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CONSOLIDATED NUCLEAR STEAM GENERATOR FOR MERCHANT SHIP APPLICATION

A Conceptual Design

August 1962

The Babcock and Wilcox Company
Atomic Energy Division
Lynchburg, Virginia

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REACTOR TECHNOLOGY

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August, 1962

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Submitted to
THE UNITED STATES ATOMIC ENERGY COMMISSION
By

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Atomic Energy Division
Lynchburg, Virginia

CONTENTS

CONTENTS (Cont'd)

List of Tables

Tables (Cont'd)

List of Figures

Figure

The Following Figures Follow Page 67

Figures (Cont'd)

Figures (Cont'd)

INTRODUCTION AND CONCLUSIONS

Introduction

As a part of the advanced planning by the Maritime Reactors Branch of the AEC, the Babcock & Wilcox Company (B&W) was requested to investigate new pressurized water reactor concepts for shipboard application.

It was felt that the experience gained by B&W in the design, construction, and testing of the NS Savannah, along with new ideas developed by B&W and other agencies, would lead to an improved design, which would permit a nuclear powered merchant ship to be constructed at a cost close to that of a conventional merchant ship. However, as the government stated, the ultimate goal (the production of a design competitive with conventional designs) might require the construction of at least three generations of nuclear powered ships.

The program began early in 1960 with a review of proposed designs to establish the significant capital and operating costs. This review indicated that capital and operating costs could be reduced by:

1. Simplifying the primary system.
2. Using less expensive construction materials.
3. Using a soluble poison for lifetime reactivity control.
4. Reducing the primary system pressure.
5. Generating slightly superheated steam (70 F).
6. Improving the design of the core and the control system.

As the work progressed and more information became available from tests on the Savannah's power plant, it became evident that the auxiliary systems would have to be simplified if costs were to be reduced significantly. Subsequent work directed towards this goal led to a new concept: a compact integral boiler reactor fully submerged within a flooded containment tank (defined in this report as a Consolidated Nuclear Steam Generator).

In the integral boiler reactor, over-all space and weight requirements are reduced by consolidating the entire primary system (including pressurizer, steam generators, and pumps) into one pressure vessel. This concept originated under B&W Company sponsored development program, was incorporated in the Upgrading Program after B&W established its technical feasibility. The flooded containment concept, suggested by the AEC, combines secondary shielding, waste control, emergency decay heat removal, and containment (pressure suppression) functions, and eliminates secondary concrete, waste tanks, and the drain collection system and tanks.

Since incorporating the integral boiler reactor concept, the main effort has been to bring together the foregoing features into a workable design for a Consolidated Nuclear Steam Generator (CNSG). Although all of these features would probably have to be placed in a nuclear power plant to achieve a status competitive with conventional plants, it was decided to complete a current-status design first to enable a comparison for determining the competitive position reached by nuclear power.

This report presents the present conceptual design, which can be built today with only a limited amount of confirming research and development. Also covered are the work remaining to be done and the alternate features that would be incorporated in a potential power plant as incentives for the further development of a nuclear powered merchant fleet.

Conclusions

This study has shown that the Pressurized Water Reactor (PWR) concept may be further simplified and improved economically. Specifically, the Consolidated Nuclear Steam Generator (CNSG) design of a PWR shows for the first time that a nuclear power plant can be competitive in merchant marine application on an unsubsidized basis. Since this design depends largely on the existing PWR technology, its feasibility can be established easily, and a detailed design of such a power plant for a second nuclear powered ship is now warranted. The study further concluded that a prototype was neither required nor desirable. For the first time, calculated operating costs for the nuclear powered ship, on an unsubsidized basis, equals those of a conventionally powered ship. These costs were obtained for a power plant of marketable size (27,000 shp) in a cargo ship operating on a long essential trade route with a large number of monthly sailings. On a subsidized basis, the nuclear ship will require an additional $900,000 approximately to have equal operating costs.

Also, the following general conclusions may be reached. Unless the United States intends to eliminate its merchant marine, higher speed ships with more mechanization and containerization must be utilized to compete in world trade. As higher performance ships are built, nuclear power, with its lower fuel costs and less frequent refueling requirements, will be more economical than fossil fueled plants.

A mistake in past studies has been the attempt to place nuclear power plants into existing ship designs. The potentials of nuclear power can be fully realized only when its inherent characteristics (lower fuel costs at high power and no fuel storage requirements) are factored into the original design of a ship. Designs of a prototype ship utilizing advanced cargo handling techniques and higher speeds are now being developed, and nuclear power can be incorporated easily now in this new over-all design effort.

Based on the above conclusions, B&W recommends that the detailed design of a nuclear power plant for a second ship be started as soon as possible (preferably in the fiscal year 1963). If there is a delay before such design can be initiated, a limited effort should be permitted on the improvement, refinement, and research and development for the reference design submitted in this report. Since better power plants will always be necessary in this internationally competitive field, this study should incorporate, to the extent possible, the advanced features of the goal reactor design submitted here.

1. ECONOMIC PERSPECTIVE

Shipping is probably the most important means of transporting goods on an international basis. Since world-wide economy and trade are expanding, shipping requirements are increasing.

Expanding markets tend to break down inhibiting trade barriers. In Europe, low tariffs facilitate commerce. To compete with the European markets, it is necessary for the United States to seek further reciprocal tariff cuts.

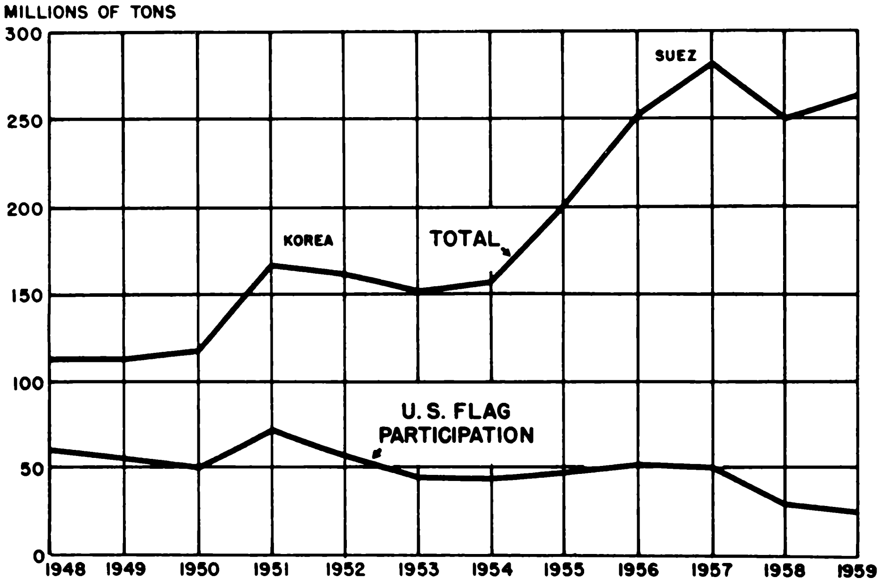
Although economical transportation is important, reducing costs is not enough. In many instances, slow transportation has been replaced by faster and more flexible conveyances at a sacrifice of higher costs. Often, a manufacturer is willing to pay extra if his goods will reach their destination quickly. The speed of ship transportation relates to the entire operation including cargo handling and distribution.

Sea transportation is economical, but it is comparatively slow. As a result, a manufacturer in the United States tends to ignore the European market in favor of the home market, where his goods may be sent quickly by rail.

In spite of expanding international trade, the American Merchant Marine is declining. (See the graphs on page 6.) Although America is the world's greatest industrial nation, her shipbuilding ranks eighth in terms of commercial ship production.

At present, the lack of vigor in the American shipping industry is usually blamed on high costs. However, as far back as 1830 when American shipping was at its peak, the cost of building and operating American ships was higher than in Europe. But through hard work, continuous technological improvement, and imaginative practices (for example, establishing regularly scheduled packets) America managed to maintain a competitive lead until Europe took the technical lead by adopting iron ships. America could regain its lead in shipping if the same competitive drive and progressive attitude existed today as was present more than a century ago.

Decline of American shippings' share of world trade . . .

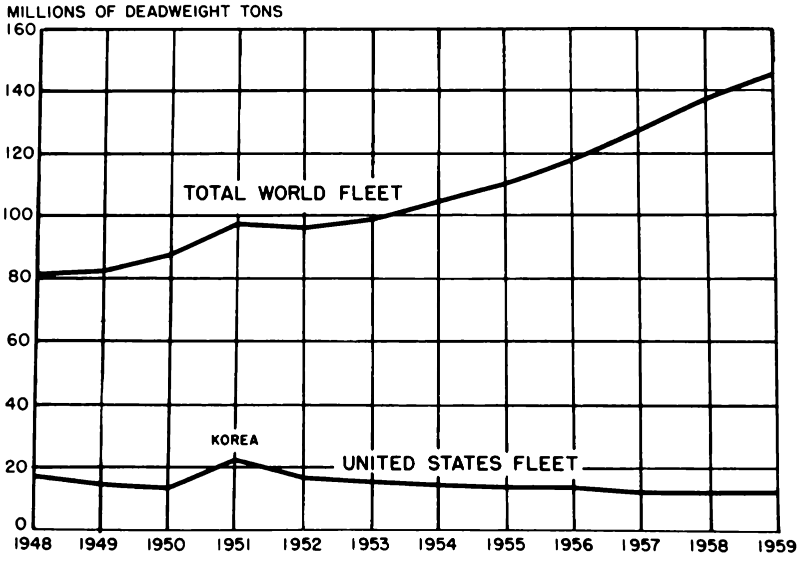


1948	1949	1950	1951	1952	1953	1954	1955	1956	1957	1958	1959
52.7	48.1	42.6	42.9	35.6	29.5	27.8	23.6	20.5	17.8	11.7	9.7

PERCENTAGE OF TOTAL - U. S. FLAG

Source: Martime Administration

reflected in shrinking of active U. S.-flag fleet



NUMBER OF SHIPS

	1948	1949	1950	1951	1952	1953	1954	1955	1956	1957	1958	1959
World Fleet	10,648	10,699	11,213	12,228	12,078	12,256	12,680	13,066	13,649	14,266	14,892	15,154
U. S. Fleet	1,415	1,212	1,099	1,955	1,408	1,234	1,142	1,072	1,097	983	960	939

Source: Maritime Administration

Marine Engineering/Log

June 15, 1961; Vol. LXVI, No. 7

A profitable, competitive American Merchant Marine can be an effective instrument for promoting national policy. Under special circumstances, national defense may require American controlled shipping to promote trade as Russia is now doing in the Cuban situation. An efficient merchant marine is needed not only for its direct effect, but also to promote technical progress in general.

The Merchant Marine Act of 1936 and its supplements are attempts to provide an effective American Merchant Marine through the use of subsidies and limited research and development. This act has helped, but subsidies stifle initiative and offer a poor incentive for improvement. Since subsidy costs have increased from 50 million dollars in 1950 to 260 million dollars in 1960, it appears that a different approach is needed by the United States.

Technical and managerial improvements must be made to enable the American Merchant Marine to compete with other nations and to improve the efficiency of the world-wide transportation system. All phases of American shipping must be improved. Improved scheduling, reduced red tape, and an increase in over-all speed are especially important.

Improvements in cargo handling are also important. Due to labor laws and high crew wages in America, cargo handling is a managerial as well as a technical problem. Many potential improvements may be made to the ship's machinery and to the methods of automation.

To meet these previously described objectives, the best vessel is a high-speed dry cargo ship with advanced cargo handling equipment. Due to the characteristics of this type of ship, nuclear power will be a definite asset. Since the horsepower will be high, low fuel costs and the absence of fuel weight and volume are important. Also with nuclear power, the disadvantage of the capital cost will be less.

Since the Savannah's power plant design was completed about four years ago, great advances have been made in nuclear power technology, as is shown by several reactor plant studies sponsored by the Maritime Administration and by a gas-cooled reactor experiment for ship propulsion under construction by the AEC.

Foreign countries are on the verge of building and operating nuclear powered merchant ships. If the American shipbuilding industry does not advance rapidly, the technical lead in this important field will be lost to other nations.

Section 5 of this report shows that the CNSG design is competitive with conventional power in a 23-knot cargo ship with advanced cargo handling. Therefore, it is concluded that, under present technology, nuclear power is the best choice for higher speed cargo ships essential in a competitive American Merchant Marine of the future.

2. GOAL AND REFERENCE DESIGNS

The CNSG delivers 20,000 shaft horsepower normal and 22,000 maximum. This horsepower, selected by B&W and the AEC, permits direct comparison with NS Savannah and other studies and approaches the normal shaft horsepower of ships now being built under the Maritime Administration's 20-knot replacement program. Average normal shaft horsepower for these ships is approximately 17,000.

The CNSG is an advanced pressurized water reactor developed for maritime application. Each of the advanced features listed below is technically sound and is included in a current-status reference design.

2.1. IBR

Simply stated, the IBR is a "package" concept; that is, the entire primary system is consolidated into a single pressure shell. This concept can be developed in different ways depending on the end use of the reactor system. For example, it can be made tall and narrow or short with a larger diameter. The reactor plant requires relatively little space — an important consideration for shipboard application. Also, the compact primary system permits a significant reduction in shielding weight. An IBR can be designed and built by present techniques.

2.2. Self-Pressurization

Self-pressurization involves the operation of a pressurized water reactor with a steam space in the reactor vessel (instead of a separate pressurizer) to absorb primary coolant density changes.

The steam in the reactor vessel is in thermal equilibrium with the water in the reactor. Therefore, the pressure in the primary system is the saturation pressure corresponding to the temperature at the steam-water interface. In the CNSG design, this is the core exit temperature. This system eliminates the pressurizer and its heaters (a large complicated component). The self-pressurized concept has been analyzed extensively. These analyses indicate that it may be incorporated with complete confidence into a current reactor plant and the system can be designed for a load following without control rod motion.

2.3. Once-Through Steam Generator

In this generator, the water makes one passage and emerges as steam. There is no device to separate water and steam; superheated steam is produced.

This steam generator reduces space and weight requirements and produces cost savings because the steam drums, the risers, and the downcomers are eliminated. It also permits operations with constant steam pressure, which permits the steam system to be designed for a lower pressure. If a natural circulation steam generator is used with a pressurized water reactor, the steam pressure rises at the low loads.

Since several once-through steam generators, including two for nuclear service, have been built by B&W, the feasibility of this feature is established.

2.4. Wet Containment

Several improvements result from filling the containment vessel with water. The containment can be designed for low pressure since the water will condense the flashing steam and will absorb the energy from a primary system rupture without any appreciable buildup in pressure. Also, the water cools the components (such as pumps and control rod drives), removes the decay heat (Item 2.5), and serves as a waste disposal tank under certain conditions.

All these functions can be based on current technology except energy absorption from a primary system rupture. Research concerning energy absorption is proceeding at the Alliance Research Center. The results obtained are encouraging, and the concept appears practical.

2.5. Decay Heat Removal Through the Pressure Vessel Shell

Decay heat removal is a major problem in a nuclear reactor design. It is possible for a ship to be stranded. Although this accident may not be considered credible, it is possible to provide protection for a period of days without unduly affecting the over-all plant cost. An amount of heat equal to decay heat four hours after shutdown is transferred continuously from the reactor vessel through the vessel shell to the containment water. A layer of lead bonded to the reactor vessel controls the amount of heat transmitted and provides shielding.

This method of removing heat from the reactor vessel is considered infallible because it does not depend on machine or manual operation. Heat removal is provided for two days with the ship on dry land and indefinitely with the ship in the water. Some heat is lost from the reactor at all times, but its economic value is small.

The present design depends on lead bonded to the reactor vessel wall. Lead bonded to steel is a regular commercial practice, but it must be demonstrated for the temperature and geometry of the CNSG vessel.

2.6. Simplified Auxiliary Systems

The auxiliary systems have been simplified (compared to conventional PWR plants) by combining functions, by using components which do not require auxiliary systems, by using the wet containment, and by reducing system functional requirements judiciously.

2.7. Soluble Poison Control

The initial excess reactivity necessary to permit high uranium burnup and isotope buildup in the CNSG is controlled by a neutron poison (boric acid) dissolved in the primary water. Soluble poison control was used because:

1. A flatter flux distribution can be achieved with soluble poison than with control rods. This permits a smaller core to be used because of the more even heat distribution. It also permits higher average fuel burnup since the limitation on burnup is at the point of the maximum power generation.

2. The reduction in the number of control rods and mechanisms (from about 21 to 7) results in considerable capital cost savings.

3. Space in the pressure vessel can be utilized more efficiently since the space taken up by control rod extensions is reduced.

4. Design of the core and reactor internals is simplified.

Boric acid is used for reactivity control to cool some reactors (e.g. Yankee) to room temperature, and is planned for many plants presently being designed and built, but data on its use in strong radiation fields is limited.

2.8. Low Pressure and Temperature Design

Reactor operating temperature and pressure (520 F, 812 psia) produce favorable component costs, core size, and nuclear material efficiency; but these effects are offset partially by decreased thermal efficiency.

Although previous experience shows that temperature generally has little effect on the total fuel cost, the temperature selected here leads to a low equipment cost.

During this study, a great many potential improvements in consolidated closed-cycle water reactor designs were recognized, but time did not permit their incorporation. The examples below indicate areas that may be explored in the further improvement of the CNSG concept.

1. Improvements in Core Design — The core design presented in this report is based on a power peaking ratio of three (an assumption based on past experience). The nuclear calculations performed during the study actually showed a power peaking factor of 2.4.

   Since the allowable burnup of uranium in the reactor is based on the burnup at the highest point with a power peaking factor of three, the average burnup was 15,000 MW-d/ton. A peaking factor of 2.4 would permit an average burnup of 18,750 MW-d/ton, which would result in a longer core life and lower fuel costs.

   The core design presented here has a local peaking factor of 1.2. Improvements in the mechanical design of the core could reduce this factor to one. In this case, the average burnup could be 22,500 MW-d/ton. Because core thermal design was based on the calculated power peaking factor of 2.4 plus a safety factor to allow for the uncertainties due to shuffling, further analysis may allow more reduction in this factor and a still smaller core.

2. Reduction in Size and Weight — Changes in parameters, such as core outlet temperature, would permit a marked decrease in the steam generator surface and therefore, a decrease in the size and weight of the

CNSG plant. Also, a re-examination of the design may permit more heat to be extracted through the steam phase, which will result in better utilization of space within the reactor vessel. Mechanical design changes, such as different types of pumps and control rod drive mechanisms, may permit further size and weight reductions.

3. Elimination of Control Rods — In the CNSG design, control of reactivity due to shim and temperature defect is accommodated by adjusting the soluble poison concentration. Control rods are used for transients, emergency shutdown, and xenon transients. Analog studies of the Savannah have shown that control rods are not as necessary as once believed for emergency transients. Scram action is not necessary on the Savannah for any accident except a primary system rupture. This experience and analog studies made of the CNSG design indicate that, by proper design procedures, a core might be designed so that control rods were unnecessary. Such a design would include margins against burnout in transients, a selection of core parameters, such as length-to-diameter and metal-to-water ratios, and an improved design of the chemical control system.

4. Spectral Shift — The boric acid used for shim and temperature deficit control in the CNSG design could be replaced by heavy water. The heavy water concentration would be maximum at the beginning of core life, and diluted steadily throughout life. This concept (the spectral shift control reactor) has been under development by B&W and the AEC for some time.

A brief study of a spectral shift core was made for the CNSG (Section 8.3). This study consisted of computing the required enrichment and fuel cost for a core originally designed for boric acid control. There was a slight reduction in fuel cost for the spectral shift core. With the adjustment of core parameters, the spectral shift fuel costs probably could be reduced further.

Improvements in technology will make spectral shift more desirable. As core burnups increase, the increase in breeding ratio attending spectral shift more favorably affects fuel costs. Therefore, the incorporation of spectral shift in the CNSG design seems desirable for the future.

5. <u>Inexpensive Materials of Construction</u> — Throughout the reference design, stainless steel is in contact with primary water. Operating experience available now indicates that carbon steel would be satisfactory in most applications at a considerable reduction in cost. This will require the use of an alkali borate since carbon steel requires high pH or spectral shift for shim control.

6. <u>Automation</u> — The CNSG plant is automated so that load changes can be absorbed with no attention by the operator except to move the throttle valve to initiate the change. After the plant is started up, the only operator attention required routinely, other than being alert for abnormal operation, is to operate the chemical control system.

The actual operation of the chemical control system can be performed automatically by standard techniques. The decision whether to operate this system requires a xenon reactivity calculator. Such calculators are feasible, however, and one is being supplied by B&W for the ATR project. Therefore, it is considered that the complete automation of the CNSG plant operation is feasible with a minimum of development and additional equipment.

7. <u>Design for Simplified Installation</u> — The CNSG pressure vessel can be installed as a unit in a minimum of time. After this component is installed, instrumentation and control requires the most installation time. This item accounts for 30% of the installation labor for the entire reactor plant including collision barrier and nuclear foundation. The cost and the time involved could be decreased by packaging the control and instrumentation into a small number of modules and assembling the cable runs before installation.

3. REFERENCE DESIGN DESCRIPTION

3.1. General Description

The reference design includes the nuclear steam generating portion of the propulsion plant. It consists of a nuclear reactor, a once-through steam generator, a pressure vessel, four primary circulating pumps, auxiliary systems, and a water filled containment.

The major portion of the nuclear steam generating plant is confined within the walls of the containment, but a small portion of the auxiliaries are located outside. These auxiliaries include the poison injection pumps, the sampling apparatus, demineralizers for the makeup water, several small water storage tanks, and a ventilating fan for the gas space in the containment vessel.

The containment vessel, approximately 19 feet in diameter and 38 feet high, contains the reactor vessel, the containment heat exchangers, the condensing tanks, and approximately 390, 000 pounds of water.

The reactor vessel, which is completely submerged in the water, is approximately 10 feet in diameter and 27 feet high. It houses the core, the control rods, the once-through steam generator, the four primary circulating pumps, and the primary water. The vessel is approximately 3 inches thick and is insulated from the water by an 8-inch layer of lead bonded directly to the vessel. The lead also serves as a gamma shield and limits the flow of heat from the primary water to the containment water. Heat losses are limited to 2.2×10^6 Btu/hr which is the decay heat generation rate of the core four hours after reactor shutdown. This mechanism serves as the emergency decay heat removal system.

The reactor vessel internals are arranged so that the core is located near the bottom of the vessel and flow is upward. Water leaving the core flows up through a central riser containing the control rod extensions. At the top of the riser, the water spills radially out toward the vessel wall where it makes another 90 degree turn and then starts down through the shell side of the once-through steam generator.

The steam generator is rated at 61.7 MW and produces 224,000 lb/hr of steam. The temperature and pressure at the turbine throttle is 515 F and 402 psia. Primary water discharged from the steam generator enters the four circulating pumps in parallel. These pumps, located 90 degrees apart outside the reactor vessel, discharge to a common plenum at the core inlet. Because of this location, they are easily maintained.

The primary system is self-pressurized and uses boric acid for lifetime control. A steam space of approximately 190 ft^3 is provided at the top of the reactor vessel to absorb pressure fluctuation caused by temperature changes of the primary water. At full power, the pressure in the steam space is 812 psia when the core outlet temperature is 520 F.

The boric acid concentration is controlled by feeding demineralized water to the reactor vessel and by bleeding steam and water as required. Bleed from the reactor vessel is routed to the condensing tanks which are submerged in the containment water, or directly to the sea water. Demineralized water or poison is injected into the vessel by two positive displacement pumps.

Provisions are made to sample the primary, the containment, and the secondary water independently.

Fill and drain functions for the containment, the primary and the secondary systems are performed by dockside equipment through nozzles in the respective systems.

All heat losses from the primary, part of the secondary, the chemical control, and part of the relief systems ultimately are rejected to the containment water. Containment water heat exchangers maintain the containment water at approximately 110 F by rejecting the heat load to the sea.

3.2. Reactor Vessel Arrangement

3.2.1. General Description

Basically, the CNSG reactor is a closed cycle reactor system. The total primary system and the steam generator are contained within a single pressure vessel. This is accomplished by providing a closed circuit in which the flow is upward through the core and chimney, down through the steam generator, through the circulating pumps, and back to the core.

The basic pressure vessel (94-inch ID) will consist of a cylindrical shell section approximately 16.5 feet long with a hemispherical head welded at the bottom and a vessel closure flange at the top. The closure head will be an elliptical head welded to a head closure flange at its lower end which will mate with the vessel closure flange. Nozzles for feedwater inlets, steam outlets, poison injection, primary water injection, primary steam bleed, control rod drives, level indicators, and pressure indicators will be located in this top head. Nozzles for temperature indicators will be located in the vessel in the region just below the main flange. The over-all length of the unit will be approximately 26 feet. At the bottom of the vessel, four forgings welded into the hemispherical head will mount the pump stalks. In the region above the core approximately 2 feet up from the core support cylinder, a forging welded to the shell courses will attach the reactor vessel support skirt to the shell. An air gap will be provided between the support skirt and the vessel wall to reduce the heat release to the surrounding water during normal operation.

The vessel will be designed for an internal pressure of 1100 psig at a temperature of 560 F. The reactor will normally operate at a pressure of 800 psig with a temperature of 520 F.

The support skirt will be welded to the ship's structure at the bottom and will provide a support for the reactor vessel. The vessel will be supported laterally at the top by tangential rods mounted on the flange (above the main closure flange) which mounts the control rod drive support structure. The bottom head will be insulated with a 3-inch blanket of Fiberglas insulating wool (Owens Corning Fiberglas Corp., type TW-F No. 900 or the equivalent).

The arrangement inside the vessel will include a core support cylinder welded to the pressure vessel at the bottom to support the 37 fuel elements through the core container, and two outer thermal shields which will be supported by and welded to this cylinder by gusset plates at the bottom of this cylinder.

The once-through heat exchanger above the core will be composed of 300, 0.75-inch OD × 0.057 minimum wall tubes swaged to the 0.5-inch OD at the inlet tube sheets. These tubes will be contained within an inner and outer cylindrical baffle. The outer baffle (1.5 inches thick) will be bolted to the under side of the closure flange and will support the tube bank by using six radial top and bottom support struts welded to the outer shell which mount strap hangers extending down through the tubes. These tubes are supported along the length of the strap hangers. The inner cylindrical baffle (1 inch thick) will be welded to the six radial top and bottom struts and will form the basic chimney. Baffling, to assure that mixing occurs in the chimney, will attach to the inner shell. This will consist of a ring baffle approximately 10.5 feet above the active core to which a structure with a conical end extending downward will be attached. This will direct the flow to the outer periphery of the chimney and then back to the center region in the area of the ring baffle. The control-rod drive extension shaft shrouds will be supported at the top by the control-rod drive mechanism nozzles and by a support structure. This structure will provide lateral guidance and will be mounted inside the closure head on specially provided pads. The baffling in the chimney provides no support for the extension shaft shrouds.

3.2.2. Lead Cladding

Lead will be bonded to the reactor vessel in the areas indicated:

1. to shield the reactor vessel shell from direct exposure to cold containment water,
2. to provide a heat transfer surface in the heat exchanger and head area for emergency decay heat removal to the containment water, and,
3. to reduce gamma shielding costs by bringing the lead in as close as possible to the vessel wall.

Lead will be bonded to the support skirt, the vessel shell (in the heat exchanger area), and to the closure head (Fig. 3.2-1). In the area of the main flanges at the top and the pump extensions at the bottom, a canning to permit hot vessel expansion will be used to keep water from this portion of the unit. Lead shielding is not needed in the closure flange area, and the pumps will be shielded externally. The lead bonding to the vessel surface in the area of the heat exchanger and the head will be of a high integrity (95-97% contact area) for heat transfer considerations. In the area of the support skirt, the bond is not important from this standpoint.

Ultrasonic inspection (100%) will be performed in the heat exchanger and head area to assure bonding integrity. Gamma probing will be conducted on the support skirt area only. Since the thickness of lead on the other areas was established by heat transfer considerations instead of by shielding, density is not a criteria.

The lead will be installed by the Insmetal (International Shielding Metal) Ferrolum process. This process employs the strictest manufacturing and testing specifications and should provide satisfactory service.

3.2.3. Steam Generator

The steam generator was integrated with the head

1. to place the tube sheets in the most advantageous location for maintenance and accessibility,
2. to permit the easiest arrangement of replacing the steam generator and of performing general maintenance on the unit outside the reactor vessel, and
3. to give the smallest vessel diameter possible with full core exposure for refueling.

The weight of the head and steam generator is approximately 106, 000 pounds. To remove this unit, a cask to reduce the dose rate during transfer would be needed with an equivalent lead thickness of 3 inches surrounding the sides of the steam generator portion. This weight would add approximately 92, 000 pounds to make the total weight 198, 000 pounds. The cask will not provide complete shielding and will not include a bottom. Its design will be based on a compromise between maximum shielding and maximum lift on the crane. Personnel will not be permitted in the area during this transfer.

The weight of the head with steam generator attached appears large. Since this assembly must be removed at each refueling, its weight was checked against crane capacity available in American shipyards. Representative capacities are tabulated below.

Newport News	150 Long Tons
Sun Shipbuilding	100 Long Tons
Bethlehem, Quincy	300 Long Tons
Avondale	650 Long Tons
Maryland Dry Dock	100 Long Tons

3.2.4. General Description of Control Rod Drives

The control rod drive mechanisms will be mounted at the top of the reactor vessel on the closure head and will operate submerged in a water-filled containment containing 10 gm/ℓ of ammonium or potassium tetraborate. All drive connections will be water tight. Each complete rod driveline will include a control rod drive mechanism arranged with its actuating shaft which extends vertically downward from the mechanism. The drive mechanism will be attached at its lower end to a nozzle provided on the head. The actuating shaft of the control rod drive mechanism will be attached to the control rod by an extension shaft and a disconnect coupling. Insertion will be in the down direction. The control rod drive system will consist of seven on-off regulating mechanisms and a control panel. These mechanisms will be the canned type driving through a magnetic coupling arrangement.

A coupling on the extension shaft will permit the drive mechanism shaft to be uncoupled without raising the poison section of the control rod.

Positive means (other than an energy source) will hold the control rods in the down position after a scram. The same positive means will hold the rods in the "in" position if the ship should ever capsize.

Fast insertion of the drive mechanism at the same speed as normal rod motion (20 inches per minute, maximum) may be accomplished with the ship in a 90 degree position. A support structure will be used to mount any equipment required by the drives in this area. A control programmer will be provided. It will allow all rods to move individually, or manually or automatically in selected groups. A sequence switching unit will be provided to select the operation sequence of any mode. If a power failure or a malfunction of any component or circuit occurs, the control rod drive system will fail safe by causing the rods to scram or to remain motionless. The control rods will never be withdrawn. The scram delay time from the receipt of the scram signal to 2/3 completion of rod travel will not exceed two seconds. The design of the mechanism and scram circuit will enable the drive mechanism to scram individually, simultaneously, or in groups. By referring to the control instrumentation, the control rod may be positioned within an accuracy of ± 1 inch. This tolerance will include position indication system accuracies.

3.2.5. Refueling and Handling

The general philosophy of refueling and maintenance will be based on the use or modifications of NMSR equipment where possible. From the preliminary study made, it appears that the fuel cask, the internals cask, and the manipulator presently used on the Savannah could be modified for use on this reactor. Temporary parts, which could be removed to make the equipment ready for Savannah use, would be substituted. A cask consisting of lead shielding with a steel structure would be used to remove the head and steam generator combination. This cask will be a sleeve-type cylinder to fit over the closure head flange and rest on the machined lugs previously welded to the vessel closure flange. These lugs will provide radial alignment for the cask. Guidance will be provided on the inside of the cask for the head as it is lowered or raised. An upper flange on the cask will act as a lip to raise the cask with the head. This total lift of 198,000 pounds could be handled by several shipyards. The immediate area will be evacuated during transfer of the head and steam generator assembly. The general procedure for refueling will be as follows:

1. Lower the water level in the containment to a point below the vessel closure flange.
2. Remove the access hatch cover at the top of the containment.
3. Disconnect all lines (steam, feed water, etc.) passing through the containment cover.
4. Remove the main cover of the containment.
5. Remove all piping and canning from the head area.
6. Disconnect all wiring and lines to drives and other lines as necessary.
7. Remove the canning around the reactor closure flange.
8. Remove the closure studs.
9. Lower head cask on to vessel closure flange.
10. Engage and raise head to the upper position to lift the cask and to remove the assembly to the dockside pit for storage or maintenance on drives.

11. Remove or shuffle fuel and rods in a manner similar to the Savannah procedure.

12. Boric acid will be used to poison the core so that the rods can be removed separately from the fuel.

13. The reverse of the above would be performed to close the reactor.

14. The main circulating pumps would be removed as follows:

    Lower the water in the containment to a level below the pump flanges.

    Lower the water in the reactor to a level above the core and below the pump flanges.

    Remove the lead shielding cover on the outside of the pumps.

    Remove the canning around the pumps, and

    unbolt the pumps and remove for replacement or repair.

15. Maintenance on the other components within the containment will be done by removing complete units or by affecting minor repairs by direct entry into the area. The lead shielding around the pressure vessel will provide sufficient shielding during shutdown to permit personnel to work in the area of the pumps and other auxiliary equipment.

It is estimated that the refueling procedure will take approximately seven days. Refueling (core shuffling) will be required approximately once every two years under the conditions of service described in Section 5. Frequency of refueling has not been optimized for any particular service at this time.

3.3. Containment

The reactor plant containment consists of a water-filled vessel approximately 19 feet in diameter and 38 feet high. An air space is provided above the water to accomodate increases in water volume due to increases in containment temperatures. The design pressure of the gas space (50 psig) will cause the containment relief valve to open.

A design pressure was chosen after the functional requirements and the space limitations imposed on the containment were carefully considered. This design was based on holding a minimum amount of water to:

1. maintain submergence of the reactor vessel during an 80 degree roll,
2. provide adequate cooling of electrical equipment during normal operation,
3. provide a general heat sink,
4. provide a geometry adequate for equipment maintainance within the containment, and to
5. provide storage of primary system wastes if normal waste disposal methods cannot be used. The discharge of liquid wastes to the containment water is not a normal operating procedure, but if normal waste disposal methods cannot be used, discharge to the containment water is permissible and shielding is adequate.

To perform these functions, the design required approximately 390,000 pounds of water. This amount of water and the 780-ft^3 air space will limit the final pressure in the air space to 50 psig if all of the primary water is mixed with the containment water.

The containment water also serves as a portion of the biological shield and as a decay heat storage sink. As a heat storage sink, the water can store decay heat for 37 hours before the pressure in the gas space reaches 50 psig. (See Fig. 3.3-1.)

3.4. Shielding

Of primary interest in the CNSG shield design are the attainment of low environmental dose rates in personnel access spaces, minimum volume shielding to conserve valuable shipboard space, and minimum shield weight. Shielding and radiation considerations are closely related to placement of the primary heat exchangers with respect to the core in the pressure vessel, decay heat removal through the vessel wall to the containment water, the placement of secondary heat exchangers and condenser tanks within the containment, and the use of a vapor suppression-type wet containment. The shield has been designed to reduce full power dose rates from neutrons and gamma photons outside the shield to less than 0.8 mrem/hr. This full-power design dose rate is equivalent to 5 rem/yr at an average (full-time) reactor power level of 50 MWt as in the NS Savannah shield design.

The compact CNSG shield (Fig. 3.4-1) combines the functions of a primary and a biological shield. It consists of thermal shields and primary coolant water inside the reactor vessel, an 8-inch lead segment located near or bonded to the pressure vessel wall over the entire height of the reactor vessel, the containment water and containment vessel walls, a 3-inch thick layer of lead, and a 6-inch thick polyethylene layer outside the containment around the plane of the reactor core radial centerline. Figures 3.4-2 and 3.4-3 show the neutron flux radial profile, and Figure 3.4-1 shows the gamma dose rate as a function of the outer lead segment thickness. Also, double-bottom shield water tanks beneath the reactor vessel, which contain a minimum of 3 feet of water, will restrict operating and shutdown dose rates to 3.6 mr/hr, and lead shadow shielding around the exterior of the primary coolant pumps (where activated primary coolant is circulated outside the reactor vessel shield) will be required. If it is necessary to design the shield to accommodate high fission product leakage from the reactor core (approximately 5% pin clad failure), the containment condensate tanks may be shadow shielded with 4.5 inches of lead on the side adjacent to the containment tank wall.

With this shield design, the contributions to the dose rate at the shield exterior along the core radial centerline are:

Source	Dose rate, mrem/hr
Primary gamma	0.12
Secondary gamma	0.52
Neutron	0.01
	0.65

Since at the shield exterior, the neutron contribution to the total dose rate will be at a maximum on the core radial centerline, neutrons are not expected to contribute to more than 10% of the dose rate at any location. The necessity for shielding the secondary coolant steam leaving the reactor vessel has been eliminated in the reactor design by maintaining a 2-foot separation between the bottom of the coolant coils and the reactor core. The calculated maximum N^{16} dose rate resulting in the secondary system at the condenser hot well is less than 0.3 mr/hr. Also, the 8-inch lead layer over the upper portion of the reactor vessel is sized on decay heat removal considerations. Only 4.5 inches of lead would be required in this region to attenuate the N^{16} gamma radiation from the activated primary coolant. In the region directly above the reactor vessel, penetrations in the head for control rod drives and plenum outlets, which would normally present streaming problems, are compensated by containment water shielding over the vessel.

3.5. Steam Generator

The once-through steam generator is located within the reactor vessel in the annulus between the chimney riser and the reactor vessel wall. It is divided into three sections, each with its own feedwater inlet and steam outlet. Each section may operate independently and can be isolated. It is possible to operate with two or even one of the three sections if leaks or other difficulties make it desirable.

The three feedwater supply lines enter through the head of the reactor vessel where they are joined to three plenums containing the tube sheets. The tubes pass down through the steam void to the bottom of the tube bank. At this point, the tubes start to spiral around the chimney and up the annulus until they emerge from the primary water and pass through the steam zone. At the reactor vessel head, the tubes again connect with three tube sheets. Steam leaving the tube sheets is channeled into three steam lines which are routed out of the containment vessel. The tube sheets are accessible from the outside of the reactor vessel for plugging operations. The steam generator was designed to transfer 61.7 MWt for the production of steam at 397 psig at 515 F. These steam conditions exist when the primary inlet temperature is 520 F and the feedwater inlet temperature is 348 F. The total heat transfer area required is 7706 ft^2. This area does not include an allowance for fouling.

The tube bank, approximately 8.5 feet high, consists of 300 tubes each with a 0.75-inch OD. The average length of each tube is 131 feet. The tubes are wound in the annulus so that the vertical cross section of the tube bank forms a parallelogram. The parallelogram is skewed from the horizontal by 30 degrees to prevent the tubes from being uncovered when the ship rolls.

The tubes are supported with a series of vertical hangers connected to six radial struts. The struts are anchored to the chimney baffle at one end and to the outer tube bank baffle at the other end. The entire tube bundle is supported from the reactor vessel head, and is removed from the reactor vessel with the head.

3.6. Primary Pumps

The primary system contains four primary pumps located on the outside periphery of the reactor vessel and spaced 90 degrees apart. They are mounted on four extensions of the reactor vessel. The suction enters the pump through an annulus located within each reactor extension. When the primary water reaches the pump, it makes a 180-degree turn just before it is picked up by the centrally located rotor. The rotor then discharges the water down the central duct to the core inlet plenum.

Each pump is driven by a canned electric motor which is cooled by the natural convection of the containment water. A total developed head of 8 psi is produced at a capacity of 1.5×10^6 lb/hr. During cold startup, the required net positive suction head for the pumps is supplied by pressurizing the reactor vessel with hydrogen. During normal operation, it is supplied by the self-pressurization feature of the reactor. A check valve is included in the pump discharge piping to prevent a shutdown pump from windmilling and to prevent flow bypassing through the core.

3.7. Core

3.7.1. Mechanical Description

3.7.1.1. General

Basically the core will consist of 37 fuel elements (hexagonal in cross section) enclosed within a hexagonally shaped core container. The effective diameter of the core, approximately 49 inches, will be divided into three zones as follows:

1. 18 outer elements (Zone 3),
2. 18 inner elements (Zone 2).
3. 1 center elements (Zone 1).

The elements will be divided into two categories. Type A elements will provide for rod travel through the bundle and type B will not. There will be 13 total elements of the A type and 24 of the B type in the core. Each A-type element will contain 108 pins and each

B-type will contain 121 pins. The total number of pins will be 4308. Zones 3 and 2 will each contain six A and twelve B-type elements. The center element (Zone 1) will be an A-type element.

The 37 elements will be located in the hexagonally shaped core container. This container will be composed of 12 segmented stainless-steel castings machined on the inside to produce the contour formed by the 37 hexagonally shaped fuel elements. These segments will be bolted together at the edges to form the container. A plate bolted to the top of this basket will form a flange which will bolt to the core support cylinder welded to the vessel. The lower grid plate, which supports the fuel elements, will be bolted to the bottom of the core container. The grid plate will consist of 37 flow tubes welded to a top and bottom plate to form a load carrying structure. The fuel elements will rest on the top plate of this structure with their lower nozzle castings extending into the flow tubes to provide coordinate location. Indexing lugs on the topplate of the lower grid plate will mate with slots on the fuel element to provide for rotational alignment. A cylindrical flow baffle surrounding the follower section of the control rods will bolt to the under side of the grid plate and will act as a shroud to prevent primary water from impinging on the rods. Cross members at the bottom of this shroud will serve as stops for the rods when they are coupled and uncoupled to the extension shafts. A lateral support plate, which serves to restrain the fuel elements in a lateral direction during ship maneuvering conditions, will rest on top of the core container and will be indexed to it by lugs and slots. This plate will also serve to restrain the fuel elements and plug rods if the ship should ever capsize.

A seal pressing on the lateral support plate will prevent leakage from the cold to hot leg of the primary coolant. This seal will permit relative thermal expansion between the internals and the reactor vessel.

3.7.1.2. Control Rods

The seven control rods will be of a cross-sectional Y shape. Each rod will pass through the upper support plate, the fuel bundle, and the lower grid plate.

The poison section will consist of three boron modified stainless-steel plates with a boron content of $1.7 {}^{+0.30}_{-0.10}$ wt% (92%-enriched in B^{10}) jacketed by three stainless-steel angles plug welded together at the center and welded to a stainless-steel guide strip at the edges. Expansion gaps will be included for expected boron growth under irradiation and to provide an expansion chamber for gas release to limit the internal gas pressure to a value consistent with design stresses and deflections.

The follower portion of the rod will fill the water gap as the poison is withdrawn to reduce flux peaking. This portion will consist of three Zircaloy angles welded back to back and riveted to a stainless-steel transition piece which will be welded to the stainless-steel angles surrounding the B^{10} plates. The transition piece will have provisions to accommodate the differential thermal growth between the stainless portion and the Zircaloy part of the rod.

At the top, three stainless-steel extension plates will be welded to the stainless steel angles. A handling knob will be welded to the center of the extension plates. The stainless-steel edge strip on each blade will be chrome plated to provide smooth sliding action as the rod travels through the upper support and the lower grid plate. The rods will pass freely through the fuel elements in the Y-shaped area receiving their guidance from the top support plate and the lower grid structures. The over-all length of the rod to the top of the handling rod will be approximately 8 feet, 4 inches. The poison section will be 38 inches long, and the Zircaloy portion including the 4-inch stainless-steel transition piece will be 4 feet, 1 inch long.

3.7.1.3. Plug Rods

Six Zircaloy plug rods, made in a Y-shaped cross section, will be disposed in the outer periphery elements. They will be made of three Zircaloy angles positioned back to back and welded along the outer edge. A handling knob will be located at the top. These rods will be stationary and will reduce the water gap in the six outer control rod fuel elements which will be shuffled to the inner zone at the first reloading. The Y section will be enclosed within a 6.25-inch diameter circle. Each blade will be 0.625 inches thick and will be guided at the top support plate and at the lower grid. Each rod will rest on the lower grid plate and will be retained by the top lateral support plate.

3.7.1.4. Fuel Elements

Each A and B-type element will consist of four or more spacer grids sandwiched between an upper and a lower stainless-steel casting. Six Zircaloy tie rods, located at the vertices of the hexagon shape, fasten mechanically to the top and bottom castings with cap screws to complete the assembly. One spacer grid will be at the top and one at the bottom. The remaining grids will be equally spaced between the top and the bottom. The A grids will have control rod access areas provided, but the B grids will not. These grids will be positioned and retained axially and radially by the tie rods.

The fuel rod assembly (approximately 45-1/8 inches long) will consist of a Zircaloy-2 tube to which Zircaloy-2 end caps will be welded. An expansion chamber for internal gas release will be provided, and mechanically compacted fuel and swaging may be employed. The final rod diameter will be 0.460 inches with a nominal wall thickness of 0.021 inches on the cladding. The resulting fuel density will be 92% ± 2%.

The nominal enrichment is:

1. Zone 1 1.6% ± 0.068%
2. Zone 2 2.1% ± 0.068%
3. Zone 3 3.1% ± 0.068%

The total kg of U^{235} is:

1. Zone 1 1.45
2. Zone 2 36.9
3. Zone 3 54.5

The total U^{235} loading is 92.85 ± 1% kg per core, and the total UO_2 loading is 4100 ± 1% kg per core.

The active length of the fuel will be 42 inches. The over-all length of the element will be 57 5 inches with an approximate weight of 350 pounds.

The maximum clad temperature was 576 F with an external design pressure of 1100 psig and a coolant temperature of 560 F. The fuel cladding thickness was arrived at on the basis of a free standing tube with no help from the fuel.

3.7.2. Physics Description

The CNSG reactor design was based primarily on the advancement of the new and old concepts developed for small power plants. Triangular pitched pins, in reshuffable fuel elements, which have been zone loaded for lowest power peaking are ideas taken from previous NS Savannah upgrading work. Improvements in the mechanical design of the fuel elements and fuel distribution in the core, and the use of a soluble poison control system have resulted in further power cost reduction. More detailed calculations on void fractions effects have given further basis to the feasibility of the self-pressurized design. Detailed descriptions of the various calculational models and methods employed in the nuclear design phases are given in this report.

3.7.2.1. Reactor Cycle Lifetime

The reactor cycle lifetime was calculated for 485 days of operation at 63 MWt between shuffles. This results in a cycle fuel exposure of 15,000 MW days per ton of contained uranium oxide.

Core Loading

The core was loaded in three radial zones. The central zone (Region I) consists of a single fuel element and is replaced at the end of each cycle. The second radial zone (Region II) consists of the six elements surrounding Region I plus the next ring of twelve elements. This zone is also removed at the end of each cycle. The third radial zone (Region III) consists of the outer ring of eighteen fuel elements of the core. The fuel elements of Region III are shuffled into Region II after one cycle of operation. The detailed enrichments and loadings at the beginning and end of both the initial and the equilibrium cycle are given in Table 3.7.2-1.

Table 3.7.2-1. Reactor Mass Balance

Region	1 1 fuel element	2 18 fuel elements	3 18 fuel elements
First cycle			
Initial time zero			
U^{235} enrichment, wt %	1.60	2.10	3.10
U^{235}, kg	1.5	36.9	54.5
Total uranium, kg	90.6	1762.2	1762.2
Total plutonium, kg	---	---	---
Final— 485 days			
U^{235} enrichment, wt %	0.84	1.16	2.30
U^{235}, kg	0.75	20.2	40.1
Total uranium, kg	89.4	1734.2	1741.9
Total plutonium, kg	0.5	9.3	6.2
Equilibrium cycle			
Initial time zero			
U^{235} enrichment, wt %	1.60	2.30	3.10
U^{235}, kg	1.5	40.1	54.5
Total uranium, kg	90.6	1741.9	1762.2
Total plutonium, kg	---	6.2	---
Final— 485 days			
U^{235} enrichment, wt %	0.82	1.48	2.30
U^{235}, kg	0.73	25.3	40.0
Total uranium, kg	89.2	1716.5	1741.9
Total plutonium, kg	0.50	12.78	6.1

Power Peaking

The zone enrichments were chosen to give the lowest radial power peaking while they maintained the desired fuel irradiation limits under the reshuffle scheme. Figures 3.7.2-1 through 3.7.2-4 show power peaking for several core conditions. Figure 3.7.2-1 shows the gross radial power to average power profile at the beginning and end of the equilibrium cycle. This profile was taken from the radial lifetime calculation where the void distribution was homogenized in the axial direction. Figures 3.7.2-2, 3.7.2-3 and 3.7.2-4 are the gross axial and the radial power to average power contour maps over the core as calculated by the two dimensional codes. These figures show the effects of the ship's motion on the gross peaking factor. The power distribution changes with ship's motion, but there is no noticeable increase in the total peaking. These calculations were based on void distribution as a function of ship's motion (Section 4. 7. 3). Fuel irradiation limitations were based on an over-all peak to average power factor of three to one. Figures 3. 7. 2-2, 3. 7. 2-3, and 3. 7. 2-4 show that the gross axial and the radial factor total is approximately two to one. Calculations of local gaps (such as control rod follower regions) show a maximum factor of 1. 2 to one. Combining the gross and local factors gives a calculated total peaking factor of approximately 2. 4 to one. Uncertainties in the peaking effects of the void distribution, the soluble poison distribution, and regulator rod motion over core life necessitate the use of a larger peaking factor (three to one) in this design study. Multidimensional lifetime calculations would probably substantiate these lower values, but since no provision for this type of work was made in this study, the three-to-one ratio was used.

Soluble Poison Control

Core criticality was controlled throughout life by varying the soluble poison concentration in the moderator. In this study, H_3BO_3 was assumed to be the poison. Figure 3. 7. 2-5 shows the variation of boric acid as a function of core lifetime. The amount of acid is given as gm of H_3BO_3 per kg of H_2O. Figure 3. 7. 2-6 gives the variation of poison during the first five days of life. The poison rate of change during this period is important in sizing pipes and other parts of the control system. Figure 3. 7. 2-7 shows the reactivity worth of H_3BO_3 as a function of the moderator temperature for the equilibrium cycle at the beginning of life. This figure is used to establish the boric acid concentration necessary for startup.

Two dimensional calculations in R-Z geometry were used as a check on the initial soluble poison concentration. The radial lifetime calculation was set up with the voids homogenized in the axial direction. Two dimensional cases showing the proper void distribution gave approximately the same initial excess reactivity, and predicted the same initial poison concentration for control.

Reactivity Balance and Neutron Economy

Neutron balances for the beginning and end of core life are tabulated in Table 3.7.2-3. These balances have been normalized to the total absorptions in the fissionable material. The leakage term includes all neutrons leaking from the fuel region into the shroud and the reflector regions.

Table 3.7.2-2. Equilibrium Cycle Reactivity Balance

Core condition	k_{eff}
Cold, clean, zero power	1.214
Hot, clean, zero power	1.166
Hot, clean, full power	1.156
	δk_{eff}
Temperature deficit	0.048
Doppler deficit	.008
Void deficit*	.002
Equilibrium Xe and Sm	.038
Equilibrium Xe	.030
Fuel burnup and isotope buildup	0.118

* Void deficit does not reflect ship's motion.

Table 3.7.2-3. Equilibrium Cycle Neutron Balance

	Zero days			485 days		
Element	Resonance	Thermal	Total	Resonance	Thermal	Total
U^{235}	0.164	0.695	0.859	0.109	0.511	0.620
U^{236}	.003	---	.003	.007	.001	.008
U^{238}	.377	.122	.499	.364	.128	.492
Pu^{239}	.017	.120	.137	.038	.307	.345
Pu^{240}	.020	.002	.022	.068	.007	.075
Pu^{241}	0.001	.003	.004	0.010	.025	.035
Xe^{135}	---	---	---	---	.036	.036
Sm^{149}	---	0.006	0.006	---	0.011	0.011

Table 3.7.2-3 (Cont'd)

Element	Zero days Resonance	Zero days Thermal	Zero days Total	485 days Resonance	485 days Thermal	485 days Total
H_2O	---	0.066	0.066	---	0.069	0.069
Zircaloy-2	0.001	.005	.006	0.001	.005	.006
H_3BO_3	.026	.166	.192	---	---	---
Fission Products	.017	.012	.029	.044	.037	.081
Leakage	0.084	0.139	0.223	0.083	0.152	0.235

Figures 3.7.2-8 and 3.7.2-9 show the variation of the integrated conversion ratio and $\bar{\eta}$ (average number of neutrons produced per absorption in fissionable fuel) with core life. A maximum conversion ratio of 0.54 was reached at the end of the equilibrium cycle.

Fast and thermal flux profiles are shown in Figures 3.7.2-10 and 3.7.2-11. These profiles were taken from the radial lifetime calculation of the equilibrium cycle. The thermal flux shape toward the center of the core reflects the homogenized void distribution used in the core life model.

3.7.2.2. Operational Parameters

Various coefficients and reactivity deficits pertinent to the operation analysis of the CNSG reactor were calculated. These parameters pertain to the beginning and end of core life.

Temperature and Doppler Effects

Reactivity coefficients and deficits are given in Table 3.7.2-4 for the temperature and the Doppler effects over the lifetime. The temperature coefficient ($1/k \frac{\partial k}{\partial T_m}$) was calculated as a function of temperatures for the start of life conditions. (See Fig. 3.7.2-12.) At each base point in temperature, the poison concentration was changed to maintain a critical reactor. A change in moderator temperature was then imposed on the critical core to determine the coefficient at that point. The Doppler coefficient ($1/k \frac{\partial k}{\partial T_f}$) reflects a change in fuel temperature.

Table 3.7.2-4. Temperature and Doppler Effects Equilibrium Cycle

	Zero days	485 days
Temperature deficit	$-4.8\%\ \delta k$	$-4.7\%\ \delta k$
Operating temperature coefficient	$-2.2 \times 10^{-4}\ 1/k \frac{\partial k}{\partial T_m}$	$-2.0 \times 10^{-4}\ 1/k \frac{\partial k}{\partial T_m}$
Doppler deficit	$-0.8\%\ \delta k$	$-0.7\%\ \delta k$
Operating Doppler coefficient	$-0.13 \times 10^{-4}\ 1/k \frac{\partial k}{\partial T_f}$	$-0.12 \times 10^{-4}\ 1/k \frac{\partial k}{\partial T_f}$

Void Effects

Various void distributions reflecting ship's motion were studied in two dimensions. Figures 3.7.2-2, 3.7.2-3 and 3.7.2-4 show the resulting power distributions. The resulting reactivity effects were compared with a case having zero voids. The actual void reactivity coefficients ($1/k \frac{\partial k}{\partial ICV}$) were expressed in terms of the integrated core void (Fig. 3.7.2-13). Relative reactivity worths of the various distributions are given in Table 3.7.2-5. These cases were calculated with an acid concentration corresponding to about 230 days of core life (approximately 3 gm H_3BO_3 per kg H_2O).

Table 3.7.2-5. Void Reactivity Effect

Void distribution	Void deficit, % δk	Integrated core void, %ICV	Void coefficient, $1/k \frac{\%\delta k}{\%\delta\, ICV}$
Gravitational force = 1.6	0.18	0.33	-5.4×10^{-4}
Gravitational force = 1.0	0.23	0.38	-6.1×10^{-4}
Gravitational force = 0.4	0.40	0.51	-7.8×10^{-4}

Xenon Transients

The reactivity change resulting from a power reduction may be important to the CNSG core operation. As a limiting case the change in reactivity (due to Xe buildup and decay) following a full power shutdown (100 to 0% operating power) was calculated as a function of time. Figures 3. 7. 2-14 and -15 show the resulting change in reactivity and the variation in soluble poisons necessary to maintain a critical condition. As will be seen in Section 3. 8. 2. 2, xenon transients are normally handled by control rods, rather than soluble poison.

Effective Delay Fraction and Delay Constants

The effective delay neutron fractions and the corresponding decay constants were calculated for six delay groups to help in performing operational analysis studies. An effective neutron life of approximately 30 microseconds was estimated. (See Table 3. 7. 2-6 for the resulting delay neutron data.) Weighting information for these calculations were taken from the radial lifetime results.

Table 3. 7. 2-6. Effective Delay Fractions and Decay Constants Equilibrium Cycle

Delay group	Zero days		485 days	
	$\bar{\beta}_{eff}$	λ_{eff}	$\bar{\beta}_{eff}$	λ_{eff}
1	0. 00022	0. 01251	0. 00017	0. 01261
2	. 00151	0. 03066	. 00125	0. 03050
3	. 00141	0. 11388	. 00113	0. 11745
4	. 00292	0. 30665	. 00228	0. 31352
5	. 00102	1. 1613	. 00080	1. 1535
6	0. 00036	3. 1064	0. 00029	2. 9941

3. 7. 2. 3. Fuel Costs

Through the use of zirconium cladding, a wet lattice, and relatively long core lifetimes, very low fuel costs were obtained.

Table 3.7.2-7 lists the dollar charges and the corresponding mills/shp.

The total fuel cost was calculated under the following general headings:

1. Fabrication — Charges based on fuel handling and manufacturing estimates were received from the vendor. Inventory changes at 4-3/4% during fabrication and transportation charges are included.

2. Fissionable Material — The cost is the initial value of the fuel loading minus the value of the exposed fuel at the end of core life.* Plutonium credit is taken only on the mass of plutonium removed from the core at the end of each cycle. A value of $9.50 per gram of Pu in fuel regions I and II was assumed.

3. Inventory — An inventory rate of 4.75% is used to calculate the charge for the time period the fuel is in core and is being reprocessed. Core power generation time was based on a power factor of 0.833. It was also assumed that inventory charges would be paid periodically, and the fuel value at the end of each pay period would be recalculated. This method of payment reduces the value of fuel on which the inventory is paid to the average value over the core life. Shipping, cooling, reprocessing, and reconversion were assumed to take six months.

4. Transportation and Insurance — Charges were based on the AEC analysis for the spent fuel shipment for the Yankee Atomic Electric Plant ($10 per kg of contained uranium). This charge covers shipping fuel regions I and II at the end of each cycle.

5. Reprocessing and Reconversion — Charges are based on the latest AEC estimates. Reprocessing rates are $16,988 per day of operation. Core enrichments allow approximately 1000 kg to be processed daily.** An additional three-day turn-around period is added.

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* See Reference 6 of Section 8.3.

** See Reference 7 of Section 8.3.

Reconversion of the low enriched uranyl nitrate into UF_6 is charged at $5.60 per kg for enrichments of 5% or less. Plutonyl nitrate is converted to metal plutonium buttons at an estimated $1500 per kg of contained plutonium. Reprocessing and reconversion losses were taken from the latest AEC press releases. Reprocessing losses were assumed to be 1%. Reconversion losses were assumed to be 0.3% for uranium and 1.0% for plutonium.

Table 3.7.2-7. Fuel Cost Summary

	First cycle		Equilibrium cycle	
Cost item	$	mills/shp-hr	$	mills/shp/hr
Fabrication	325, 300	1.27	167, 000	0.65
Net fissionable material	276, 400	1.08	226, 300	0.88
Inventory	51, 600	0.20	56, 500	0.22
Reprocessing and reconversion	114, 100	0.45	120, 600	0.48
Transportation and insurance	18, 200	0.07	18, 000	0.07
Total	785, 600	(3.07)	588, 600	(2.30)

3.7.3. Thermal and Hydraulic Design

The thermal and hydraulic design of the core is based on an evaluation of the burnout heat transfer, the fuel melting limitations, the steam voids, and the flow stability. Thermal and hydraulic data of core performance are shown below.

Reactor power, MWt	62.4
Coolant flow rate, lb/hr	6.0×10^6
Core operating pressure, psia	816
Average core outlet temperature, F	520
Core inlet temperature, F	490.4
Heat transfer surface, ft^2	1814
Average heat flux, Btu/hr-ft^2 62.4 MWt	117,500
Maximum heat flux, Btu/hr-ft^2 62.4 MWt	376,500
Burnout heat flux - Surface heat flux, 62.4 MWt	4
Reactor power for fuel melting, MWt	80.2 (128.4%)
Reactor power for burnout, MWt	106 (170%)
Power density, kw/ℓ total core volume	47
Power density, kw/ℓ core coolant volume	79

3.7.3.1. Power Distribution

The power distribution used for the thermal and hydraulic analyses shown in Figure 3.7.3-1 was obtained from the distribution plot shown in Section 3.7.2. The axial distribution corresponds to 100% power and a radial position of 17 cm. Flux levels were increased 25% to account for additional peaking resulting from fuel shuffling.

3.7.3.2. Hot Channel Factors

When thermal limitations are established for a reactor core, consideration must be given to the effects of manufacturing tolerances. For this design, all of the adverse effects were assumed to occur in the region of maximum heat flux. Also, the manufacturing tolerances were assumed to cause the heat generation rate to be greater than the nominal value and the flow to be reduced. Therefore, the enthalpy rise of the coolant is increased and the fuel temperatures are greater.

Hot channel factors resulting from manufacturing tolerance evaluation are tabulated below.

F''_q - surface heat flux — 1.067

F_θ - film temperature drop — 1.074

$F\Delta_h$ - coolant enthalpy rise — 1.138

The surface heat flux factor is the ratio of the increased heat flux (due to manufacturing tolerances) to the nominal heat flux. The film temperature drop factor is the ratio of the increased film temperature rise to the nominal temperature rise. The coolant enthalpy rise factor is a measure of the coolant enthalpy rise resulting from the increased heat generation rate and flow reduction.

3.7.3.3. Fuel Melting

The maximum heat generation rate in the fuel pin is limited so that the fuel will not exceed its melting temperature of 5000 F. The limiting reactor power was obtained by using the following maximum-to-average power ratios and hot channel factors.

Power ratio	
Axial and radial	2.00
Fuel shuffling	1.25
Local	1.20
Hot channel factor	
F''_q	1.067
Total	3.2

The resulting reactor power for fuel melting is 80.2 MWt, which corresponds to a maximum heat generation rate of 17 kw/ft of the fuel pin.

3.7.3.4. Flow Distribution

The calculation model used to determine the coolant distribution consists of a series of typical channels connected in parallel. The flow in each channel is varied until the total pressure drop in each channel is equal and until the sum of the flow in all channels is equal to the total flow.

The pressure loss increase due to boiling in the hotter channels is almost offset by the increase in density head. This compensating effect maintains the flow in the hotter channels so that the flow distribution across the core is nearly uniform.

3.7.3.5. Void Distribution

The operating pressure of a self-pressurized reactor is determined b the saturation pressures of the water at the steam-water inner face. Mixing baffles in the riser section of the core assure that the reactor coolant is completely mixed before it reaches the surface. Therefore, the average reactor outlet temperature is identical to the saturation temperature in the steam dome.

Because steam will occur in any channel having a heat generation rate greater than the core average, the CNSG core normally operates with a steam void. The void distribution for the CNSG configuration was calculated at 100% power. (See Fig. 3.7.3-2.) The core was grouped into four volumes with the radial heat generation rate for each volume corresponding to its average radial heat generation rate.

Before a reference design was established, void analyses were performed on a core configuration similar to the CNSG core. These void distributions were used in supporting studies and as guides for the CNSG design.

The calculated voids for the CNSG reactor are lower than those obtained while establishing trends. Decreasing the voids has been accomplished by increasing the reactor coolant flow. This change causes a smaller coolant temperature rise across the core.

The power distribution and the resulting voids are interdependent. To obtain an accurate solution, these two variables must be solved simultaneously. In practice, this requires an iterative solution. The voids used to calculate the power shapes were greater than those calculated after the power shapes were calculated. Therefore, further iterations were not made because the first iteration was judged to be conservative.

3.7.3.6. Burnout Power

Burnout power is defined as the maximum power that can be removed from the core without departure from nucleate boiling.

Channels having the largest peaking factors are farther from film boiling than other channels within the lattice. The peaking resulting from a large water channel is nullified by the additional cooling provided by the large channel. Therefore, the local peaking factor was not used directly to establish the burnout power.

The correlation used to predict the onset of film boiling is similar to that presented in the WAPD Report 188. The power level corresponding to the onset of film boiling is 106 MWt.

3.7.3.7. Flow Stability

A channel flow versus a pressure drop curve having more than one flow rate with the same pressure drop is a necessary (but not sufficient) condition for flow instability in a parallel channel flow configuration. When this occurs in a heated channel, there are three flow rates having the same pressure drop. When the flow rate is such that the pressure losses are mostly from a single-phase flow (either steam or water) the pressure loss curve nearly follows the velocity squared path. The unstable condition only occurs when the dominating pressure drop is from two-phase flow.

Figure 3.7.3-3 shows the pressure drop versus the flow curve for a typical channel of the CNSG core. The curve covers the range where two-phase flow may dominate. The flow per pin in the CNSG core is about 1250 lb/hr, and the curve indicates that the necessary condition for flow instability is not present.

3.8. Instrumentation and Control

3.8.1. Introduction

The CNSG instrumentation and control equipment is in the central control room with the propulsion plant instrumentation and control. This control room also contains the control panels for these systems. Only those auxiliaries which can be controlled more effectively from a central area rather than locally will be mounted on the control panels.

The instrumentation and control is accomplished by the Nuclear Steam Generator Instrumentation and Control System. This system contains four subsystems to measure and control the variables in the over-all system. They are:

1. Reactor control system.
2. Nuclear instrumentation system.
3. Reactor safety system.
4. Process instrumentation and control system.

The indicators, alarms, switches, and control setpoint adjustments for these subsystems are located on the Nuclear Steam Generator Control Panel (NSGCP) in the central control room. The cabinets housing the relays, controllers, computing circuits, interlocks, etc., for these subsystems are not on the NSGCP because they do not require operator attention.

The electronic circuits in these subsystems use solid-state components whenever possible to minimize space requirements where they have proved more reliable than their thermionic counterparts.

All measurement, control, and computing circuits use electrical components. All final control elements are electric or electro-pneumatic.

As in any ship's operation, the propulsion load must be under manual control by the engineer at all times for safety and maneuverability. The demand made by turbine-generators on the steam supply is determined by the size of their electrical loads. The instrumentation and control system operates the reactor and its auxiliaries so that they automatically follow the propulsion and electrical load demands imposed on the plant during all steady-state and normal transient operation.

3.8.2. Reactor Control System

3.8.2.1. General

The reactor control system enables the reactor power level to follow ± 1% per second load transients over the range of 10 to 100% power. Over-all reactivity control is divided among three reactivity sources: the central control or regulating rod, the outer ring of six control or shim rods, and the soluble poison concentration.

Because the reactor is largely self-regulating, the reactor control system is not complex. (See Fig. 3.8-2.) The reactor control system maintains constant pressure in the primary system. Since the primary pressure established in the reactor vessel steam space corresponds to the saturation temperature of the water leaving the core, controlling constant primary pressure maintains constant core outlet temperature. Figure 3.8-1 shows the primary system temperature versus load characteristics with this form of control. The drooping characteristic of the average temperature with a load provides reactivity to offset the void and Doppler effects, and takes maximum advantage of the self-regulating characteristics of the reactor. The once-through boiler eliminates the drooping characteristic of secondary steam pressure associated with natural circulation boilers which is further aggravated by drooping primary temperature.

The control system output positions the central control rod to achieve the desired reactor power level. A pressure deadband in the control system minimizes the control rod duty cycle. Outside the deadband, the center rod moves at a constant speed to restore the pressure within the deadband.

Upper and lower flux limits are imposed on the control system to prevent the reactor from being driven to overflux scram or subcritical during unusual transients. At full load, the central control rod is in the upper 10% of stroke, and at a zero-propulsion load, it is in the lower 10% of its stroke.

The other six control rods and the soluble poison concentration are controlled manually. The position of the central control rod relative to the load is used as an index to manually operate the six shim rods and to change the soluble poison concentration. The shim rods are used for short term reactivity changes (primarily xenon). The soluble poison concentration is used for long term reactivity changes (primary burnup) and for temperature defect during a cold shutdown.

3.8.2.2. Control Rod Motion During Xenon Transient

During normal steady-state operation with equilibrium xenon concentration, the operator uses the regulating rod position relative to the load to adjust soluble poison concentration. At full power, the regulating rod should be in the upper 10% of its stroke; at 10% power, it should be in the bottom 10% of its stroke; at 50% power, it should be in the middle 10% of its stroke. This relationship results from the fact that the reactivity of the regulating rod is roughly equal to the Doppler and void reactivity.

The regulating rod and shim rods are used to take care of all xenon transients resulting from load changes. Soluble poison concentration is not changed for xenon transients. However, checks can be made at intervals during xenon transients to determine when long term reactivity changes require soluble poison concentration changes.

The largest frequently occurring load change is propulsion turbine shutdown. This corresponds roughly to a reduction in load from 100 to 10% of full power. The plant conditions at the start of this transient are those which prevail after a long run at full power. Xenon is at equilibrium for 100% power, shim rods are fully withdrawn, and soluble poison concentration is adjusted to place the regulating rod in the upper 10% of its stroke.

Figure 3.8-3 shows a 100 to 10% transient with a startup at 100 hours after shutdown. The 100 to 10% transient is summarized as follows:

0 hours — Throttle closed; steam flow drops to base load of 10%; regulating rod inserted to offset Doppler and void effects.

4 hours — Xenon poisoning peaks; regulating rod is partially withdrawn to offset xenon.

15 hours — Regulating rod bottoms and shim rods start in to offset decaying xenon. An alternate method would position shim rods to hold the regulating rod in the bottom 10% of its stroke.

50 hours — Xenon reaches equilibrium for 10% of its load; shims level off.

100 hours — Throttle opened to 70% of steam flow; shim rods withdrawn to offset Doppler and void effects.

115 hours — Shim rods full out as xenon builds in; throttle opened to 100% steam flow; regulating rod withdrawn to offset Doppler and void effects.

150 hours — Xenon reaches equilibrium for 100%; regulating rod back to original position at 0 hours.

At no time during this transient is it necessary or desirable to change the soluble poison concentration. However, if the shim rods are inserted past the 25% withdrawn position during the xenon decay between 15 and 50 hours, soluble poison concentration must be increased to preserve the shutdown margin. If the regulating rod is withdrawn to its full-out position during the xenon buildup from 115 to 150 hours, the soluble poison concentration must be decreased to restore the rod to its upper 10% of travel.

The most difficult point in returning to full power during a 100 to 10% transient would occur at the peak of the xenon poisoning. Figure 3.8-4 and the following tabulation summarize the effects of returning to full power at the xenon poisoning peak.

4 hours — Xenon poisoning peaks; throttle opened to 100% steam flow; regulating rod full out and temperature drops to 498 F to offset Doppler and void effects.

6 hours — Temperature returns to normal and regulating rods start in to offset burnout of xenon poisoning.

8 hours — Xenon levels off as the xenon burnout rate equals the xenon buildup rate.

11 hours — Xenon returns to equilibrium for 100% power; regulating rod is back to original position at 0 hours.

Figure 3.8-5 shows a 100% to zero power transient with the plant maintained hot. Startup occurs 100 hours after shutdown. This transient is summarized as follows:

0 hours — Steam load is completely shut down or transferred to an auxiliary source; regulating rod and fraction of shim rods are inserted to offset Doppler and void effects.

4 hours — Shim rods and regulating rod are full out to offset loss of reactivity from xenon poisoning.

8 hours — Xenon poisoning peaks; temperature drops to 510 F to offset xenon poisoning.

12 hours — Temperature is returned to normal and regulating rod starts in to offset decaying xenon.

20 hours — Regulating rod bottoms and shim rods start in to offset decaying xenon.

70 hours — Xenon decays back to its zero value; shim rods level off.

100 hours — Steam load is raised to 70%; shim rods are withdrawn to offset Doppler and void effects.

121 hours — Shim rods are full out as xenon builds up; steam load is raised to 100%; regulating rod is withdrawn to offset Doppler and void effects.

150 hours — Xenon reaches equilibrium for 100% power; regulating rod is back to original position at 0 hours.

At no time during this transient is it necessary or desirable to change the soluble poison concentration unless the shim rods insert past 25% of withdrawn during xenon decay or the regulating rod is not in its upper 10% of stroke after xenon buildup.

If the operator elects to reduce the soluble poison concentration between 4 and 11 hours to keep the temperature up, he will probably have to increase the concentration sometime after 40 hours as the xenon decays out. Reducing soluble poison concentration during the first 20 hours after load reduction is desirable only if the temperature falls below 490 F.

The most difficult point in returning to full power during a 100% to zero power transient would occur at the peak of the xenon poisoning. Figure 3.8-6 and the following tabulation show the effects of returning to full power at the xenon poisoning peak.

8 hours — Xenon poisoning peaks; the throttle is opened and regulated to prevent temperature from dropping below 490 F; the drop in temperature offsets Doppler and void effects.

11 hours — There is a sufficient burnout of xenon poisoning to offset Doppler and void effects for raising the power level to 100% without dropping the temperature below 490 F; temperature starts to rise when the load levels off at 100%.

13 hours — Temperature returns to normal and regulating rod starts in to offset the burnout of xenon poisoning.

15 hours — Xenon levels off as xenon burnout rate equals the xenon buildup rate.

25 hours — Xenon returns to equilibrium for 100% power; regulating rod is back to original position at 0 hours.

Figure 3.8-7 shows a 100 to 60% transient with a startup at 100 hours after shutdown. This transient is summarized as follows:

0 hours — Steam flow is reduced to 60%; regulating rod is partially inserted to offset Doppler and void effects.

3 hours — Xenon poisoning peaks; regulating rod is partially withdrawn to offset xenon poisoning.

30 hours — Regulating rod bottoms and xenon reaches equilibrium for a 60% load.

100 hours — Throttle is opened to 100% steam flow; regulating rod is withdrawn to offset Doppler and void effects.

125 hours — Xenon reaches equilibrium for 100% power; regulating rod is back to original position at 0 hours.

3.8.3. Nuclear Instrumentation System

3.8.3.1. General

The nuclear instrumentation system provides reactor neutron level information to the operator and to the reactor safety system from source level to 150% of full power. The system consists of three ranges with a minimum overlap of one decade between ranges. (See Fig. 3.8-8.)

Two channels in both the source and the intermediate ranges ensure reliable operator data and safety actions. Three channels in the power range reduce the number of false shutdowns.

3.8.3.2. Detectors

The detectors for each of the three ranges are located in thimbles which extend from flanges on the face of the containment vessel to positions around the circumference of the reactor vessel at the core level. With this arrangement, containment access is not required for servicing or for replacing the detectors. Also, detector lead running and shielding are simplified.

3.8.3.3. Source Range Channels

The source range consists of two pulse counting channels using BF_3 proportional counters as detectors. These channels indicate and record the neutron log count rate from 1 to 10^6 counts per second, and indicate the reactor period from − 30 through infinity to + 3 seconds. A short positive period sounds the annunciator. A trip signal from the intermediate range cuts off the high voltage to the proportional counters before the neutron flux rises to the level of detector burnout. The same circuit automatically restores the high voltage as the neutron flux decreases through the trip point.

3.8.3.4. Intermediate Range Channels

The intermediate range consists of two logarithmic channels using compensated ion chambers as detectors. These channels indicate and record ion current from 10^{-10} to 10^{-4} amperes, and indicate the reactor period from − 30 through infinity to + 3 seconds. The log N signal actuates a bistable circuit to cut off the high voltage supply to the source range detectors. The output of a short period bistable circuit in the period signal goes to the reactor safety system. A short period annunciator sounds prior to the safety system action to permit operator corrective action. The period safety action, but not the annunciator, is manually bypassed by a start-run switch above 10% of full power where temperature and void feedbacks preclude period accidents. The output of a high level bistable circuit in the log N signal goes to the reactor safety system for level safety action during a startup below 10% power. A high level annunciator sounds prior to safety system action to permit operator corrective action. The level safety action and the annunciator are manually bypassed by a start-run switch above 10% power.

3.8.3.5. Power Range Channels

The power range consists of three linear channels using uncompensated ion chambers as detectors. These channels indicate and record the linear power level from 0 to 150% of full power. The linear power signals are auctioneered to provide high and low flux limit signals to the reactor control system.

Ship's motion in the vertical plane will produce periodic or transient variations in the neutron flux around the steady-state flux value. This is especially true at full power where the void fraction contribution to reactivity is greatest or at partial power under natural circulation conditions. The transient flux peaks may exceed the normal overflux scram setting and still not have sufficient time duration to produce unsafe thermal peaking in the core. A flux-time integrator modifies the flux inputs to the safety system to make them indicative of thermal peaking in the core.

The overflux trip setting is a direct function of the number of primary pumps being operated. The high flux level bistable trip-set point is modified by inputs from bistables which monitor the operating status of each primary pump. Dangerous thermal conditions in the core do not appear until sometime after the loss of one or more primary pumps. The loss of one or more pumps is annunciated immediately, but the reduction of the overflux scram level is delayed to permit sufficient time for the operator to reduce the plant load to a safe level consistent with the number of pumps in operation. An alternate design would include the automatic reduction of plant load.

Normal operation of the reactor over the normal maneuvering transients requires the power to be regulated with the center control rod. The six shim rods are totally withdrawn. Power peaking in the core precludes operation at 100% rated power with the shim rods inserted into the core. Since the shims are operated manually, administrative rules will prevent the shims from being inserted at full power. Limit switches on the shim rods reduce the overflux trip point to a power level which is safe regardless of the rod configuration when the shim rods are inserted.

3.8.4. Reactor Safety System

The reactor safety system protects the reactor against conditions which could immediately or ultimately damage the reactor or its auxiliaries. (See Fig. 3.8-9.) Two protective actions are provided — rod stop and scram. Rod stop prevents all withdrawal signals from reaching the seven control rod drives, and is used for those conditions which may be approaching scram points but do not justify immediate scram action. It does not block rod insertion commands. Scram releases all seven control rods from their drives so they fall by gravity into the core and reach the bottom within two seconds. Scram action is backed by the automatic rundown of the rod drives.

All safety system actions will be preceded by annunciators indicating the source of the impending safety system action to permit operator corrective measures. Two scram amplifiers permit the system to be tested without shutting down the reactor.

The inputs to the safety system originate in the nuclear instrumentation system and the process instrumentation system. Due to the simplicity of the CNSG very few inputs to the reactor safety system are required. The inputs to the safety system are given below.

3.8.4.1. Nuclear Instrumentation

Scrams

1. High neutron level from power range channels.
2. High neutron level from intermediate range channels (below 10% power).
3. Short period from intermediate range channels (below 10% power).

Rod Stop

Short period from intermediate range channels (below 10% power).

3.8.4.2. Process Instrumentation

Scrams

1. High primary pressure.
2. Low primary pressure (locked out during warmup).

3. Low reactor water level.

Rod Stop

High primary pressure.

3.8.4.3. Miscellaneous

Scrams

1. Manual.
2. Loss of vital bus power.

A coincidence circuit on the power-range over-flux trips requires the coincidence of two out of three inputs before scram action can occur. Auctioneering may be selected by a switch on the coincidence circuit. All other signals require only a single input for safety action.

3.8.5. Process Instrumentation System

3.8.5.1. General

The process instrumentation system controls the operation of the process systems and supplies input signals to the reactor control and safety systems. No instrumentation and controls are provided to control the ship's conventional equipment and systems except where the controls affect the operation of the reactor. Local indications and controls are used except where remote indications, controls, and alarms are mandatory to operate the auxiliaries from the NSGCP. An operational description of the instrumentation provided for the variables of the process systems is described below.

3.8.5.2. Primary System

The primary system (including the pressure vessel) is a combination heat sink for the reactor, a heat exchanger for the secondary system, and a pressurizer for reactor control. It can absorb sizeable load changes without upsetting the operation of the process systems. This inherent safety feature eliminates fast and elaborate controls to maintain safe operations. However, the following measurements are made to provide information and control for the operation of the system and the power plant.

Pressure

The reactor vessel pressure is measured by two electrical pressure transmitters. The pressure signals, indicated on the NSGCP, are alarmed for high and low pressures, are transmitted to the reactor control system for reactor power level control, and are transmitted to the safety system for high and low pressure scram.

Temperature

Thermocouples in each primary pump measure the pump motor operating temperature. The temperatures, indicated on the NSGCP, actuate the alarms for excessively high pump-motor temperatures.

Thermocouples connected to the piping of the suction side of each pump measure primary water temperature as it circulates from the reactor core to the pumps and back through the core. The signals, indicated on the NSGCP, permit the operator to limit manually primary water temperature buildup during startup to approximately 40 F per hour until operating temperatures are reached.

Level

The level in the reactor vessel is sensed by several magnetostrictive probes installed in stillwells attached parallel to the innermost steam generator downcomers. These probes do not detect a continuous level, but they detect and indicate the level at the exact fixed position of each probe as the water level touches the face of each respective probe. The level signal is indicated on the NSGCP by a graphic arrangement of lights representing the exact position of each respective probe and level in the reactor vessel. High and low alarms and low level scram are actuated by the selected probes.

System Operation

At plant startup, the primary pumps are actuated from the NSGCP. The reactor plant and primary system then are manaually controlled until the primary system pressure and temperature reach normal operating conditions. Once pressure and temperature reach steady-state operating conditions, the controls can be switched to automatic operation. At this time, primary pressure is used as the index to control the control rod drive mechanisms and the reactor power level.

Pressure buildup in the reactor vessel (primary system) is indicative of the reactor output. To control the reactor (power level) by primary system pressure without continuous and minute motions of the control rods, a pressure deadband of ± 50 psi of normal set point pressure is used. Primary system pressure varies over this deadband without initiating control rod motion. If the primary pressure exceeds the upper limits of the deadband, the reactor control system actuates control rod insertion and cuts back on the reactor power level. Conversely, if primary pressure decreases below the lower limits of the deadband, the reactor control system actuates control rod withdrawal and increases the reactor power level.

The piping to each primary pump is equipped with a backflow check valve with a small hole in the plate so that high temperature water circulates through the pump which is stopped during normal operation. This eliminates the need for a coldleg interlock if a down pump is restarted during steady-state operating conditions.

3.8.5.3. Chemical Control System

The chemical control system provides water and soluble poison injection into the primary system in response to reactor control requirements; provides water makeup and soluble poison injection into the containment; controls plant liquid effluent discharge; samples the water contained in the primary, secondary, and containment systems; and controls the secondary system water quality and the purification of plant water makeup.

The chemical control system controls are manually actuated from the NSGCP. Manual operation is used because the small capacity for injecting water and soluble poison into the primary system and containment vessel makes the effects on reactor control when small equipment is operated to perform the above functions. Also, since the necessity to make control changes develops slowly, manual operation for the electric motor pumps and stop valves is satisfactory. Measurements made in the system for information and control are described below.

Level

Level of the three interconnected condensing tanks is measured by several magnetostrictive probes installed in the walls of one of the three tanks. These probes do not detect continuous level, but they detect and indicate the level at the exact fixed position of each probe as the water level touches the face of each respective probe. The level is indicated on the NSGCP by a graphic arrangement of lights representing the exact position of each respective probe and level in the selected condensing tank. High and low level alarms are provided on the NSGCP.

Pressure

The pressure of the interconnected condensing tanks is measured by an electrical pressure transmitter and is indicated along with a high pressure alarm on the NSGCP.

Flow

Separate feed and bleed flow measurements are made by turbine-type flow meters and are indicated on the NSGCP.

Indicating Lights and Switches

The selector switches on the NSGCP used to actuate the electric motor pumps and stop valves are equipped with color coded lights to indicate the state of equipment operation.

3.8.5.4. Containment System

The containment system operates during steady-state operation without any control action except for local manual operation of the sea pumps, gas blowers, block valves, etc., when required. It is designed to assure safe environmental conditions outside the containment vessel without automatic control action after a maximum credible accident. Measurements made in this system are described below.

Level

The containment water level is measured by an electrical differential pressure transmitter and is indicated along with high and low-alarms on the NSGCP.

Pressure

The pressure of the gases above the water level is measured by an electrical pressure transmitter and is indicated on the NSGCP.

Flow

Sea water flow to the sea water heat exchangers is measured by an electrical differential pressure transmitter and is indicated locally. A low flow alarm is provided on the NSGCP.

Temperature

The containment water temperature is measured at thirty separate points around the containment vessel by separate swaged thermocouples inserted through the containment vessel walls. The measurements are transmitted to a multi-point recorder on the NSGCP which actuates the high temperature alarms.

3.8.5.5. Secondary System

The secondary system consists of the feedwater and the high pressure steam systems from the feedwater pumps to the main turbine, the turbine generator, and the steam bypass valve. The feedwater piping branches to three separate headers downstream from the feedwater flow nozzle. This piping enters the reactor vessel and becomes three separate once-through steam generators to produce high pressure superheated steam. The three separate steam headers leaving the reactor vessel unite forming the main steam header to supply the superheated steam for plant and hotel loads. The three separate once-through steam generators are valved so that the units can be separately isolated if a tubing rupture occurs inside the reactor vessel. The instrumentation measurements and the controls for the feedwater through the high pressure steam system are described below.

Pressure

Feedwater pressure is measured immediately upstream from the feedwater regulator and is indicated on the NSGCP.

Differential Pressure

An electrical differential pressure transmitter measures the pressure drop across the feedwater regulator. The signal is transmitted to a controller which regulates the feedwater pump speeds to provide the head needed to maintain a constant pressure drop across the feedwater regulator for all propulsion and plant loads.

Steam Pressure

Steam pressure is measured by an electrical pressure transmitter located in the main steam header upstream from the turbine throttle control valve. It is indicated on the NSGCP where high-low alarms are provided. The signal is also transmitted to the three-element feedwater controller.

Feedwater Flow

Feedwater flow is measured by an electrical differential pressure transmitter connected across a flow nozzle located downstream from the feedwater regulator (upstream from the steam generator). The flow signal is indicated on the NSGCP where high-low flow alarms are provided. The signal is also transmitted to the three-element feedwater controller.

Steam Flow

Steam flow is measured by an electrical differential pressure transmitter connected across a flow nozzle located in the main steam header. The flow signal is indicated on the NSGCP where high-low flow alarms are provided. The signal is also transmitted to the three-element feedwater controller.

Feedwater Temperature

Feedwater temperature is measured by thermocouples installed in the feedwater piping upstream from the feedwater regulator. The signal is indicated on the NSGCP.

High Pressure Steam Temperature

Thermocouples installed in each steam generator outlet piping measure the high pressure steam temperature from the steam generator. This measurement, indicated on the NSGCP, supplies supervisory information for preparing heat balance data on the separate steam generator. The measurements should indicate the steam temperature (from 4 to 5 F) in the reactor vessel dome. The measurement can be compared with steam table data to check the accuracy of the primary system pressure measurements.

Thermocouples installed in the main steam header upstream from the steam flow nozzle measure the temperature of total steam flow. This is indicated on the NSGCP where high-low alarms are provided. The low temperature alarm set at 470 F alerts the operator to reduce main steam pressure to maintain superheat in the steam to the turbine until the cause for the low temperature is corrected. An alternate design would automatically reduce the main steam pressure under low temperature conditions.

Radiation Monitoring

A radiation monitoring unit located outside the containment vessel senses for radioactive gases in each steam generator outlet piping. A radiation level indicates tube leakage from the primary to the secondary system. The signal is indicated locally, and high level alarms are provided on the NSGCP.

System Operation and Controls

During plant startup, the electric motor feedwater pump is manually started from the control room when the primary system pressure reaches 240 psia. Feedwater is circulated through the steam generators, the steam bypass valve, and the deaerator or condenser back through the feedwater pump. System pressures and temperature are built up by and simultaneously with the primary system temperature buildup. As system minimum operating temperatures are reached, the turbine generators are rolled (by turbine manufacturer instructions) and synchronized, the main turbine feedwater pump is started, the electric motor feedwater pump is stopped, and the steam bypass valve is closed to divert the available steam to the operating turbines and hotel loads. Once the turbines are operating, the plant load controls are switched to automatic. The three-element feedwater control system is the major control system used to follow plant load changes. It uses a combination of the main steam pressure, steam flow, and feedwater flow signals to control the feedwater regulator proportionately with the load changes. Steam flow is the main index of boiler load. Steam pressure is an index of proper matching between steam flow and feedwater flow. As the load changes, the system controls the feedwater flow causing it to equal the steam flow plus or minus a bias in feedwater flow proportional to the deviation in steam pressure from 412 psi. For the normal load transients, feedwater flow leads steam flow by 5 to 10% to maintain a constant steam pressure.

Interlocks and Trips

The feedwater pumps have governor protection to limit pump speeds to deliver maximum flow rates of 120% of the flow required for 100% reactor power level.

The steam bypass valve is interlocked with a coincidence of minimum steam pressure and temperature signals to trip all turbines and to start automatically the electric motor feedwater pump if the coincidence of those signals drops to 150 psi and 400 F.

3. 9. Auxiliary Systems

3. 9. 1. Chemical Control System

This system controls the water chemistry for the entire plant. Soluble poison is to be used to control the excess reactivity added to the core for core lifetime and to provide the temperature defect. Except for the initial plant startup, infrequent maintenance and refueling, cold shutdowns are not expected to occur. The plant will normally operate at some finite power level.

During reactor startup, the chemical control system will be used to inject demineralized water into the reactor vessel. This will reduce the soluble poison concentration until the operating temperature at some power level is attained. Subsequently, water feed and primary coolant bleed will reduce the soluble poison concentration in response to fuel burnup. Primary coolant bleed from the reactor will normally be discharged through a monitored line into the ship's propeller wash. Based on the corrosion product and fission product activities scaled from previous work on the Savannah Upgrading Program (see Section 8.2 of this report), the activity in the primary coolant will be approximately 120 $\mu c/ml$. From the Pritchard Report,* it can be shown that the activity, resulting from the primary bleed discharge into the propeller wash, will be less than the weighed mean partial permissible concentration (ppc) for Zones 3B and 4B after the activity is dispersed by the ship's displacement. While the ship is in port or in restricted waters, the condensing tanks are capable of holding the primary bleed until it can be discharged at sea.

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*See Reference 4 of Section 8. 1.

This system also contains provisions for increasing the soluble poison concentration in the primary system. Throughout most of the core life, a steam bleed from the reactor vessel will provide the required poison reconcentration in the primary system for a shutdown to 200 F. The condensate tanks are able to receive approximately 8000 lb of steam from the primary system. An 8000-lb steam loss from the reactor vessel increases the primary system poison concentration by about 20 to 25%. Near the end of core life, the poison concentration in the primary water is not adequate to shut down the reactor by bleeding steam. Thus, following the steam bleed, a soluble poison injection from the poison supply tank will be used to increase the poison concentration. A soluble poison injection is also used for an emergency shutdown. To prevent reactor startup following a primary system rupture, the containment water is treated with sufficient soluble poison to shut the reactor down cold. The chemical control system provides makeup water and soluble poison makeup to the containment. The requirements for soluble poison makeup will be determined by sampling.

Demineralizers are installed in the condensate line behind the condensate pumps. A bypass around the demineralizers enables any part of the condensate flow to be passed through the demineralizers. Two demineralizers are provided in parallel so that the resins in one can be regenerated while the other is on line. No shielding or special resin handling equipment is required for the demineralizers since only non-radioactive secondary water passes through them. Conventional regeneration equipment is used.

Makeup for the secondary system is obtained from the ship's fresh water tanks and must pass through a demineralizer before entering the main secondary steam. Makeup for the primary system is obtained from the secondary system since, with the once-through steam generators, the secondary water must be maintained at a quality suitable for this purpose.

Contamination from sources within the secondary cycle caused by corrosion or erosion of the materials in the cycle must be minimized. In addition to full or part flow condensate demineralization, oxygen and carbon dioxide are maintained at very low values and pH is kept within limits which will provide satisfactory protection to the materials. This level of pH, usually in the range of 8.5 to 9.5, can be obtained by adding an alkaline volatile material such as ammonia, cyclohexylamine, or morpholine.

Good mechanical deaeration will reduce dissolved oxygen in the feedwater to less than 0.007 ppm. To achieve even lower values and to provide against upsets in deaerator operation, supplementary chemical deaeration employing hydrazine is used. Hydrazine is a volatile material and does not contribute to the dissolved solids content of the feedwater.

3. 9. 2. Relief System

3. 9. 2. 1. General Description

The relief equipment protects the reactor vessel, the condensate tanks, the containment vessel, the steam generators, the secondary system equipment, and the containment heat exchangers from excessive pressure.

Reactor Vessel Relief Equipment

Relief valves protect the reactor vessel from excessive pressures. They relieve the steam generated by that part of the decay heat which is not removed through the vessel walls. It is assumed that the high pressure and the high temperature scrams function properly. The primary relief valves discharge to the condensate tanks in the chemical control system.

Steam Generator Relief Equipment

If a steam generator is filled with water and is isolated, the relief valve protects it from excessive pressure. The water from these relief valves is discharged under water to the containment vessel. The set pressure exceeds that of the primary valves to prevent the release of primary water to containment water if a tube has ruptured.

Secondary Relief Equipment

The secondary relief valves relieve the steam generated when the turbines are down and the reactor is operating at maximum power. These valves discharge to the atmosphere.

Containment Relief Equipment

This equipment operates only if the ship is stranded out of the water. The decay heat is transferred from the reactor to the containment water. After 37 hours (assuming no heat loss from the containment), the pressure reaches 50 psig. The relief valves are capable of relieving the steam generated from decay heat after 37 hours of decay, and they discharge to the atmosphere.

Containment Heat Exchanger Relief Equipment

If a containment heat exchanger is isolated, the relief valve protects it from excessive pressure. The relief valves discharge to the sea water overboard-discharge line down stream of the last block valve. The set pressure of these valves exceeds the maximum containment pressure to prevent the release of containment water.

Condensate Tank Relief Equipment

If a condensate tank is full, the relief valves are able to relieve at a rate equal to that at which steam or water is discharged to the tank.

3.9.2.2. Operating Information

Normally, all valves are closed. They have the following characteristics:

Equipment protected	Valve no.	Set pressure psig	Back pressure psig	Flow, lb/hr
Reactor	1	1100	400	
	2	1150	400	Total both valves
Steam generator	1	1220	50	(10 gpm)*
	2	1220	50	(10 gpm)
	3	1220	50	(10 gpm)
Secondary	1	495	--	336,000
	2	515	--	Total both valves
Containment	1	50	--	1,280
Condensate tanks	1	400	50	23,000 or (67 gpm)
Containment heat exchanger	1	65	--	(0.05 gpm)
	2	65	--	(0.05 gpm)
	3	65	--	(0.05 gpm)
	4	65	--	(0.05 gpm)

*Numbers in parentheses are water; all others are steam.

3.9.3. Containment System

3.9.3.1. Description

The containment system consists of a containment vessel and cooling and ventilating equipment.

Containment Vessel

The containment vessel contains the reactor, the control rod drives, the primary pumps, and the condensate tanks all submerged under water. The reactor vessel is completely covered until the ship exceeds an 80-degree vertical roll. The water provides the cooling for the primary pump motors, the control rod drive motors, the primary water sample, and the condensate tanks. The water also removes heat from the reactor through the lead covered vessel walls. The total heat loss through the vessel walls and the control rod drive housing and other connecticns is limited to approximately 2.2×10^6 Btu/hr when the containment water is at 110 F. This heat loss is sufficient to prevent the popping of the primary relief valves when no other means of removing decay heat is available.

Following a reactor vessel rupture, the energy stored in the primary system water is released to the containment. This energy is absorbed by the containment water while it raises the temperature by approximately 35 F. The air space is sufficient to limit the pressure rise caused by the addition of primary system water and by expansion of the containment water.

Following a reactor vessel rupture, the energy stored in the primary system water is released to the containment.

If the primary system ruptures at the bottom of the reactor vessel, virtually all the primary water will leave the system. This raises the containment pressure to 50 psig and the temperature of the containment water to 145 F.

If the primary system ruptures near the top of the reactor vessel, some of the water will remain in the primary system. Then, the containment pressure will increase to only 15 psig. In this case, the reactor, which is scrammed when the primary system ruptures, will become critical again when the reactor temperature declines to about 240 F. It will again become subcritical when the water level in the reactor vessel decreases to about the core midplane. Due to the heat generated while the reactor is critical, the containment water temperature in this case is approximately 153 F.

The temperature and pressure in the gas space will be approximately 255 F and 50 psig at the end of 37 hours when the ship is beached if there is no heat removal from the containment water.

The containment water also serves as a portion of the biological shield.

Decay Heat Removal

Decay heat is removed from the reactor during normal shutdown and during an emergency. During normal shutdown, decay heat can be removed by:

1. The dissipation of heat from the primary water through the reactor vessel walls and head.
2. Simultaneously with (1) above, heat can be removed through the steam generator, and
3. at primary water temperatures of less than 150 F by using dockside circulation and heat exchanger equipment.

To provide for emergency decay heat removal, heat will be continuously dissipated through the head and walls of the reactor vessel during normal operation and during the emergency. This method has the advantage of a system that is capable of operating at all times without depending on mechanical equipment or on the human element to put it into operation. The obvious disadvantage is the continuous loss of 1% of the rated reactor power at all times. However, since this mechanism has a 100% degree of reliability, the disadvantage is compensated.

A thermal resistance design for the reactor vessel required the consideration of several factors. Some of the more important of these considerations were

1. the limitation of the ΔT that could be imposed on the reactor pressure vessel,
2. the ability of the insulating material to render 100% protection of the vessel from the water surrounding it,
3. the ability of the insulating material to withstand thermal and radiation damage,

4. the ability of the insulating material to yield predictable and constant heat transfer data, and
5. the economics associated with any scheme proposed.

A number of proposals, including banded and insulated vessels, were examined. Of these, the bonded lead vessel appeared to satisfy the majority of the requirements.

With the given area of the reactor vessel available for heat transfer, it was found that an 8-inch layer of lead would yield the desired heat flow. A reduction in lead thickness may be possible by using properly spaced thermal resistances on the outside surface of the lead. This would require further study and development.

Cooling Equipment

The containment water is cooled by four vertically mounted, single-pass shell and tube heat exchangers. During normal operation, these heat exchangers employ natural convection on the tube side for containment water cooling, and forced circulation of sea water on the shell side. They will be fabricated of a 70-30 copper-nickel alloy and will be of an all welded construction to prevent sea water leakage into the containment.

Since sea water leakage into the containment is conceivable, all materials within the containment in contact with water will be constructed so chloride ions will cause no damage. Because of activation, it will not be desirable to contaminate the containment water with sea water; therefore, maintenance of a leak-tight system will be required.

During an emergency when all power aboard the ship is lost, the heat exchangers will be able to remove heat from the containment water by natural circulation of sea water on the shell side and natural circulation of containment water on the tube side. This is accomplished by bypassing the two pumps and by using the natural circulation inlet and outlet lines.

Ventilation Equipment

The radioactive gases and hydrogen are controlled in the containment vessel by diluting and stacking. All gases leaving the containment vessel are filtered and monitored before stacking.

3.9.3.2. System Operation

The initial fill of the containment is performed at dockside with demineralized water containing 9370 ppm of boric acid and sufficient potassium hydroxide to give a pH of 8.5 to 9.5. At a filling rate of 150 gpm, it should take approximately five hours to fill the containment vessel. At the normal level of 33 feet, the water covers the rod drive housings by approximately 3 feet.

During normal operation, a sea water circulating pump supplies 500 gpm of sea water to the containment heat exchangers (approximately 125 gpm per heat exchanger). The exact flow to each heat exchanger is governed by the cooling required for each primary pump motor. A flow balance can be achieved by observing the temperature of each primary pump and by manually throttling the sea water outlets from each heat exchanger.

If all power on board is lost (dead ship), the operator should perform the following:

1. open the natural circulation inlet valve,
2. open the natural circulation outlet valve,
3. isolate the sea water circulating pumps, and
4. determine the ship's free board and prepare to bring the ship to the full load line if the ship is not at this position at the time of power failure.

A return to normal operation can be accomplished by closing the natural circulation inlet and outlet valves, and by putting the pump on line again. If both sea water circulating pumps are lost without the loss of ship's power, the bypass line around the sea water circulating pump or the natural circulation lines will be opened. The sea water outlet temperatures will show the operator the best method to obtain the highest heat removal rate. This operation will extend the containment operating time.

During normal ship operation, the operator will attempt to maintain one atmosphere of absolute pressure in the containment gas space. This will be done by venting the gases through an absolute filter and a 1000-cfm fan, and then by routing the gases up the stack. A bypass line and a radiation monitor allow the operator to obtain radiation levels of the gases before venting.

Drainage of the containment water is accomplished by attaching a line from dockside to a 3-inch connection on the containment. The connection is provided with a blind flange and a gate valve.

3.10. Propulsion Plant

The design of the propulsion plant, performed by the Maritime Administration, is similar to the conventional merchant ship steam plant. A high pressure and a low pressure turbine are connected in parallel to a double reduction gear to drive a conventional propeller designed to absorb 20,000 shp at about 105 rpm. The low quality or low enthalpy of the steam supply from the reactor (402 psia and 515 F) as compared to a conventional boiler supply (600 psia at 850 F) produces a high steam rate (lb/shp/hr) that requires the use of comparatively large turbines and condenser. The reactor steam, having about 70 F superheat, is piped directly to the manual throttle valve and thence to the high pressure turbine steam chest without moisture separation. In the cross-over line from high to low pressure turbine the excess moisture in the steam is removed mechanically by a separator. Steam is bled from selected stages of the high and low pressure turbines or the cross-over line to heat the feed water, in three stages, to the temperature required for optimum cycle efficiency. The plant is controlled manually by a throttle valve or nozzle control valves. Propulsion plants for both amidship and astern positions were considered. There are two basic differences between the midship propulsion plant and the stern plant: in the length of line shafting, and in the size and shape of the main condenser. The stern plant has a shorter condenser with a larger cross section due to the restricted athwartship clearance. The auxiliary equipment is basically the same for both plants, but is arranged differently to suit space limitations.

Electric power for auxiliary motor-driven pumps and for miscellaneous uses is supplied by two 600 kw - 440 v - ac steam turbine generators. An automatic starting, diesel-driven generator of 75 kw capacity supplies emergency power.

A heat balance for this plant was prepared by Professor Holm of Webb Institute, and was used in the Maritime Administration's plant design. (See Fig. 3.10-1.)

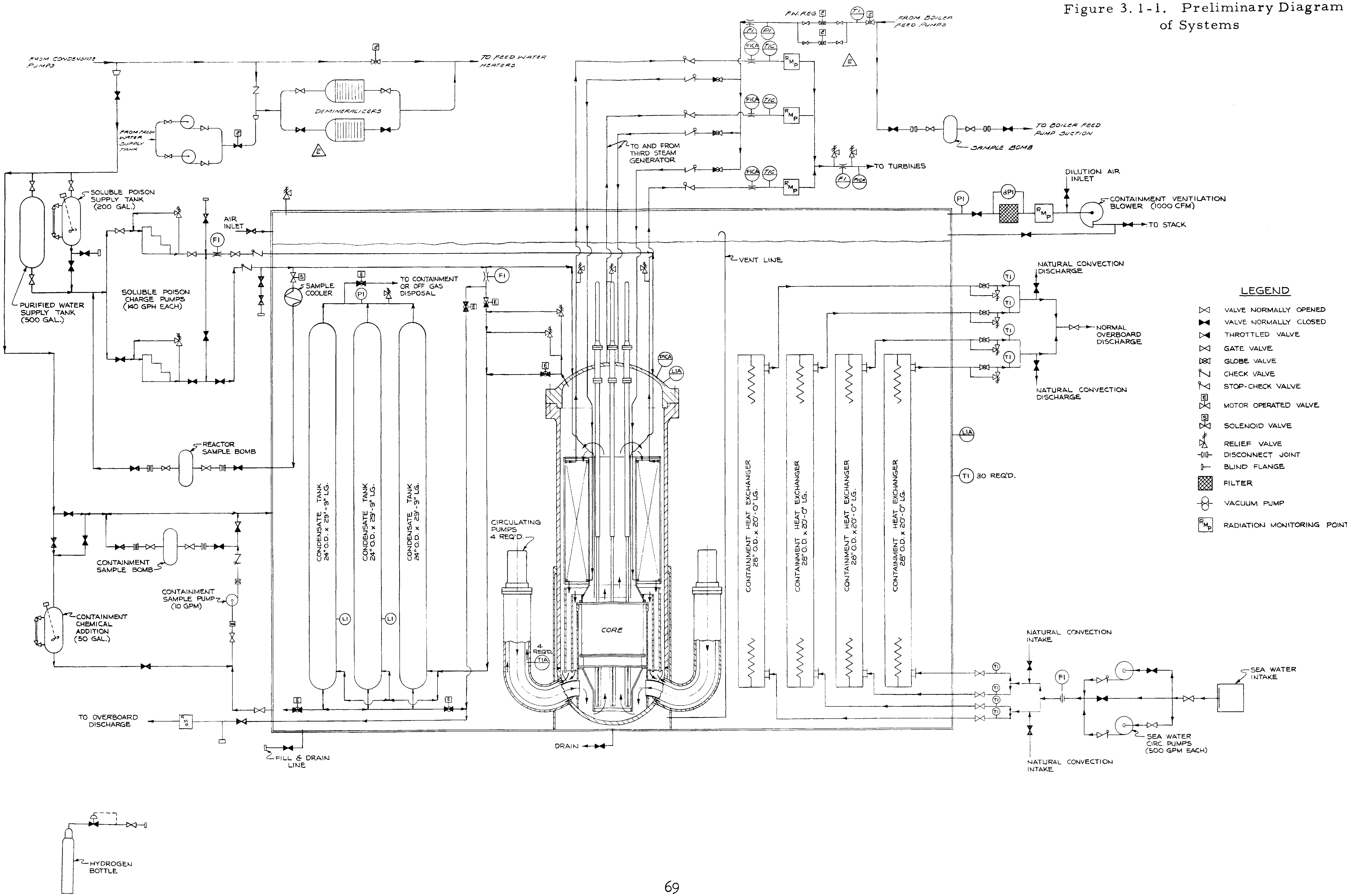


Figure 3.1-1. Preliminary Diagram of Systems

Figure 3.2-1. Arrangement of Reactor and Containment

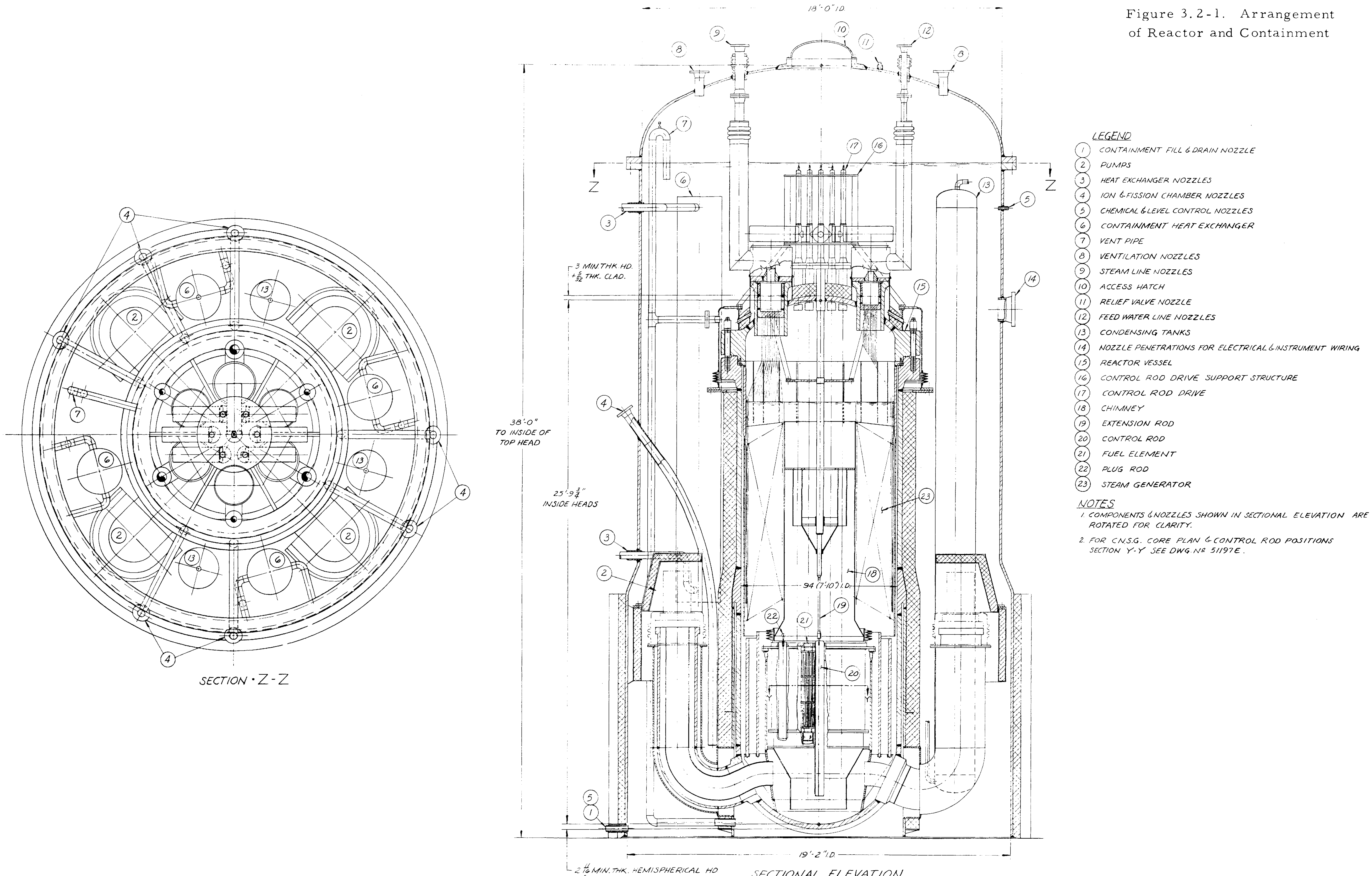


Figure 3.3-1. Containment Pressure Temperature Buildup After a Shutdown (No Heat Removed from the Containment)

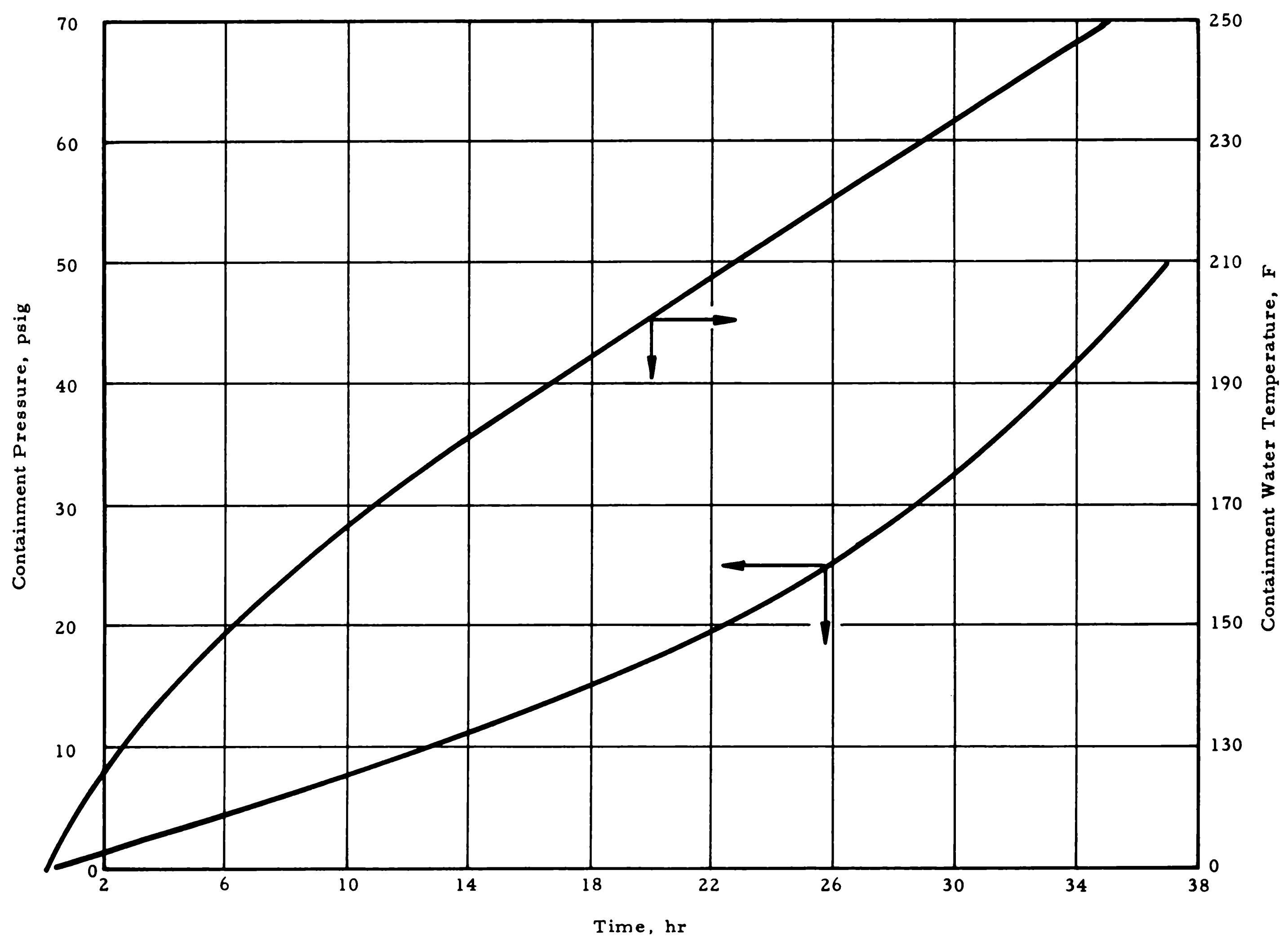


Figure 3.4-1. Gamma Ray Dose Rate at Shield Exterior Vs Thickness of Outer Lead Segment – Side ℄

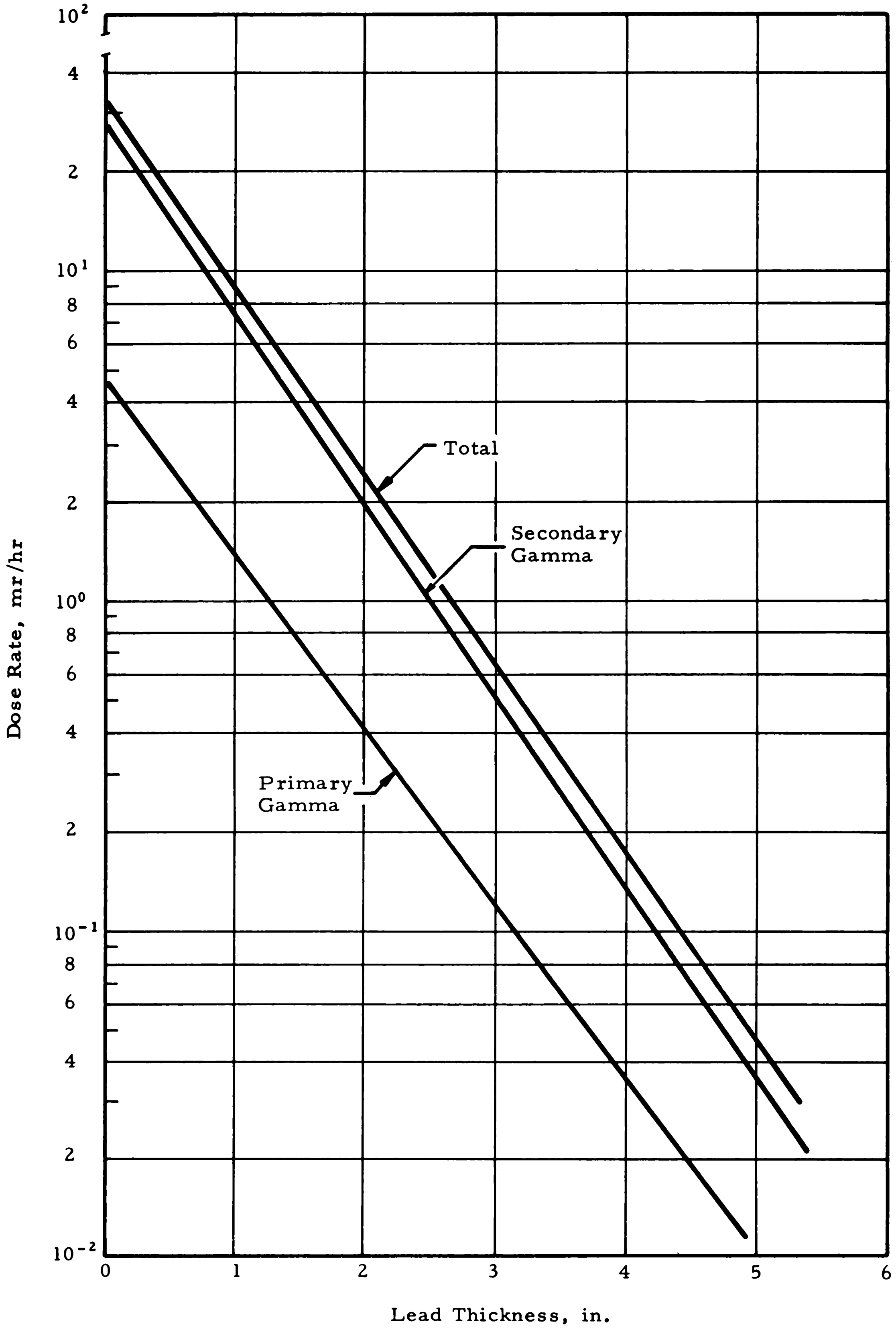


Figure 3.4-2. Two-Group Diffusion Theory Neutron Fluxes – Side $\mathrm{C_L}$

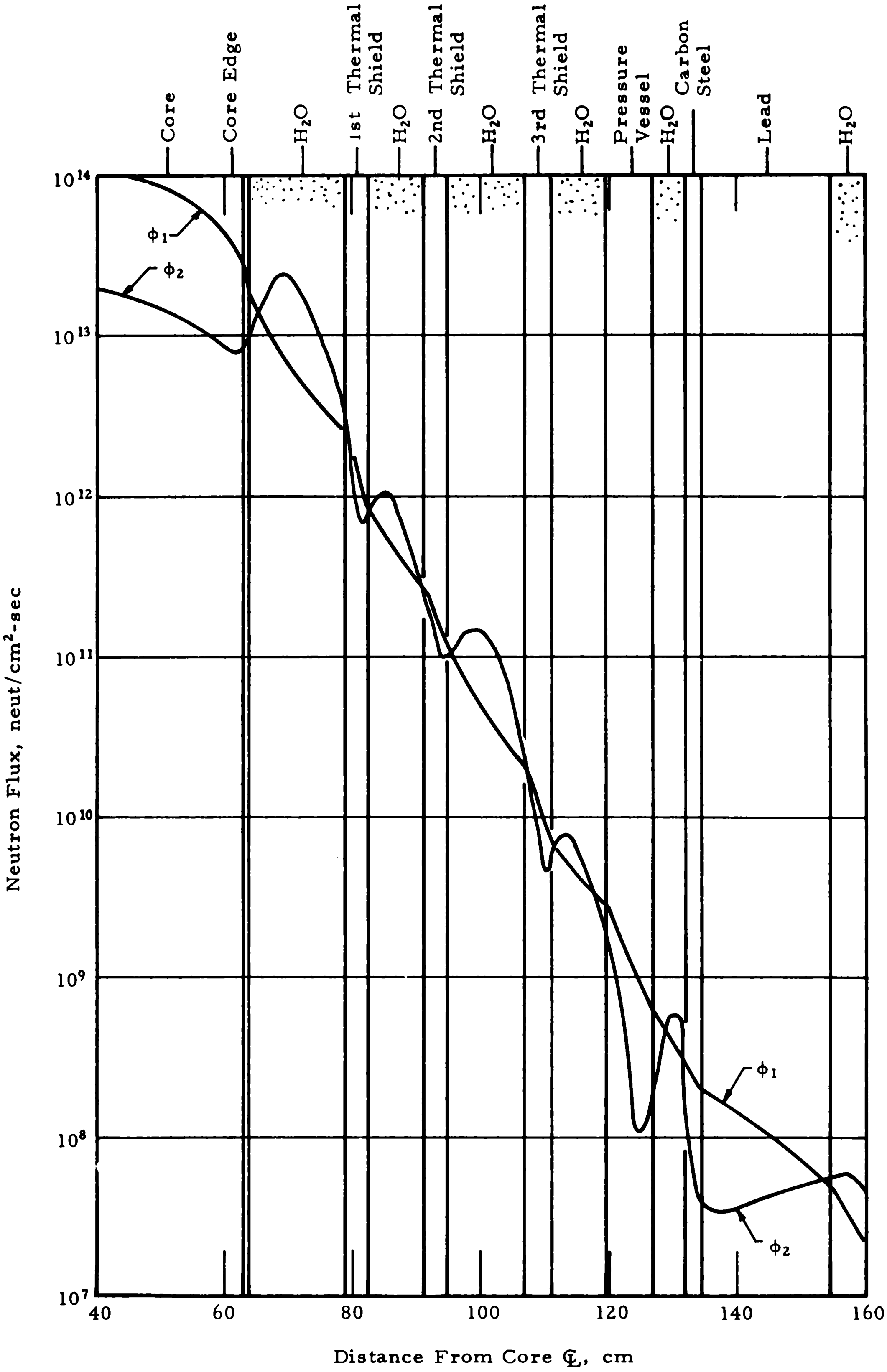


Figure 3.4-3. Fast Neutron Removal Theory Flux – Side ℄

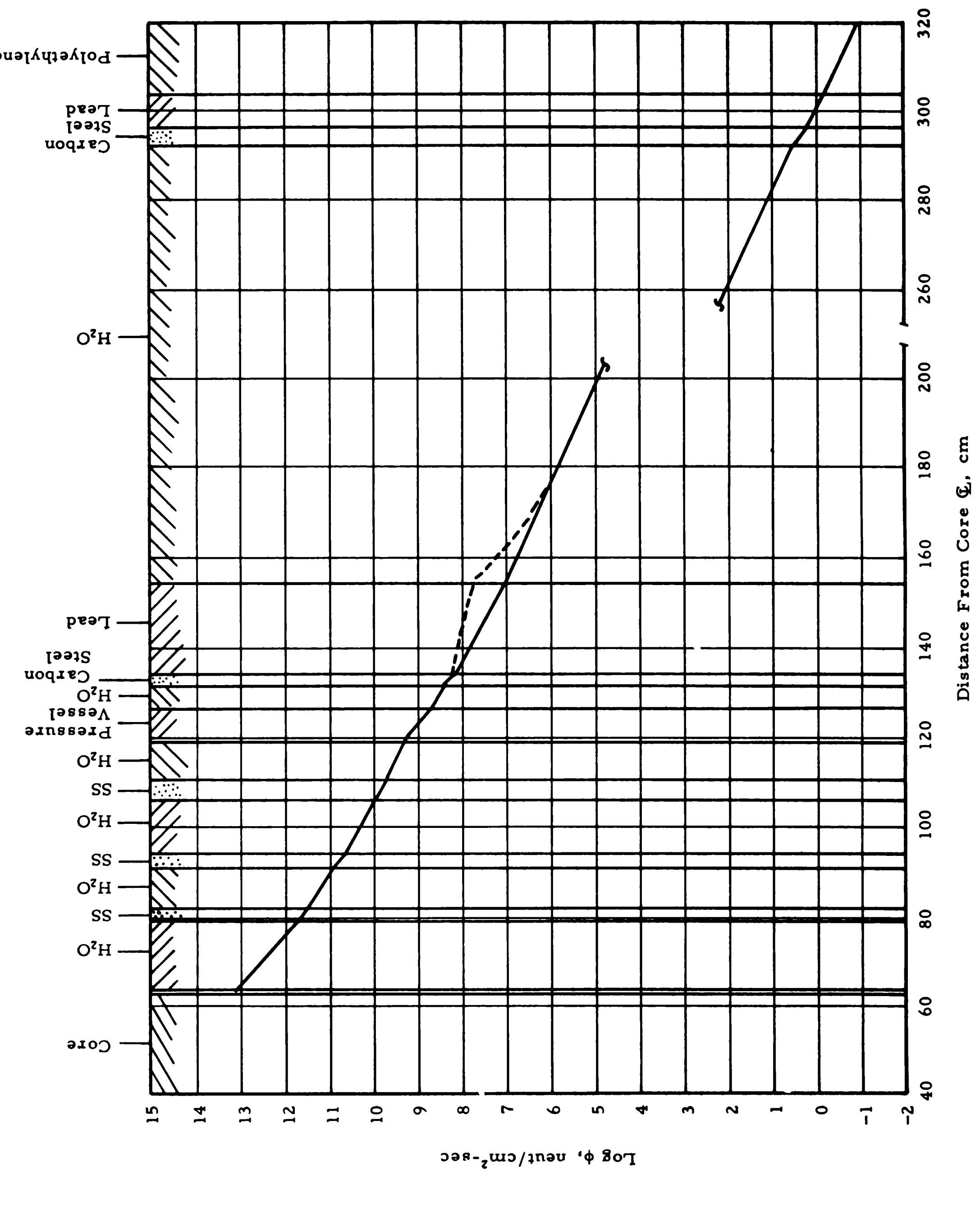


Figure 3. 7. 1-1. Core Plan and Control Rod Positions

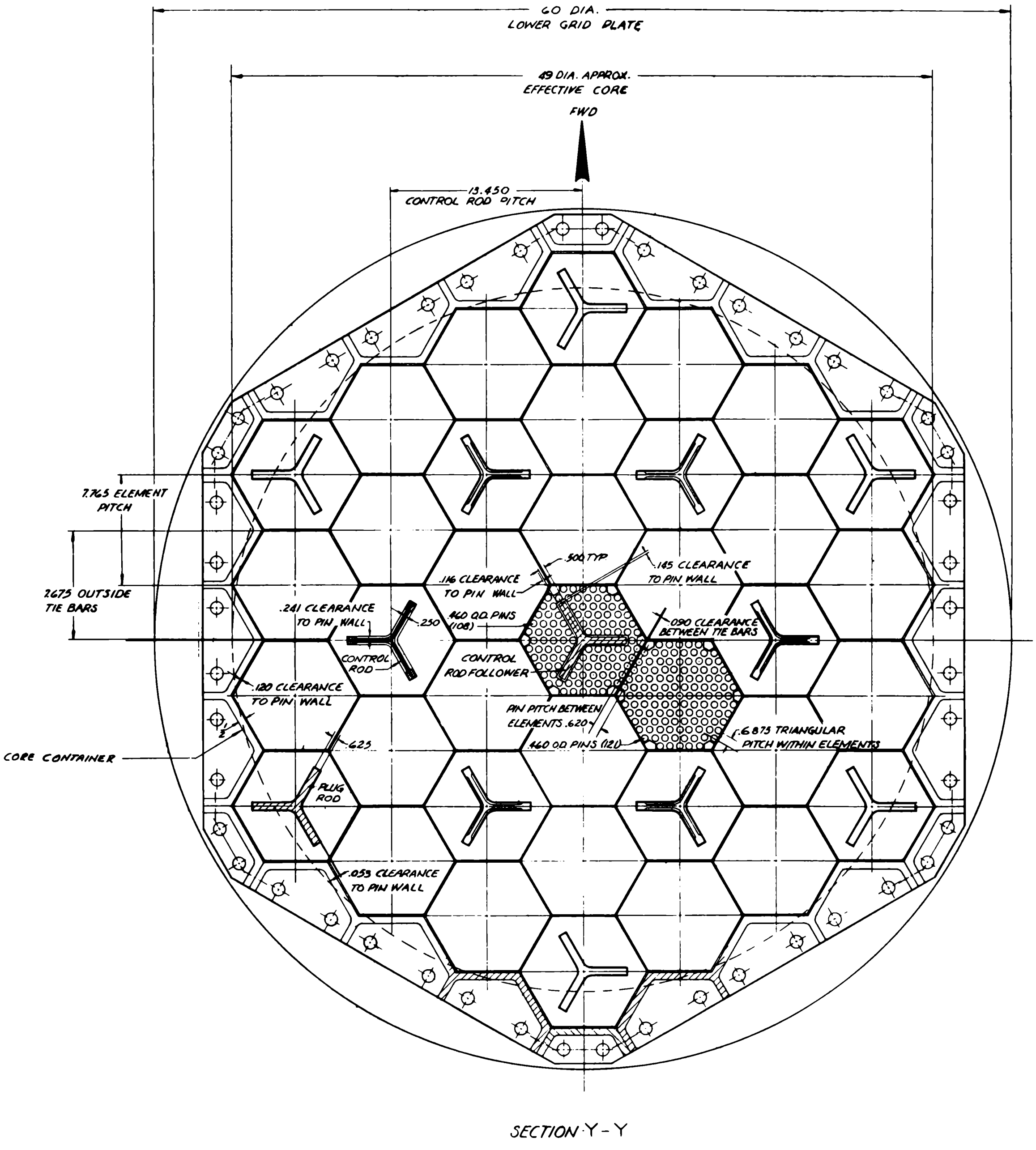


Figure 3.7.2-1. Radial Power to Average Power Profile at Beginning and End of Equilibrium Cycle

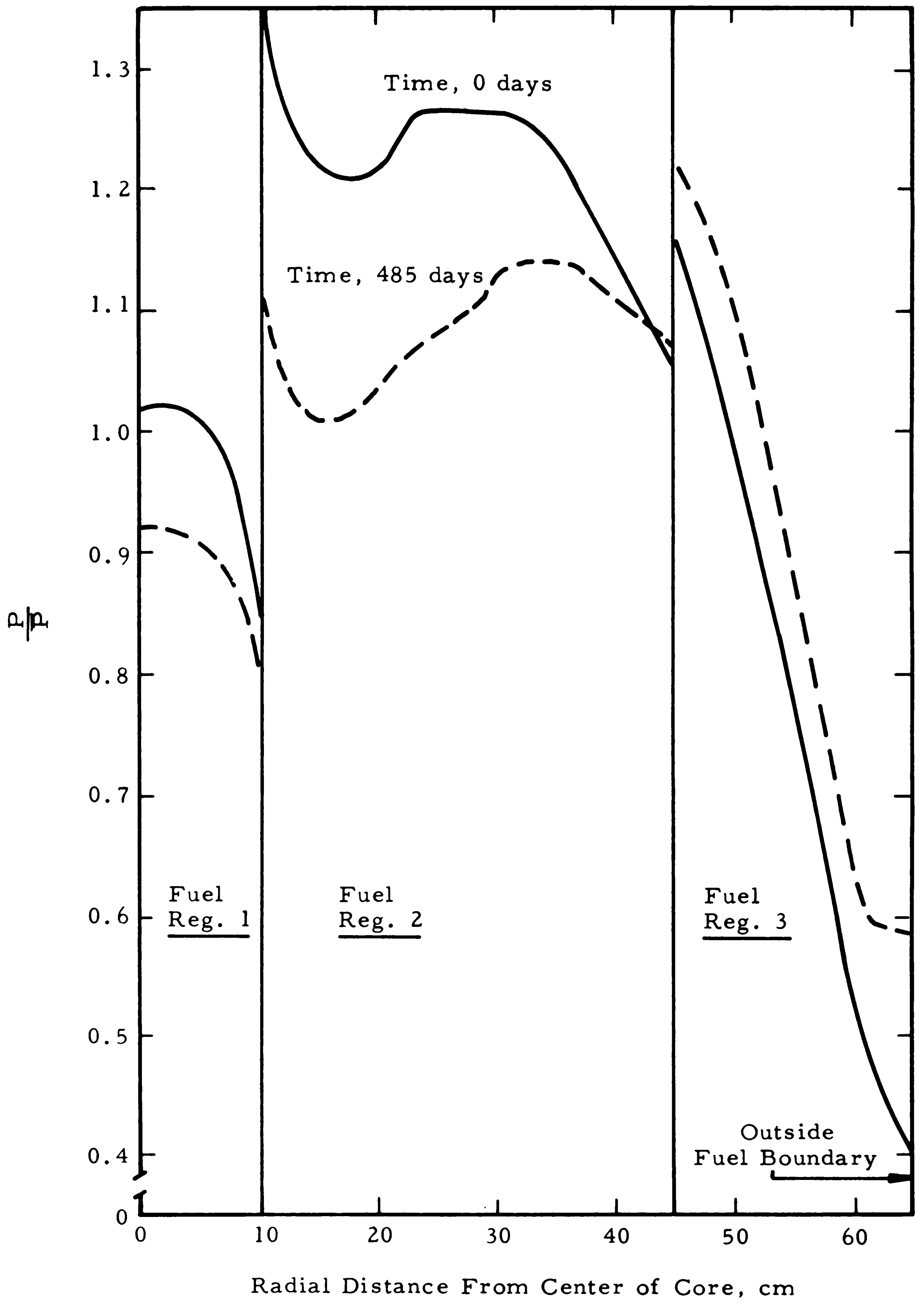


Figure 3.7.2-2. Gross Power to Average Power Contour
Equilibrium Cycle – Start of Life

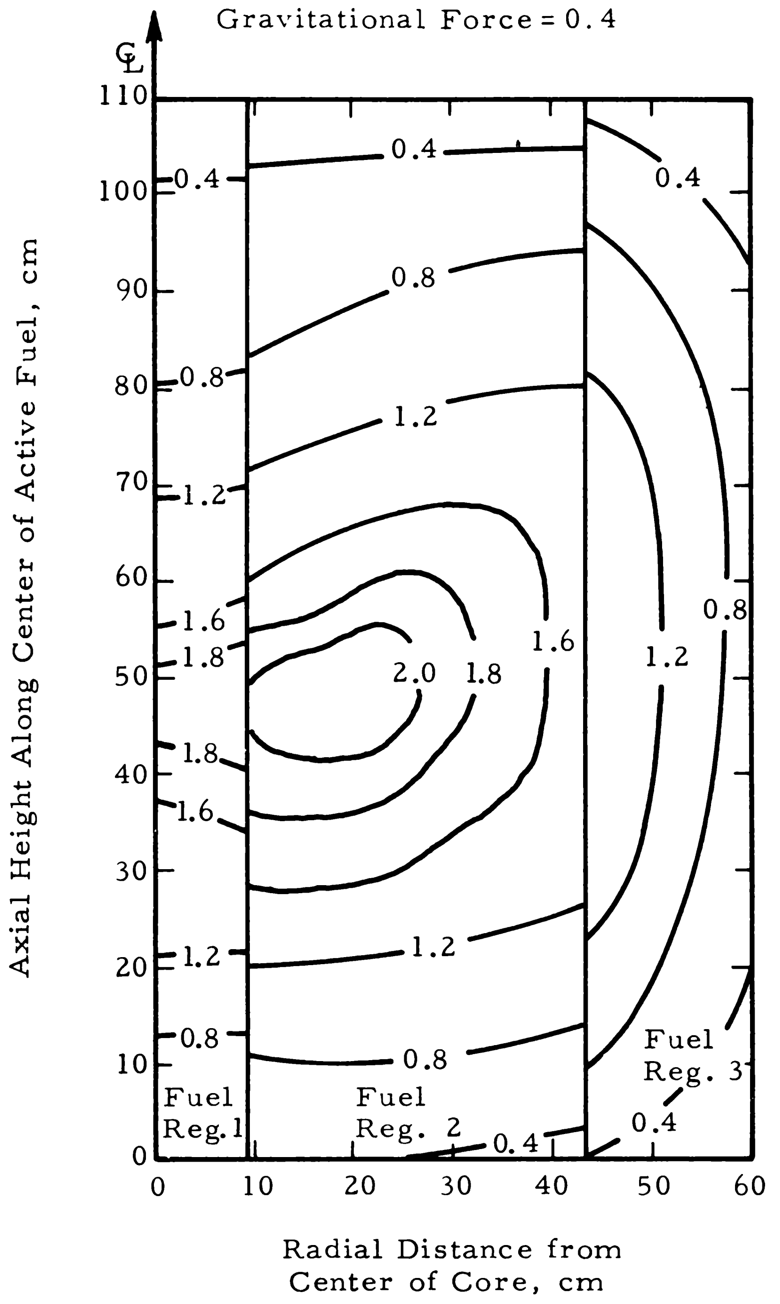


Figure 3.7.2-3. Gross Power to Average Power Contour Equilibrium Cycle – Start of Life

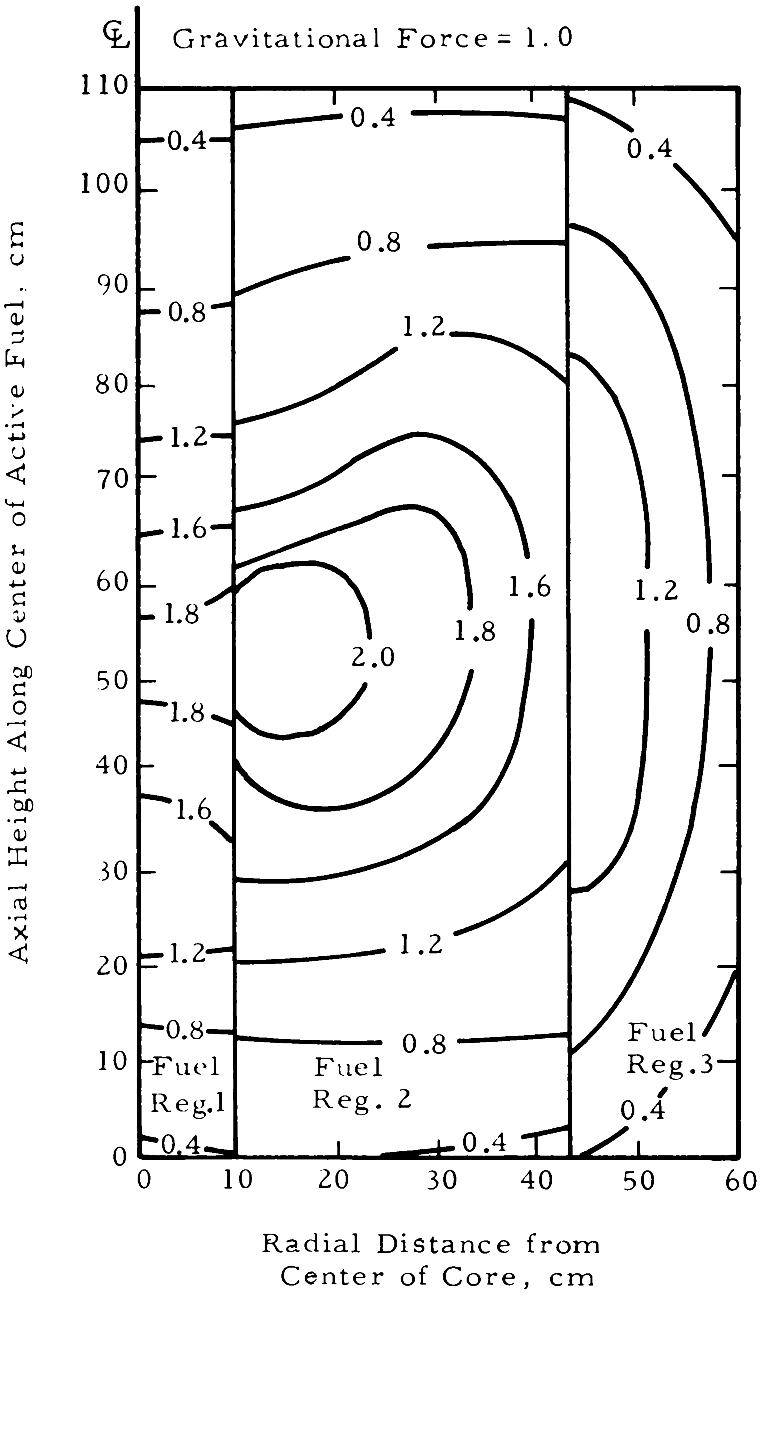


Figure 3.7.2-4. Gross Power to Average Power Contour
Equilibrium Cycle – Start of Life

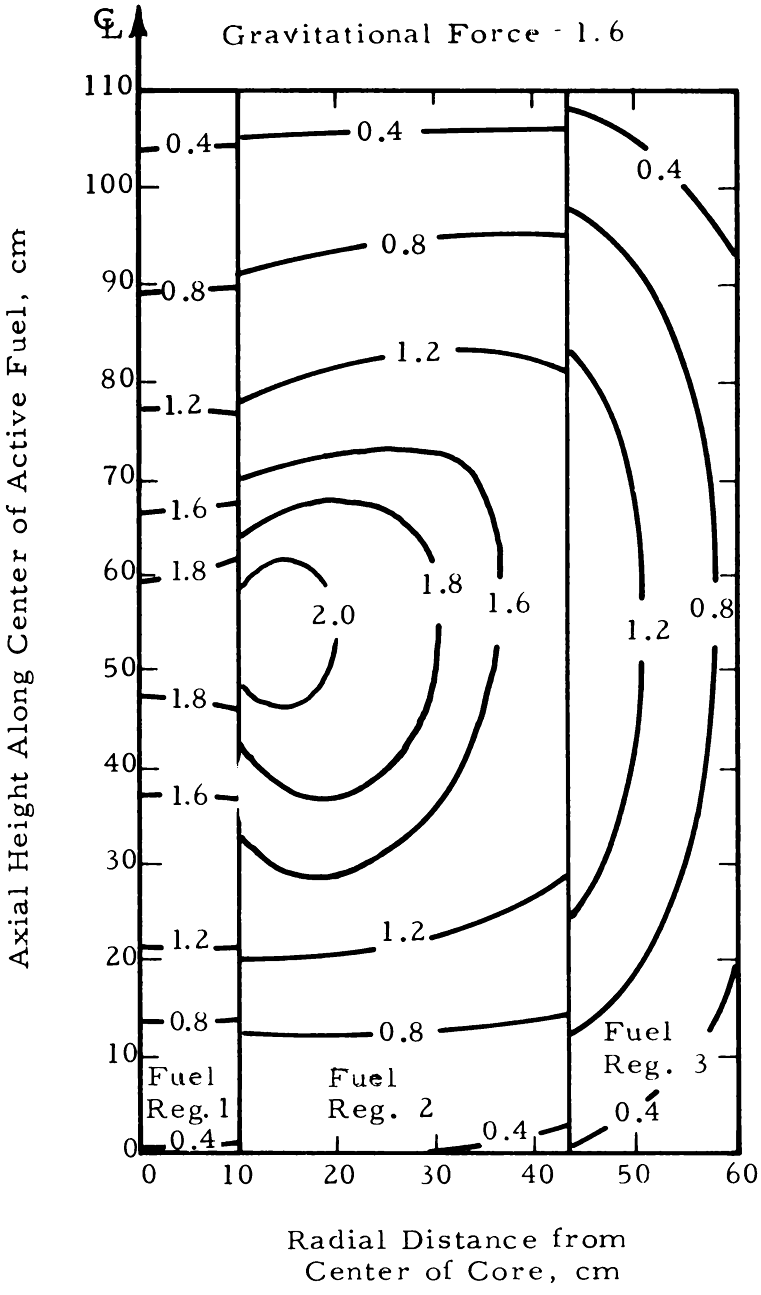


Figure 3. 7. 2-5. Soluble Poison Variation Over Core Life First and Equilibrium Cycles

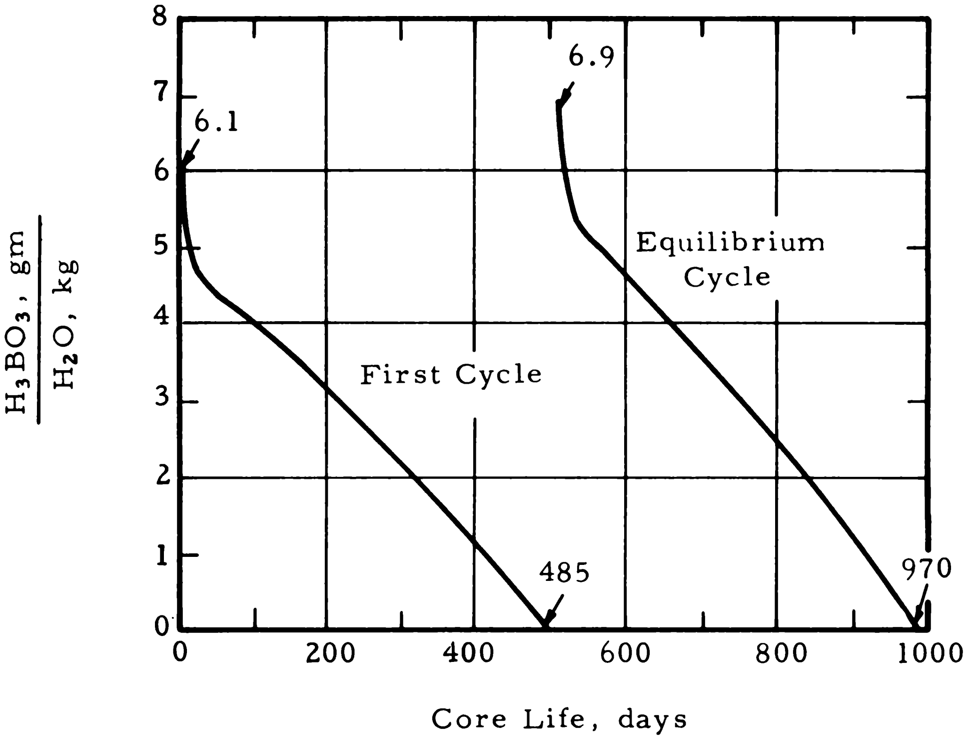


Figure 3. 7. 2-6. Initial Soluble Poison Variation

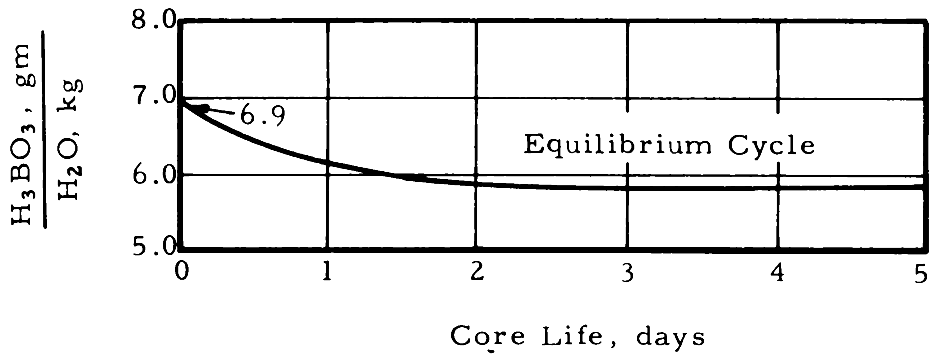


Figure 3.7.2-7. Soluble Poison Reactivity Worth as a Function of Moderator Temperature (Equilibrium Cycle —Beginning of Life)

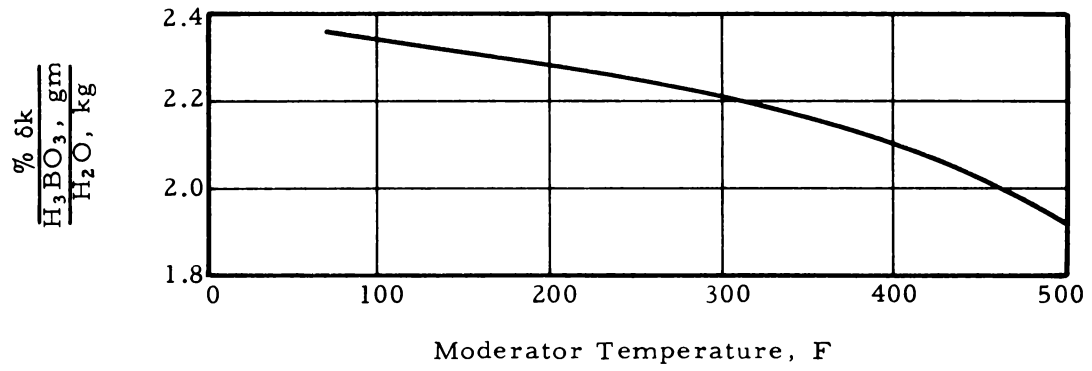


Figure 3.7.2-8. Integrated Conversion Ratio as a Function of Core Life (Equilibrium Cycle)

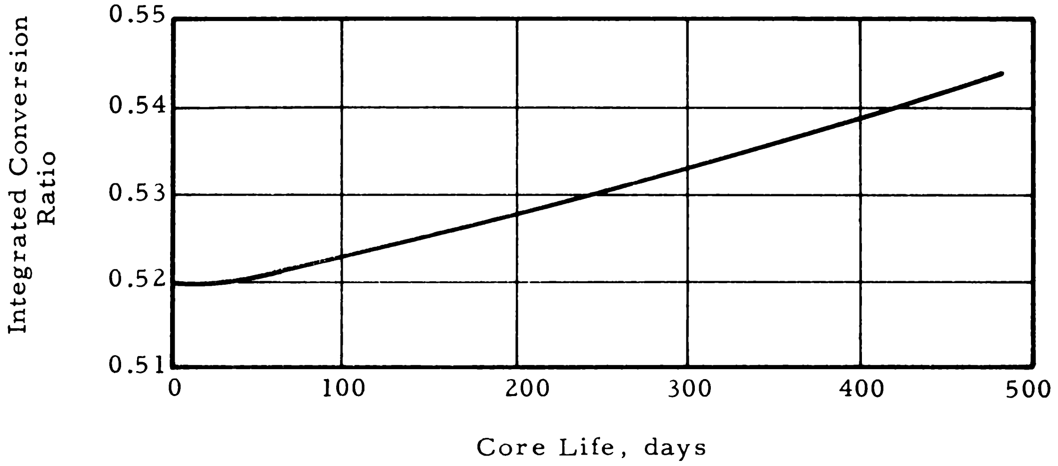


Figure 3.7.2-9. $\bar{\eta}$ as a Function of Core Life (Equilibrium Cycle)

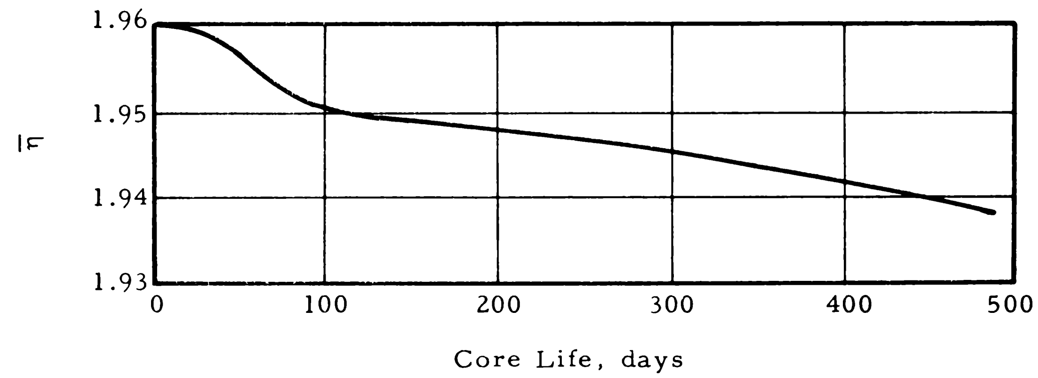


Figure 3.7.2-10. Radial Fast and Thermal Flux Profile – Beginning of Life (Equilibrium Cycle)

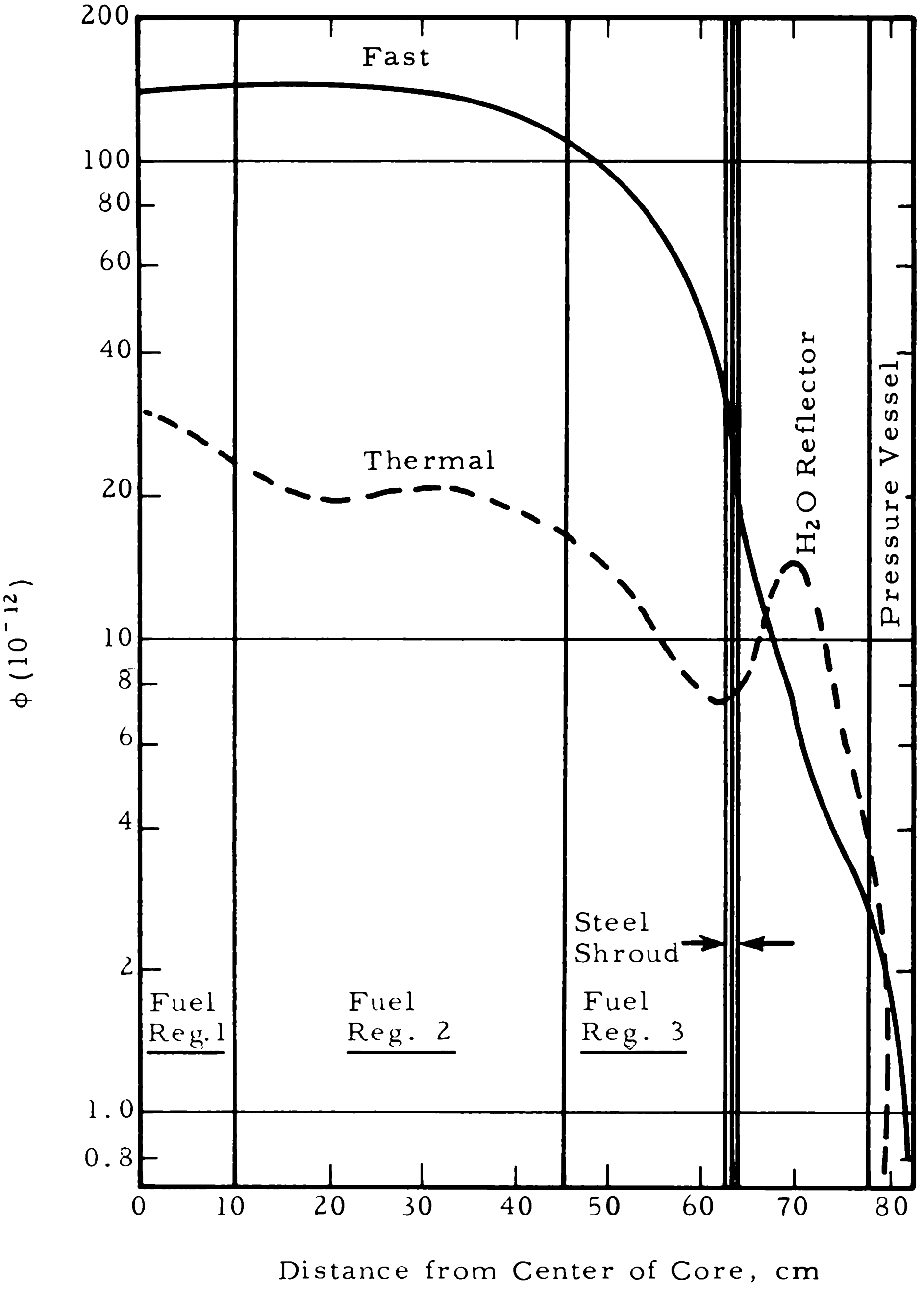


Figure 3.7.2-11. Radial Fast and Thermal Flux Profile – End of Life (Equilibrium Cycle)

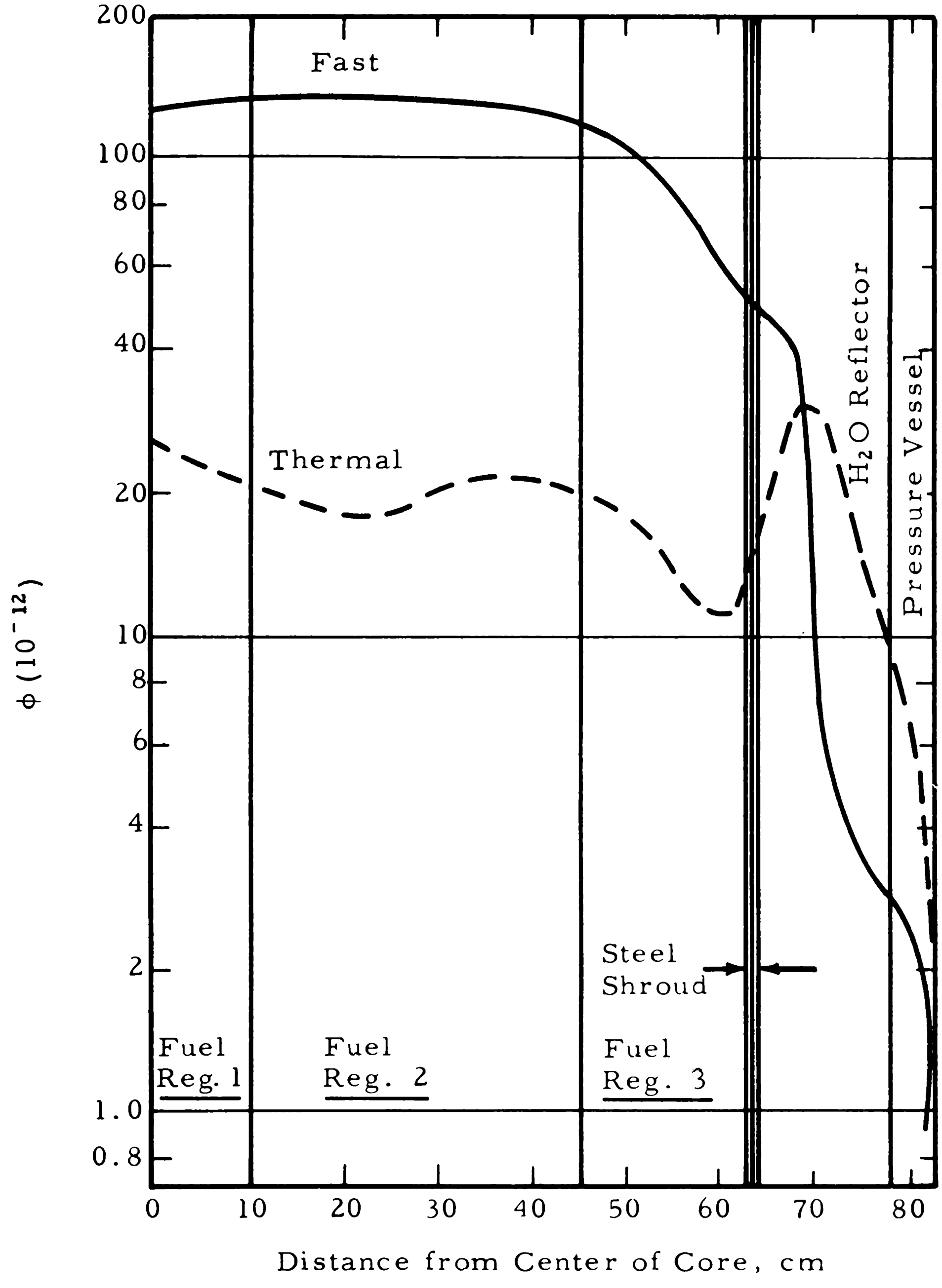


Figure 3.7.2-12. Temperature Coefficient Vs Core Temperature

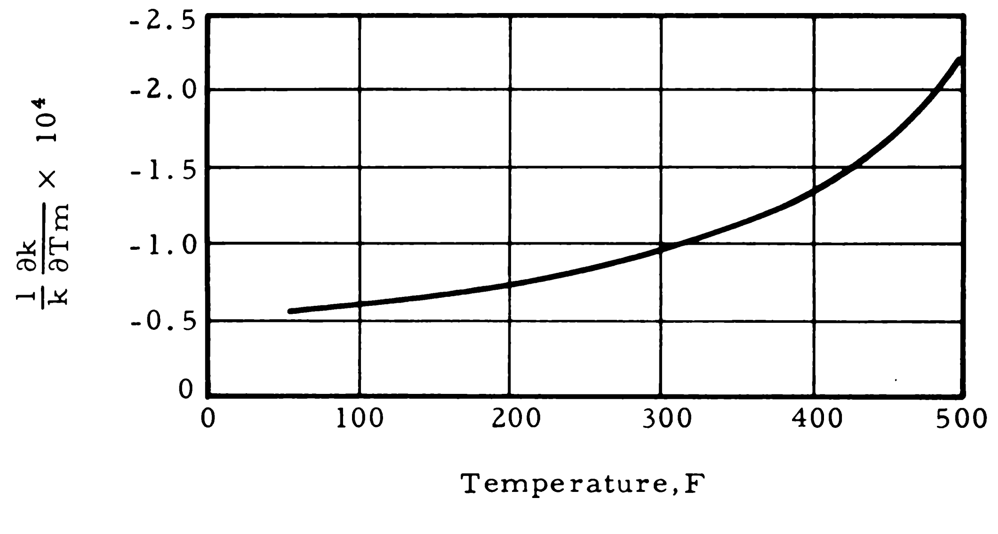


Figure 3.7.2-13. Void Coefficient Vs Integrated Core Void

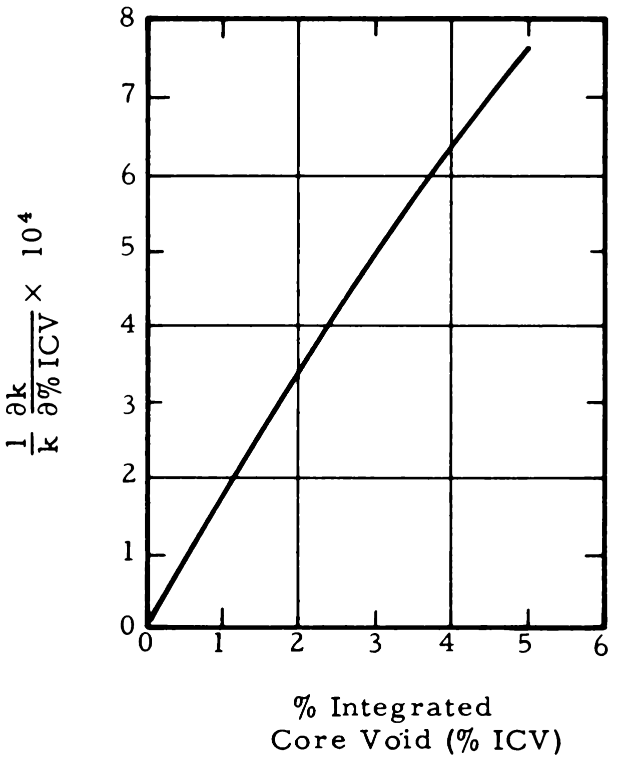


Figure 3.7.2-14. Reactivity Response to Xenon Variation Following Full Power Shutdown (100% to 0% Power)

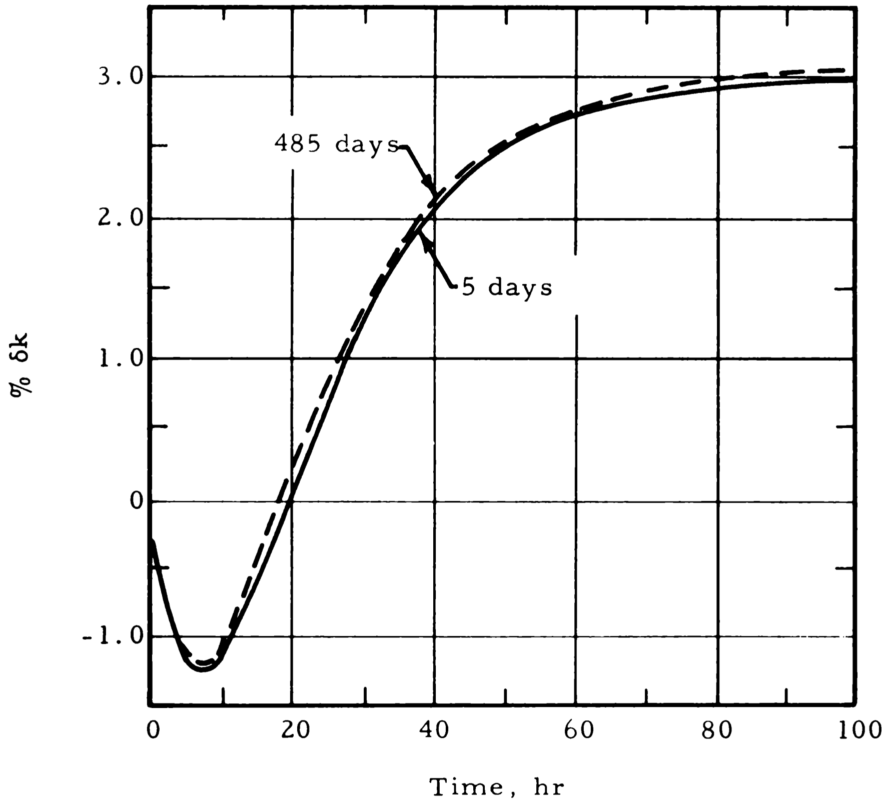


Figure 3.7.2-15. Soluble Poison Change Necessary to Maintain Critical Core After Full Power Shutdown

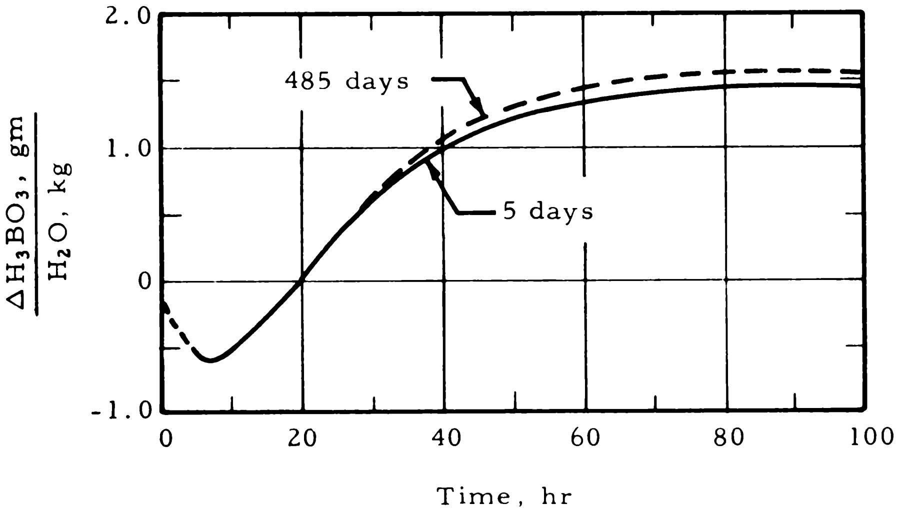


Figure 3.7.3-1. Axial Power Distribution

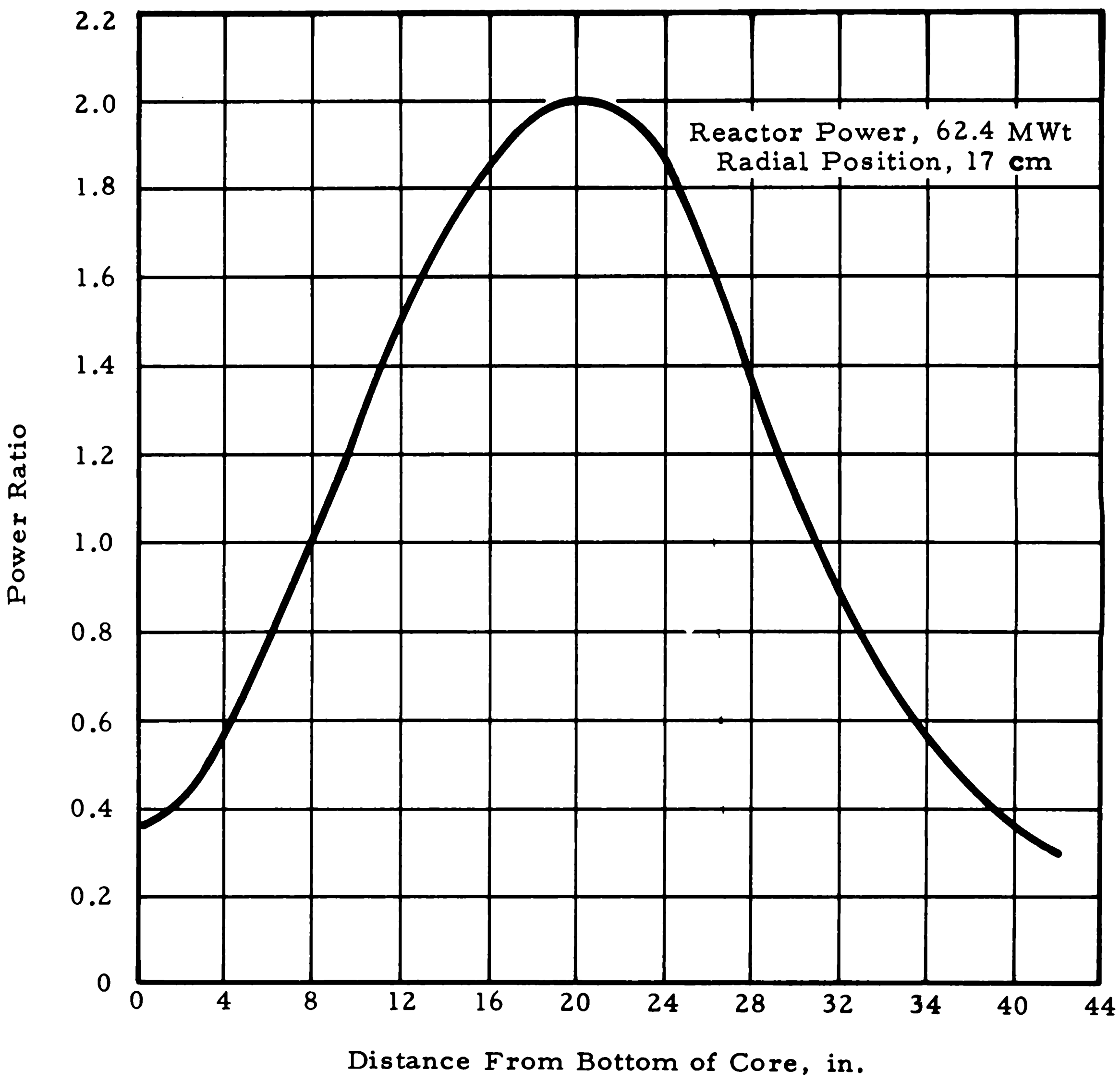


Figure 3. 7. 3-2. Steam Volume Fraction

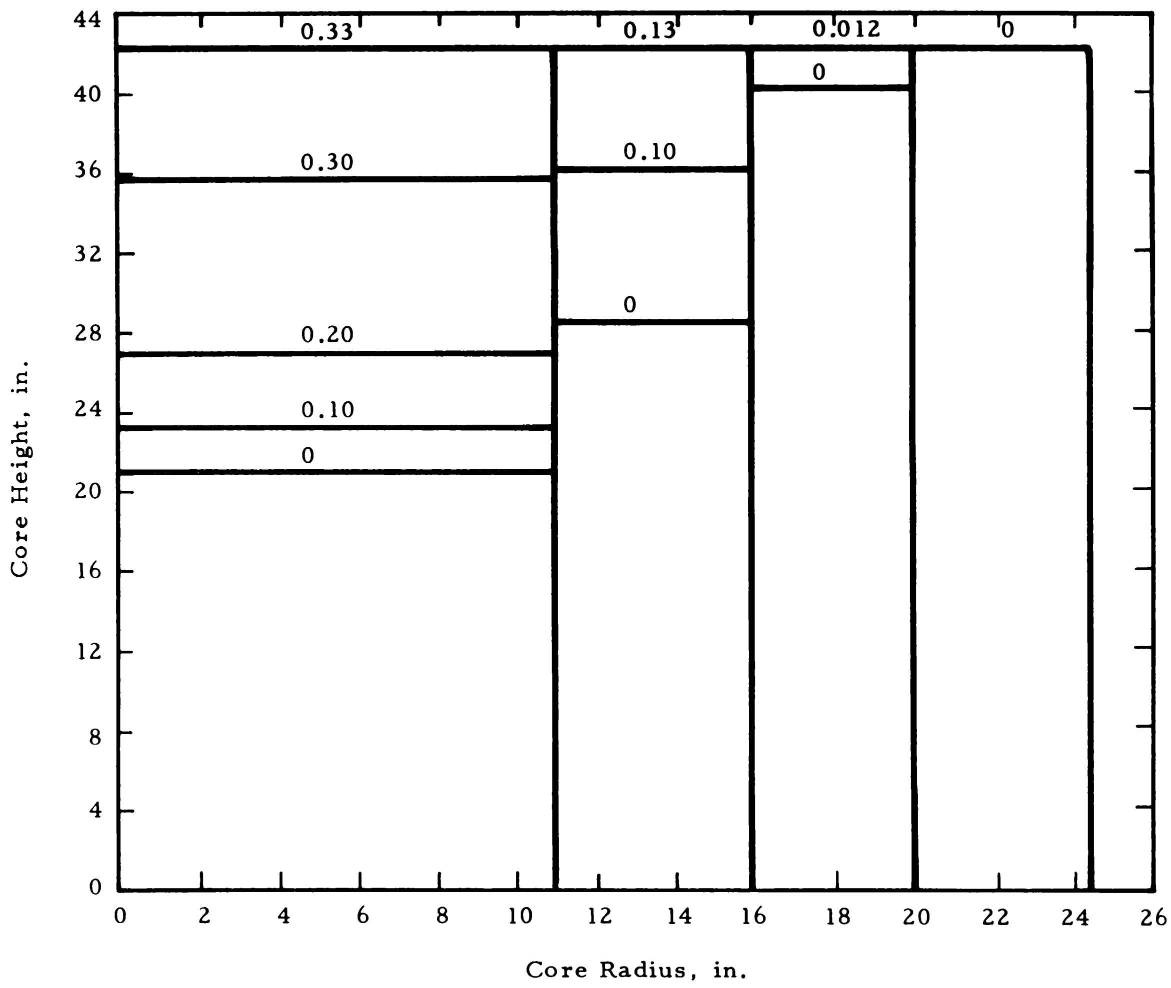


Figure 3.7.3-3. Pressure Drop Vs Flow per Pin

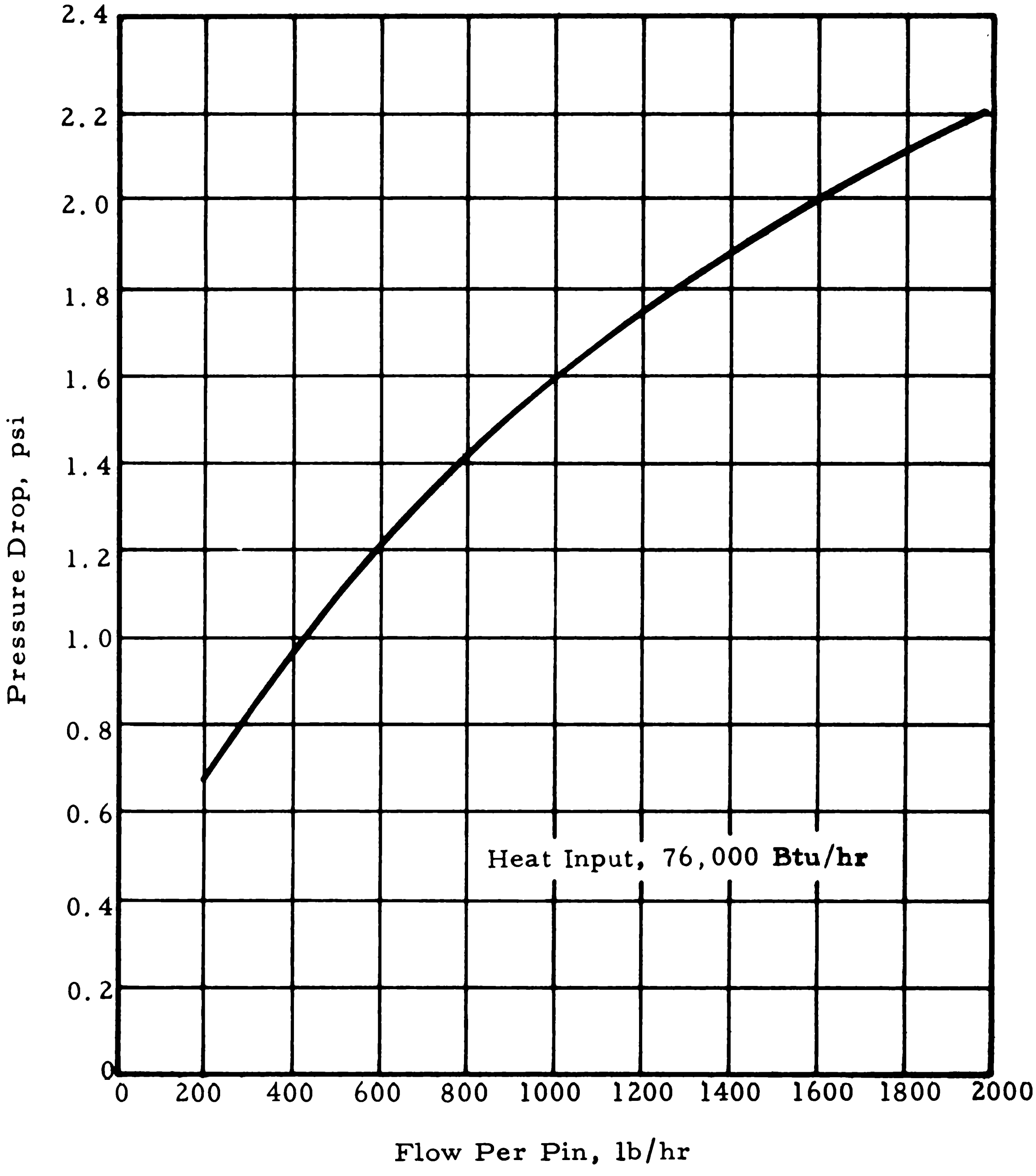


Figure 3.8-1. Primary Temperature Vs Load with Constant Primary Pressure Control

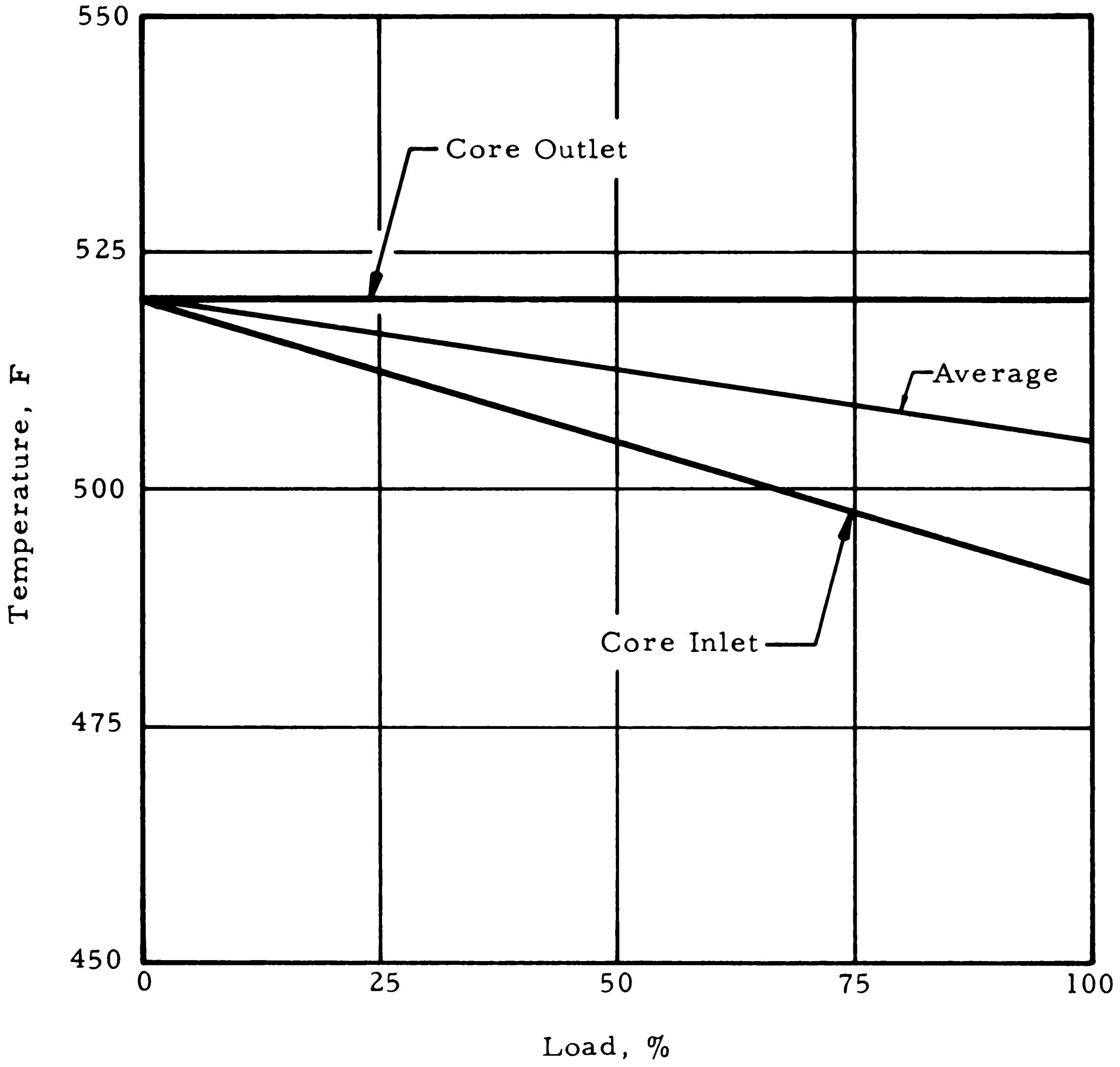


Figure 3.8-2. Reactor Control System
(Block Diagram)

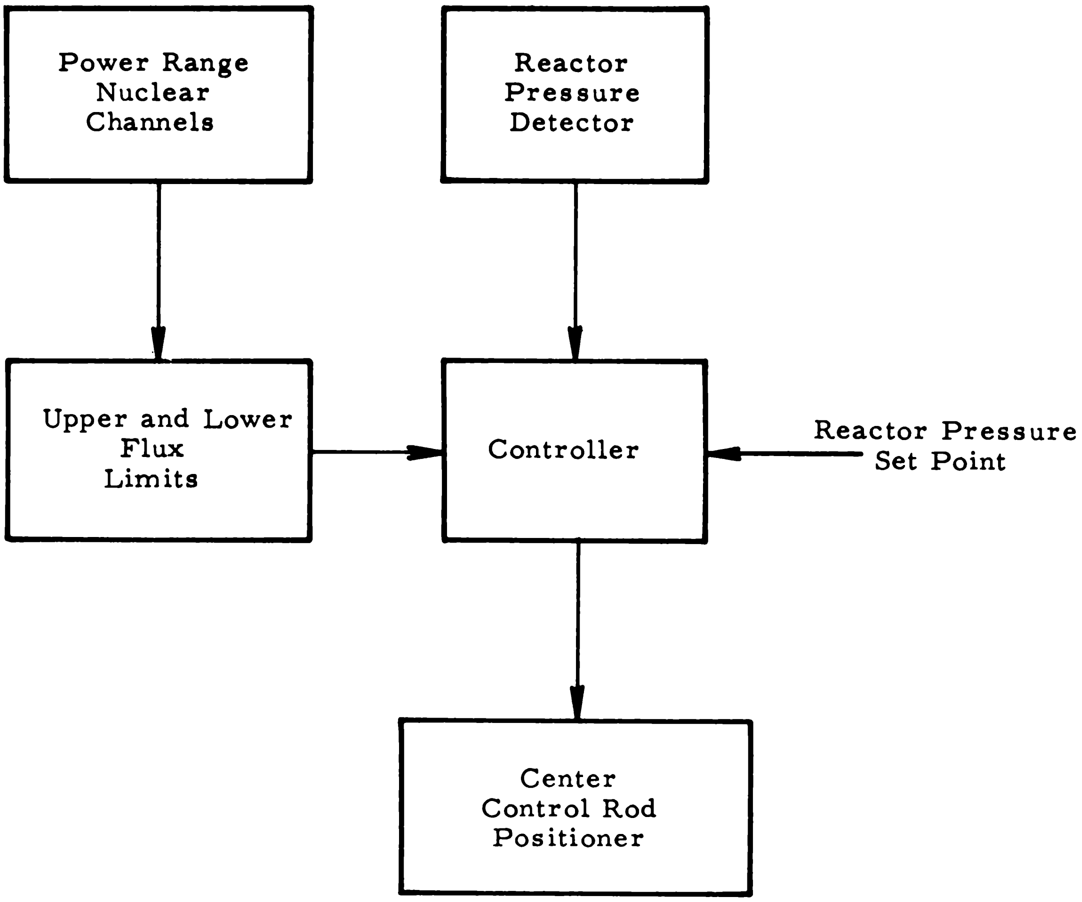


Figure 3.8-3. Rod Positioning for Xenon Transients Resulting from Load Change (Full to Base Power)

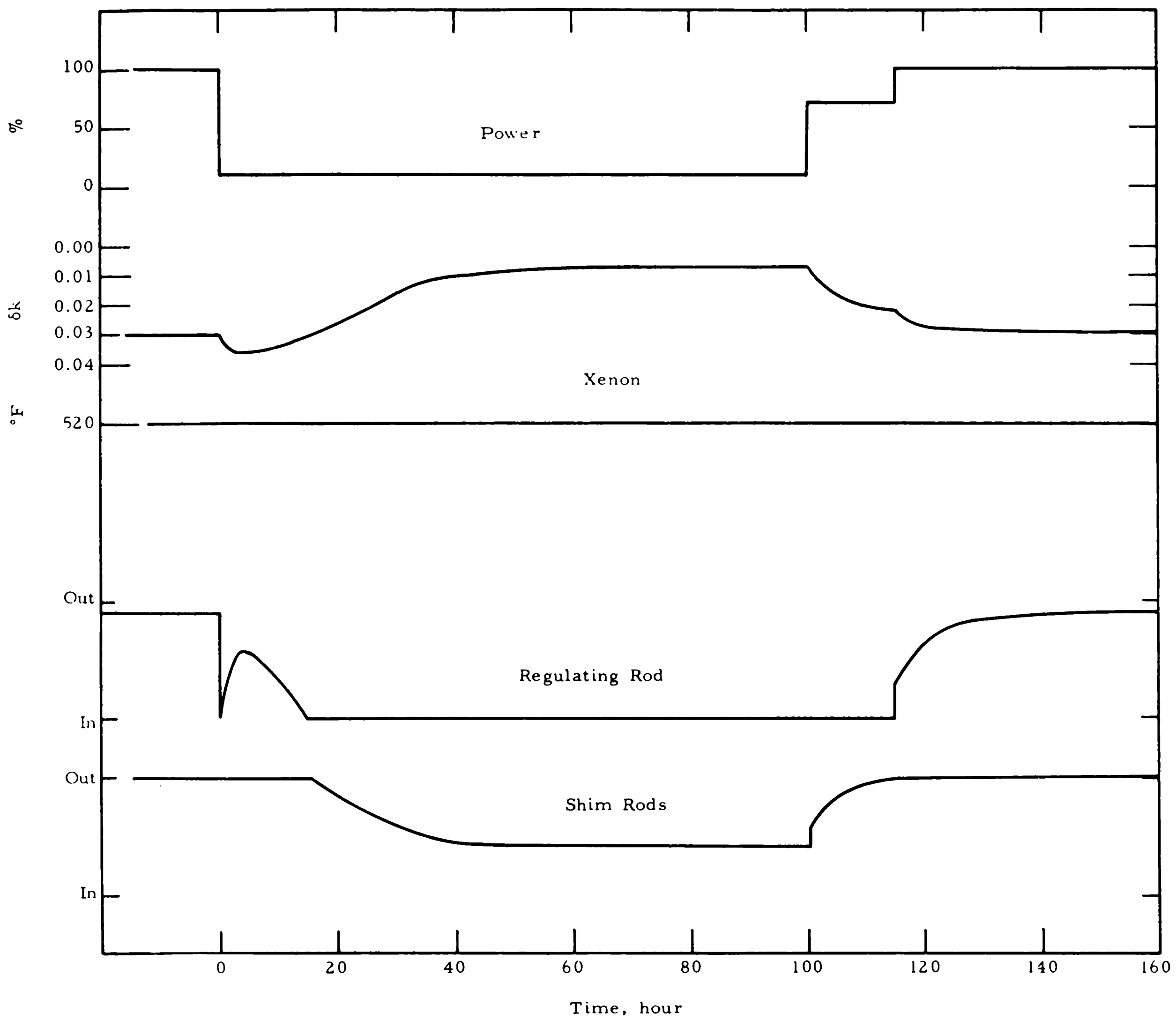


Figure 3.8-4. Return to Full Power at Xenon Peak at 100% to 10% Transient

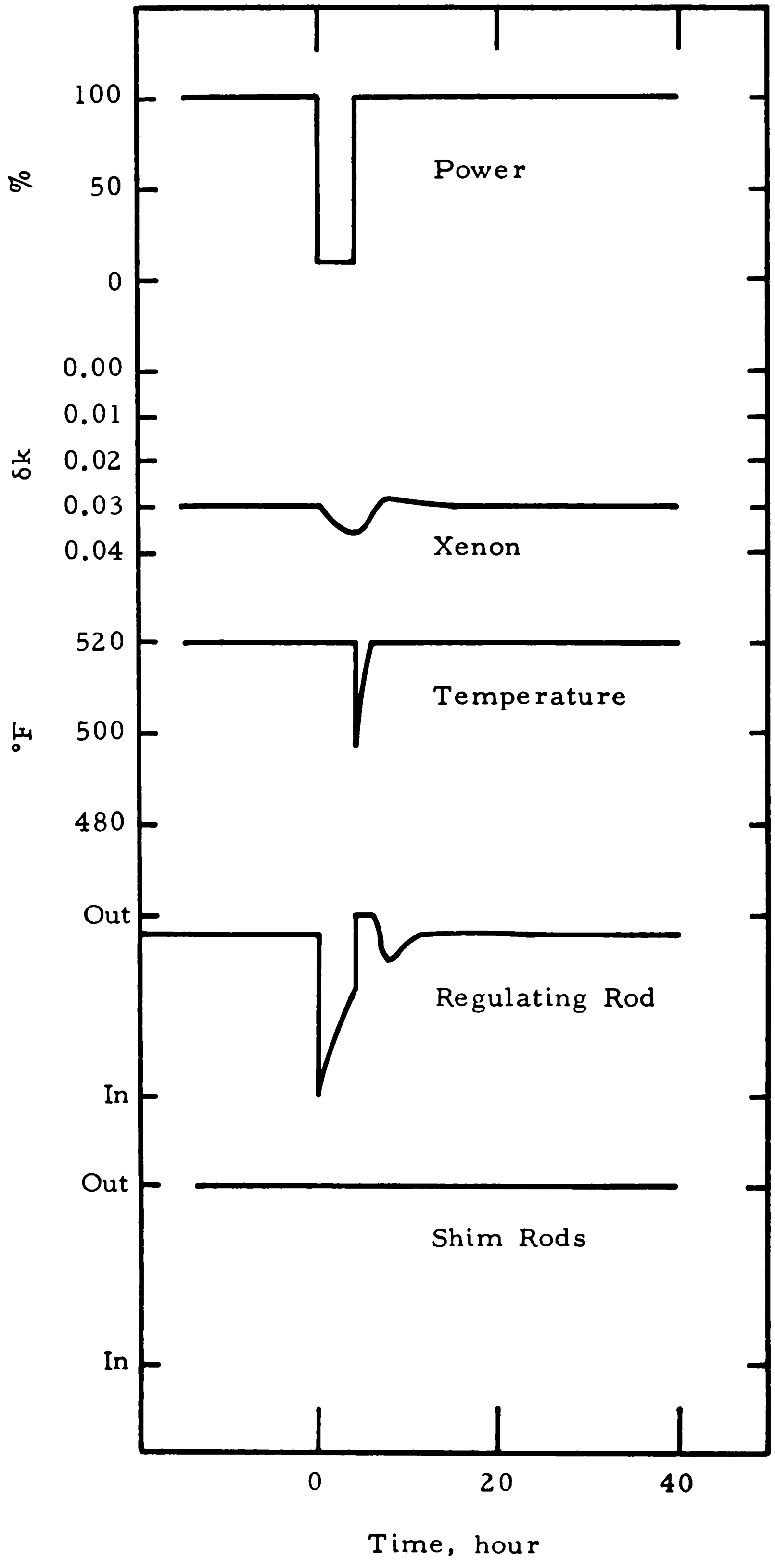


Figure 3.8-5. Rod Positioning for Xenon Transients Resulting from Load Change (Full to Zero Power)

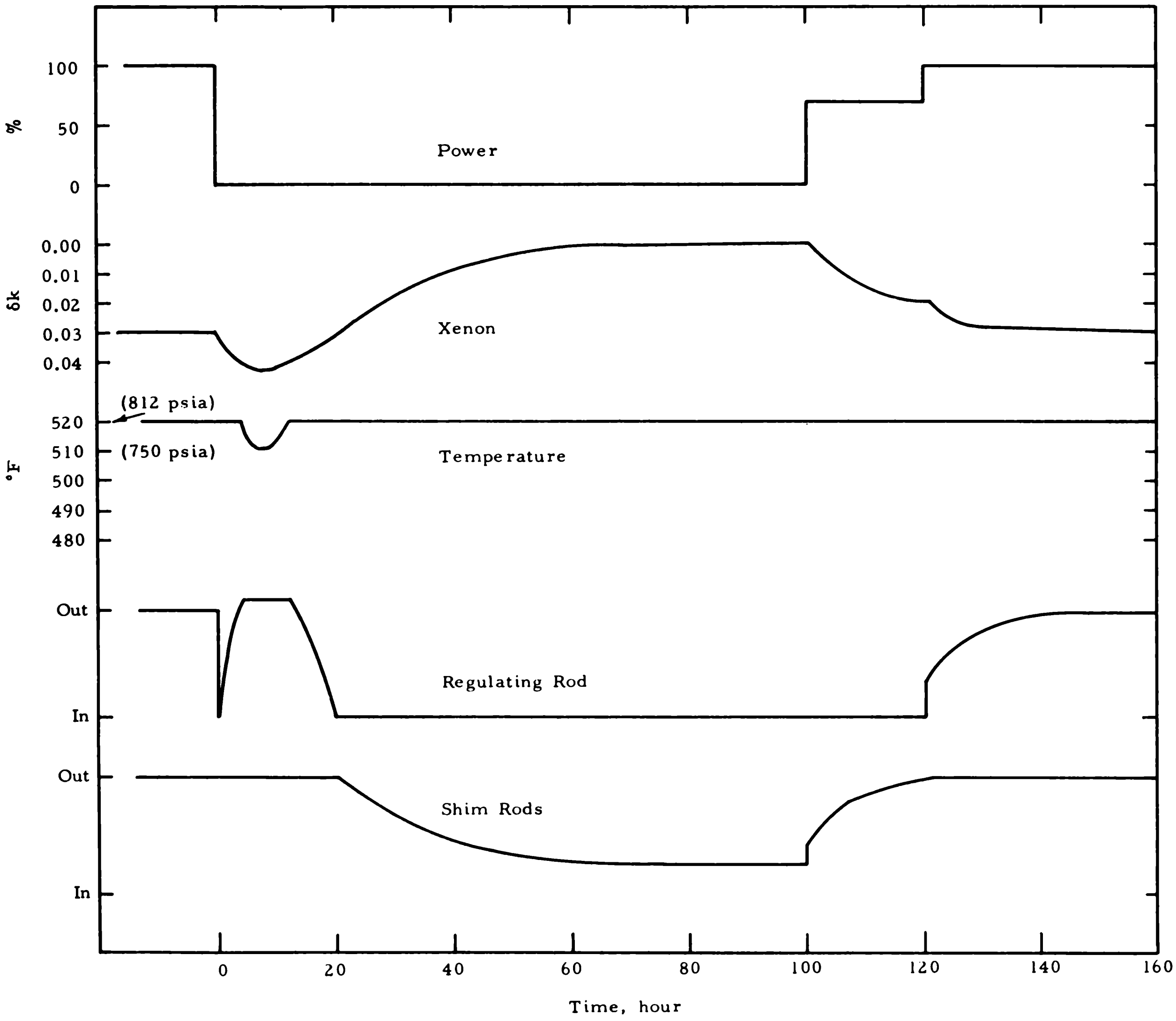


Figure 3.8-6. Return to Full Power at Xenon Peak at 100% to Zero Power Transient

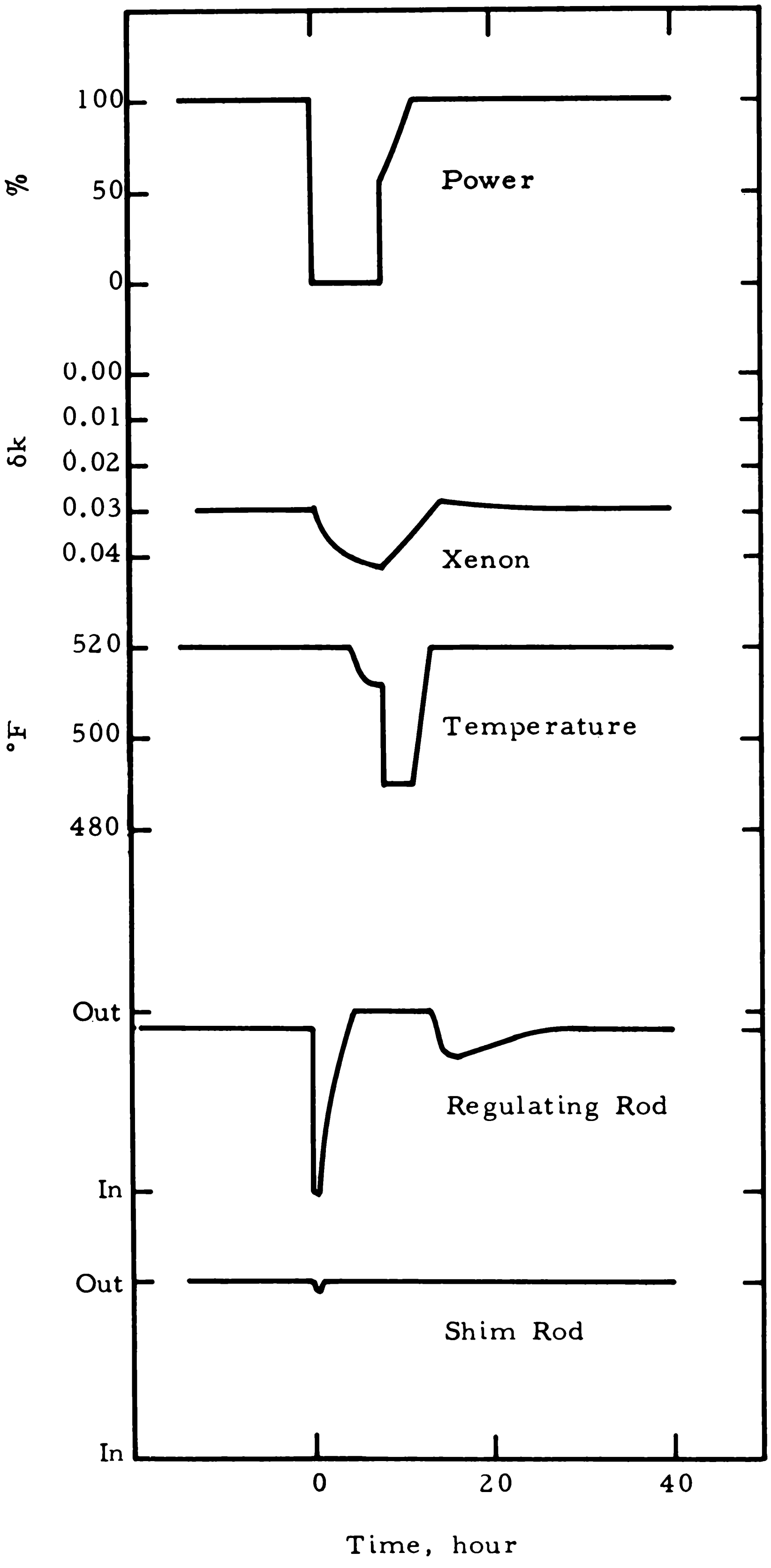


Figure 3.8-7. Rod Positioning for Xenon Transients Resulting from Load Changes (Full to 60% Power)

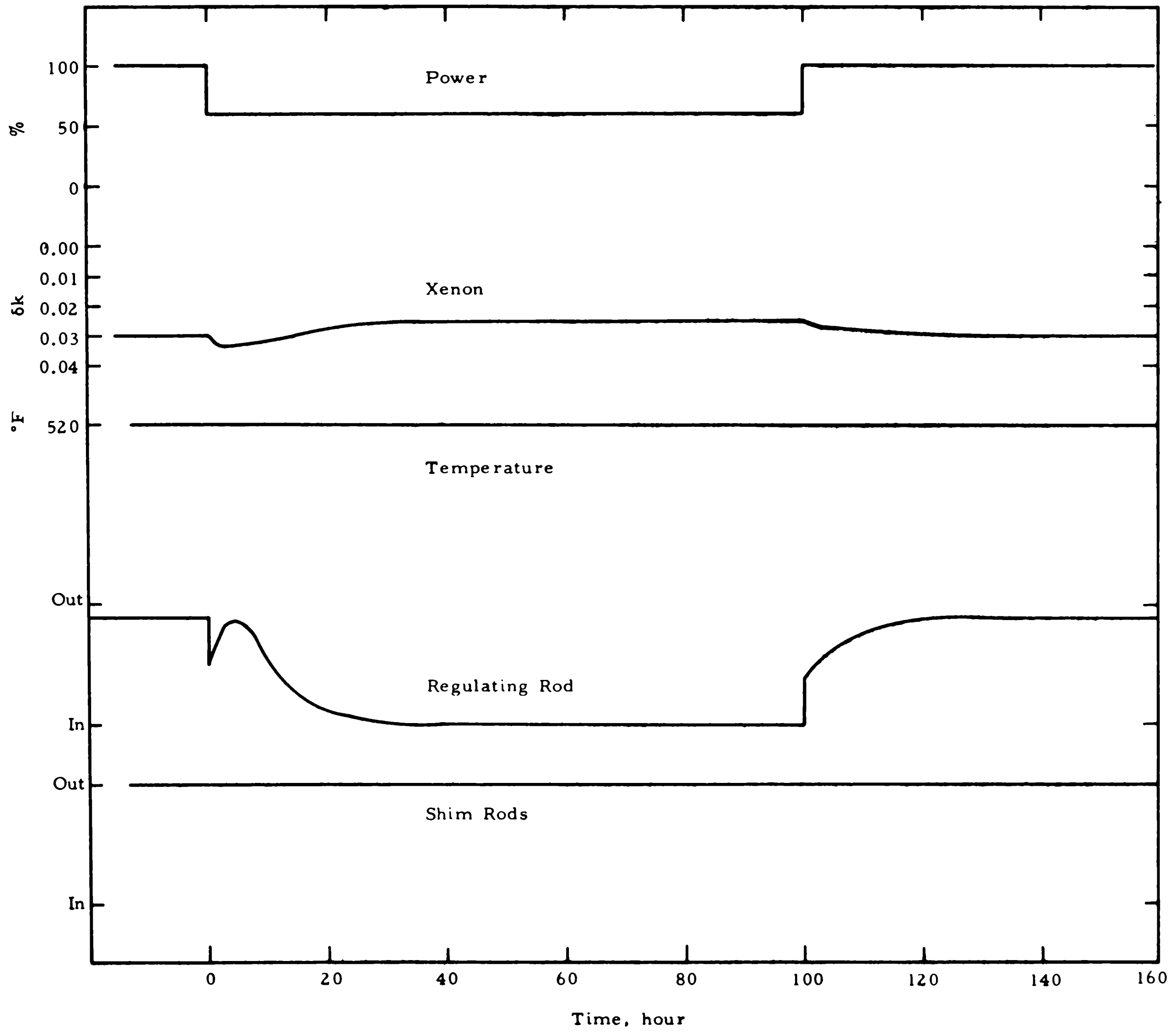
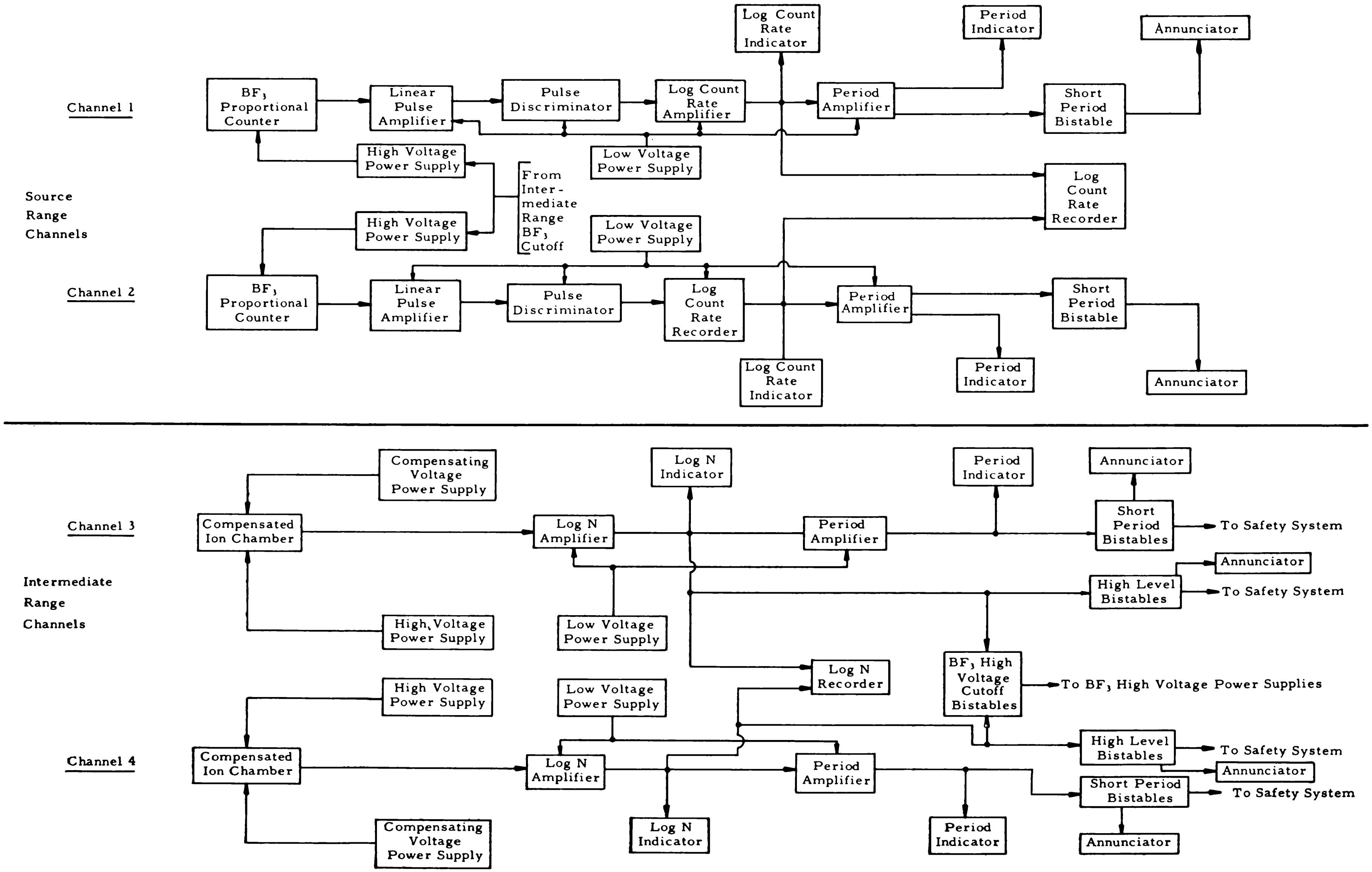


Figure 3.8-8. Nuclear Instrumentation Diagram



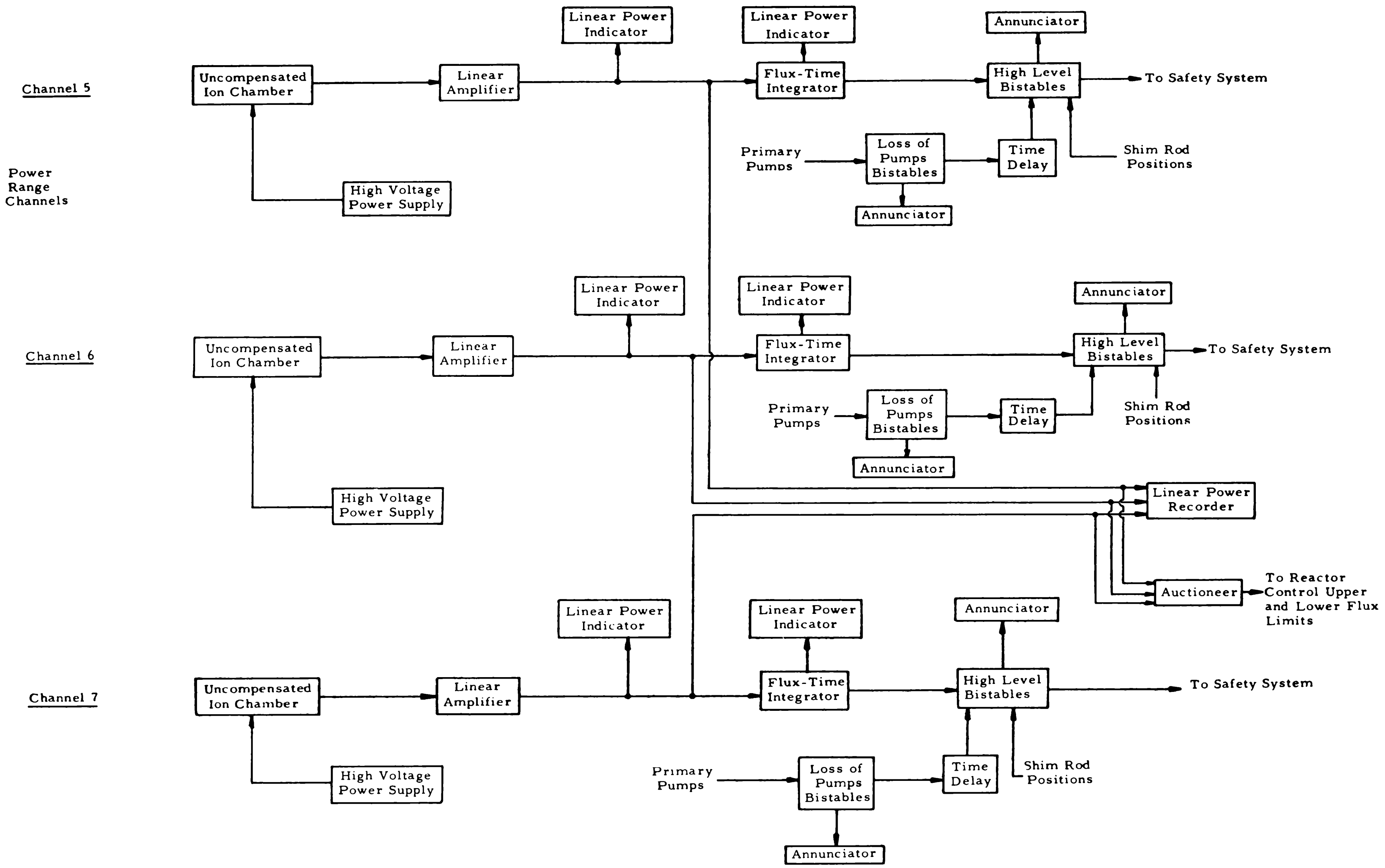

Channel 5
Power Range Channels
Uncompensated Ion Chamber
High Voltage Power Supply
Linear Amplifier
Linear Power Indicator
Linear Power Indicator
Flux-Time Integrator
Annunciator
High Level Bistables
To Safety System
Primary Pumps
Loss of Pumps Bistables
Annunciator
Time Delay
Shim Rod Positions
Channel 6
Uncompensated Ion Chamber
High Voltage Power Supply
Linear Amplifier
Linear Power Indicator
Linear Power Indicator
Flux-Time Integrator
Annunciator
High Level Bistables
To Safety System
Primary Pumps
Loss of Pumps Bistables
Annunciator
Time Delay
Shim Rod Positions
Linear Power Recorder
Auctioneer
To Reactor Control Upper and Lower Flux Limits
Channel 7
Uncompensated Ion Chamber
High Voltage Power Supply
Linear Amplifier
Linear Power Indicator
Linear Power Indicator
Flux-Time Integrator
Annunciator
High Level Bistables
To Safety System
Primary Pumps
Loss of Pumps Bistables
Annunciator
Time Delay
Shim Rod Positions

Figure 3.8-9. Safety System Diagram

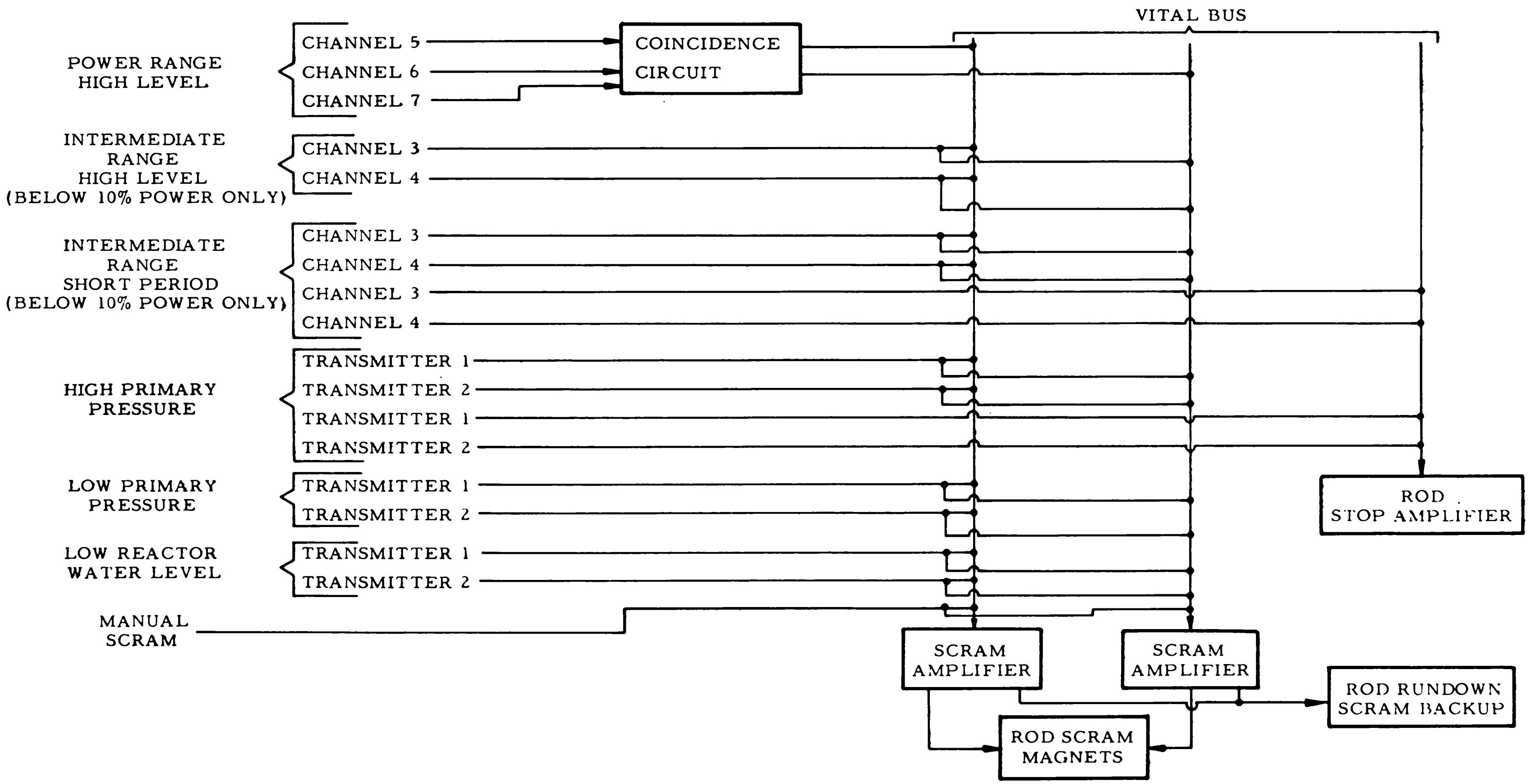
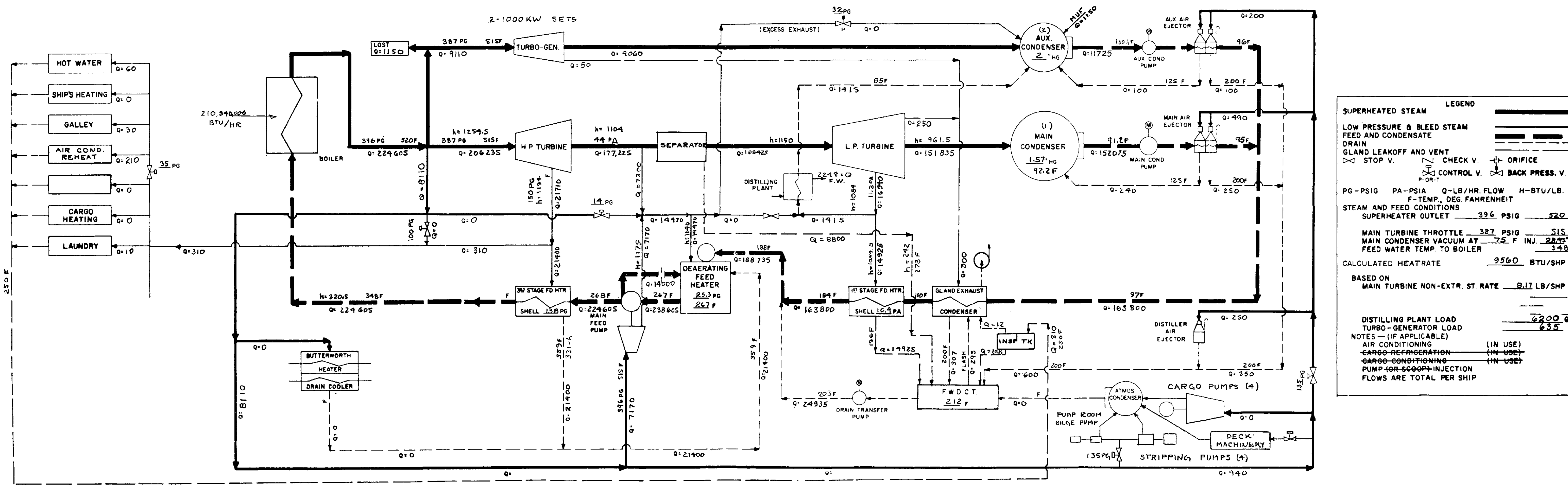


Figure 3.10-1. Heat Balance
and Flow Diagram



4. SHIP AND PLANT ARRANGEMENT

Volume is important for a shipboard power plant. To evaluate this aspect, the Maritime Administration considered the use of the CNSG in two typical merchant ships. This consideration is described here.

The CNSG plant requires no more space for installation than does today's conventional machinery of same power. In fact, the reactor, together with its containment vessel, is physically smaller than the boilers that it replaces.

Figures 4-1 to 4-4 show how the plant could be installed in a large cargo ship of the Mariner class and in a tanker of the 43 to 46000 deadweight ton size. In each case, the installation is made without sacrificing cargo space.

As shown by Figures 4-1 and 4-2 for the Mariner cargo ship, the machinery arrangement is indentical with that of the conventionally powered ship except for the simple substitution of the reactor and containment vessel for the boilers. The superstructure overhead would be rearranged to provide a casing for servicing directly above the reactor head. The uptake casing required for conventional machinery is eliminated. The conventional Mariner machinery space is arranged with tanks along the shell on each side. In the nuclear ship, the space required for these tanks is utilized for the collision barrier.

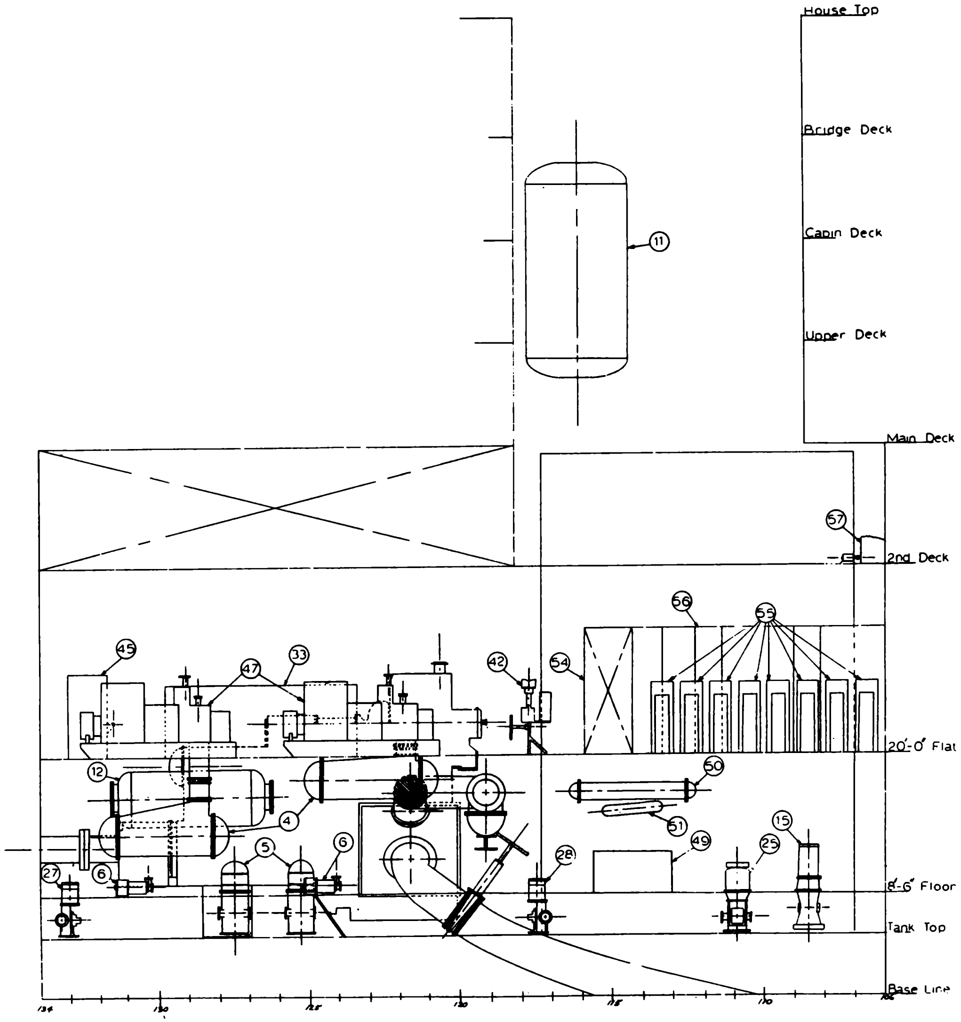
In the tanker, the reactor and containment vessel are on the tank top immediately forward of the propulsion turbine. This does not represent a simple exchange of reactor for boilers, since the basic conventional tanker machinery arrangement has boilers aft of the turbine and above the propeller shaft. This location does permit cooling of the reactor by natural thermal circulation if required.

Location of the reactor forward of the propulsion turbine requires relocation of the forward engine room bulkhead and rearrangement of the pump room. By taking over space originally allocated to the after fuel oil tanks, these adjustments are made without sacrificing cargo space.

The space originally required for boilers and uptake casings is available for other purposes. This permits smaller superstructures aft and eliminates the poop.

Since the collision barrier is of cellular construction, some equipment can be located partly within its confines if necessary.

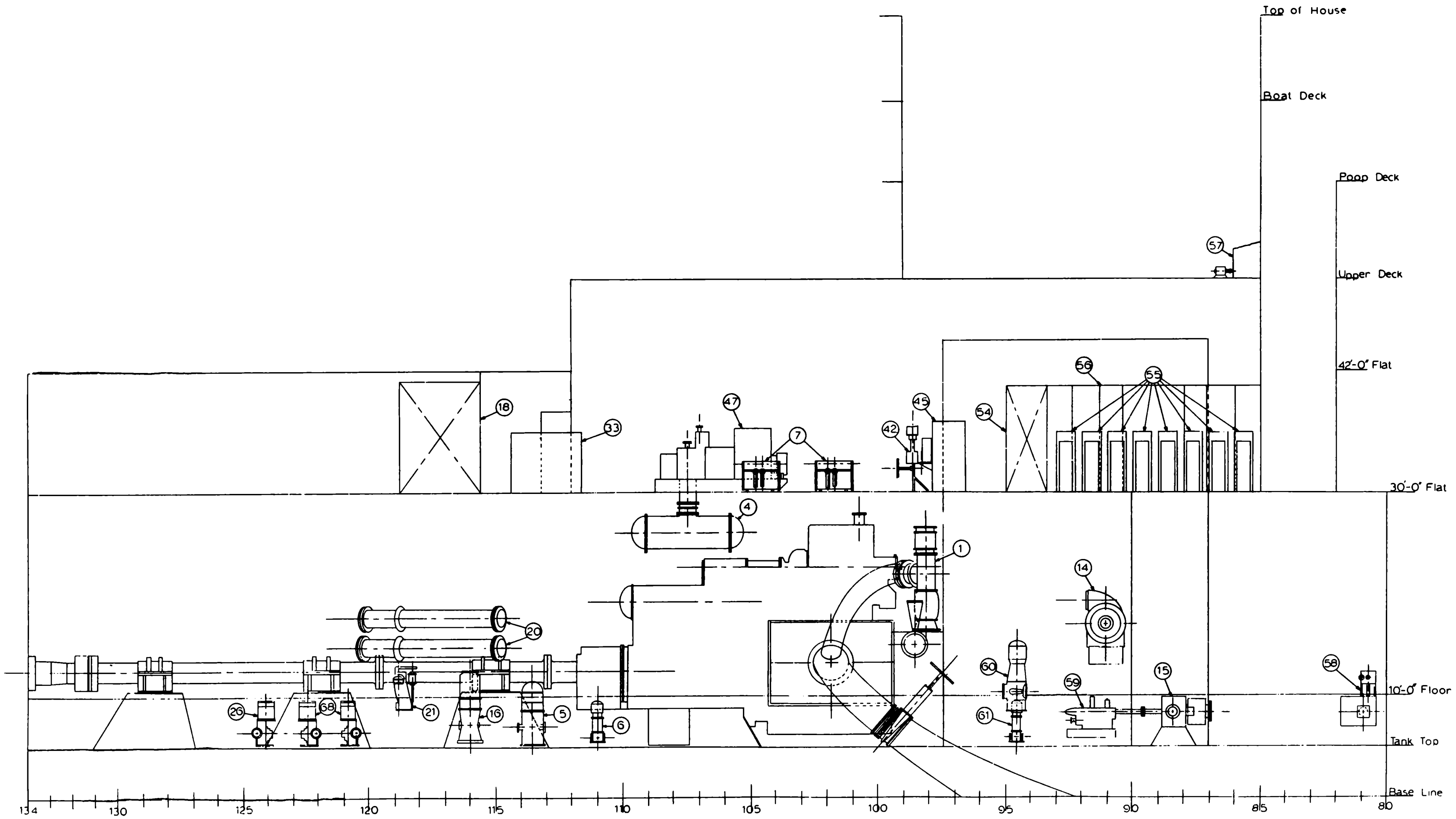
Figure 4-1. CNSG Plant (20, 000 shp) in Mariner (C4-S-1a) Cargo Ship



LIST OF MACHINERY

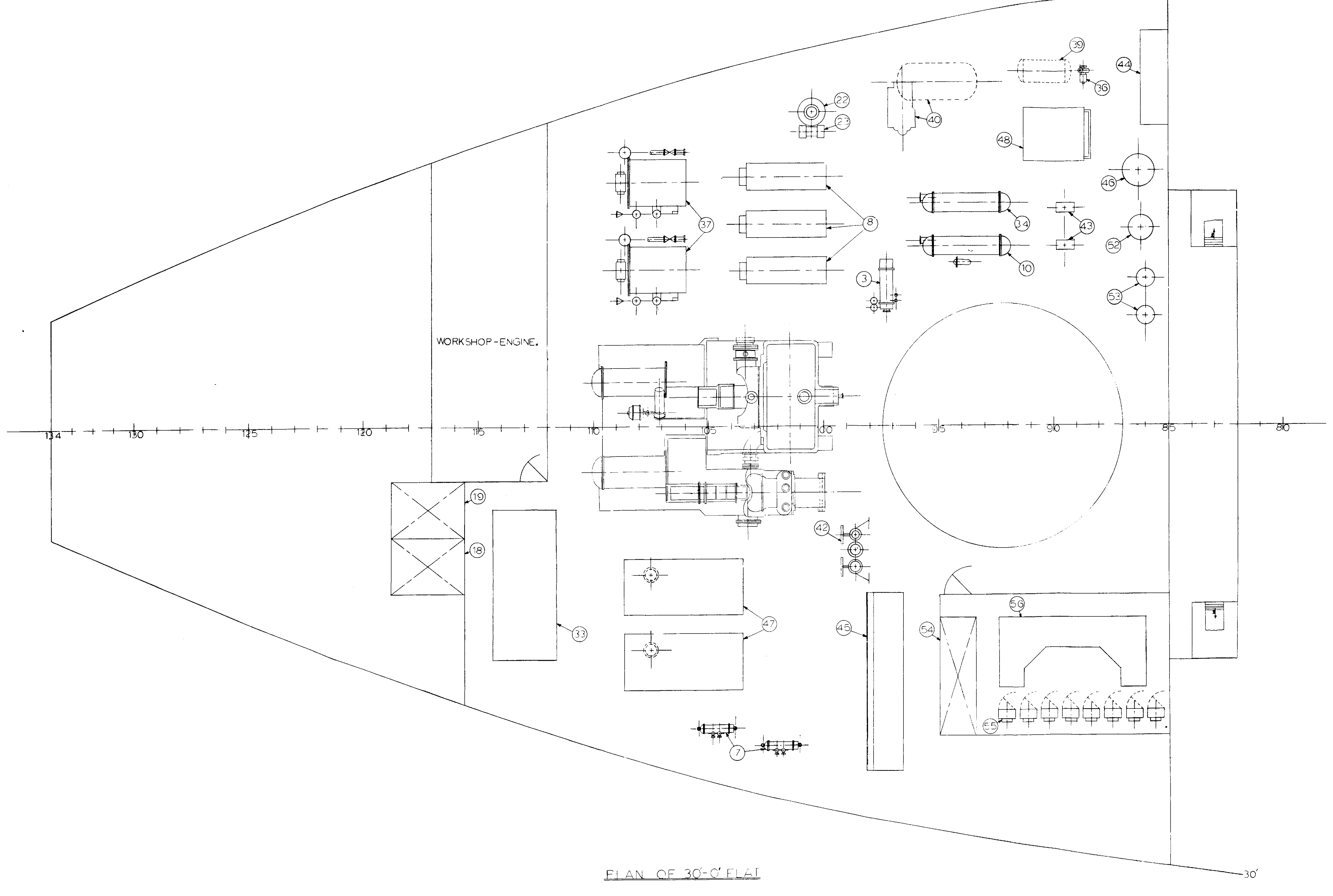
NO	NAME
1	NOT USED
2	NOT USED
3	NOT USED
4	AUXILIARY CONDENSER
5	AUXILIARY CONDENSER CIRCULATOR
6	AUXILIARY CONDTE. PUMP
7	NOT USED
8	NOT USED
9	NOT USED
10	NOT USED
11	SECOND STAGE D.C. FEED HEATER
12	STEAM SEPARATOR
13	NOT USED
14	NOT USED
15	CARGO OIL TRANSFER PUMP
16	NOT USED
17	NOT USED
18	NOT USED
19	NOT USED
20	NOT USED
21	NOT USED
22	NOT USED
23	NOT USED
24	NOT USED
25	FIRE PUMP
26	NOT USED
27	BILGE & BALLAST PUMP
28	CARGO REFER. CONDSR. CIRCULATOR
29	NOT USED
30	NOT USED
31	NOT USED
32	NOT USED
33	AIR CONDITIONING UNIT
34	NOT USED
35	NOT USED
36	NOT USED
37	NOT USED
38	NOT USED
39	NOT USED
40	NOT USED
41	NOT USED
42	MAIN THROTTLE VALVE
43	NOT USED
44	NOT USED
45	SWITCHBOARD
46	NOT USED
47	TURBO-GENERATOR
48	NOT USED
49	CARGO REFER. COMPRESSOR
50	CARGO REFER. CONDENSER
51	CARGO REFER. RECEIVER
52	NOT USED
53	NOT USED
54	CUBICLE FOR CONSOLE EQUIP.
55	REACTOR CONTROL & RADIATION MONITORING EQUIP.
56	CONTROL CONSOLE
57	C. V. VENT BLOWER, FILTER, & RADIATION MONITOR

Figure 4-3. CNSG Plant (20, 000 shp) in 43-46, 000 dwt Tanker

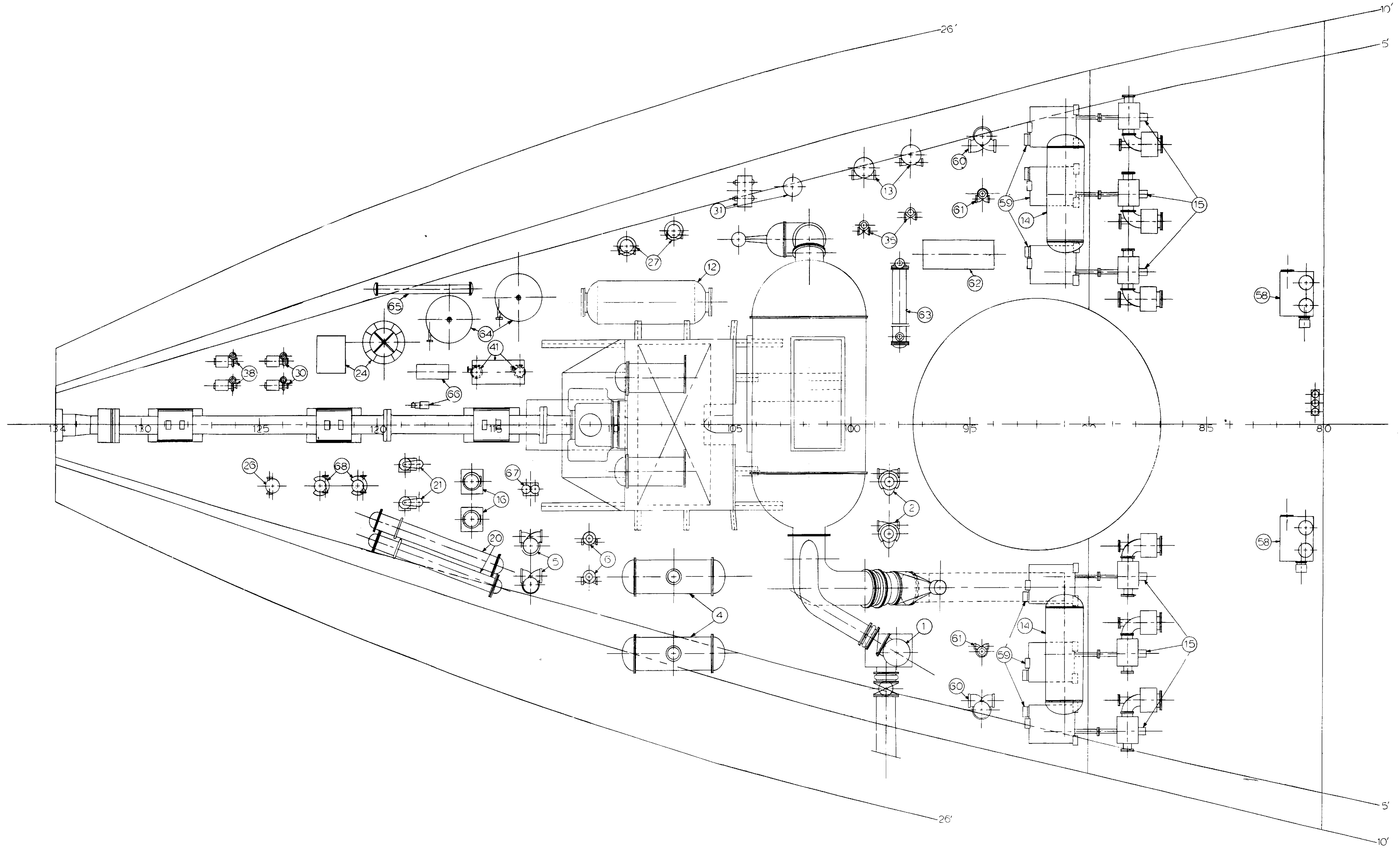


ELEVATION-STBD. SIDE LKG. INBD.

Figure 4-4. CNSG Plant (20,000 shp) in 43-46,000 dwt Tanker



PLAN OF 30'-0" FLAT

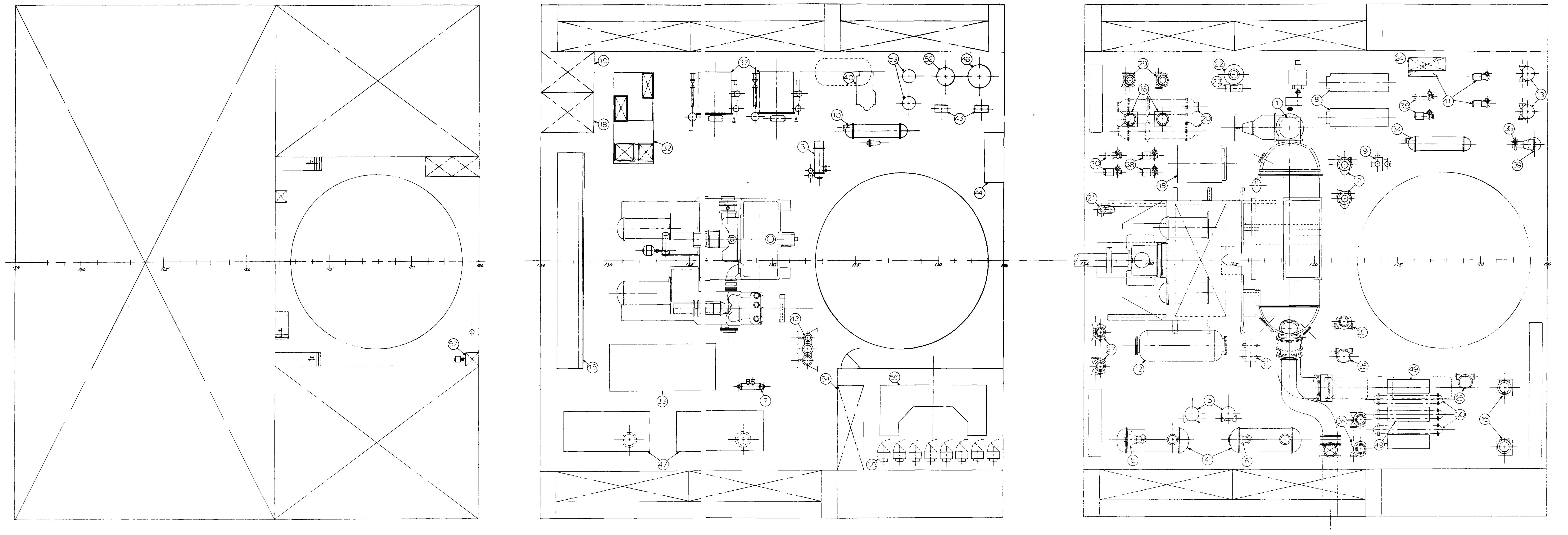


PLAN OF LOWER LEVEL

LIST OF MACHINERY

NO.	NAME
1	MAIN CONDENSER CIRCULATOR
2	MAIN CONDTE. PUMP
3	MAIN AIR EJECTOR
4	AUXILIARY CONDENSER
5	AUXILIARY CONDENSER CIRCULATOR
6	AUXILIARY CONDTE. PUMP
7	AUXILIARY AIR EJECTOR
8	MAIN FEED PUMP
9	NOT USED
10	FIRST STAGE HEATER & GLAND EXH. CONDSR.
11	SECOND STAGE D.C. FEED HEATER
12	STEAM SEPARATOR
13	SEA WATER CIRC. PUMP
14	ATMOSPHERIC CONDENSER
15	CARGO OIL TRANSFER PUMP
16	LUBE OIL SERVICE PUMP
17	LUBE OIL GRAVITY TANK
18	LUBE OIL SETTLING TANK
19	LUBE OIL STORAGE TANK
20	LUBE OIL COOLER
21	LUBE OIL PURIFIER
22	CONTAMINATED EVAPORATOR
23	CONTAMINATED EVAPORATOR FEED PUMP
24	DRAIN INSPECTION TANK & HOTWELL
25	NOT USED
26	FIRE & BILGE PUMP
27	BILGE & BALLAST PUMP
28	NOT USED
29	NOT USED
30	SANITARY PUMP
31	PRIMING PUMPS & VACUUM TANK
32	NOT USED
33	AIR CONDITIONING UNIT
34	THIRD STAGE FEED HEATER
35	MAKE UP PUMP
36	CONTAINMENT SAMPLING PUMP
37	L. P. DISTILLING PLANT
38	FRESH WATER PUMP
39	CONTAINMENT CHEMICAL ADDITION TANK
40	SHIPS SERVICE AIR COMPRESSOR & RECEIVER
41	FRESH WATER DRAIN COLLECT. TANK & DRAIN PUMP
42	MAIN THROTTLE VALVE
43	SOLUBLE POISON CHARGE PUMP
44	SAMPLING BOMB CABINET
45	SWITCHBOARD
46	PURIFIED WATER SUPPLY TANK
47	TURBO-GENERATOR
48	SHIPS STORES REFER. COMPRESSOR UNIT
49	NOT USED
50	NOT USED
51	NOT USED
52	SOLUBLE POISON SUPPLY TANK
53	MAKE UP DEMINERALIZER
54	CUBICLE FOR CONSOLE EQUIP.
55	REACTOR CONTROL & RADIATION MONITORING EQUIP.
56	CONTROL CONSOLE
57	NOT USED
58	STRIPPING PUMP
59	CARGO OIL PUMP TURBINE
60	ATMOS. CONDSR. CIRCULATOR
61	ATMOS. CONDSR. CONDTE. PUMP
62	FIRE & BUTTERWORTH PUMP
63	BUTTERWORTH HEATER & DRAIN COOLER
64	DE-OILER
65	CONTAMINATED DRAIN COOLER
66	DE-OILER PUMP
67	LUBE OIL SUCT. STRAINER
68	S.W. SERVICE PUMP

Figure 4-2. CNSG Plant (20, 000 shp)
in Mariner (C4-S-1a) Cargo Ship



PLAN OF 2nd DECK

PLAN OF 20'-0" FLAT

PLAN OF LOWER LEVEL

LIST OF MACHINERY

NO.	NAME
1	MAIN CONDENSER CIRCULATOR
2	MAIN CONDENSATE PUMP
3	MAIN AIR EJECTOR
4	AUXILIARY CONDENSER
5	AUXILIARY CONDENSER CIRCULATOR
6	AUXILIARY CONDENSATE PUMP
7	AUXILIARY AIR EJECTOR
8	MAIN FEED PUMP
9	PORT FEED PUMP
10	FIRST STAGE HEATER & GLAND EXH. COND.
11	SECOND STAGE D.C. FEED HEATER
12	STEAM SEPARATOR
13	SEA WATER CIRC. PUMP
14	NOT USED
15	CARGO OIL TRANSFER PUMP
16	LUBE OIL SERVICE PUMP
17	LUBE OIL GRAVITY TANK
18	LUBE OIL SETTLING TANK
19	LUBE OIL STORAGE TANK
20	LUBE OIL COOLER
21	LUBE OIL PURIFIER
22	CONTAMINATED EVAPORATOR
23	CONTAMINATED EVAPORATOR FEED PUMP
24	DRAIN INSPECTION TANK & HOTWELL
25	FIRE PUMP
26	FIRE & BILGE PUMP
27	BILGE & BALLAST PUMP
28	CARGO REFER. COND. CIRC. PUMP
29	DISTILLER COND. & CARGO CONDITIONING CIRC. PUMP
30	SANITARY PUMP
31	PRIMING PUMPS & VACUUM TANK
32	CARGO CONDITIONG UNIT
33	AIR CONDITIONING UNIT
34	THIRD STAGE FEED HEATER
35	MAKE UP PUMP
36	CONTAINMENT SAMPLE PUMP
37	L. P. DISTILLING PLANT
38	FRESH WATER PUMP
39	CONTAINMENT CHEMICAL ADDITION TANK
40	SHIPS SERVICE AIR COMPRESSOR & RECEIVER
41	FRESH WATER DRAIN COLLECT. TANK & DRAIN PUMP
42	MAIN THROTTLE VALVE
43	SOLUBLE POISON CHARGE PUMP
44	SAMPLING BOMB CABINET
45	SWITCHBOARD
46	PURIFIED WATER SUPPLY TANK
47	TURBO-GENERATOR
48	SHIPS STORES REFER. COMPRESSOR UNIT
49	CARGO REFER. COMPRESSOR
50	CARGO REFER. CONDENSER
51	CARGO REFER. RECEIVER
52	SOLUBLE POISON SUPPLY TANK
53	MAKE UP DEMINERALIZER
54	CUBICLE FOR CONSOLE EQUIP.
55	REACTOR CONTROL & RADIATION MONITORING CABINET
56	CONTROL CONSOLE
57	C.V. VENT. BLOWER, FILTER & RADIATION MONITOR

5. ECONOMIC ANALYSIS

5.1. Introduction

The goal of the Federal Government's merchant ship reactor program is the development of economical nuclear power for merchant ships. Therefore, each new design must be analyzed for economic attractiveness and technical feasibility. In this analysis the CNSG power plant is considered for a specific ship in a specific trade using one of several existing essential trade routes. The ship is the design currently used by the Maritime Administration in its ship development and improvement program.

5.2. Ship Application

Ideally, a comparison between conventional and nuclear power plants for ship propulsion is made by selecting a particular trade route and designing a ship of each type to serve it. Both power plants are designed for maximum efficiency, and both ships have identical schedules and cargo space. On these bases, the cost of operating each ship is computed and compared.

The foregoing procedure could not be used in this study owing to limited manpower and time and the lack of a naval architect. Therefore, one basic design was used for both the conventional and the nuclear ship.

Chosen for this anlysis was a high-speed dry cargo ship with improved cargo handling and storage. The preliminary design has been developed by the Maritime Administration and designated as PD-108.[1] The major parameters of this ship are listed below.

PD-108 Parameters

Length between perpendiculars, ft-in.	528-0
Beam moulded, ft-in.	81-0
Draft moulded full load, ft-in.	28-0

Displacement full load, tons	18,800
Deadweight, tons	9,850
Total under deck bale, ft^3	798,083
Normal shaft horsepower	27,500
Speed, knots	22.75
Cruising radius, miles	10,000

PD-108 was chosen for these reasons:

1. It represents the latest technology in ships and is thus likely to be representative of ships built in the near future.

2. Since the American Merchant Marine has more dry cargo ships than any other kind, they are of particular interest to the future market.

3. Dry cargo ships are often subsidized in order to meet foreign competition; hence, the Federal Government is directly interested in improving the economy and reducing operating costs for such ships.

4. The Maritime Administration is using this design in present studies on improvement of cargo handling.

Study of a preliminary design for the conventional ship showed that the CNSG machinery would fit into the conventional machinery space (see Section 4). The Number One deep tanks and the fuel oil settling tanks, not utilized by the nuclear ship, were considered available for cargo and added about 500 tons to the ship's cargo capacity. Weight estimates showed that when both ships are carrying identical cargoes the displacement leaving port is about 1,700 tons greater for the conventional ship. This difference is due to the fuel requirement.

5.3. Trade Route

Two factors affecting the choice between nuclear and conventional power are the length of the route and the cost of fuel. These parameters are presented for selected trade routes in Table 5-1. The route numbers and the frequency of service are from Reference 2. The length of the trade routes are approximate as the particular ports of call are unknown. Since fuel oil is cheaper in the United States, fuel oil will be bought at the United States end of the voyage when practicable. As the length of the voyage increases, the probability of this practice diminishes and the average price paid for fuel oil increases. On the longer voyages, fuel costs are further increased because the ship spends more time steaming at sea. The longer trade routes therefore tend to favor nuclear power.

The trade route assumed for this analysis is from the east coast of the United States to Japan, a part of the Maritime Administration's essential trade route No. 12. The fastest, most modern American ships are used in this trade. The ship is assumed to serve three ports in the United States and two in Japan. The schedule derived for this service is shown below.

Condition		Time per voyage, days
Sea transit		35.5
Canal transit		2
Local interport transit		1.6
Inport		5.6
Margin		1.0
	Total days	45.7

Time in port is based on Reference 1, and time at sea on operation at rated sea speed (22.75 knots). No attempt was made to fit a given frequency of service, since the ship was not designed solely for this trade.

The conventional ship was assumed to carry a cargo of 5,700 tons. Although not a full cargo, this tonnage is considered practical since cargo ships do not always carry a full load. Maximum cargo is 7,000 tons, but only 6,500 tons of typical cargo could be carried without excessively tight storage. Because of its additional capacity, the nuclear ship was assumed to be filled on a volume basis to the same percentage of capacity as the conventional ship. The capacity then is 6,000 tons, and limitations for a typical cargo are 7,000 tons on a volume basis and 8,700 tons on a weight basis.

Table 5-2 shows that the average displacement during the voyage is approximately 16,000 tons for both ships. This displacement requires 24,100 shaft horsepower to propel the ship at rated speed, allowing a 15% margin for seaway and bottom fouling. This horsepower was used in calculating fuel consumption.

Table 5-1. Distance and Cost of Fuel for Selected Tade Routes

Route no.	Route	Sailings per month	Distance, miles	Fuel oil cost, $/barrel U. S.	Foreign	Average
12	US Atlantic Ports & Far East	15	9,700	2.215	3.45	2.83
15A	US Atlantic Ports to South & East Africa	5-6	7,500	2.215	3.20	2.68
15B	US Gulf Ports to South & East Africa	1-2		2.20	3.20	2.70
16	US Atlantic Ports to Australia-New Zealand	2	11,000	2.215	3.62	2.89
22	US Gulf Ports to Far East	9-11	9,100	2.20	3.45	2.82
26	US Pacific Ports to Western Europe	5	9,100	2.34	3.03	2.68
27	US Pacific Ports to Australia	2-3	7,000	2.34	3.62	2.98
29	US Pacific Ports to Far East	29-37	6,000	2.34	3.45	2.90

Table 5-2. Ship Displacements and Weights

Item	Conventional ship, long tons	Nuclear ship, long tons
Light ship displacement	8,722	9,676
Crew, stores, fresh water and lube oil	366	366
Cargo	5,700	6,000
Fuel at end of voyage	242	0
Displacement at end of voyage	15,030	16,042
Fuel consumed	2,420	0
Displacement at beginning of voyage	17,450	16,042
Average displacement	16,240	16,042

5.4. Reactor Plant Costs

Reactor plant costs, which are based on the design presented in this report, were complicated due to the use of a 27,500 shaft horsepower plant for the economic studies (the basic design was for a 20,000 shaft horsepower plant). The Maritime Administration supplied cost data for installation, collision barrier, containment, and auxiliary system piping. These costs included shipyard engineering. B&W supplied data for nuclear equipment and contractor engineering. Cost data are shown in Table 5-3.

Economic studies were based on the assumption that the power plant is the second of its kind to be built. It was further assumed that five "second of a kind" ships were built at one time for one owner, since construction of five ships at the same time reduces the cost of engineering and increases the efficiency of installation. This is current practice for the construction of cargo ships.

Costs for the pressure vessel, head, and steam generator were estimated by B&W's Boiler Division according to the procedures used in preparing fixed-price bids for commercial equipment. The cost was $2,200,000 for the first of a kind and $1,700,000 for each of the second-generation units. The Boiler Division also estimated the cost of control rods and plug rods. Costs for the pressure vessel internals (lower grid plate, upper lateral support plate, core container, and follower shroud) were estimated by B&W's Atomic Energy Division based on previous prices for similar equipment.

The cost of cladding the reactor with lead was provided by a vendor who would perform the operation in B&W's shop. The cost of lead and vessel handling was estimated by the Boiler Division, which will supply the lead and maneuver the vessel during the cladding operation. Vendors also furnished costs for control rod drives and primary pumps. Instrumentation costs were interpolated from previous vendor quotations for similar systems. The costs for systems are for components such as pumps, heat exchanger, and valves. The cost of piping was included in the Maritime Administration's estimates for equipment to be furnished by the shipbuilder. Most of the costs for components were obtained from vendors.

Table 5-3. Reactor Plant Costs

Item	Cost for 20,000-shp ship	Cost for 27,000-shp ship
Pressure vessel, head & steam generator	$1,700,000	$1,870,000
Pressure vessel internals	142,200	142,200
Pressure vessel lead cladding	145,000	167,000
Control rod drives	231,000	231,000
Control rods, plug rods & source	127,000	127,000
Primary pumps	192,500	192,500
Containment cooling system	82,500	101,500
Chemical control system	58,300	71,800
Relief system	4,290	5,280
Instrumentation	341,000	352,500
Total - Equipment furnished by Contractor	$3,023,790	$3,260,780
B&W engineering		218,000
Shipyard equipment, installation & engineering		1,068,000
Changes under contract		114,400
Startup		200,000
Owner's engineering		102,000
Total - power plant		$4,963,180

In addition to the $500,000 savings for pressure vessel and steam generator, the second-generation control rod drives and primary pumps will cost $82,000 less than their prototype. Other costs are unchanged. The foregoing costs, based on a 20,000-shp plant, had to be adjusted for the 27,000-shp plant.

The core is one of the components that determines the size of the pressure vessel. Late in the design study, it was found that the size of the core could be reduced by using 0.400-inch-diameter fuel pins instead of the 0.46-inch pins of the reference design. The size of the core for the 27,000-shp design, determined by scaling up the 20,000-shp, 0.400-inch fuel pin core, was found to be the same as that for the 20,000-shp, 0.460-inch fuel pin core. Therefore, the size of the pressure vessel had to be increased only to accommodate the increased area required for steam generator heat transfer. This was done by increasing the reactor vessel height 2-1/2 feet and maintaining the diameter constant. The cost of the vessel and lead cladding was increased in proportion to the size. The cost of reactor internals, control rods, plug rods, and control rod drives was unchanged. The cost of the pumps was unchanged since changes in capacity have little effect on the cost of low horsepower canned rotor pumps. Pump capacity was assumed to increase in direct proportion to reactor power level. Systems costs were increased in proportion to the two-thirds power of the horsepower. Instrumentation costs were essentially unchanged. The cost of remotely controlled valves, included under instrumentation, was increased in the same proportion as were systems. The shipyard items estimated by the Maritime Administration were based on 27,000-shp and "five of a kind".

The changes under the contract were taken as 2-1/2% of the cost of the plant, excluding startup and owner's engineering. Since this is a "second of a kind" plant, there seemed to be little need for further expensive changes. No escalation was used because the immediate past shows little change in costs from year to year.

5.5. Ship Costs

The cost of steel, outfit, and propulsion machinery for the nuclear ship, and all costs for the conventional ship, were furnished by the Maritime Administration. Cost of owner's engineering was $200,000 for the conventional ship and $300,000 for the nuclear ship. The latter cost included $30,000 per ship for additional hazards studies. Cost of owner's engineering for the hull and outfit of both ships was taken as $136,500. Of the remaining owner's engineering for the nuclear ship, $61,500 was levied against the propulsion machinery and $102,000 against the nuclear machinery.

The cost of both ships includes 7-1/2% for interest during construction. This standard item has been adopted for other studies and is dependent on the construction schedule for each ship. At this time, however, there is insufficient information on which to base variations.

5.6. Annual Costs

The cost of fuel oil for the conventional ship was based on Esso International Corporation quotations for February, 1962. These are spot prices in the U. S., less the voluntary reduction on the East Coast, and spot prices abroad. The fuel cost used is the average of U. S. and foreign costs, since the trade route is long enough to require refueling at both ends. This procedure differs from that used in the Three-Tanker Study[3] which used an average expected price of $2.70 per barrel during the life of the ship. This was, and still is, well above the current market price of fuel oil in the port (Philadelphia or New York) at which the ships evaluated were assumed to refuel.

The actual price of fuel oil over the next twenty years is of course speculative. Although the procedure used in this report may be no more correct than one based on a price extrapolated from previous long-term trends, it does correspond to current practice and is slightly more conservative than long-term averages. Fuel cost for the nuclear ship is that derived for the equilibrium fuel cost for the core in a 20,000-shp plant generating 2.3 mils per shaft horsepower hour. The fuel cost for a 27,000-shp plant was assumed to be the same, although it might be slightly less because of the smaller fuel inventory per megawatt hour of heat generated.

Wages, overtime, and subsistence were assumed to be the same for both nuclear and conventional ships. The crew of the nuclear ship should be smaller, but wage rates may be higher.

The Maritime Administration supplied costs for maintenance and repair of the conventional ship. They also suggested using $120,780 for the conventional portion of the nuclear ship. Nuclear maintenance and repair was estimated as $35,000 per year, which is approximately 0.7% of the capital cost of the nuclear steam supply system. Use of the 0.7% factor was suggested in Reference 5 and is based on central station power plant experience.

Insurance for hull and machinery, and war risk insurance, was taken as 1.35% of the investment. Protection and indemnity insurance was taken as $3.50 per gross ton. Nuclear liability insurance was based on Reference 4 and taken as 0.7% of the nuclear and propulsion plant cost. This procedure differs from previous studies. The procedure used in Reference 3 considers the nuclear liability insurance premium for a tanker as a percentage of ship investment. This procedure was unsatisfactory considering that the same power plant, though placed in different ships, should entail approximately the same risk and extent of liability. The use of 0.7% of the power plant investment relates the premium to the power plant directly. This is based on shoreside experience.

Depreciation and interest was based on 5% interest plus 1/2% mortgage insurance, twenty-five-year life, and 2-1/2% scrap value. These parameters are in line with recent practices for Title XI mortgages which are commonly used by shipowners.

5.7. Subsidies

To determine costs on a subsidized basis, the following operating subsidy rates from Reference 1 were used:

Item	Percentage of cost
Wages	70
Subsistence	12
Maintenance and repair	40
P & I Insurance	70
H & M Insurance	22

Construction subsidy was assumed to be 50% of construction cost.

To obtain the data shown in Table 5-6, the cost of the ship to the owner was reduced by a factor of two. The insurance premiums were then computed using the procedure just described and the reduced cost of the ship to the owner. The subsidy rates listed were then applied to these premiums. Wages, subsistence, and maintenance and repair were obtained by applying the subsidy rates to the data in Table 5-5. Depreciation and interest was based on the reduced cost of the ship to the owner in accordance with the parameters described in the foregoing section.

5.8. Conclusions

On an unsubsidized basis, the transportation cost is virtually identical for both ships. There is a 1% differential, which is within the accuracy of the estimate. Under subsidy, the nuclear ship is more economical to operate than the conventional ship. Here the difference in cost is great enough to indicate a clear advantage for the nuclear ship. Transportation costs for both ships are presented in Tables 5-4 and 5-5.

5.9. Non-Capitalized Costs

Non-capitalized costs — non-recurring items which do not appear in the "second of a kind" analysis — are shown below.

First of a kind engineering	$ 2,950,000
Crew training	200,000
Research & Development	922,900
Refueling machinery	Undetermined
Fuel costs in excess of equilibrium	170,000
Total	$ 4,242,900

Table 5-4. Transportation Cost Without Subsidy

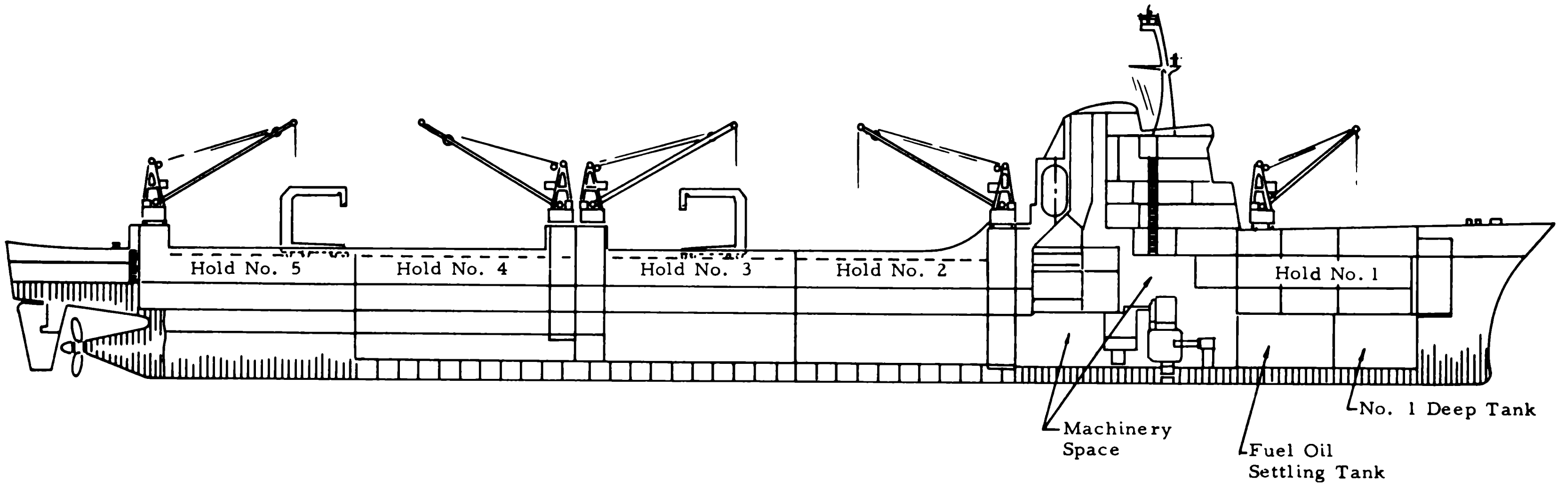
Item	Conventional ship	Nuclear ship
Salient parameters		
Sea speed, knots	22.75	22.75
Voyages per year	7.66	7.66
Average cargo load, long tons	5,700	6,000
Cargo per year, long tons*	87,300	91,900
Ship costs		
Steel	$ 3,550,000	$ 3,550,000
Outfit	6,350,000	6,350,000
Conventional machinery	4,600,000	4,240,000
Nuclear machinery	------	4,960,000
Subtotal	$ 14,500,000	$ 19,100,000
Interest during construction	1,090,000	1,430,000
Cost to owner	15,590,000	20,,530,000
Annual costs		
Wages, overtime, etc.	$ 505,500	$ 505,500
Subsistence	26,900	26,900
Stores, supplies & equipment	59,140	59,140
Fuel	706,000	379,000
Maintenance & repairs	140,780	155,780
Insurance	245,000	313,000
Nuclear liability insurance	-----	70,000
Voyage expense	36,400	36,400
Canal fees	133,000	133,000
Pallets	27,000	27,000
Depreciation & interest	1,154,000	1,520,000
Total	$ 3,033,720	$ 3,225,720
Transportation cost; $/ton	$ 34.75	$ 35.10

* Cargo is carried both directions.

Table 5-5. Transportation Cost With Subsidy

Item	Conventional ship	Nuclear ship
Cost of ship to owner	$7,795,000	$10,265,000
Annual costs		
Wages, overtime, etc.	151,500	151,500
Subsistence	23,700	23,700
Stores, supplies, & equipment	59,140	59,140
Fuel	706,000	379,000
Maintenance & repair	84,500	93,500
Insurance	83,200	106,300
Nuclear insurance	-----	70,000
Subtotal	$1,108,040	$ 883,140
Voyage expense	36,400	36,400
Canal fees	133,000	133,000
Pallets	27,000	27,000
Depreciation & interest	577,000	760,000
Total	$1,881,440	$ 1,839,540
Transportation cost; $/ton	$21.55	$20.00

Figure 5-1. High-Speed Dry Cargo Ship
(Maritime Administration Design PD-108)



6. RESEARCH AND DEVELOPMENT

6.1. Introduction

Some features of the CNSG design will require experimental verification. Each of these features has been studied thoroughly by utilizing the available experimental knowledge. On this basis, these features are feasible, but only actual experiments will provide the detailed design information necessary to build the CNSG.

The research and development program presented here is necessary to the construction of the CNSG design. This scope was chosen to provide a complete, consistent plan. Other research needed to improve the technology of pressurized water reactors beyond the CNSG design has been suggested to the AEC.

Consideration was given to the construction of a CNSG prototype. Prototypes have been built for several advanced reactors to hasten the proof of their feasibility and to limit the investment made in a concept before it is proved attractive. A CNSG prototype is neither desired nor required.

It has been argued that use of a prototype will speed up development by reducing shipboard testing time. However, on the Savannah, most shipboard testing resulted from quality control problems and regulatory body requirements. Both of these problems will recur regardless of whether a prototype exists or not. There will be some reduction in testing time due to preliminary testing performed on the prototype if it exists.

The construction of a prototype and a shipboard power plant takes approximately the same length of time. Incorporating the plant in a ship does not cause any appreciable delay since fabrication of the power plant must start before that of the ship. The tests accomplished in a prototype are less complete than those performed on board ship. The prototype cannot reproduce ship operating conditions exactly, particularly ship motion and vibration. One of the purposes of the first CNSG is to convince shipowners that a CNSG is economical and will work satisfactorily. For these purposes, a ship based plant will be much more effective than a prototype. (See also Section 8.9.)

6.2. Boric Acid

Boric acid is specified for lifetime shim control and for override of the excess reactivity needed to offset the temperature deficit. Although several installations report limited use of boric acid for these purposes, further information regarding the effect of radiation on boric acid is required for confirmation. (See Section 7.4 for current data on the use of soluble poisons for reactivity control.)

A series of experiments using the MARAD loop at ORNL has been recommended, but no cost estimates are available. The boric acid costs given in Section 6.7 cover liaison with ORNL and the incorporation of their work into the CNSG design.

6.3. Lead Bonding

Lead is bonded directly to the reactor vessel to provide shielding and to control the heat flow out of the reactor vessel.

Several vendors have been contacted to develop cost and feasibility information. Although one vendor believes that lead can be successfully bonded to a CNSG reactor vessel, no successful demonstration has been performed under the conditions of temperature and vessel size specified for the current design. Therefore, B&W recommends that a limited number of confirmation tests be performed.

The potential experimental apparatus is a 14-foot-long, 1 foot diameter pressure vessel having 8 inches of lead bonded to the outside surface. Internal electric heaters are used inside the vessel which is positioned vertically and immersed in a tank of water. This vessel is cycled between hot and cold conditions, and is operated hot for long periods. Dial indicators and thermocouples indicate the lead bond conditions.

6.4. Pressure Suppression

The water in the containment has many uses including the suppression of pressure due to primary system ruptures. For this purpose, it is necessary to understand the behavior of hot high-pressure water suddenly released into relatively cool water at atmospheric pressure.

A facility has been built at the B&W Research Center to investigate this phenomenon. (See Section 8.1.) Tests of ruptures up to 6 inches in diameter have been performed. The facility has been shut down since the current test series has ended. To date, the tests have shown the feasibility of pressure suppression, but have not developed the theory well enough to extrapolate to larger ruptures or different geometries. B&W recommends that further tests be performed in this facility to better establish the theory. Such tests will cost approximately $160,000.

If theory can not be firmly established in the present facility or if regulatory agencies insist on full size tests, an allowance of $585,000 is included in research and development cost estimates.

6.5. Containment Materials

For safety, a soluble poison must be added to the containment water. Since boric acid is not compatible with the cheaper construction materials used in the containment at low pH, potassium hydroxide will be added to raise the pH to a satisfactory level of from 8.5 to 9.5. Corrosion tests will confirm this process.

6.6. Vapor-Liquid Interface

The vapor-liquid interface in the CNSG is due to self-pressurization. The water at this interface contains boric acid. Corrosion information is not available for certain high-strength materials (Inconel, 17-pH, etc.) commonly used for control-rod driveline components. If these materials are selected, corrosion tests will be necessary.

6.7. Estimated Cost of Research and Development

		Cost, $
Boric Acid*		20,720
Lead Bonding		173,000
Pressure Suppression		745,000
Containment Materials		130,000
Vapor-Liquid Interface		76,000
	Total	1,144,720

* For liaison and incorporation of information only.

7. ALTERNATE DESIGNS

7.1. Introduction

The CNSG design evolved from several studies described here.

The initial designs were based on the IBR concept with some components from the Savannah design. This approach was sufficient in the early phases to develop some of the design concepts. However, the use of the same core in both natural and forced circulation systems did not present an adequate design comparison. Since the forced circulation system could have a smaller core, the ground rules were changed to use components from a previous design study. The features of the reference design were based on these initial design studies and on other information developed by B&W.

Additional studies were made concerning the particular application of the concept and the incorporation of certain spectral shift control aspects.

7.2. Preliminary Design

The preliminary design directly preceded the reference CNSG design. Figure 7.1 shows a cross section of the reactor and the containment. All CNSG design elements are present in the preliminary design, but the following improvements have been made.

1. Pump Location — The pumps have been moved away from the core to permit ready maintenance. Also, space for the suction and discharge passages is not restricted.

2. Heat Exchanger Tube Sheet Configuration — The heat exchanger may be removed with the head. Therefore, a smaller riser is used, and the pressure vessel ID is reduced from 108 to 94 inches. This change also decreases the pressure vessel volume by about 30%. The tube sheets located on the head makes tube plugging easier.

7.3. Compacted Nuclear Steam Generator

A compacted reference design was studied to determine if the height could be decreased. Table 7.1 gives the major parameters for this study. The height can be decreased only if the reactor diameter is increased.

Table 7.1. Compacted Nuclear Steam Generator

Maximum shp	30,000
Reactor power, MWt	86
Over-all thermal efficiency, %	26
Operating pressure (primary), psia	812
Design pressure, psig	1,100
Coolant core outlet temperature, F	520
Steam temperature, F	515
Steam pressure at the throttle, psig	400
Total heat transfer area, ft^2	11,316
Coolant flow rate, lb/hr	6×10^6
Number of tube circuits	247
Average tube circuit length, ft	175
Tube size, OD × thickness, in.	1 × 0.083
Number of inlet headers	4
Number of outlet headers	4
Number of primary pumps	8
Capacity of pumps, gpm	1850
Number of control rods	6
Metal-to-water ratio (unit cell)	0.70
Fuel assembly geometry	hexagonal
Reactor vessel ID, in.	124
Outside containment height	29 ft, 3 in.
Secondary shield, OD, ft	22

7.4. Spectral Shift Study

The effects of substituting D_2O for boric acid in shim control were studied to determine if decreased fuel costs could compensate for D_2O costs. The fuel cost reduction was small (2.31 mills/shp-hr versus 2.35 mills/shp-hr). (See Section 8.3.3.)

7.5. Civil Defense Ship

A nuclear powered cargo ship to be converted into an emergency mobile power plant was considered. Ground rules, established by the Maritime Administration, stated that only single reactor propulsion plants with extra size and equipment for the emergency conditions were to be used. Although it was recognized that the turbo-generator condenser system was a large factor in the convertible plant design, B&W was to consider only the nuclear steam generating unit.

To establish the ship's general feasibility, a preliminary core design, a fuel cost summary, and a plant arrangement and weight summary were made.

The following arbitrary assumptions were made for this study.

1. Normal ship propulsion
   Shaft horsepower 30,000
   Auxiliary power, kwe 1,600
2. Emergency power generation, MWe 100
3. Potential core capacity must be maintained during normal operation to provide emergency power for at least six months.
4. The reduced maneuvering requirements and the state of emergency warrants operation with less conservatism than is necessary for ship propulsion.
5. The design should not incorporate features requiring development.
6. The plant should be able to fit the size and weight restrictions of a 30,000-shp ship.

A design was selected to incorporate a natural circulation IBR for normal propulsion service. The integral heat transfer surface is oversized to produce the emergency power at a reduced steam pressure. Steam water separation takes place in external steam drums connected by risers and downcomers to the side of the boiler tubes. Forced circulation of the primary fluid increases the flow during the emergency operation.

A pin-type canless stainless-steel clad core is used. This core has been loaded to provide just over five normal full power years before it burns into the emergency reserve of 72,000 MWd (6 months at 400 MWt).

An average power density in the core, obtained during the emergency operation, is higher by a factor of 2.7 than is permitted by present ship propulsion design limits. This increase is attributed to:

1. The overpower core design point can be set at 120% instead of 130% because of the reduced requirements for power transients.

2. By removing the control rods and holding down the excess reactivity with soluble poisons, a better peak-to-average power distribution can be obtained. This factor is appreciable for radial power distribution and provides small benefits in axial power distribution. The use of soluble poison for emergency use not exceeding six months is acceptable, even though it is not now approved for long term ship propulsion.

3. At present, the maximum heat flux is limited to provide a confidence level of 99.4% based on available burnout data. By reducing this level to 97.9%, a 20% heat flux increase can be attained.

4. By doubling the primary water flow, the power can be increased by 60%. This requires operation with primary circulating pumps. The normal operation flow requirements would be supplied by natural circulation. Bypass gates must be closed when the conversion to emergency power is made.

5. The reduced requirements for power transients when operating to produce emergency electricity permits the margin to be reduced between primary system operating pressure and the vessel design pressure. The resulting higher temperatures permit 5% additional power from the same core.

To obtain 100 MWe from the core during an emergency, the core size must be increased by a factor of 399 MWt/82.5 MWt × 2.7 = 1.79. Preliminary core and plant data are given below.

1. Normal Propulsion Power, 30,000 shp + 1600 MWe

Containment vessel design pressure, psig	181
Primary design pressure, psig	2000
Primary operating pressure, psig	1750
Average primary temperature, F	506
Primary δ temperature, F	46
Primary flow (natural circulation) lb/hr	5.2×10^6

Steam pressure, psig	500
Feed temperature, F	340
Steam flow, lb/hr	316,000
Q reactor, Btu/hr	282×10^6
Q reactor, MWt	82.5

2. Emergency Power, 10,000 kwe

Primary operating pressure, psig	1900
Average primary temperature, F	521
Primary δ temperature, F	57
Primary flow (forced circulation) lb/hr	20×10^6
Steam pressure, psig	400
Steam rate at 10 in. Hg absolute, lb/kwh	14.9
Steam flow, lb/hr	1,490,000
Feed temperature, F	220
Q reactor, Btu/hr	$1,362 \times 10^6$
Q reactor, MWt	400

3. Core

Equivalent diameter, in.	61
Active height, in.	67
Average enrichment, %	4.19
Total UO_2, kg	14,120
Average burnup, MWd/MWt	16,000
Core life, MWd	226,000
2738 days at 82.5 MWe	
or 1865 days at 82.5 MWe + 180 days at 400 MWe	
or 0 days at 82.5 MWe + 565 days at 400 MWe	

4. Component Weight, lb

Reactor vessel, core, internals, tubes, and drives (shipping weight of vessel = 550,000 lb)	756,000
Steam drums, piping, pumps and insulation	174,000
Platforms and steels	80,000
Auxiliary systems	100,000
Water — primary	137,000
— secondary	29,000
Containment vessel	524,000
Shielding	4,442,000
Total	4,996,000
Comparable figure for 69-MWt NS Savannah	6,000,000

5. Fuel Costs (based on refueling after 1865 days at 82.5 MWe even though 72,000 MWd of available power remains in the core.)

Fabrication, mills/shp-hr	0.81
Fissionable material	0.87
Inventory (assumed 70% load factor)	1.30
Transportation and insurance	0.08
Processing and conversion	0.32
Sub-total	3.39
Control rods (one set per core)	0.13
Working Capital	0.22
Total	3.74

6. Fuel Costs, mills/shp (based on refueling after 2738 days at 82.5 MWe while completely using the core.)

Fabrication	0.52
Fissionable material	0.87
Inventory	1.18
Transportation and insurance	0.05
Processing and conversion	0.20
Sub-total	2.82
Control rods	0.08
Working capital	0.42
Total	3.32

For emergency operation, the turbine back-pressure is maintained at 10 inches Hg to keep from greatly enlarging the condenser. The optimum combination of condenser, turbine or multiple turbines, and the cycle parameters has not been determined.

The external, vertically mounted canned pumps, which require no development, are accessible for maintenance. However, undesirable external primary piping is required.

The tube bundles may be replaced from inside the vessel. Individual tubes can be plugged only by removing the bundle. With sixteen bundles, however, they may be isolated by capping the risers and downcomers. Severe leakage may be confined remotely by closing the steam throttle valve and isolating one steam drum system.

An integral pressurizer will utilize the electric heater elements installed through the pressure vessel wall. Subcooling, accomplished by a multi-plate flow baffle and thermal barrier, maintains a higher temperature in the pressurizer region than in the lower regions.

7. Principal Alternatives

Several alternative designs are listed below.

1. A concentric core design may be used which has a central replaceable core for normal power and a concentric annular core (normally completely suppressed by control rods) for the higher power level. The central core would be replaced only as it is used up in commercial operation, and the standby-emergency core costs would not increase the normal power fuel costs.

2. The reactor vessel size could be reduced by installing the excess-standby heating surface in an external boiler. This would require more piping and complex arrangements.

3. A once-through secondary steam system with some superheat for normal power operation may be used. The conversion will be to a saturated recirculation cycle for emergency power.

4. Further study will reduce the total height of the containment vessel and will make it more acceptable for some ships.

7.6. Advanced High Power PWR Concepts

The economic perspective of American shipping is reviewed in Section 1 of this report. Basically, advancement in this industry requires technical improvements to increase the productivity of maritime labor. Major areas to be exploited are propulsion system analyses, automation, and cargo-handling.

Improvements in machinery should be made for better operational economy. Concerning this subject, the WALRUS[1] report basically calls for increases in ship's speed, cargo handling, and mechanization. According to this report it is profitable to operate 30-knot cargo ships; and 40-knot ships are technically feasible, but are not commercially feasible for a 3000- to 4000-mile range. The report also recommends that on a long term basis, prototype ships of advanced design should be built to establish the feasibility of new developments.

Improvements in ship design and machinery have been based on the use of fossil fuels. Now that nuclear fission is a demonstrated source of energy, marine power plant designs should include the possible use of nuclear fuels.

With large horsepowers, nuclear power becomes more economical. Also, more horsepower means higher ship speeds. Because nuclear fission permits the release of large quantities of energy from a small space, nuclear power can be used to produce economical ships of high speed with practically unlimited range. Such ships cannot be realized, even for a 3000- to 4000-mile range, with a conventional plant because of the space and displacement required to carry the large amounts of fuel oil. Since a nuclear ship eliminates the fuel carrying problems and permits unlimited operating range, the naval architect has more leeway to optimize ship designs for foreign commerce and world-wide national defense.

The development of power reactors has involved several concepts using various fluids. However, H_2O has more background experience and knowledge behind it than any other fluid. Since the pressurized water reactor is an outgrowth of years of experience by the steam boiler industry, all nuclear propelled ships use this reactor concept. This experience permits an orderly development of advanced designs with more engineering integrity and realistic cost estimates than can be achieved with any other reactor concept, and these designs are particularly applicable to marine propulsion.

Since reliability, weight, and space are major considerations in the design of propulsion machinery, our marine PWR designs have aimed at consolidation of the plant and systems while minimizing research and development. The CNSG is one result of this type development. Table 7.2 compares the CNSG and the advanced PWR designs with the Savannah power plant.

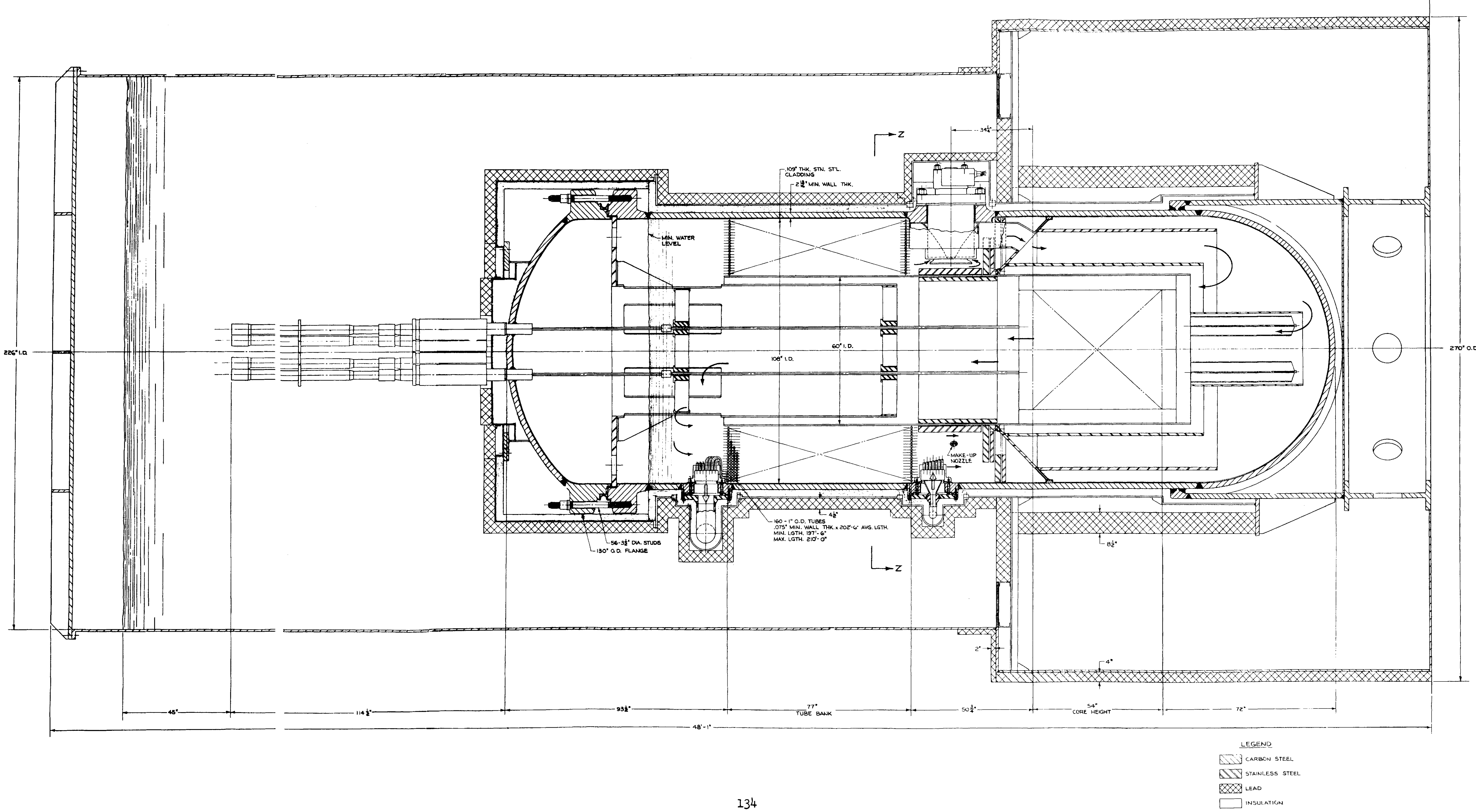
Table 7.2 emphasizes the advances made in marine PWR designs in a relatively short time. Further efforts in design and consolidation will result in increased outputs and weight, space, and costs can be reduced without extensive nuclear research and development. Also, there is considerable design flexibility since forced or natural circulation may be used in any combination on the primary and secondary sides, and a wet or dry containment may be used. Therefore, it is concluded that the PWR concept with its advanced design will result in economical, high speed ships of unlimited range.

Table 7.2. Reactor Plant Parameters

Reference Unit	A Savannah	B IBR	C CNSG	D CNSG	E IBR	F UPIR
Steam Separation	External Boilers	External Drums	Once Through	Once Through	Internal	Internal
Circulation	Forced	Natural	Forced	Forced	Natural	Natural
MW-t, normal	63.5	62	62.4	86	200	400
shp, normal	20,000	20,000	20,000	27,500	65,000	135,000
Primary Pressure, psia	1750	1750	815	815	1700	3500
Average Primary Temperature, F	508	508	505	505	545	710
Secondary Pressure, psia	470	470	415	415	515	1500
Secondary Temperature, F	460	460	515	515	470	595
Steam per hour, lb	245,000	240,000	225,000	310,000	770,000	1,600,000
Containment length	50 ft, 6 in.	25-ft diam	19-ft, 6-in. diam	19-ft, 6-in. diam	34-ft diam	34 ft
width	35-ft diam	--	--	--	--	--
height	51 ft	43 ft, 6 in.	38 ft	40 ft, 6 in.	56 ft	56 ft
Plan Area, ft^2	1485	490	300	300	900	900
Wet Weights						
Reactor System,* lb	1,989,000	1,385,000	919,000	1,031,000	3,290,000	3,345,000
Secondary Shield, lb	3,508,000	1,715,000	615,000	650,000	2,807,000	2,807,000
Total, lb	5,497,000	3,100,000	1,534,000	1,681,000	6,097,000	6,152,000
Total, long tons	2,452	1,384	685	750	2,720	2,746
lb/shp	275	155	77	61	93	46
shp/ft^3 of Containment	0.51	1.16	1.77	2.38	1.59	3.30
Approximate Speed, knots	21	21	21	23	27	33

* Includes vessel, control drives, internals, core, auxiliary systems, containment vessel, support steel, primary shielding, and insulation.

Figure 7-1. Dual Purpose Plant
(82.5/400 MWt)



8. SUPPORTING STUDIES

8.1. Hazards Analysis

8.1.1. Preliminary Hazard Studies

The objective of the preliminary hazard studies has been the comparison of the CNSG to the Savannah for similar accidents. For this comparison, all areas external to the primary system (e.g., secondary system and ship structural integrity) were assumed to be equally safe in design. The following accidents were compared:

1. Reactivity Accidents
2. Mechanical Failures
3. Ship Accidents
4. Maximum Credible Accident

8.1.2. Safety Aspects of a Water-Filled Containment

The inherent characteristics of a water-filled containment provide unique safety features that are unobtainable in a standard "dry" containment. These features are available at all times and require neither operator action nor automatic operation of any mechanical device. The most important of these features are:

1. In the event of an MCA, the maximum pressure buildup within the containment is on the order of one-tenth that of a "dry" containment of equivalent diameter. Furthermore, the peak pressure is only a transient condition, thus the pressure-driving force required to expel fission products from the containment exists for only a short period of time.

2. There is a good possibility that the containment water will re-enter the primary system after the initial blowdown, and thus tend to prevent core meltdown and the subsequent release of fission products. Further study is required to confirm this possibility.

3. In the event of a core meltdown, a large quantity of the fission products released will be dissolved in the containment water, thus reducing the environmental hazard to the surroundings. In this sense, the water presents an additional containment barrier to the release of fission products to the environment.

4. In the improbable event of the complete loss of cooling water and all electrical power, the containment water will have sufficient heat storage capacity to safely absorb the total integrated decay heat from the reactor for a minimum period of 37 hours. This time period is based on zero heat loss from the containment to its surroundings.

8.1.3. Reactivity Accidents

To evaluate rod mishandling accidents, the maximum rates of reactivity addition for the CNSG and the Savannah cores were compared. The rates were found to be approximately equal. Since the kinetic characteristics of the two cores are comparable, it is concluded that rod mishandling accidents will not result in a radiological hazard to the environment.

Since reactivity is also controlled by the boric acid concentration in the primary system, the rate at which reactivity can be added to the core through mal-function of the soluble poison system was analyzed. With the soluble poison system operating at maximum dilution capacity, the rate of reactivity addition to the core is less than that possible in a rod mishandling accident. Therefore, any accident involving the operation of the soluble poison system will result in a reactivity addition that is less severe than that from rod mishandling accidents.

If the primary system is accidentally cooled, reactivity is added to the core due to the negative temperature coefficient of the moderator. This accident would add reactivity to the CNSG at a rate comparable to that for the Savannah, so no severe excursion will occur. No cold, stagnant volumes of water can accumulate in the CNSG, so operational interlocks for the prevention of cold-water accidents are not required. If the primary system is cooled to a low temperature (exact temperature depends on core life) without an addition of boric acid, the reactor can go critical even after the control rods have scrammed. The most severe cooling transient, and therefore the most rapid reactivity addition, would occur during the loss of coolant accident analyzed in the next section. Even in this accident, there is no destructive excursion.

8.1.4. Mechanical Failures

8.1.4.1. Primary Coolant Leak

Preliminary evaluation indicates that the primary system rupture accident will not be more severe than the same accident for the NS Savannah.

Method of Analysis

Leak analyses were conducted to determine the reactor conditions during blowdown. Two kinds of leaks were assumed: bleedoff of water only, and bleedoff of steam only. These two cases represent the two extremes, and results indicate that the steam leak probably represents the worst case.

The following assumptions were made for the leak analyses:

1. Decay heat is neglected. This is a good assumption for rapid transients because the integrated heat during blowdown is small. For slow transients, this assumption is conservative because the decay heat will decrease the cooling rate.

2. Stored heat in the reactor vessel and internals is neglected. This is a good assumption for rapid transients because of the relatively long time constant for heat transfer to the coolant. For slow transients, this assumption is conservative, because the transfer of stored heat to the coolant will decrease the rate at which the coolant temperature decreases.

3. The initial conditions for the transient are assumed to be normal operating conditions of 100% power.

For the water leak analysis, the coolant was divided into a hot mass (12, 000 lbs at 520 F) and a cold mass (22, 100 lbs at 490 F). The leak is assumed to occur in the bottom of the vessel. Water is bled from the cold mass, and the hot mass flashes steam to fill the volume.

For the steam leak analysis, the rupture is assumed to occur in the steam dome. The reactor vessel is divided into a liquid volume and a steam volume. It is assumed that the coolant temperature can not drop below the saturation temperature of 240 F without causing inflow of containment water with a large soluble poison concentration. Therefore, the worst condition for criticality is at 240 F before the containment water enters the reactor vessel.

Results of Analysis

The results of these analyses are shown in Figure 8.1-1. For a water leak, the core is less than half filled with liquid when the coolant temperature drops to 495 F. The transient time to reach this condition is estimated at 1.5 seconds for a rupture equivalent to a 24-inch-diameter hole and 200 seconds for an equivalent 2-inch-diameter hole. After scramming, it is probably that the core will not go critical during blowdown for this case. The core may suffer a partial meltdown due to inadequate cooling. Should meltdown occur, the accident should not be more severe than the same accident for the NS Savannah.

For a steam leak, the core is still covered with liquid when the coolant temperature reaches 240 F, and the core may possibly go critical. The time required to reach this condition is estimated at 4 seconds for an equivalent 24-inch-diameter leak and about 600 seconds for an equivalent 2-inch-diameter leak. Because of reduced soluble poison concentration and the slightly larger temperature and Doppler deficits, end-of-life conditions represent the worst case. The δk associated with the very slow decay of Xe^{135} is not considered because it can not contribute to a nuclear burst, and at worst it can only contribute to partial fuel melting if the containment water does not enter the core in 15 to 20 hours following the accident. The estimated reactivity changes for a steam leak at the end of life are:

	δk
Control rod worth (cold)	−0.040
Doppler reactivity	+0.009
Moderator cooling, 500 to 240 F	+0.040
Void Collapse (conservative -- probably negative with flashing)	+0.003

	δk
Increase in soluble poison concentration from steam bleed and cooling, 0.2 gm/ℓ	−0.006
Net reactivity at 240 F after steam blowdown (not including Xe^{135} decay)	+0.006

If this reactivity is stepped into the reactor, a destructive excursion will not occur. Even if no credit is taken for the change in the soluble poison concentration, the maximum positive reactivity that can be inserted rapidly is 1.2% δk. In experiments at the SPERT reactor with NS Savannah fuel pins, step additions up to approximately 1.4 to 1.5% δk were made, and the Doppler coefficient limited the power excursion and prevented damage to the reactor. A more severe condition can not be visualized.

More detailed analysis may show that the wet containment scheme will provide better fission product containment than a dry containment. Further analysis may also show that the core will not melt, although no effort was made to determine the adequacy of the cooling in this investigation. The assumptions in this analysis are conservative, and the results show the pessimistic consequences of a primary system rupture.

8.1.4.2. Loss of Power to the Primary System Pumps

No detailed information on pump design and coastdown time was available when this analysis was made; therefore, the accident could not be completely analyzed. The CNSG primary system characteristics are very good for promoting natural circulation. Thus, the flow coastdown would be much less severe than in the Savannah. On this basis, it is believed that detailed calculations will show that this accident presents neither a hazard nor a serious operational problem.

8.1.5. Ship Accidents

The general subject of ship accidents involving the Savannah has been completely covered in the Final Safeguards Report, BAW-1164, Vol. II. In the event of a collision or a grounding accident, a ship containing a CNSG power plant will have the same structural integrity as the Savannah, and no intolerable hazard exists. The structural integrity probably will be greater since the CNSG containment vessel is approximately 16 feet smaller in diameter than that of the Savannah, thereby allowing an additional 8 feet of space for strengthening of the hull structure. An environmental analysis has been performed for the case of a ship that has sunk at sea.

No credible nuclear incident can cause the ship to sink. However, non-nuclear incidents such as collision, fire, or severe storm can cause a ship to sink in spite of any additional structural integrity built into the design. One major difference is evident when comparing the hazards involved from the sinking of the Savannah to those from the sinking of a CNSG. The Savannah will remain subcritical when the primary coolant water is replaced by sea water, but the CNSG may become supercritical if the borated primary coolant is replaced by salt water of lower cross section. The consequences and the environmental hazard resulting from the replacement of primary water with sea water in a CNSG that has sunk at sea have been analyzed.

For this analysis, the incident is assumed to cause the ship to sink, but damage to the primary system is not assumed. Also, it is assumed that prior to, or during the process of sinking, the reactor has been scrammed either by direct operator action or by one of the safety devices such as loss of electrical power.

Provisions for decay heat removal have been incorporated in the reactor vessel design. With the primary system at operating temperature, sufficient heat is conducted through the vessel wall to the containment water to remove the decay heat generated in the core. Immediately after the ship sunk, the reactor would be in a subcritical state. The events that could occur after sinking depend on the design of the reactor vessel, containment vessel, and ship as well as on the nature of the accident, the location of the sinking relative to the surrounding land area, and environmental conditions such as water depth and ocean currents. Due to the extremely complicated nature of the problem, the exact sequence of events leading to the maximum fission product release is difficult to predict. To overcome this problem, it was assumed that at the time of release to the ocean the fission product inventory in the CNSG core is comparable to that in the Savannah core following 600 days operation at 69 MW, as corrected for the CNSG power level of 63 MW.

Of the fission products in the core, 100% of the rare gases, 50% of the halogens, and 1% of the solids are assumed to be released to the sea water. This corresponds to the release assumed in the proposed AEC Site Guide Criteria. It should be noted that the fission product release in the Site Guide is based on a 100% core meltdown. In the case considered here, there is no reason to believe that gross core melting will occur, although some damage is possible. Therefore, the assumed fission product release in this preliminary analysis is quite conservative.

An accident in which the containment and the primary systems are ruptured when the ship sinks in a harbor is considered incredible. Furthermore, if the ship should sink, there will be sufficient time to carry out salvage operations to prevent the release of radioactivity.

If a sinking accident occuring at sea should result in a release of radioactivity, the environmental hazard would depend on the characteristics of the ocean area in which the accident occurred. The poorest diffusion characteristics, and consequently the worst related hazard, occurs near the shore. Therefore, an accident occurring in the ocean zone, which includes the area extending from 2 to 12 miles from the coastline, is analyzed. In the water areas beyond 12 miles, the mean concentration of fission products, and therefore the potential hazard, is reduced by a factor of 40 for fishing areas and by a factor of 200 for the open sea.

The appendix lists the mean concentration of isotopes in sea water following the accident. For all isotopes, the mean concentration is less than the maximum permissible concentration (MPC) listed for drinking water in 10 CRF 20.[2] Since the general public does not consume sea water, per se, the activity can be transmitted only by consuming seafoods that have been exposed to the radioactive environment. The analysis assumes that a person receives his entire protein requirement for a one-year period from fish that have lived their entire lifetime in the radioactive environment. In an actual case, a person would not receive his entire protein requirement from eating only radioactive fish; furthermore, the radioactive fish that he did eat would only be subjected to the radioactive environment for periods ranging from one day to one year. The activity transmitted to this person would be less than the calculated value. Even with these conservative assumptions, the total radioactivity ingested by a person from the radioactive fish is only 7% of the amount that would be ingested by drinking water containing the occupational MPC for one year.

The noble gases, which are primarily beta emitters and are not retained by marine organisms, will transmit no activity to the general public. Therefore, they are not considered. Wherever salvage is possible, no radioactivity, and therefore no gases, will be released. Assuming that salvage is not possible below 100 meters, the gases must diffuse vertically through this depth of water. Generally, the surface waters are less than 100 meters thick. Below this depth, stratification occurs and vertical mixing is very slow. Since most of the noble gas isotopes have very short half-lives, they will have essentially disappeared before reaching the surface (Kr^{85} is an exception, but its yield is less than 0.2%). There is also indication that the noble gases will be dissolved in the water[1], thereby minimizing vertical movement.

As a preliminary evaluation of a sinking accident, an instantaneous release of fission products to the sea from a completely melted core at the end of its life is very conservatively assumed. The resulting maximum radioactivity ingested by the public during the first year following the accident is less than 7% of that considered tolerable for radiation workers. Therefore, a credible sinking accident occurring in a CNSG-powered ship does not result in an undue hazard to the public.

8.1.6. Vapor Suppression Test Program

A test facility has been constructed at The Babcock and Wilcox Company's Research Center in Alliance, Ohio, to obtain preliminary information on the effectiveness of a water-filled containment in absorbing the energy released from a rupture in the primary system. Figure 8.1-2 is an over-all view of the facility, and Figure 8.1-3 is a schematic representation of the test apparatus. Besides obtaining information on condensation, the facility has instruments to obtain information on the magnitude and the duration of pressure pulses generated by the rupture, and also the magnitude of the force on the containment shell directly in the path of release from the rupture.

The facility was completed in November, 1961, and testing is in progress. Tests have been conducted with 1- and 3-inch sized ruptures at 520 F, and at pressures of 900 to 1900 psig. The peak pressure exerted on the containment tank was 8 psi for the 1-inch size and 24 psi for the 3-inch size. The duration of the pressure pulses is on the order of 20 milliseconds (ms). The pressure pulses exerted on the tank are detected by piezoelectric quartz crystals and are indicated on oscilloscopes set for single sweep. Polaroid photographs are taken of these traces for analysis.

The test program has not been completed, but the test results available to date show that the forces exerted on the containment tank are considerably less than those predicted by theory. The containment tank, formerly an oil storage tank designed for static head, is constructed with a 0.25-inch carbon steel plate. Due to the low design pressure, initial testing has been conducted with the top access opening uncovered. During the actual tests, no steam or water was observed to be expelled from the tank. This indicated that condensation was complete, and was true even when the containment water was lowered to within 2 feet above the rupture.

Future tests with larger sized ruptures are planned. Also, the dynamic force of the jet issuing from the rupture will be measured.

8.1.7. Maximum Credible Accident

A large rupture of the primary system would result in the greatest release of fission products from the core and the primary system. The sequence of events for this accident would be essentially the same as that of the Savannah. No credit, other than pressure reduction, is claimed for the presence of water within the containment.

Resulting doses to people near the ship are not excessive. As a maximum, all of the fission products released from the core can be assumed to pass through the containment water to the vapor space located above. If this were possible, the total activity released from the containment would be similar in magnitude to that for the Savannah. The maximum dose rate from a MCA on the Savannah was calculated to be 200 mr to the thyroid at a distance of 2000 meters from the source for a 24-hour exposure[7].

In the assumed MCA, a very pessimistic method of analysis was used. Evidence[1, 3] shows that fission products are retained in water to a considerable degree. In an experiment using tracer isotopes, 100% of the water soluble isotope Rb^{86}, 96% of the insoluble isotope Y^{90}, and 86% of the volatile isotope I^{131} remained in the water. These isotopes were released under 10 feet of synthetic sea water. Experiments under conditions simulating a pipe rupture are required to determine accurately the fission product retention of water for this application. Therefore, no credit was taken in this analysis of the MCA.

To allow the release of fission products from the containment, core meltdown or loss of cladding integrity must first occur. It is believed that the containment water will re-enter the reactor vessel immediately after blowdown has been completed (this is shown by the initial testing at the Alliance Test Facility). However, more testing and analysis is required to determine the rate at which the water re-enters. Also, it is believed that further testing and analysis will prove that gross core meltdown will not occur after a rupture in the primary system, and that the quantity of fission products released will be considerably less than that assumed in the analysis of the MCA.

The foregoing analyses have dealth with the most serious accidents to the primary system of the CNSG. Although these analyses have been preliminary, it is believed that detailed analyses will prove the assumptions made in obtaining results to be conservative. The CNSG should present no undue hazard to the general public.

8.2. Shielding

The following shielding studies, which did not affect the design of the reactor shielding proper, were carried out to assess plant operating feasibility and the ease of maintenance or access:

1. Shutdown shielding required above the core.
2. Dose rates and shielding requirements for air transfer of heat exchanger and vessel head.
3. Shielding required for fission product leakage from the core.
4. Shielding required for control rod coupling.
5. Shutdown shielding below the reactor vessel.
6. Instrumentation.

Shutdown shielding water requirements above the reactor core are shown in Figure 8.2-1 for a decay time of six hours after shutdown; the dose rate at the water surface is plotted as a function of water depth over the core. At this decay time, full-time access for fuel handling can be permitted with 13 feet of water over the core, and limited-time access with 9 to 10 feet of water.

Figure 8.2-2 shows the dose rate as a function of lead thickness at a distance of 10 feet from the shield surface during transfer of the heat exchanger surfaces and the reactor vessel head. The calculation of source activity on the heat exchanger surface is based on these assumptions:

1. 50 mg/dm^2 of corrosion product material is built up on all heat exchanger surfaces.

2. Corrosion product distribution is scaled from Task 26 NS Savannah Upgrading studies, and a total corrosion product activity of 4800 curies is used for the system.

3. Activation of heat exchanger stainless steel is based on there being 0.2 wt % cobalt in the stainless, and saturation activation of the 5.2-year Co^{60} activity.

4. No decay of corrosion products is taken into account.

As can be seen from these assumptions, particularly No. 3 and No. 4, the calculated dose rates plotted in Figure 8.2-2 are undoubtedly pessimistic. Despite this, the feasibility of performing the head removal operation is evident: Even when the heat exchanger surfaces are shielded by the "3-inch lead equivalent", and under the pessimistic assumptions made, the dose rates 50 feet from the shield will be in the range of from 15 to 25 mr/hr. Note that the "Total Dose Rate" curve of Figure 8.2-2 applies only to the bottom of the tube bundle, where activation of the metal is important.

Also studied was the effect of fission product leakage from the core in the event of clad defect or failure. The resultant dose rates were calculated by scaling the results of an NS Savannah computer calculation, which utilized Chalk River experimental UO_2 fission product release rates, to the volume power and flow conditions of the CNSG

This program calculates the buildup of individual isotopes in many chains, under continued leakage The following were assumed for the CNSG calculation:

1. Reactor operates for 300 days before leakage starts.

2. Leakage from core continues for 100 days, after which coolant fission product activity is considered during reactor operation with no shutdown decay.

3. No credit for plating or hideout of nuclides on surfaces inside the reactor vessel.

4. Five % of the fuel volume is exposed to coolant water as a result of pin-hole clad defects.

5. Primary coolant is transferred without dilution to the condenser tanks inside the containment.

The results are shown in Figures 8.2-3 and 8.2-4.

In Figure 8.2-3, the dose rates outside the containment vessel, including the effects of water shielding, are shown as a function of lead thickness. Water dose buildup was used in this calculation. Figure 8.2-4 shows the shielding requirements (lead thickness versus dose rates) for the removal of control rod couplings and lower portion of the control rods.

In the event of dry dock work such as hull-scraping and painting, the double-bottom water tanks below the reactor serve as a biological shield. Assuming 3 feet of water in these tanks, the dose rate at the bottom of the ship three days after shutdown was calculated to be 3.6 mr/hr. An additional foot of water in the tanks will reduce this dose rate to about 1 mr/hr if lower levels are required.

The thermal flux and the gamma dose rates were predicted at several positions to determine the location of nuclear instrumentation within the reactor vessel. Full power operating levels in the 2-inch water-filled space between the pressure vessel and the 8-inch layer of lead beyond the vessel were found to be 6.0×10^8 neutrons/cm^2-sec thermal flux, and 1.1×10^8 mr/hr total gamma dose rate. The corresponding levels near the outer surface of the 8-inch lead slab in water were 5.6×10^7 neutrons/cm^2-sec thermal flux, and 3.8×10^4 mr/hr total gamma dose rate.

8.3. Physics

8.3.1. Nuclear Calculation Methods

Basic nuclear core calculations were made by using methods developed during the design of the NS Savannah core, the Consolidated Edison core, and various recent power plant proposals. Critical experiments have verified many of these methods. Core lifetimes and various criticality calculations to obtain reactivity effects were done with proved one-dimensional codes. Two-dimensional studies were carried out to verify certain one-dimensional results and to solve directly void distribution effects.

8.3.1.1. Criticality and Lifetime Models

Criticality and lifetime calculations were made with the modified two-group diffusion model. The modification is made to account for the fission occurring in the epicadmium energy region. This model is described by

$$D_1 \nabla^2 \phi_1 - W_1 \phi_1 + \nu_C (f_1 \phi_1 + f_2 \phi_2) = 0 \tag{1}$$

$$D_2 \nabla^2 \phi_2 - W_2 \phi_2 + \beta_1 \phi_1 = 0 \tag{2}$$

where

W_1 = removal coefficient from the fast group

$= \Sigma_1^A + \beta_1 + D_1 B^2$.

Here,

Σ_1^A = absorption removal cross section for fast group

β_1 = transfer coefficient for fast group

$= P_T \Sigma_1$

P_T = total resonance escape probability

Σ_1 = slowing down cross section for fast group

$D_1 B^2$ = fast group leakage

f_1 = fast group source coefficient

$= P_{28} \Sigma_1 \epsilon \Lambda \sum_1 \eta_f^{res} (1-P_f)$.

P_{28} = U^{238} resonance escape probability

P_f = resonance escape probability of fissionable element (f)

Λ = resonance competition factor for resonance elements

$$= 1 - \pi P_E / \sum_E (1 - P_E) \quad E \pm 28$$

ϵ = fast fission factor

η_f^{res} = average number of neutrons produced per resonance neutron absorbed in fissionable element (f)

f_2 = thermal group source coefficient

$$= \epsilon \sum_f \eta_f^{th} \overset{A}{\underset{2f}{\Sigma}} ,$$

$\overset{A}{\underset{2f}{\Sigma}}$ = thermal absorption cross section of fissionable element (f)

η_f^{th} = average number of neutron produced per thermal neutron absorbed in fissionable element (f)

W_2 = removal coefficient from thermal group

$$= \overset{A}{\underset{2}{\Sigma}} + D_2 B^2 .$$

$\overset{A}{\underset{2}{\Sigma}}$ = absorption removal cross section for thermal group

D_2B^2 = thermal group leakage

ν_c = critical eigenvalue = $1/k_{eff}$.

Region constants used in solving the two-group diffusion equations were generated by a forty-group spectral diffusion calculation. The spectral code treats each reactor region as a bare homogeneous reactor. It was assumed that the isotope buildup and the fuel burnup would have a negligible effect on the diffusion properties of the core, and these changes would be felt basically in the thermal and in the resonance absorption properties. Therefore, moderator dependent coefficients and constants, such as the fast and thermal diffusion coefficients (D_1, & D_2) and the age (τ), were derived from the spectral code as a function of the moderator void fraction. Coefficients involving the absorption properties of the core were generated as a function of the moderator void fraction and the boron concentration in the core. These absorption coefficients were changed over core life to account for isotope buildup and fuel burnup.

The thermal and resonance contributions to various reactivity values and changes are computed as

$$k_{eff} = k_1 + k_2. \tag{3}$$

The resonance portion is

$$k_1 = \epsilon\ P_{28}\ \Lambda \sum_f (1 - P_f)\ \eta_f^{res} \Big/ 1 + \tau\ B^2. \tag{4}$$

Here, $\dfrac{1}{1 + \tau\ B^2}$ = fast group non-leakage probability

τ = age to thermal energy, cm^2

B^2 = total buckling, cm^2.

The thermal portion is

$$k_2 = \epsilon\ P_T\ \bar{f}\ \eta_f^{th} \Big/ (1 + \tau\ B^2)\ (1 + L^2\ B^2). \tag{5}$$

Here, $\dfrac{1}{(1 + \tau\ B^2)\ (1 + L^2\ B^2)}$ = total non-leakage probability

L^2 = thermal diffusion length, cm^2

$\bar{f}$ = thermal utilization factor.

The thermal utilization factor is

$$\bar{f} = \sum_f \Sigma_{2f}^A \Big/ \sum_f \Sigma_{2f}^A + \sum_E \Sigma_{2E}^A\ \frac{\phi E}{\phi f}\quad E \neq f. \tag{6}$$

Here, f = all fissionable elements

E = all non-fissionable elements

$\Sigma^A_{2f}\Sigma^A_{2E}$ = macroscopic absorption cross sections calculated from homogenized number densities and microscopic absorption cross sections which have been corrected for temperature, neutron energy distribution and have been non-1/V conditioned when applicable

$\frac{\bar{\phi}E}{\bar{\phi}f}$ = advantage factors for non-fissionable element obtained by ratioing the average flux values from thermal flux profiles generated by cylindrical P_3 pin cell calculations.

The resonance escape probability is defined for all elements as

$$P_E = \text{esp.} -\left(\frac{N_E R_E}{\xi \Sigma_s}\right) \qquad (7)$$

where

N_E = homogenized number density of element (E)

R_E = resonance integral of element (E)

$\xi \Sigma_s$ = resonance slowing down power (cm^{-1}) of the medium.

The total resonance escape probability is then defined as

$$P_{total} = \pi P_E \, . \qquad (8)$$

The resonance integrals (R_E) for all core elements except U^{238} were taken as the infinitely dilute values. The effective resonance integral for U^{238} was calculated by an empirical equation for single pin values and was corrected for lattice, temperature, and spectrum effects in the following manner. The single pin resonance integral was calculated by

$$R^{\text{Single Pin}}_{\text{Total}} = A + B\sqrt{S/M} \qquad (9)$$

where

A = 5.25

B = 26.60

S/M = surface to mass ratio of the oxide

This equation, based on measurements performed by Eric Hellstrand, was modified to include the 1/V portion of R_{28}.[1]

The total single pin value is divided into a surface term and an effective volume term from the data published by L. W. Nordheim.[2] A curve fit to the results of Nordheim gives a polynomial equation in terms of the pin radius:

$$K_s = h_0 + h_1 r + h_2 r + - - - - h_n r \tag{10}$$

where

K_s = surface fraction of the single pin integral

r = pin radius

h_1 = polynomial coefficients.

Consequently, the surface and volume terms are

$$R_S^{\text{Single Pin}} = K_s\, R_{\text{total}}^{\text{Single Pin}} \tag{11}$$

$$R_V^{\text{Single Pin}} = (1 - K_s)\, R_{\text{total}}^{\text{Single Pin}} \tag{12}$$

A Dancoff correction (γ) was applied to the surface term to account for the self shielding effect of the closely packed fuel pins. This correction factor was based on the methods of Dancoff and Ginsburg.[3]

The lattice value of the resonance integral is then defined as

$$R_{\text{total}}^{\text{lattice}} = R_V^{\text{Single Pin}} + \gamma\, R_s^{\text{Single Pin}} \tag{13}$$

where

$$\gamma = 1 + \frac{V_0}{2r\, V_m\, \Sigma_m^{res}} \tag{14}$$

Here,

V_0 = oxide volume fraction

V_m = volume fraction of everything in the pin cell except oxide

r = oxide radium,

Σ_m^{res} = resonance scattering macroscopic cross section of everything in the pin cell except the oxide.

The increased resonance integral value resulting from the broadening of the U^{238} resonance peaks at operating temperatures is accounted for in the Doppler resonance coefficient as applied to the results of Equation (13).

$$R^{\text{Lattice}}_{\text{Total at Power}} = \left[R^{\text{Single Pin}}_{V} + \gamma\, R^{\text{Single Pin}}_{s}\right]\left[1 + a\,(T_2 - T_1)\right] \qquad (15)$$

where

T_2 = oxide average temperature at power, F

T_1 = ambient temperature, F

a = Doppler coefficient (a value of 0.6×10^{-4} per °F was used).

An initial attempt to correct for the resonance competition between the U^{235} and U^{238} in the fuel pin was made with the forty-group spectral code. The resonance integral (R_{28}) was calculated from Equation (15) as a function of the moderator void fraction. These values were used in the spectral calculations to generate diffusion coefficients. In addition to varying the void fraction, the U^{235} content was also made a variable. This resulted in a family of curves for D_1, D_2, τ, and P_{28} as a function of the void fraction and the U^{235} content. The diffusion coefficients and τ were insensitive to the U^{235} content, but P_{28} was sensitive to both variables. The P_{28} values, as calculated by the spectral code, reflect the spectrum effects on the resonance absorption in U^{238} and the resonance competition with U^{235}. Redefining the resonance escape probability of the spectral codes by Equation (7) and solving for R_{28} produced an effective resonance integral corrected for resonance competition. A family of curves were then calculated for R_{28}^{eff} as a function of the void fraction and the U^{235} content.

A given U^{235} core loading dictates the set of effective resonance integrals (as a function of the void fraction) to be used in calculating core criticalities and lifetimes. Void fraction dependent coefficients D_1, D_2, τ, $\xi\Sigma_s$, and R_{28}^{eff} used are shown in Figures 8.3-1, 8.3-2, 8.3-3, 8.3-4, and 8.3-5. Reactor element absorption cross sections and resonance integral are given in Table 8.3-1.

Table 8.3-1. Absorption Cross Sections and Resonance Integrals

Element (E)	τ/a (e), barns	R_E, barns
U^{238}	1.76	variable with void fraction
U^{236}	3.94	220
U^{235}	422	445
Pu^{239}	1017	588
Pu^{240}	164	7400
Pu^{241}	908	1600
Pu^{242}	42	-
Xe^{135}	2.229×10^6	-
Sm^{149}	5.597×10^4	-
Fission 25	109.3	630
Products 49	94.7	469
H_2O	0.450	0.17
Zr-2	0.165	0.17
B_{10}	2810	1918

All listed cross sections have been corrected for temperature neutron energy distribution, flux depression, and non-1/V distribution when applicable.

The two-group model was used to determine core criticality for various reactor conditions. In conjunction with a isotope deletion routine at normalized power conditions, the same model was used to calculate the reactor lifetime. Criticality was maintained during the lifetime by changing the soluble poison concentration.

8.3.1.2. Two-Dimensional Study

Two-dimensional studies were performed to determine the void and the fuel zoning effects on core reactivity. These studies were made with the PDQ code[4] in R-Z geometry. The PDQ code is a two-dimensional criticality code that solves the few-group, neutron diffusion equations for one to four lethargy groups in either rectangular or cylindrical coordinates. Two-dimensional calculations performed on this core used four lethargy groups as described by the following equations:

$$D_1 \nabla^2 \phi_1 - (\Sigma_{a1} + \Sigma_1 + D_1 B_z^2) \phi_1 + \frac{1}{\lambda} \sum_{i=1}^{4} V_1 \Sigma_{fi} \phi_1 = 0 \qquad (16)$$

$$D_2 \nabla^2 \phi_2 - (\Sigma_{a2} + \Sigma_2 + D_2 B_z^2) \phi_2 + \Sigma_1 \phi_1 = 0 \qquad (17)$$

$$D_3 \nabla^2 \phi_3 - (\Sigma_{a3} + \Sigma_3 + D_3 B_z^2) \phi_3 + \Sigma_2 \phi_2 \doteq 0 \qquad (18)$$

$$D_4 \nabla^2 \phi_4 - (\Sigma_{a4} + \Sigma_4 + D_4 B_z^2) \phi_4 + \Sigma_3 \phi_3 = 0 \qquad (19)$$

where

D_i = diffusion coefficient of group (i)

ϕ_1 = flux level of group (i)

Σ_{ai} = macroscopic absorption cross section for group (i)

Σ_i = macroscopic removal cross section for group (i)

$D_i B_z^2$ = leakage from group (i)

Σ_{fi} = macroscopic fission cross sections of group (i)

V_i = average neutron produced per fission in group (i).

Various coefficients used in the two-dimensional calculation were generated by the forty-group spectral code. This code is designed to generate forty group weighted coefficients for each of the desired lethargy groups.

8.3.1.3. Operational Analysis Data

The operational analysis of this reactor necessitated the calculation of various reactivity coefficients, effective neutron delay fractions, effective decay constants. The Doppler reactivity coefficients

$$\left(\alpha_D = \frac{1}{K} \frac{\partial K}{\partial T_f} \right)$$

and the moderator temperature coefficient

$$\left(\alpha_M = \frac{1}{K}\frac{\partial K}{\partial T_m}\right)$$

were calculated at the beginning and end of core life. Void coefficients

$$\left(\alpha_V = \frac{1}{K}\frac{\partial K}{\partial \rho_m}\right)$$

were calculated from two-dimensional cases at the beginning of core life. Reactor criticality calculations were made for slight changes in the U^{238} resonance integral and in the moderator temperature to obtain these coefficients from the following equations:

1. Doppler Reactivity Coefficient

$$\alpha_D = \frac{1}{K}\frac{\partial K}{\partial T_f} = \frac{K_0 - K_1}{K_0 - T_f} \tag{20}$$

where

K_0 = initial core reactivity

K_1 = core reactivity resulting from an integral (R_{28}) change

δT_f = effective fuel temperature change reflected by the R_{28} change.

2. Moderator Temperature Coefficient

$$\alpha_M = \frac{1}{K}\frac{\partial K}{\partial T_m} = \frac{K_0 - K_1}{K_0 - T_m} \tag{21}$$

where

K_0 = initial core reactivity

K_1 = core reactivity resulting from a change in the moderator temperature

δT_m = change in moderator temperature.

3. Void Coefficient (two-dimensional calculations)

$$\alpha_V = \frac{1}{K}\frac{\partial K}{\partial 1CV} = \frac{K_0 - K_1}{K_0 - \delta 1CV} \tag{22}$$

where

K_0 = initial core reactivity with no void in the system

K_1 = core reactivity with a specific void distribution

δICV = change in integrated core void.

4. Effective Delay Fraction and Decay Constant

Effective delay group fraction (β_{eff}) and decay constants (λ_{eff}) were calculated for the beginning and end of core life. All data was taken from the radial lifetime calculations. The ${}_j\beta_{eff}$ and ${}_j\lambda_{eff}$ values for six delay groups were source weighted at time zero and 485 days.

$$ {}_j\beta_{eff} = \sum_i \beta_i W_i \tag{23} $$

$$ {}_j\lambda_{eff} = \sum_i \lambda_i W_i \tag{24} $$

where

i = elements U^{235}, U^{238}, Pu^{239}

j = delay groups 1 through 6

${}_j\beta_i$ = delay fraction of element i and delay group j

${}_j\lambda_i$ = decay constant of element i and delay group j

W_i = source weighting function defined as

$$ W_i = \frac{\nu_i F_i}{\sum_i \nu_i F_i} \tag{25} $$

Here,

F_i = fission rate of element i

ν_i = average neutron per fission in element i.

An additional correction was placed on the effective delay fraction to account for the relative importance of the delay versus the fission neutron. Since the delay neutron has a lower initial energy than the fission neutron, the probability of causing a fission is greater, and its relative importance is higher. The effective delay fraction is then defined as

$$_j\overline{\beta}_{eff} = \frac{qd}{qf}\left({}_j\beta_{eff}\right). \tag{26}$$

Here, qd/qf is the ratio of the slowing down densities at the thermal energy for delayed and fission neutrons. The quantity qd/qf was calculated by the forty-group spectral code. It was assumed that all of the delayed neutrons were emitted in the same energy band (0.498 Mev — 0.388 Mev). The ratio qd/qf had a value of 1.025 at time zero and at the end of life. The effective neutron lifetime[5] was approximated by the equation

$$\ell^* \approx \frac{\rho}{V\Sigma_a} \tag{27}$$

where,

ℓ^* = neutron lifetime

ρ = core reactivity

V = neutron velocity

Σ_a = thermal absorption cross section.

If a change in the 1/V absorption cross section of the core equal to one over the neutron velocity is made, then the resulting change in the core reactivity approximates the neutron lifetime of the system.

8.3.2. Supplementary Nuclear Study

A second core design, based on the same general layout as the reference core but with smaller pins, was investigated under two different control schemes. The secondary design maintained the same metal-to-water ratio (0.68) as the original design with 0.40-inch OD pins at a pitch of 0.598 inches. The Zr-2 cladding was reduced to a thickness of 0.0185 inches. Individual fuel elements consist of the same number of pins and the same basic hexagonal shape. The resulting core has an active fuel height of 42 inches and an equivalent diameter of 42.6 inches.

Fuel loading was broken down into the same three basic radial zones as were previously employed. Zone shuffling methods were also of the type described in the basic design section.

Two methods of core control were used. The first approach was to use the same soluble poison system as was proposed for the reference core. Boric acid was used as the poison. This core will be referred to as CNSG-A in this report. The second control method used was the spectral shift control system. This system consists of a moderator mixture of light (H_2O) and heavy (D_2O) water where the relative concentration of D_2O to H_2O is varied over the core life to maintain core criticality. The lack of moderating power at the higher D_2O concentration provides the necessary reactivity holddown while the effective conversion ratio of the reactor system is increased. A 4-inch thick natural uranium blanket was placed around the basic core to improve neutron economy. The blanket is replaced and the old one is reprocessed at the end of every second cycle. The spectral shift version will be referred to as the CNSG-B core.

Both cores were calculated to produce 63 MWt for 400 days. Based on a power-to-average power factor of 2.75 to 1, the fuel received approximately 16,400 MW day per ton average irradiation during a complete tenure in the core.

The resulting lifetime parameters are compared to the reference design in Table 8.3-1, and fuel costs for the equilibrium cycle are also given. Fuel costs for the equilibrium SSCR recycle core are given under column CNSG-B' in Table 8.3-2. In the case of recycling, which means refabrication of the bred fuel, the conversion ratio will approach 0.83, and hence the fissionable material cost becomes quite low.

Table 8.3-2. CNSG Parameter Comparison Equilibrium Cycles

	CNSG	CNSG-A	CNSG-B	CNSG-B'
Metal to water ratio	0.68	0.68	0.68	
Pin pitch, in.	0.6875	0.5980	0.5980	
Pin OD, in.	0.460	0.400	0.400	
Clad thickness, in.	0.0210	0.0185	0.0185	
Active fuel height, in.	42	42	42	
Equivalent core diameter, in.	49.5	42.6	42.6	
Blanket thickness, in.	-	-	4	
Method of control	H_3BO_3	H_3BO_3	D_2O	
Initial poison concentration gm H_3BO_3/kg H_2O	6.9	6.3	-	
Initial D_2O concentration, %	-	-	60	
Integrated conversion ratio	0.54	0.53	0.66	
Total UO_2 loading, kg	4079	3074	3074	
Initial U^{235} loading, kg	96	79	67	
MWd/ton/cycle	7500	8200	8200	
Equilibrium cycle fuel cost				
Fabrication	0.65	0.73	0.87	0.72
Fissionable material cost	0.88	0.87	0.61	0.29
Inventory	0.22	0.19	0.15	0.15
Reprocessing and reconversion	0.48	0.50	0.59	0.52
Transportation and insurance	0.07	0.06	0.09	0.09
Total fuel cost	2.30	2.35	2.31	1.77

8.4. Selection of Soluble Poison

8.4.1. Abstract

Covered here is the selection of a suitable soluble poison for lifetime control of the CNSG. Although it is difficult to conclude that any one poison is the most suitable, it can be stated that of those tested and described in the literature, a boron-containing compound probably is the most desirable.

To settle conclusively the issue of selecting a soluble poison, research and development is necessary. A program for research and development utilizing the NS Savannah loop at ORNL has been suggested and is discussed here.

8.4.2. Introduction

Selection of a soluble poison for lifetime control of a reactor plant is difficult for the following reasons.

Lack of Information and Security Restrictions

The literature reviewed gives no information on the long-time use of boric acid in a reactor plant. It is understood that considerable data have been gathered under the Naval Reactor Program at the Bettis Atomic Power Laboratory. It is also known that the Yankee Atomic Electric Company plant at Rowe, Massachusetts has used boric acid for control of the reactor. Attempts are being made to get information from these two facilities.

Use of Varied Materials

The possible use of carbon steel as a material of construction in the CNSG complicates the selection of a soluble poison. This is particularly true with boric acid, which is entirely incompatible with carbon steel. Most writers conclude that boric acid is a good soluble poison, but not when in contact with carbon steel.

Classified Reports

Since certain reports on the subject are classified, the information therein cannot be discussed.

8.4.3. Discussion of Experience With Soluble Poisons

8.4.3.1. Statement of Problems

Cohen[1] lists a number of problems to be considered when soluble poison is used in a reactor:

1. The poison must be economical and readily available.

2. Commercial grade purity should be such that further treatment is unnecessary.

3. The macroscopic cross section should be adequate in the entire range from freezing point to critical temperature. Also, cold water solubility should be such that small volume is required for subcritical shutdown. The microscopic cross section should be high enough to permit the most concentrated solution required to be dilute (not more than 5%).

4. Properties of the soluble poison must not give heat transfer problems.

5. The poison must be stable at high temperatures; no decomposition, deposition, or crystallization.

6. It must be stable in a radiation field and must not increase water radiolysis.

7. It must not increase shielding requirements.

8. It must not be more corrosive than H_2O.

9. It must not cause more wear than would occur with pure H_2O.

10. It must not interfere with normal operation of the purification system.

11. The rate and amount of crud deposition must be kept to a minimum.

12. The poison must not decrease the tolerances of waste disposal effluents.

13. Concentration of poison must lend itself to rapid, accurate measurement.

8.4.3.2. Properties of Various Poisons

Shapiro[2] outlines some of the problems associated with the use of chemical poisons in reactors. Gaseous and liquid compounds of boron are discussed, and emulsions, soluble solids, and slurries are considered. Shapiro concludes that soluble solids seem most suitable. However, when dissolved in water most soluble solids hydrolyze to form acid solutions, which may be undesirable. A discussion of each of these groups follows.

Chemical Properties of Gaseous Compounds of Boron

The chemical properties of gaseous compounds of boron are given below.

Chemical formula	Melting point, C	Boiling point, C	Remarks
B_2H_5Br	- 104	+ 10	Hydrolyzes: HBO_2 + HBR + H_2
BCL_3	- 107	+ 12	Hydrolyzes: HBO_2 + HCL
B_2H_5CL	- ?	- 78 (18 mm)	Highly unstable
BF_3	- 127	- 101	Hydrolyzes (hot): HBO_2 + HF
B_2H_6	- 165	- 92	Hydrolyzes: HBO_2 + H_2 (thermally unstable)
B_4H_{10}	- 120	+ 18	Hydrolyzes: HBO_2 + H_2 (thermally unstable)

From information given in the table, it can be concluded that none of the poisons listed are suitable for use in the coolant. In an auxiliary system, the halide gases may be applicable if a compatible combination of solvent, structural material, and gas can be found.

Liquid Nuclear Poisons

Liquid nuclear poisons were studied to determine their applicability as soluble poisons. Chemical properties of these compounds are given in the following tabulation.

Chemical formula	Melting point, C	Boiling point, C	Remarks
BBr_3	- 46	90 (740 mm)	Hydrolyzes: $HBO_2 + HB_r$
B_5H_9	- 47	0 (66 mm)	Hydrolyzes: $HBO_2 + H_2$ (spontaneously flammable in air)
B_5H_{11}	- 123	65	Hydrolyzes: $HBO_2 + H_2$
B_5H_{10}	- 65	0 (7.2 mm)	Hydrolyzes: $HBO_2 + H_2$
B_2H_5I	- 110	0 (78 mm)	Hydrolyzes: $HBO_2 + HI + H_2$
Hg	- 39	360	Insoluble in water
ReO_3Cl	4.5	131	Probably hydrolyzes to acid solution.

These data indicate that none of these compounds is suitable for use in primary water.

Emulsions

Mercury, which is insoluble in water, may be considered for use as an emulsion, but more work is needed to prove its applicability.

Slurries

The main problems associated with slurries are abrasion and erosion, which eliminates them from further consideration.

Soluble Solids

Of the compounds in this group, boron in some soluble, solid form seems most suitable. Sulfates of rare earths and cadmium, indium and hafnium would probably hydrolyze and form acid solutions. Deposition of the noble metals (since they are easily reduced) is likely. However, recent work done at AECL[3] indicates that cadmium sulfate is a suitable soluble poison and that its use in the reactor coolant system would create few if any adverse effects. In comparing cadmium sulfate with boric acid, AECL has concluded that cadmium sulfate is better for the reasons given below.

Cadmium does not produce alpha particles, as does boron [B^{10} (n, α) Li^7], and therefore does not increase radiolysis of the water through particle generation.

Cadmium is more easily removed from the coolant by ion exchange. A disadvantage of cadmium sulfate is that it does deposit on system surfaces to some extent. At pH of 6.5, AECL found a cadmium thickness of 0.1 $\mu g/cm^2$. As the pH rose from 6.5, the deposition increased, reaching 3.5 $\mu g/cm^2$ at pH 7.5. These cadmium deposits were difficult to remove, so AECL, in an effort to minimize deposition, tried sulfuric acid, ammonia, and cyanides. There was still some deposition, but to a lesser extent.

In spite of the fact that cadmium does deposit in reactor systems, the Canadians do not consider it serious enough to prevent the use of cadmium for shim control in the CANDU moderator. When cadmium is to be used, the moderator will be charged with 10^{-6} moles per liter of excess acid to minimize cadmium deposition. It should be noted that the poison will be used solely for shim control and will be removed from the coolant shortly after plant startup. This work, then, will not yield conclusive information for use in designing the CNSG for lifetime control with soluble poison.

8.4.3.3. Radiation Stability of Soluble Poisons

A study at Westinghouse[4] investigated the effects of irradiation on potassium tetraborate, ammonium pentaborate, and boric acid solutions. Fast electron radiation decomposed potassium tetraborate solutions to form hydrogen, oxygen, and hydrogen peroxide, but pre-irradiation hydrogen additions virtually eliminated gas production up to a total dose of 2.7×10^8 rads. The ammonium pentaborate solutions, even with added hydrogen, proved to be unstable to radiation, and produced large amounts of hydrogen, oxygen, hydrogen peroxide, and ammonia; the boric acid, however, proved quite stable and yielded only small amounts of hydrogen, up to 6.8×10^7 rads. The following conclusions were reached in this work.

Ammonium Pentaborate

After irradiation of ammonium pentaborate solutions containing 3.9 grams boron per liter and a concentration of hydrogen of 17 to 25 cc hydrogen per liter of water, 6.5 millimolar hydrogen and 3.2 millimolar nitrogen showed up as decomposition products. The decomposition had not reached steady state values after a total dose of 6×10^7 rads. It was therefore concluded that added hydrogen does not stop the decomposition of ammonium pentaborate solutions.

Boric Acid

Irradiation of boric acid solutions in the concentration of 3.9 grams boron per liter produced a concentration of hydrogen of 0.16 millimolar at a dose of 6.8×10^7 rads. No oxygen or hydrogen peroxide was produced.

Boric acid was further tested[5] at ANL. The behavior of solutions of boric acid in inpile irradiation was investigated as a function of boric acid concentration. The study included the effect of hydrogen, hydrogen peroxide, and potassium iodide on radiation of boric acid solutions. Figure 8. 4-1 shows the concentrations of products formed in solutions of 0.1 molar boric acid. It was concluded that solutions of up to 0.02 molar boric acid may be irradiated without decomposition. At higher concentrations of boric acid, decomposition of water occurs at a rate of 1.83 millimolar hydrogen and hydrogen peroxide per minute per boric acid. The gamma flux produces mainly hydrogen and hydroxyl radicals. This same flux recombines the hydrogen and hydrogen peroxide, which are formed as a result of the $[B^{10}\ (n, \alpha)\ Li^7\ ]$ reaction. Figure 8. 4-2 shows the effect of initially dissolved hydrogen on hydrogen production in boric acid solutions under irradiation. Figure 8. 4-3 gives the effect of boric acid concentration on hydrogen production in air-free solutions under irradiation.

Potassium Tetraborate Solutions

These solutions, containing 3.9 grams boron per liter, were irradiated with fast electrons over the range of 6×10^4 to 7×10^7 rads to produce hydrogen peroxide and oxygen. Each product reached a steady state value after a total dose of about 5×10^7 rads. The steady state values were about 4 millimolar hydrogen, 1.2 millimolar hydrogen peroxide, and 1 millimolar oxygen.

In another test with potassium tetraborate solutions, 25 to 30 cc hydrogen per liter of water was added prior to irradiation. The solutions were then exposed to total doses up to 2.7×10^8 rads. Essentially no decomposition products were formed. It was then concluded from this work that potassium tetraborate, with hydrogen in the concentration given above, was the most suitable borate compound for use as an emergency chemical shutdown poison.

8.4.3.4. Thermal Stability of Soluble Poisons

A loop test[6] was conducted to determine the thermal stability of boric acid at 600 F and 2000 psig. The loop water contained 100 cc hydrogen per liter, and boric acid was added in increments until a concentration of 0.21 moles boric acid per liter was attained. No deposition was observed in the loop at the conclusion of the test.

8.4.3.5. Solubility

The complete solubility of a soluble poison is necessary in order to minimize deposition and to prevent heat transfer and pressure drop problems. A further problem may arise if borate poisons deposit on the fuel cladding, since an increase in water radiolysis may result from the alpha flux produced. The particle flux may also disturb the corrosion film and lead to increased corrosion. Boron-containing compounds, being considered as poisons, possess a positive temperature coefficient of solubility. Figures 8.4-4 through 8.4-7 show solubility versus temperature for boric acid,[7] potassium tetraborate,[8] ammonium pentaborate,[8] and sodium tetraborate[8]. Cadmium sulfate has a negative temperature coefficient of solubility above 165 F, which limits its use as a soluble poison unless suitable complexes can be found to increase its solubility. The solubility[8] of cadmium sulfate versus temperature is shown by Figure 8.4-8.

8.4.3.6. Corrosion

Static beaker tests[9] were conducted with specimens of AISI 304 stainless steel and AISI 4135 carbon steel in saturated solutions of boric acid. Samples of these materials were exposed in boric acid solutions of 5 wt % (70 F) and 13 wt % (140 F), both of which are approximately saturation conditions. After a four-week exposure the stainless steel specimens had not been attacked, but the carbon steel had developed considerable corrosion product scale. The attack on carbon steel at 70 F in 5 wt % boric acid was greatly reduced by adding base to neutralize the solution. A 13 wt % boric acid solution severely attacked the carbon steel at 140 F, even when adjusted to neutral pH. Figure 8.4-9 shows the pH of solutions of boric acid at 20 C. The results of carbon steel and stainless steel tested during acidic exposure are given in the table below.

Table 8.4-1. Weight Changes of Carbon Steel and Stainless Steel During Acidic Exposure

Material	Specimen no.	Cumulative weight change, mg/dm^2 2 weeks	4 weeks
Corrosion in Saturated Boric Acid Solution at 70 F			
SS AISI 304	1	- 0.7	0
	2	- 0.4	+ 1.1
	5	- 0.4	+ 0.7
	6	- 0.4	+ 0.7
	9	(- 6.6)	(- 6.9)
	11	0	− 0.3
	13	- 1.6	- 1.1
	15	0	+ 0.5
CS AISI 4135	17A	- 616	- 1032
	19A	- 614	- 983
	21A	- 532	- 860
	22A	- 560	- 903
	25A	- 414	- 623
	26A	-	- 636
Corrosion in Saturated Boric Acid Solution at 140 F			
SS AISI 304	3	0	0.4
	4	0	0.4
	7	0	0
	8	0	- 0.3
	10	- 12.9	-19.6
	12	- 9.7	-13.9
	14	- 2.6	- 3.2
	16	0.3	- 0.3
CS AISI 4135	18A	- 3680	-6370
	20A	- 4290	-7890
	23A	- 3350	-7000
	24A	- 4470	-7000
	27A	- 2180	-3890
	28A	- 2300	-4240

The weight changes of carbon steel after acidic exposure to neutral pH solutions of boric acid are presented below. (Corrosion occurred when carbon steel was exposed to saturated boric acid solutions with pH control.)

Table 8.4-2. Weight Changes of Carbon Steel After Acidic Exposure

Specimen no.		Cumulative weight change, mg/dm^2 2 weeks	4 weeks	8 weeks
17B	70 F	- 0.9	---	- 2.5
19B	70 F	- 0.2	---	- 1.6
21B	70 F	- 3.1	---	- 2.5
22B	70 F	- 1.1	---	- 5.5
25B	70 F	- 1.8	---	10.1
26B	70 F	- 0.8	---	9.4
18B	140 F	- 5.2	- 118.5	- 2432
20B	140 F	- 9.8	- 124.8	- 1180
23B	140 F	- 0.4	- 116.0	- 1313
24B	140 F	- 0.6	- 133.6	- 1297
27B	140 F	- 0.8	- 36.6	- 1025
28B	140 F	- 0.7	- 7.5	- 1157

Another test[10] was conducted to determine the corrosion resistance of various materials in boric acid solutions. The materials were tested in a 1% boric acid solution at 600 F. The following information, taken from the report, covers the results obtained for each material.

Material	Discussion
304 austenitic stainless steel	Showed excellent resistance to general and pitting corrosion as well as to stress corrosion cracking in dynamic, semi-static, and static tests. No difference in corrosion resistance of this material when subjected to water phase, steam phase, or steam-water interface.
Modified 304 austenitic stainless steel (contained 300, 500, 12,000 ppm boron)	Gave excellent corrosion resistance in dynamic loop.

Material	Discussion
Anodized and unanodized aluminum (type 6061 and 6063)	Fair to excellent corrosion resistance. (9 to 57 mg/dm^2-mo rate)
SA 212 carbon steel	Poor corrosion resistance. (2450 to 3140 mg/dm^2-mo rate)
Brass (85% Cu, 15% Zn) and 90 - 10 copper-nickel alloy)	Excellent corrosion resistance except when specimens were half-submerged — then subject to general and pitting corrosion at water line.
Stellite 6 K and Stellite 12	Both stellites exhibited excellent corrosion resistance. Surface finish significantly affects the corrosion rate of Stellite 12. A rough ground surface corrodes about twice as fast as a 125-microinch finish.
Hafnium	Excellent corrosion resistance. No significant difference in corrosion rate of specimens made with crystal bar or sponge hafnium.
17 - 4 PH alloy	Excellent resistance to general corrosion.
Croloy 16 - 1, type 17 - 7 PH, Zr - 2, Inconel X and Haynes 25 alloys	All showed excellent corrosion resistance.
Silver alloy (80% Ag, 15% Cd, 5% In)	Relatively high corrosion rate. (100 mg/dm^2-mo)

8.4.4. Research and Development Proposal

Since little work has been done on the long-time use of soluble poisons in an inpile loop or reactor system, a research and development program must be conducted before a soluble poison can be selected for lifetime control of the CNSG. Attempts are being made to obtain relative information now being developed by the Westinghouse Atomic Power Department and the Yankee Atomic Electric Company. This information will be used in setting up the research and development proposed in the following pages.

The NS Savannah loop in the Oak Ridge Reactor has been suggested for this work. In addition, screening corrosion tests of various materials in potassium tetraborate and ammonium pentaborate solutions will be conducted in static autoclaves to determine the corrosion behavior of materials in the foregoing soluble poison environments prior to the entry of specimens into an inpile loop. Boric acid will not be used in this phase of testing since sufficient information is available[10] on the corrosion of materials in boric acid solutions in autoclave tests.

8.4.4.1. Program Objectives

The main objective is to select a soluble poison for lifetime control of the CNSG. Additional objectives are

the determination of chemical stability and radiation effects of soluble poisons in an inpile reactor loop, and

the determination of materials compatibility with soluble poisons.

8.4.4.2. Ground Rules

1. The soluble poison selected will be used for lifetime control of the CNSG.

2. The major materials contacting the primary fluid are stainless steel, carbon steel, and Zircaloy.

3. There will be minimum control of water chemistry. (No bypass purification of primary coolant.)

4. The reactor plant will operate with flooded containment.

8.4.4.3. Information to be Obtained from Program

1. The test loop will be run first with no addition of soluble poison in order to determine reference equilibrium conditions. Information developed with the use of soluble poisons later in the program will then be based on a reference set of conditions.

2. For each soluble poison tested, data on corrosion rates and film thicknesses will be recorded for all materials exposed, inpile and outpile. The thickness of soluble poison buildup will be determined for each material tested both inpile and outpile. The distribution of corrosion products between coolant and system surfaces will also be investigated.

3. Information will be gathered on the increase in radiolysis of water due to the presence of soluble poison.

4. Information will be obtained on the distribution of soluble poison between coolant and vapor space.

8.4.4.4. General Description of Test Program

Reference Loop Test — No Soluble Poison

The loop will be run first to determine equilibrium conditions which will be used as a reference for subsequent tests with soluble poisons in the coolant. The loop will be initially filled with deionized and degassed water, and no chemical additions will be made to the loop water during the test. The demineralizer will be bypassed in all phases of the test program. Operation of a primary coolant system without a bypass purification system has been suggested under Task 26 of the NS Savannah Upgrading Program.

This operation is not expected to lead to significantly higher activities and dose rates in the coolant, as has been shown in a modified purification system performance test[11] at the Shippingport Atomic Power Station. The plant was operated for 1475 EFPH without side-stream demineralization. During this operation reference water specifications for reactor coolant were maintained within the prescribed limits. The results indicate no discernible increase of crud deposition in the reactor coolant system during operation without purification. Further, the results indicate no significant increase in radiation levels from operation without purification.

Coupons of materials to be used in the CNSG will be placed in inpile and in outpile sections of the loop. The loop will be operated until equilibrium conditions are reached for corrosion film thickness, pH, dissolved and undissolved solids, and radiolytic gases (H_2 + O_2). Analyses for these variables will be made at appropriate intervals throughout the test to determine the time required to reach equilibrium and to provide data for plotting curves. When the data indicate that equilibrium conditions have been attained, the specimens of material will be removed from the loop and analyzed for activity pickup.

Since the activity of corrosion and fission products in the water, the volume of the loop, and the area of the specimens will then be known, the ratio of activity of the specimens to that of the coolant can be obtained. When this is completed, the specimens will be weighed to determine weight gain. The loop will then be flushed, and specimens of all materials added in preparation for the first run with soluble poison.

First Soluble Poison Test — Use of Boric Acid

The loop will be filled with deionized, degassed water and charged with a boric acid concentration of 10 grams per liter — the reference concentration for the CNSG. This test will determine the effect of boric acid on corrosion film thickness and on dissolved and undissolved solids in the water. In addition, the production of radiolytic gases will be measured. Also to be determined are the distribution of poison between coolant and vapor space, and the buildup of poison in the loop surfaces.

It should be noted that carbon steel specimens will not be tested in boric acid solutions, since the corrosion behavior of carbon steel in the concentrations of boric acid being considered in this test are known to be too high to permit its use as a material of construction. Other materials will be tested with boric acid. Solutions containing potassium tetraborate and ammonium pentaborate will be tested with all materials (including carbon steel) in outpile static autoclaves prior to testing materials inpile.

The loop should be operated long enough to give adequate data on the use of soluble poisons for lifetime control. The exact period of time, dependent on the data obtained during the test, cannot be stated definitely at this time. At appropriate times during the test, samples will be removed, analyzed for activity pickup, and weighed to determine weight change. At the end of the test with boric acid, all specimens will be removed, and the loop will be drained and flushed in preparation for the next test.

It is desirable to place fuel rods in the inpile section of the loop in order to determine crud deposition on fuel element surfaces as a result of heat flux and radiation.

Second Soluble Poison Test — Use of Potassium Tetraborate

The loop will be filled with deionized, degassed water and will contain potassium tetraborate at a concentration to be specified later. New specimens of all materials will be tested inpile and outpile. This test will be conducted in similar fashion to the boric acid test, and analyses for the same items will be made. The length of this test will also be specified later, the exact time being dependent upon data obtained during the test. At the end of the test, all specimens will be removed, and the loop will be drained and flushed in preparation for the next test.

Third Soluble Poison Test — Use of Ammonium Pentaborate

The loop will be filled with deionized, degassed water and will contain ammonium pentaborate at a concentration to be specified later. New specimens of all materials to be tested will be placed inpile and outpile prior to start of the test. The pH resulting from the use of this poison will range from 9 to 11, depending on the concentration of ammonium pentaborate specified. This test will be conducted in similar manner to the previous soluble poison tests; the exact length, to be specified later, will be determined by data obtained during the test. At the end of the test, all specimens will be removed, and the loop will be drained and flushed in preparation for other tests.

Other Soluble Poison Tests

Other soluble poison tests will be conducted as necessary, dependent upon the results of the boric acid, potassium tetraborate, and ammonium pentaborate tests.

8.5. Core Thermodynamics

A reduction in fuel pin diameter was proposed as a means of reducing the size of the core. Accordingly, heat transfer calculations were performed using reference design thermal and hydraulic conditions. Thermal data for the smaller core are given below.

Table 8.5-1. Data for 0.400-inch Diameter Fuel Pin Core

Reactor power, MWt	62.4
Coolant flow rate, lb/hr	6×10^6
Core operating pressure, psia	816
Average core outlet temperature, F	520
Core inlet temperature, F	490.4
Heat transfer surface, ft^2	1578
Reactor power for fuel melting, MWt	86
Reactor power for burnout, MWt	112
Power density, kw/ℓ of core	60
Power density, kw/ℓ of coolant	100

8.6. Once-Through Steam Generator Studies

8.6.1. Abstract

The transient response and control characteristics of a once-through steam generator for the CNSG have been investigated by analog simulation.

These investigations, while preliminary, indicate that the transient behavior of the once-through steam generator is stable with respect to water carryover to the turbine, steam temperature, pressure, and steam flow. The investigations also indicate that control may be initiated from either the feedwater flow control valves or the turbine throttle valves.

8.6.2. Introduction

The once-through steam generator for the CNSG is designed to produce a moderate amount of superheat (about 70 F) at a moderate pressure (about 400 psig).

The generator receives heat from the reactor, which operates with a constant outlet temperature and varying inlet and average temperatures. Feedwater, introduced to the tube side, is heated, evaporated, and superheated along the length of the unit.

As feedwater flows through the unit, heat absorption takes place in four heat transfer sections, defined as follows.

Economizer section — receives feedwater at the temperature of the last stage feedwater heater and heats the feedwater to saturation temperature at operating pressure. Heat is transferred by convection-conductance.

Nucleate boiling section — receives water at saturation temperature from the economizer section and evaporates the water to x quality (x = % steam by weight). Heat is transferred by nucleate boiling. At some quality x, which is a function of pressure and heat flux (Btu/hr-ft^2), film boiling begins.

Film boiling section — receives water at quality x and completes the evaporation process. Heat is transferred by film boiling, which may be considered as a form of convection-conductance.

Superheat section — receives saturated steam from the boiling section and heats the steam to the desired temperature. Heat is transferred by convection-conductance.

In terms of heat transfer surface requirements, the percentage of total surface of each section at full power is approximately as follows:

Economizer	10%
Nucleate boiling	60%
Film boiling	15%
Superheat	15%

Initial studies of the once-through steam generator were carried out by hand to determine the steady state performance of the unit. The tabulations listed in this section are taken from these initial studies and shown to illustrate the general characteristics of the once-through steam generator.

As power decreases, the heat transfer surface required in each section decreases, although the total surface is constant. This can be seen from

$$S = \frac{Q}{U_o} \Delta Tm$$

where

S = heat transfer surface
Q = power
U_o = over-all heat transfer coefficient
ΔTm = log mean temperature difference.

As the steam and feedwater flow decrease, Q decreases and U_o decreases in economizer, film boiling, and superheat sections. U_o in the nucleate boiling section decreases slightly as the boiling film ΔT and the pressure change, since the nucleate boiling heat transfer coefficient is a function of ΔT and pressure. As Q decreases, however, the average primary system temperature increases, causing the ΔTm's in each section to increase. The net result of a power decrease is to reduce the heat transfer surface requirements of each section. At 20% power the approximate surface proportions are

Economizer	5%
Nucleate boiling	7%
Film boiling	3%
Superheat	85%*

A comparison of heat transfer surface proportions at full power and low power shows that the length, mass, and energy of each section changes significantly as power changes. (See Table 8.6-1.)

An analog simulation of the once-through steam generator investigated the manner in which heat transfer surface, length, volume, pressure, stored mass, stored energy, and flow rates change during transient conditions in order to determine the stability and maneuvering ability of the unit. These investigations also were to determine the basic features of a control system that would permit reliable operation

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*(The actual superheater surface required is only 8.5%, but the remainder of the surface after economizer and boiling surface requirements are met, acts as an inefficient superheater.)

of the unit through specified transients. It must be noted that the analog simulation of the once-through steam generator is preliminary, and that the results of the simulation indicate the basic transient characteristics of the actual unit.

8.6.3. Summary of Results

Analyses of various simulated transients, including step and ramp changes in feedwater flow and ramp changes in steam flow, indicate that the once-through steam generator can exceed the specified maneuvering rate of 20 to 100% power in 80 seconds and the propulsion turbine trip out from 100% power provided that the following conditions are met:

1. Changes in power are initiated by the feedwater flow.

2. Or, if changes in power are initiated by the steam flow, the feedwater flow must be made to lead the steam flow by about 7% within 5 seconds after the steam flow change.

3. For turbine tripout from 100% power, provisions must be made to bypass a small quantity of steam.

With control as specified above, steam outlet conditions are relatively constant at 515 F throughout the transient if the total surface is large enough to accommodate the feedwater flow rate of 107%.

The amount of excess surface required is proportional to the upper limit of feedwater flow. For example, if the feedwater flow never exceeds 110%, an excess surface of 10% is sufficient to prevent loss of superheat.

Table 8.6-1. Mass and Energy in Steam Generator

	Mass, lb				Energy, Btu			
Percent power	20	40	80	100	20	40	80	100
Economizer section								
Primary water	203	273	446	511	110,000	144,000	224,000	253,000
Tube metal	818	1,082	1,720	1,960	56,000	73,000	110,000	123,000
Secondary fluid	209	274	428	483	68,000	93,000	157,000	184,000
Nucleate boiling section								
Primary water	339	763	2,375	3,505	183,000	397,000	1,196,000	1,736,000
Tube metal	1,360	3,035	9,200	13,460	95,000	209,000	604,000	870,000
Secondary fluid	24	55	166	243	20,000	45,000	137,000	201,000
Film boiling section								
Primary water	138	194	320	394	75,000	103,000	164,000	201,000
Tube metal	552	771	1,260	1,560	39,000	54,000	84,000	102,000
Secondary fluid	2.6	3.6	5.9	7.2	3,000	4,000	8,000	8,700
Superheat section								
Primary water	5,060	4,880	2,810	1,585	2,745,000	2,600,000	1,445,000	81,000
Tube metal	20,450	18,290	11,000	6,200	1,446,000	1,356,000	735,000	42,000
Secondary fluid	88	87	49	28	108,000	107,000	61,000	34,000

8.6.4. Physical Model

8.6.4.1. Description

The physical model from which steady-state data and constants for the analog model were derived was a preliminary design of a once-through steam generator developed under AEC contract. Power level, steam flow, materials and operating conditions were based on this preliminary design. Several subsequent changes have been made, including the number of tubes, tube material, power level and steam flow. Time limitations precluded the use of design numbers associated with the final reference design in the analog simulation.

The following parameters of the model were used for the analog simulation and for the present reference design once-through steam generator:

	Model	Reference design
Power rating, Btu/hr	202×10^6	213×10^6
Primary flow rate, lb/hr	6×10^6	6×10^6
Primary inlet temperature (reactor outlet), F	520	520
Primary outlet temperature (reactor inlet), F	492	490.6
Steam flow at full power, lb/hr	209,000	224,000
Steam temperature and pressure, F/psig	515/435	515/400
Number of tubes	146	300
Tube OD & minimum wall thickness	1-in. OD × 0.075-in. min. wall	0.75-in. OD × 0.062-in. avg wall
Average over-all tube length, ft	192*	131
Primary flow area, ft²	8.0	10.9
Secondary flow area, ft²	0.555	0.635
Metal	Carbon steel	Stainless steel
Metal mass, lb	23,000	17,600
Primary coolant mass per ft of tube, lb/ft	97	85
Secondary coolant mass at full power, lb	730	735
Metal time constant, sec	0.18	0.28

* Base actually varied for length studies.

Comparison of variables such as primary mass per foot of tube length, metal mass, metal time constant, and secondary mass, all of which affect time response, indicate that transient data for the model studied are applicable to the reference design.

8.6.4.2. Proposed Method of Control

The coupling of the steam generator to the primary system and reactor is such that the release of stored energy from the primary system to the steam system parallels steam and/or feedwater flow. The reactor changes power level after some time delay, while the primary system conditions change with boiler demand. (See Section 8.7.) For example, in a power increase, the increase in feedwater and steam flow causes an increase in heat transfer rate from the primary system. The increase in heat transfer rate causes the primary outlet temperature from the steam generator to decrease. With a decrease in primary cold leg temperature, the primary average temperature decreases, and the specific volume of the primary system tends to decrease, thereby decreasing primary system pressure. With a decrease in primary system pressure, control rod motion, together with inherent reactivity effects, will maintain a constant core outlet temperature.

During the transient, the length, stored mass, and stored energy of the economizer, boiling, and superheater sections change. Preliminary analysis indicated that the feedwater flow must lead the steam flow for the greater part of a transient in order to maintain proper stored mass and stored energy relationships throughout the transient. Past experience with high performance once-through steam generators has indicated that operation with a constant outlet pressure, and with power demands transmitted directly to the feedwater control device, is suitable for the CNSG. This type of control prepares the steam generator, with respect to mass and energy inventory, for the transient.

8.6.5. Analog Model

8.6.5.1. General

The analog model developed to simulate the once-through steam generator was made up in four sections corresponding to the heat transfer sections of the unit: economizer, boiling, superheat No. 1, and superheat No. 2.

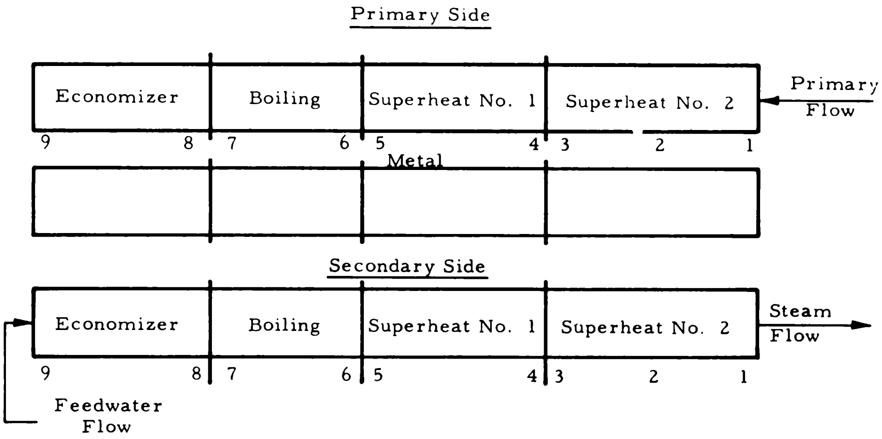

Diagram of Model

In the diagram above, the odd numbers 1, 3, and 5 denote boundaries of each section, and the even numbers denote averages. For example,

primary inlet flow to superheater No. 2 = Wp1,

steam flow out of superheater No. 2 = Ws1,

average metal temperature in superheater No. 2 = Tm 2,

average metal skin temperature on primary side in superheater No. 2 = Tmp 2.

and average metal skin temperature on secondary side in superheater No. 2 = Tm 2.

The description, equations, and assumptions for each section of the model are on the following pages. The basic computer wiring diagram is shown in Figure 8.6-1.

8.6.5.2. General Features and Assumptions

Analysis of the steam generator prior to the analog work indicated that the feedwater flow would have to lead the steam flow during power changes if proper mass inventory, steam pressure, and temperature were to be obtained in the unit. As a result, the analog model was constructed so that power changes were accomplished by changes in feedwater flow.

Analysis of the steam temperature distribution in the superheater indicated that a minimum of two sections on the analog would be required for simulation of the superheater. The film boiling section was lumped with the nucleate boiling section, a combination that little affected the results of the analog runs, since the amount of heat transferred in the film boiling section is about 6% of the latent heat of evaporation. (The enthalpy boundary of the film boiling section is a function of steam pressure and heat flux. For the steam pressure and heat flux under consideration, film boiling begins at a quality of 94-95%.) The analog model treats the boiling zone as one section, making no correction for the lumped film boiling section. Estimated error in total boiling length is 1%.

Constant steam pressure and zero secondary pressure drop were assumed for the analog model. Since it is planned to design the control system to permit constant steam generator outlet pressure, the major consequences of a constant pressure analog model are given below.

With a constant outlet pressure, the inlet pressure to the steam generator must increase to provide the ΔP necessary for flow. As the pressure at the steam generator inlet increases there is an increase in average density, outlet enthalpy, outlet temperature, and average temperature of the economizer section. The boiler inlet enthalpy, inlet temperature, average enthalpy, average temperature, outlet temperature, and outlet enthalpy also increase with pressure, and the heat of vaporization, hfg, decreases with pressure. The superheater inlet temperature, inlet enthalpy, and average density increase with increasing pressure.

The values of temperature and enthalpy used in the analog model are for 100% power. The errors in length and heat transfer rate, which occur at power levels below 100%, and the maximum error in the various sections are as follows:

<u>Economizer Section</u>

Constant secondary outlet temperature and enthalpy cause a 10% error in length and heat transferred. This amounts to 0.6 foot in length and causes a primary temperature error in the economizer section of less than 1 F.

<u>Boiling Section</u>

Errors in length and heat transfer resulting from constant boiler section inlet temperature and enthalpy are less than 2%. This small error results from the nearly constant value of the saturated vapor enthalpy, hf, (boiling section outlet enthalpy), from using the average pressure in the boiling section to calculate the saturation temperature, and from the LMTD. The error in heat transferred causes an error of less than 1 F in the primary ΔT in the boiling section.

Superheater Section

Since the superheater inlet enthalpy, hg is essentially constant, little error results in the heat transfer. The deviation in saturation temperature has very little effect upon the LMTD because of the close approach temperature (5 F) between the steam outlet and the primary inlet. This close approach causes the LMTD to be essentially constant for small variations (5 F ΔT) in steam inlet temperature. The major error in the analog simulation is associated with the mean temperature difference (MTD) in each section. The MTD error occurs because the primary and secondary temperature in each section are assumed to be linear. Following is a comparison of the over-all ΔT obtained from linear temperatures and the LMTD's at full power.

	MTD or ΔT_o	LMTD	% Error
Economizer section	94	85	10
Boiling section	53	51.5	3
Superheater no. 1	47	45	4
Superheater no. 2	16	14	14

Analysis of LMTD's and the values of ΔT_o indicate that the percent error in each section is nearly constant over the power range from 20 to 120%. The resulting errors then give steady state values that are in error, but the transient changes simulate those of the actual unit.

The following additional assumptions were made:

1. Primary density, specific heat, and heat transfer coefficient are constant in each section.

2. Metal density, specific heat, and thermal conductivity are constant in each section.

3. Heat flux per unit length in each section is constant.

4. Primary and metal section boundaries coincide with secondary boundaries, and there is no heat transfer, for example, from the primary economizer section to the secondary boiling section.

5. Metal temperature gradient is linear.

6. The secondary boiling heat transfer coefficient is constant. Economizer and superheater heat transfer coefficients vary with flow.

7. The initial conditions for the analog model were set up so that nominal variation of feedwater or steam flow would be from 10 to 60 lb/sec. Steam flow in the physical model was 58 lb/sec at 100% power. In terms of percentage of full flow, 10 lb/sec would be 17% of full flow, and 60 lb/sec would correspond to 102% of full flow. For the overpower transients, feedwater and steam flow were taken to be 66 lb/sec and 72 lb/sec respectively. These values correspond to 114% and 124% of full power respectively.

Equations

The equations beginning on Page 152 describe the once-through steam generator.

W_p Primary flow rate, lb/second. Subscripts denote location: W_{p_1} = inlet to superheater No. 2, W_{p_3} = outlet from superheater No. 2 = inlet to superheater No. 1, W_{p_5} = outlet from superheater No. 1 = inlet to boiling section, W_{p_7} = outlet from boiling section = inlet to economizer section, W_{p_9} = outlet from economizer section.

W_s Secondary flow rate, lb/second. Subscripts denote location: W_{s_9} = feedwater flow into economizer section, W_{s_7} = flow of saturated water out of economizer section, W_{s_5} = flow of saturated vapor out of boiling section, W_{s_3} = flow of superheated steam out of superheater No. 1, W_{s_1} = flow of superheated steam out of superheater No. 2 = steam flow to turbines, W_{f_9} = flow between water and steam phase.

T_p Primary temperature, (F). Subscripts denote location as in flow. Even numbers, i.e., T_{p_2}, T_{p_4}, T_{p_6}, and T_{p_8} are average temperatures in superheater No. 2, superheater No. 1, boiling section, and economizer section, respectively.

T_{mp} Tube metal skin temperature (F) at surface in contact with primary system. Subscripts T_{mp_2}, T_{mp_4}, etc., denote location as given above.

T_m Average tube metal temperature (F) at t/2, where t/2 is one-half the thickness of the tube wall. Subscripts T_{m_2}, T_{m_4}, etc., denote location as given above.

T_{ms} Tube metal skin temperature (F) at surface in contact with secondary fluid. Subscripts T_{ms_2}, T_{ms_4}, etc., denote location as given above.

T_s Secondary temperature, (F). Subscripts T_{s_1}, T_{s_2}, T_{s_3}, T_{s_4}, etc., denote location as given above.

L	Length of tubing in given section. Subscripts denote section, i.e., L_2, L_4, L_6, L_8 are superheat No. 2, superheat No. 1, boiling, and economizer sections, respectively.
M	Mass of fluid, lb. Subscripts denote location.
C_p	Specific heat, Btu/lb-F. The second p in C_{pp} denotes primary and the m in C_{pm} denotes metal.
ρ	Density, lb/ft^3. Subscripts ρ_p, ρ_m, ρ_s denote primary, metal, and secondary, respectively.
v	Specific volume, ft^3/lb.
V	Volume, ft^3.
h	Secondary enthalpy Btu/lb. Subscripts h_1, h_2, h_3, etc., denote location as given above.
U	Local heat transfer coefficient, Btu/sec-ft^2-F.
Q_p	Heat transfer rate (Btu/sec) to or from primary system. Subscripts Q_{p_2}, Q_{p_4}, Q_{p_6}, etc., denote location as given above.
Q_s	Heat transfer rate (Btu/sec) to or from secondary system. Subscripts Q_{s_2}, Q_{s_4}, etc., denote location as given above.
A_p	Primary flow area (ft^2) constant.
A_s	Secondary flow area (ft^2) constant.
A_m	Cross-sectional area of tube metal, ft^2.
N	Number of tubes.
D	Tube outside diameter, inches.
$\dot{M}$ or $\dot{L}$ $\dot{T}$ or $\dot{V}$	First derivative of quantity $= \frac{dn}{dt}$ or $\frac{dL}{dt}$ or $\frac{dT}{dt}$ or $\frac{dV}{dt}$.
f	Function.
hf	Enthalpy of saturated liquid, Btu/lb.
hg	Enthalpy of saturated vapor, Btu/lb.
Mg	Mass of saturated vapor, lb.

M_f Mass of saturated liquid, lb.

A_1, A_2, B_1, B_2, C_1, C_2 — Constants.

1. Economizer Equations

a. Primary

Heat Balance

$$C_p \, (\dot{M}T)_{p_8} = W_{p_7} \, C_{pp} \, T_{p_7} - W_{p_9} \, C_{pp} \, T_{p_9} - Q_{p_8}$$

$$T_{p_8} = \left(T_{p_7} + T_{p_9}\right) / 2$$

Mass Balance

$$\dot{M}_{p_8} = A_p \rho_p \, \dot{L}_8 = W_{p_7} - W_{p_9}$$

where

$$\dot{L}_8 = \frac{W_{s_9} - W_{s_7}}{A_s \rho_{s_8}}.$$

Heat Transferred From Primary Coolant To Metal

$$Q_{p_8} = N \, \Pi \, D_p \, L_8 \, U_{p_8} \left(T_{p_8} - T_{mp_8}\right)$$

where $N \, \Pi \, D_p \, U_{p_4}$ is constant.

b. Metal

Heat Balance

$$C_{pm} \, (\dot{M}T)_{m_8} = Q_{p_8} - Q_{s_8} + \dot{M}_{m_8} \, C_{pm} \, T_{m_7}^{*}$$

Mass Balance

$$\dot{M}_{m_8} = A_m \rho_m \, \dot{L}_8$$

where $\dot{L}_8$ is given in Equation (4).

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*The last term is the heat added as a result of the boundary moving.

Boundary Condition at Surface of Secondary Side

$$\text{Heat Flux} = U_{s_8}\left(T_{ms_8} - T_{s_8}\right) = \left(\frac{K_m}{(\Delta X)_{m/2}}\right)\left(T_{m_8} - T_{ms_8}\right) \bullet \tag{8}$$

$$\text{Let} \qquad \frac{K_m}{(\Delta X)_m} = U_m \text{ (constant)} \tag{9}$$

$$\text{then} \qquad U_{s_8}\left(T_{ms_8} - T_{s_8}\right) = 2\,U_m\left(T_{m_8} - T_{ms_8}\right). \tag{10}$$

$$T_{m_8} = \left(T_{mp_8} + T_{ms_8}\right)/2$$

$$\text{or } T_{mp_8} = 2\,T_{m_8} - T_{ms_8} \tag{11}$$

$$U_{s_8} = f\left(W_{s_9}\right) \tag{12}$$

c. Secondary

Mass Balance

$$\dot{M}_{s_8} = A_s\,\rho_{s_8}\,\dot{L}_8 = W_{s_9} - W_{s_7} \tag{13}$$

$$\dot{L}_8 = \frac{W_{s_9} - W_{s_7}}{A_s \rho_{s_8}} \quad \text{(same as 4)}$$

Heat Balance (Constant Pressure)

$$\dot{M}_{s_8} h_{s_8} = W_{s_9} h_{s_9} - W_{s_7} h_{s_7} + Q_{s_8} \tag{14}$$

$$h_{s_7} = h_{fs} \tag{15}$$

$$Q_{s_8} = N\,\Pi\,D_s\,L_8\,U_{s_8}\left(T_{ms_8} - T_{s_8}\right) \tag{16}$$

2. Boiler Equations

a. Primary

Heat Balance

$$C_{pp}\,(\dot{M}T)_{p_6} = W_{p_5}\,C_{pp}\,T_5 - W_{p_7}\,C_{pp}\,T_7 - Q_{p_6} \qquad (17)$$

$$T_{p_6} = \left(T_{p_5} + T_{p_7}\right)/2 \qquad (17a)$$

Mass Balance

$$\dot{M}_{p_6} = A_p \rho_p\,\dot{L}_6 = W_{p_5} - W_{p_7} \qquad (18)$$

where

$$\dot{L}_6 = \frac{W_{s_7} - W_{s_5}}{A_s \rho_{s_6}}\,. \qquad (19)$$

Heat Transferred From Primary Coolant to Metal

$$Q_{p_6} = N\Pi\,D_p\,L_6\,U_{p_6}\left(T_{p_6} - T_{mp_6}\right) \qquad (20)$$

where $N\Pi\,D_p\,U_{p_6}$ is constant.

b. Metal

Heat Balance

$$C_{pm}\,(\dot{M}T)_{m_6} = Q_{p_6} - Q_{s_6} - \dot{M}_{m_8}\,C_{pm}\,T_{m_7}{}^{*} + \dot{M}_{m_8} + \dot{M}_{m_6}\,C_{pm}\,T_{m_5}{}^{**} \qquad (21)$$

Mass Balance

$$\dot{M}_{m_6} = A_m \rho_m\,\dot{L}_6 \qquad (22)$$

where $\dot{L}_6$ is given by Equation (26).

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*Heat lost as economizer boundary moves down stream.

**Heat added as boiling section boundary moves downstream at a rate equal the sum of the rates of change of the economizer and the boiler lengths.

Boundary Condition at Secondary Side Metal Surface

$$\text{Heat Flux} = U_s \left(T_{ms_6} - T_{s_6}\right) = \left(\frac{K_m}{(\Delta X)_{m/2}}\right)\left(T_{m_6} - T_{ms_6}\right) \tag{23}$$

$$U_{s_6} \left(T_{ms_6} - T_{s_6}\right) = 2\, U_m \left(T_{m_6} - T_{ms_6}\right) \tag{24}$$

$$U_{s_6} = \text{Constant}\ U_m = \text{Constant}$$

$$T_{m_6} = \frac{T_{ms_6} + T_{mp_6}}{2} \text{ or } T_{mp_6} = 2\, T_{m_6} - T_{ms_6} \tag{25}$$

c. Secondary (Constant Pressure Assumed)

Mass Balance

$$\dot{M}_g = W_{fg} - W_{s_5} \tag{26}$$

$$\dot{M}_f = W_{s_7} - W_{fg} \tag{27}$$

$$A_s \rho_{s_6} \dot{L}_6 = \dot{M}_{s_6} = \dot{M}_g + \dot{M}_f \tag{28}$$

Volume Balance

$$A_s \dot{L}_6 = \dot{V}_{s_6} = \dot{M}_g V_g + \dot{M}_f V_f \tag{29}$$

Heat Balance

$$\dot{M}_g h_g + \dot{M}_f h_f = W_{s_7} h_{s_7} - W_{s_5} h_{s_5} + Q_{s_6} \tag{30}$$

$$Q_{s_6} = N \Pi D_s L_6 U_{s_6} \left(T_{ms_6} - T_{s_6}\right) \tag{31}$$

3. Superheater No. 1 — Constant Length = 10 feet

a. Primary

Heat Balance

$$C_{pp}\,(\dot{M}T)_{p_4} = W_{p_3}\,C_{pp}\,T_{p_3} - W_{p_5}\,C_{pp}\,T_{p_5} - Q_{p_4} \tag{32}$$

$$T_{p_4} = \left(T_{p_3} + T_{p_5}\right)/2 \tag{33}$$

Mass Balance

$$\dot{M}_{p_4} = W_{p_5} - W_{p_7} = 0 \text{ (Fixed Length)} \tag{34}$$

Heat Transferred From Primary Coolant To Metal

$$Q_{p_4} = N\,\Pi\,D_p\,L_4\,U_{p_4}\left(T_{p_4} - T_{mp_4}\right) \tag{35}$$

where $N\,\Pi\,D_p\,L_4\,U_{p_4}$ is constant.

b. Metal

Heat Balance

$$C_{pm}\,(\dot{M}T)_{m_4} = Q_{p_4} - Q_{s_4} - \left(\dot{M}_{m_8} + \dot{M}_{m_6}\right)C_{mp}\,T_{m_5} + \left(\dot{M}_{m_8} + \dot{M}_{m_6} + \dot{M}_{m_4}\right)C_{pm}\,T_{m_3} \tag{36}$$

Mass Balance

$$\dot{M}_{m_4} = 0 \text{ (Fixed Length)} \tag{37}$$

Boundary Condition at Surface of Secondary Side

$$U_{s_4}\left(T_{ms_4} - T_{s_4}\right) = \left(\frac{K_m}{\Delta X_{m/2}}\right)\left(T_{m_4} - T_{ms_4}\right) \tag{38}$$

$$\frac{K_m}{(\Delta X)_m} = U_m = \text{Constant}$$

$$\text{Then } T_{ms_4} = \frac{2\,U_m\,T_{m_4} + U_{s_4}\,T_{s_4}}{2\,U_m + U_{s_4}} \tag{39}$$

$$T_{m_4} = \left(T_{mp_4} + T_{ms_4}\right)/2 \text{ or } T_{mp_4} = 2\,T_{m_4} - T_{ms_4} \tag{40}$$

c. <u>Secondary</u> (Constant Pressure)

<u>Mass Balance</u>

$$\dot{M}_{s_4} = AL_4\dot{\rho}_{s_4} = W_{s_5} - W_{s_3} \tag{41}$$

<u>Volume Balance</u>

$$\dot{V}_{s_4} = (\dot{M}V)_{s_4} = 0 \text{ (Fixed Length)} \tag{42}$$

<u>Heat Balance</u>

$$(\dot{M}h)_{s_4} = W_{s_5}\,h_{s_5} - W_{s_3}\,h_{s_3} + Q_{s_4} \tag{43}$$

<u>Enthalpy</u>

$$h_{s_5} = h_{gs} \tag{44}$$

$$\frac{h_{s_5} + h_{s_3}}{2} = h_{s_4} \tag{45}$$

$$h_{s_4} = C_1 + C_2\,V_{s_4} \tag{46}$$

$$\dot{h}_{s_4} = C_2\,\dot{V}_{s_4} \tag{47}$$

$$T_{s_4} = A_1 + A_2\,h_{s_4} \tag{48}$$

$$Q_{s_4} = N\Pi\,D_s\,L_4\,U_{s_4}\left(T_{ms_4} - T_{s_4}\right) \tag{49}$$

where $U_{s_4} = f\left(W_{s_5}\right)$.

4. Superheater No. 2

a. <u>Primary</u>

<u>Heat Balance</u>

$$C_{pp}\,(\dot{M}T)_{p_2} = W_{p_1}\,C_{pp}\,T_{p_1} - W_{p_3}\,C_{pp}\,T_{p_3} - Q_{p_2} \tag{50}$$

$$T_{p_2} = \left(T_{p_1} + T_{p_3}\right)/2 \tag{51}$$

<u>Mass Balance</u>

$$\dot{M}_{p_8} + \dot{M}_{p_6} + \dot{M}_{p_4} + \dot{M}_{p_2} = W_{p_1} - W_{p_9} = 0$$

$$\dot{M}_{p_4} = 0 \text{ (Fixed Length)} \tag{52}$$

<u>Heat Transferred From Primary Coolant to Metal</u>

$$Q_{p_2} = N\,\Pi\,D_p\,L_2\,U_{p_2}\left(T_{p_2} - T_{mp_2}\right) \tag{53}$$

where $N\,\Pi\,D_p\,U_{p_2}$ is Constant.

b. <u>Metal</u>

<u>Heat Balance</u>

$$C_{pm}\,(\dot{M}T)_{m_2} = Q_{p_2} - Q_{s_2} - \left(\dot{M}_{m_8} + \dot{M}_{m_6} + \dot{M}_{m_4}\right) C_{pm}\,T_{m_3} \tag{54}$$

<u>Mass Balance</u>

$$\dot{M}_{m_8} + \dot{M}_{m_6} + \dot{M}_{m_4} + \dot{M}_{m_2} = 0 \text{ (Total Length Constant)} \tag{55}$$

<u>Boundary Condition at Surface of Secondary Side</u>

$$U_{s_2}\left(T_{ms_2} - T_{s_2}\right) = \left(\frac{K_m}{\Delta X_{m/2}}\right)\left(T_{m_2} - T_{ms_2}\right) \tag{56}$$

where $U_{s_2} = f\left(W_{s_5}\right)$, since $\left(W_{s_5} = W_{s_3}\right)$.

$$\left(\frac{K_m}{\Delta X_m}\right) = U_m = \text{Constant},$$

$$\text{then} \quad T_{ms_2} = \frac{2\,U_m\,T_{m_2} + U_{s_2}\,T_{s_2}}{2\,U_m + U_{s_2}} \,. \tag{57}$$

$$T_{m_2} = \left(T_{mp_2} + T_{ms_2}\right)/2 \text{ or } T_{mp_2} = 2\,T_{m_2} - T_{ms_2} \tag{58}$$

c. Secondary

Mass Balance

$$\dot{M}_{s_2} = A_s\,L_2\,\dot{\rho}_{s_2} = W_{s_3} - W_{s_1} \tag{59}$$

Volume Balance

$$\dot{V}_{s\ total} = 0 = \left(\dot{M}V\right)_{s_8} + \left(\dot{M}V\right)_{s_6} + \left(\dot{M}V\right)_{s_4} + \left(\dot{M}V\right)_{s_2}$$

$$\left(\dot{M}V\right)_{s_4} = 0 \text{ (Constant Length)} \tag{60}$$

Heat Balance

$$(\dot{M}h)_{s_2} = W_{s_3}\,h_{s_3} - W_{s_1}\,h_{s_1} + Q_{s_2} \tag{61}$$

Enthalpy

$$\left(h_{s_1} + h_{s_3}\right)/2 = h_{s_2} \tag{62}$$

$$h_{s_2} = C_1 + C_2\,V_{s_2} \tag{63}$$

$$\dot{h}_{s_2} = C_2\,\dot{V}_{s_2} \tag{64}$$

$$T_{s_2} = A_1 + A_2\,h_{s_2} \tag{65}$$

$$Q_{s_4} = N\,\Pi\,D_s\,L_2\,U_{s_2}\left(T_{ms_2} - T_{s_2}\right) \tag{66}$$

$$h_{s_2} = B_1 + B_2\,\rho_{s_2} \tag{67}$$

8.6.6. Results

Transient analyses of the once-through steam generator were divided into two phases: transient stability investigations, and control characteristics. Table 8.6-2 lists the runs and the type of transient for each.

8.6.6.1. Transient Stability Investigations

Initial transient runs were carried out to determine the flow, length, and temperature characteristics of the unit by varying the feedwater flow in ramps and steps. Table 8.6-3 lists the results. Figures 8.6-2, 8.6-3, and 8.6-4 are typical recorded curves showing the response of section flow, length, and temperature for a feedwater flow ramp of 10 to 60 lb/sec in 20 seconds.

Figure 8.6-2 gives flow rates for feedwater economizer, boiling section, superheater No. 1, and superheater No. 2 (steam). This figure also shows the following:

1. The economizer flow rate begins to increase as feedwater flow increases; however, the economizer flow increases at a rate of about 20 lb/sec^2 for 18 seconds and then reaches 60 lb/sec after 53 seconds. The integrated difference between the feedwater flow rate and the economizer flow rate represents the storage of mass in the economizer section during the transient.

2. The boiler flow rate lags the economizer flow rate. Maximum rate of change of the boiler flow is about 1.56 lb/sec^2 while that of the boiler inlet is about 2.0 lb/sec. Boiler flow reaches 60 lb/sec in about 60 seconds. Integrated difference in boiler flow and economizer flow represents the storage of mass in the boiling section during the transient.

3. The superheater No. 1 flow nearly parallels the boiler flow. A slight difference in flow rates is to be expected, since the density of this section increases slightly as the average temperature decreases. This increase causes an increase in stored mass during the transient, which causes the outflow to lag the inflow.

4. The superheater No. 2 flow (steam flow) leads the flow from superheater No. 1. The maximum rate of change of steam

flow is about 1.8 lb/sec^2 while that of superheater No. 1 is about 1.5 lb/sec^2. The integrated difference in these flow rates represents the loss of stored mass in superheater No. 2 during the transient.

Figure 8.6-3 shows the length changes in the economizer, boiling, and superheater No. 2 sections for the 20-second feedwater ramp. It also shows that length changes in each section parallel the flow in each section, and that changes in length occur in a smooth manner; that is, there are no oscillations that would indicate a "hunting" condition.

Figure 8.6-4 shows temperature changes in several locations in the steam generator model for the 20-second feedwater ramp (up and down). The total changes in the temperatures agree with predicted values, and the rate of change of T_{p_8} agrees with the value used in Section 8.7.

Figures 8.6-4, 8.6-5, and 8.6-6 are typical recorded curves of flow rates and lengths for feedwater steps. Figure 8.6-5 gives flow for the economizer, boiling section, superheater No. 1, and steam section for a feedwater step from 10 to 60 lb/sec and from 60 to 5 lb/sec with a feedwater ramp back to 10 lb/sec in 6 seconds. This curve shows that the economizer flow (that is, the quantity of feedwater that reaches saturation enthalpy at any instant) reaches 60 lb/sec in about 20 seconds. Boiler flow lags the economizer flow and reaches 60 lb/sec in about 30 seconds. Flow from superheater No. 1 is identical to the boiler flow since the density and thus the mass change in this section is very small. Superheater No. 2, or steam flow, leads boiler flow and superheater No. 1 flow, because superheater No. 2 is losing mass; thus, the outflow must be greater than the inflow.

Mass checks were made by integrating the inflow and outflow of each section,

$$\frac{dM}{dt} = W_{in} - W_{out}, \text{ or } dM = W_{in} - W_{out}\, dt,$$

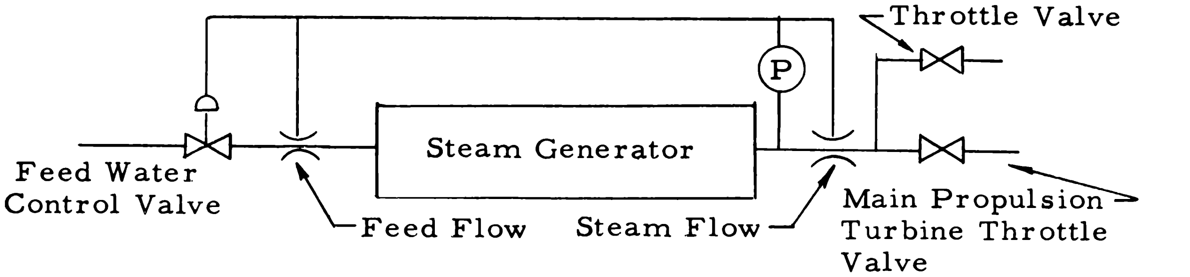
and comparing it with the mass obtained by $M = V\rho$ where $V = LA$. These values agreed within 2% in each section.

Figure 8.6-6 shows the length of the economizer, boiling section, and superheater No. 2 for feedwater steps of 10 to 60 lb/sec and 60 to 10 lb/sec. The economizer length changes by about 6 feet while the boiling section and superheater lengths change by 87 and 94 feet respectively. For this transient, the total surface is greater than that required. To maintain a steam temperature of 515 F at a flow of 60 lb/sec, the required length of superheater No. 2 is about 20 feet. As shown on the curve, the length of superheater No. 2 is about 40 feet.

Figure 8.6-7 shows the length of superheater No. 2 for feedwater steps from 10 to 60 lb/sec, 10 to 66 lb/sec, and 10 to 72 lb/sec. The 10 to 60 lb/sec case is identical to that of Figure 8.6-9. For a feedwater step from 10 to 66 lb/sec (this would correspond to 110% power), the superheater No. 2 length drops to about 25 feet, a length sufficient to maintain steam temperature at 515 F. For a feedwater step from 10 to 72 lb/sec, the superheater length drops to about 10 feet, a length insufficient to maintain 515 F steam temperature. The enthalpy curve plotted at the top of Figure 8.6-10, for the 10 to 72 lb/sec step, shows that steam temperature drops to about 497 F.

8.6.6.2. Control

Transient stability investigations were carried out by stepping or ramping feedwater flow and observing the response of other variables. Analysis of possible control schemes for the once-through unit indicated that it would be desirable to initiate transients from the steam side of the unit via the propulsion turbine throttle or the turbine generator throttles. At the same time, we recognized that feed-water flow must lead the steam flow for the greater part of the transient in order to prevent evaporation of the entire mass in the unit. With this in mind, a three-element control system comprised of steam flow measurement, steam pressure measurement, and feedwater flow measurement was investigated. Basic operation of the three-element control system follows:



The basic equation for this system is

$$W_{fw} = W_{stm} + K(\Delta P_e)$$

where

W = flow rate

fw = feedwater

stm = steam

K = Gain of pressure signal

ΔP_e = Steam pressure error.

As the turbine throttle valve or valves are opened, steam flow increases, and steam pressure begins to drop. Thus, the steam flow increase and the pressure decrease tend to open the feedwater control valve. If the gain of the pressure signal can be made to reach the proper magnitude, the feedwater flow can be made to lead the steam flow, after some delay time. Since the control devices are electrical, major delay times are in the pressure sensing device and in the response time of the feedwater control valve. Total delay time between the steam and pressure signal and the opening of the feedwater control valve has been estimated to be less than 2 seconds.

With this control scheme in mind, several transient runs were made to determine the feedwater flow required for given steam flow ramps. Since the analog model could be controlled only by changes in feedwater flow, several runs had to be made to obtain a steam ramp. Results of these runs are shown in Table 8.6-4. Figure 8.6-8 is a typical recorded curve showing the feedwater flow required and the change in length of superheater No. 2 for steam flow ramps of 10 to 60 lb/sec and 60 to 10 lb/sec in 80 seconds each way. Figure 8.6-8 shows that ramp changes in steam flow, in this case 10 to 60 lb/sec in 80 seconds, can be accomplished by increasing the feedwater flow rate in three ramps. The first ramp would cause the feedwater flow to change by about 1.75 lb/sec^2 for 8 seconds. The second ramp would be at 0.625 lb/sec^2 for 66 seconds (this gives an overshoot to 62 lb/sec), and the third ramp would reduce the feedwater flow from 62 to 60 lb/sec in 6 seconds. A similar procedure is required for the down transient except that the feedwater "undershoot" is to 6 lb/sec and the feedwater flow then increases to 10 lb/sec in about 8 seconds.

The length of superheater No. 2 decreases at a rate compatible with the increase in steam flow rate. When the steam flow rate reaches 60 lb/sec, the length of superheater No. 2 reaches an equilibrium value.

Figure 8.6-9 shows the steam pressure variation when feedwater flow lags the steam flow at the start of an up transient. This curve was derived from hand calculations that assumed a 3-second delay time in the steam system. After this period, feedwater flow increased to 110% of the steam flow in 2 seconds and held at 110% of the steam flow until steam flow reached 100% flow. The steam pressure drops by about 10 psi in 3 seconds and begins to rise again after the feedwater flow overtakes the steam flow. Within 10 seconds, steam pressure has returned to the original value.

Figure 8.6-10 shows the relation of several variables to a steam ramp of 10 to 60 lb/sec and 60 to 10 lb/sec if the reactor is self regulating (that is, with no control rod motion). In this case the primary system inlet temperature to the steam generator (core outlet temperature) decreases after the delay time (10 seconds) between the steam generator outlet and the core inlet. This primary inlet temperature was ramped in at 0.2 F/sec for 100 seconds. In order to run the problem, the superheater No. 2 surface had to be increased by 30 feet to account for the decrease in LMTD as primary system temperature decreased. The boiling section outlet temperature approximates that used for the primary system studies (Section 8.7).

The feedwater flow and steam flow valves for the feedwater flow step decrease shown in the down transient portion of Figure 8.6-5 also may be considered as turbine trip conditions. The figure shows that steam flow lags feedwater flow. This indicates that a steam bypass or a steam dump system is necessary to relieve the quantity of steam produced after the feedwater valve and propulsion turbine valves have been closed.

8.6.7. Summary of Results and Conclusions

When controlled by feedwater flow, either by initiating the transient or by leading the steam flow after a short time delay, the transient response of the once-through CNSG is stable with respect to steam flow and steam temperature. When the feedwater flow lags the steam flow for short periods of time, pressure swings of about 10 psi may be expected.

The excess heat transfer surface provided in the superheater is directly related to the overpumping that the feed pumps can produce. For example, if the feed pumps can be limited to 110% of full flow for short periods of time (10 seconds or less), an excess of about 5% in heat transfer surface is adequate.

The major conclusion is this: when feedwater flow responds to the control system in such a manner that feedwater flow leads steam flow for power increase and power decrease transients, the once-through steam generator is a reliable, rapidly responding component of the CNSG.

Table 8.6-2. Tabulation of Transient Runs

Run no.	Transient time	Variables investigated
1	Feedwater flow ramp, 10 to 60 lb/sec in 10 sec	
2	10 to 60 lb/sec in 4 sec	
3	10 to 60 lb/sec in 20 sec	Steam flow, section length, and primary and steam temperatures
4	10 to 60 lb/sec in 40 sec	
5	10 to 60 lb/sec in 80 sec	
6	60 to 10 lb/sec in 80 sec	
7	Feedwater flow step, 10 to 60 lb/sec	
8	60 to 10 lb/sec	
9	60 to 5 lb/sec and ramp from 5 lb/sec to 10 lb/sec in 10 seconds	Steam flow; section length, and steam temperature
10	10 to 66 lb/sec	
11	66 to 10 lb/sec	
12	10 to 72 lb/sec	
13	72 to 10 lb/sec	
14	Steam flow ramp, 10 to 60 lb/sec in 80 sec	
15	60 to 10 lb/sec in 80 sec	
16	10 to 60 lb/sec in 50 sec	
17	60 to 10 lb/sec in 50 sec	
18	10 to 60 lb/sec in 40 sec	Feedwater flow and superheater No. 2 length
19	60 to 10 lb/sec in 40 sec	
20	10 to 66 lb/sec in 80 sec	
21	66 to 10 lb/sec in 80 sec	
22	10 to 72 lb/sec in 80 sec	
23	72 to 10 lb/sec in 80 sec	
24	Various feedwater ramps and steps to simulate turbine generator load changes	Steam flow and section lengths
25	Steam flow ramps to simulate turbine generator load changes	Steam flow and section lengths
26	Feedwater ramp with variable primary inlet temperature	Section lengths, and primary outlet temperature

Table 8.6-3. Response of Section Length, Temperature, and Steam or Feedwater Flow to Changes in Feedwater or Steam Flow

	Run No. 1	2	3	4	5	6	7
W_{s9} -Feedwater Flow Change	10-60	10-60	10-60	10-60	10-60	10-10	10-60
Ramp Time	10	4	20	40	80	80	Step
W_{s1} -Total Change	10-60	10-60	10-60	10-60	10-60	60-60	
Time to Reach 2/3 Final Value	10		18	30	61	61	
Time to Reach Final Value	50		53	67	109	109	
W_{s5} -Total Change	10-60		10-60	10-60	10-60	60-60	
Time to Reach 2/3 Final Value	15		20	34	64	64	
Time to Reach Final Value	60		63	73	114	114	
W_{s3} -Total Change	10-60		10-60	10-60	10-60	60-10	
Time to Reach 2/3 Final Value	15		20	34	64	64	
Time to Reach Final Value	60		63	73	114	114	
W_{s1} -Total Change	10-60		10-60	10-60	10-60	60-10	10-60
Time to Reach 2/3 Final Value	13	10	18	33	60	60	8
Time to Reach Final Value	55	25	58	70	109	109	30
L_8 -Total Change	6.2	6.2	6.2	6.2	6.2	6.2	
Time to Reach 2/3 Final Value			18	33			
Time to Reach Final Value			53	65			
L_6 -Total Change	82.8	82.8	82.8	82.8	82.8	82.8	
Time to Reach 2/3 Final Value			20	35			
Time to Reach Final Value			63	70			
L_2 -Total Change	89	89	89	89	89	89	
Time to Reach 2/3 Final Value			23	35			
Time to Reach Final Value			60	70			
T_{p8} -Total Change			21		21	21	
Time to Reach 2/3 Final Value			24		70	63	
Time to Reach Final Value			60		109	109	
T_{p6} -Total Change			10		10	10	
Time to Reach 2/3 Final Value			27		65	73	
Time to Reach Final Value			60		112	112	
T_{m8} -Total Change			33		33	33	
Time to Reach 2/3 Final Value			23		55	60	
Time to Reach Final Value			65		112	112	
T_{mp6} -Total Change			8		8	8	
Time to Reach 2/3 Final Value			30		75	67	
Time to Reach Final Value			60		110	110	

	Run No. 8	9	10	11	12	13	14	Units
W_{s9} -Feedwater Flow Change	60-10	10-60	60*-10	10-66	66*-10	10-72	72*-10	lb/sec
Ramp Time								sec
W_{s1} -Total Change								lb/sec
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
W_{s5} -Total Change								lb/sec
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
W_{s3} -Total Change								lb/sec
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
W_{s1} -Total Change	60-10	10-60	60-10	10-66	66-10	10-72	72-10	lb/sec
Time to Reach 2/3 Final Value	4	8	4					sec
Time to Reach Final Value	40	30	24					sec
L_8 -Total Change		6	6					ft
Time to Reach 2/3 Final Value		8	14					sec
Time to Reach Final Value		30	30					sec
L_6 -Total Change		88	88					ft
Time to Reach 2/3 Final Value		9	9					sec
Time to Reach Final Value		50	50					sec
L_2 -Total Change		94	94	109	109	124	124	ft
Time to Reach 2/3 Final Value		9	9	10	13	11	14	sec
Time to Reach Final Value		50	50	60	60	60	60	sec
T_{p8} -Total Change								°F
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
T_{p6} -Total Change								°F
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
T_{m8} -Total Change								°F
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec
T_{mp6} -Total Change								°F
Time to Reach 2/3 Final Value								sec
Time to Reach Final Value								sec

*Feedwater flow stepped down to 5 lb/sec until steam flow approaches 10 lb/sec. Feedwater flow is then ramped to 10 lb/sec in 6 seconds.

Table 8.6-4. Response of Feedwater Flow and Superheater No. 2 Length to Steam Flow Ramps

	Run No.									
	15	16	17	18	19	20	21	22	23	24
W_1-Steam Flow Change-lb/sec	10-60	60-10	10-60	60-10	10-60	60-10	10-66	66-10	10-72	72-10
Ramp Time-sec	80	80	50	50	40	40	80	80	80	80
W_9-Initial Rate of Feedwater Flow Change-lb/sec^2	1.5	1.5	2.3	2.3	2.4	2.4				
Final Rate Feedwater Flow Change-lb/sec^2	.625	.625	1.0	1.0	2.0	2.0				
L_2-Total Change	94	94					109	109	124	124
Time to Reach 2/3 Final Value	28	23					32	23	36	26
Time to Reach Final Value	65	65					65	65	65	65

8.7. Primary System Analog Studies

8.7.1. Introduction

Analog computer studies were made to determine the basic operational characteristics of the CNSG during normal load changes. The effects of ship's motion on performance were also studied.

The parameters of a preliminary CNSG design were used in the analog stuides so that the transient characteristics of the concept could be evaluated concurrently with the reference design analyses. Table 8.7-1 summarizes the parameters used in the analog studies. The results of the preliminary CNSG analog studies can be qualitatively related to the reference CNSG since the two designs have the same basic features.

The CNSG is a self-pressurized reactor with forced circulation and an integral steam generator. The primary system was studied separately from the secondary system by applying the load demand directly to the primary side of the steam generator. All reactor power changes were effected without control rod motion, thus the CNSG is simulated as a self-regulating reactor. The reactor demonstrated stable operating characteristics for all of the imposed transients including severe conditions of abnormal operation such as step changes in power demand.

The following sections describe the analytical procedure, the results, an analysis of results, and conclusions.

8.7.2. Analytical Procedure

8.7.2.1. General

Figure 8.7-1 shows the basic model and the symbols used to develop the CNSG analog simulation. The block diagrams and the analog computer wiring diagrams for power increasing and power decreasing transients are presented in Figure 8.7-2 through 8.7-5. The analytical development of the component models is also described in this section.

Table 8.7-1. Preliminary CNSG Design Parameters

Reactor	
Maximum power to propeller, shp	22,000
Reactor maximum operating power, MWt	69
Reactor normal operating power, MWt	63.5
Overall thermal efficiency, %	24
Coolant flow rate, lb/hr	6×10^6 (Forced Circulation)
Operating pressure, psia	812
Coolant outlet temperature, F	520
Coolant inlet temperature, F	487
Core*	
Fuel	UO_2
Cladding	Zircaloy
No. zones	3
Initial fuel loading, kg U (wt % U^{235})	
Zone I	129.7 (1.90)
Zone II	2,208.8 (1.90)
Zone III	2,208.8 (2.70)
Total	4,547.3 (2.29)
Fuel rod OD, in.	0.460
Cladding thickness, in.	0.022
Fuel rod pitch, triangular, in.	0.686
Metal-to-water ratio (unit cell)	0.70
Fuel assembly, geometry	Hexagonal Clusters
No. fuel rods/full assembly	124
No. support rods/assembly	3
No. fuel assemblies	37
No. fuel rods/core	4,348
No. support rods/core	111
Core diameter, equivalent, in.	50.1
Core diameter, maximum, in.	53.9
Core length, in. (active)	54.0
Length to equivalent diameter ratio	1.078
Core volume, $in.^3$ (active)	10.7×10^4
Average burnup, MWd/MT-U	16,550
Days at operating power/fuel cycle	613
Heat transfer	
Total heat transfer area, ft^2	2,356
Average heat flux, maximum power, Btu/hr-ft^2	99,960
Maximum heat flux, maximum power, Btu/hr-ft^2	287,200

*The preliminary core is physically the same as the Task 25 Core I.[1]

Table 8.7-1 (Cont'd)

Heat transfer (Cont'd)	
Maximum heat flux, 1.25 maximum power, $Btu/hr\text{-}ft^2$	358,860
Peak/average heat flux, 1.25 maximum power	3.59
Burnout power, % maximum power	158
Average fuel power density, maximum power, kw/ft	3.53
Maximum fuel power density, maximum power, kw/ft	10.15
Fuel power density for fuel melting, kw/ft	18.4
Fuel melting power, % maximum power	181
Power density, total active core, kw/ℓ	39.2
Power density, core coolant volume, kw/ℓ	66.9

Control system	
Type of rod	Y-shaped, solid neutron absorbing rod with Zircaloy follower
No. of rods	6
Type drive	Rack & pinion
Core entry	Top
Total reactivity control in control elements, $\delta k/k$	0.265 (including soluble poison)
Excess reactivity, $\delta k/k$	0.235
Initial shutdown margin, $\delta k/k$	0.030

Primary system	
Primary system coolant volumes, ft^3	
Steam generator	175
Reactor	63
Cold Leg - Steam generator outlet to boiler inlet	651
Total cold mass volume	889
Hot Leg - Reactor outlet to steam generator inlet, nominal value	329
Steam volume, nominal value	280
Total primary system	1,498
Primary system coolant masses, nominal, lb	
Water mass	59,765
Steam mass	500
Stored heat, nominal, Btu	32×10^6
Transit times, sec	
Steam generator outlet to reactor inlet	19
Reactor outlet to steam generator inlet	12

Table 8.7-1 (Cont'd)

Nuclear parameters

Moderator coefficient

Moderator temperature, F	$\alpha_M \times 10^4$ ($\delta k/k$-F) Beginning of life	End of life
100	- 0.2	- 0.2
200	- 0.5	- 0.4
300	- 1.0	- 0.8
400	- 1.9	- 1.5
500	- 3.0	- 2.4

Doppler coefficient

	$\alpha_D \times 10^5$ ($\delta k/k$-F)
Beginning of life	- 1.16
End of life	- 1.24

Delayed neutron fractions and decay constants

Delayed group "i"	Beginning of life $\overline{\beta}_{\text{eff}_i}$	$\overline{\lambda}_i$ (sec^{-1})	End of life $\overline{\beta}_{\text{eff}_i}$	$\overline{\lambda}_i$ (sec^{-1})
1	0.00022	0.0125	0.00017	0.0126
2	.00156	0.0306	.00127	0.0305
3	.00145	0.1140	.00114	0.1180
4	.00300	0.3060	.00230	0.3140
5	.00102	1.1580	.00080	1.1520
6	.00036	3.0940	.00029	2.9790
$\sum_{i}^{6} \beta_i$	0.00761	--	0.00597	--

Void coefficient

$\alpha_V = -\ 0.14\%\ \delta k/\%$ Void

Effective neutron lifetime

$\ell^* = 4.1 \times 10^{-5}$ seconds

Table 8.7-2. Summary of Void-Reactivity Studies[a] Preliminary, Core I (100% Power-69 MWt)

Operating pressure	Coolant temperature reactor inlet	Reactor power	Average core void fraction	Average moderator density	δk_{void}	Minimum burnout ratio	Approximate burnout power level
psia	F	% rated power	$\bar{\alpha}$	lb/ft^3		$[q''_{BO}/q''_{(x)}]$ min	% rated power
812	460	103	0----0	50.25	0----0	(b)	
		125	0.0085	49.39	(a)	2.381	175
		180	0.0672	46.14	- 0.0072	0.887	
812	479	72	0----0	49.30	0----0	(b)	
		94	0.0093	48.70	(b)	3.191	162
		125	0.0497	46.58	- 0.0073	1.929	
		156	0.0954	44.39	- 0.0150	1.167	
812	487	58	0----0	49.30	0----0	(b)	
		100[c]	0.0407	46.80	- 0.0064	2.775	158
		125	0.0824	44.71	- 0.0118	1.944	
812	500	36	0----0	48.75	0----0	(b)	
		63	0.0310	46.98	- 0.0049	4.701	150
		125	0.01764	40.00	(b)	1.634	

(a) All values in this table are based on first iteration for void distribution and power distribution (void distribution is based on power distribution with no voids in core: $(P^*/\bar{P})_R = 1.52$, $(P^*/\bar{P})_A = 1.5$, sine).

(b) Not calculated.

(c) Normal operating conditions.

Table 8.7-2 (Cont'd)

Operating pressure	Coolant temperature reactor inlet	Reactor power	Average core void fraction	Average moderator density	δk_{void}	Minimum burnout ratio	Approximate burnout power level
psia	F	% rated power	$\bar{\alpha}$	lb/ft³		$[q''_{BO}/q''_{(x)}]$ min	% rated power
600	440	77	0----0	51.34	0----0	(b)	
		94	0.0085	50.65	(b)	4.023	164
		125	0.0467	48.56	- 0.0068	2.276	
		156	0.0993	45.85	- 0.0147	1.249	
600	460	44	0----0	50.66	0----0	(b)	
		94	0.0784	46.41	- 0.0104	3.292	153
		125	0.1585	42.36	(b)	1.872	
600	479	13	0----0	50.02	0----0	(b)	
		64	0.1754	41.22	- 0.0239	5.163	146
		125	0.3382	33.40	(b)	1.657	
1045	479	127	0----0	49.02	(b)	(b)	
		180	0.0228	47.26	- 0.0043	1.143	186
1045	500	91	0----0	48.31	0----0	(b)	
		150	0.0411	45.66	- 0.0067	1.412	167
1045	518.5	58	0----0	47.39	0----0	(b)	
		100	0.0346	45.36	- 0.0058	2.357	154
		130	0.0821	43.09	- 0.0126	1.593	

(a) All values in this table are based on first iteration for void distribution and power distribution (void distribution is based on power distribution with no voids in core: $(P^*/\bar{P})_R = 1.52$, $(P^*/\bar{P})_A = 1.5$, sine).

(b) Not calculated.

8.7.2.2. Reactor Kinetics Model

The reactor kinetics equations given below assume the neutron flux to be space independent.

$$\dot{n} = \frac{n\delta k}{\ell^*} - \frac{n\beta}{\ell^*} + \sum_{1}^{5} \lambda_i C_i \tag{1}$$

$$\dot{C}_i = \frac{n\beta_i}{\ell^*} - \lambda_i C_i . \tag{2}$$

The fifth and sixth groups of delayed neutrons are lumped to form one equivalent group.

8.7.2.3. Doppler Reactivity Model

$$\delta k_D = \alpha_D \overline{T}_F . \tag{3}$$

At the beginning of life

$$\delta k_D = -(1.16 \times 10^{-5})(\overline{T}_F). \tag{4}$$

At the end of life

$$\delta k_D = -(1.24 \times 10^{-5})(\overline{T}_F). \tag{5}$$

8.7.2.4. Moderator Temperature Reactivity Model

$$\delta k_M = \alpha_M (\overline{T}_R - \overline{T}_{R_{T=0}}) \tag{6}$$

where δk_M is zero at the start of the transient.

$$\alpha_M = f(\overline{T}_R) \approx a_1 + a_2 \overline{T}_R . \tag{7}$$

At the beginning of life

$$\delta k_M = (2.3 - 0.01066\, \overline{T}_R)(10^{-4})(\overline{T}_R - T_{R_{T=0}}) \tag{8}$$

and at the end of life

$$\delta k_M = (2.628 - 0.01013\, \overline{T}_R)(10^{-4})(\overline{T}_R - T_{R_{T=0}}). \tag{9}$$

8.7.2.5. Void Reactivity Model

Since the self-pressurized CNSG has limited bulk boiling in the hotter flow channels, the reactivity is affected by the quality and the distribution of steam voids in the core. The reactivity in voids was calculated for different steady-state operating conditions to provide a basis for predicting reactivity changes during transients[1,2,3]. The results of these calculations are summarized in Table 8.7-2 and are shown in Figures 8.7-6 through 8.7-11. The calculated void coefficient is - 0.14% $\delta k/\%$ void.

The following equation approximates the void reactivity:

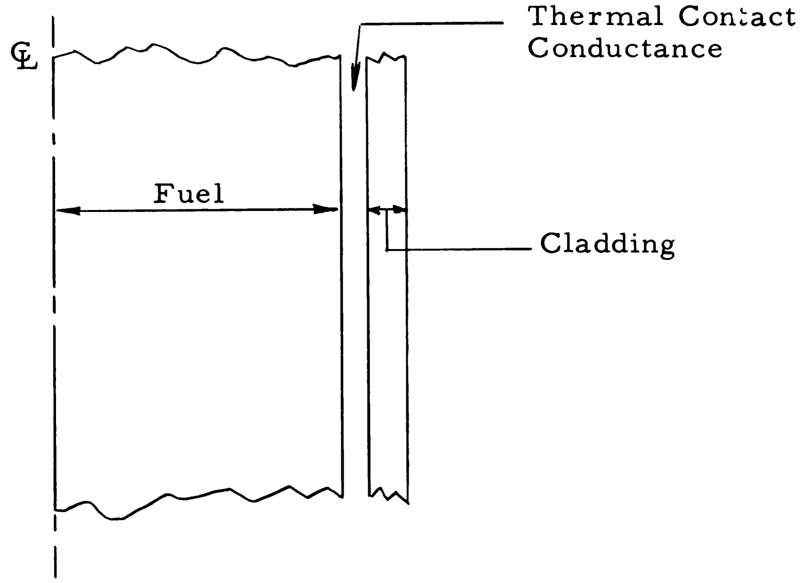
$$-\delta k_V = 0.0233 \frac{Q_P}{Q_{P_{100\%}}} - 0.000524\,(T_{sat} - T_4) - 0.000033\,(T_{sat} - 520)\frac{Q_P}{Q_{P_{100\%}}} \quad (10)$$

The first two terms describe a family of straight lines with the same slope whose intercepts vary with the coolant inlet subcooling (F). The last term is a pressure correction. Changes in primary system pressure are also reflected in the subcooling term. The slope in the void reactivity equation corresponds to the void reactivity versus power curve for 812 psia and 33 F subcooling. The equation is conservative since the CNSG pressure swings are between 812 psia and 1025 psia, and the slope decreases with increasing pressure.

Steady-state void distribution calculations were based on the reactor power distribution with no voids or soluble poison in the core. Several iterations are required to match the void and power distributions. Two iterations were performed to determine the void reactivity at 812 psia and 125% power, and the reactivity value was found to be only about 60% of the value after one iteration. The void reactivity equation is based on the results of one iteration and should provide a conservative model since it overestimates the void reactivity at full power.

8.7.2.6. Reactor Fuel Model

The time rates of change of the average fuel temperature and the thermal power are based on the transient characteristics of a single fuel rod with a heat generation rate equivalent to the core average. Digital computer calculations were made to determine the transient characteristics of an average fuel rod with the fuel region divided into 10 sections and the physical properties variable with temperature. A simplified analog model was developed with approximately the same transient characteristics as the digital computer model. The analog model has three regions: fuel, thermal contact conductance between the fuel and the cladding, and the cladding.



$\bar{T}_F$ - Average fuel temperature

T_{F_o} - Fuel temperature at outside surface

T_{Ci} - Clad temperature at inside surface

$\bar{T}_C$ - Average clad temperature

T_{Co} - Clad temperature at outside surface

Q_F - Reactor heat generation rate (the total reactor heat is assumed to be generated in the fuel)

The fuel temperature equation is

$$\dot{\bar{T}}_F = \frac{Q_F}{(MC_p)_F} - \left(\frac{hA}{MC_p}\right)_F (\bar{T}_F - T_{Fo}) \qquad (11)$$

where h_F is an equivalent heat transfer coefficient for heat transfer across the fuel. The constants were evaluated to obtain good agreement with the digital computer calculations; the fuel time constant is approximately 5 seconds. The heat generation rate in the fuel (Q_F) is assumed to be directly proportional to the average neutron flux.

The heat transfer across the bond between the fuel and the cladding is

$$(hA)_{FC}\,(T_{Fo} - T_{Ci}) = (hA)_F\,(\overline{T}_F - T_{Fo}). \tag{12}$$

The cladding temperature equation is

$$\dot{\overline{T}}_C = \frac{(hA)_{FC}}{(MC_p)_C}\,(T_{Fo} - T_{Ci}) - \left(\frac{KA}{X}\right)_C \left(\frac{1}{MC_p}\right)_C (T_{Ci} - T_{sat}). \tag{13}$$

Local boiling exists at the surface of the fuel rod; therefore, the equation for the clad surface temperature is

$$T_{Co} = T_{sat} + \Delta T_{sat} \approx T_{sat}\,. \tag{14}$$

The power transferred to the coolant is

$$Q_P = \left(\frac{KA}{X}\right)_C \left(T_{Ci} - T_{sat}\right)\,. \tag{15}$$

8.7.2.7. <u>Reactor Coolant Model</u>

The reactor heat balance equation is

$$\dot{\overline{T}}_R = \left(\frac{W}{M_R}\right) T_4 - \left(\frac{W}{M_R}\right) T_1 + \frac{Q_P}{\left(MC_p\right)_R}. \tag{16}$$

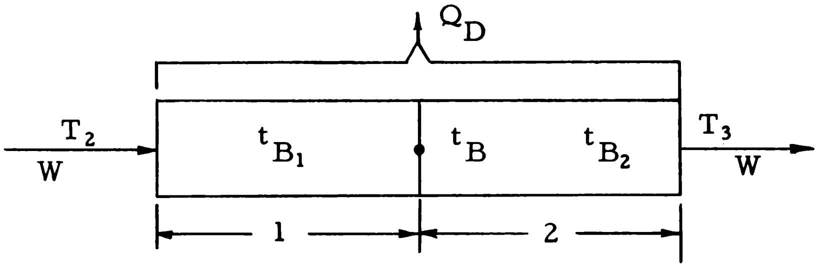
The assumed linear coolant temperature rise is

$$\overline{T}_R = \frac{T_4 + T_1}{2}. \tag{17}$$

The bulk coolant may have a net steam quality at the core exit during a power increase, pressure decrease transient. In this case, the coolant outlet temperature calculated from the reactor heat balance exceeds the saturation temperature. Nuclear calculations to determine the moderator temperature reactivity are based on an outlet temperature equal to the saturation temperature. Therefore, the power increase model includes the following equation for the average reactor coolant temperature which is fed to the moderator temperature reactivity model:

$$\overline{T}_{R_M} = \frac{T_4 + T_{sat}}{2}\,. \tag{18}$$

8.7.2.8. Steam Generated Model



The steam generator model is a two-section simulation of the primary side only, and the load demand (Q_D) is applied directly to the primary side. The equations are given below.

Section 1

$$\dot{t}_{B_1} = \left(\frac{W}{M_{B_1}}\right) T_2 - \left(\frac{W}{M_{B_1}}\right) t_B - \frac{0.5}{(MC_p)_{B_1}} Q_D \tag{19}$$

$$t_{B_1} = \frac{T_2 + t_B}{2} . \tag{20}$$

Section 2

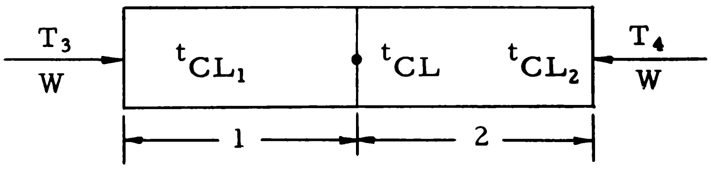
$$\dot{t}_{B_2} = \left(\frac{W}{M_{B_2}}\right) t_B - \left(\frac{W}{M_{B_2}}\right) T_3 - \frac{0.5\, Q_D}{(MC_p)_{B_2}} \tag{21}$$

$$t_{B_2} = \frac{t_B + T_3}{2} . \tag{22}$$

The average coolant temperature in the steam generator is calculated by

$$\overline{T}_B = \frac{t_{B_1} + t_{B_2}}{2} . \tag{23}$$

8.7.2.9. Delay Model, Cold Leg



The transport delay between the steam generator outlet and the reactor inlet is approximately by two first order delays represented by the following equations.

Section 1

$$\dot{t}_{CL_1} = \left(\frac{W}{M_{CL_1}}\right)\left(T_3 - t_{CL}\right) \tag{24}$$

$$t_{CL_1} = \frac{T_3 + t_{CL}}{2} . \tag{25}$$

Section 2

$$\dot{t}_{CL_2} = \left(\frac{W}{M_{CL_2}}\right)\left(t_{CL} - T_4\right) \tag{26}$$

$$t_{CL_2} = \frac{t_{CL} + T_4}{2} . \tag{27}$$

When fast transients are imposed on the system, the delay model introduces spurious oscillations in the reactor inlet temperature (T_4) during the delay time. A comparator circuit was used to hold the reactor inlet temperature constant during this period.

The arithmetic average used to calculate the average cold leg temperature in the power decrease model is

$$\overline{T}_{CL} = \frac{t_{CL_1} + t_{CL_2}}{2} . \tag{28}$$

The integrated average cold leg temperature used in the power increase model is

$$\dot{\overline{T}}_{CL} = \left(\frac{W}{M_{CL}}\right)\left(T_3 - T_4\right) . \tag{29}$$

8.7.2.10. Cold Mass Temperature Model

The average temperature of the cold mass is approximately equal to the volume weighted average of the steam generator, the cold leg, and the reactor average temperatures.

$$\overline{T}_{CM} \approx \left(\frac{V_B}{V_{CM}}\right) \overline{T}_B + \left(\frac{V_{CL}}{V_{CM}}\right) \overline{T}_{CL} + \left(\frac{V_R}{V_{CM}}\right) \overline{T}_R \tag{30}$$

where

$$V_{CM} = V_B + V_{CL} + V_R. \tag{31}$$

8.7.2.11. Pressure Model

The analog simulation of the CNSG has different pressure models for power increase and power decrease transients which allow simplification without invalidating the conclusions. Tests of the NS Savannah pressurizer provide a basis for evaluating the pressure model since the rate and the magnitude of the pressure changes approximate those of the CNSG studies. The pressure decrease model checks exactly with the NS Savannah pressurizer tests, and the pressure increase model predicts the pressure changes to a ± 10% accuracy. The pressure transients calculated by the simplified CNSG pressure model are also in good agreement with the results of analog studies[4] made with a more complex model based on SM-1 pressurizer tests. The major uncertainty in the pressure model is caused by the dissimilarities between the self-pressurized reactor and a pressurizer.

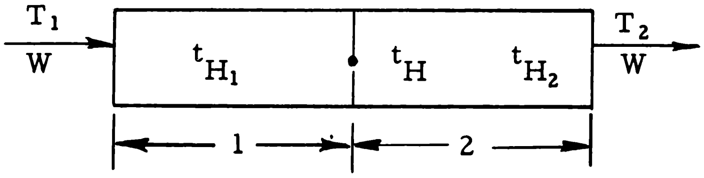
Pressure Model - Power Decrease

When the reactor power is reduced, the average fuel temperature and the core void fraction decrease to cause a positive reactivity change. With no control rod motion, the compensating negative reactivity comes from a moderator temperature rise. During the load initiated transient, the reactor power lags the demand, and the heat input to the primary coolant exceeds the heat removed. The coolant temperature rises, and the resulting expansion compresses the steam in the steam dome. The mass of steam is assumed to remain constant during compression, and the specific volume of the steam is assumed to follow the saturation line. The steam saturation temperature may exceed the water temperature in the hot leg, and the model does not include the heat transfer equations to bring the two temperatures to an equilibrium. Therefore, the pressure model for a power decrease is not valid beyond the peak pressure.

The average coolant temperature in the primary system is calculated as a volume weighted average of the hot and cold mass temperatures:

$$\bar{T}_{sys} = \left(\frac{V_H}{V_H + V_{CM}}\right)\bar{T}_H + \left(\frac{V_{CM}}{V_H + V_{CM}}\right)\bar{T}_{CM}. \tag{32}$$

The hot leg temperature is calculated by the equations for the transport delay between the reactor outlet and the steam generator inlet.



The transport delay is approximated by two first order delays represented by the following equations.

Section 1

$$\dot{t}_{H_1} = \left(\frac{W}{M_{H_1}}\right)\left(T_1 - t_H\right) \tag{33}$$

$$t_{H_1} = \frac{T_1 + t_H}{2}. \tag{34}$$

Section 2

$$\dot{t}_{H_2} = \left(\frac{W}{M_{H_2}}\right)\left(t_H - T_2\right) \tag{35}$$

$$t_{H_2} = \frac{t_H + T_2}{2}. \tag{36}$$

The hot leg temperature, assumed to be an arithmetic average, is

$$\bar{T}_H = \frac{t_{H_1} + t_{H_2}}{2}. \tag{37}$$

The pressure equations are

$$v_f = 0.00665 + 0.0275 \frac{\overline{T}_{sys}}{10^3} \tag{38}$$

$$V_g = V_{total} - M_f v_f = 1498 - M_f v_f \tag{39}$$

$$v_g = \frac{M_g}{V_g} . \tag{40}$$

The steam mass (M_g) is assumed to be constant and is dependent on the initial steam volume. The saturation temperature and pressure equations (linear approximations between 520 and 545 F) are almost exact:

$$T_{sat} = 641.2 - 216.6 v_g \tag{41}$$

$$P_{sat} = - 3222.8 + 7.76 T_{sat}. \tag{42}$$

Pressure Model - Power Increase

As the reactor power increases, the average fuel temperature and the average void fraction also increase and effect a negative reactivity change. The compensating positive reactivity results from a decrease in the self-regulating reactor coolant temperature. For a load initiated transient, the reactor power lags the demand and the heat removed exceeds the heat input to the primary system; consequently, the coolant temperature and the pressure decrease. The coolant temperature in the hot leg is calculated from an overall heat balance on the primary system, the steam saturation temperature is assumed to equal the hot leg temperature. This assumption implies that, if the expansion process tends to drop the steam temperature below the hot leg temperature, flashing will occur to maintain temperature equilibrium.

The primary system heat balance equation is

$$q_t = q_{t_o} + \Delta q_t = M_f h_f + M_g h_g . \tag{43}$$

$M_g h_g$ is less than 2% of the total stored heat and is assumed to be constant. M_f is also assumed to be constant since the mass of water flashed to steam is small and can be neglected. Therefore,

$$\dot{q}_t = (\dot{q}_{t_o} + \Delta q_t) = \Delta\dot{q}_t = Q_D - Q_P \approx M_f \dot{h}_f \qquad (44)$$

where $h_f = 1.235\,\overline{T}_{sys} - 130.3$ (linear approximation, 490 F to 540 F). (45)
$\overline{T}_{sys}$ is given by Equation (32).

Equations (44), (45), and (32) are solved for the hot leg temperature which equals the steam saturation temperature:

$$T_H = T_{sat} = C_1 + C_2\,\Delta q_t + C_3\,\overline{T}_{CM} \qquad (46)$$

where C_1, C_2, and C_3 vary with the initial steam volume. The pressure is calculated from Equation (42).

8.7.3. Results

Analog studies were made for changes in load demand between 20 and 100% rated power. All analog runs were made with no control rod motion so the transient response is determined by the interaction of void, Doppler, and moderator temperature reactivities. The maximum design maneuvering rate for a load increase is from 20 to 100% power in 80 seconds; a turbine trip (100 to 20% step) is the most severe load decrease transient. In addition to the expected operational transients, abnormal conditions were imposed on the CNSG simulator by varying key parameters. (See Table 8.7-3.)

Table 8.7-3. Parameter Variations for Analog Studies.

Parameter	Values studied
Power range	20% to 100%, 100% to 20%
Transient time, sec	Step, 20, 40, 80
Steam volume (at rated power), ft^3	230, 280, 330
Void reactivity	Nominal, 1/3 nominal, 2 × nominal, 3 × nominal, 1/3 nominal (+ void coefficient)
Moderator temperature reactivity	Nominal BOL, nominal EOL, 1/3 nominal EOL
Doppler reactivity	Nominal BOL, nominal EOL
Time constant on load demand, sec	0, 5, 10, 20
Reactivity change caused by vertical acceleration, %	+ 0.087, − 0.217, (0.152 sine $\omega\tau$ − 0.065)

8.7.3.1. Effects of Transient Time

General

Figures 8.7-12 through 8.7-17 show the effects of variable transient times (rate of load change). The nominal end of life nuclear parameters and a 280-ft^3 steam volume at 100% power (corresponds to 210 ft^3 at 20% power) were used in the analog model.

Figures 8.7-16 and 8.7-17 show that the pressure rise is immediate, but the pressure decrease is delayed for a time period equal to the transport delay between the steam generator and the hot leg. As the load demand drops, the steam generator coolant temperature rises and the water expands. This causes an immediate compression of the steam bubble. The process is reversed for a pressure decrease, but it is assumed that as the steam expands, flashing occurs to maintain the steam temperature in equilibrium with the hot leg temperature. The mass of water flashed and the cooling effects are small so the hot leg temperature and the pressure do not drop significantly until the cooling effects in the steam generator are transported to the hot leg ($\approx$19 sec).

Power Decrease Transient Time

In a power decrease transient, the immediate pressure rise causes a collapse of voids and a positive reactivity change. The reactor power rises until the hotter water from the steam generator reaches the reactor to cause negative moderator temperature and void reactivity changes. The void reactivity effect becomes positive again as the reactor power decreases. The pressure rise rate depends on the rate of the load change. Therefore, the initial power rise caused by void collapse is greater for short transient times. For a 100 to 20% step change in the load demand, the average neutron flux rises to 120% and the thermal power rises to 112% rated power. For an 80 second transient, the neutron flux overshoot is only 4%.

The transient time does not affect the peak pressure, but it does affect the time at which it occurs. The pressure peaks at 1025 psia in about 2 minutes for a step change and in about 3 minutes for the 80 second transient. The calculated equilibrium pressure at 20% power is 980 psia, but the analog model does not include the heat transfer equations to lower the steam temperature. (See Section 8.7.2.11.)

Power Increase Transient Time

The reactor power does not change in a power increase transient until the colder water from the steam generator reaches the reactor. The cold water causes a positive change in moderator temperature reactivity and an initial positive change in void reactivity. As the reactor power rises, the Doppler and void reactivity changes are negative, and the moderator temperature reactivity change is positive. The initial reactor power rise increases as transient times become shorter, but even for a step change, there is no power overshoot.

The magnitude of the downward pressure swing is independent of the transient time, but the rate of change of pressure is slightly greater for the rapid transients. The pressure decreases from 980 to 845 psi in about three minutes for a step change and in about four minutes for an 80 second transient.

8.7.3.2. Effects of Steam Volume

Power Decrease Transients

Figure 8.7-18 shows that the peak pressure increases as the initial steam volume decreases. The surge volume is about 64 ft^3 for each of the cases shown, but the mass of steam compressed varies with the steam volume. The lower limit on the peak pressure corresponds to the hot leg temperature when a steady-state condition is reached. For a 330-ft^3 initial steam volume, the steam saturation temperature equals the hot leg temperature at the end of the transient. Therefore, a further increase in steam volume would not reduce the peak pressure. If the steam volume is too small, the peak pressure may be too large to justify the additional vessel thickness required. The optimum steam volume for a self-regulated reactor is a compromise between space, weight, and cost factors.

As the steam volume decreases, there is an increase in the pressure rate of change and in the power overshoot caused by pressure collapse of the voids. The steam volume effects on the reactor power for a 100 to 20% load step are shown in Table 8.7-4.

Table 8.7-4. Effects of Steam Volume - Power Decrease 100 to 20% Step

Steam volume, ft^3	230	280	330
Peak pressure, psia	1085	1025	980
Peak neutron flux, %	124	120	116
Peak thermal power, %	116	112	109
Initial average fuel temperature rise, F	70	55	40

Power Increase Transients

Figure 8.7-19 shows that the initial steam volume does not have a large effect on the downward pressure swing during a power increase. The mass of water in the primary system is decreased as the initial steam volume is increased. Therefore, for the same heat removal, the hot leg temperature and the pressure decrease more for the larger steam volumes. This effect is small since the total water mass in the primary system is about 60,000 lb, and the mass change corresponding to a 50-ft^3 change in steam volume is a small fraction of the total. The steam volume does not have a significant effect on the reactor power for an increase in the load demand.

8.7.3.3. Effects of Void Reactivity

Initial analog computer runs were made with the void reactivity model described in Section 8.7.2.5. Additional runs to study the effects of voids were made with the coefficients of the void reactivity equation multiplied by factors of zero, two, three, one-third, and minus one-third (positive void coefficient). These studies show that the feasibility of this concept is not limited to a narrow range of void reactivity coefficients.

Power Decrease Transients

The analog computer runs shown in Figures 8.7-12 through 8.7-19 were made with the nominal void reactivity model. Figure 8.7-20 shows the primary system response to a turbine trip with three times the nominal void reactivity. The effects of the void reactivity on reactor power for a turbine trip (100 to 20% power step) are summarized in Table 8.7-5.

Table 8.7-5. Effects of Void Reactivity - Power Decrease (Nominal δk_V = - 0.60% at 100% Power)

Void reactivity	Nominal	2 × Nominal	3 × Nominal	Zero	1/3 Nominal (+)
Peak neutron flux, %	120	124	124	100	100
Peak thermal power, %	112	116	116	100	100
Initial rise of average fuel temperature, F	55	70	70	0	0

Figure 8.7-21 shows the effects of void reactivity on operating pressure for a turbine trip and no control rod motion. As the negative reactivity in voids becomes larger, the moderator must take a greater temperature change to compensate. Also, the pressure swing is greater. The moderator temperature reactivity coefficient becomes stronger as the moderator temperature increases. This helps to reduce the pressure swing for the large values of void reactivity. For a positive void coefficient, the void and the moderator temperature reactivities act in the same direction to compensate for the Doppler reactivity change, and the pressure swing is small.

During a power decrease transient, the pressure change and the void-induced power overshoot become smaller as the void reactivity coefficient becomes less negative. As a consequence of the lower value of the peak flux, the minimum neutron flux reaches a lower level. The analog studies show that it is possible for a reactor with a strong negative moderator temperature coefficient and a weak negative or a positive void reactivity coefficient to shut down following a turbine trip. Shutdown following a turbine trip can be avoided by using a control system or by properly selecting design parameters for a self-regulating reactor.

Power Increase Transients

Figure 8.7-22 illustrates the primary system response to a step increase in load from 20 to 100% with no control rod motion and three times the nominal void reactivity. The increased void reactivity has no adverse effects on the reactor power.

Figure 8.7-23 shows the changes in the primary system pressure for the different values of void reactivity. When the void reactivity is greater than nominal, the moderator temperature must decrease to compensate for the additional negative reactivity, and the downward pressure swing is greater. Large pressure swings can be avoided by proper design.

Even though extreme variations in the void reactivity coefficient were used in the analog runs, the CNSG exhibited stable operating characteristics for all the transients imposed.

8.7.3.4. Effects of Moderator Temperature Reactivity

For a load initiated transient, the normal void and Doppler reactivity changes are in the same direction; and with no control rod motion, the compensating reactivity results from a change in moderator temperature. A strong moderator temperature coefficient reduces the temperature change required, and the pressure swing is less. Figure 8.7-24 shows the effect of a weak moderator temperature coefficient. For this analog computer run, the moderator temperature and the void reactivities have one-third of their nominal values; and, even with the small void reactivity, the pressure change is about 300 psi. Other analog runs with varying moderator coefficients also showed the desirability of a strong, negative moderator temperature coefficient in the self-regulating reactor.

8.7.3.5. Effects of Doppler Reactivity

The values of the Doppler coefficient at the beginning of life ($- 1.16 \times 10^5$ $\delta k/k$-F) and at the end of life ($- 1.24 \times 10^{-5}$) were used in the analog computer runs. The reactivity change is approximately the same for both values (~ 0.35%) since the average fuel temperature changes only about 300 F between ~0 and 100% power. Since the Doppler reactivity opposes a power change, some mechanism of reactivity compensation must be used during each power change. Control rods may be moved, but a self-regulating reactor utilizes the change in moderator temperature which makes a small Doppler deficit desirable.

8.7.3.6. Effects of Secondary System

The analog simulation imposes the load demand directly on the primary side of the steam generator. Because the secondary system serves as a buffer between the turbine and the primary system, the analog simulates more severe conditions than can be realized in the physical system. Analog runs were made with different time constants on the load demand to approximate the attenuation effect of the secondary system. Figure 8.7-25 is a typical run with a 10-second time constant on the load demand, and Figures 8.7-26 and 8.7-27 show the the effects of different time constants on operating pressure. The stored heat in the primary system is so large that a short delay on the demand does not have a significant effect on the primary system transient response.

Analog studies of the once-through steam generator verify that the simplified steam generator model used in the primary system studies well represents the steam generator outlet temperature as a function of time.

These studies indicate that the basic transient characteristics of the primary system are not greatly influenced by the secondary system.

8.7.3.7. Effects of Ship's Motion

Preliminary studies were made to determine the effects of ship's motion on the performance of the CNSG.[5,6,7] These studies concluded that with forced circulation, the bulk flow rate would vary only 1.5% for ± 0.8-g accelerations. Therefore, the reactor flow rate is assumed to be constant (6×10^6 lb/hr) for the analog studies.

Steady-state calculations of the core flow distribution for different values of apparent gravity show that vertical accelerations may cause significant changes in flow and void distributions in the core. These calculations do not account for power redistribution with changes in void distribution or for the reactor power level change with changes in void reactivity. Based on the calculated void distributions at 100% power, the void reactivity changes associated with vertical accelerations were conservatively estimated to be + 0.087% for + 0.6 g and - 0.217% for - 0.6 g. These reactivity variations were programmed as a sine input to the reactor kinetics model:

$$1000\ \delta k_g = 1.52 \text{ sine } \omega\tau - 0.65. \tag{47}$$

The transient response of the reactor to this driving function includes all of the feedback effects that relate power to Doppler, void, and moderator temperature reactivities.

The maximum vertical accelerations of a reactor on a ship powered by the CNSG would probably occur on a period of 6 to 10 seconds (depending on ballast) with a duration of not more than three successive cycles.[8] Analog runs were made with the sine input period varying from 2 to 30 seconds. The results of the analog studies are illustrated in Figure 8.7-28 and are summarized in Table 8.7-6.

Table 8.7-6. Effects of Ships's Motion (Sine Input with a Period of from 2 to 30 seconds)

Parameter	Maximum variations	
δk_g (driving function), %	+ 0.087	- 0.217
δk_{total}, %	+ 0.150	- 0.25
Average neutron flux, %	+ 40	- 30
Thermal power, %	+ 4	- 8
Average fuel temperature, F	+ 10	- 35
Pressure, psi	------	- 10

The void reactivity model has a 0.5-second time constant which corresponds to the void sweep time. The phase difference between the void reactivity feedback and the sine driving function causes a total reactivity change which exceeds the input for the shorter periods. Although the reactivity change is greater in the negative direction, the neutron flux swing is greater in the positive direction because the period for the flux rise is shorter than that for the decay. The duration of the positive swing is less than that of the negative swing, and the integrated power is negative as indicated by the thermal parameters.

The ship's motion studies show no evidence of resonant conditions or of tendencies toward instability even though the acceleration effects and the period range have been treated conservatively.

8.7.4. Analysis of Results

8.7.4.1. Evaluation of Analog Model

The results of the analog computations can be no better than the computer model. The component models of the CNSG are confidently simulated with the possible exception of the void and the pressure models. The pressure model has been checked against pressurizer test results, and it accurately predicts the pressure swings. The use of a nuclear heat source with instantaneous pressure-feedback effects to maintain the vessel pressure subjects the model to some uncertainty. The void model is based on analytical methods successfully used in the design of other operating water reactors. The major uncertainties in this model are associated with the effects of the self-pressurization on the steam voids in the core. The amount of reactivity in the voids is small, and the analog studies show that large variations in the void reactivity do not cause any serious operational problems.

The analog model is considered to be an adequate simulation of the CNSG, and the basic operational characteristics demonstrated are valid to establish the attractiveness of the concept and the preliminary design feasibility.

8.7.4.2. Comparison of Preliminary CNSG and Reference CNSG

The analog computer studies were made with the preliminary CNSG design parameters. This design has been optimized and improved. The design parameters having a major effect on the transient performance are compared in Table 8.7-7. The preliminary and reference core arrangements and physical dimensions are basically the same. Soluble poison was used in the nuclear calculations for the reference core so the nuclear constants are different (except for the delayed neutron fractions and decay constants). There is a significant difference in the mass and volume of the primary system.

The reference CNSG has a nominal steam volume of 190 ft^3 and a water volume of 505 ft^3 compared with 280 and 1218 ft^3 for the preliminary design. The calculated maximum pressure swing for the reference CNSG with no control rod motion is from 812 psia to 1060 psia compared with 812 psia to 1025 psia for the preliminary design. The larger pressure swing in the reference design is primarily a result of the weaker moderator temperature reactivity coefficient.

The coolant transport times between the steam generator and the reactor are about one-half as long in the reference CNSG as in the preliminary CNSG. The effects of changes in the load demand will be effective in the reactor in a shorter time, and the reference reactor power will follow the load more closely than will the preliminary reactor.

The nuclear constants do not differ greatly for the two designs. The reference core has a weaker moderator temperature coefficient, but the weaker void coefficient is partially compensating. The effects of voids on the transient performance of the reference reactor will be less than on the preliminary reactor because of the smaller values of the void deficit, the void coefficient, and the reactivity changes caused by vertical accelerations.

This comparison leads to the conclusion that the transient characteristics of the two designs are basically similar, and the results of the preliminary CNSG analog studies can be qualitatively related to the reference CNSG.

Table 8.7-7. Comparison of Key Parameters Preliminary and Reference CNSG

Parameter	Preliminary	Reference
Reactor power, MWt	69	62.4
Operating pressure, psia	812	812
Reactor flow rate, lb/hr	6×10^6	6×10^6
Coolant outlet temperature, F	520	520
Coolant inlet temperature, F	487	490
Fuel	UO_2	UO_2
Cladding	Zircaloy	Zircaloy
Fuel rod OD, in.	0.460	0.460
Primary system coolant volumes, ft^3 (approximate)		
Steam generator	175	80
Cold leg	651	375
Reactor (includes volume between reactor and 1st thermal shield)	63	60
Hot leg, nominal	329	180
Steam volume, nominal	280	190
Total primary system	1498	695
Primary system coolant masses, nominal, lb		
Water mass	59,765	34,100
Steam mass	500	340
Primary system stored heat, nominal, Btu	32×10^6	18×10^6

Table 8.7-7 (Cont'd)

Parameter	Preliminary	Reference
Transit times, sec		
Steam generator outlet to reactor inlet	19	10
Reactor outlet to steam generator inlet	12	6
Nuclear parameters		
Moderator coefficient at 500 F, $\delta k/k\text{-}F \times 10^4$		
Beginning of life	- 3.0	- 2.2
End of life	- 2.4	- 2.0
Doppler coefficient, $\delta k/k\text{-}F \times 10^5$		
Beginning of life	- 1.16	- 1.30
End of life	- 1.24	- 1.20
Void coefficient, % δk/% void	- 0.14	- 0.06
Reactivity in voids, %		
1 g	- 0.60	- 0.23
1.6 g	- 0.51	- 0.18
0.4 g	- 0.82	- 0.40

8.7.4.3. Stability

The CNSG simulator without the benefit of a control system, was subjected to a variety of severe abnormal operating conditions. The natural response of the reactor provided stable operation for all situations, and there was no evidence of instability. It is concluded that the CNSG has a high degree of inherent stability and that it is capable of meeting the maximum requirements for ship's maneuvering with or without a control system.

8.7.4.4. Control

The excellent load-following characteristics of the CNSG allow a variety of control schemes to be considered. The preliminary studies indicate that it may be possible to operate a modified CNSG over the entire life without the use of control rods. Further analysis is necessary to establish the practicality of the complete elimination of control rods since such studies are outside the scope of the work reported here. If complete self-regulation can be achieved, the costs and complexities of control rods and drives can be eliminated as a major stride toward simple and economic nuclear power for ship's propulsion.

The reference design of the CNSG has seven control rods. A constant pressure control system takes maximum advantage of these rods by reducing the pressure swing and power overshoots caused by pressure changes. This control method was selected for the reference CNSG. (See Section 4.8.) The control system has not been studied on the analog computer, but on the basis of the self-regulating control studies, a constant pressure control system should provide simple and effective power regulation.

The reactor may be allowed to follow the load demand without control rod motion and then the reactor may be manually shimmed to the desired steady state. The reference design will allow this control flexibility as a step toward proving self-regulation.

8.7 4.5. Overpower Scrams

Ship's motion studies show that the neutron flux may vary appreciably with maximum vertical accelerations of the reactor. The use of an instantaneous flux scram set for 125% might initiate spurious shutdowns. Therefore, a flux integrating scram circuit is required for the CNSG.

The analog studies show that the reactor power exceeds 100% for a turbine trip and for maximum vertical accelerations. For the nominal preliminary parameters, the thermal power overshoots to 112%, and the average fuel temperature rises 50 F when the turbine trips. The maximum vertical accelerations cause a peak thermal power of 104% and an average fuel temperature rise of 10 F. Since void reactivity effects are less in the reference CNSG than in the preliminary CNSG, the power overshoots should not be as large as those in the analog studies. The CNSG burnout and fuel melting safety margins are adequate to insure that fuel rod damage does not occur during a transient power overshoot.

8.7.5. Conclusions

The conclusions of the preliminary design analog computer studies are listed here.

1. The analog simulation demonstrates the basic operational characteristics of the CNSG, and engineering confidence in the concept is established. There are some uncertainties in the pressure model and the void model of the simulation, but these uncertainties may be removed by a nominal amount of experimental backup.

2. The CNSG has a high degree of inherent stability and can be made to satisfy the maximum design maneuvering requirements with or without a control system.

3. The CNSG transient behavior depends on the interaction of the Doppler, the void, and the moderator temperature reactivity coefficients. The analog studies demonstrate that stable operation and the self-regulating properties are possible with a wide range of values for each of the reactivity coefficients. Optimum performance is achieved by proper design.

4. Ship's motion has some effect on the performance of the CNSG but it does not impose serious operational problems. A neutron-flux integrating scram circuit is required to prevent spurious shutdowns in rough seas.

5. Power overshoots during reactor transients are not severe and will not cause fuel element damage.

6. The basic transient characteristics of the primary system are not greatly influenced by the secondary system.

7. A constant pressure control system will take maximum advantage of the CNSG control rods and will provide effective power regulation.

8. Complete elimination of the control rods in the CNSG appears possible and is a goal worthy of further study.

List of Symbols for Analog Studies

A - Heat transfer area, ft^2

C_i - Concentration of the delayed neutrons emitted in group i

C_p - Constant pressure specific heat, Btu/lb-F

g - Acceleration due to gravity, ft/sec^2

h - Heat transfer coefficient, $Btu/sec\text{-}ft^2$

h_f - Enthalpy of water, Btu/lb

h_g - Enthalpy of steam, Btu/lb

K - Thermal conductivity, Btu/sec-ft-F

ℓ^* - Mean effective neutron lifetime, sec

M - Mass, lb

M_f - Mass of water in primary system, lb

M_g - Mass of steam in primary system, lb

n - Average thermal neutron flux, $neut/cm^2\text{-}sec$

P - Pressure, psi

Q_D - Load demand, Btu/sec

Q_F - Heat generation rate in fuel, Btu/sec

Q_P - Thermal power, Btu/sec

q_t - Total stored heat in primary system, Btu

T - Temperature, F

t - Temperature, F

V - Volume, ft^3

v_f - Specific volume of water, ft^3/lb

v_g - Specific volume of saturated steam, ft^3/lb

List of Symbols (Cont'd)

W - Coolant flow rate, lb/sec

X - Thickness, ft

α - Reactivity coefficient

β_i - Fraction of delayed neutrons in group i

δk - Reactivity

λ_i - Decay constant for i^{th} group of delayed neutrons

τ - Time, sec

ω - Frequency, radians/sec

 - Derivative with respect to time

Subscripts

B - Boiler (steam generator)

C - Cladding

CL - Cold leg, boiler outlet to reactor inlet

CM - Cold mass, coolant mass in boiler, reactor, and cold leg

D - Doppler

F - Fuel

FC - Thermal contact conductance between fuel and clad

H - Hot leg, reactor outlet to boiler inlet

M - Moderator temperature

R - Reactor

sat - Saturation

sys - Primary system water

8.8. Materials

8.8.1. Introduction

Structural materials within the CNSG will be exposed to one or more of three different environments denoted as the primary, secondary, and containment systems.

The primary system will contain neutral or lower pH water with a soluble poison addition. Since the CNSG is self-pressurizing, part of the environment will be in the vapor phase. Therefore, some materials will be exposed to a vapor-liquid interface.

The secondary system will contain pure neutral or higher pH water heated to the vapor phase, thus creating a vapor-liquid interface within the tubes of the steam generator.

The containment system will contain above neutral water (high pH) with a soluble poison addition. The containment water will be cooled by using a sea water heat exchanger. The possibility of sea water leakage from the heat exchanger creates a potential corrosion problem for the containment system materials.

8.8.2. Recommended Materials

8.8.2.1. Primary System

Zircaloy-2 is recommended for the fuel cladding and core structure where a low neutron absorption cross section is required. Based on the knowledge of its physical and nuclear properties and its extensive operating experience, this material is satisfactory for the present service conditions. However, a Zircaloy-2 spacer grid used to assemble the fuel elements mechanically will require development of a practical means of joining the grid parts. An alternate, though less desirable material for this application, would be austenitic stainless steel. In either case, the effects of fretting between the fuel rods and spacer grid would have to be determined.

Generally, structural materials in contact with primary water will be austenitic stainless steel. Components requiring higher strength or greater resistance to galling will be made of materials exhibiting corrosion resistance similar to that of austenitic stainless steel.

8.8.2.2. Secondary System

Generally, carbon steel will be used in the secondary system. However, the steam generator (heat exchanger) will require stainless-steel or nickel alloy tubes. The higher alloy steam generator tubing is necessary since the material will also be in contact with primary water and may require periodic cleaning to remove sludge deposited by boiling in the tube.

8.8.2.3. Containment System

Carbon steel is recommended as the principal structural material in the containment system. Since containment water chemistry has not been established, it may be necessary to corrosion test carbon steel and other containment materials (including copper-nickel, stainless steel, lead and suitable non-metallic coatings) to determine their compatibility within the system environment.

8.8.3. Recommended Testing

The special problems discussed in Section 8.8.2.3 indicate that considerable testing of materials and development work will be required before the final recommendation of certain structural materials can be made for their application within the CNSG. Therefore, it is recommended that the following studies and tests be considered.

1. Determine the effect of the vapor-liquid interface of the primary environment on materials such as shafting and piping.
2. Determine the compatibility of materials with the containment environment.
3. Develop a zirconium alloy spacer grid.
4. Determine the effect of fretting between Zircaloy-2-clad fuel rods and the spacer grid.

8.9. Considerations Concerning Prototype Construction

8.9.1. Introduction

Because of the following considerations, B&W concludes that no prototype CNSG should be built prior to the first shipboard installation. This conclusion considers the unknown factors and assumes a proper balance among the various factors involved.

In this study, the CNSG prototype is assumed to be a full sized (85 MWh) plant which will be tested at rated heat output. The prototype is a complete steam generator having a condenser and associated equipment required for dissipation of all of the heat generated. This choice was made in belief that problems encountered in the prototype will involve details of equipment such as that experienced with the Savannah, where, for example, the details of control rod drives caused trouble. Further, flow distribution in the core and steam generator can not be tested in any other way since heat generation and detailed geometry influence flow distribution. However, no problems in flow distribution are expected. Since considerable time and expense are required for a full-sized prototype design, this design should serve also for the ship.

8.9.2. Probability of Success

Since the CNSG is firmly based on previous water reactor technology, the consideration of limiting the investment in case of project failure does not apply.

8.9.3. Usefulness of the Product

If the nuclear powered ship proved to be uneconomical after construction and testing, it might be desirable to build the prototype. As is shown in Section 6, however, a CNSG-powered high-speed dry cargo ship is economical.

8.9.4. Technical Considerations

A prototype, of course, cannot provide the complete test that a ship installation can since it is impossible to duplicate ship's motion and vibratory problems. Also, actual service conditions often reveal problems not found in the laboratory.

The problems (leaking gaskets, etc.) revealed in the startup of the Savannah would not have been significantly changed by previous experience with a prototype. Many of these problems could have been avoided by a closer inspection of the critical components in the vendors plant.

A major problem is to satisfy the requirement of the regulatory body, which always insists on complete tests on board ship even though identical tests may have been run on a prototype.

The entire primary system of the CNSG, including control rod drives and associated equipment, can be assembled in the shop and tested. However, the primary system must be operated at room temperature, and no nuclear tests can be run in the shop. Even with these limitations it appears that most flaws could be worked out before shipment.

8.9.5. Financial Considerations

Financial considerations in the construction of a prototype or a ship are complicated by several parties with distinct interests as outlined below.

1. The AEC - The AEC, charged with developing peaceful uses for atomic energy, is interested in advancing the technology of atomic energy to the point where outside parties can use it.

2. The Maritime Administration - The Maritime Administration is charged with promoting the status of the American Merchant Marine, developing new technology for the maritime industry, and providing the necessary subsidies to permit competition. These subsidies are to assure the construction and operation of an American Merchant Marine large enough to carry out government purposes.

3. The Ship Owner - The owner's interest lies in transporting goods at a profit.

4. Power Plant Manufacturer - The power plant manufacturer's interest lies in producing goods and services to fit the needs of his customers.

The approximate costs in this section are taken, when available, from Section 5 of this report. Costs for the first ship, presented below, are divided among the various parties according to an assumed financial scheme.

Case I. Ship Construction

Non-capitalized costs - AEC operating	$ 4,500,000
Charter and operation during tests - AEC operating	2,000,000
Construction subsidy - Maritime Administration	10,000,000
Total government expense	$16,500,000
Cost of ship to owner	$10,000,000

These costs assume that the AEC pays the non-capitalized costs and charters the ship for one year to perform tests. Charter cost is based on paying for all operating expenses, unsubsidized, plus interest and depreciation based on the cost of a 50%-subsidized ship to the owner. As shown in the reference design, this arrangement is favorable to the ship owner and is consistent with present government policy for building merchant ships. The fuel cost is also less since the power plant will be operated during only part of the testing.

Case II. Land Prototype

Non-capitalized costs - AEC operating	$ 4,500,000
Prototype power plant - AEC construction	5,000,000
Supporting facilities - AEC construction	1,000,000
Operation of facility - AEC operating	1,000,000
Subsidy for a replacement conventional ship - Maritime Administration	7,750,000
Total government expense	$19,250,000

These costs were first estimated on the basis that the land prototype would be abandoned after having served its purpose. The cost of subsidizing the conventional ship is included so that Case II will compare as closely as possible with Case I. This is in the interest of the Maritime Administration since one of their functions is to provide a Merchant Marine large enough for government purposes. Under these ground rules a land prototype is less desirable because it (1) requires larger government expenditure for a comparable product, (2) involves a less complete test, (3) provides no advancement of the art, and (4) gives the operator a conventional ship.

Case III. Land Prototype with Reusable Components

Non-capitalized costs - AEC operation	$ 4,500,000
Prototype power plant - AEC construction	5,000,000
Supporting facilities - AEC construction	1,000,000
Operation of facility - AEC operating	1,000,000
Dismantling & transportation - Maritime Administration ?	1,000,000
Re-erection in ship - Maritime Administration	500,000*
Subsidy of remainder of ship - Maritime Administration	7,500,000
Total government expense	$ 20,500,000
Cost of ship to owner	$ 10,000,000

*Half of this cost is paid by the ship owner.

These costs, based on the possibility of reusing the components of the prototype in a nuclear ship, exceed those for the construction of the ship without a prototype.

8.9.6. Schedule Considerations

The construction time for a first ship is close to that for a prototype. In both cases the steam generating plant is identical. The first ship requires additional construction time for propulsion plant hull, and outfit, but these items should not delay fabrication and erection of the steam generator if care is taken in scheduling.

The steam generator for the first ship will undergo the same testing prior to initial operation as will that for the prototype. Thus, the elapsed time from initiation of the project to operation will be approximately the same. At that time the first-ship plant will have begun to demonstrate operation under service conditions, but the prototype will be operating under simulated conditions.

8.9.7. Conclusion

The construction of a CNSG in a ship offers certain advantages over the construction of a land prototype:

1. It provides a more thorough test.
2. It reduces the over-all cost.
3. It reduces the time required to reach operational status for the American Merchant Marine.

Figure 8.1-1. Volume of Water in Reactor Vessel Vs Saturation

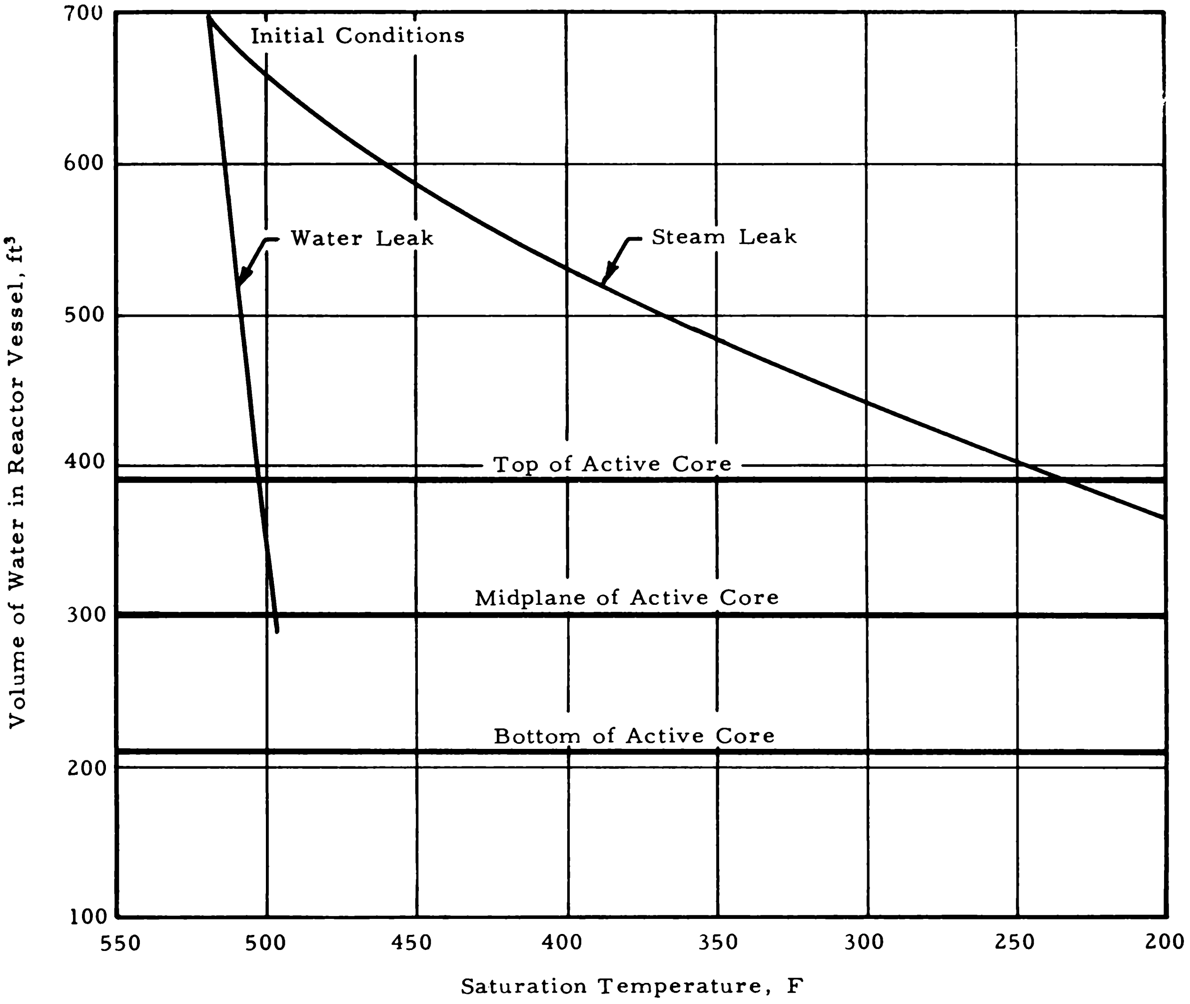


Figure 8.1-2. Vapor Suppression Apparatus

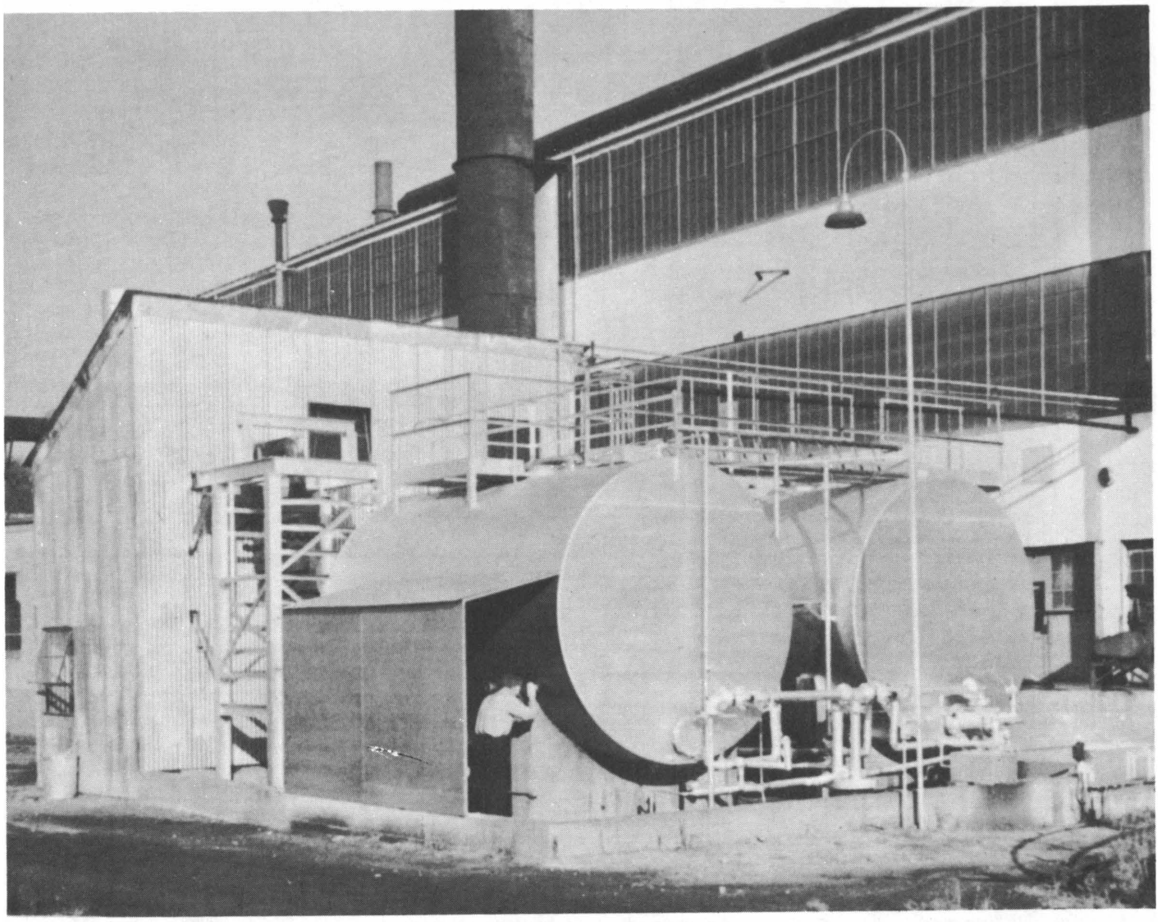


Figure 8.1-3. Vapor Suppression Apparatus (Schematic)

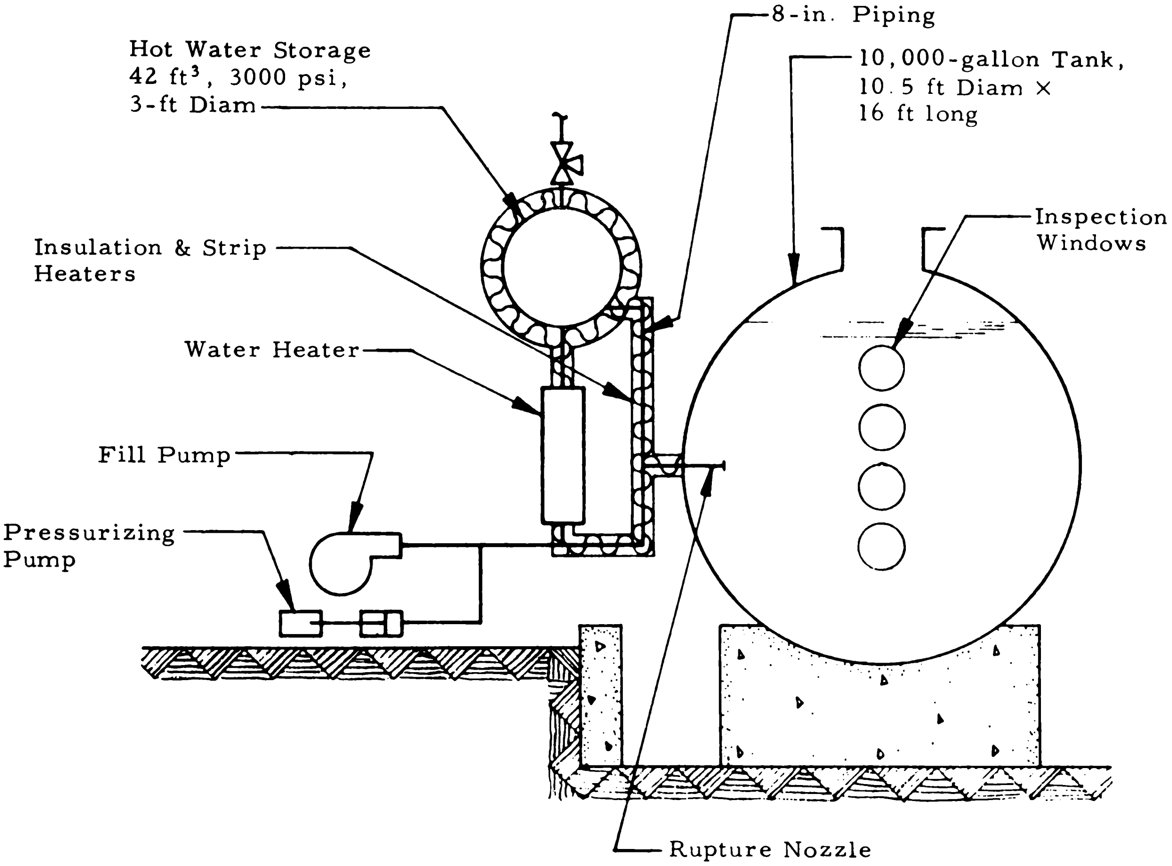


Figure 8.2-1. Dose Rate Over Core Through Water, Six Hours after Shutdown

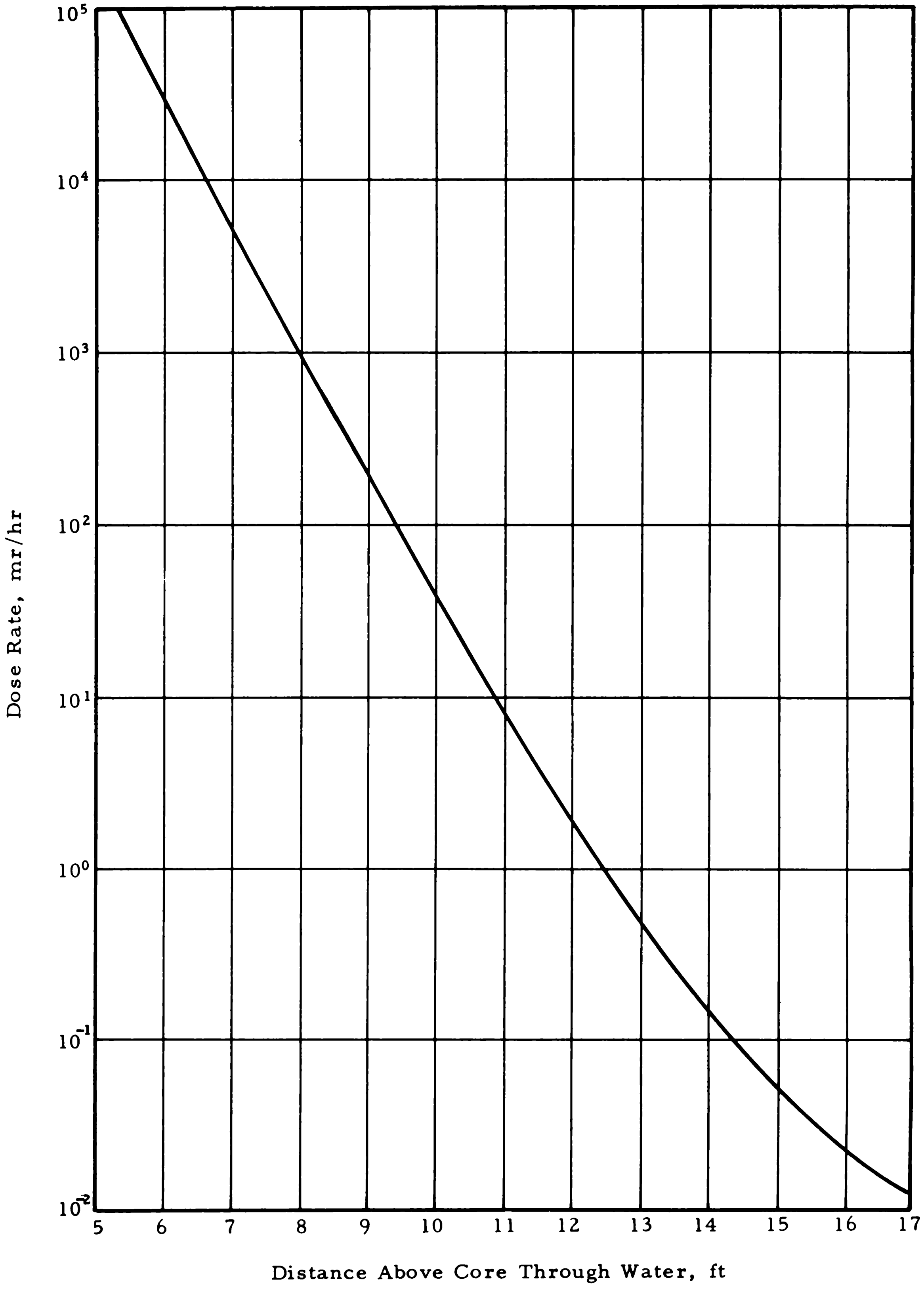


Figure 8.2-2. Total Gamma Dose Rate Through Lead, 10 feet from Heat Exchanger

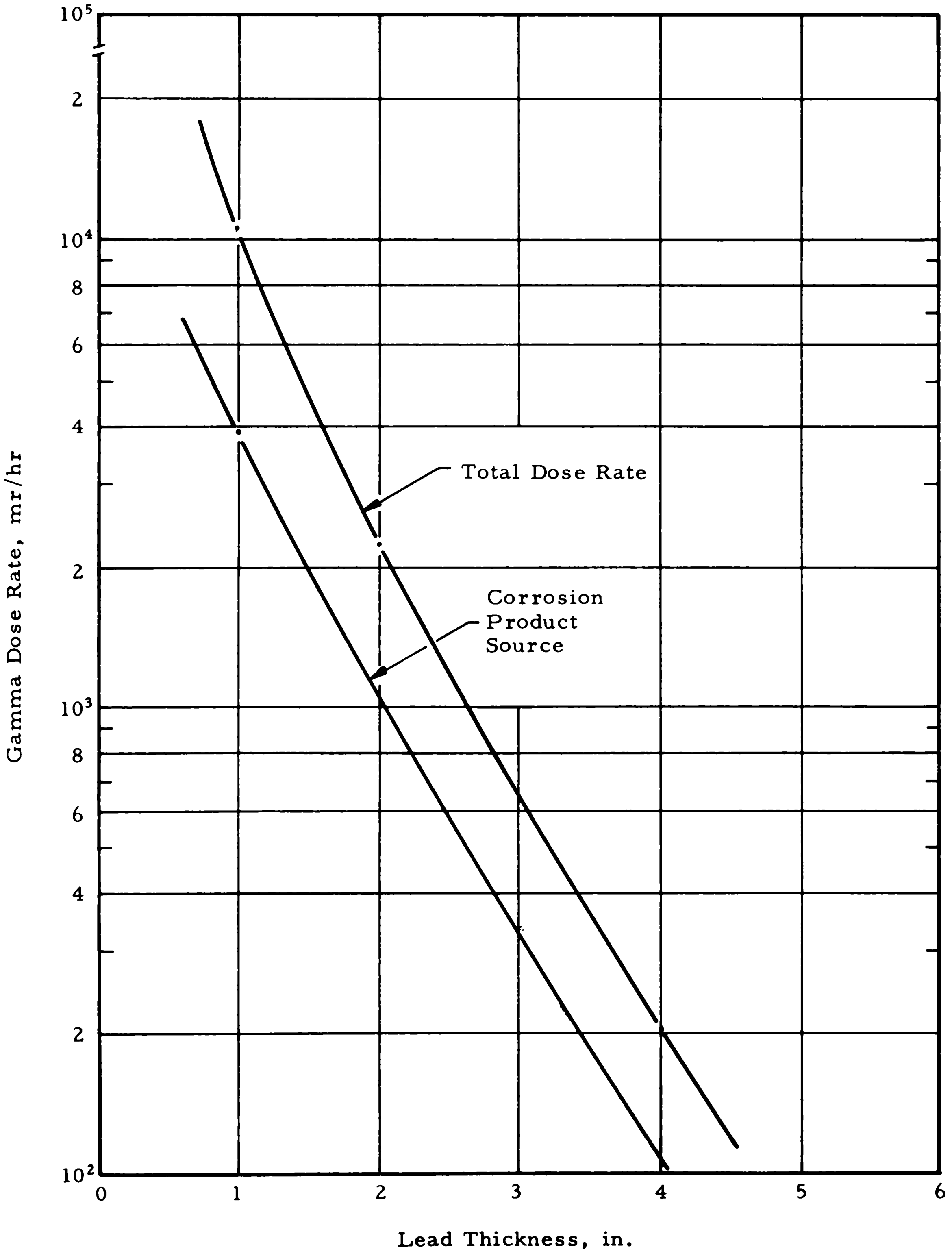


Figure 8.2-3. Containment Surface Dose Rate Due to Fission Products in Condenser Vs Thickness of Lead on Condenser

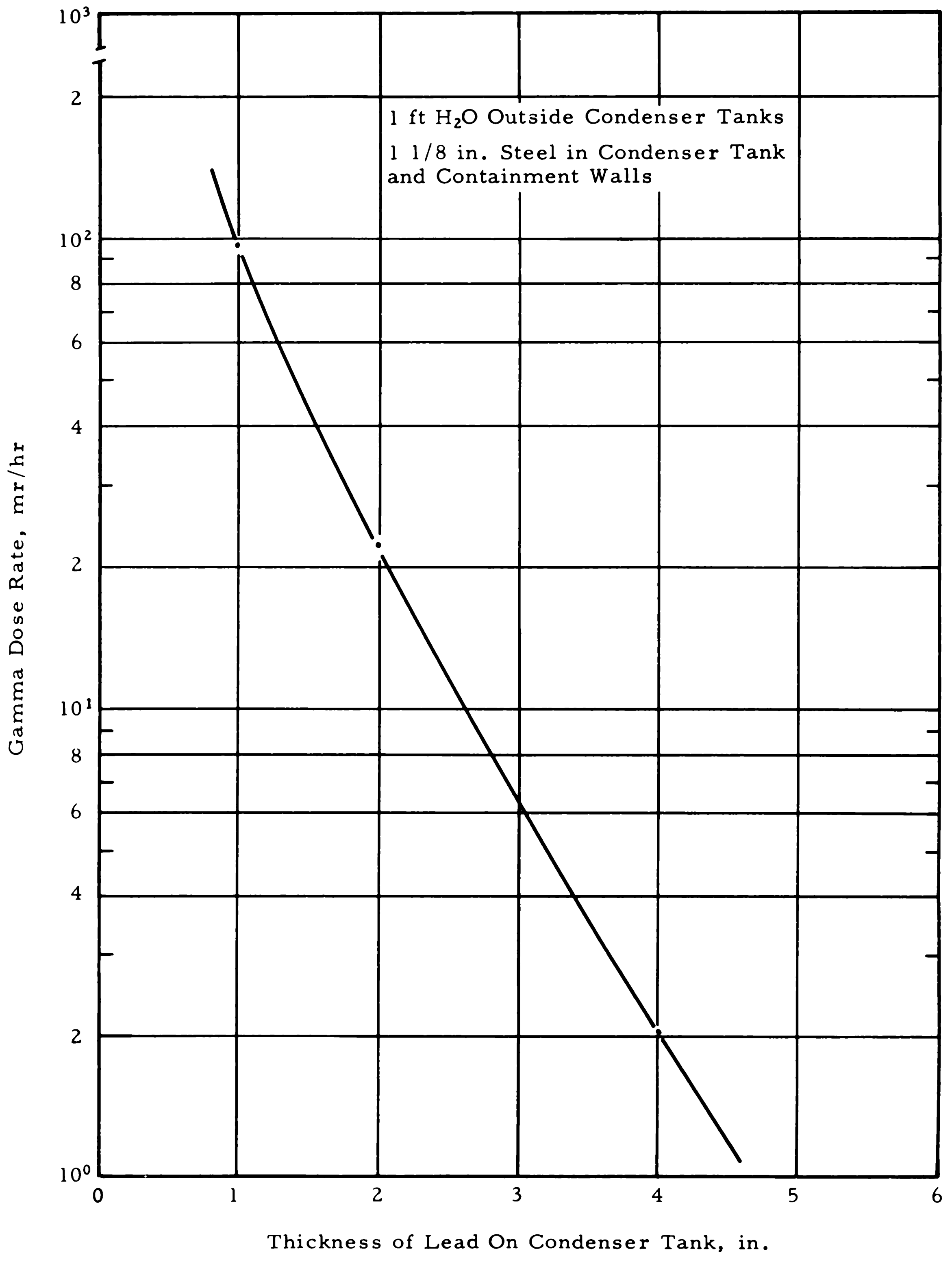


Figure 8.2-4. Dose Rate at Shield Surface Vs Lead Thickness — Control Rod Drive Coupling (0.2 wt % Co in Steel)

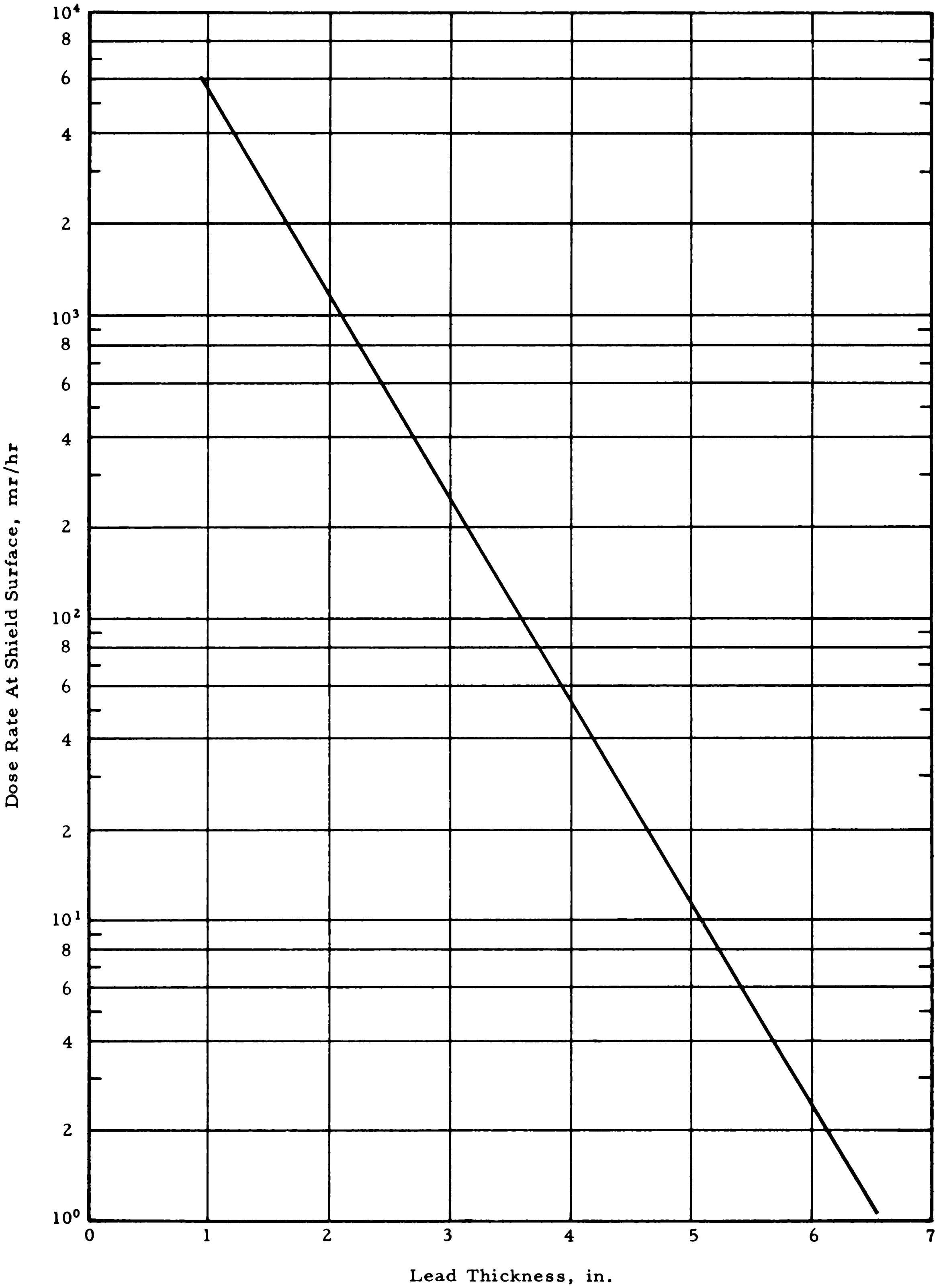


Figure 8.3-1. Diffusion Property — Variation with Void
(τ Vs % Moderated Void)

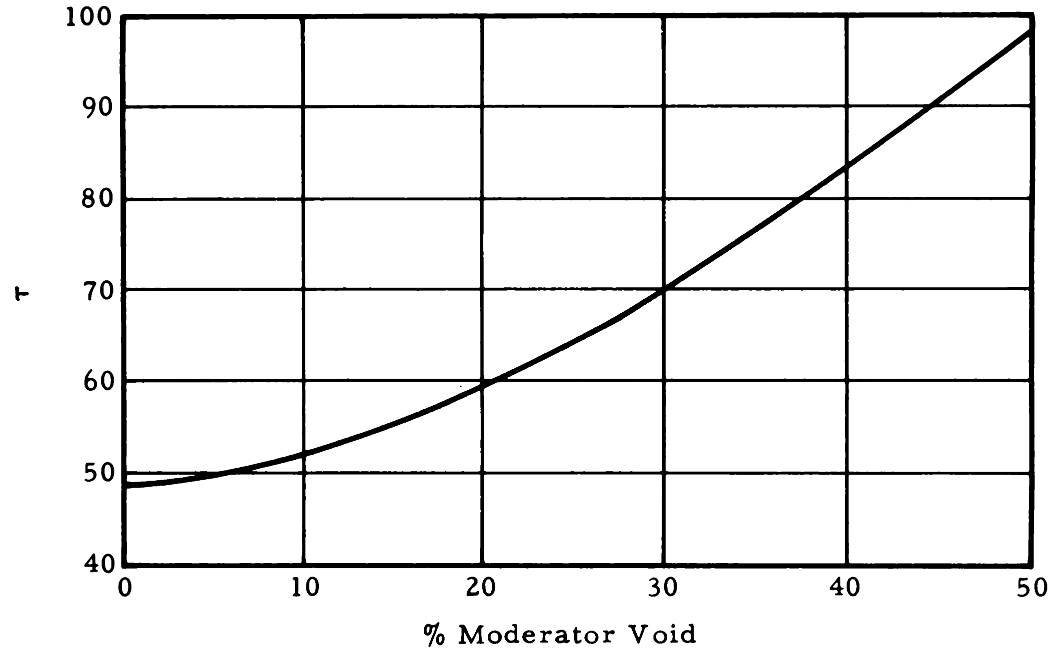


Figure 8.3-2. Diffusion Property — Variation with Void
(R_{28}^{eff} Vs wt % Fuel Enrichment)

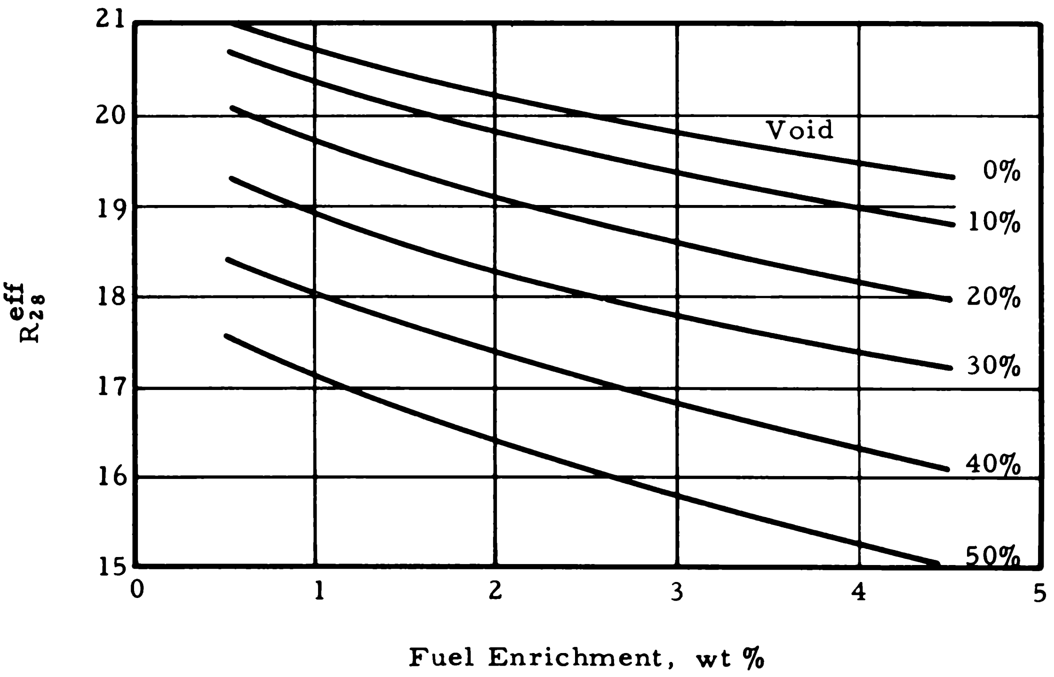


Figure 8.3-3. Diffusion Property — Variation with Void
(D_1 Vs % Moderated Void)

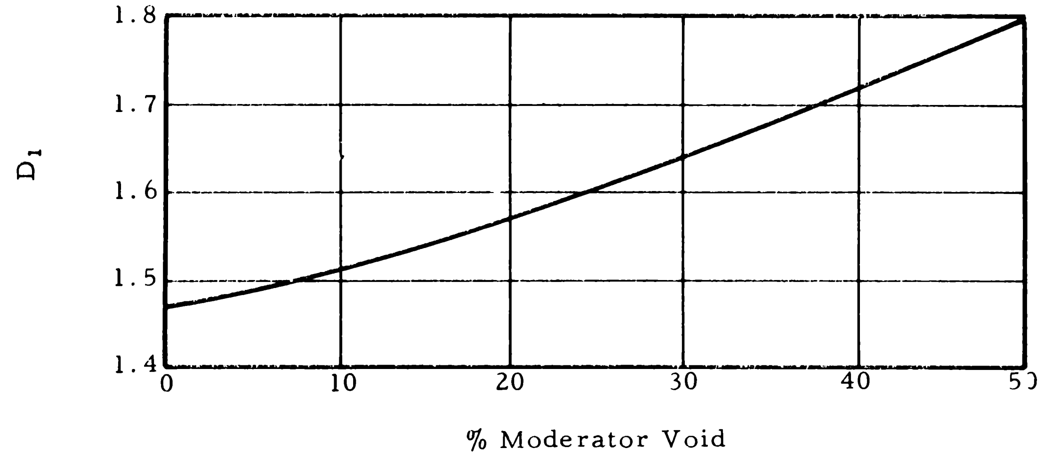


Figure 8.3-4. Diffusion Property — Variation with Void
(D_2 Vs % Moderator Void)

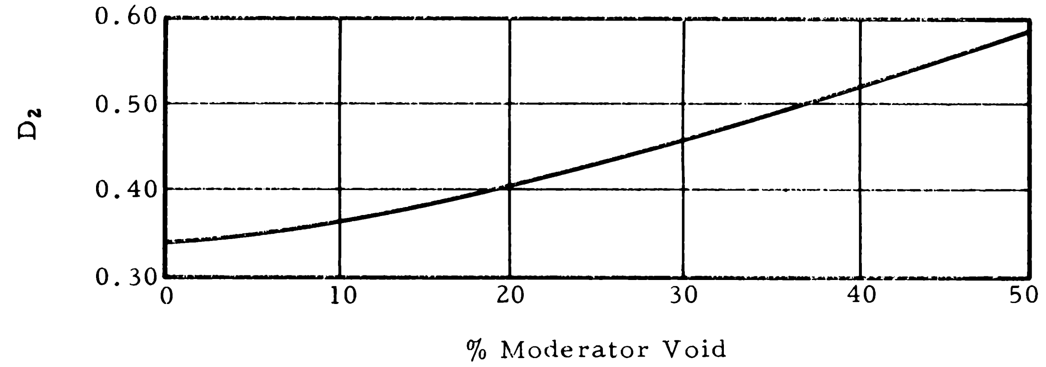


Figure 8.3-5. Diffusion Property — Variation with Void
($\xi \Sigma_s$ Vs % Moderator Void)

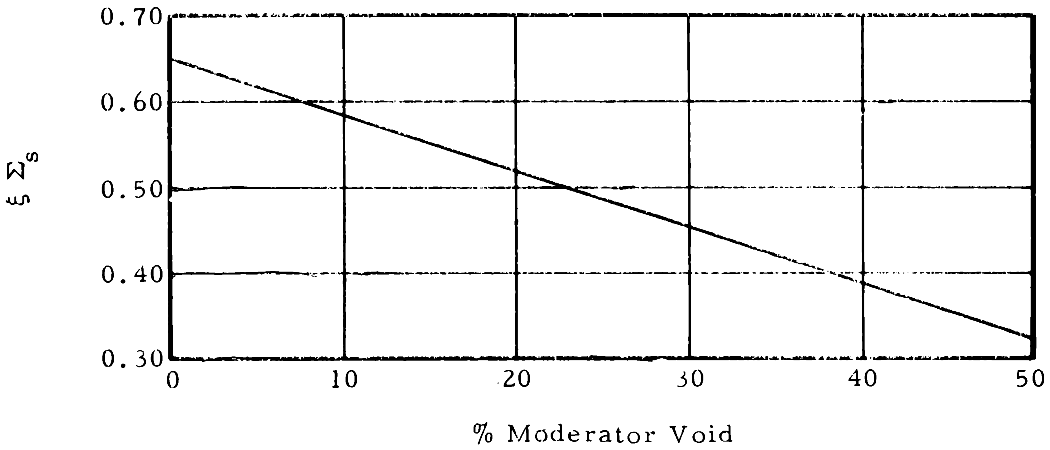


Figure 8.4-1. Decomposition Products From Irradiation of 0.10 M Boric Acid in Light Water

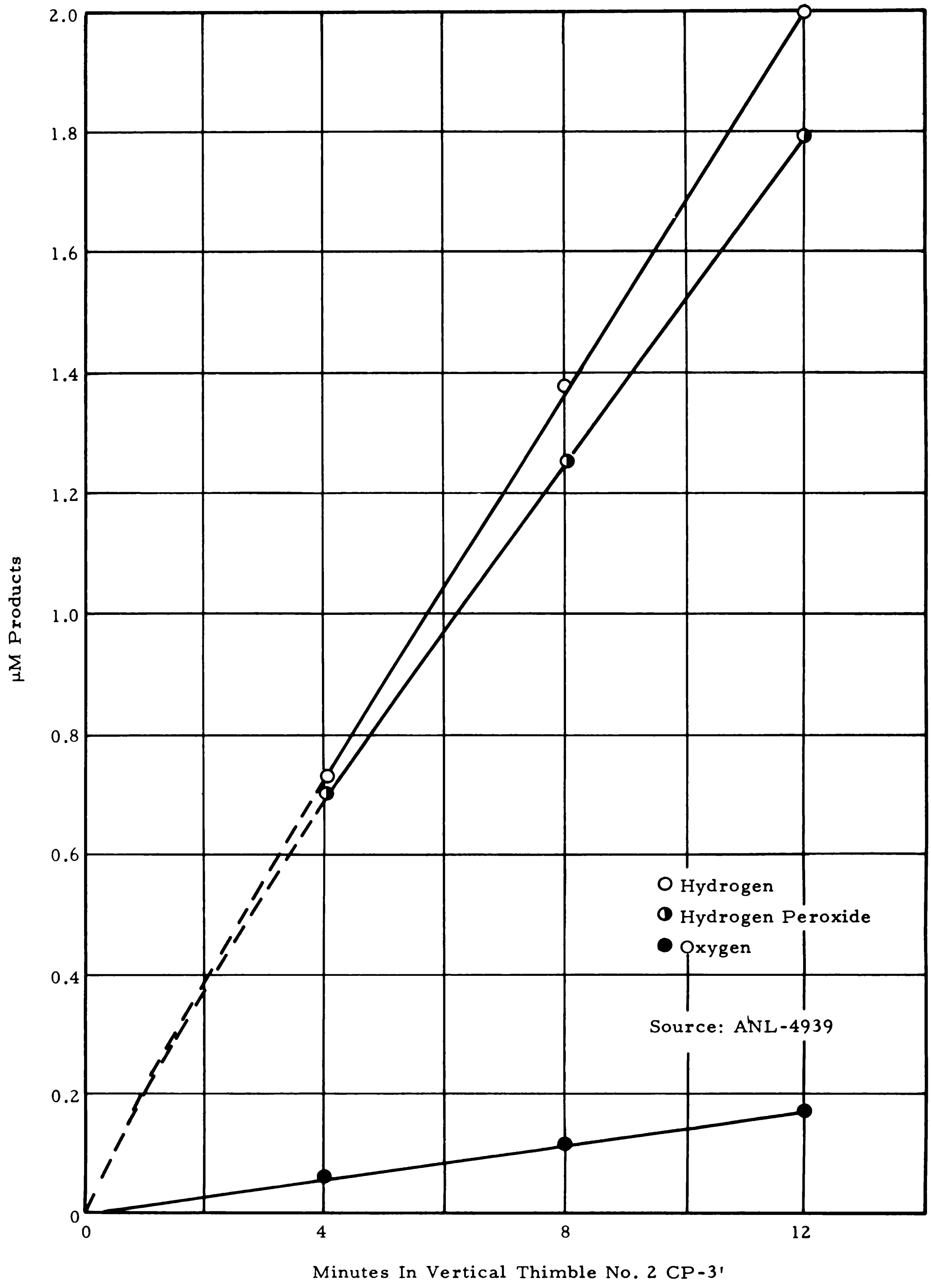


Figure 8.4-2. Effect of Initial Dissolved Hydrogen on Hydrogen Production in Boric Acid Solution Irradiated in CP-3

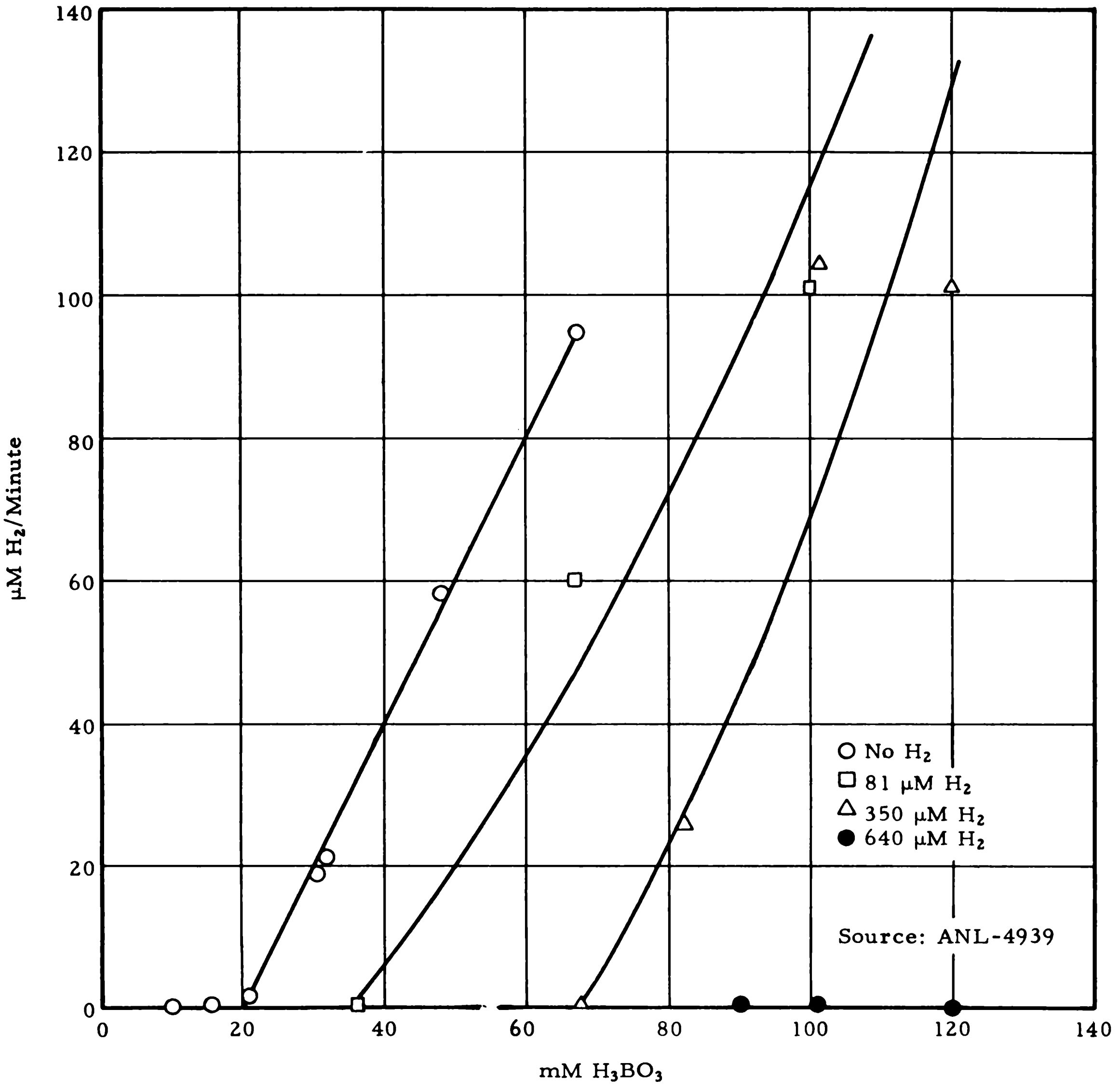


Figure 8.4-3. Effects of Boric Acid Concentration on Hydrogen Production in Air-Free Solutions Irradiated in CP-3

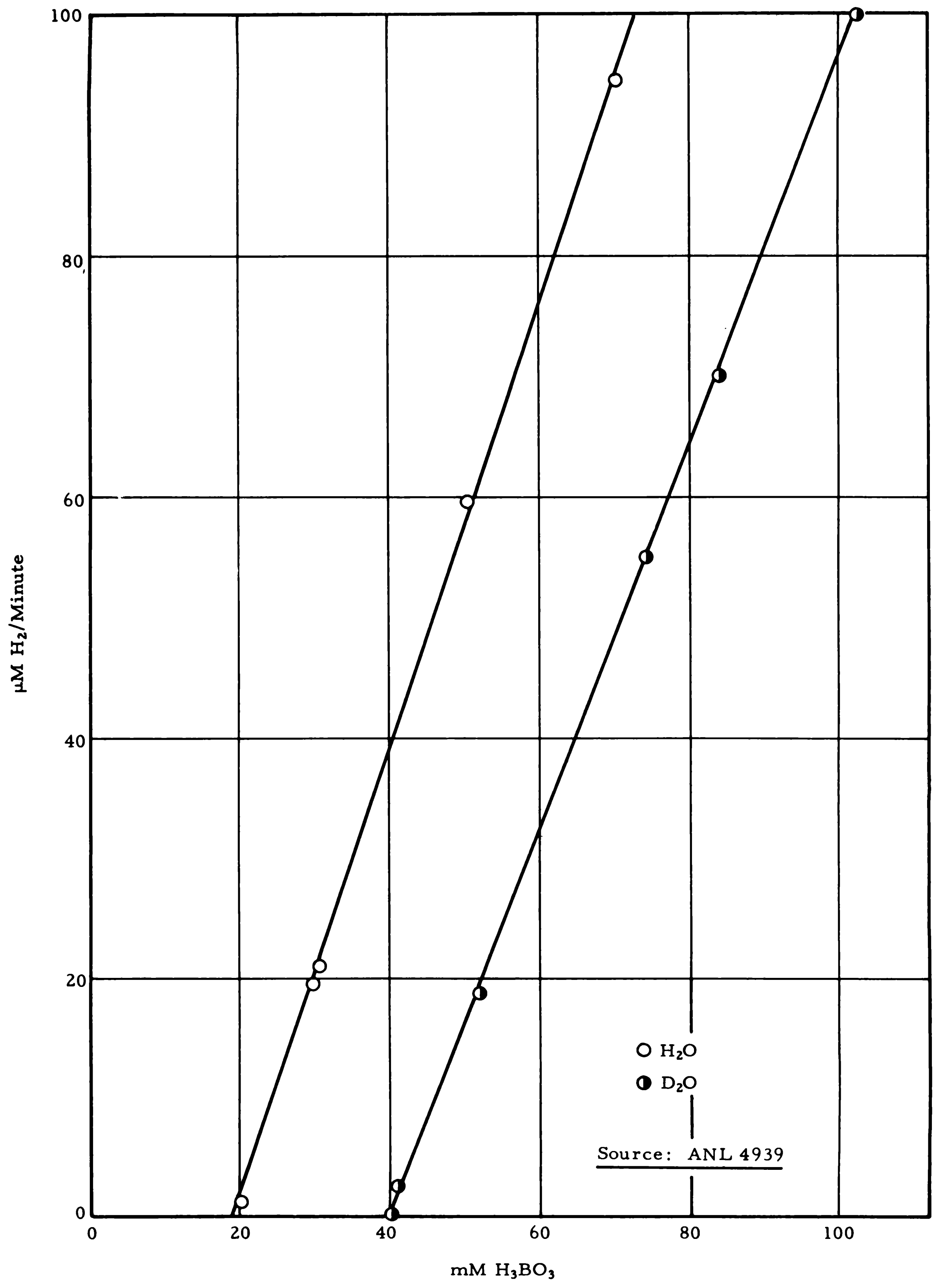


Figure 8.4-4. Solubility of Boric Acid in Water

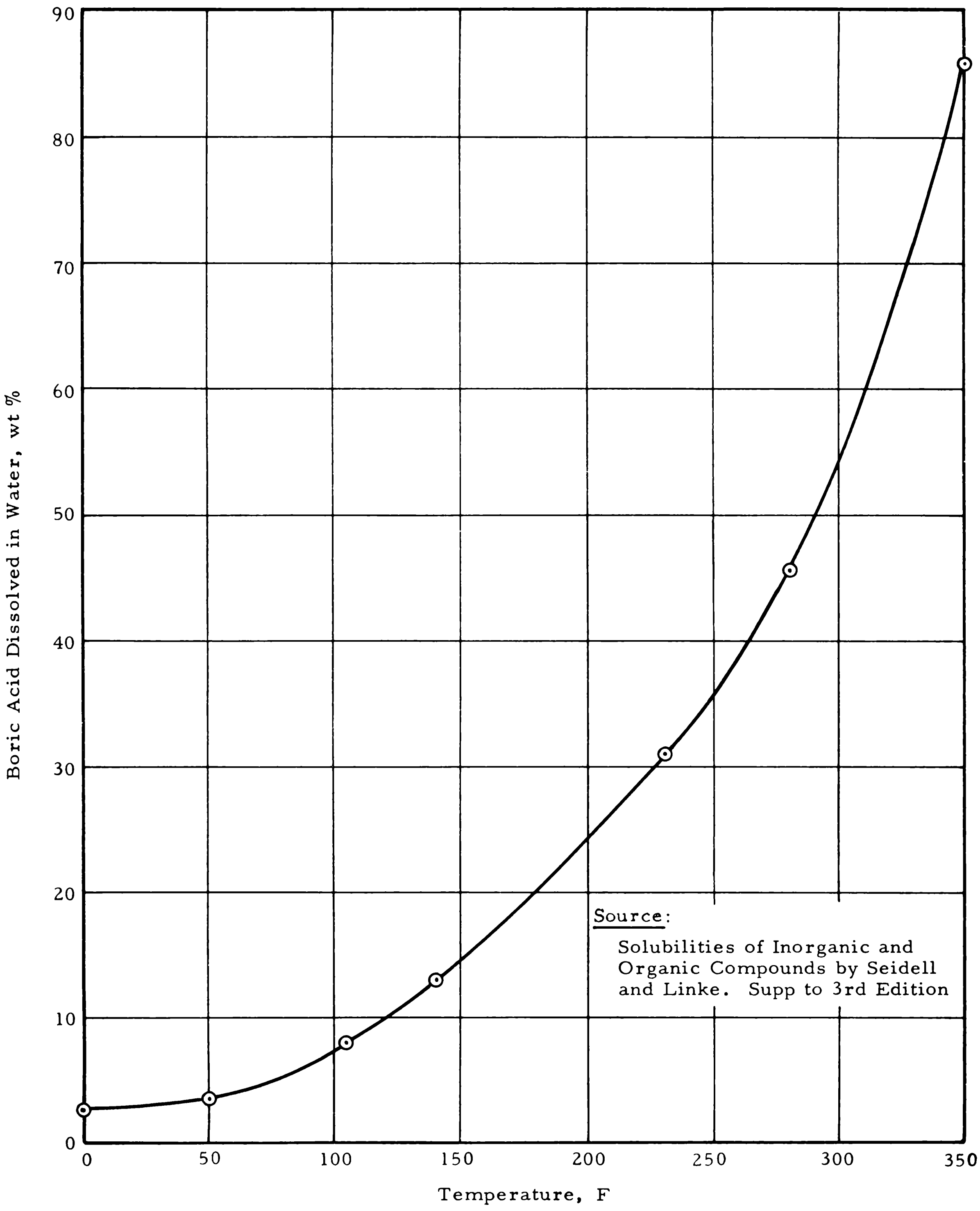


Figure 8.4-5. Solubility of Potassium Tetraborate in Water

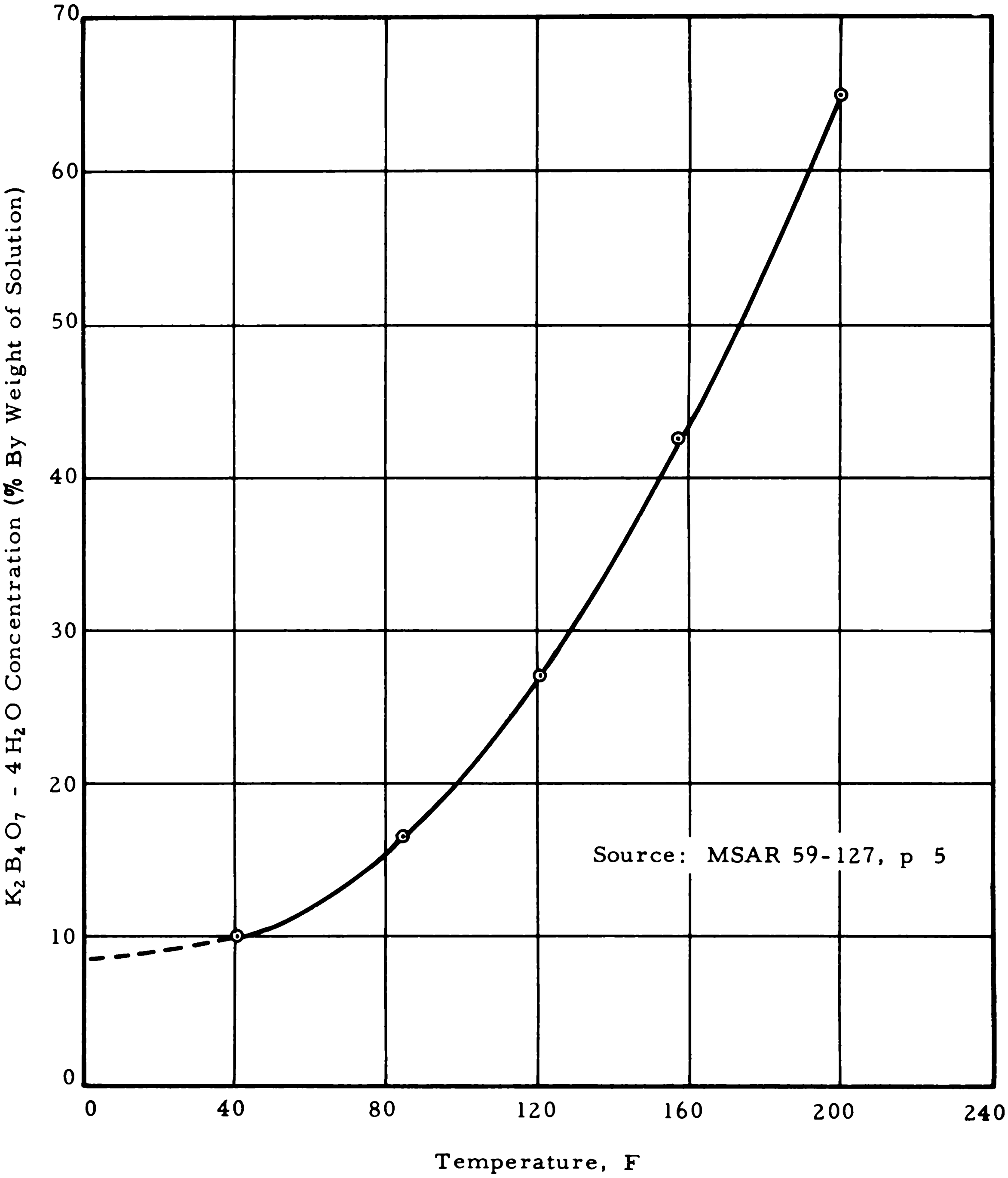


Figure 8.4-6. Solubility of Ammonium Pentaborate in Water

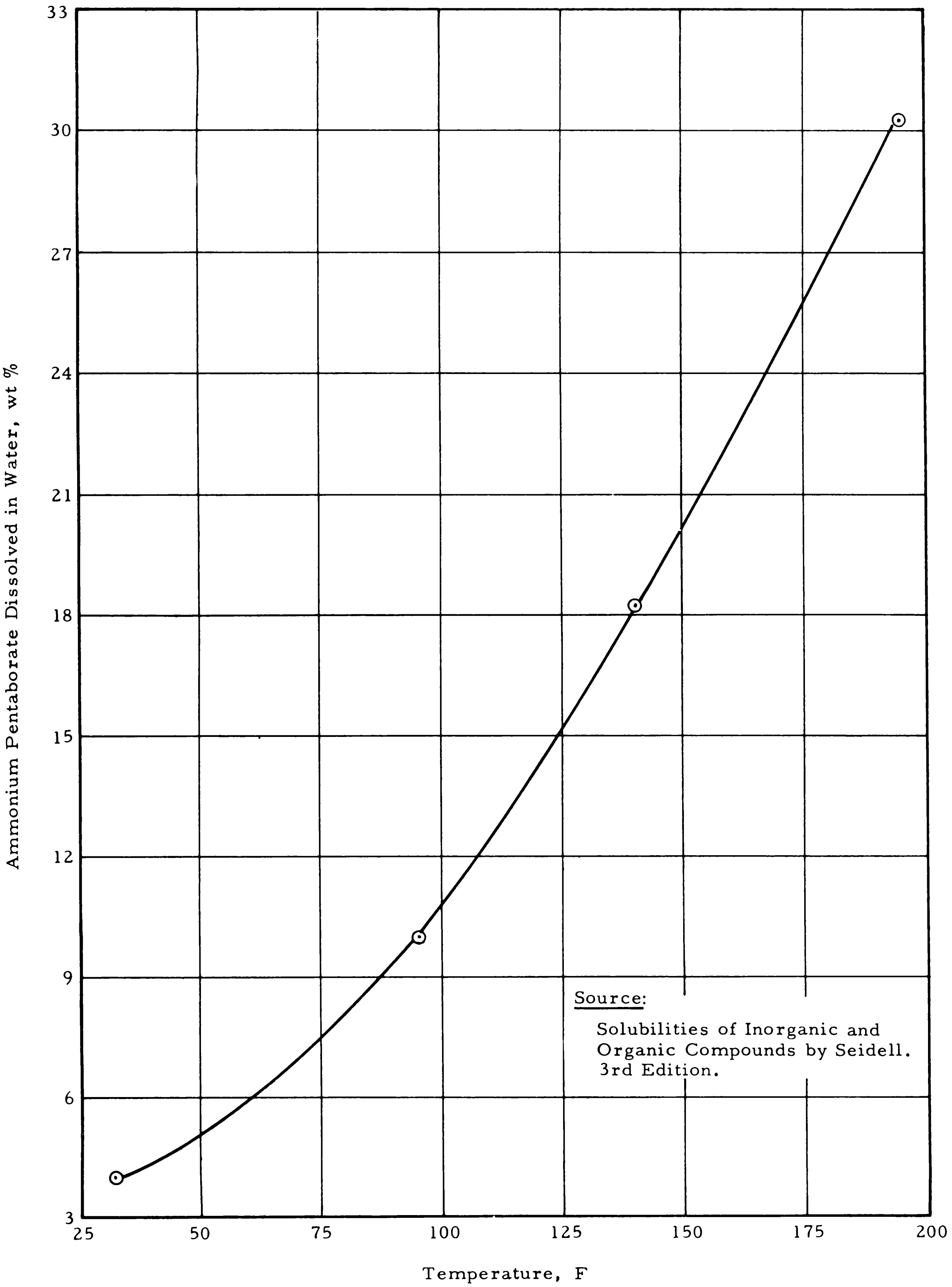


Figure 8.4-7. Solubility of Sodium Tetraborate in Water

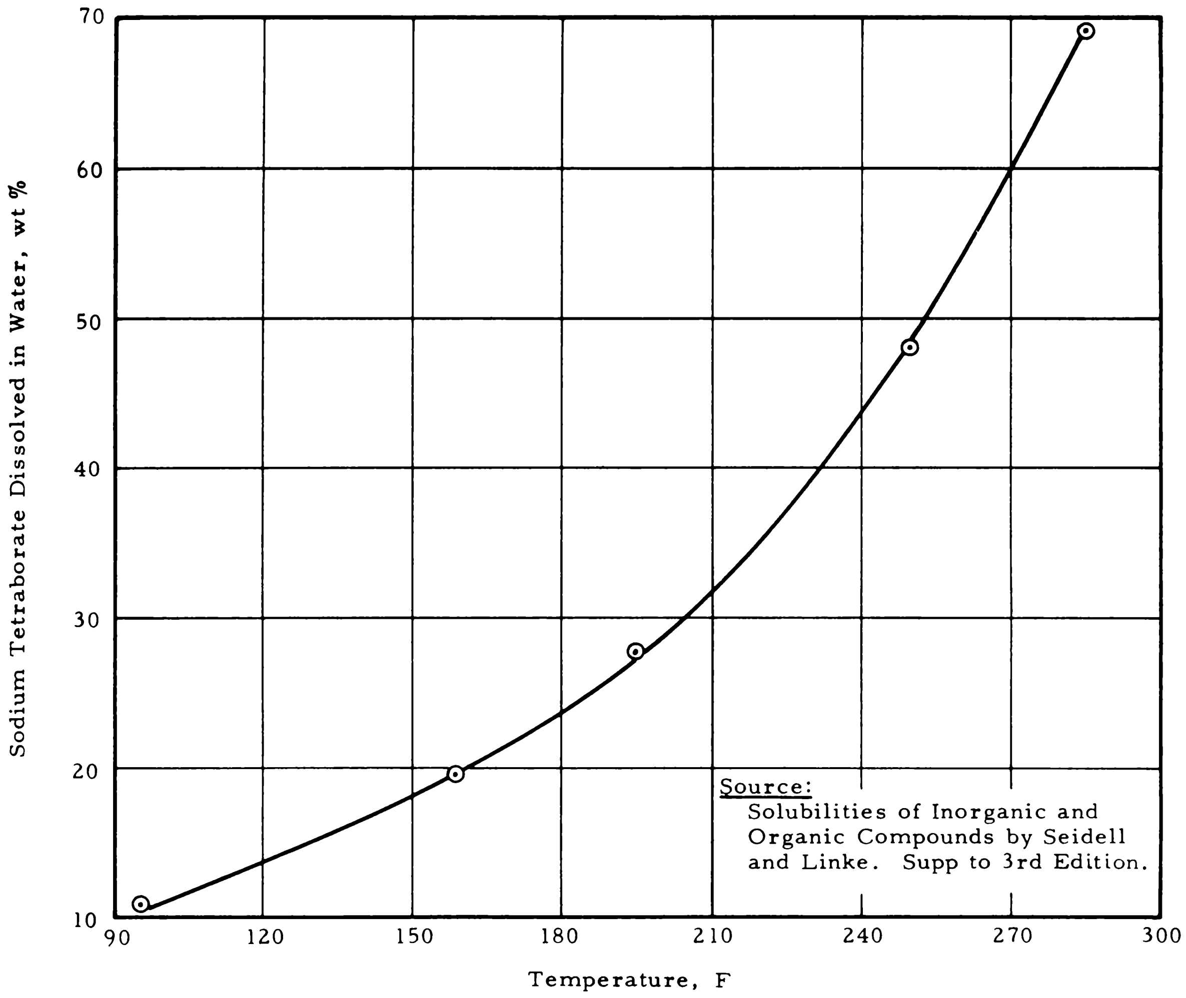


Figure 8.4-8. Solubility of Cadmium Sulfate in Water

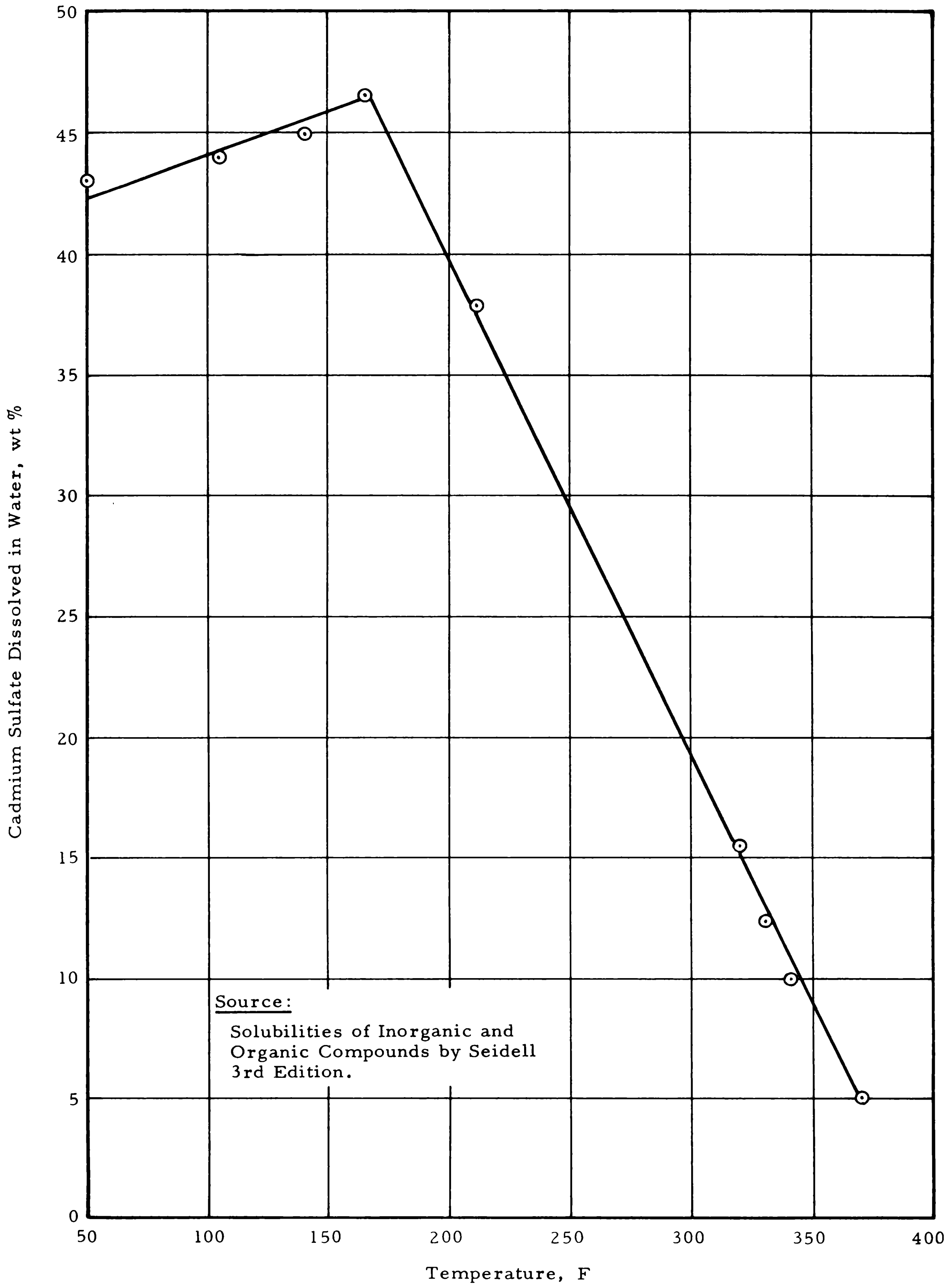


Figure 8.4-9. pH of Solution of Boric Acid at 20° C

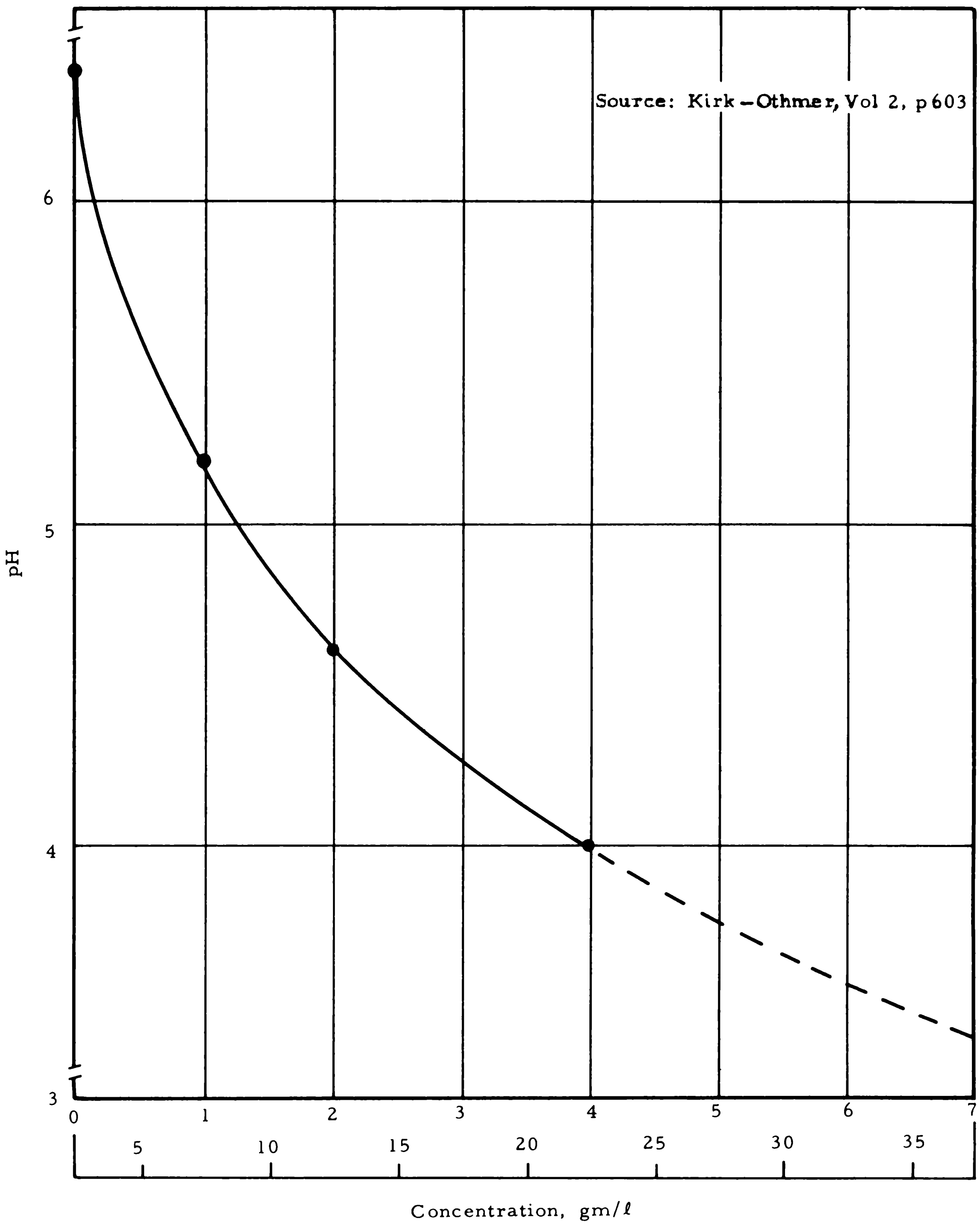


Figure 8.6-1. Basic Computer Wiring Diagram, Once-Through Steam Generator

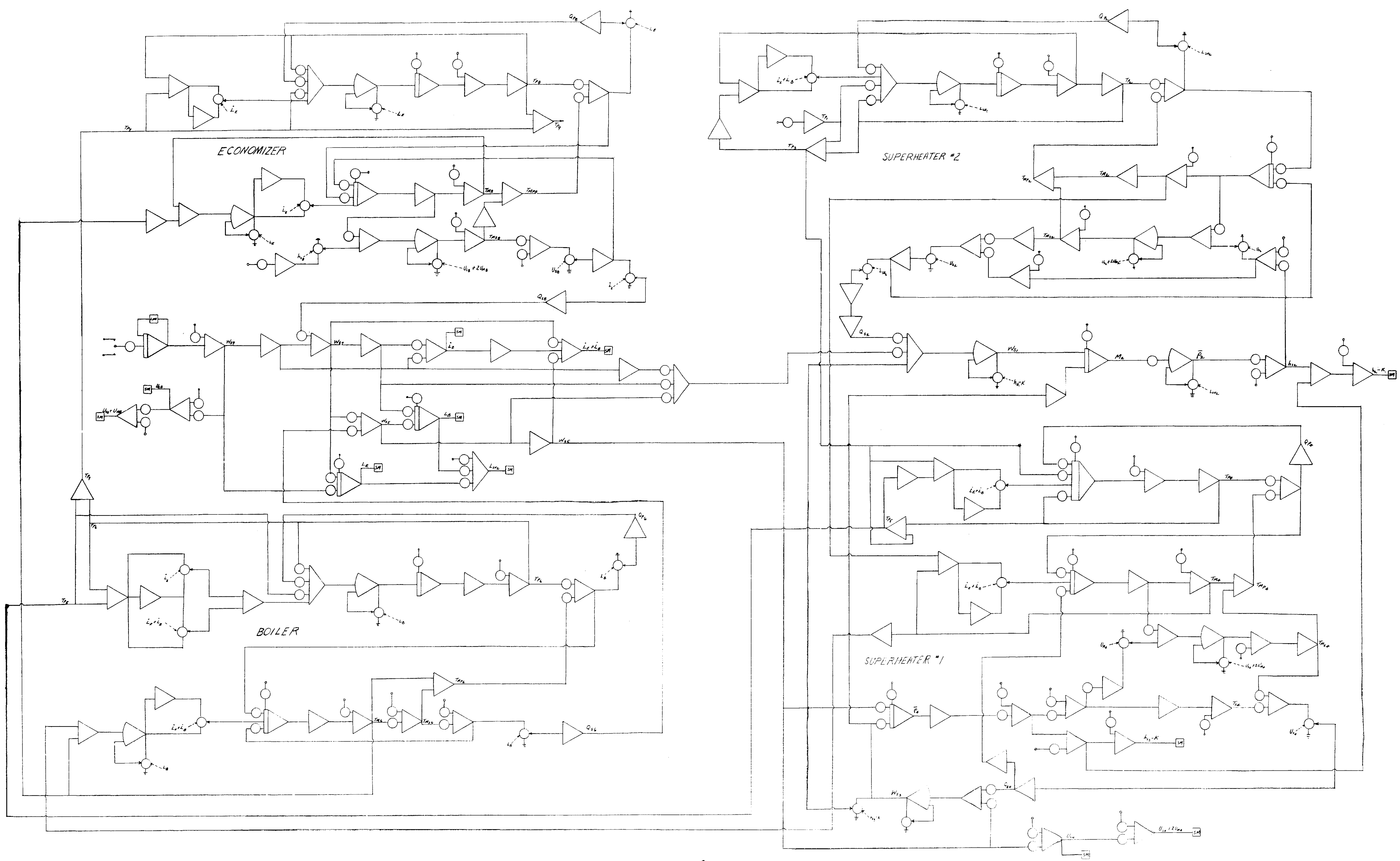


Figure 8.6-2. Secondary Flow Response of Each Section to Feedwater Flow Ramp of 60 lb/sec in 20 seconds

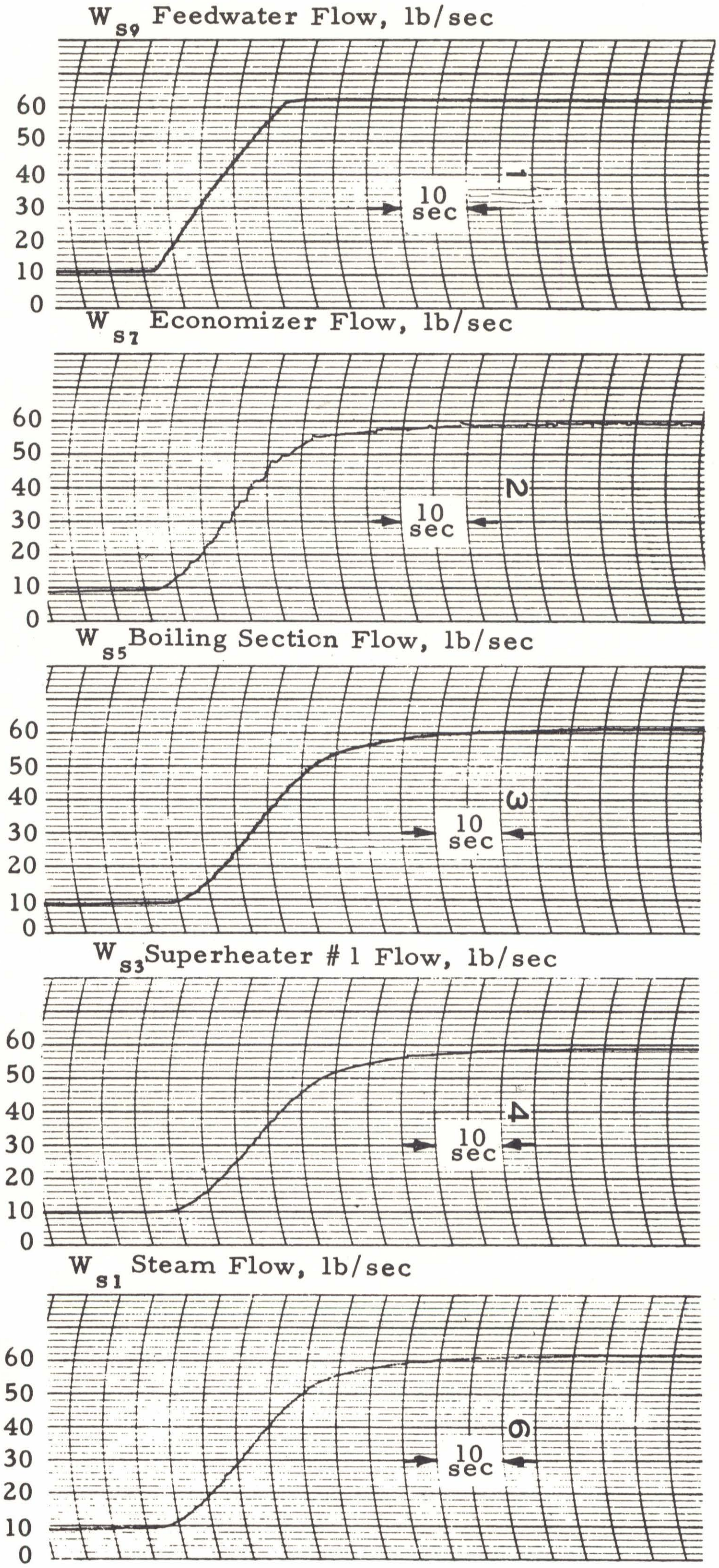


Figure 8. 6-3. Response of Section Length and Steam Flow to Feedwater Ramps of 10 to 60 lb/sec and 60 to 10 lb/sec in 20 seconds

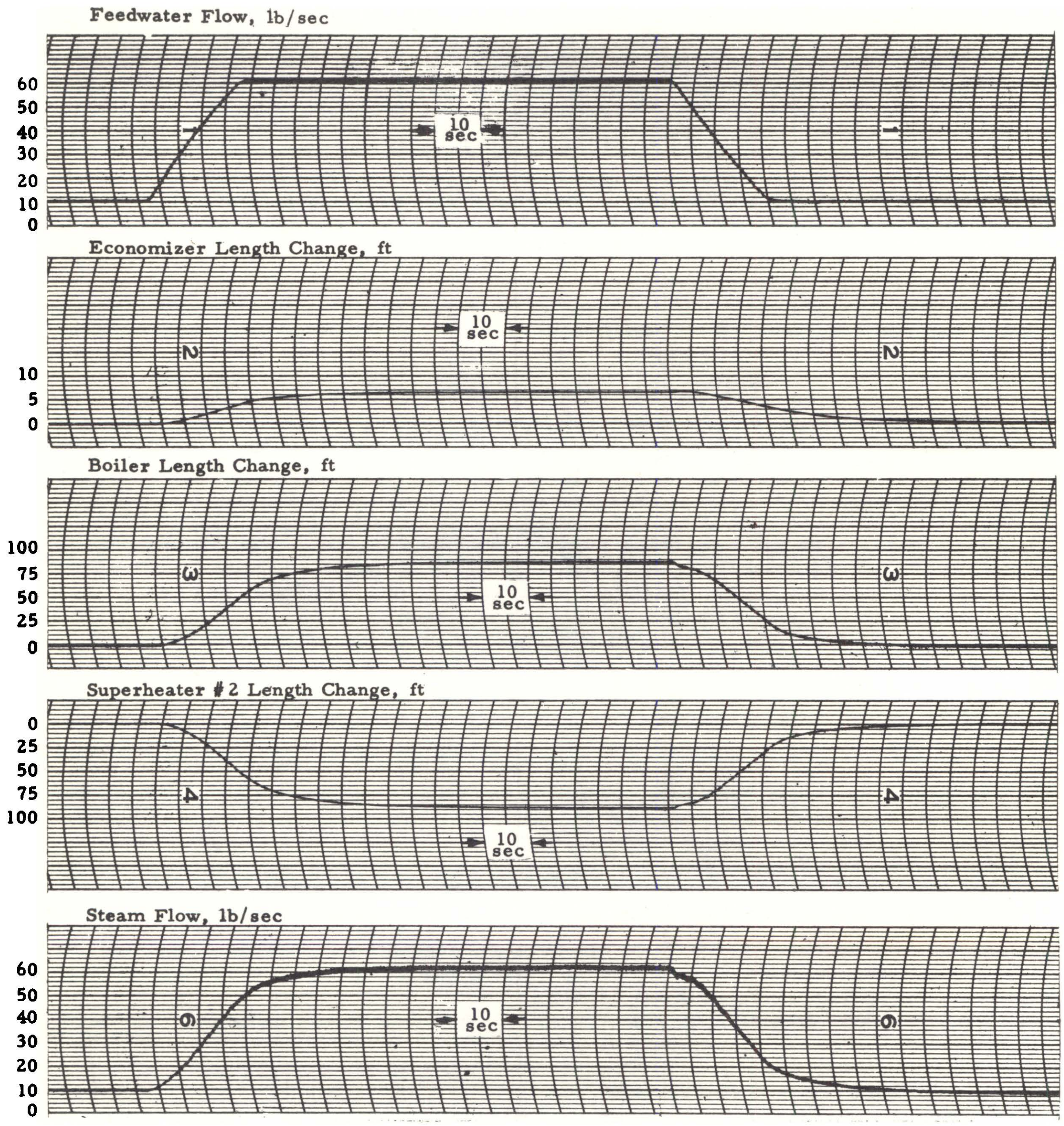


Figure 8.6-4. Response of Average Temperatures to Feedwater Flow Ramps of 10 to 60 lb/sec and 60 to 10 lb/sec in 20 seconds

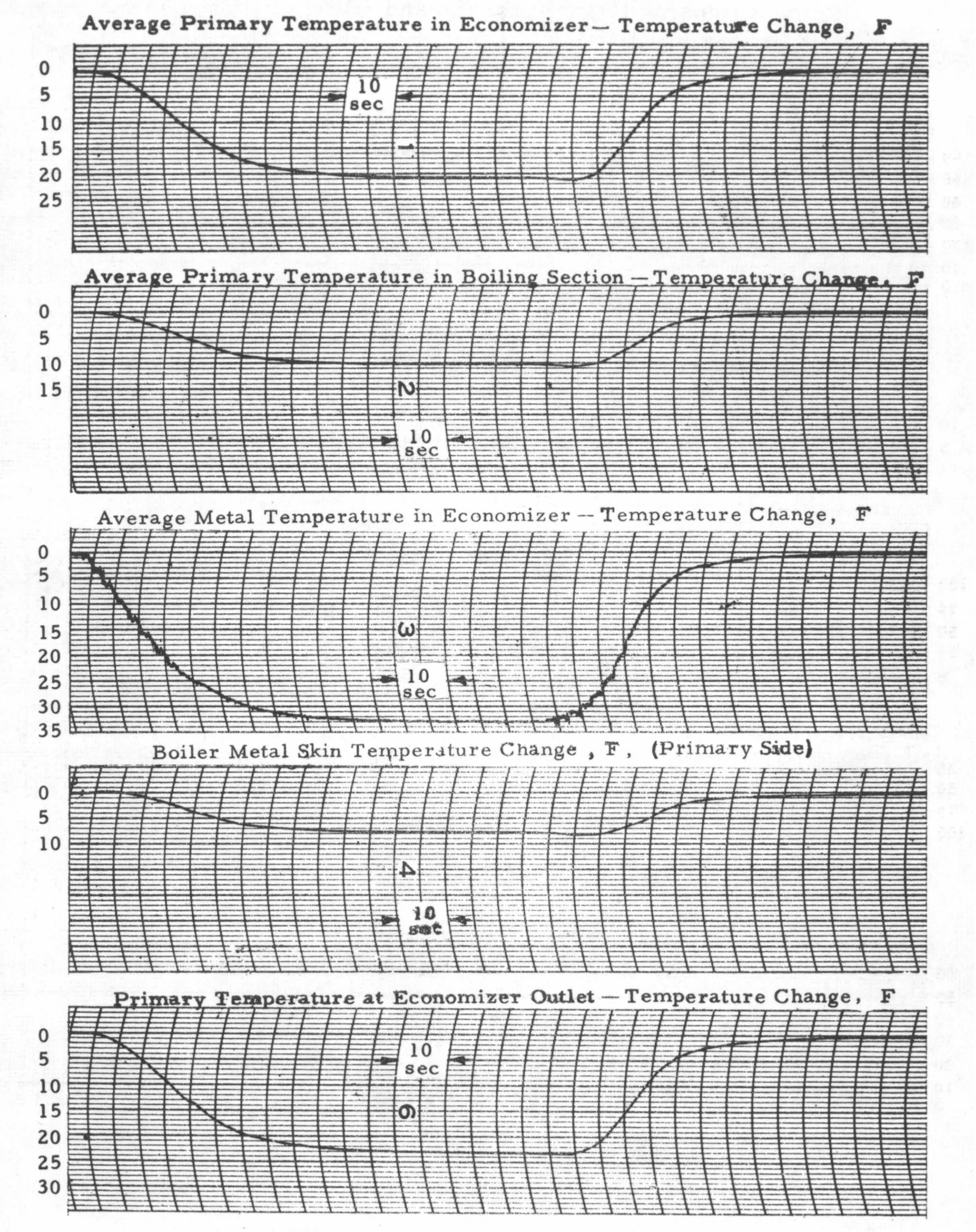
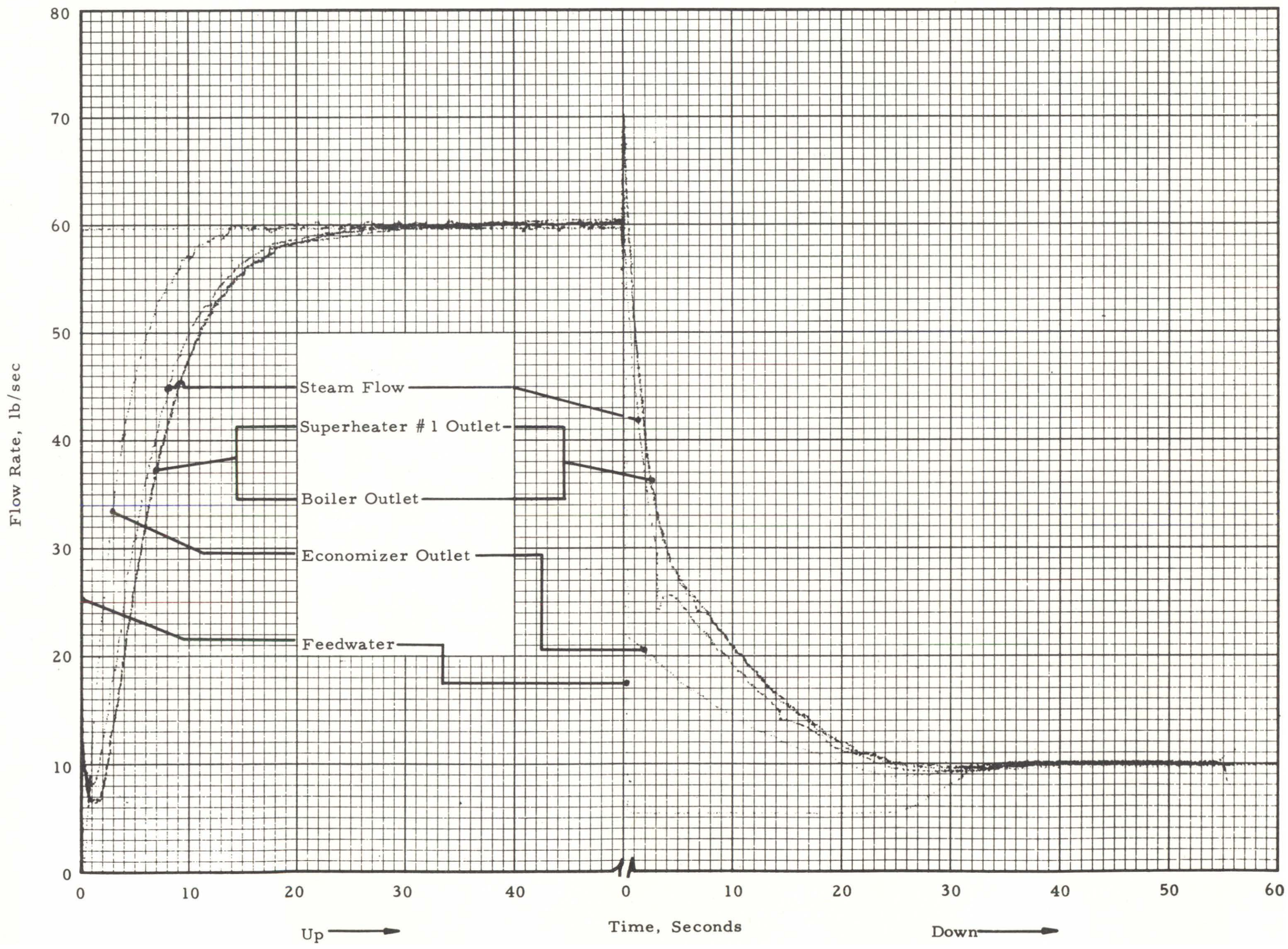


Figure 8.6-5. Response of Flow in Each Section to Feedwater Steps of 10 to 60 lb/sec and 60 to 5 lb/sec with Ramp from 5 lb/sec to 10 lb/sec in 6 seconds



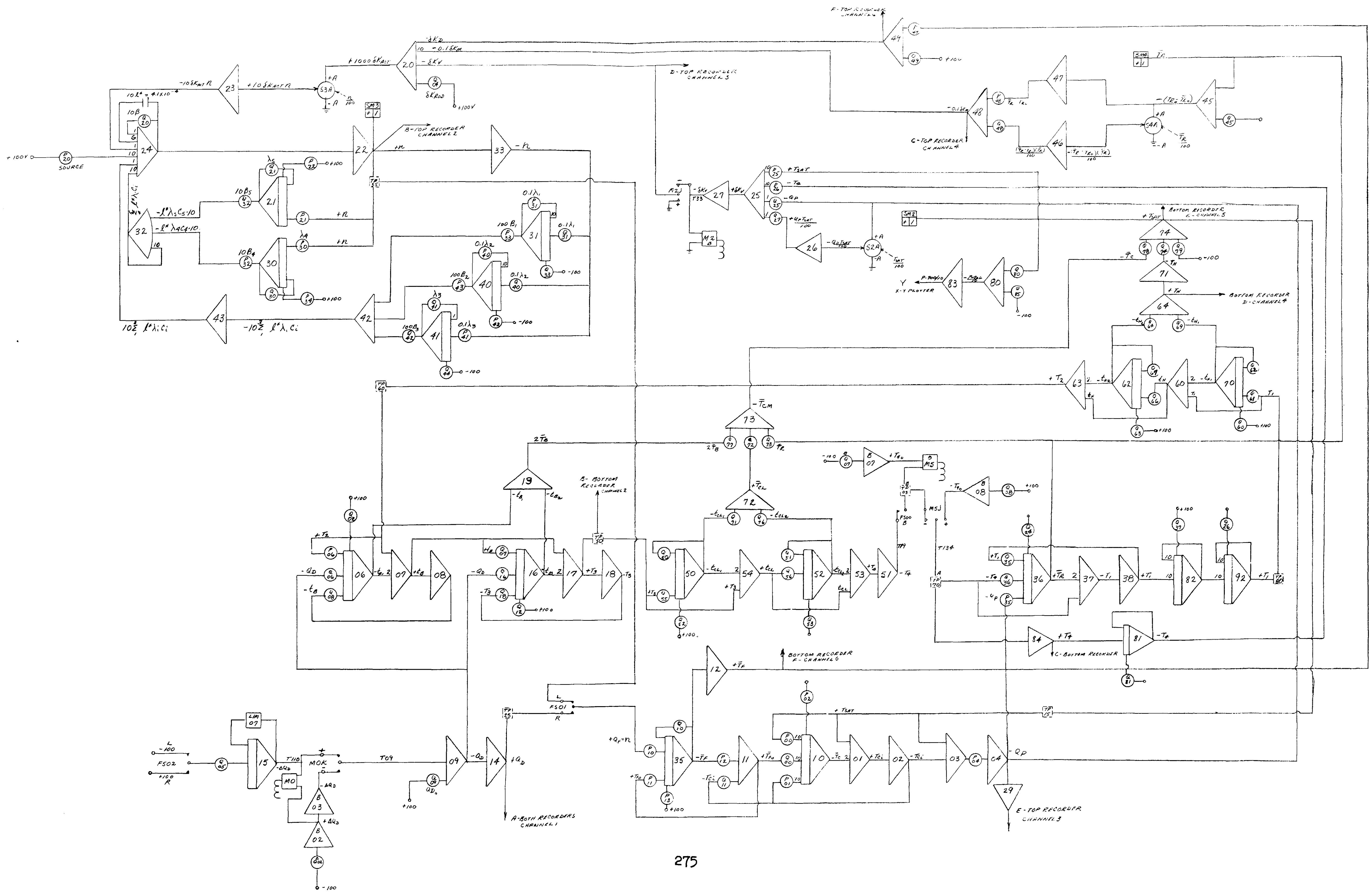


Figure 8.7-4. Analog Computer Wiring Diagram of CNSG Power Decrease Model

Figure 8.6-6. Response of Section Lengths to Feedwater Steps of 10 to 60 lb/sec and 60 to 10 lb/sec

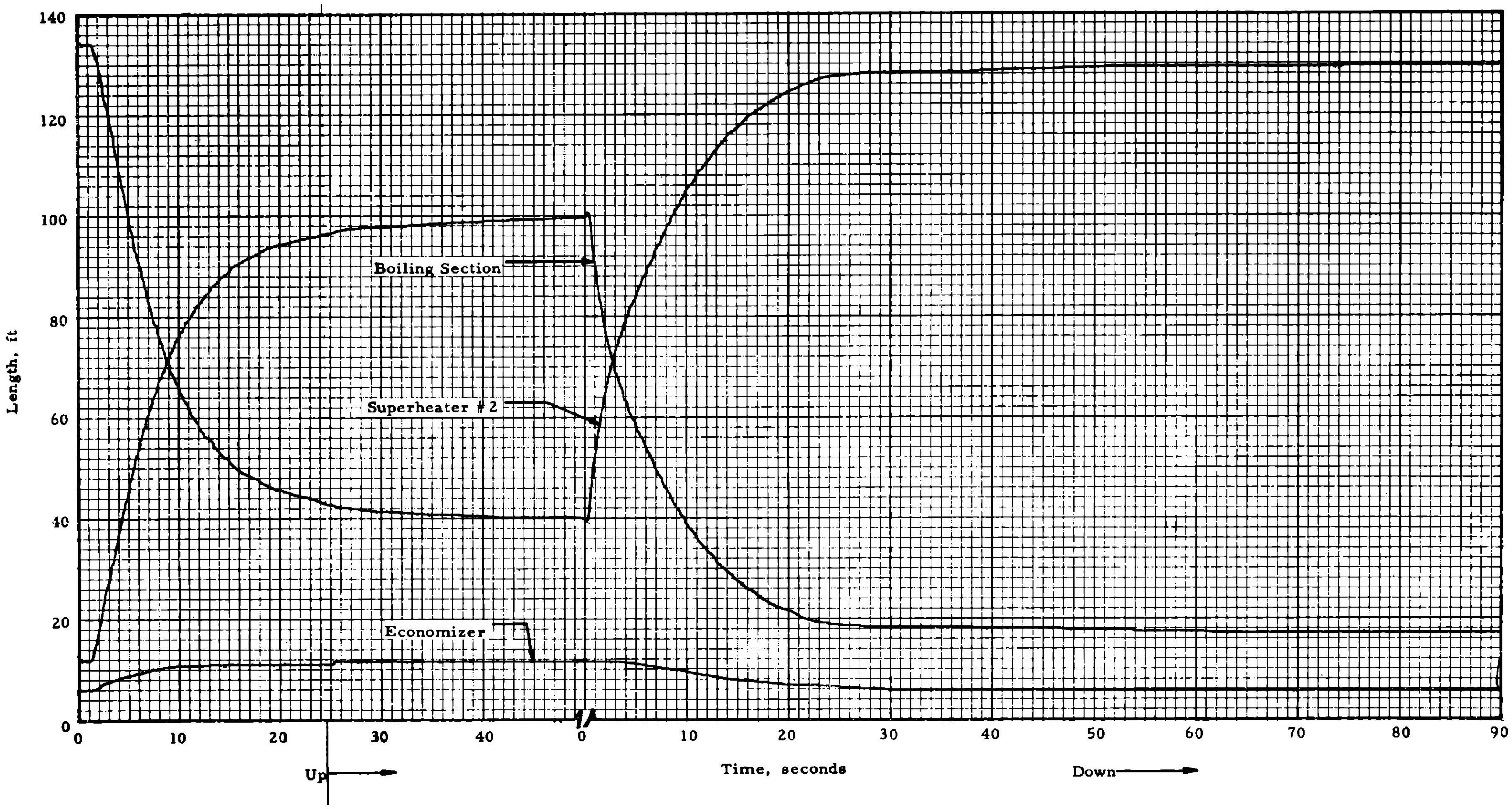


Figure 8.6-7. Response of Superheater No. 2 Length to Feedwater Steps of:
10 to 60 lb/sec and 60 to 10 lb/sec,
10 to 66 lb/sec and 66 to 10 lb/sec,
10 to 72 lb/sec and 72 to 10 lb/sec;
and Response of Superheater No. 2 Outlet Enthalpy to Feedwater Steps of 10 to 72 lb/sec and 72 to 10 lb/sec

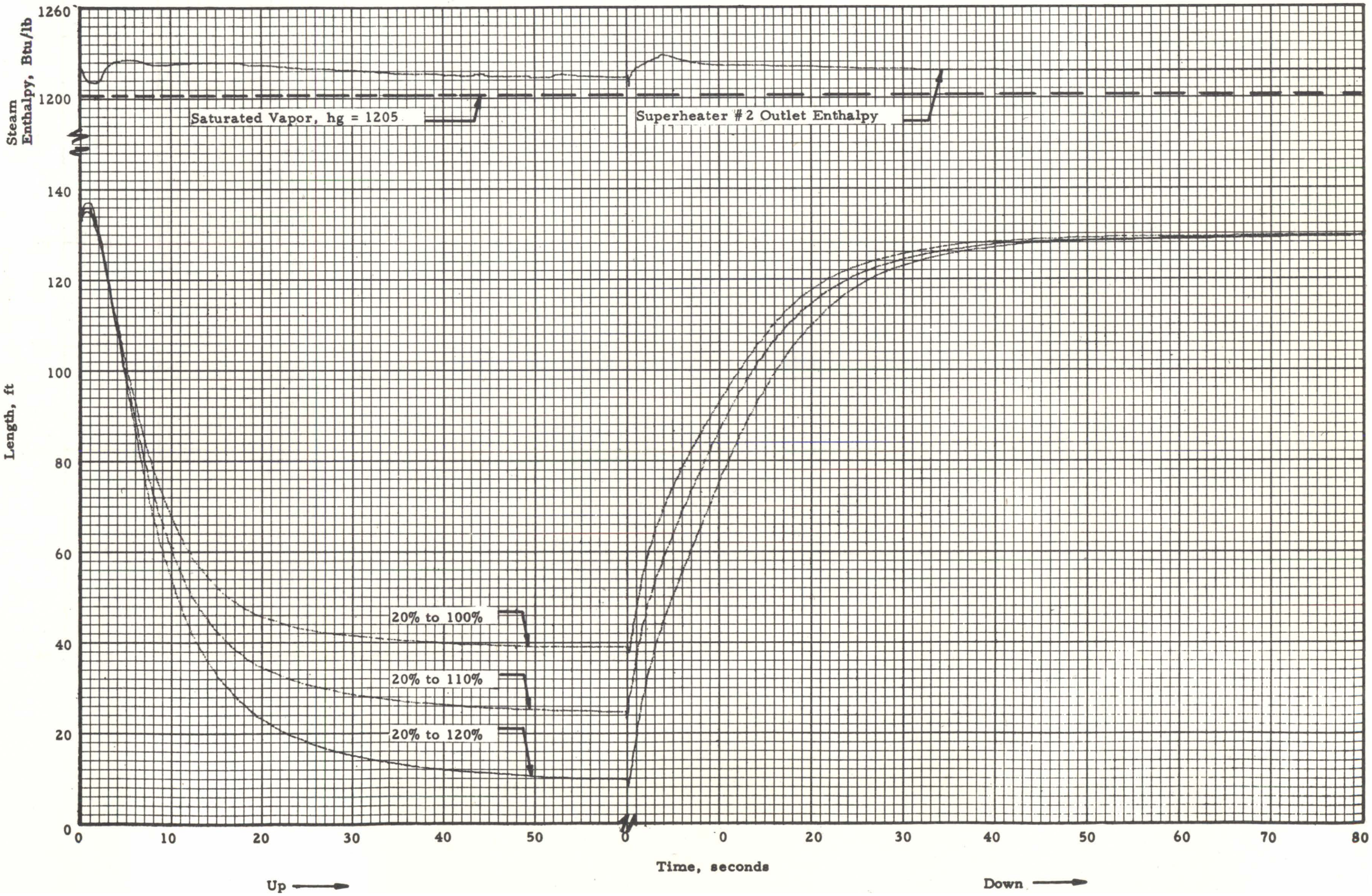


Figure 8.6-8. Response of Feedwater Flow and Superheater No. 2 Length for Steam Flow Ramps of 10 to 60 lb/sec in 80 seconds, and 60 to 10 lb/sec in 80 seconds

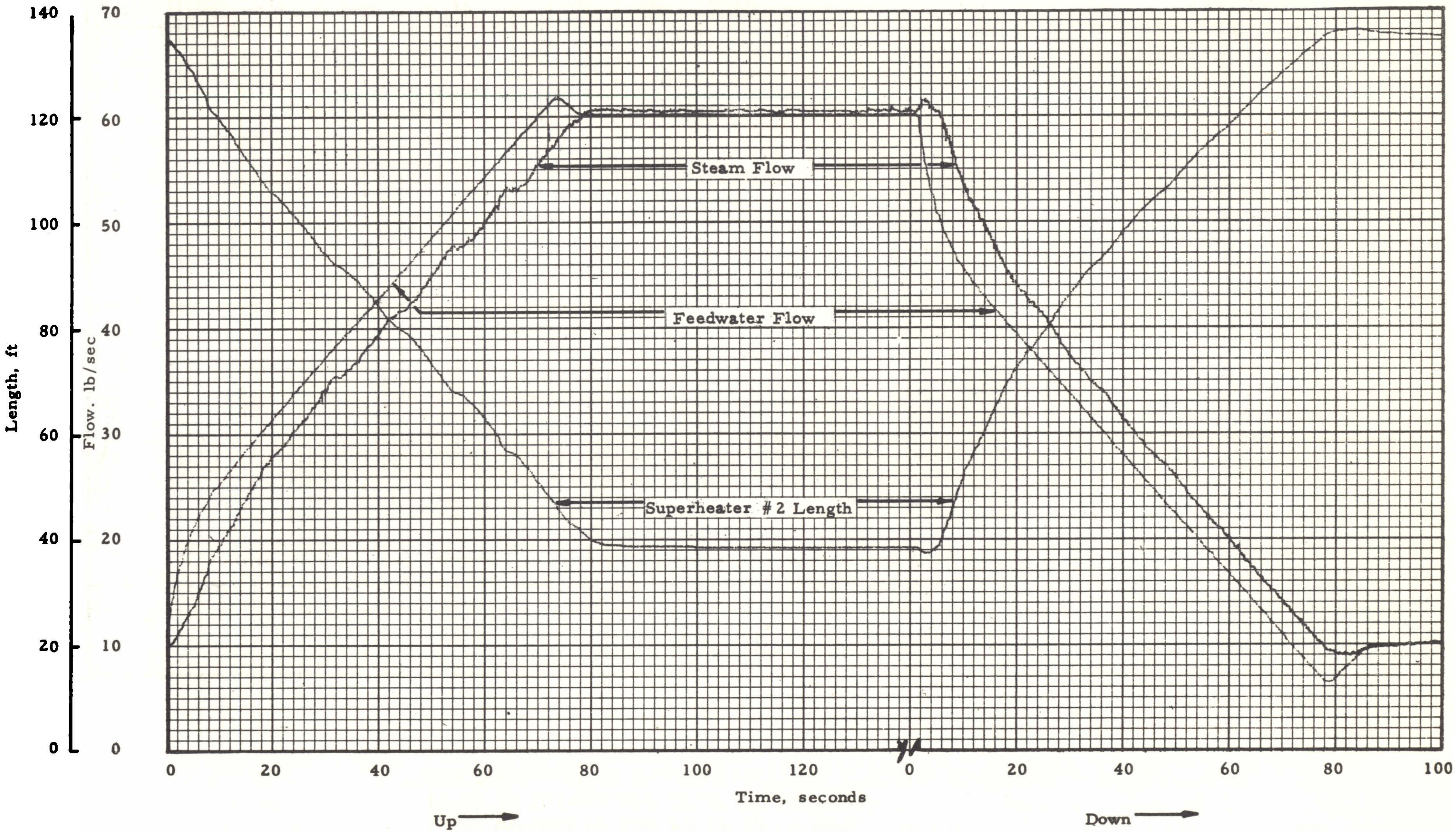


Figure 8.6-9. Steam Pressure Response to Steam Flow Ramp of 10 to 60 lb/sec in 80 seconds. Feedwater Flow Lags Steam Flow for 30 seconds, Then Increases to a Value 10% Greater Than Steam Flow in 2 seconds

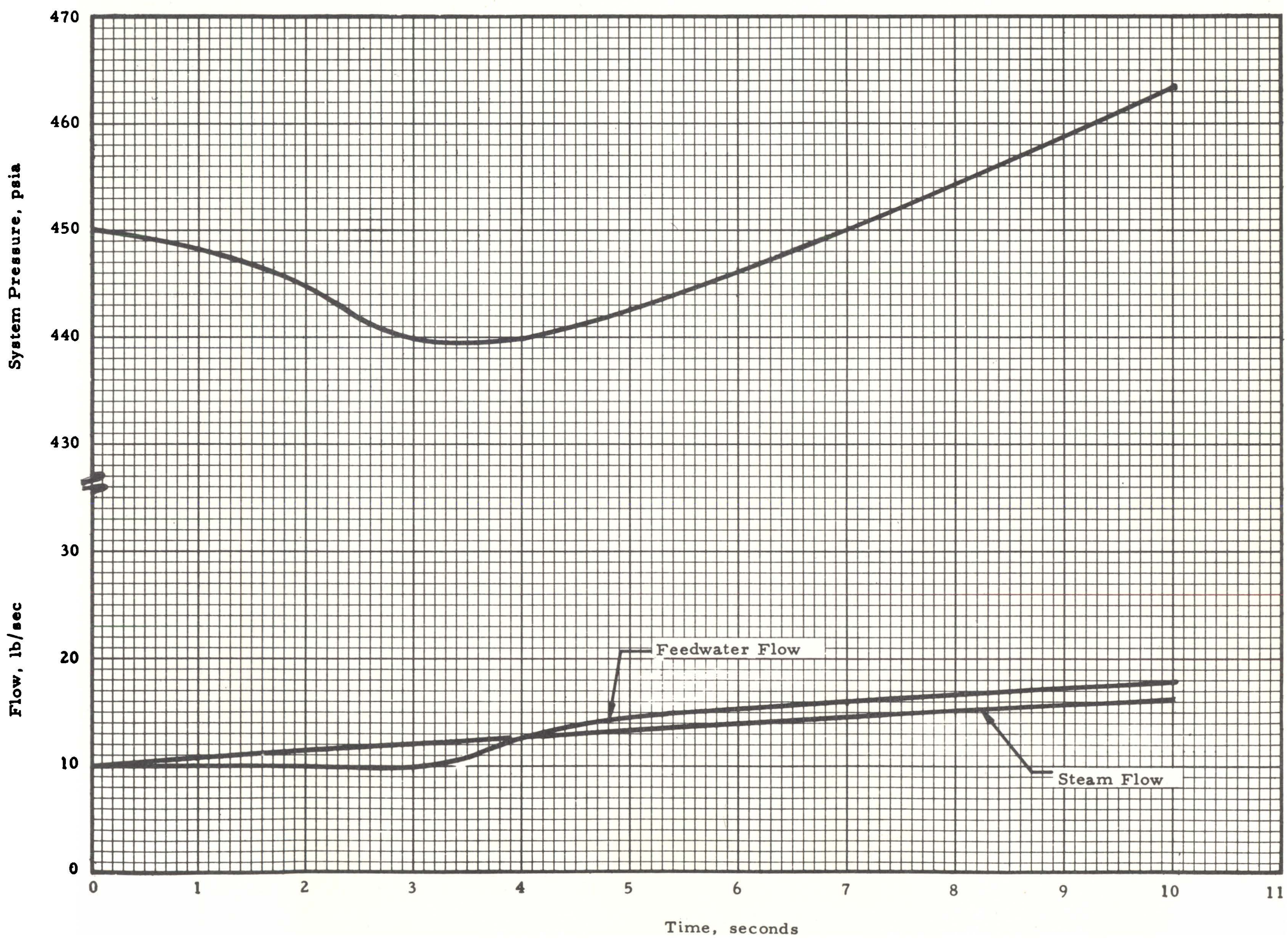


Figure 8.6-10. Feedwater Ramps, 10 to 60 lb/sec and 60 to 10 lb/sec in 80 seconds: Approximate Response of Superheater No. 2 Length and Boiling Section Outlet Temperature for Self-Regulating Reactor. Superheater Primary Inlet Temperature Drops from 520 to 500 F in 100 seconds

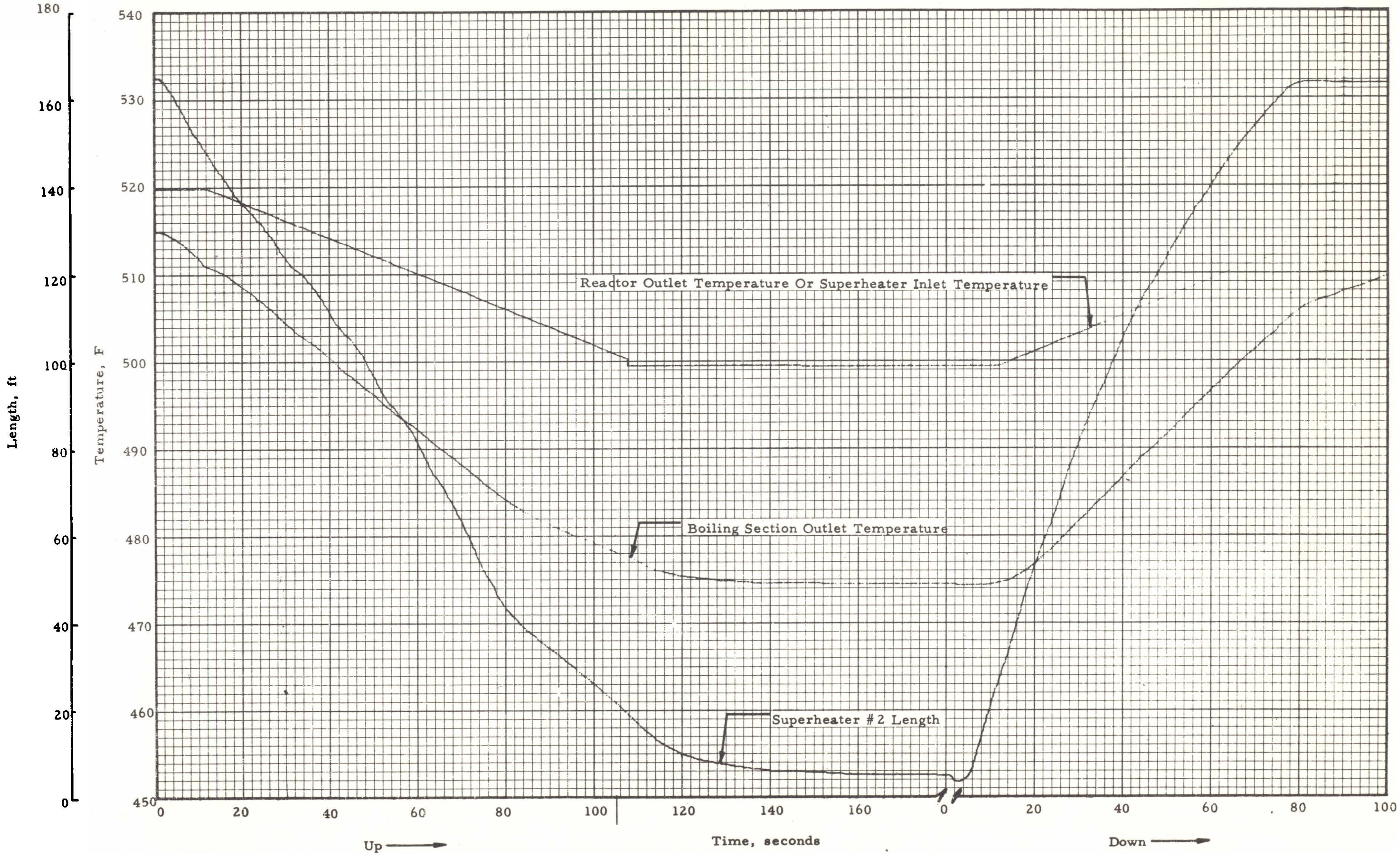
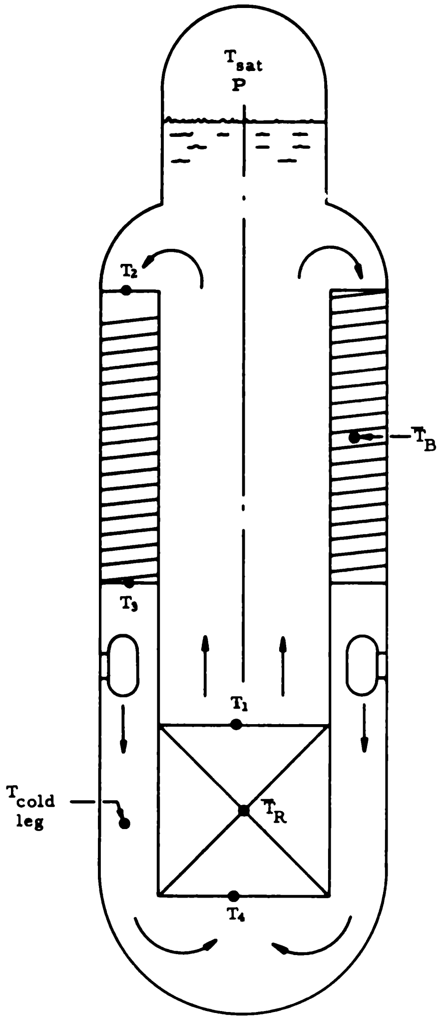


Figure 8.7-1. Basic Model of CNSG



Definition of Symbols

n	- neutron level
P	- system operating pressure
Q_D	- heat demand (Load applied to primary side of steam generator)
Q_P	- reactor heat input to coolant
$^*T_1$	- bulk coolant temperature, reactor outlet
$^*T_2$	- bulk coolant temperature, boiler inlet
T_3	- bulk coolant temperature, boiler outlet
T_4	- bulk coolant temperature, reactor inlet
$\bar{T}_B$	- average coolant temperature in steam generator
$\bar{T}_R$	- average coolant temperature in reactor
$T_{cold\ leg}$	- average coolant temperature between steam generator outlet and reactor inlet
$^*T_H$	- average coolant temperature between reactor outlet and steam generator inlet
$T_{cold\ mass}$	- average temperature of all fluid between steam generator inlet and reactor outlet
T_{sat}	- saturation temperature
$\bar{T}_{fuel}$	- average fuel temperature
$\delta k_{Doppler}$	- reactivity change due to change in average fuel temperature
δk_{temp}	- reactivity change due to change in average temperature of moderator
δk_{void}	- reactivity change due to change in average core void fraction
δk_{rod}	- reactivity change due to control rods.

*For power increase (pressure decrease):

$$T_1 = T_2 = T_H = T_{sat}$$

Figure 8.7-2. Block Diagram of CNSG Power Decrease Model

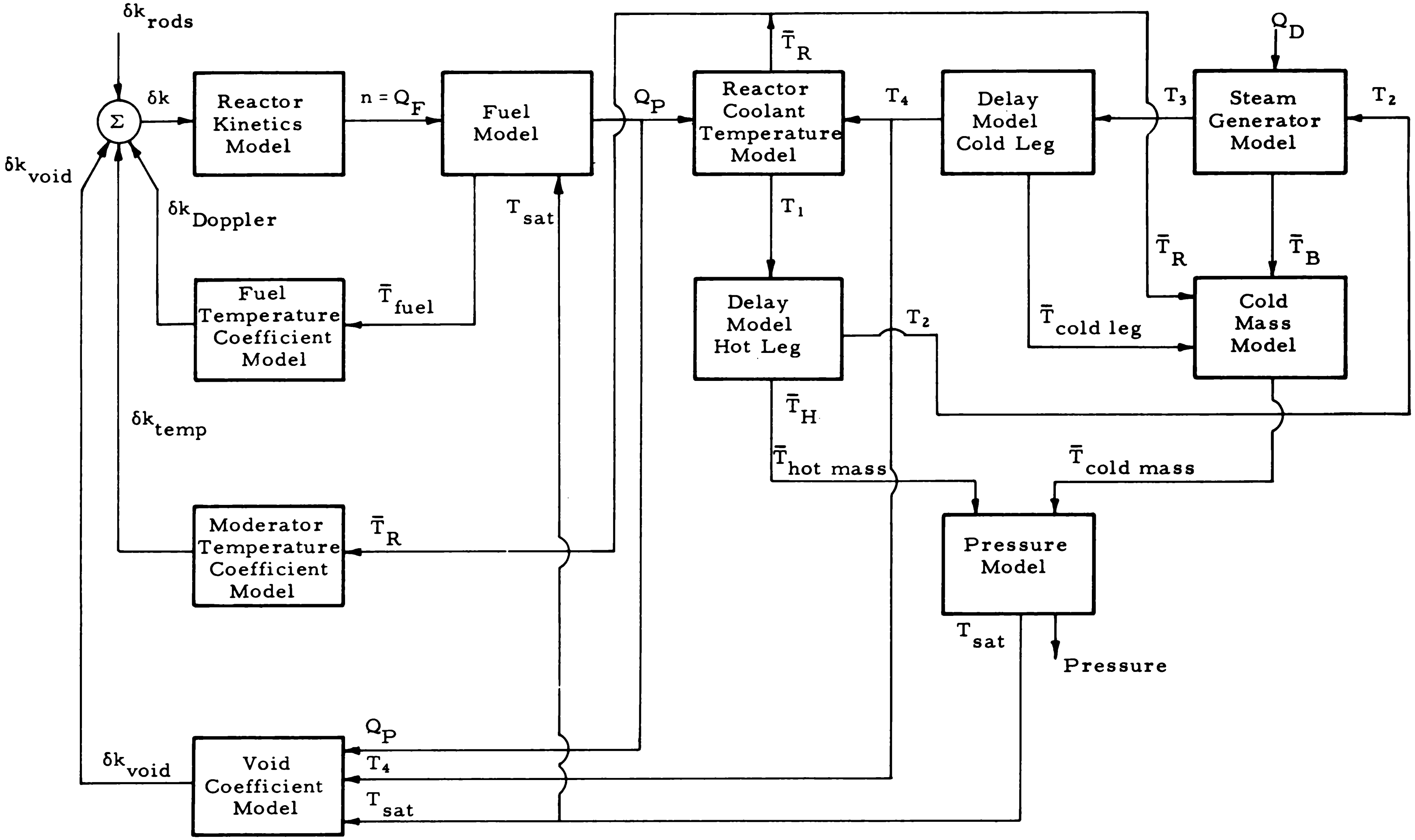


Figure 8.7-3. Block Diagram of CNSG Power Increase Model

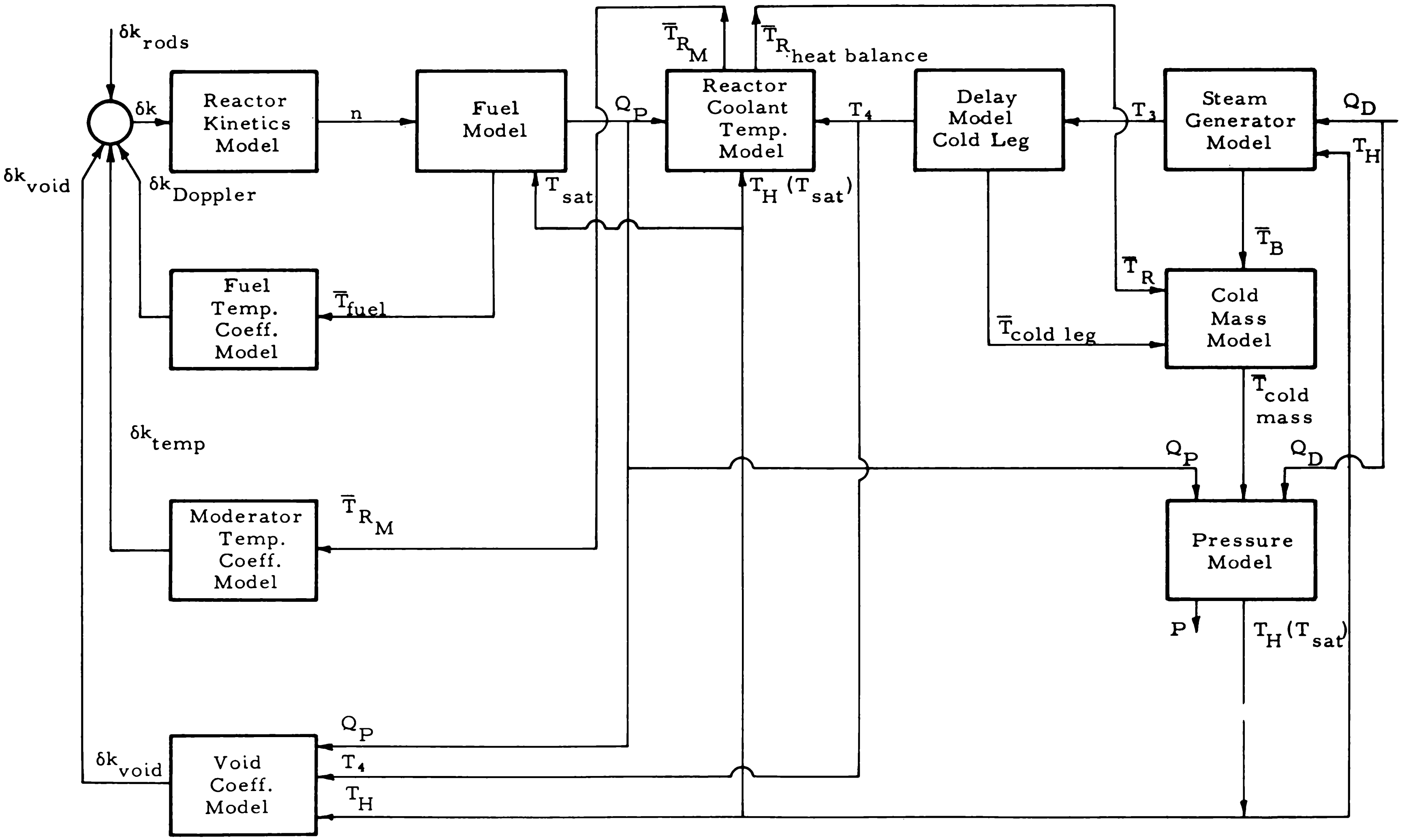


Figure 8.7-5. Analog Computer Wiring Diagram of CNSG Power Increase Model

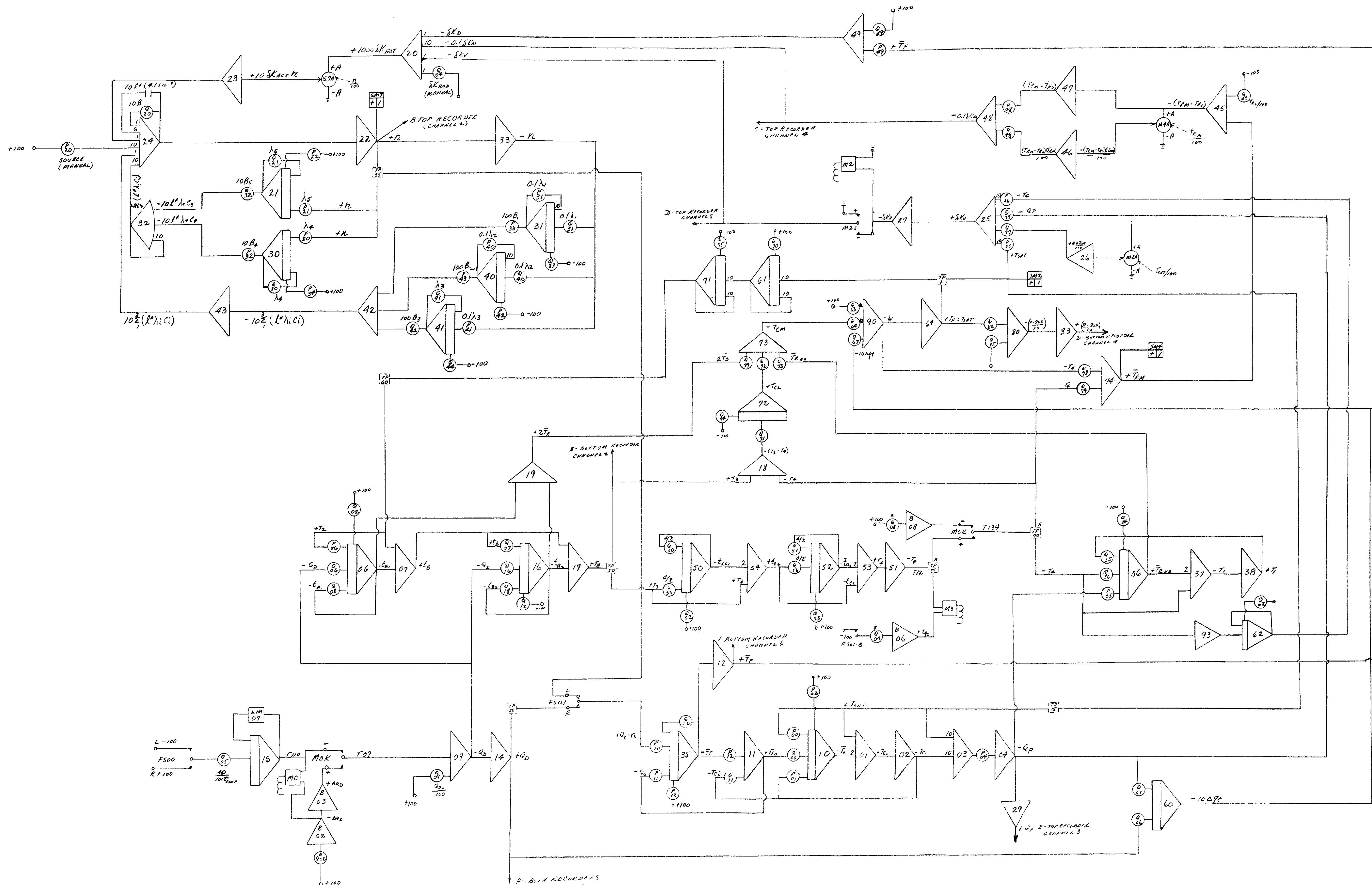


Figure 8.7-6. Average Void Fraction Vs Reactor Power for Different Inlet Temperatures — 812 psia

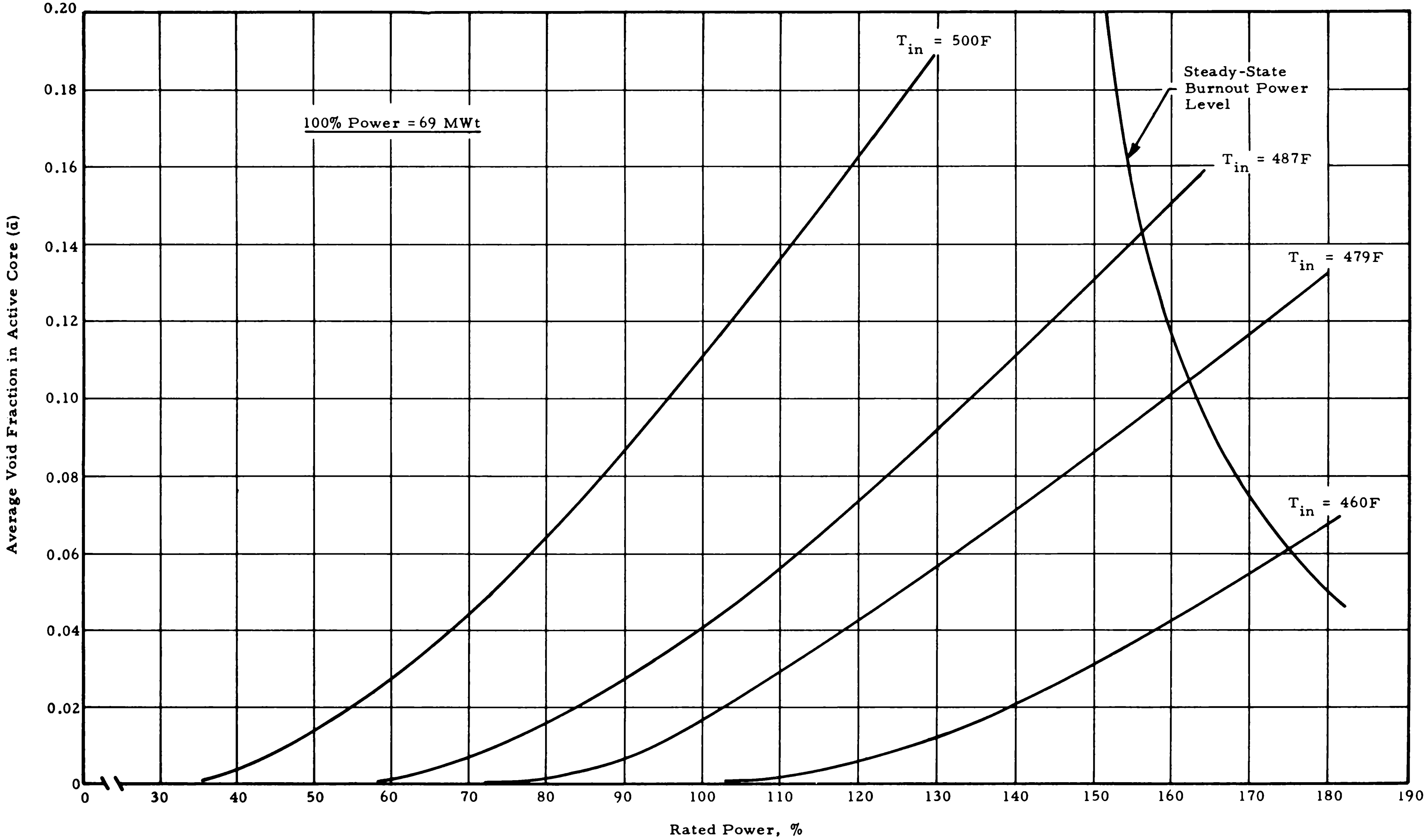


Figure 8.7-7. Average Void Fraction Vs Reactor Power for Different Inlet Temperatures — 600 psia

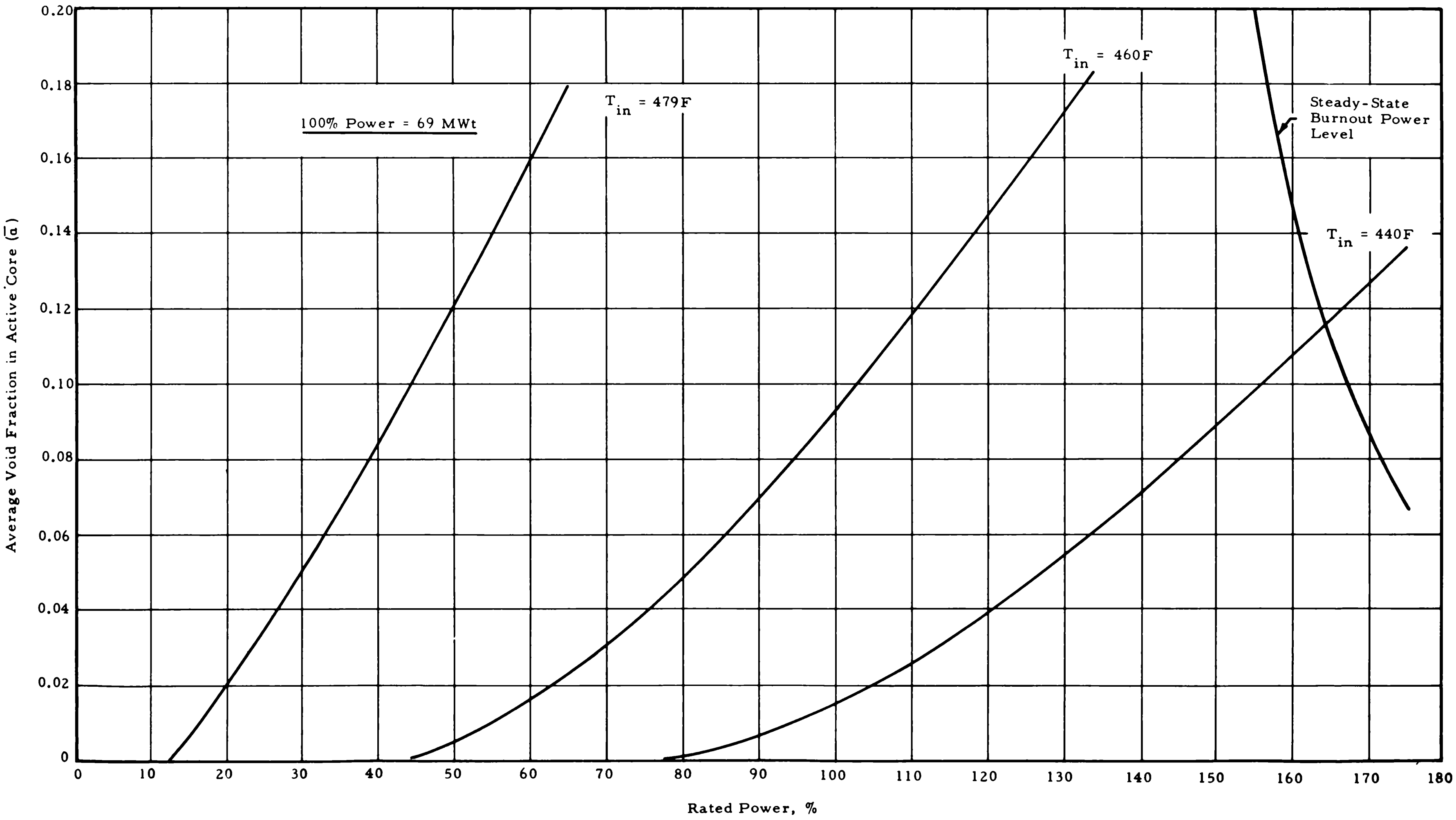


Figure 8.7-8. Average Void Fraction Vs Reactor Power for Different Inlet Temperatures — 1045 psia

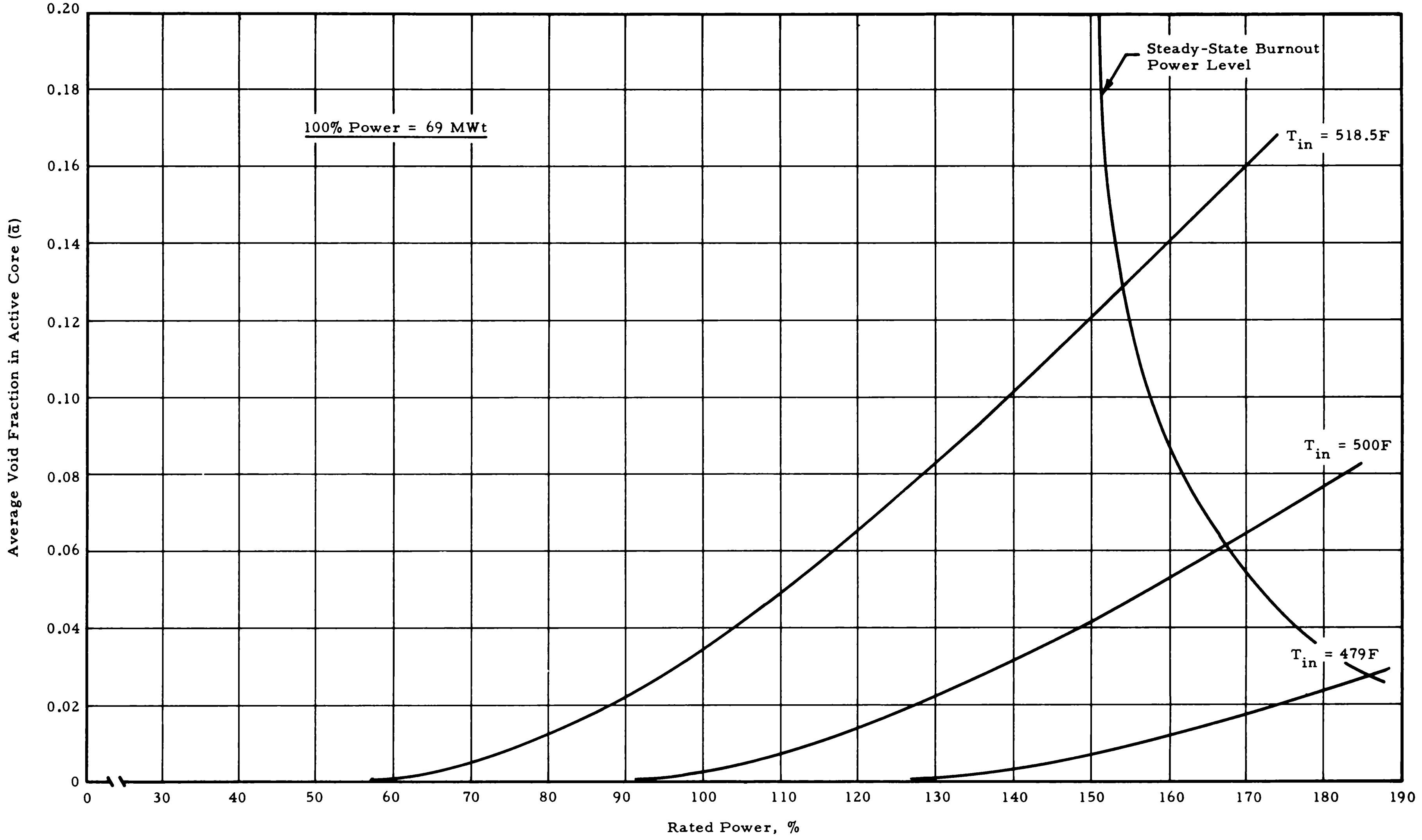


Figure 8.7-9. Reactivity in Voids Vs Average Core Void Fraction

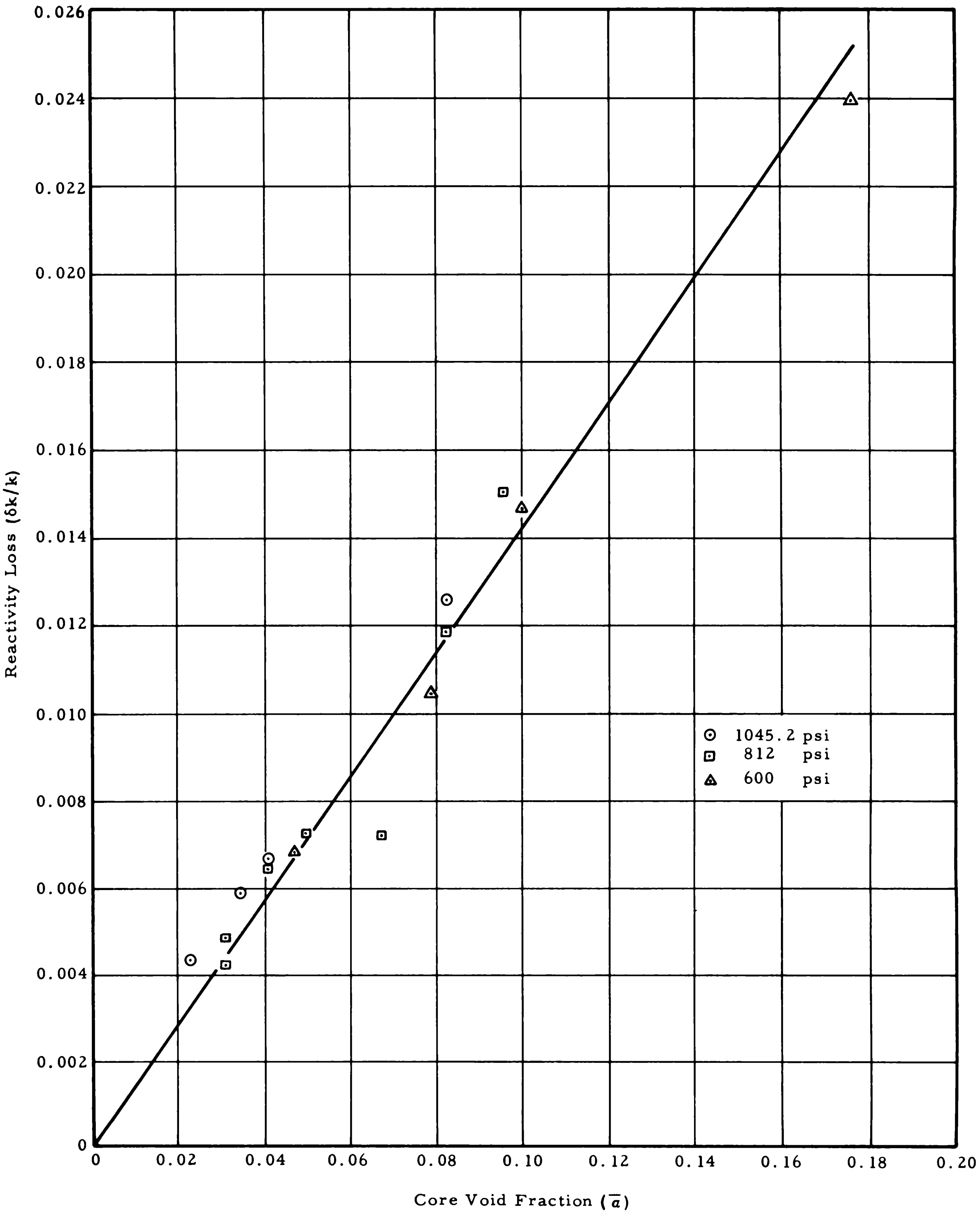


Figure 8.7-10. Void Reactivity Loss Vs Reactor Power

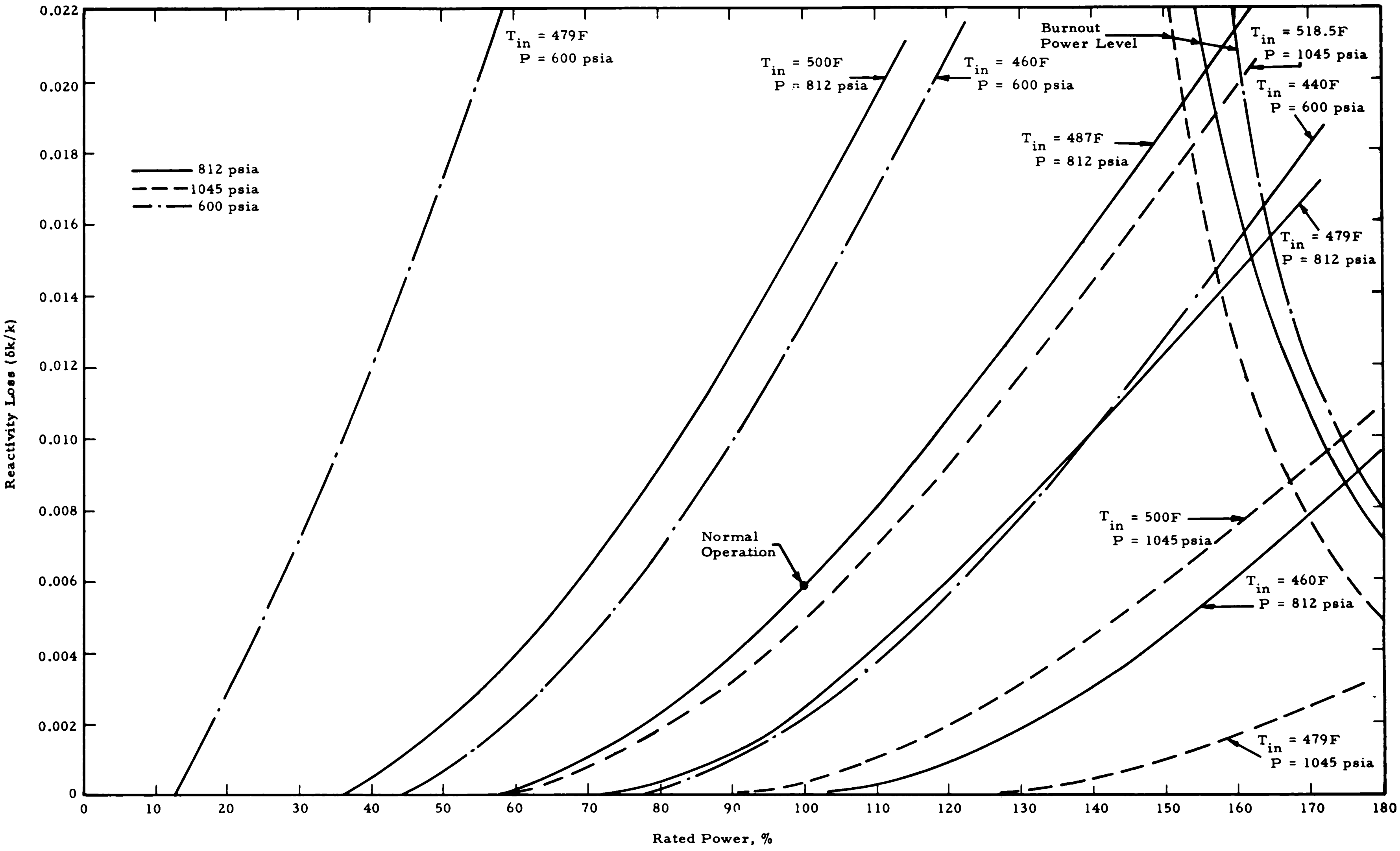


Figure 8.7-11. Void Reactivity Loss Vs Inlet Subcooling

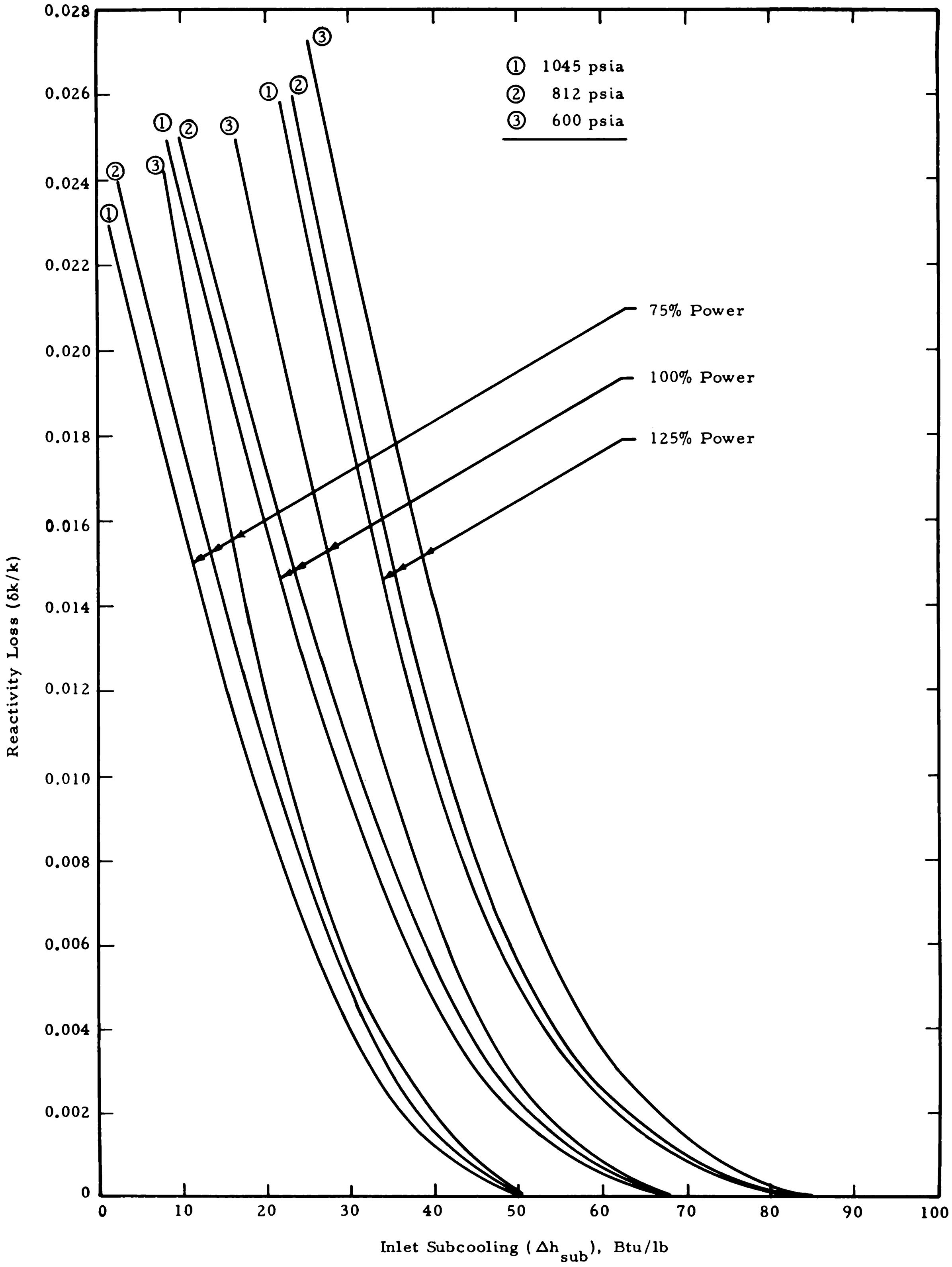


Figure 8.7-12 A. 100% to 20% Power in 80 seconds — Nominal EOL Parameters, V_{steam} = 280 cu ft

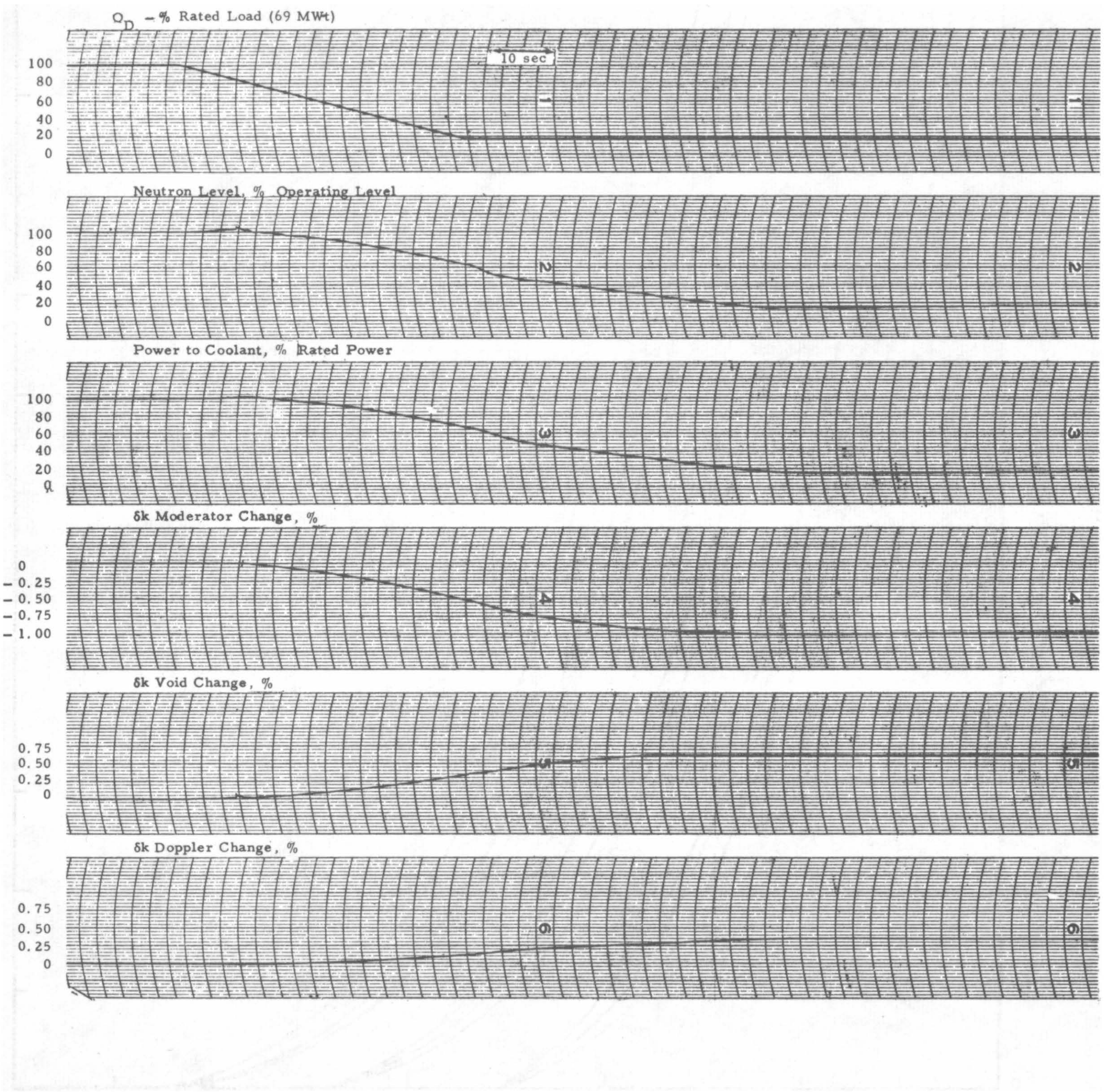


Figure 8.7-12 B. 100% to 20% Power in 80 seconds — Nominal EOL Parameters, V_{steam} = 280 cu ft

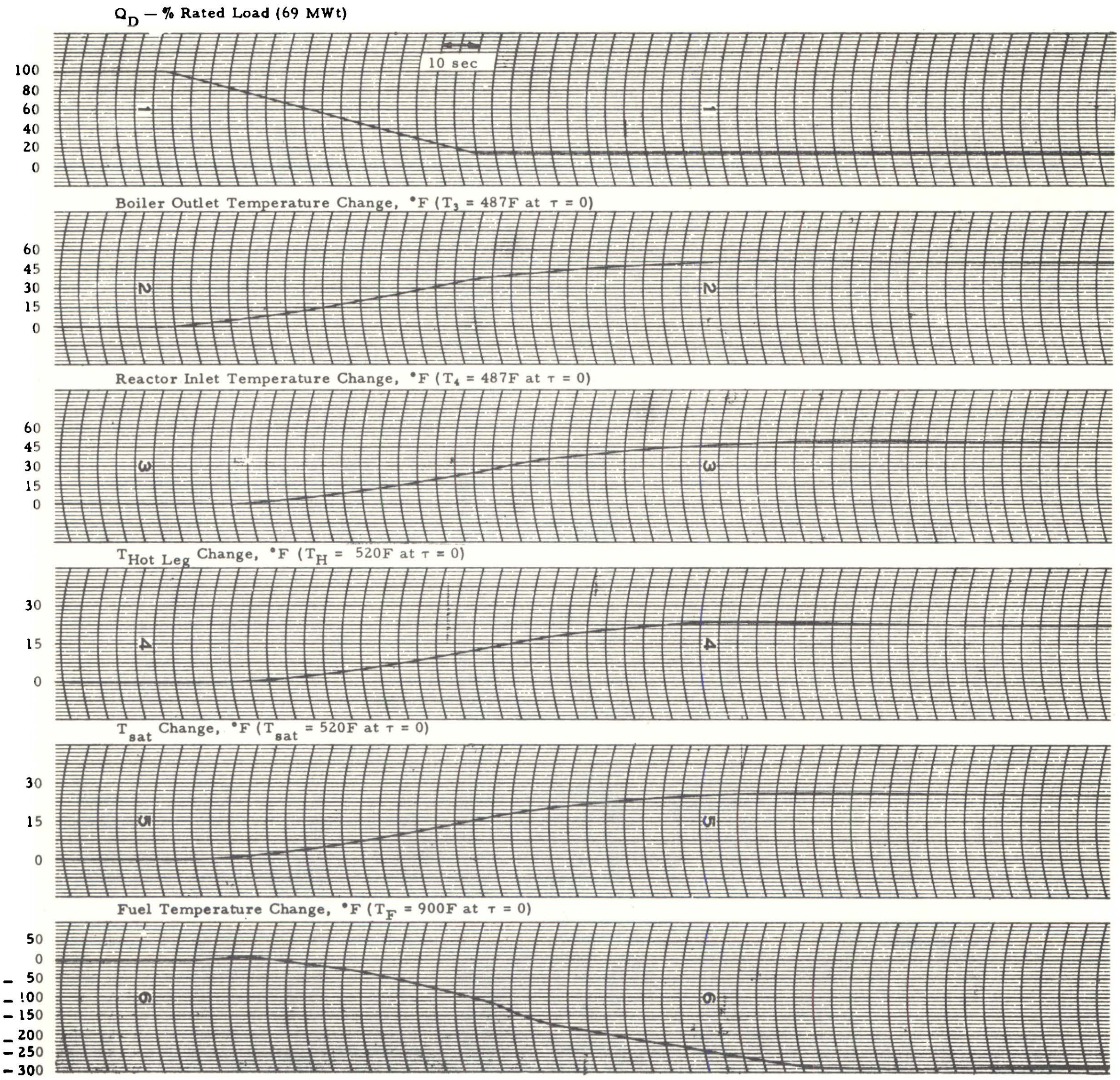


Figure 8.7-13 A. 100% to 20% Power Step — Nominal EOL Parameters, V_{steam} = 280 cu ft

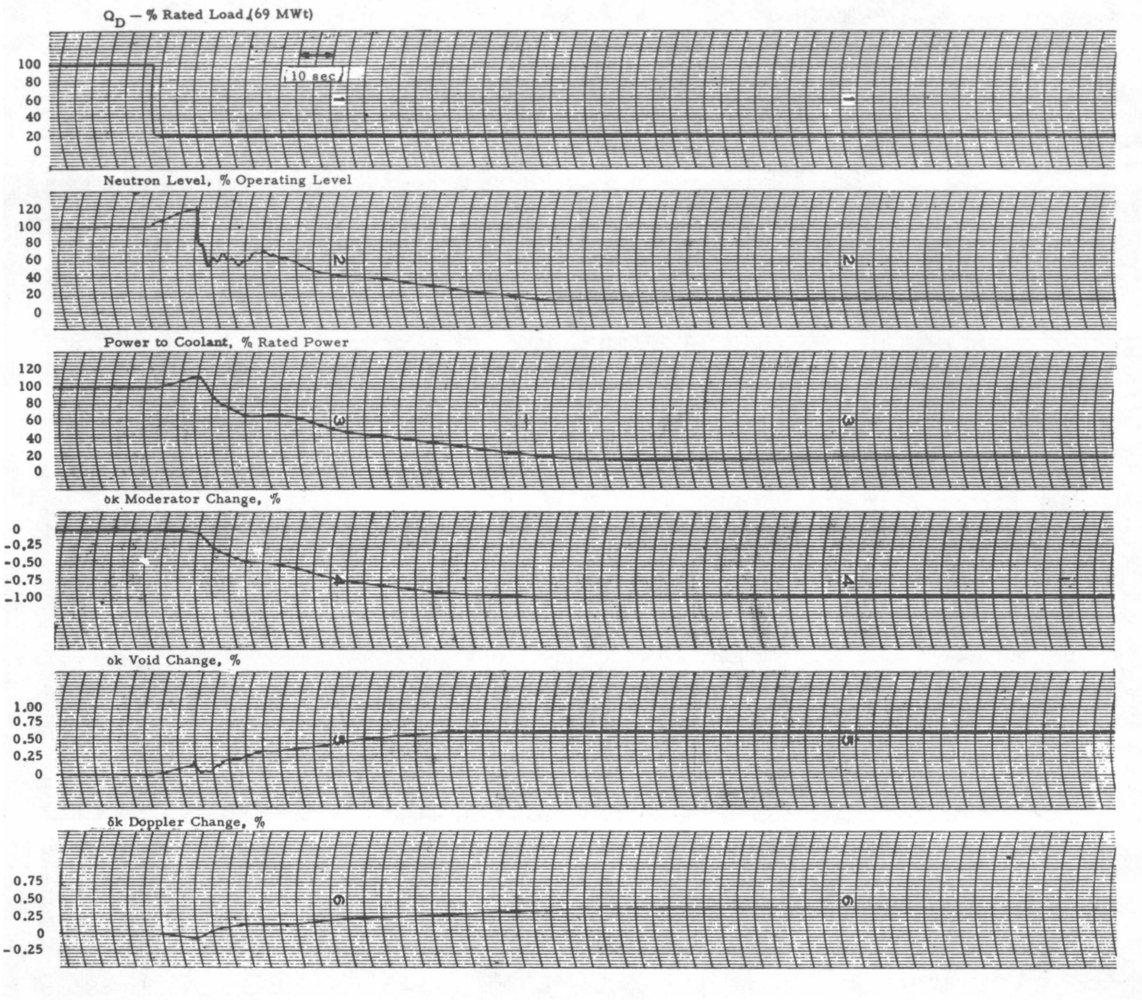


Figure 8.7-13 B. 100% to 20% Power Step — Nominal EOL Parameters, V_{steam} = 280 cu ft

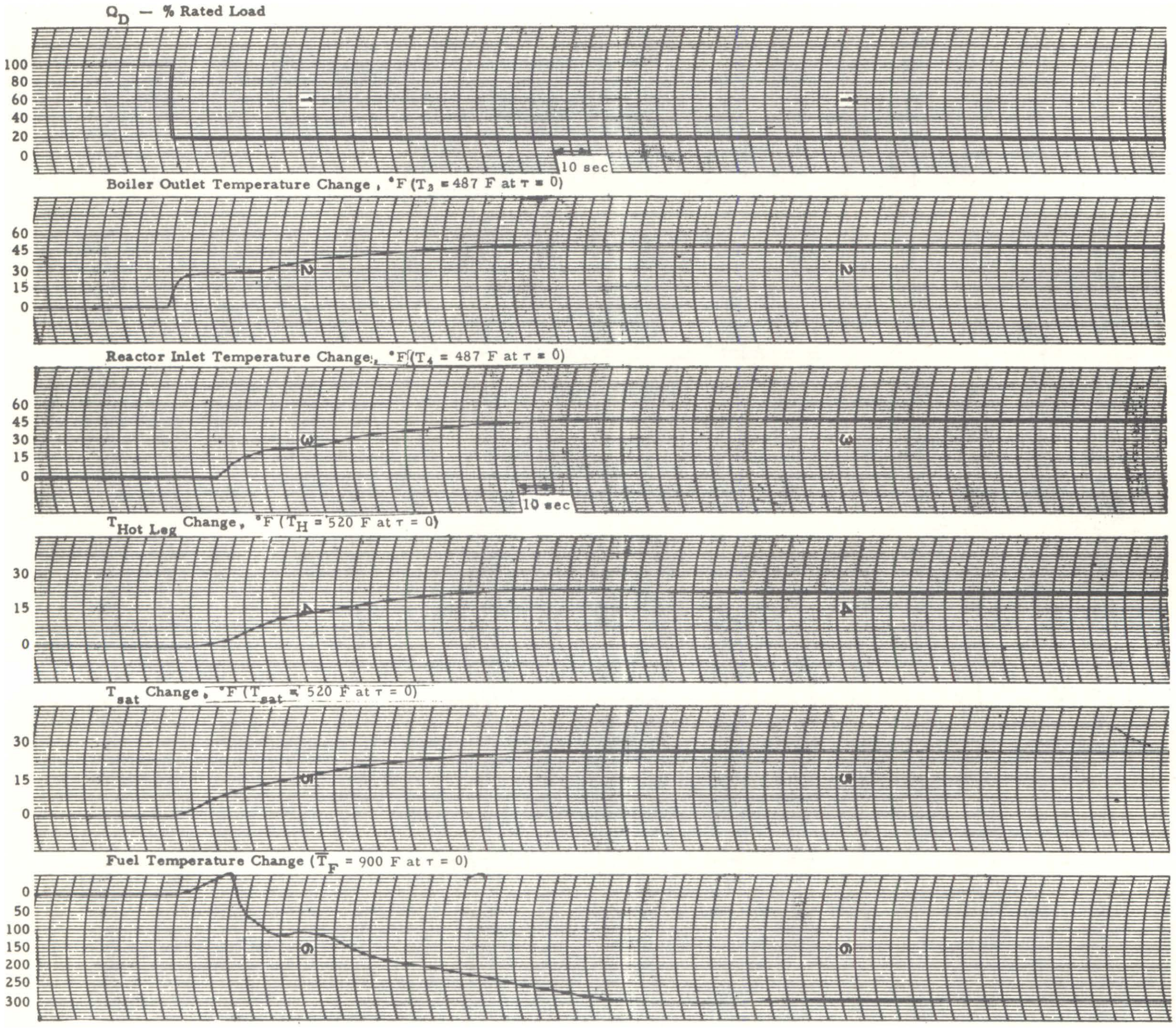


Figure 8.7-14 A. 20% to 100% Power in 80 seconds — Nominal EOL Parameters, V_{steam} = 210/280 cu ft

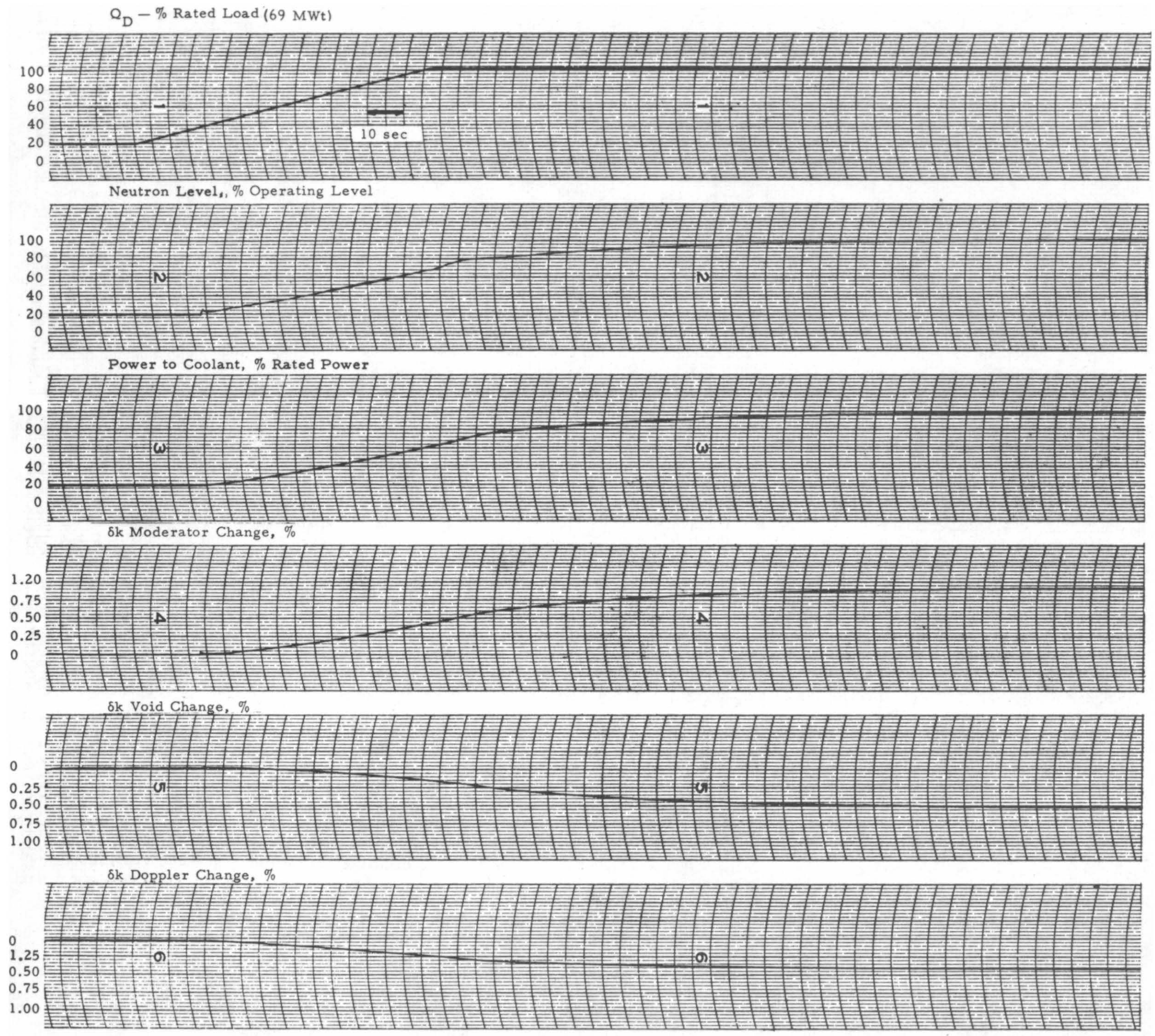


Figure 8.7-14 B. 20% to 100% Power in 80 seconds — Nominal EOL Parameters, V_{steam} = 210/280 cu ft

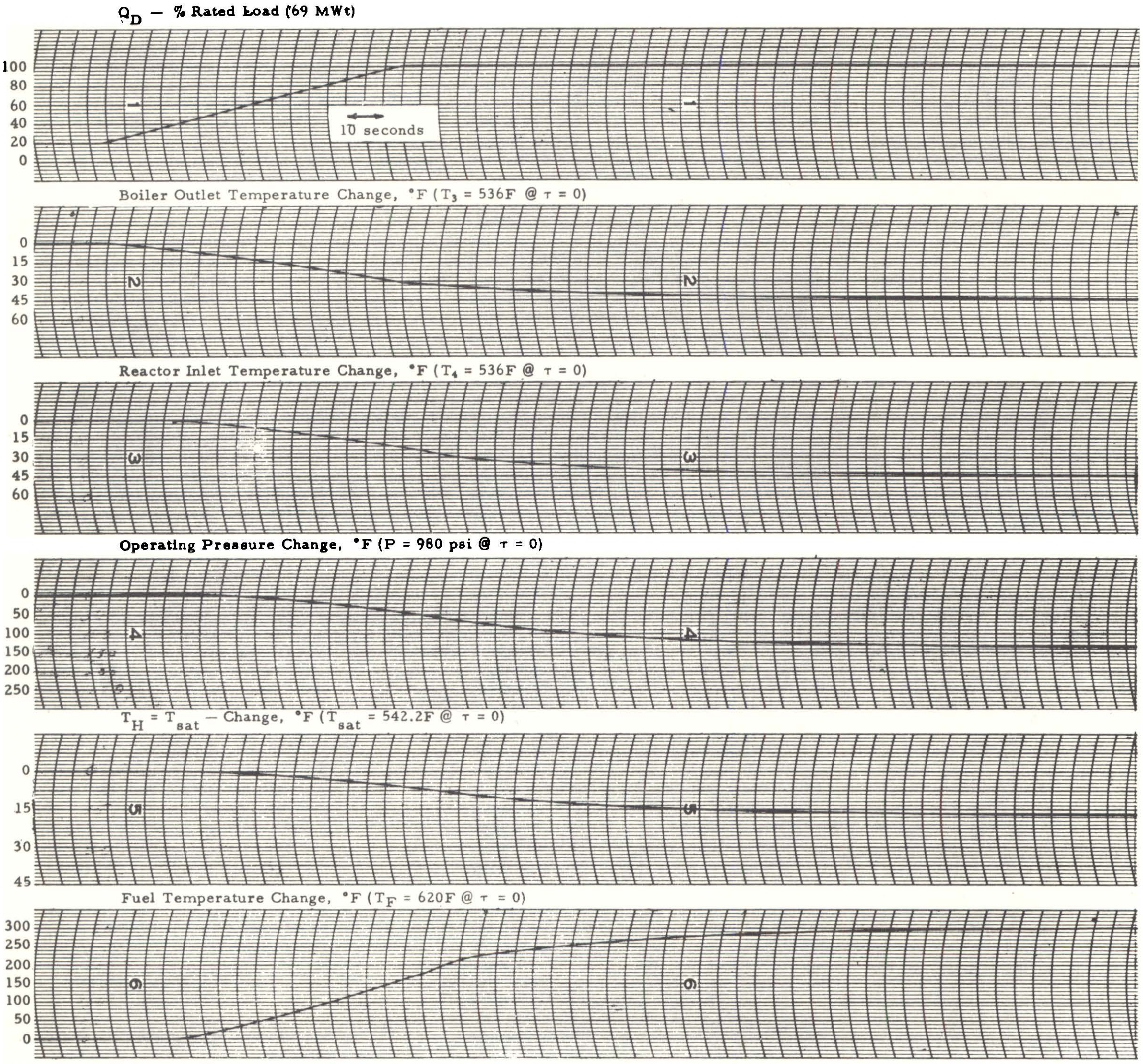


Figure 8.7-15 A. 20% to 100% Power Step — Nominal EOL Parameters, V_{steam} = 210/280 cu ft

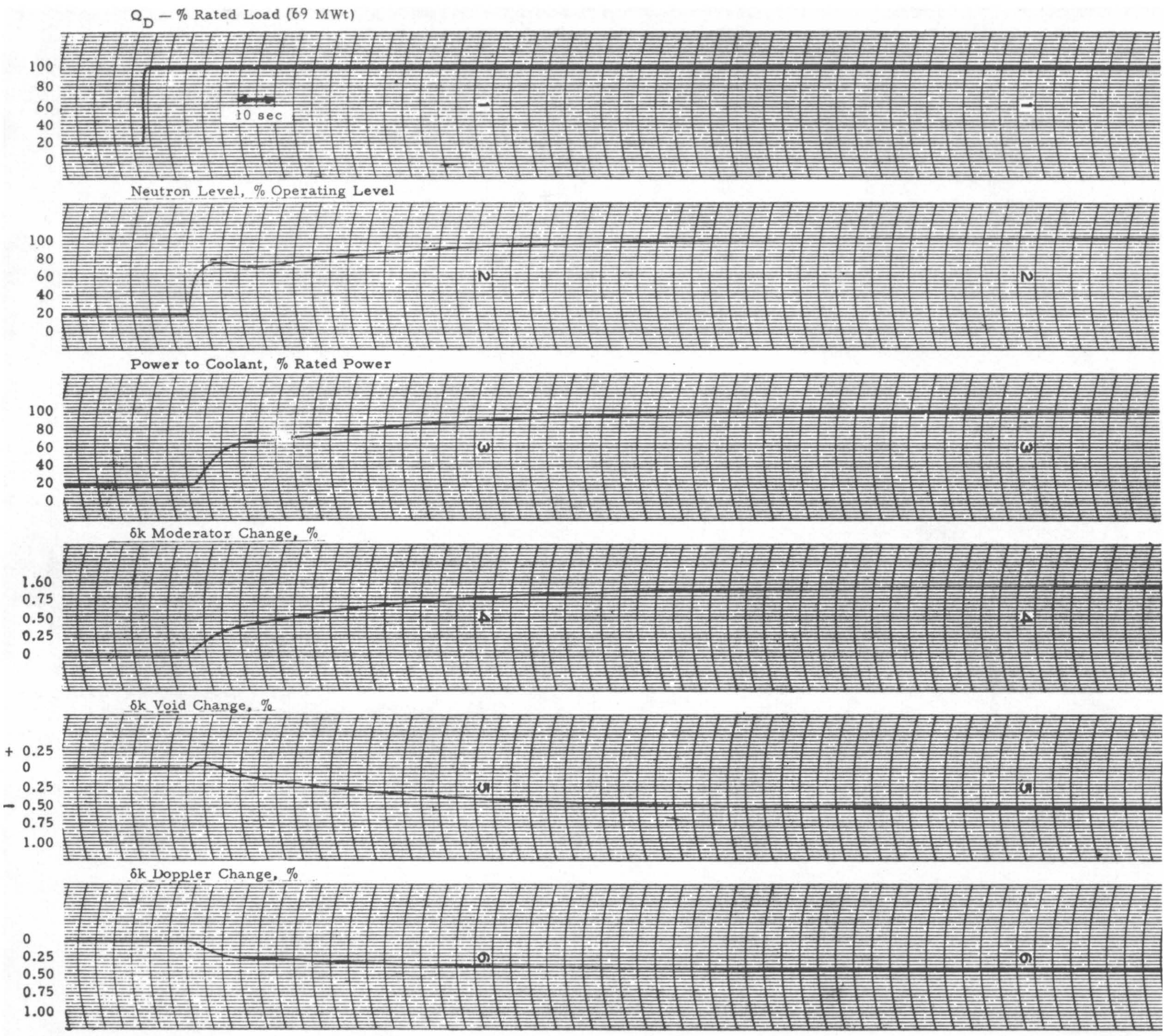


Figure 8.7-15 B. 20% to 100% Power Step — Nominal EOL Parameters, V_{steam} = 210/280 cu ft

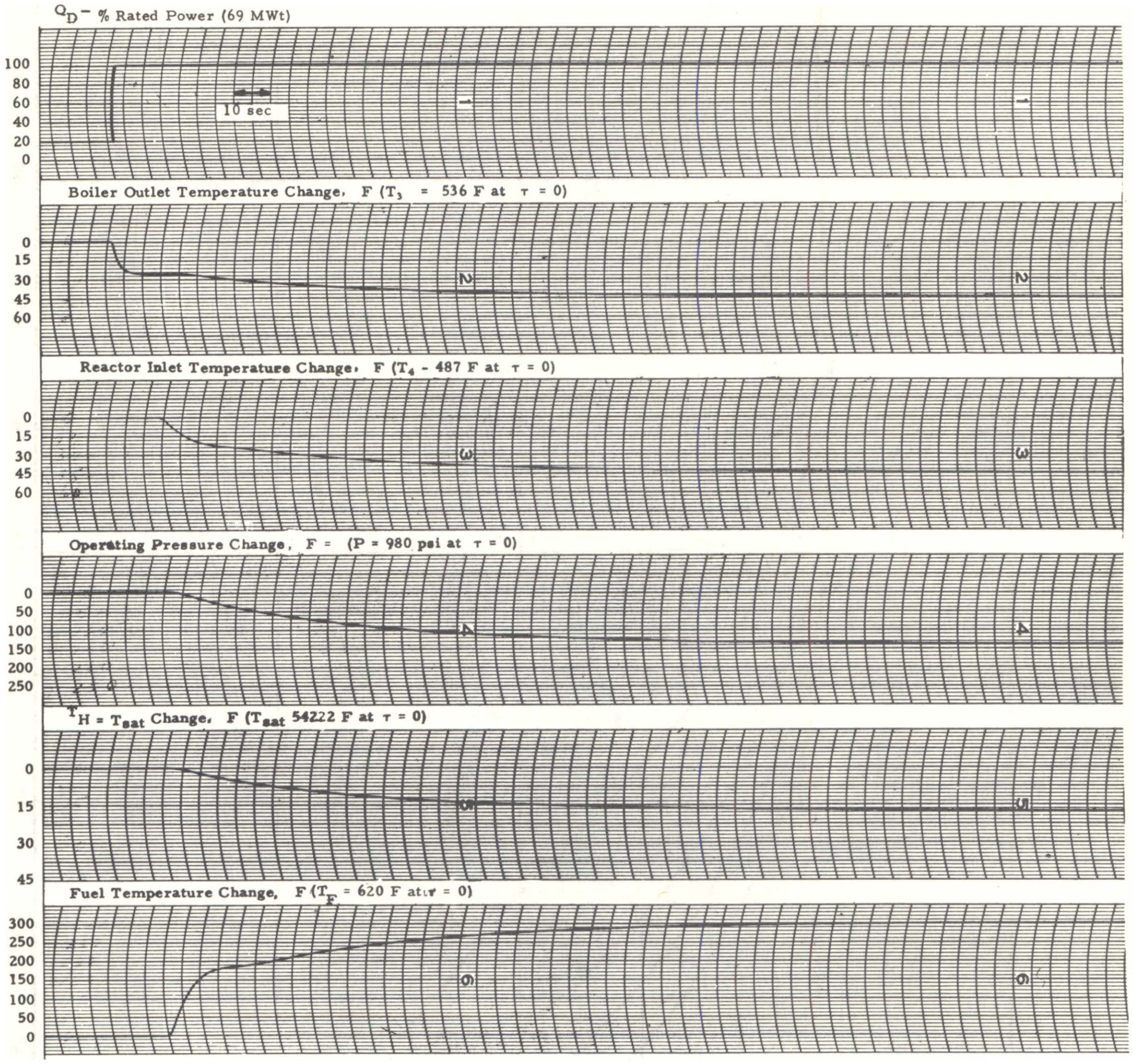


Figure 8.7-16. Effect of Transient Time on Operating Pressure — Power Decrease

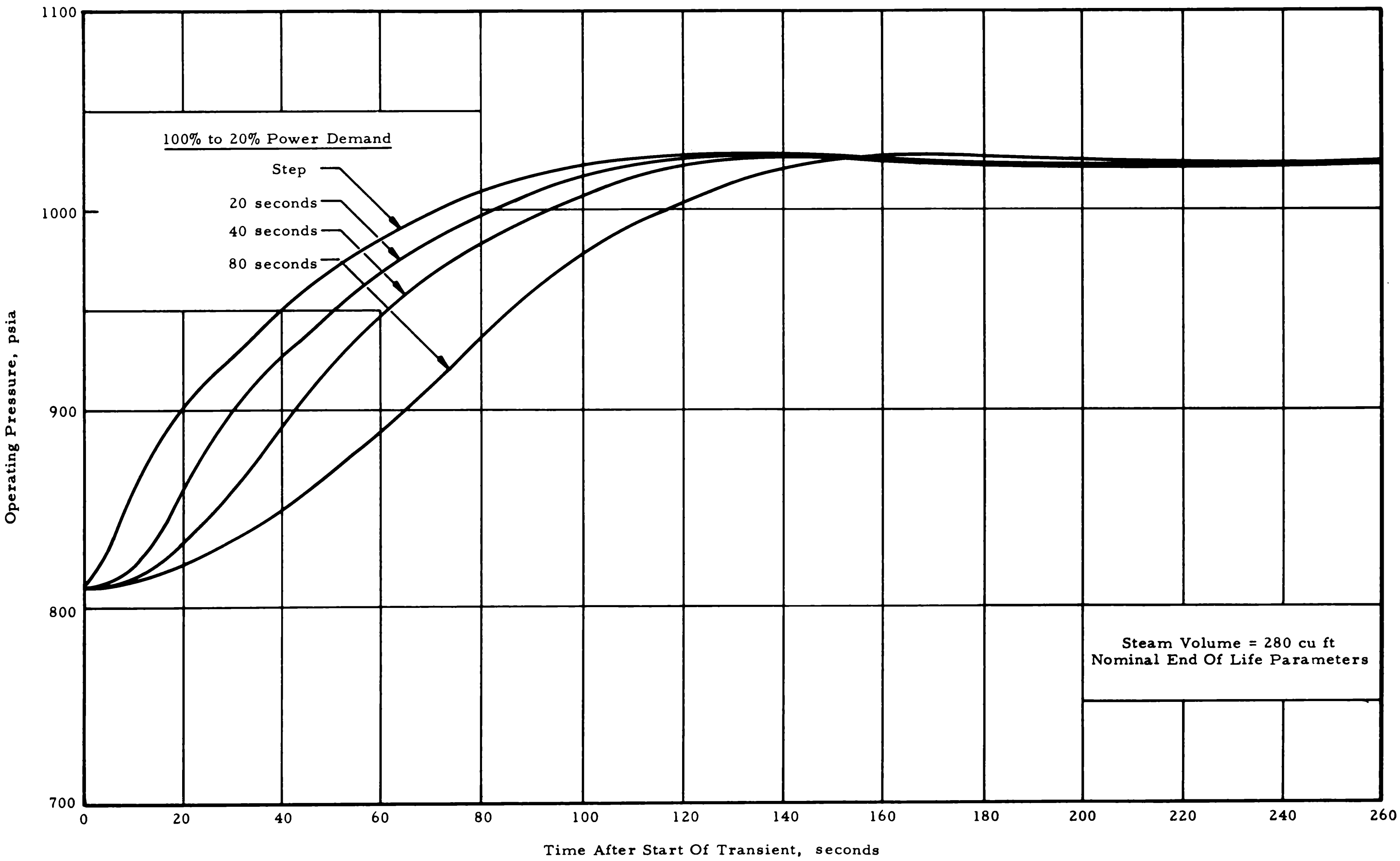


Figure 8.7-17. Effect of Transient Time on Operating Pressure — Power Increase

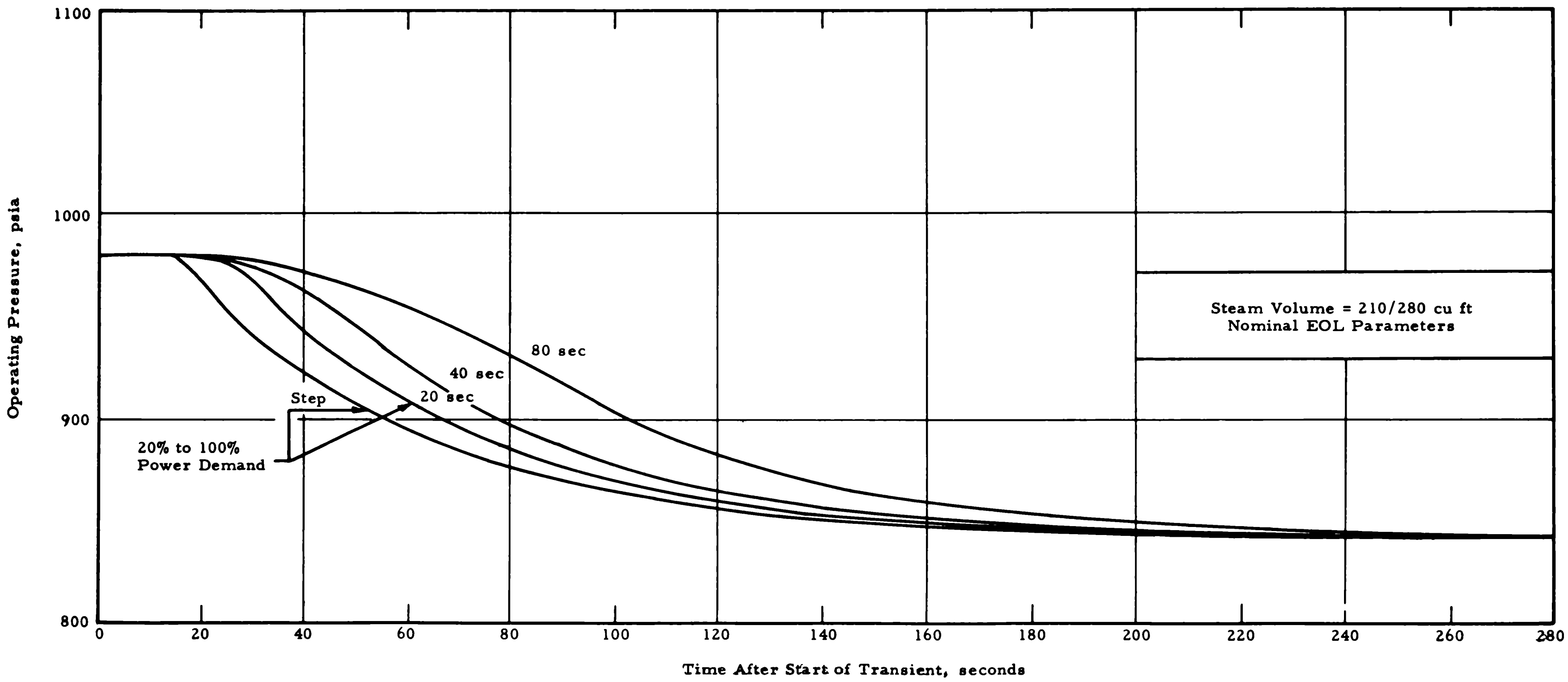


Figure 8.7-18. Effect of Steam Volume on Operating Pressure — Power Decrease

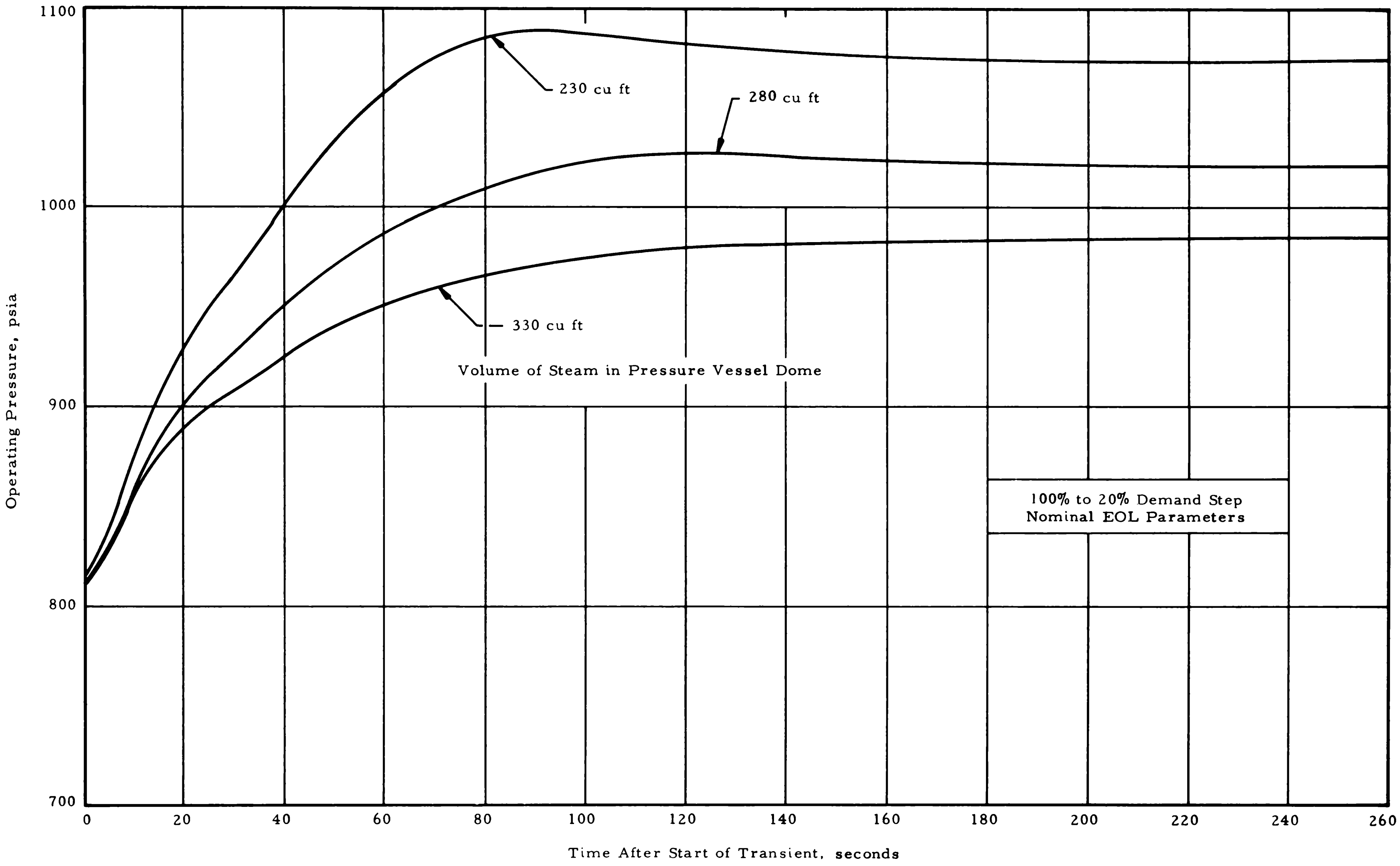


Figure 8.7-19. Effect of Steam Volume on Operating Pressure — Power Increase

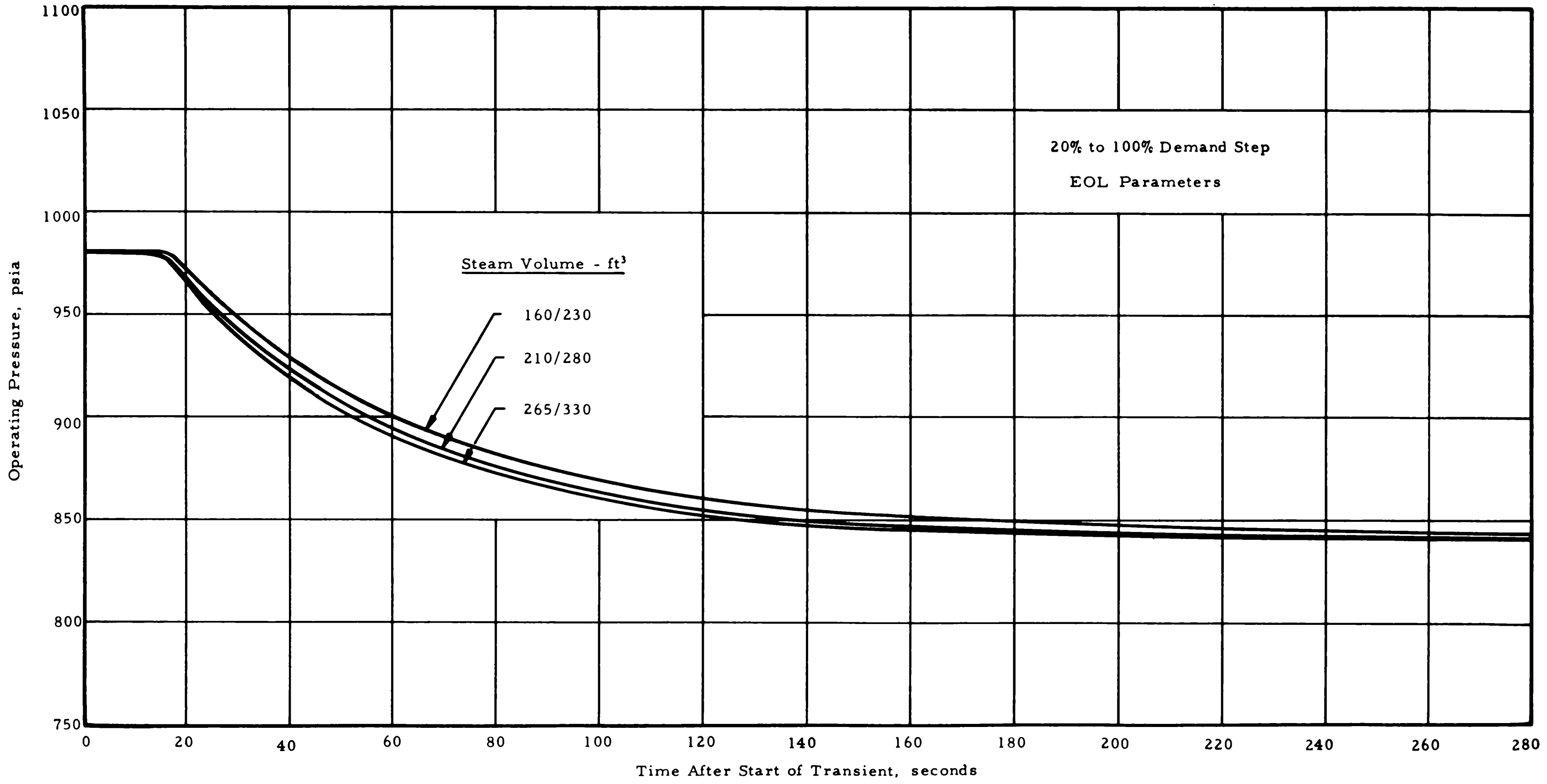


Figure 8.7-20 A. 100% to 20% Power Step — EOL Parameters with $\delta k_v = 3 \times$ Nominal, $V_{steam} = 280$ cu ft

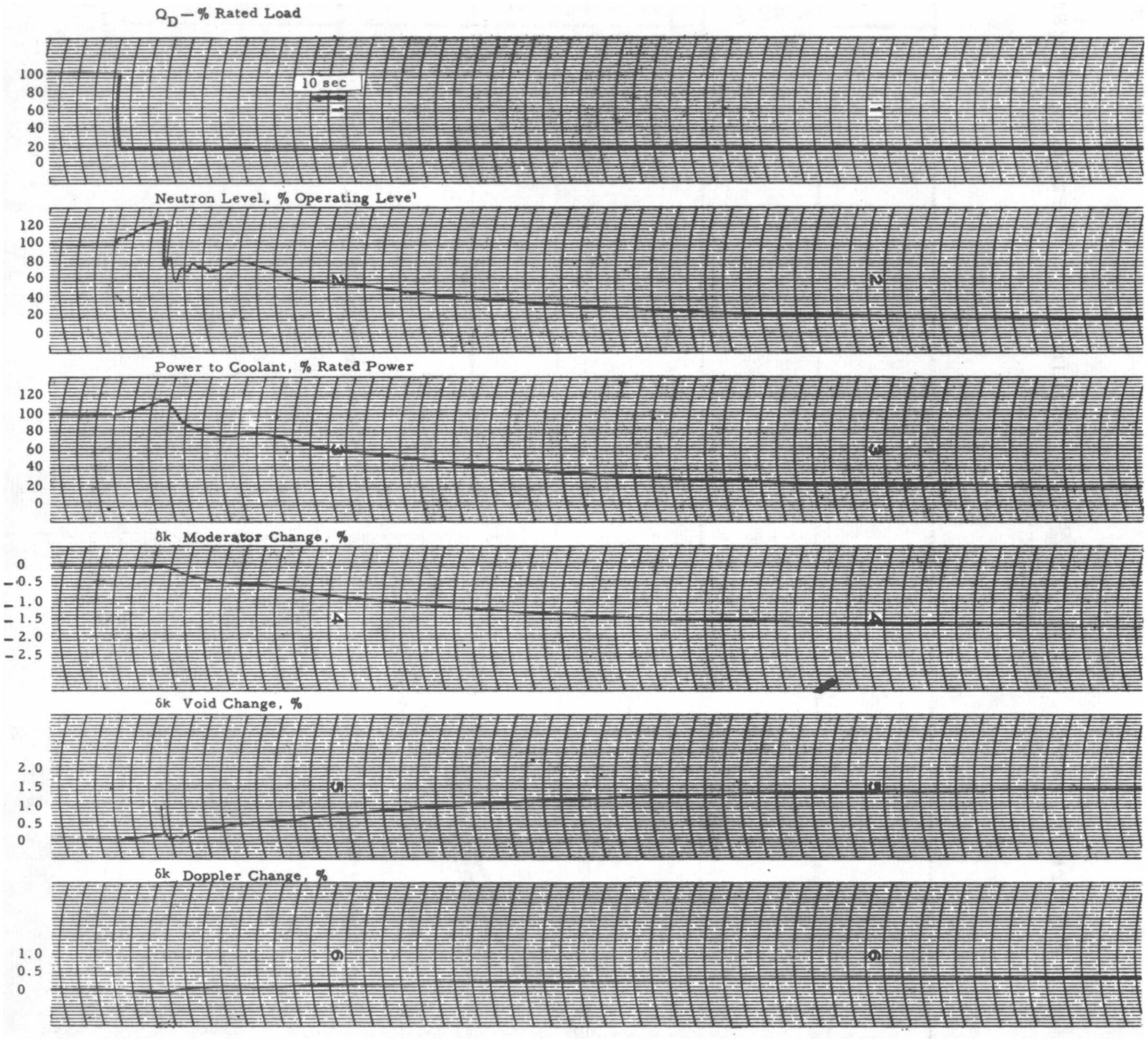


Figure 8.7-20 B. 100% to 20% Power Step — EOL Parameters with δk_v = 3 × Nominal, V_{steam} = 280 cu ft

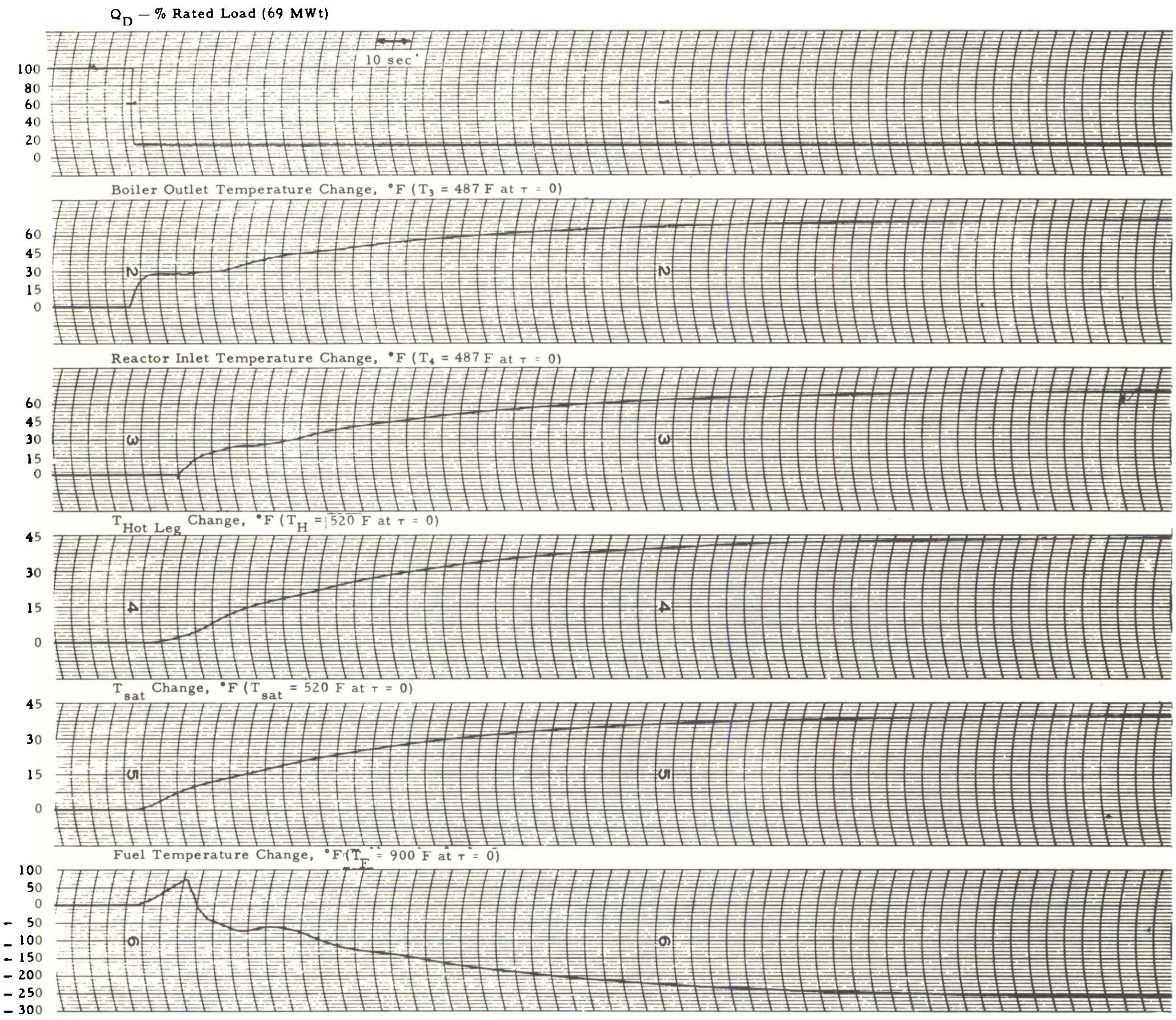


Figure 8.7-21. Effect of Void Reactivity on Operating Pressure — Power Decrease

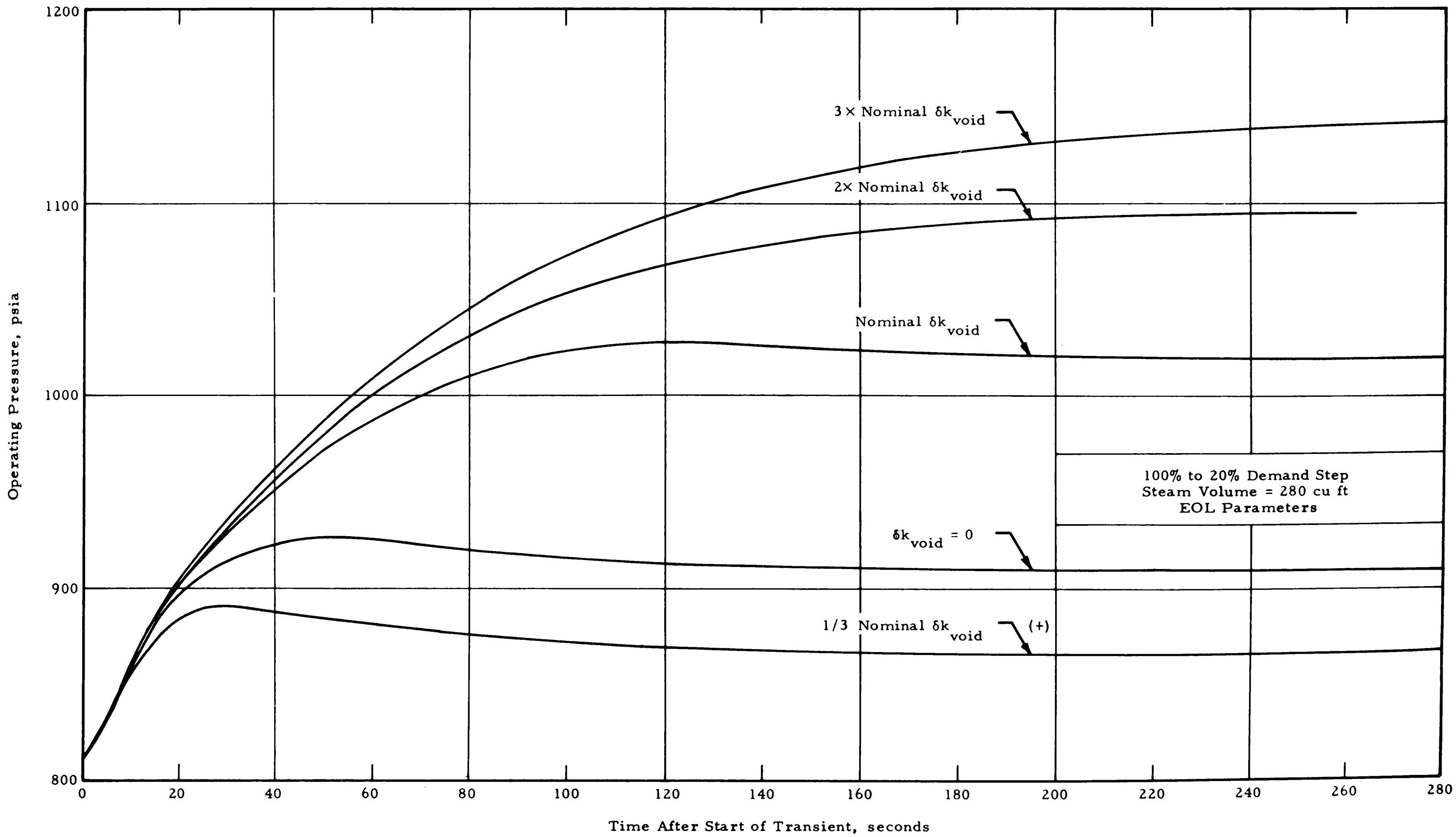


Figure 8.7-22 A. 20% to 100% Power Step — EOL Parameters with δk_v = 3 × Nominal, V_{steam} = 210/280 cu ft

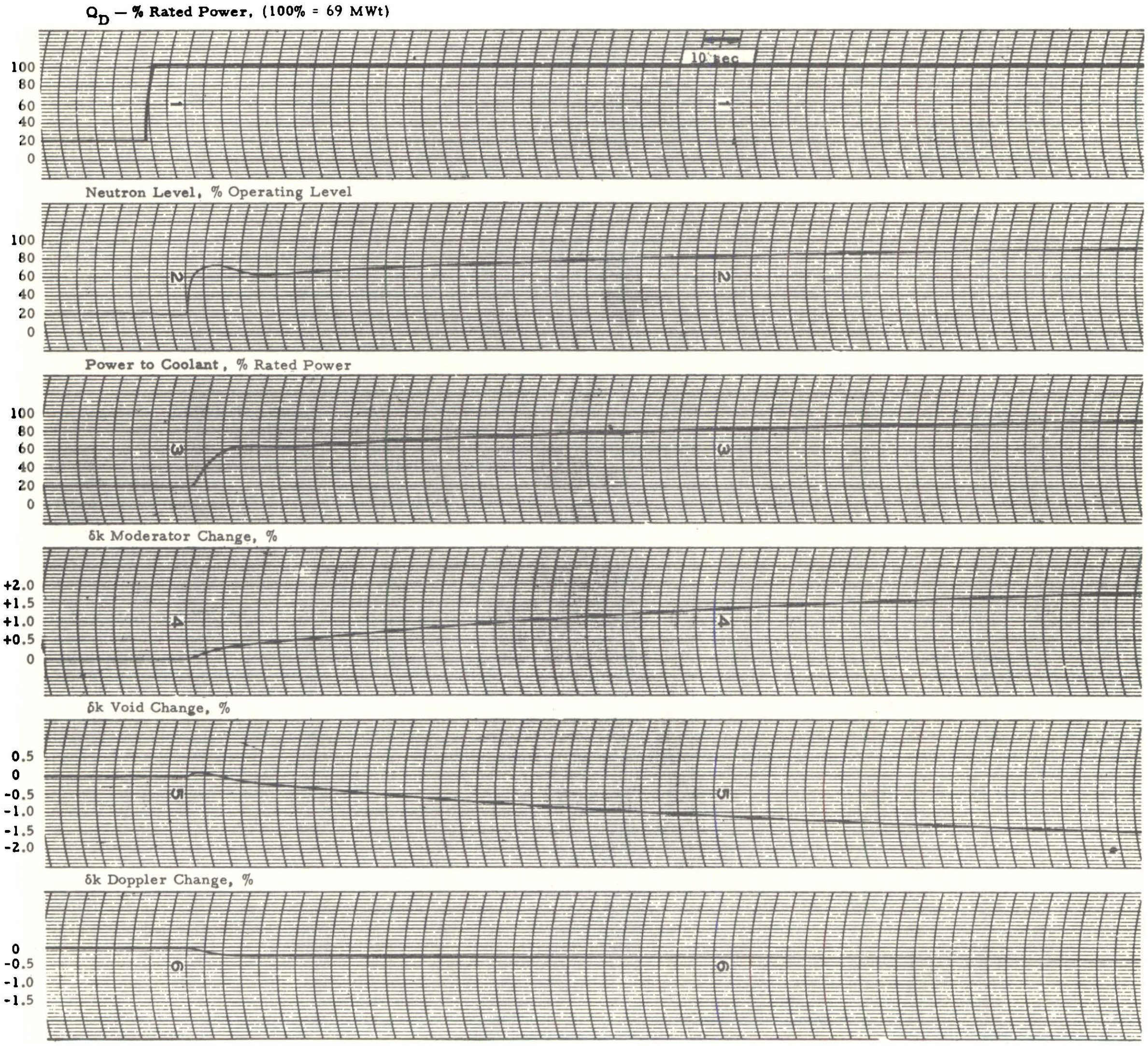


Figure 8.7-22 B. 20% to 100% Power Step — EOL Parameters with $\delta k_v = 3 \times$ Nominal, $V_{steam} = 210/280$ cu ft

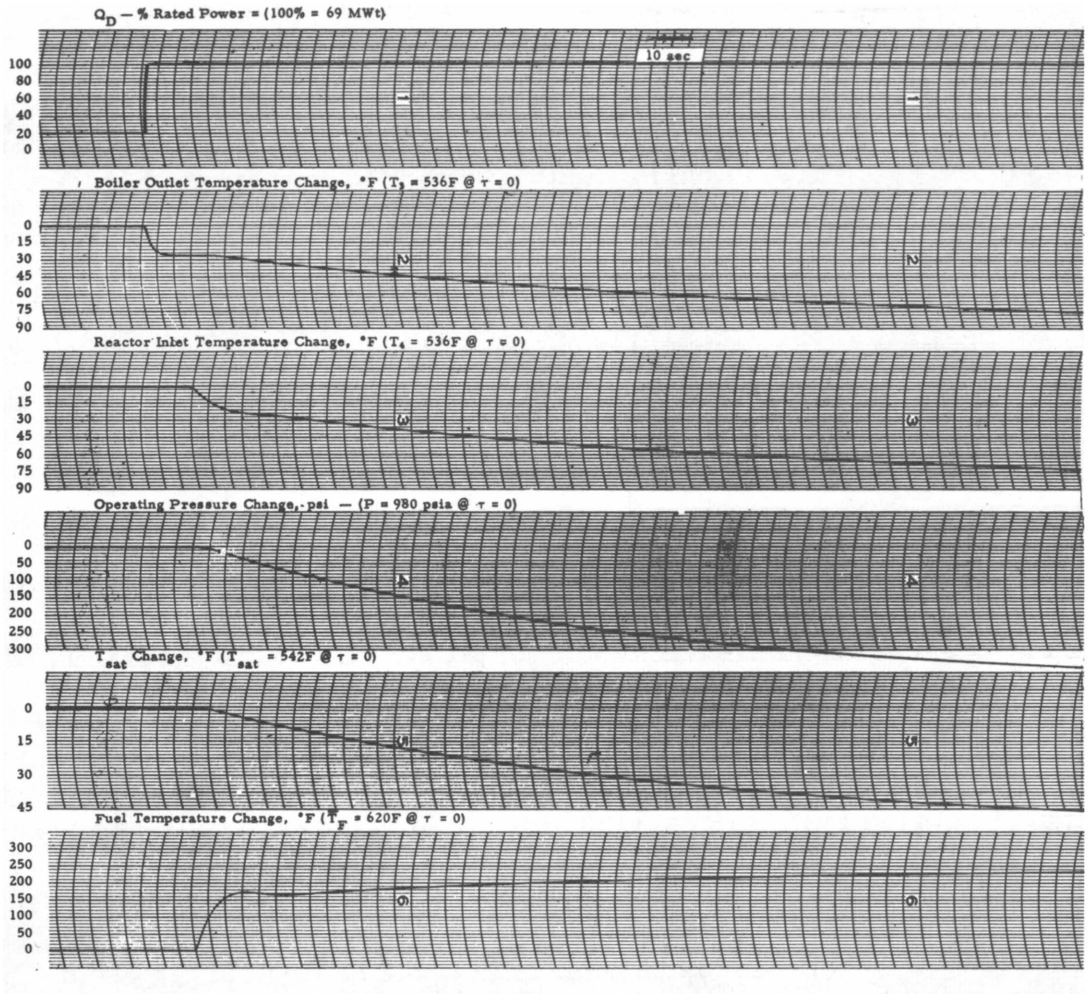


Figure 8.7-23. Effect of Void Reactivity on Operating Pressure — Power Increase

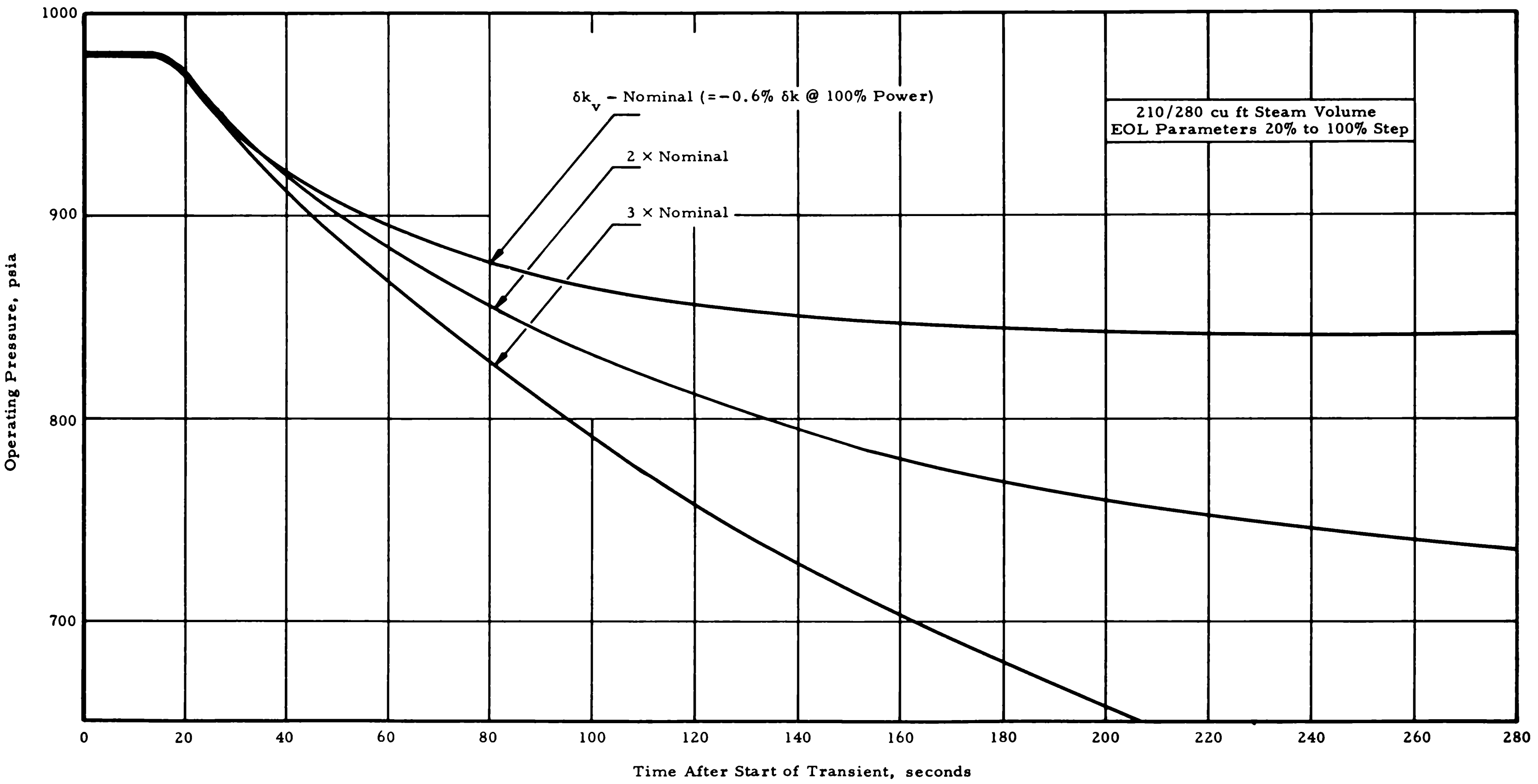


Figure 8.7-24 A. 20% to 100% Power Step — δk_m and δk_v = 1/3 Nominal, V_{steam} = 210/280 cu ft

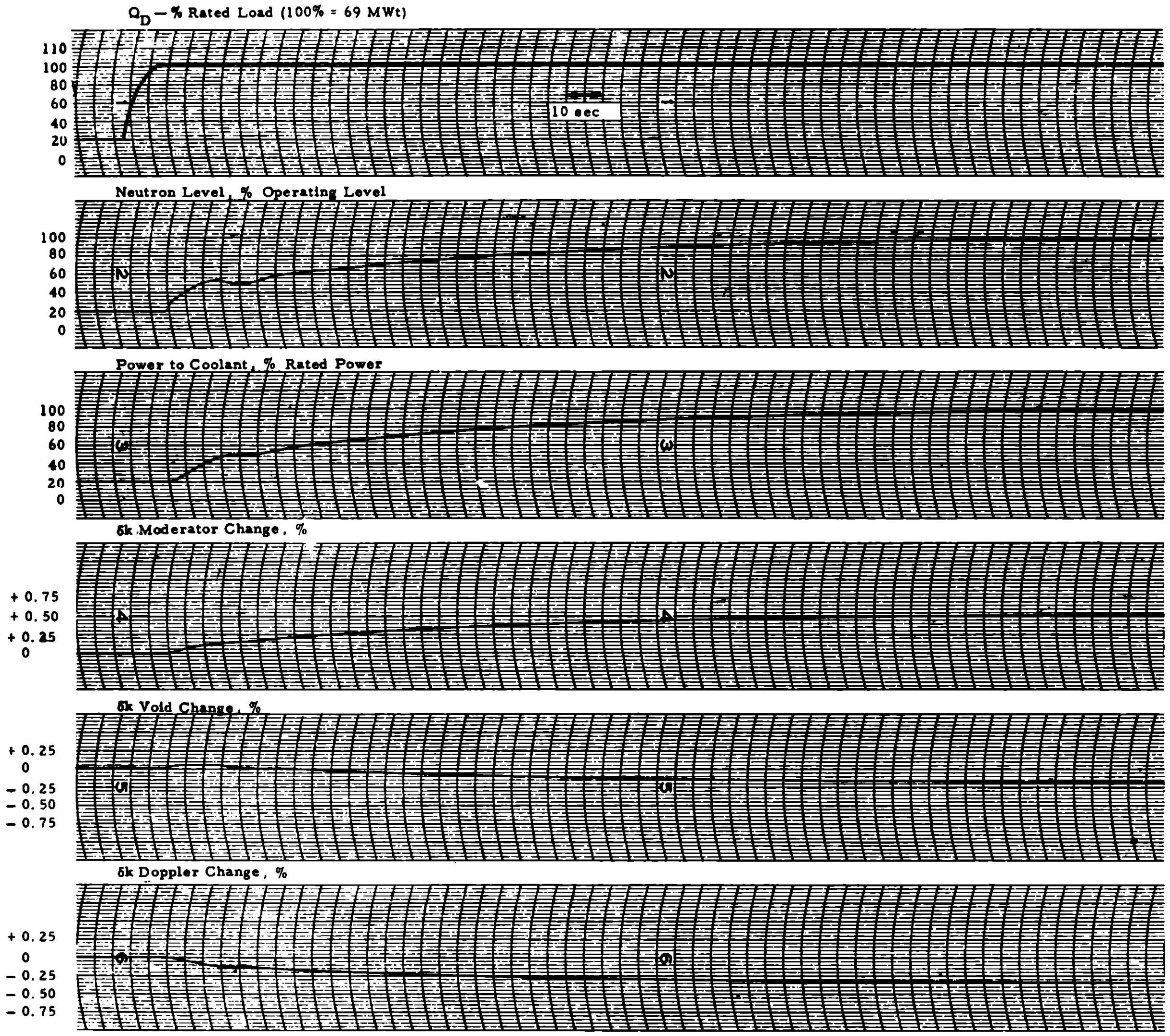


Figure 8.7-24 B. 20% to 100% Power Step — δk_m and δk_v = 1/3 Nominal, V_{steam} = 210/280 cu ft

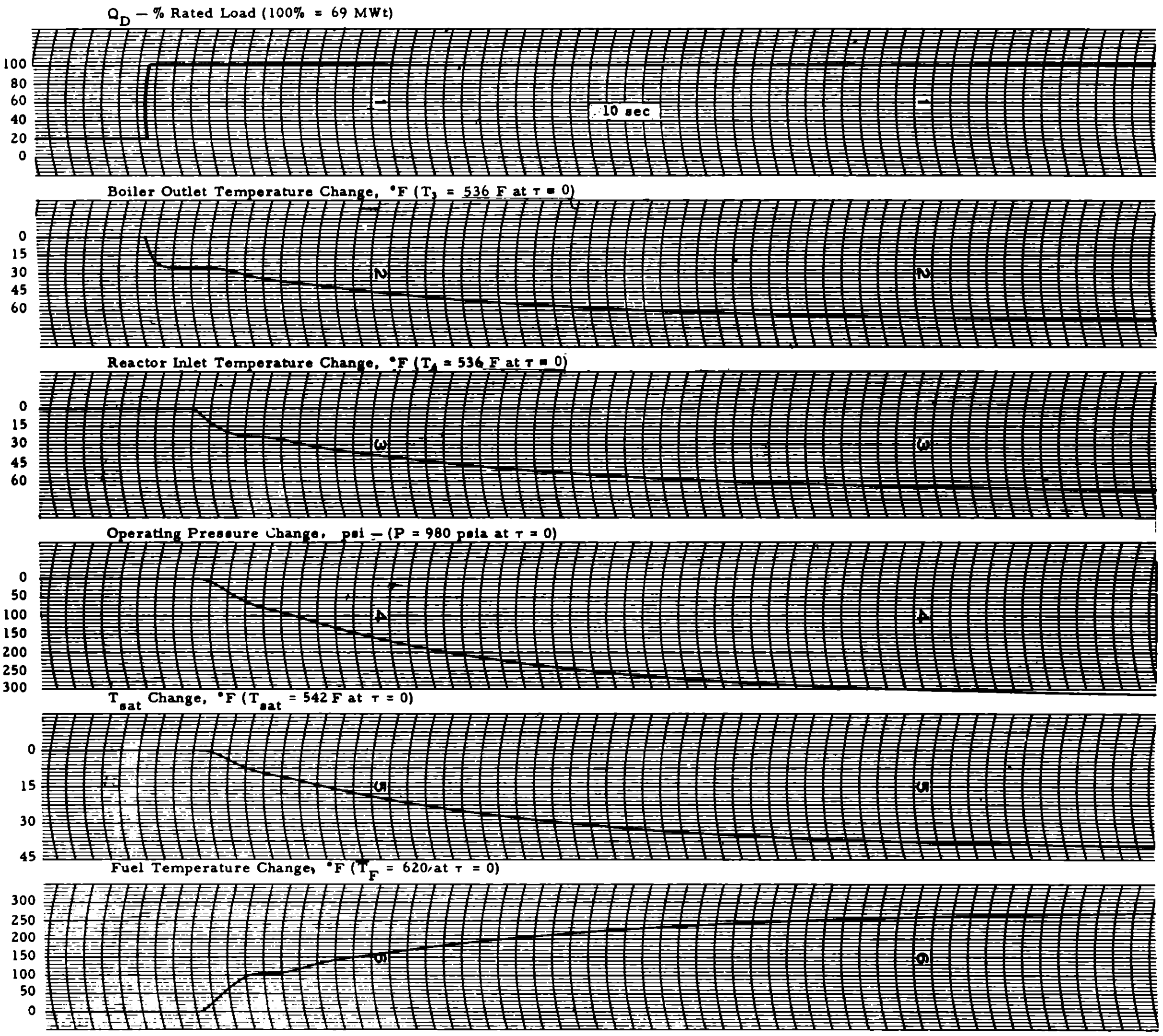


Figure 8.7-25 A. 20% to 100% Power Step with 10-second Time Constant on Load Demand — EOL Parameters, V_{steam} = 210/280 cu ft

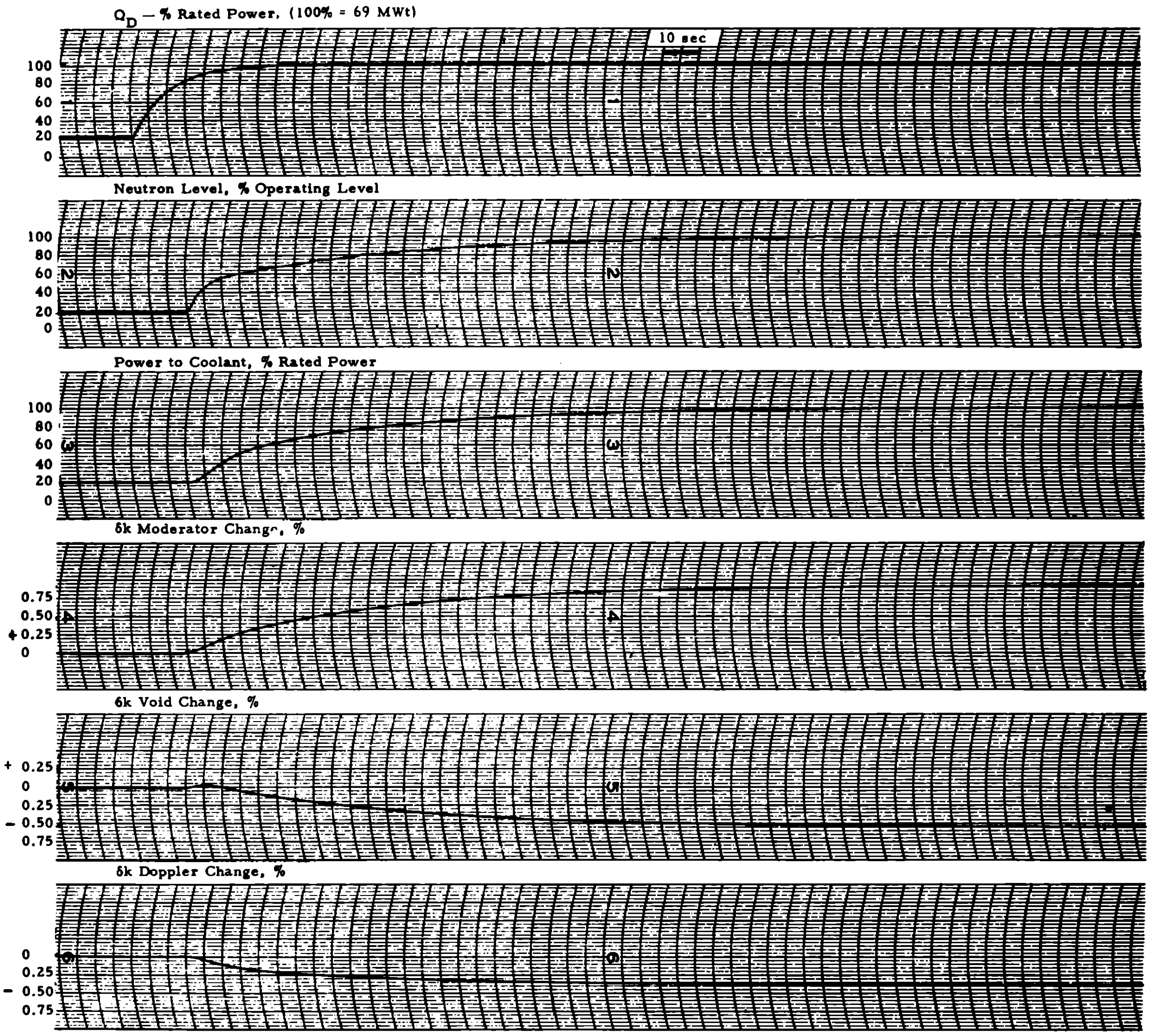


Figure 8.7-25 B. 20% to 100% Power Step with 10-second Time Constant on Load Demand — EOL Parameters, V_{steam} = 210/280 cu ft

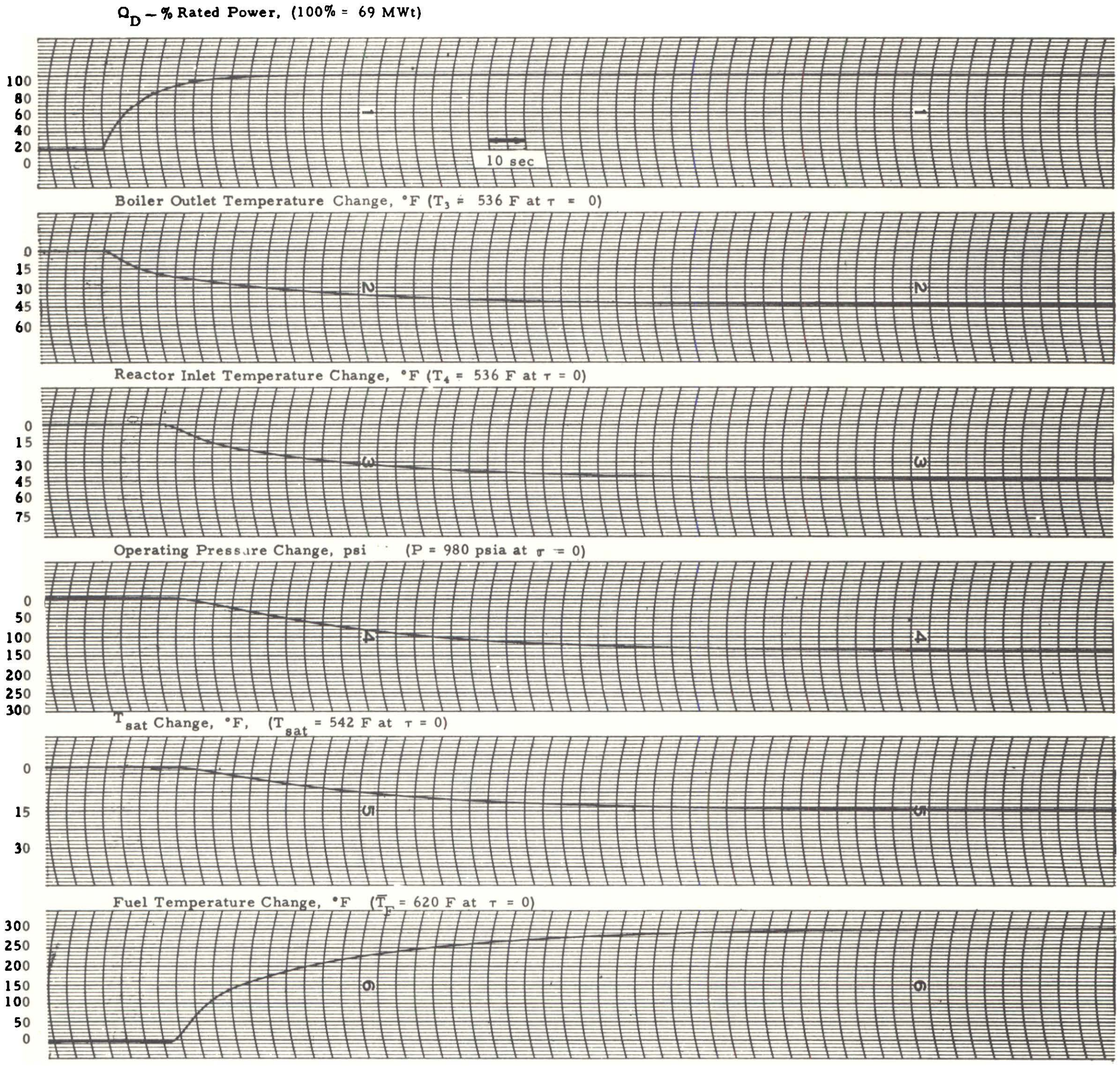


Figure 8.7-26. Effect of Load Demand Time Constant on Operating Pressure — Power Decrease

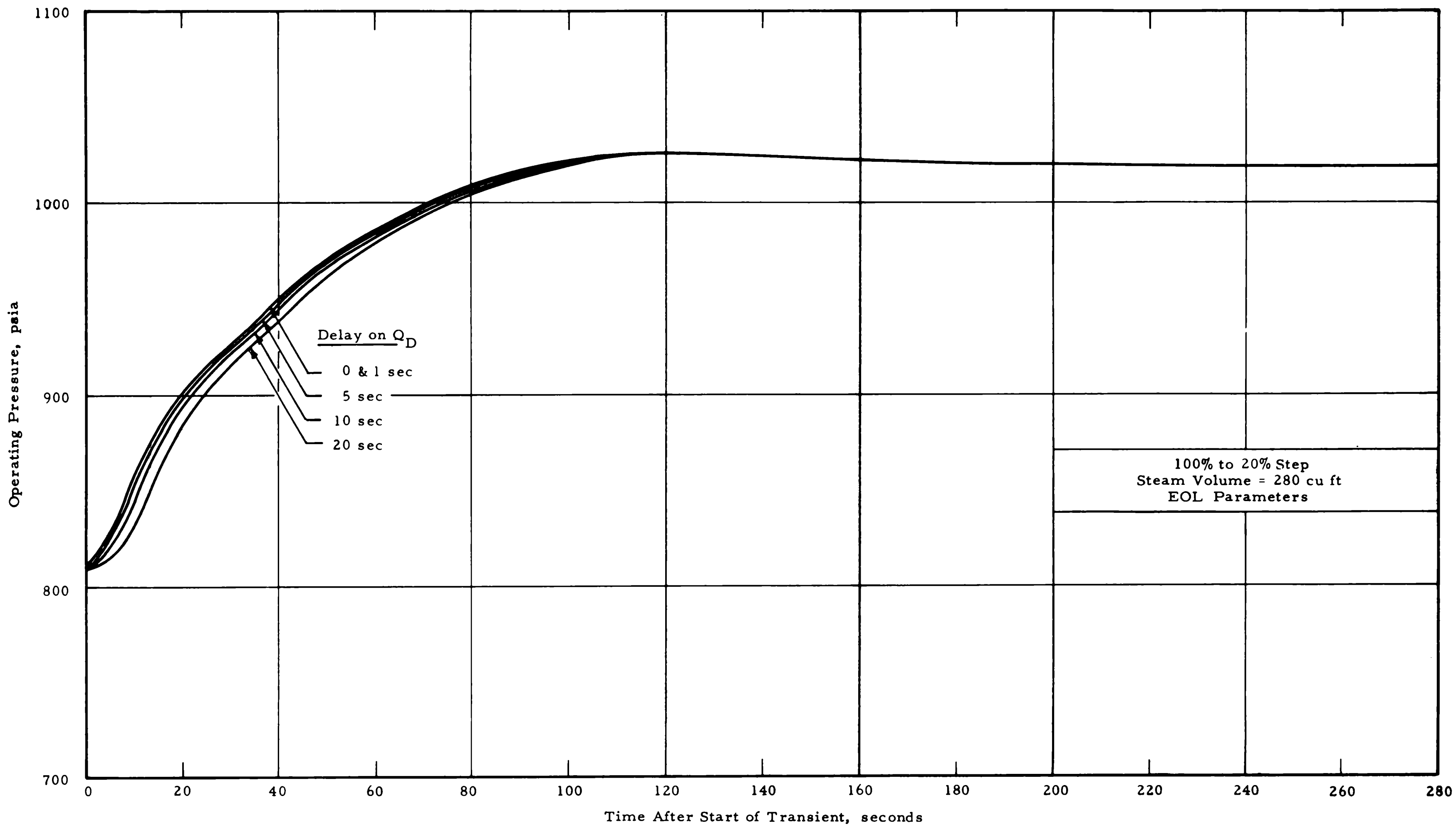


Figure 8.7-27. Effect of Load Demand Time Constant on Operating Pressure — Power Increase

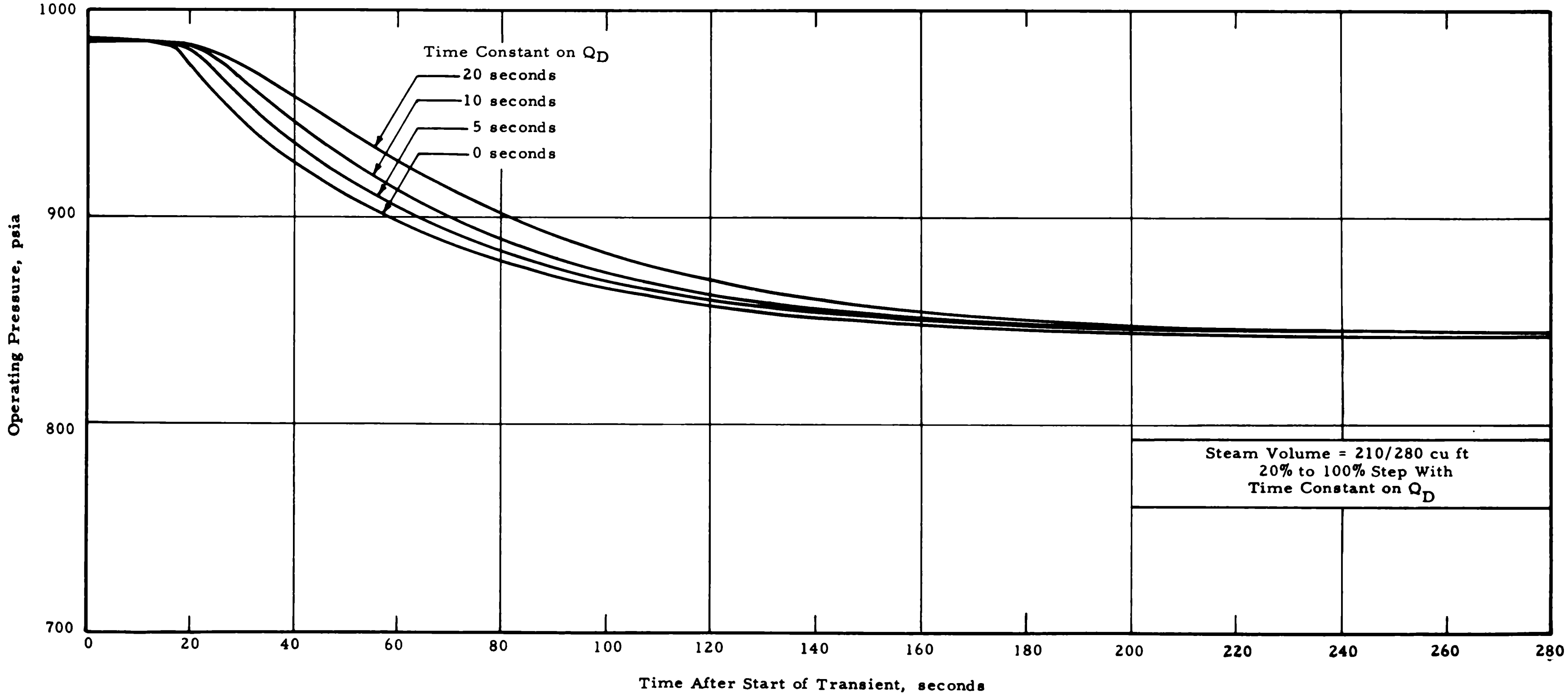


Figure 8.7-28 A. Ship's Motion Studies for 6-, 7-, 8-, and 10-second Periods — EOL Parameters, Power Increase Model

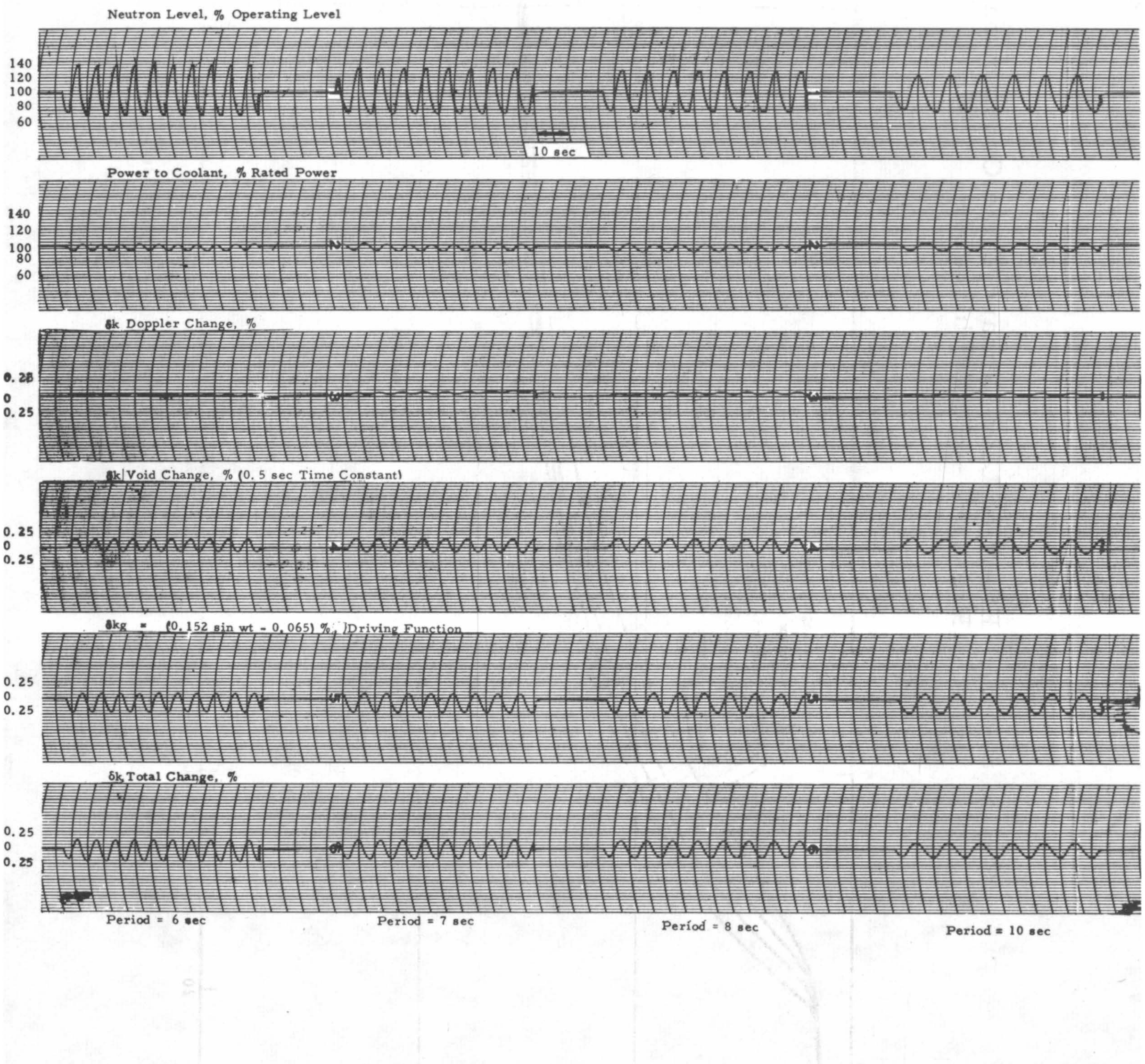
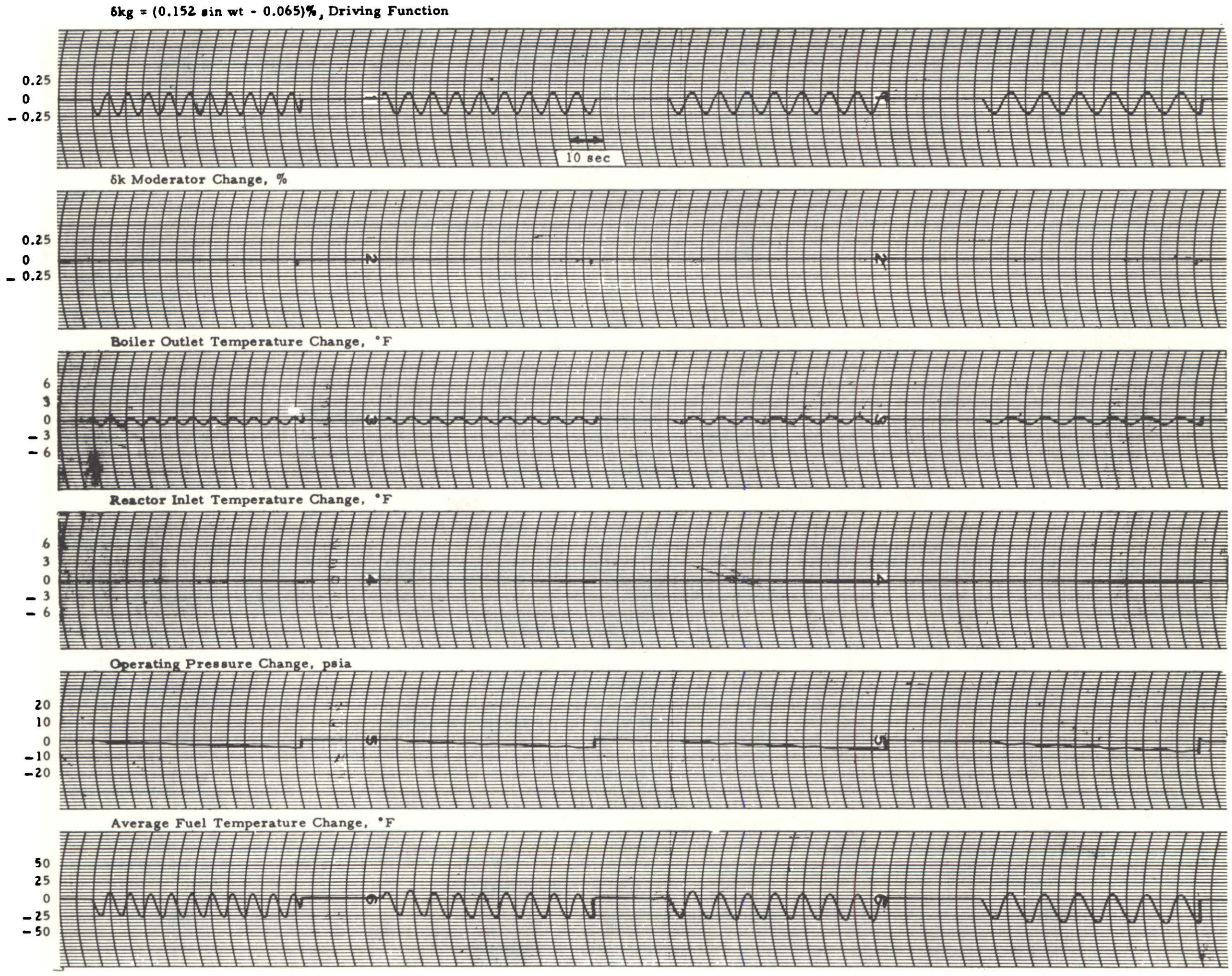


Figure 8.7-28 B. Ship's Motion Studies for 6-, 7-, 8-, and 10-second Periods — EOL Parameters, Power Increase Model



APPENDIX 1

PART ONE

Preliminary Plant Design Information Sheets for a Consolidated Nuclear Steam Generating Marine Power Plant

1. Heat Balance

1. Primary System Heat Generation

Core Output, Btu/hr	2.13×10^8 (62.4 MW)

2. Primary System Heat Losses (Btu/hr)

Decay Heat Mechanism	
Through vertical walls	1.41×10^6
Through top reactor vessel head	0.25×10^6
Control rods (estimate)	0.25×10^6
Through remainder of vessel (approximate)	0.1×10^6
Poison Bleed	Negligible
Total	2.01×10^6

3. Heat Transferred at Steam Generators

Btu/hr	2.11×10^8 (61.7 MW)

4. Secondary Heat Losses

Feed water and steam lines within containment, Btu/hr	0.15×10^6
Steam System, Btu/hr	1.4×10^6
Total	1.55×10^6

2. Primary System Design Information

1. Design Conditions

Design Pressure, psig	1100
Design Temperature, F	560
Design Flow Through Core, lb/hr	6×10^6

2. Operating Conditions

Pressure (steam dome), psia	812.4
Temperatures, F	
Core Outlet	520
Core Inlet	490.4
Average	505

3. Volumes, ft^3

Reactor Vessel Internal (empty)	1300
Reactor Vessel External (with lead)	2310
Primary Water (during operation)	695
Steam Dome (during operation)	187
Reactor Vessel Internals	373
Containment Vessel (empty)	9630
Containment Water (during operation)	6309
Containment Shielding	282
Reactor Vessel Shielding	493

4. Weights, lb

Reactor Vessel (empty) including support skirt	170,000
Primary Water (during operation)	33,900
Steam Dome (during operation)	290
Reactor Vessel Internals (includes 42,500 lb for tube bundle)	115,650
Containment Vessel (empty)	150,000
Containment Water (during operation at 110 F)	390,000
Containment Shielding (outside)	202,500
Reactor Vessel Shielding	327,530
Primary Pumps	15,000
Primary Pump Shielding	21,000
Condensing Tanks	7,500
Containment Heat Exchangers	32,000
Water in Condensing Tanks	4,000
Containment Circulating Pumps	2,000
Chemical Control Injection Pumps	2,400
Piping Valves, etc.	12,000
Instrumentation and Controls	22,400
Total	1,508,170

5. Transit Times, seconds

Reactor Outlet to Steam Generator Inlet	5.7
Steam Generator	2.9
Steam Generator Outlet to Pump Section	6.75
Pump Discharge to Reactor Inlet	3
Reactor	0.5
Total	18.85

3. Secondary System Design Information

1. Design Conditions (between inlet check)

Pressure, psig	1,220
Temperature, F	560
Total Steam Flow, lb/hr	224,605

2. Operating Conditions, psia

Pressure - Full Load at Super Heater Outlet	412
No Load to 15% Load (estimate)	420
Steam Flow from Steam Generator, lb/hr	224,605

4. Auxiliary Systems

1. Soluble Poison System

Name and formula	Boric Acid-H_3BO_3
Concentration in primary water	
Time zero - cold	9370 ppm
hot	7190 ppm
Core half burned - cold	5800 ppm
hot	2850 ppm
End of life - cold	----
hot	----
Concentration in Containment Water, minimum	9370 ppm
Poison Injection Rate maximum emergency	140 gph (15 to 20% H_3BO_3 by weight)
Feed and bleed rate (normal operation)	0.3-3 gph
Total Mass removed from Primary System over lifetime of core (no shut-down or startups)	87,600 lb
Heat rejection due to bleed	Negligible

2. Containment Fill and Drain

Fill Rate, gpm	150
Drain Rate, gpm	150
Water Level (maximum), ft	35
Normal average water temperature, F	110

3. Primary Fill and Drain

Fill Rate (cold)	50 gpm
Drain Rate (cold)	50 gpm
Make-up Rate (operating T&P)	140 gph

4. Emergency Decay Heat Removal

Amount of heat to be removed, Btu/hr	2.01×10^6

Temperature gradients across, F

Inside film	32
Scale	16
Stainless clad	8
Vessel	64
Thermal resistance (lead)	213
Scale	16
Outside film	45

5. Containment Water Heat Removal

Heat load during operation, Btu/hr

Decay heat mechanism	2.01×10^6
Pumps	0.056×10^6
Poison Bleed	Negligible
Feed water and steam lines within containment	0.15×10^6
Total	2.22×10^6

Average containment water temperature, F (normal operation)	110
Average containment water temperature after 37 hours with no heat removed, F	255
Containment water flow through HX, lb/hr	4×10^5
Sea water flow through HX, lb/hr	1.6×10^5
Heat transfer area, ft^2	4270

6. Sampling (includes samples from primary, containment, and feedwater)

Number of sampling points, liquid	3
Number of sampling points, gas	1

7. Primary Relief

Fluid	steam
Relieves to	condensing tank
Number of valves	2
Type of valves	Self-Actuating ASME Section VIII Code
Pressure Settings, psi	1100 and 1150
Temperature, F	556
Flow Total, lb/hr	23,000

8. Secondary Relief

Steam Generator	water
Relieves to	containment water
Number of valves	3

Type of valves	Self-Actuating ASME Code Section VIII Water Relief
Pressure Settings, psig	1220
Temperature	560
Flow, gpm	10
Secondary Relief	
Fluid	Steam
Relieves to	Atmosphere
Number of valves	2
Type of valve	Self-Actuating ASME Code Section VIII
Pressure Settings, psig	495 and 515
Temperature, F	470
Flow, lb/hr	336,000

9. Containment Relief

Fluid	Steam
Relieves to	Atmosphere
Number of valves	1
Type of valve	Self-Actuating ASME Code Section VIII
Pressure Settings, psig	50
Temperature, F	300
Flow, lb/hr	1,280

10. Condensate Tanks

Fluid	Steam and Water
Relieves to	Containment
Number of valves	1
Type of valve	Self-Actuating ASME Code Section VIII
Pressure Setting, psig	400
Temperature, F	450
Flow	
Steam, lb/hr	23,000
Water, gpm	67

5. Components

1. Steam Generators

Type	Once Through
Number	(1)
Thermal Duty	2.13×10^6 Btu/hr

Primary flow rate	6×10^6 lb/hr
Steam flow rate	224,000
Steam Conditions - 100% Power Level	
Pressure, psia	412
Temperature, F	512
Superheat, F	70°
Feedwater Temperature, F	348
Number of tubes	300
Tube Size, in.	0.75
Average Effective Tube Length, ft	131.04
Heat Transfer Surface, ft^2	7706
Shell Side Pressure Drop at Full Flow, psi	5
Tube Side Pressure Drop at Full Flow, psi (approximately)	100

2. Primary Pump

Type	Canned Motor, axial flow
Number	4
Capacity (each at 490 F gpm)	3900
Total Developed Head, psi	8

6. Water Conditions

1. Initial Primary Fill

Fluid	Water
Total Solids, ppm maximum (other than boric acid)	0.05
pH	5
O_2, ppm (Hydrazine Controlled)	0.01
Chlorides, ppm - maximum	0.01
Boric Acid, ppm	9370

2. Feedwater Recommended

Total Dissolved Solids, ppm	0.05
Suspended Solids	Minimum (preferably zero)
Hardness, ppm	zero
Organic, ppm	zero
Free Caustic, ppm	zero

Dissolved Oxygen, ppm	0.007 Maximum (preferably zero)
CO_2, ppm	Minimum (preferably zero)
Chlorides, ppm - maximum	0.01
Total silica (SiO_2) ppm - maximum	0.02
Total iron (Fe), ppm - maximum	0.01
Total copper (Cu), ppm - maximum	0.002

The pH valve should be adjusted to obtain a maximum 0.01 ppm of iron. This will normally require a pH valve within the range of 8.5 to 9.2 at 77 F.

3. Containment Water

Potassium Hydroxide	pH = 8.5 to 9.5
Boric Acid, ppm	9370

7. Containment Design

1. Design Basis

The containment is designed for a complete primary fluid loss. It will also adsorb decay heat for 37 hours without any heat removal following a shutdown.

2. Energy Release, approximate

Primary Fluid	33,900 lb - 16.7×10^6 Btu
Secondary Fluid	Negligible

3. Design Pressure

The storage of decay heat for 37 hrs, or for the release of all the primary water yields the maximum containment pressure, 50 psig, in the gas space.

8. Construction Materials for Primary, Secondary, and Chemical Control Systems

1. Material Contacting Primary Fluid

Plate or Strip	Stainless Steel, ASTM-A-167, Grade 3
Forgings	Stainless Steel, ASTM-A-182, F 304
Castings	Stainless Steel, ASTM-A-351, CF 8 or 8 M
Tubing	Stainless Steel, ASTM-A-213, Type 304

Cladding	Stainless Steel, ASTM-A-240, Type 304 and Stainless Steel, ASTM-A-298
Base Material (where clad)	
Plate	Carbon Steel, ASTM-A-302, Grade B
Forgings	Carbon Steel, ASTM-A-336,
Seating and Guiding Surfaces	Stellite 6 or 3
Piping	Stainless Steel, ASTM-A-376, Type 304 or 316

2. Material Contacting Secondary Fluid

Plate	Carbon Steel, ASTM-A-212, Grade B
Forgings	Carbon Steel, ASTM-A-105, Grade II

3. Packing and Gaskets

The packing and gaskets for the primary relief valves, gate valves, globe valves, and pumps are based on the vendor's recommendations.

9. Integrity Testing and Inspection of B&W Supplied Equipment for Primary, Secondary, and Chemical Control Systems

The applicable codes, standards, and regulations are listed in Section 2 of the General Technical Specifications, PD-1-191-(59-3034).

APPENDIX 1

PART TWO

Preliminary Issue System Design
Specifications for the Primary System

CONTENTS

1. Scope

Presented here is a description of the CNSG Primary System. (See Fig. 3.2-1.) This specification is for design information and is not intended for system fabrication.

2. References

1. Figure 3.2-1
2. Preliminary Specification for Cost Estimating of CNSG Reactor Pressure Vessel, No. M-3-129-0(59-3034-31)
3. General Technical Specification, B&W Specification No. PD-1-191-0(59-3034)

3. Function and General Description

1. Function

This system will remove heat from the core

1. during normal and maximum power operation of the ship,
2. during a scheduled reactor shutdown, and
3. during emergency operating and shutdown conditions.

2. General Description

The primary system will consist of a reactor vessel core, a once-through steam generator, four circulating pumps, and internal baffles. (See Fig. 3.2-1.)

The four pumps will take suction from the bottom of the shell side of the steam generators, and will force the water through the core and up a central riser. At the top of the riser, the water will spill over and enter the steam generator top.

When the pumps are not used, the primary system will operate by natural circulation up to 20% of the rated power.

2.1. Core

The core will be capable of transferring 62.4 MW of heat to the primary water as it flows up through the core at 6,000,000 lb/hr with an inlet temperature of 490 F.

2.2. Steam Space

A steam space at the top of the vessel will operate at a saturation pressure corresponding to the water outlet temperature of the core. This space will absorb fluctuations in pressure due to changes in the primary water temperature.

2.3. Once-Through Steam Generator

A once-through steam generator will be located circumferentially in the down comer section of the vessel between the chimney and the vessel wall.

Primary water will flow on the shell side of the steam generator and feedwater will flow on the tube side. The steam-generator heat transfer area will be divided into three equal sections. Each may be isolated if a tube fails. Together, the three sections of the steam generators will be able to produce 224,000 lb of steam at 515 F and 412 psia.

2.4 Main Circulating Pumps

Four axial-flow canned-motor pumps will be arranged around the outside of the reactor vessel 90 degrees apart. They will be mounted on extensions of the reactor vessel where they will be accessible for maintenance.

Each pump will carry 25% of the flow during normal ship operation. Flow to the pump will be through an annulus between the pressure containing extention of the reactor vessel and the central discharge duct.

2.5. Reactor Vessel

See Reference 2.

4. Parameters and Conditions

1. General

Design Pressure, psig	1100
Design Temperature, F	560
Core Heat Generation, Btu/hr	2.13×10^8
Heat Losses, Btu/hr	2.01×10^6
Flow, lb/hr	6×10^6
Core Inlet Temperature, F	490.4

Core Outlet Temperature, F	520
Steam Dome	
Operating Temperature, F	520
Operating Pressure, psia	812.4
Primary Water inventory, lb	33,900

2. Steam Generator

Design Pressure, psig	1,220
Design Temperature, F	560
Heat transferred, Btu/hr	2.11×10^8
Steam Conditions	
Pressure, psia	412
Temperature, F	515
Flow, lb/hr	224,605
Feedwater Conditions	
Temperature, F	348
Flow, lb/hr	224,605
Primary Water Conditions	
Flow, lb/hr	6×10^6
Inlet Temperature, F	520
Outlet Temperature, F	490.4
Dimensional Restrictions (Reference 1)	

3. Main Circulating Pumps

Capacity, gpm	3,900
Total Developed Head, psi	8
Design Temperature, psig	1,100
Operating Pressure, psig	800
Design Temperature, F	560
Operating Temperature, F	490
Fluid	Water + 9370 ppm H_3BO_3
Minimum Available NPSH, psig	40
Motor	canned
Mounting	See Reference 1
Motor cooling	Natural Convection of Containment Water

4. Check Valves for Main Circulating Pumps

Function - To prevent reverse rotation of pumps when the motor is not energized.

Requirements - Must be accessible for maintenance located on the discharge side of the pump (may be integral with pump), capable of minimum pressure drop when water is circulating through a dead pump at approximately 20% of full flow.

5. Construction Material

Plate or strip	Stainless Steel, ASTM-A-167, Grade 3
Forgings	Stainless Steel, ASTM-A-182, F 304
Castings	Stainless Steel, ASTM-A-351, Cf 8 or 8M
Tubing	Stainless Steel, ASTM-A-213, Type 304
Cladding	Stainless Steel, ASTM-A-240, Type 304 and Stainless Steel, ASTM-A-298
Base Material (where clad)	
Plate	Carbon Steel, ASTM-A-302, Grade B
Forging	Carbon Steel, ASTM-A-336
Seating and Guiding Surfaces	Stellite 6 or 3
Piping	Stainless Steel, ASTM-A-376, Type 304 or 316.

6. Relationship to Other Systems

1. Secondary System

Feedwater is pumped through the steam generators.

2. Chemical Control System

Poison concentration control and sampling.

3. Containment System

Primary water gives up approximately 2×10^6 Btu/hr continuously to the containment water. Containment water cools the primary pump and the rod drive motors.

4. Relief System

Protects the reactor vessel from excessive pressure.

7. Instrumentation Requirements

The proposed instrumentation and control will provide the necessary means for detecting, measuring, and indicating the variables of pressure, temperature, and level. Detectors, transmitters, indicators, switches, and relays comprise the instrumentation.

The detection, measurement, etc., of the variables mentioned in the preceding paragraph shall read at the following location.

1. Level

   The water level is measured and indicated continuously on the console, and a high and low level alarm is provided.

2. Pressure

   The pressure is measured and indicated on the console. At a pre-set high level, the reactor is scrammed.

3. Temperature

   The steam generator inlet and outlet temperature is measured and indicated on the console. The reactor is scrammed at a pre-set high temperature.

8. Insulation Requirements

The major portion of the reactor vessel will be covered with lead with a thickness designed to control the flow of heat from the primary water to the containment water (approximately 2×10^6 Btu/hr). See Reference 1 for portions of the vessel that will be canned.

APPENDIX 1
PART THREE

Preliminary Issue System Design Specification for Chemical Control System

CONTENTS

1. Scope

Described here is the chemical control system to be installed in a CNSG Plant. This specification is for design purposes and is not intended for system fabrication.

2. References

1. General Technical Specification, B & W Specification No. PD-1-191-0(59-3034)
2. Figure 3.1-1
3. National Research Council Publication 658, Washington, D. C., 1959.

3. Function and General Description

1. Function

1. To provide water and soluble poison injection into the primary system for reactor control requirements.
2. To inject the water makeup and soluble poison into the containment.
3. To control plant liquid effluent discharge.
4. To sample the water contained in the primary system, the secondary system, and in the containment.
5. To control the secondary system water quality and the purification of plant makeup.

2. General Description

2.1. Primary System Chemical Control and Sampling

During normal reactor operation, simultaneous fluid withdrawal and pure water injection from a purified water supply tank to the reactor at a low rate will be used to reduce soluble poison concentration in the primary system fluid. Two lines extending inside the reactor vessel from the head to the core region will provide for fluid injection and withdrawal. Fluid withdrawn from the reactor through one of these lines will pass through a flow measuring device, and will be discharged through a control valve and a radiation monitor to the ship's propeller wash. Branch line connections to this withdrawal line will provide 1) recirculation of flow through a sampling loop to an injection pump and the injection line for sampling, and 2) flow diversification to three condensing tanks if fluid holdup is required during port operation. Fluid injection will be accomplished by using one of two positive displacement pumps. The pump discharge flow is measured and matched with the withdrawal flow for simultaneous injection and withdrawal. A soluble poison supply tank will connect to the intake of the injection pumps and provide an emergency shutdown injection of poison.

Three condensing tanks, interconnected by common drain, vent, and reactor vessel effluent lines, will provide intermediate holdup for fluid or steam discharged from the reactor vessel. Two relief valves and a steam bleed valve mounted on a connection to the reactor vessel steam dome will discharge through piping to the lower part of these condensing tanks. Steam discharged to the tanks will be condensed by heat transfer to the containment water through the tank walls, and will mix with 4000 lb of water maintained in the tanks.

When design pressure is reached in the tanks, they will relieve through a common relief line to the containment. A vent line containing a control valve will connect to the common relief line to provide gas venting to the containment or to off-gas disposal facilities. A common drain line from the tanks will have one piping connection containing a control valve to provide controlled discharge to the overboard discharge line. Also, it will contain one piping connection with a control valve to provide for fluid discharge to the containment if overboard discharge is not desired.

The relief, vent, and drain mechanisms are included for operating flexibility, and normally, fluid will not be discharged to the containment by them. If primary fluid is discharged to the containment, local monitoring around the containment will determine the degree of access allowable to the outside of the containment.

2.2. Containment System Chemical Control and Sampling

A centrifugal pump, by taking suction from a soluble poison mixing tank, can add soluble poison to the containment water. This pump will also provide containment water samples by taking suction from the bottom of the containment and by recirculating the flow back to the containment. A portion of the recirculating flow established in this manner will be bypassed through a sampling loop in which samples can be taken. Appropriate valves in the lines will provide containment isolation and flow control.

Water makeup to the containment, the soluble poison mixing tank, and the purified water supply tank will be taken from the secondary system boiler-feed water downstream from the main condenser condensate pumps.

2.3. Secondary System Chemical Control and Sampling

Secondary makeup water obtained from the ship's evaporators will be stored in the fresh water supply tank. A level controlling device in the condenser hot well will indicate the amount of water to be injected into the secondary system. Two centrifugal pumps will take suction from the fresh water supply tank to force the water through one of two demineralizers before it enters the secondary system. The secondary system sampling will be provided by a sample bypass line containing a sample bomb to recirculate flow from the filter discharge to the boiler feed pump inlet.

4. Design Requirements

1. Injection Pumps (2) Positive Displacement - Adjustable Flow

Maximum Flow Rate	140 gph
Normal Range	0.3 to 3 gph
Suction Pressure	Atmospheric
Maximum Discharge Pressure	1250 psig

Temperature	150 F
Horsepower	5
Location	Outside Containment

2. <u>Condensing Tanks (3) - Vertical Installation</u>

Design Pressure	400 psig
Design Temperature	467 F
Internal Volume (each)	91 ft^3
Maximum Over-all Height	30 ft
Maximum OD	27 in.
Connections (each)	1.5-in. Schedule 40 Nozzle Top Head and Bottom Head
	1.5-in. Schedule 40 Nozzle Bottom Head with Internal Spray Ring for Steam Dispersal
	Two 1-in. Schedule 80 Level Detection Connections
Location	Inside Containment

3. <u>Soluble Poison Supply Tank (1)</u>

Design Pressure	40 psig
Design Temperature	200 F
Internal Volume	200 gal
Connections	1-in. Schedule 80 Fill Nozzle
	2-in. Agitator Flange
	1.5-in. Schedule 40 Discharge Nozzle
	Two 1-in. Schedule 80 Nozzles for Level Gauge
Location	Outside Containment

4. <u>Purified Water Supply Tank</u>

Design Pressure	40 psig
Design Temperature	120 F
Internal Volume	500 gal
Connections	1-in. Schedule 80 Fill
	1 1/2-in. Schedule 40 Discharge
Location	Outside Containment

5. Containment Poison Mixing Tank

Design Pressure	40 psig
Design Temperature	200 F
Internal Volume	50 gal
Connections	0.75-in. Schedule 80 Fill
	0.75-in. Schedule 80 Discharge
	2-in. Flange for Agitator
	6-in. Poison Addition Hand Hole
	Two 0.75-in. Schedule 80 Nozzles for Level Gauge
Location	Outside Containment

6. Containment Recirculation Pump (1)-Centrifugal Type

Capacity	10 gpm
Suction Head	10 ft
Developed Head	50 ft
Horsepower	1
Location	Outside Containment

7. Sampling Equipment (3 Identical Sample Loops)

Design Temperature	400 F
Design Pressure	1250 psig
Sample Bomb Internal Volume	2250 cc
Piping	0.25-in.

8. Makeup Feed Pumps (2) - Centrifugal Type

Capacity	25 gpm
Suction Head	10 ft
Developed Head	75 ft
Horsepower	2
Location	Outside Containment

9. Demineralizers (2)

Type of Resin	Nuclear Grade - Mixed Bed
Resin Volume	11.75 ft^3
Vessel ID	23.5 in.
Vessel Height	8 ft

Resin Bed Pressure Drop, psi	
Normal	2.5
Maximum	5
Makeup Flow, gpm	
Maximum	25
Normal	5
Effluent Maximum Impurity	0.05-ppm Solubles and Insolubles,
	0.01-ppm Chlorides-maximum
Renewal Method	Regeneration or Replacement

10. Piping, Valves and Fittings

Design and installation will be in accordance with the codes listed in Reference 1.

Permanently installed piping, valves, and fittings will be of a welded construction to handle primary coolant. Valves inaccessible during plant operation will be remotely operated.

Design temperature and pressure of piping, valves, and fittings will correspond to the source of the fluid handled (primary system bleed piping, valves, and fittings will be designed for 1100 psig at 560 F).

5. Material Requirements

Materials for the chemical control equipment associated with primary system fluid injection, withdrawal, or sampling will be stainless steel. Chemical control equipment associated with containment system and secondary system fluid handling will be carbon steel except the sampling loop components which will be of stainless steel identical to that of the primary system sampling loop.

6. Flow Diagram

See Figure 3.1-1

7. Relationship to Other Systems

1. Primary System

This system provides sampling, soluble poison concentration control, water makeup, and effluent handling for the primary system.

2. Containment System

This system provides sampling, soluble poison concentration control, and water makeup for the containment system.

The containment system will cool the condensate tanks and primary system sampling flow, store gases from vent and relief valves, and hold up liquid effluent from the condensing tanks in an emergency.

3. Secondary System

This system provides sampling and demineralized makeup for the steam generator feedwater.

The secondary system provides feed from the evaporated water supply to the demineralizers for makeup. Also, it provides regeneration flow through the demineralizers for shipboard regeneration.

8. Instrumentation

The proposed instrumentation and control will provide the necessary means for detecting, measuring, and indicating the variables of temperature, pressure, level, flow, and activity level. Detectors, transmitters, indicator switches, and relays comprise the instrumentation.

1. Level

The fluid level in the condensing tanks will be measured and indicated adjacent to the controls. Normally, about a 4000-lb water storage will be maintained within the three interconnected condensing tanks. The fluid level in the tanks located outside the containment is shown by local level gages, and is used to maintain or make up the required solution supply.

2. Pressure

Pressure within the condensing tanks will be measured and indicated adjacent to the controls.

3. Flow

Fluid flow injected and withdrawn from the primary system will be measured and indicated adjacent to the controls. During feed and bleed operation, these flows will be matched to give the desired change of poison concentration within the primary system.

4. Temperature

The temperature of the primary system sampling flow will be measured, and it will be controlled locally so it will not exceed the inlet temperature requirements of the injection pumps.

5. Activity Level

A monitoring device will determine control activities in any primary system effluent discharged to the ship's propeller wash. The activity in this flow and the accumulative quantity of discharge will be logged and controlled so it will not exceed requirements for disposal. (Reference 3.)

9. Electrical Requirements

Name of equipment	No.	No. normally operating	Horsepower
Injection Pumps	2	1	5
Primary Poison Supply Tank Agitator	1	1	1/4
Circulating Pumps	1	1	1
Containment Poison Mixing Tank Agitator	1	1	1/4
Makeup Pumps	2	1	2

10. Requirements from Other Services

Temporary connections for fill, drain, and venting during dockside maintenance and startup of the reactor plant are required.

11. Shielding Requirements

The disposal line to the ship's propeller wash may require shielding. The primary system sampling line and the effluent condensing tanks will require shielding. The quantities will be specified in the final design.

12. Insulation Requirements

None

APPENDIX 1

PART FOUR

Preliminary Issue General
Technical Specification

CONTENTS

1. Scope

These specifications enumerate the general requirements for the design of systems to be used in a shipboard CNSG. Specific requirements are given in the system specifications.

2. References

The latest editions and addenda of the following codes, specifications, regulations, and standards, when referenced in this or the system specifications, are to be considered as part of the system specifications.

1. Marine Engineering Regulations and Materials Specification for the United States Coast Guard, CG-115.
2. American Bureau of Shipping, Rules for the Classification and Construction of Steel Vessels.
3. Electrical Engineering Regulations, United States Coast Guard, CG-259.
4. ASME Boiler and Pressure Vessel Codes, Section II, Materials Specification.
5. ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels.
6. ASME Nuclear Case Interpretations.
7. Standards of the American Society for Testing Materials, ASTM.
8. American Standard Association, ASA.
9. Manufacturers' Standardization Society, MSS.
10. Standards of the American Institute of Electrical Engineers.
11. Standards of the National Electrical Manufacturers' Association.
12. Hydraulic Institute Standards.
13. Procedure for Liquid Penetrant Inspection, B&W Specification No. AMI-1.
14. Procedure for Helium Leak Testing, B&W Specification No. AMI-4-0.
15. Cleaning Procedure for Stainless Steel Equipment in Critical Application, B&W Specification No. AMP-1-0.
16. B&W Inspection, Testing and Cleaning Procedures for Non-Nuclear Instrumentation Transmitters.

17. B&W Inspection, Testing and Cleaning Procedures for Non-Nuclear Instrumentation - Flow Nozzles and Reservoirs.

18. B&W Inspection, Testing and Cleaning Procedure for Non-Nuclear Instrumentation - Diaphragm Operated Valves.

19. Federal Register Section 10, Part 20.

3. General Requirements

1. All equipment will comply with the specification and with the applicable statutes, rules and regulations, including those of the Atomic Energy Commission, the United States Coast Guard, and the American Bureau of Shipping. Where existing regulations do not apply, the ASME codes will be used as a guide and the matter will be referred to the Atomic Energy Commission for approval.

2. All equipment will be designed to function satisfactorily when the ship is at a permanent list of 15 degrees or at a permanent trim of 5 degrees. Also, the equipment will function satisfactorily during a continuous oscillating roll condition of 30 degrees from the vertical to each side with a period of approximately 14 seconds, and at a continuous oscillating pitch condition of 7 degrees from the horizontal up and down by the bow with a pitching period of approximately 7 seconds. For calculation, the permanent list and the trim condition will not be considered as additive to the oscillating roll and pitch conditions. All electrical enclosures must be designed for these conditions.

3. The equipment will be designed to withstand the following forces in addition to dead weight and mechanical loadings. These forces constitute all of those which might be imposed by the permanent list, roll, pitch, heave, etc., of the ship.

   Vertical Acceleration . . . . 0.30 g's, maximum
   Athwartship Acceleration . . . . 0.60 g's, maximum
   Fore and Aft Acceleration. . . . . 0.25 g's, maximum

   These values are not to be considered as acting simultaneously.

4. No component will have a natural resonant frequency between the critical range of exciting frequencies of 5 and 10 cycles per second.

5. All components inside the containment vessel will be protected on the outer surface against corrosion.

6. As many swing check valves will be located in a fore-aft direction as is practical.

4. Material Requirement

All materials will be in accordance with the Marine Engineering Regulations and the Materials Specification for the United States Coast Guard.

5. Water Quality

1. Primary-System Initial Fill

Total Solids (other than boric acid), ppm	0.05
O_2, (Hydrazine Controlled), ppm	0.01
Chlorides as Chloride Ion, ppm-maximum	0.01
Boric Acid, ppm	9370

2. Feedwater - Recommended

Total Dissolved solids, ppm	0.05
Suspended solids	Minimum (preferably zero)
Hardness, ppm	Zero
Organic, ppm	Zero
Free Caustic, ppm	Zero
Dissolved Oxygen, ppm	0.007 Maximum (preferably zero)
Carbon Dioxide, ppm	Minimum (preferably zero)
Chlorides as Chloride Ion, ppm-maximum	0.01
Total silica as SiO_2, ppm-maximum	0.02
Total iron as Fe, ppm-maximum	0.01
Total copper as Cu, ppm-maximum	0.002

The pH value should be adjusted to obtain a maximum 0.01 ppm of iron. This will normally require a pH value within the range of 8.5 to 9.2 at 77 F.

3. <u>Containment Water</u>

Potassium Hydroxide	pH of 8.5 to 9.5
Chlorides as Chloride Ion, ppm-maximum	1.0
Boric Acid, ppm	9370

APPENDIX 1
PART FIVE

Preliminary Issue System Design Specification for the Relief System

CONTENTS

1. Scope

This specification describes the relief system for the CNSG. (See Fig. 3.1-1.) It is for design information and is not for fabrication.

2. References

1. General Technical Specification, B&W Specification PD-1-191-0(59-3034).
2. Preliminary Schematic of CNSG Systems, Figure 3.1-1.

3. Function and General Description

1. Function

The relief system will protect the following from excessive pressure:

1. the reactor
2. the steam generator if it is isolated
3. the entire secondary system
4. the chemical control system condensate tanks
5. the containment
6. the containment heat exchangers if they are isolated

2. General Description

The relief system consists of the primary system relief equipment, the steam generator relief equipment, the secondary relief equipment, containment relief equipment, and the containment heat exchanger relief equipment.

2.1 Primary Relief Equipment

If a main turbine trip occurs with a resulting failure of the fast set back, the relief valves protect the reactor from excessive pressure. They relieve the steam generated by the decay heat in excess of the heat removed through the reactor vessel wall. The high pressure scram and the high temperature scram are assumed to function properly. The primary relief valves discharge to the condensate tank in the chemical control system.

2.2. Steam Generator Relief Equipment

If a portion of the steam generator is filled with water and is isolated, the relief valve protects this section from excessive pressure. The water from these relief valves is discharged into the containment water.

2.3. Secondary Relief Equipment

The secondary relief valves relieve the steam generated when the turbines are down and the reactor is operating at maximum power. These valves discharge to the atmosphere.

2.4. Containment Relief Equipment

If the ship is grounded, the decay heat is transferred from the reactor to the containment water. After 37 hours, assuming no heat loss from the containment, the pressure reaches 50 psig. Since the containment is designed for 50 psig, the relief valves can relieve the steam generated from decay heat. These valves relieve to the atmosphere.

2.5. Containment Heat Exchanger Relief Equipment

If a containment heat exchanger is isolated, the relief valve protects it from excessive pressure. The relief valves discharge to the sea water overboard-discharge line down stream of the last block valve. The set pressure of these valves exceeds the maximum containment pressure to prevent the release of containment water.

2.6. Condensate Tank Relief Equipment

If the condensate tanks are full, the relief valve can relieve at a rate equal to that at which steam or water is discharged to the tanks.

4. Design Requirements

1. Design Pressures, Temperatures, and Flow Rates

Equipment	No. valves	Pressure, psig	Temperature, F	Total flows, lb/hr
Primary	2	1100	560	23,000
Condensate tank	1	400	450	23,000 or (67 gpm)*
Steam Generator	3	1220	560	(10 gpm)
Secondary	2	500	470	336,000
Containment	1	50	300	1,280
Containment Heat Exchangers	4	65	300	(0.05 gpm)

*Numbers in parentheses are water; all others are steam.

2. The primary relief valves remove the decay heat in excess of that removed by the decay heat system when the reactor is scrammed by the high pressure scram or by the high temperature scram (no heat is removed by the steam generator).

3. The steam generator relief valves relieve the expansion water from an isolated steam generator when the primary system temperature is increasing at the rate of 10 F per minute.

4. The secondary relief valves relieve the maximum steam generation (125% of the design steam flow).

5. The condensate tank relief valve relieves the maximum rate of water or steam released into the tank.

6. The containment relief valves can remove the steam generated in the containment vessel by decay heat after 37 hours of decay.

7. The containment heat exchanger relief valves can discharge the expansion water when the containment temperature increases due to a reactor vessel rupture.

5. Operating Conditions

Normally, all valves remain closed. The set pressures at which the valves are required to open are:

Valves	Set pressure psig	Maximum back pressure, psig
Primary No. 1	1100	400
No. 2	1150	400
Condensate tank	400	50
Steam Generator No. 1	1220	50
No. 2	1220	50
No. 3	1220	50
Secondary No. 1	495	--
No. 2	515	--
Containment	50	--
Containment Heat Exchangers No. 1	65	--
No. 2	65	--
No. 3	65	--
No. 4	65	--

6. Material Requirements

All valves and piping associated with the primary system and the condensate tank will be stainless steel TP-304, TP-316 or the equivalent. The downstream side of the condensate tank discharge valve may be carbon steel.

All valves and piping associated with the steam generators, the secondary system, and the containment vessel will be carbon steel or the equivalent.

The sea water heat-exchanger relief valves will be of a material capable of handling sea water.

7. Flow Diagram

Figure 3.1-1 is the flow diagram of the system and is part of this specification.

8. Relationship to Other Systems

1. Reactor System

Safety valve effluent is piped to the condensate tank.

2. Steam Generators

Safety valve effluent is discharged under water to the containment vessel.

3. Secondary System

Safety valve effluent is discharged to the atmosphere.

4. Containment System

The safety valves from the condensate tank and steam generators discharge under water to the containment.

The containment relief valves discharge to the atmosphere.

The sea water heat-exchanger relief valves discharge to the sea through the overboard discharge line.

5. Chemical Control System

The chemical control system receives the primary relief valve discharge.

The condensate tank relief valve discharges to the containment.

9. Instrumentation Requirements

None

10. Electrical Requirements

None

11. Requirements from Other Services

None

12. Shielding Requirement

None

13. Location of Equipment

The secondary relief valves are located on the common steam piping outside the containment. The containment relief valve is located on top of the containment. The sea water heat exchanger relief valves are on the discharge line across the stop valve. All other equipment is inside the containment vessel.

14. Insulation Requirements

The secondary relief valves are canned.

The primary and the steam generator relief valves are insulated through the discharge connection with 2.5 inches of canned insulation.

APPENDIX 1
PART SIX

Preliminary Issue System Design Specification for the Containment System

CONTENTS

1. Scope

Described here is the containment system to be installed in a CNSG-type plant. (See Fig. 3.1-1.) This specification gives design information and is not for fabrication.

2. References

1. General Technical Specification, B&W Specification No. PD-1-191-0 (59-3034).
2. Preliminary Schematic of CNSG Systems, Figure 3.1-1.

3. Function and General Description

3.1. Function

The containment system performs the following functions:

1.1 Contains the primary water following a reactor vessel rupture.

1.2 Contains sufficient heat capacity to absorb decay heat for 37 hours.

1.3 Serves as a heat sink for:

1. Decay heat removal during normal and emergency conditions.
2. Motors of the primary water circulating pumps.
3. Motors of the control rod drives.
4. Primary water sampling heat exchanger.
5. Condensation in the condensate tanks.
6. Condensate tank relief valve.
7. Steam generator relief valve.

1.4 Contains all gases until disposal is convenient.

1.5 Serves as a portion of the biological shield.

3.2. General Description

Following a reactor vessel rupture, the energy stored in the primary system is released to the containment. The heat is absorbed by the containment water by raising the temperature approximately 35 F. The air space is sufficient to limit the pressure rise caused by the addition of primary water and by expansion of the containment water.

The decay heat is removed through a section of the reactor vessel. The reactor vessel is covered with lead sized (in thickness) to limit the heat transfer to 2.2×10^6 Btu/hr when the containment water is at 110 F.

The containment water serves as a heat sink. This heat is removed from the containment water by four containment heat exchangers. The heat exchangers are served by two sea water circulation pumps of two natural circulation lines. During normal operation, one circulation pump is used.

The natural circulation is used for a dead ship condition when power is not available to the pumps. The ship must be riding at its load line or natural circulation is not possible.

The containment contains sufficient water to limit the temperature and pressure rise for a ship which is stranded. At the end of 37 hours, the pressure will be 50 psig at 255 F.

The radioactive gases and hydrogen are controlled in the containment gas space. Provisions are made for dilution before the discharge to the atmosphere through the stack.

The fill and drain provisions consist of a gate valve with a suitable connector and adaptors for other type connectors.

4. Design Requirements

1. Containment Vessel

Design Pressure, psig	50
Design Temperature, F	300
Volume, ft^3	
Air Space	780
Water	6 050

2. Sea Water Pumps (2)

Flow Rate, gpm	
Maximum	500
Normal	500

Suction Head, ft	9-20
Total Developed Head, ft	50
Horsepower	10
Location	Outside Containment

3. Sea Water Heat Exchangers (4)

Heat Load, Btu/hr (each)

Maximum	7.5×10^5
Normal	6.0×10^5

3.1. Sea Water-Natural Circulation

Sea Water (shell side) (each)

Flow, nat circ, lb/hr	$\approx 4 \times 10^4$
Inlet Temperature, F	85
Outlet Temperature, F	100

Containment Water (tube side)

Flow, nat conv, lb/hr	$\approx 1 \times 10^5$
Inlet Temperature, F	113
Outlet Temperature, F	107

3.2. Sea Water-Forced Circulation, gpm

The heat exchangers are inside the containment mounted parallel to the containment center line.

4. Ventilation Blower

Capacity, cfm	1000
Developed Head, in.	3
Horsepower	1
Location	Outside Containment

5. Piping and Fittings

All piping and fittings will be of the thickness required by the codes listed in Reference 1. Most of the piping and fittings inside the containment vessel will be of a welded construction. All hookups will be welded except the sea water heat exchanger which will be flanged. The piping outside the containment vessel shall be of a flanged construction.

Electrolytic corrosion must be prevented in the bimetallic connection.

5. Material Requirement

   1. Carbon Steel

      1. Pipe and fittings for fill and ventilation lines
      2. Containment vessel
      3. Ventilation valves

   2. Bronze

      1. Sea water pumps
      2. Fittings (sea water circuit)
      3. Ventilation blower
      4. Valves (sea water circuit)

   3. 90-10 Cu Ni

      1. Piping (sea water line)

   4. 70-30 Cu Ni

      1. Tubes for sea water heat exchangers
      2. Tube sheet for sea water heat exchanger
      3. Shell for sea water heat exchanger

6. Flow Diagram

   Figure 3.1-1 is the flow diagram of the system and is part of this specification.

7. Relationship to Other Systems

   1. Reactor System

      This system removes 2.2×10^6 Btu/hr of heat from the reactor.

   2. Relief System

      The containment water cools the condensate tank, condenses and cools the steam or water from the condensate tank relief valve and the steam generator relief valves.

   3. Ship's Water Supply

      The ship's water supply provides makeup.

4. Chemical Control System

The containment water cools the sample line heat exchanger for the primary water. The chemical control system controls the additives in the containment water, and provides the sampling equipment for the containment water.

8. Instrumentation

The proposed instrumentation and control will detect, measure, and indicate the variables of pressure, temperature, level, flow, and radiation. Detectors, transmitters, indicators, switches, and relays comprise the instrumentation.

The detection, measurement, etc., of the variables mentioned in the preceding paragraph shall read at the following locations:

1. Level

The containment water level is measured and indicated on the console. A high and low level alarm is provided.

2. Pressure

The containment pressure is measured and indicated on the console.

3. Flow

The sea water flow to the sea water heat exchangers is measured and indicated locally. A low flow alarm is provided.

4. Temperature

The temperature of the containment is measured at approximately 30 points, and the average temperature is indicated on the control console. An alarm is sounded at a pre-set high temperature. The temperature at the inlet and outlet of each sea water heat exchanger is measured, and is indicated on a panel-mounted duplex indicator in the machinery room. A selector station permits an operator to change to any of the four heat exchangers.

5. Radiation

A radiation monitor is used in the containment ventilation line and is indicated locally. An alarm is sounded in the control room at a pre-set high level when the stack discharge valve is open.

9. Electrical Requirements

Name of Equipment	No. Installed	No. Normally Operating	Horsepower Rating
Sea Water Pumps	2	1	10
Ventilation Blower	1	0	1

The sea water pumps controlled from the main control room also will require a local control. During normal operation, power is required continuously to provide sufficient sea water for the sea water heat exchangers.

The ventilation blower controlled from the main control room also will require local controls.

10. Requirements from Other Services

None

11. Insulation Requirements

None

APPENDIX 2
Dispersion of Radioactivity Into the Ocean

Reference 4 has developed a diffusion model for predicting the rate at which radioactive particles are dispersed through the ocean. This reference was written to determine the quantity of radioactivity that could be safely discharged continuously from a large number of nuclear powered ships. The accident analyzed in this section is a single occurrence. The probability of more than one nuclear powered ship sinking in the same location within a period of several years time is negligible. Therefore, the method of analysis used here differs slightly from that presented in Reference 4 of Section 8.1.

The general equation derived in this reference has been rearranged to aid in calculation:

$$s_m \doteq \left[\frac{8\, t_{1/2}}{9\, DAT^2}\right]^{2/3} \left[\frac{n}{2\,\pi\, DP^2}\right]^{1/3} M \qquad (1)$$

where

s_m = mean concentration of an isotope at the end of a time period T, μc/ml

$t_{1/2}$ = time interval to replace 50% of the water with new water from an adjacent uncontaminated area, sec

D = layer thickness of volume, m

A = area of a particular marine region in which the ship sinks, m^2

T = time period of investigation, sec

n = correction factor for confined waters, dimensionless

P = diffusion velocity, m/sec

M = total activity of specific isotope released, μc.

Using the values given in the reference material, this equation can be evaluated as

$$s_m = 3.16 \times 10^{-10}\ M. \qquad (2)$$

The total activity of a specific isotope, M, to be used in the above equation was obtained from Reference 5 of Section 8.1. The value for the isotope activity at the end of 600-days operation at 69 MW for the Savannah core was corrected for the CNSG power level of 63 MW and is used in the analysis. Only those isotopes having a half life exceeding 6 hours and a fission product yield above 0.2% are considered important in the analysis. The activities of these isotopes are listed in Column 3 of Table 2-1.

In determining the quantity of fission products released, the proposed AEC Site Guide Criteria considers gross meltdown of the fuel to occur. As a result, 100% of the noble gases, 50% of the halogens, and 1% of the solids in the fission product inventory are assumed to be released to the sea water. While there is no reason to believe that gross melting of the core will occur, considering that an infinite heat sink (the ocean) is available for core cooling, these percentages of fission product inventory are considered to be released to the sea. The value of the isotope activity released to the sea is listed in Column 4 of Table 2-1.

The equation derived for the diffusion model takes no credit for the decay of individual isotopes. Therefore, the mean concentration obtained from the equation is corrected for the radioactive decay of the specific isotope. The resulting concentrations (Column 6 of Table 2-1) are average concentrations for the first year after fission product release. For comparison, the maximum permissible concentration in water for occupational exposure is listed in Column 5 of the table.

Comparison of Columns 5 and 6 shows that no isotope concentration exceeds the maximum permissible. Since sea water is not usually consumed by the general public, the only method by which radioactivity can be transmitted is by consuming the sea food inhabiting the environment. This method of analysis is used in Reference 4, Section 8.1.

Marine organisms take up from their environment and their food and incorporate into their bodies those elements required for their maintenance, growth, and reproduction. The proportion of various elements required by the organisms differs from the actual proportions in the environment. This results, therefore, in the concentration of some elements. A person who obtained all his protein requirement from marine sources would eat about 1.5 kg per week. Since the major portion of the fission products released from the accident would float around the ocean in similar manner to a radioactive cloud, the marine organism mainly affected would be fish. If the fish are contaminated with a radioisotope, the maximum permissible concentration in the fish corresponding to the maximum permissible concentration in drinking water can be calculated by multiplying the latter by the ratio

$$\frac{\text{water ingested per week}}{\text{fish ingested per week}} = \frac{2200 \times 7}{1500} \cong 10.$$

Therefore, the concentration of an isotope in fish can be ten times the allowable concentration in drinking water for a comparable dose of radiation.

The factor by which fish concentrate various isotopes was obtained from References 4 and 6 of Section 8.1. Where no data were available for a specific isotope, the factor for a known element in the same group and subgroup of the period table was used. This factor, together with the ingestion factor and the maximum permissible concentration obtained from the Code of Federal Regulations, is used to determine the maximum permissible concentration for the environment. The value of the MPC for the environment of the specific isotopes analyzed is listed in Column 8 of Table 2-1.

To determine whether the combined isotope activity is greater than that allowed, the fraction of the allowable activity for each isotope was obtained (Column 9 of Table 2-1). The total activity indicates that the dose of radiation received by a person is below the maximum permissible. It is concluded that no undue hazard exists to the general public as a result of the assumed accident.

Table 2-1. Activity of Isotopes after 600 days Operation

Isotope	Half-life	Total Curies In Core	Total Curies Released	MPC per 10 CFR 20	Sea Water Average Concentration, (T = 1 year), μc/ml	Concentration Factor	MPC for Environment, μc/ml	Mean Conc of Isotope / MPC for Environment
Sr-89	51d	2.37×10^5	2.37×10^3	3×10^{-4}	1.58×10^{-7}	1	3×10^{-4}	0.00053
Sr-90	28 yr	1.2×10^4	1.2×10^2	4×10^{-6}	3.78×10^{-9}	1	4×10^{-6}	0.00095
Y-90	64.2h	1.2×10^4	1.2×10^2	6×10^{-4}	3.78×10^{-9}	1.2	5.0×10^{-4}	<0.00001
Sr-91	9.7h	2.89×10^5	2.89×10^3	2×10^{-3}	1.52×10^{-9}	1	2×10^{-3}	<0.00001
Y-91m	50.3m	1.73×10^5	1.73×10^3	1×10^{-1}	9.08×10^{-10}	1.2	8.3×10^{-2}	<0.00001
Y-91	58d	2.96×10^5	2.96×10^3	8×10^{-4}	2.24×10^{-8}	1.2	6.7×10^{-4}	0.00003
Y-93	10h	3.3×10^5	3.3×10^3	8×10^{-4}	1.78×10^{-10}	1.2	6.7×10^{-4}	<0.00001
Zr-95	63d	3.44×10^5	3.44×10^3	2×10^{-3}	2.83×10^{-7}	4	5×10^{-4}	0.00057
Nb-95m	84h	3.44×10^5	3.44×10^3	3×10^{-3}	2.83×10^{-7}	2	1.5×10^{-3}	0.00019
Nb-95	35d	3.44×10^5	3.44×10^3	3×10^{-3}	1.41×10^{-7}	2	1.5×10^{-3}	0.00009
Zr-97	17h	3.22×10^5	3.22×10^3	5×10^{-4}	2.96×10^{-9}	4	1.25×10^{-4}	0.00002
Nb-97m	1m	3.22×10^5	3.22×10^3	3×10^{-2}	2.96×10^{-9}	2	1.5×10^{-2}	<0.00001
Nb-97	74m	3.22×10^5	3.22×10^3	3×10^{-2}	2.96×10^{-9}	2	1.5×10^{-2}	<0.00001
Mo-99	67h	3.22×10^5	3.22×10^3	5×10^{-3}	1.17×10^{-8}	2	2.5×10^{-3}	0.00001
Ru-103	41d	2.32×10^5	2.32×10^3	2×10^{-3}	1.25×10^{-6}	100	2×10^{-5}	0.06250
Rh-103m	54m	2.32×10^5	2.32×10^3	4×10^{-1}	1.25×10^{-6}	100	4×10^{-3}	0.00031
Te-131m	30h	2.41×10^4	2.41×10^2	2×10^{-3}	3.88×10^{-10}	1	2×10^{-3}	<0.00001
Te-131	24.8m	1.78×10^5	1.78×10^3	2×10^{-3}	2.86×10^{-9}	1	2×10^{-3}	<0.00001
I-131	8.05d	1.78×10^5	8.9×10^4	6×10^{-5}	9.32×10^{-8}	1	6×10^{-5}	0.00155
Te-132	77.7h	2.35×10^5	2.35×10^3	9×10^{-4}	9.83×10^{-9}	1	9×10^{-4}	0.00001
I-132	2.33h	2.35×10^5	1.16×10^5	2×10^{-3}	4.85×10^{-7}	1	2×10^{-3}	0.00024
I-133	21h	3.34×10^5	1.67×10^5	2×10^{-4}	1.95×10^{-7}	1	2×10^{-4}	0.00098
I-135	6.7h	3.09×10^5	1.55×10^5	7×10^{-4}	5.64×10^{-8}	1	7×10^{-4}	0.00008
Cs-137	30yr	1.06×10^4	1.06×10^2	4×10^{-4}	3.34×10^{-8}	1	4×10^{-4}	0.00008
Ba-137m	2.6m	9.77×10^3	9.77×10^1	5×10^{-3}	3.08×10^{-8}	1	5×10^{-3}	<0.00001
Ba-140	12.8d	3.26×10^5	3.26×10^3	8×10^{-4}	5.42×10^{-8}	1	8×10^{-4}	0.00007
La-140	40h	3.26×10^5	3.26×10^3	7×10^{-4}	5.42×10^{-8}	1.2	5.8×10^{-4}	0.00009
Ce-141	32d	3.1×10^5	3.1×10^3	3×10^{-3}	1.27×10^{-7}	1.2	2.5×10^{-3}	0.00005
Ce-143	33h	3.2×10^5	3.2×10^3	1×10^{-3}	5.73×10^{-9}	1.2	8.3×10^{-4}	<0.00001
Pr-143	13.7d	3.2×10^5	3.2×10^3	1×10^{-3}	5.32×10^{-8}	1.2	8.3×10^{-4}	0.00006
Ce-144	290d	2.37×10^5	2.37×10^3	3×10^{-4}	5.23×10^{-7}	1.2	2.5×10^{-4}	0.00209
Pr-144	17.5m	2.37×10^5	2.37×10^3	1×10^{-3}	5.23×10^{-7}	1.2	8.3×10^{-4}	0.00063
Nd-147	11.3d	1.33×10^5	1.33×10^3	2×10^{-3}	1.98×10^{-8}	1.2	1.67×10^{-3}	0.00001
Pm-147	2.52yr	4.69×10^4	4.69×10^2	6×10^{-3}	1.34×10^{-7}	1.2	5×10^{-3}	0.00003
							Total	0.07127

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