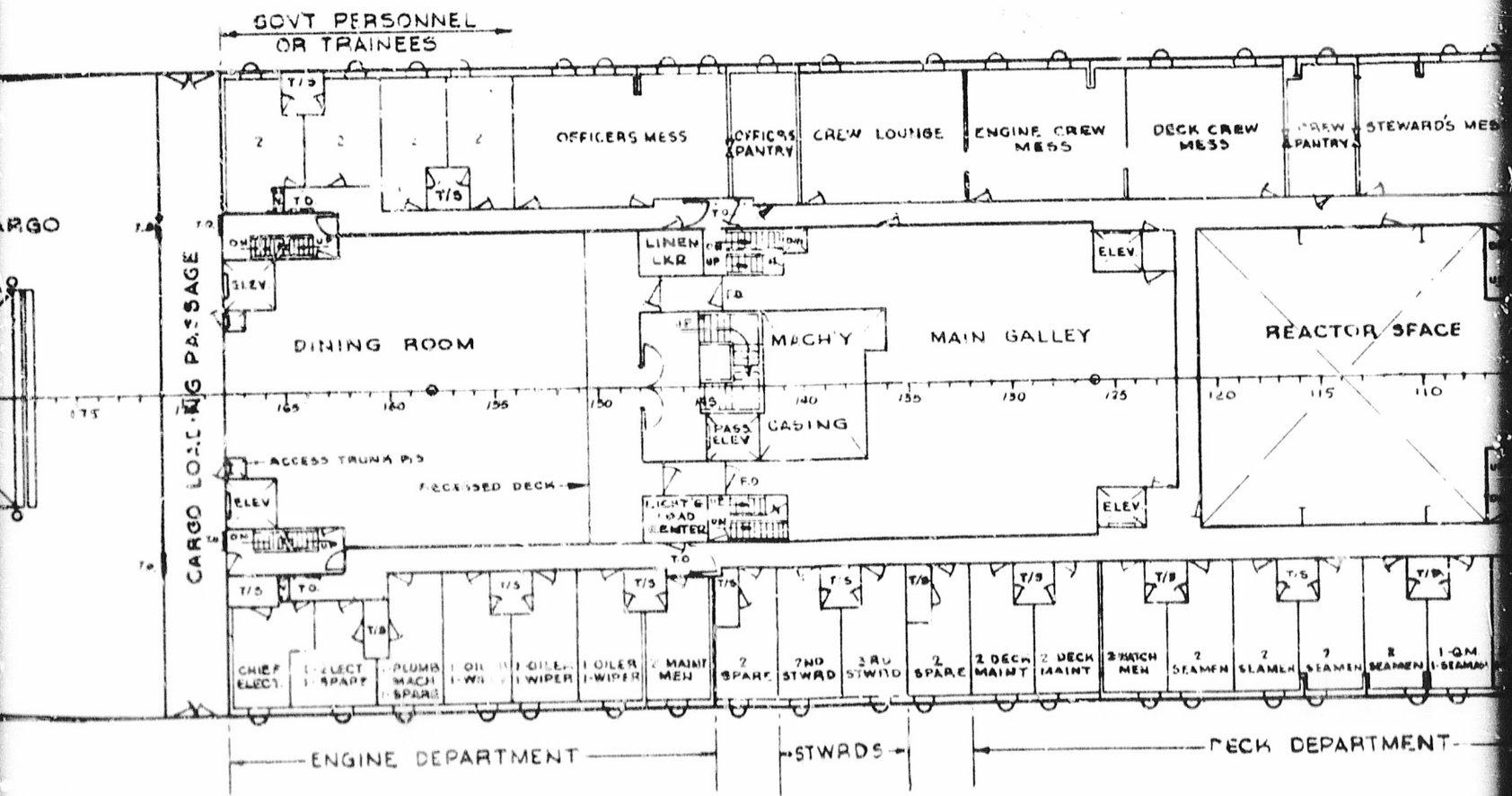
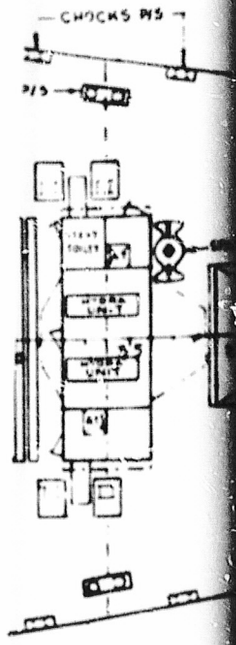
FIGURE 62


2 ROLLER CHOCK P/S
16" BITT P/S
R.W.T.
18 BITT P/S
1 ROLLER CHOCK P/S
STERN CHOCK (CLOSED)
GIRDER 2'-6" ABV. DK P/S
DECK LOCKER
DECK TOILET
FREIGHT OFFICE
TONNAGE HATCH
HOLD No 7
25'-0" x 26'-0"
HOLD No 6 (TRUNKED HATCH)
23'-0" x 26'-0"
AWNING OVER
CAPSTAN P/S
BOUNDING MACHINE
ASSIST. PURSER
PURSERS CLERK
NURSE
STWDESS
T/S
1+1
230
225
220
215
205
200
195
190
185
180


HATCH OVER
ENGINE STORES
BOSUN'S STORES
STEERING GEAR
BOSUN'S STORES
BOSUN'S STORES
SHAFT ALLEY ESC TRK
LAMP & PAINT RM.
TONNAGE WELL
GENERAL CARGO
GENERAL CARG
T.O.
205
200
195
185
180


CHIEF PURSER'S S.R.
SPARE CABIN
CHIEF PURSER'S OFF.
PANTRY
PASS'R LAUNDRY
BELL BOY STATION
POOL
LINEN LKR
TANK
PURSERS WORK ROOM
MAIN LOBBY
MAIN ENTRANCE
MACHY
PASS ELEV.
140 CASING
WOMENS WARD
MENS WARD
PASSENGER ENTRANCE
DOCTORS OFFICE
CREW ENTRANCE
DISPENSARY
REACTOR SPACE
FAN
ISOLATN WARD
CREW WARD
HEALTH PHYSICS LABORATORY
LKR
FURN STOW.
CHIEF STEWARD'S OFFICE
CHIEF STEWARD'S S.R.
DOCTORS S.R.
LAB ASSIST
PASSENGER STATEROOMS P/S
OFFICERS P/S
PASSENGER STATEROOMS P/S


GOVT PERSONNEL OR TRAINEES
OFFICERS MESS
OFFICERS PANTRY
CREW LOUNGE
ENGINE CREW MESS
DECK CREW MESS
CREW PANTRY
STEWARD'S MESS
LINEN LKR
DINING ROOM
MACHY
MAIN GALLEY
REACTOR SPACE
ELEV
PASS ELEV
CASING
ACCESS TRUNK P/S
RECESSED DECK
CARGO LOADING PASSAGE
CHIEF ELECT.
1 ELECT 1 SPARE
PLUMB MACH 1 SPARE
1 OILER 1 WPR
1 OILER 1 WIPER
2 MAINT MEN
2 SPARE
2ND STWRD
3RD STWRD
2 SPARE
2 DECK MAINT
2 DECK MAINT
2 WATCH MEN
2 SEAMEN
2 SEAMEN
2 SEAMEN
2 SEAMEN
1 QM 1 SEAMAN
ENGINE DEPARTMENT
STWRDS
DECK DEPARTMENT



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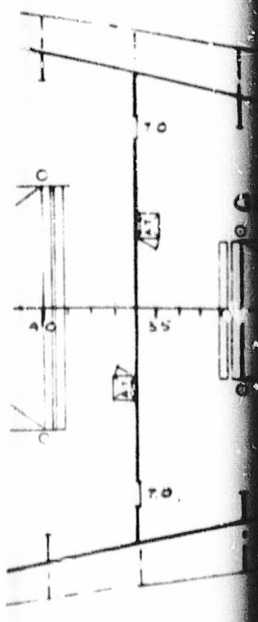

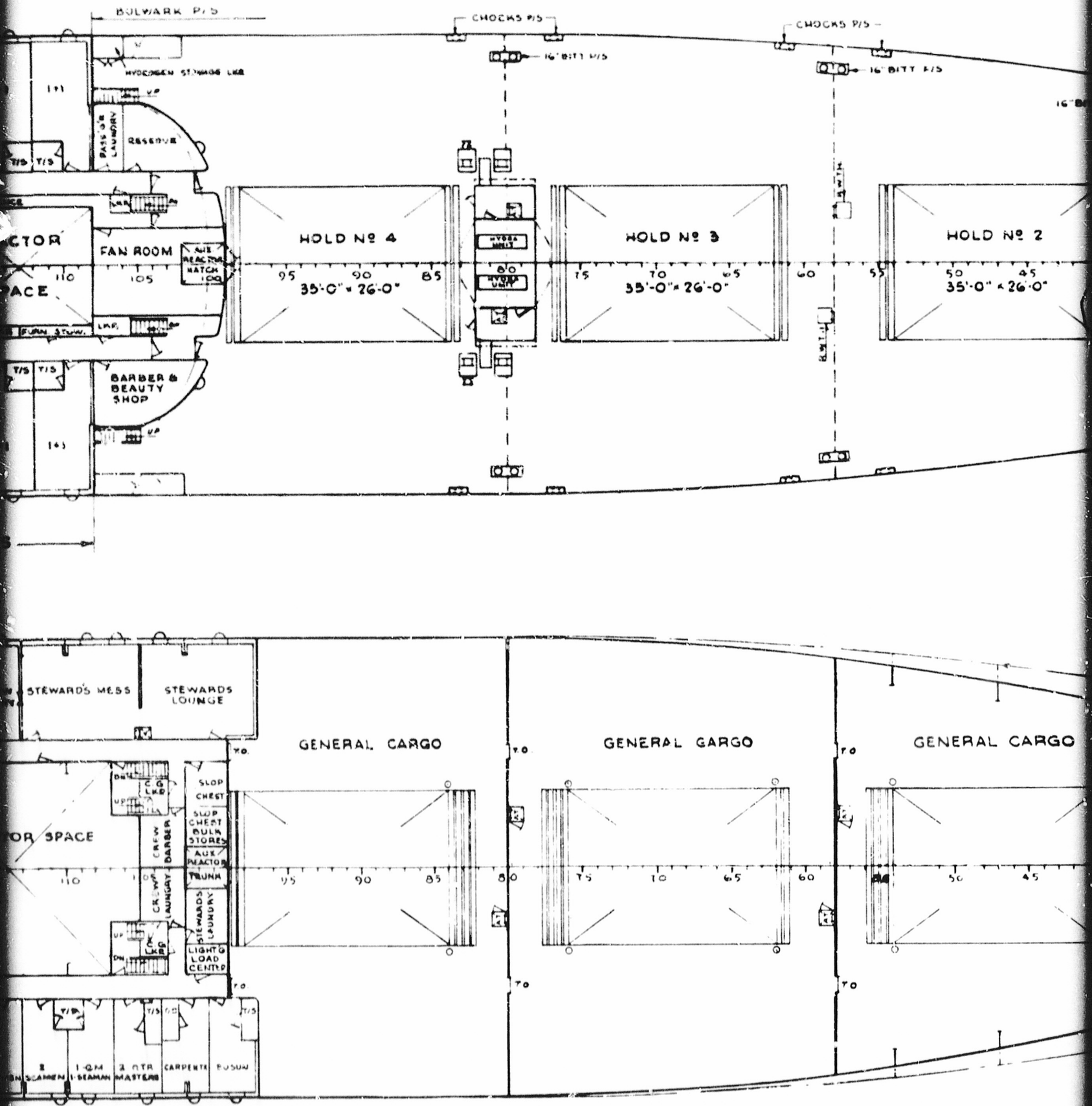
BULWARK P/S
CHOCKS P/S
CHOCKS P/S
16" BITT P/S
16" BITT P/S
HYDROGEN STORAGE LKR
RESERVE
PASS'R LAUNDRY
FAN ROOM
AIR REACTOR HATCH
HOLD Nº 4
HOLD Nº 3
HOLD Nº 2
35'-0" x 26'-0"
35'-0" x 26'-0"
35'-0" x 26'-0"
BARBER & BEAUTY SHOP
STEWARD'S MESS
STEWARDS LOUNGE
GENERAL CARGO
GENERAL CARGO
GENERAL CARGO
SLOP CHEST
SLOP CHEST BULK STORES
AUX REACTOR TRUNK
CREW BARBER
CREW LAUNDRY
STEWARDS LAUNDRY
LIGHT & LOAD CENTER
2 SEAMEN
1 QM 1 SEAMAN
2 QTR MASTERS
CARPENTER
BOSUN

3

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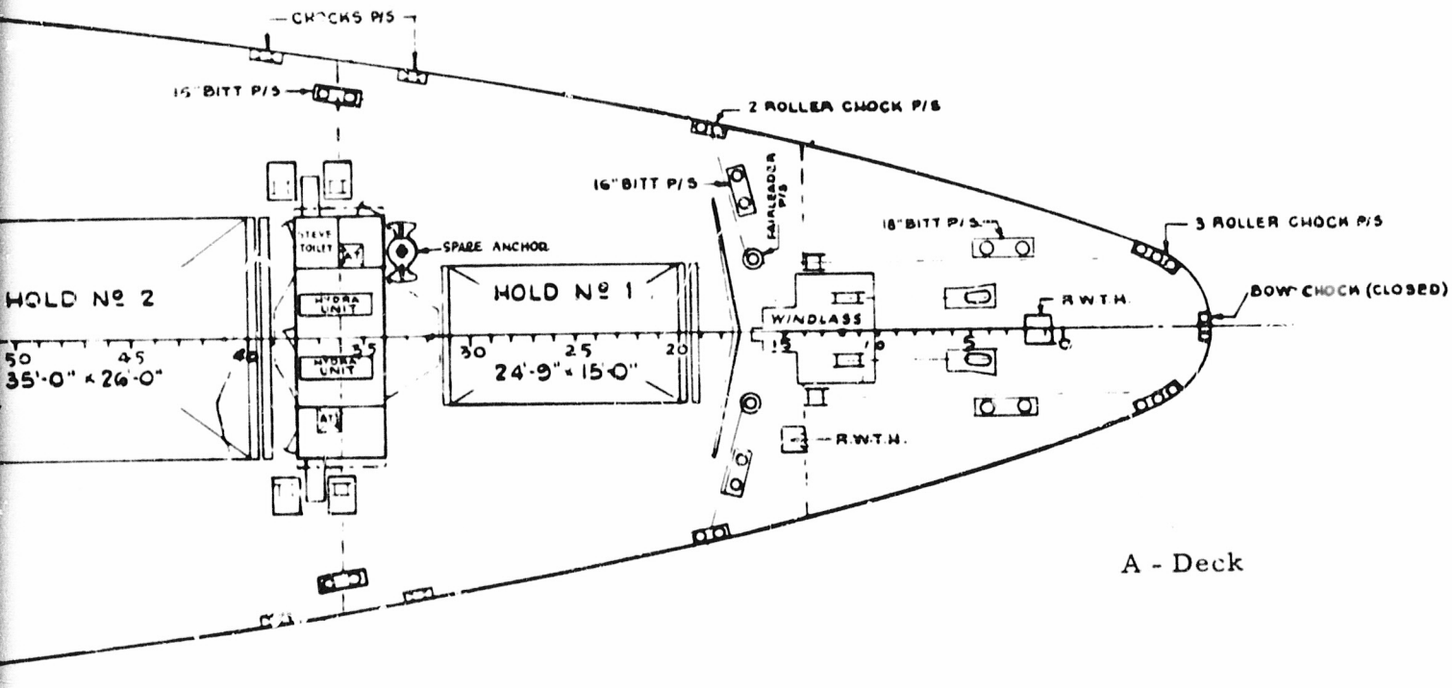
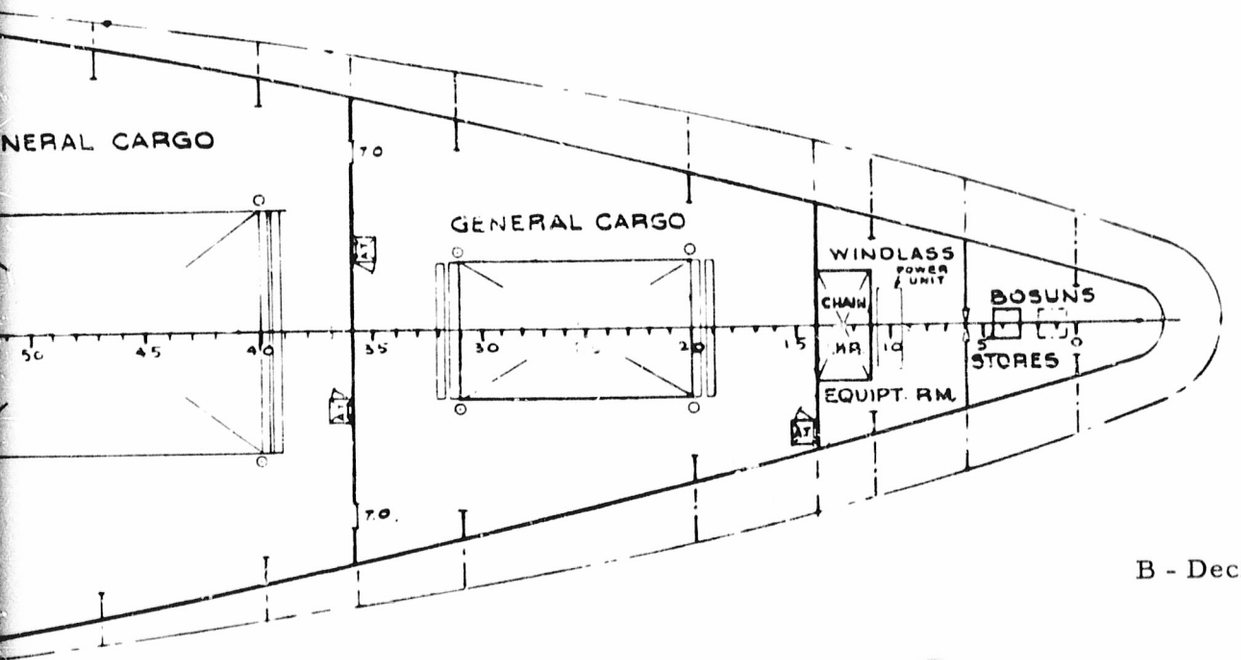
FIG. 63: "A" AND "B" DECK PLANS



A - Deck



B - Deck

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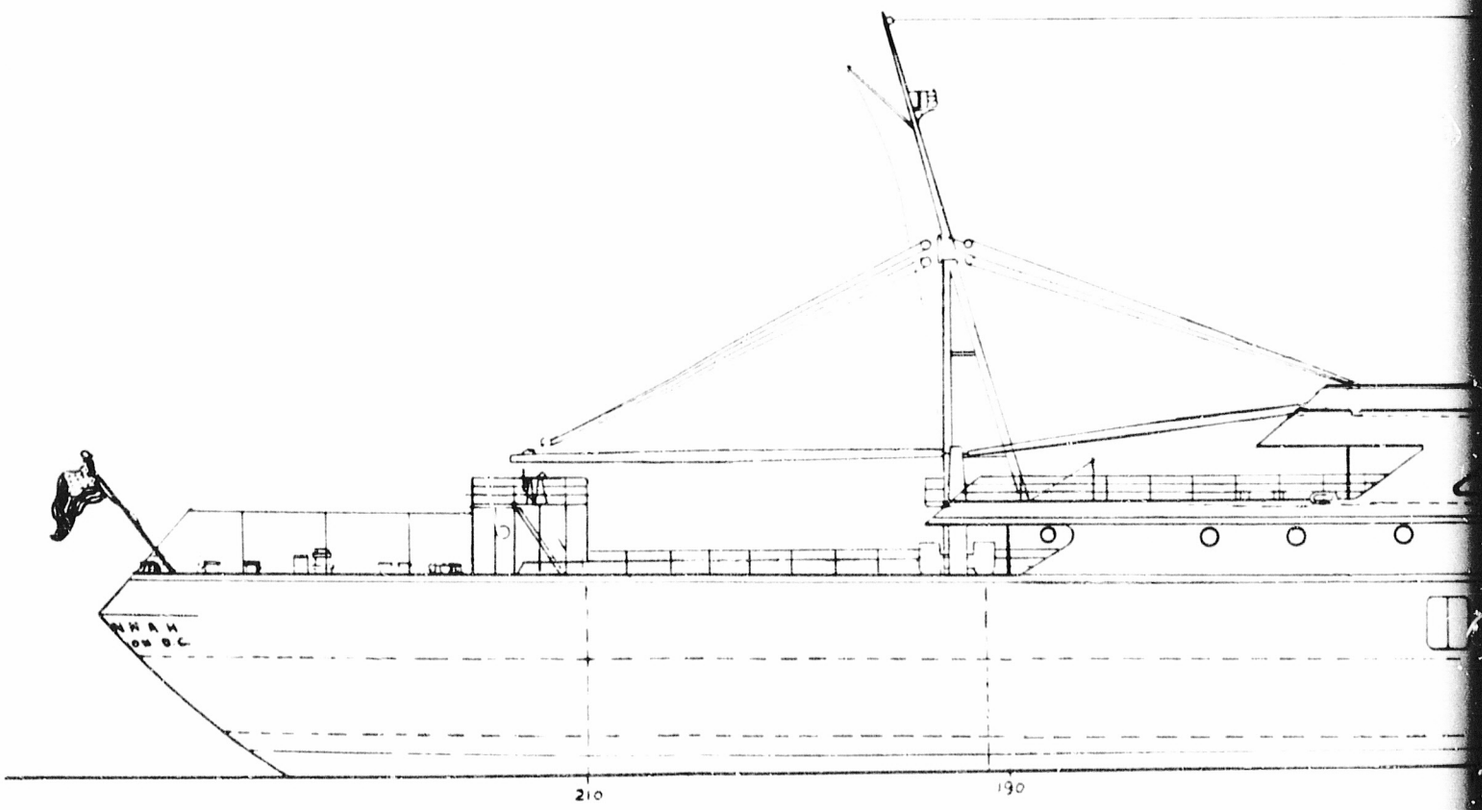
FIGURE 63


210
190

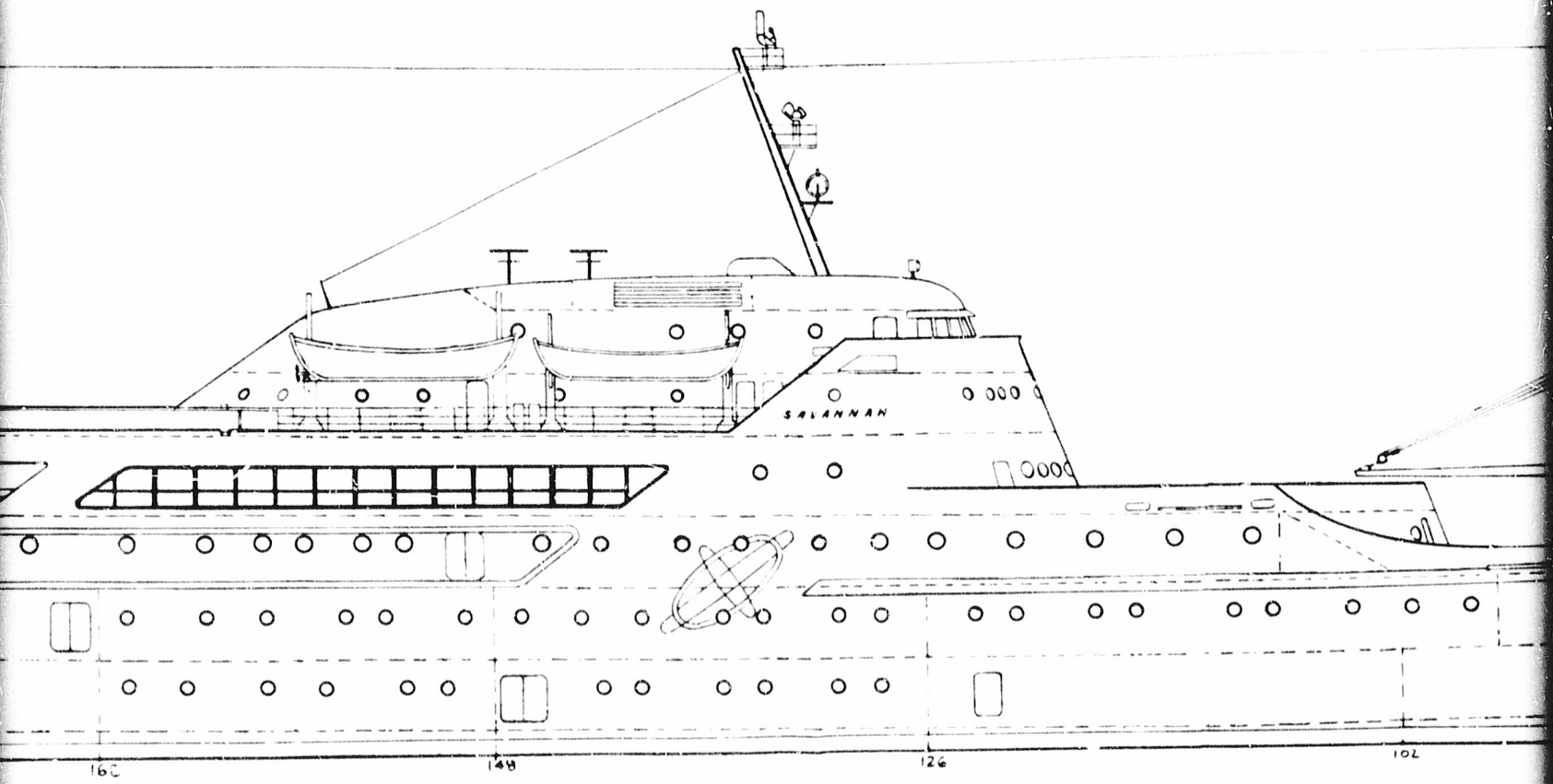


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FIG. 64:

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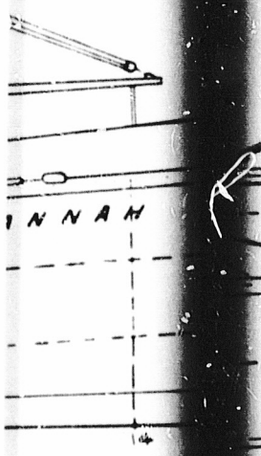

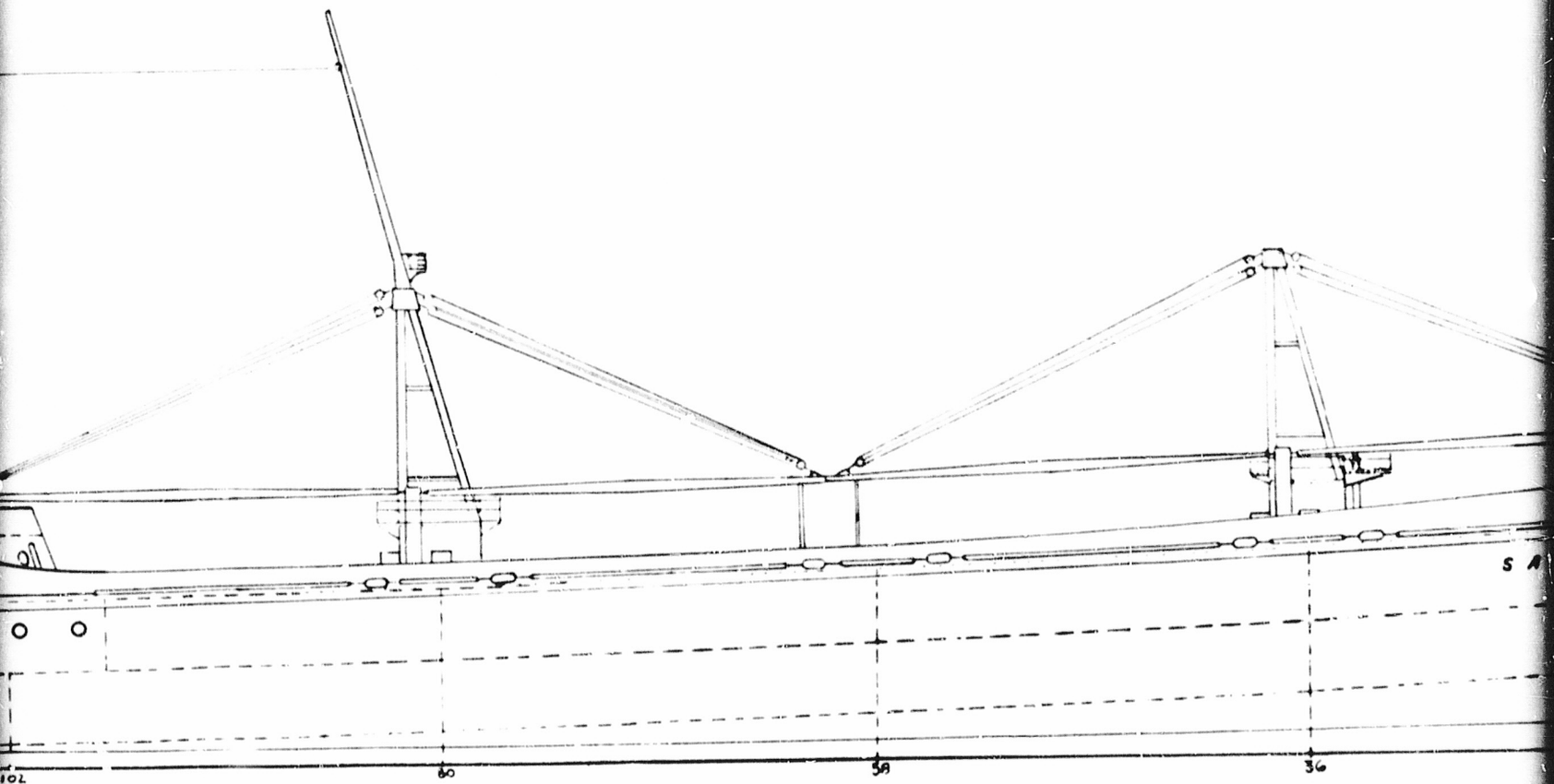

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George G. Sh

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FIG. 64: OUTBOARD PROFILE

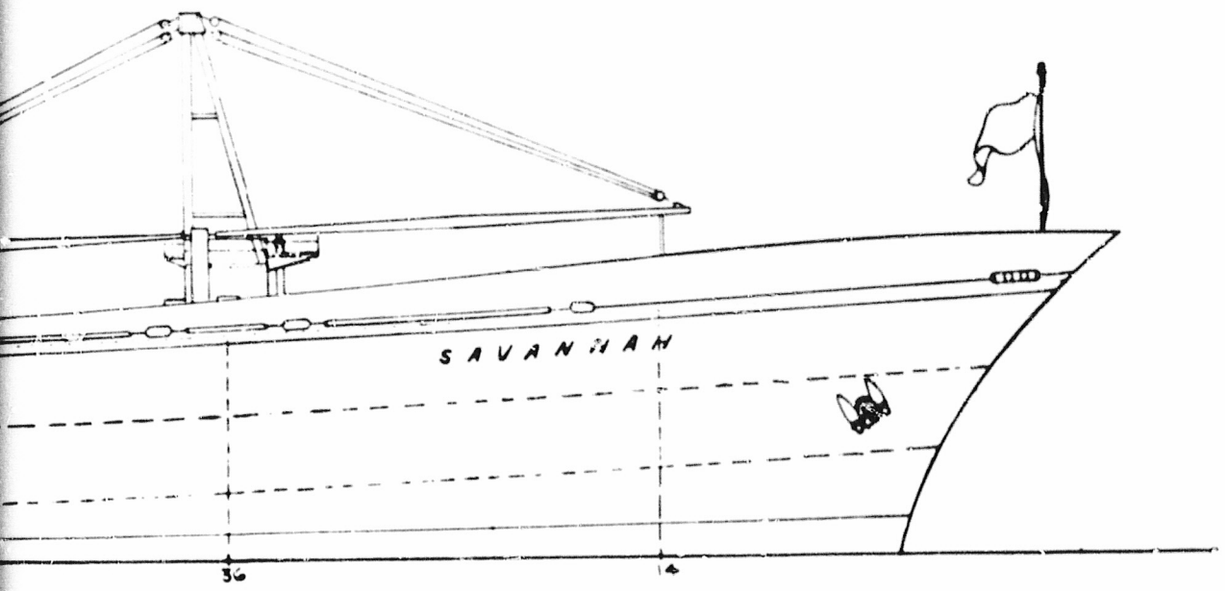

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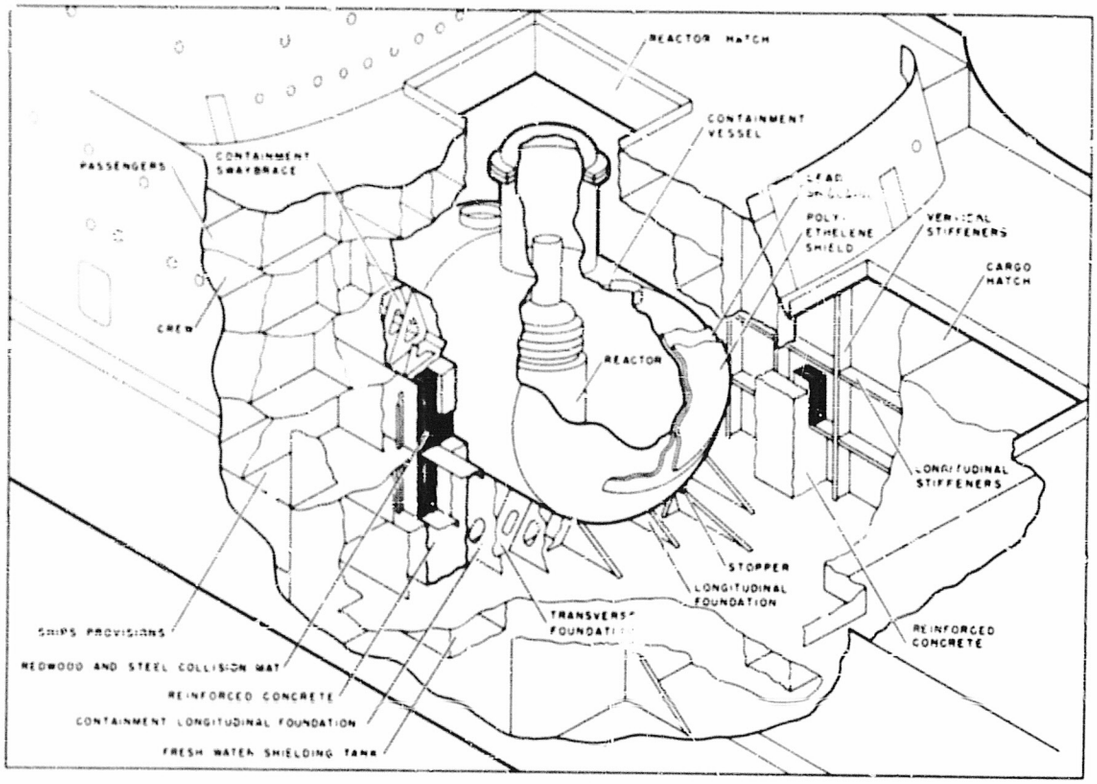
FIGURE 64

FIG. 65: CUT-AWAY VIEW OF NS SAVANNAH SHOWING STRUCTURE



The Babcock & Wilcox Co.
Atomic Energy Division

FIGURE 65

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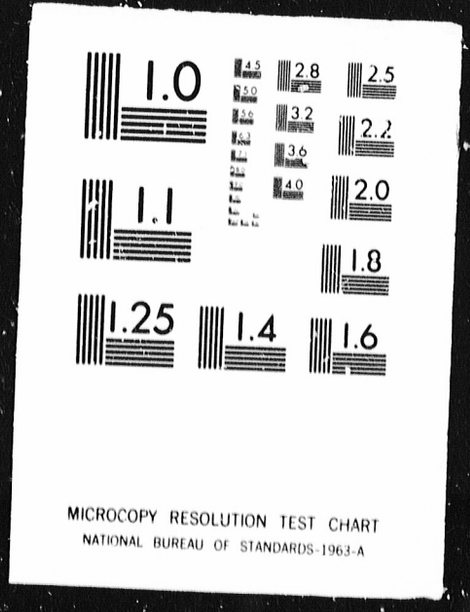

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