THE CNSG II
A CONCEPTUAL MERCHANT SHIP
NUCLEAR REACTOR DESIGN

September 1963

Facsimile Price \$ 7.60
Microfilm Price \$ 2.51

Available from the
Office of Technical Services
Department of Commerce
Washington 25, D. C.

AEC Contract No. AT(30-1)-3206
B&W Contract No. 59-3085

Submitted to
THE UNITED STATES ATOMIC ENERGY COMMISSION
By
THE BABCOCK & WILCOX COMPANY
Atomic Energy Division
Lynchburg, Virginia

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

ABSTRACT

The Babcock & Wilcox Company has designed the Consolidated Nuclear Steam Generator II (CNSG II), an improved pressurized water reactor that is more economical than conventional power plants for high shaft horsepower and high load factor applications. The entire primary system is contained in one pressure vessel, resulting in an extremely compact and light system. The refueling cycle is completed in only 10 days.

The CNSG II design presented here is for 66,000 shaft horsepower, but basic plant information is given for designs of 15,000, 22,000, and 30,000 shp.

CONTENTS

	Page
1. INTRODUCTION	1-1
1.1. Design Factors	1-1
1.2. General Reference Design	1-3
2. ECONOMICS	2-1
2.1. Capital Cost Estimate	2-2
2.2. Fuel Cost	2-3
2.3. Weight and Space of Reactor Plant	2-4
3. SAFETY CHARACTERISTICS	3-1
4. OPERATIONAL PROCEDURES	4-1
4.1. General Characteristics	4-1
4.2. Normal Operation	4-1
4.3. Reactor Plant Warmup and Cooldown	4-2
4.4. Decay Heat Removal	4-2
4.5. Effect of Ship's Attitude and Motion	4-3
4.6. Refueling	4-3
4.7. Maintenance	4-5
4.8. Response to Accidents	4-6
5. DESIGN	5-1
5.1. General Description	5-1
5.2. Containment	5-2
5.3. Reactor Vessel Arrangement	5-2
5.4. Auxiliary Systems	5-9
5.5. Shielding	5-13
5.6. Instrumentation and Control	5-15
5.7. Erection	5-15
6. DESIGN HISTORY	6-1
6.1. Introduction	6-1
6.2. Pressurized Water Reactor Design	6-1
6.3. Integral Boiling Reactor	6-3
6.4. Wet Containment	6-3
6.5. CNSG I, Phase I	6-5
6.6. CNSG I, Phase II	6-6
6.7. Burnable Poison Core Design	6-7
6.8. CNSG II Design	6-8
APPENDIX	
Plant Information Sheets	A-1

List of Figures

Figure

The Following Figure Follows Page 1-4

- 1-1. CNSG II General Arrangement

The Following Figures Follow Page 2-4

- 2-1. CNSG Cost Estimate
- 2-2. Annual Fuel Cost (66,000 shaft horsepower)
- 2-3. Unit Fuel Cost (66,000 shaft horsepower)
- 2-4. Comparative Size of Containments
- 2-5. Weight Vs Horsepower Comparison for Nuclear Plant

The Following Figure Follows Page 4-6

- 4-1. Heat Balance and Flow Diagram

The Following Figures Follow Page 5-6

- 5-1. CNSG II Reactor and Containment Arrangement
- 5-2. Core Plan and Control Rod Positions
- 5-3. CNSG II Hot Channel Design (66,000 shaft horsepower)
- 5-4. Preliminary Systems Design

The Following Figures Follow Page 6-9

- 6-1. PWR Plant Arrangement (30,000 shaft horsepower)
- 6-2. IBR Plant Arrangement
- 6-3. IBR Containing Bayonet Tube Boiler
- 6-4. IBR General Arrangement
- 6-5. Dual Purpose Plant (82.5/400 MW-t)
- 6-6. CNSG I, Reactor and Containment Arrangement
- 6-7. Advanced CNSG Scheme I

1. INTRODUCTION

In 1962 The Babcock & Wilcox Company (B&W), working under an AEC contract, developed the Consolidated Nuclear Steam Generator (CNSG) design. This was an advanced pressurized water reactor design that included a large number of new ideas, such as wet containment, consolidation of the primary system into a single pressure vessel, and once-through steam generators. The CNSG achieved economic parity with conventional power plants for high-speed, long-distance cargo ship service.

B&W's first CNSG design was for a 20,000 shaft horsepower (shp) plant. However, after ship application studies by George G. Sharp, Inc. showed that proper application for nuclear power plants was in the 60,000-shp range, the Atomic Energy Commission requested B&W to modify the original design to produce a 66,000-shp CNSG. The Commission also requested B&W to study means of simplifying the refueling operation and making the power plant more compact, lighter, and less costly. In addition to studying the reference design at 66,000 shp, B&W was asked to extrapolate the basic information to power plants producing from 15,000 to 66,000 shp. The results of these efforts is the CNSG II design presented in this report.

1.1. Design Factors

The CNSG II was designed specifically as a marine reactor and includes many features that are of major importance to marine operators:

1. Reliability - The plant has backups for all rotating, electrical, and electronic equipment and can continue to operate with a section of the boiler isolated. The design is based to a major degree on components or concepts that have been proved in operating power plants. The power plant is basically self-limiting and cannot be damaged by control system failures.

2. Simplicity - Elimination of all frills from the design provides a plant that is easily understood by the operator and is simple to operate and maintain.

3. Maintenance - The containment and all equipment exposed to radioactivity are arranged for easy, quick access. Whenever possible, sensitive equipment is located outside of the containment for ready access.

4. Radiation safety - The radioactivity level in the machinery space during operation is 1.0 mrem/hr. All sources of radioactivity are closely contained.

5. Ease of refueling - Arrangement of the reactor plant has reduced refueling time to 10 days or less.

6. Weight and space - Both of these factors are significantly improved over those of previous marine power plants.

7. Cost - The CNSG II is designed for minimum cost consistent with the foregoing factors. The low-enriched UO_2 core reduces fuel costs significantly. Through simplicity and attention to such parameters as temperature and pressure, capital costs have been minimized.

8. Ease of installation - To reduce the time required for plant assembly and shipyard installation, the system can be installed in a small number of individual units. For instance, delivery of the entire primary system as a single, sealed, clean unit, would minimize cleanliness requirements in the shipyard during erection.

B&W achieved the desired characteristics by basing its design on a firm foundation of past experience. Elements contributing to the excellence of the design are as follows:

1. Consolidated design - Containment of the entire primary system in one vessel provides great simplicity and control over radioactivity, reduces weight and space requirements, and provides an easily installed system.

2. Pressurized water - Being a pressurized water reactor the CNSG II benefits from U. S. Navy and NS Savannah experiences afloat, and from the experience of many other reactors ashore.

3. Once-through steam generator - The once-through steam generator facilitates consolidation and greatly simplifies steam generator construction.

4. Self-pressurization - The self-pressurization eliminates the need for a separate pressurizer and its controls.

5. Wet containment - Wet containment is a great advancement in reactor safety and design simplicity.

6. Low-enriched UO_2 core - This core, of the type used in the Savannah, provides very low fuel costs and inherent self-regulation.

7. Simplified refueling - Arrangement of the reactor plant minimizes the number of operations required for refueling. The control rod drives are not removed during refueling, nor is it necessary to make and break the usual electrical and piping connections. These, along with other requirements, greatly minimize the time spent before and after actual fuel handling.

1.2. General Reference Design

A cross section of the CNSG II is shown in Figure 1-1. The pressure vessel, which contains the reactor core, steam generator, and circulating pumps, is largely filled with water for transporting heat and moderating neutrons.

The core, centrally located in the reactor vessel (see Figure 1-1), consists of 16 fuel elements assembled from Zircaloy tubes containing low-enriched UO_2 . Water flows upward around these fuel tubes, absorbs heat, and then leaves the core at a temperature of 544 F. Above the core the water turns radially outward, passes through the pumps, and then flows down through the steam generator, which consists of helical-coiled Inconel tubes. The heat absorbed by the water is transferred to the steam generator to make superheated steam (400 psig and 523 F), which in turn drives the turbines. Leaving the steam generator, the water flows downward for a short distance, turns radially inward through the flow baffle, and then turns upward and re-enters the core.

To provide for maximum accessibility, the reactor vessel is arranged so that maintenance and repair functions, except for control rod drives, are performed at the top of the system. The control rod drives are maintained through access tubes from the outside of the vessel so that the containment will not have to be drained.

The refueling time has placed severe economic penalties on nuclear reactors previously proposed for marine application. The CNSG II incorporates the following features, which eliminate these objectionable characteristics of nuclear reactors.

The core is arranged so that a minimum of equipment must be dismantled for the refueling process. Fuel elements are retained by bottom latches, thus eliminating the conventional upper grid plate; control rod drives are placed below the core. The reactor must be refueled once every 4 years; refueling is estimated to require 10 days or less.

The steam generator, placed concentrically around the core, has inlets and outlets at the top of the vessel to permit easy plugging of the tubes if necessary. Vertical positioning of the steam generator relative to the core was selected so that N^{16} activity will be low in the steam to the propulsion turbine.

The reactor vessel is surrounded by a water-filled containment (Figure 1-1). The containment is required for safety in case of a failure in the primary system. The water in the containment will absorb the energy released from a system rupture, as well as shield personnel from radiation during operation.

Figure 1-1. CNSG II General Arrangement

2. ECONOMICS

The CNSG II, in selected applications, will propel a ship more economically than a conventional power plant.

The major factors involved in an economic analysis are the capital cost and the fuel cost. The weight of the power plant and its fuel are also involved, since it may affect the size of the ship and the horsepower required to propel it.

At the present time, construction of a nuclear ship will probably include government participation. The nonrecurring costs associated with the first nuclear power plant of a new kind are too great to be recovered in building the first several plants. The federal government has generally assumed the burden of these nonrecurring costs to encourage the development of nuclear power. For example, the government constructed the NS Savannah as a necessary step in the development of marine nuclear power that could not possibly be commercially justified. The government has also financed several projects to develop technology for more advanced nuclear power plants.

Government policy for construction of a second nuclear ship has not been completely formulated. An indication of its possible form can be obtained from the Garmatz Bill (HR 1071), which is presently before Congress. This bill, an amendment of the 1936 Merchant Marine Act, provides that the Secretary of Commerce can, under certain circumstances, grant the following financial aid to the constructor of a nuclear ship:

1. Provide payment of development costs.
2. Provide payment for all or part of the excess cost of the nuclear ship over the conventional ship.
3. Waive or reduce charges for nuclear fuel.
4. Assist in planning or designing shore facilities.

These aids are given at the discretion of the Secretary of Commerce. Before an aid is granted, the Secretary must determine whether it will be beneficial to the government according to criteria specified in the bill. The financial aid of the Garmatz Bill or similar bills is not provided after nuclear power is fully developed. The merits of nuclear power must then first be judged by the normal economic rules. One major factor the federal government considers in deciding whether to grant aid to a nuclear ship is the economic merit of the reactor concept when fully developed. This is judged by cost estimates for the second ship of a given kind.

The second-of-a-kind costs are used in the economic analysis given in the following sections since these are the figures of real interest to prospective ship owners and the AEC.

2.1. Capital Cost Estimate

The capital cost estimate is the estimated cost of the reactor plant as installed in the ship. This includes the equipment and services supplied by the reactor manufacturer and the shipyard. Costs are given for a range in plant sizes from 15,000 to 66,000 shp. Since the reference design for the study was the 66,000-shp size, cost estimates were made for this size and extrapolated for the other sizes.

In an estimate of second-of-a-kind costs, it is assumed that an identical reactor plant has been constructed previously and the first plant has absorbed all of the nonrecurring costs, such as design, development, and safeguards reviews. The only engineering needed for the second-of-a-kind construction is that required to procure, expedite, and test equipment and to modify the safeguards reports for application to the second ship. The largest single entry in the cost estimate is the reactor vessel and its internals, including the steam generator. (The cost of this item was estimated by B&W's Barberton Works.)

The cost of primary circulating pumps for the CNSG II was based on estimates from pump manufacturers. Cost of control rods and control rod drives were based on recent cost estimates for similar components. Cost of auxiliary systems was based on the flow diagram (Figure 5-4). The characteristics for the required components were developed from this diagram. Their costs were then estimated by using similar, recent costs as a basis. The instrumentation item in

the cost estimate includes the automatic valves shown on the flow diagram, the nuclear instrumentation system, the safety system, and the power plant control system. Cost of these items was based on recent quotes for similar equipment. The sum of these entries is the cost of nuclear contractor-furnished equipment. Nuclear contractor engineering cost was based on Reference 1. Shipyard costs were extrapolated from similar information developed by a naval architect. The shipyard cost of a second ship is the sum of the expense of the following items: erection of the nuclear equipment; procurement of piping, fittings, and valves for auxiliary systems; procurement and erection of the containment and shielding; shipyard engineering; and overhead.

Indirect costs (startup testing, owner's engineering, and changes under the contract) are estimated at 7-1/2% of direct costs.

Figure 2-1 shows the estimated capital cost of the nuclear power plant for a range of horsepower.

2.2. Fuel Cost

The fuel cost for the CNSG II is much lower than that of conventional ships and also lower than other nuclear power plants of the same size. These low fuel costs are achieved through the efficient use of low-enriched uranium. Fuel cost for the 66,000-shp CNSG II are estimated at 1.84 mills/shp-hr.

Nuclear fuel is bought for a ship at intervals. In the case of the CNSG II, it is necessary to buy a new core or batch of fuel once every 4 years. Actually, since only the federal government may own nuclear fuel by law, the ship owner leases the fuel itself. He only buys the assembly containing a form of the fuel that can be used safely in the reactor. Only a portion of the fuel in each core is consumed. After this portion of the fuel is consumed, nuclear poisons build up to the point where the nuclear chain-reaction stops. At this time, the core must be removed from the ship, and the remaining fuel must be recovered and purified. The portion of the nuclear fuel present, but not consumed, has a high value. Hence, the lease charges on fuel that is not consumed are an appreciable portion of the total fuel cost. The fuel costs can be divided into the following categories:

1. Fabrication - Charge paid to the core manufacturer when the core is delivered.
2. Fissionable material - Charge for the fuel actually consumed during operation.
3. Inventory - Continuing charge on the amount of fuel present in the core.
4. Transportation, insurance, reprocessing, and conversion - Charges concerning the processes required to return the fuel to the government in such a form that it can be used in another reactor.

Figure 2-2 shows the estimated annual fuel cost for 66,000 shp. Because of the inventory cost, which does not vary whether the core is used or not, it has been necessary to show the utilization factor on the graph (Figure 2-2). This is the average percent of full power at which the reactor operates. For approximate studies, it may be assumed that the annual fuel bill is proportional to the horsepower. Figure 2-3 shows the same information in the form of mills/shp-hr. These fuel costs are based on the present prices. The future is expected to bring further reduction in fuel costs, due to continuing research programs being carried out by the government and private industry. For example, B&W is constructing a laboratory to carry out research on recycled fuel. Because of these programs, it is estimated that fuel costs 10 years from now will be about 1.3 mills/shp-hr, or about 70% of the present levels.

2.3. Weight and Space of Reactor Plant

Figure 2-4 presents the size of reactor plants for various horsepower. Figure 2-5 presents the reactor plant weight, excluding collision barrier, as a function of horsepower.

Figure 2-1. CNSG Cost Estimate

Figure 2-2. Annual Fuel Cost (66, 000 shaft horsepower)

Figure 2-3. Unit Fuel Cost (66,000 shaft horsepower)

Figure 2-4. Comparative Size of Containments

Figure 2-5. Weight Vs Horsepower Comparison for Nuclear Plant

3. SAFETY CHARACTERISTICS

The CNSG II possesses several improvements that enhance the plant's safety. Basically, however, this concept represents water-cooled reactor technology incorporating safety features that have been extensively reviewed and approved by various regulating bodies. In addition, many years of operating experience in numerous water-cooled reactor plants have demonstrated the safety and reliability of this system.

The CNSG II core design is similar in most respects to the low-enriched UO_2 rod-type core for the NS Savannah and other water reactor power plants. This concept possesses the following similar features: (1) a large prompt negative power reactivity coefficient due to the Doppler effect, (2) a large negative coolant temperature coefficient of reactivity, (3) a large heat capacity in the fuel oxide and metal, (4) a large heat capacity in the reactor coolant, and (5) a single coolant-moderator material.

The concept of self-pressurization with a gas over-pressure provides a reactor having no significant voids in the core. Slow excursions owing to small reactivity insertions ($<0.3\%$ of reactivity) are readily limited by the negative coolant density coefficient without control action. Power excursions with reactor periods as low as 3 milliseconds and reactivity additions of approximately $1.5\% \Delta k$ can be experienced without core damage with a scram action of many seconds. Kinetic experiments in the SPERT I facility, using NS Savannah critical experiment fuel pins, have verified these conclusions. This series of experiments demonstrated that a reactor period of 3.2 milliseconds caused by a step input of reactivity of $1.6\% \Delta k$ produced a peak power of 7,480 MW, an integrated energy of 114 MW-seconds, and a peak pressure of 15 psi. However, the peak fuel temperature was more than 100 F below melting; the core received absolutely no damage; and the reactor remained operative.

Reactivity induced excursions as serious potential accidents are ruled out in the CNSG II due to the absence of a source of cold, dense

water, the self-regulating properties consistent with slow rod speeds and simple control, and the inherent capability of the core to withstand large nuclear excursions without damage. In addition to the favorable effects of the water coolant in limiting nuclear excursions, the primary system water acts as a large thermal flywheel, providing an attenuation of thermal perturbations between the reactor and the steam side of the boiler. This provides a ready source of stored energy for increased load demands or a heat storage medium when needed, thus allowing a large mismatch of reactor power and boiler output during transients. This simplifies control requirements (minimizes sources of reactivity excursions) and reduces the probability and extent of temperature and pressure excursions.

These many strong, self-regulating safety features have already reduced the stringent requirements placed on external safety systems by early reactors. For example, early reactors required safety systems capable of scramming in only a few thousandths of a second, whereas present day water reactors containing oxide fuel rods are adequately protected by scrams that will act in several seconds. Furthermore, present day technology gives every indication that as more experience is gained this dependence on external safety systems may be reduced even further, thus providing a safer, as well as a more reliable power plant.

The concept of self-pressurization with a gas overpressure provides a reactor having no voids in the core. Therefore, the motion of the ship in heavy seas cannot affect the reactor stability through reactivity changes brought about by the effect of gravity variations on the voids.

The maximum credible accident for the CNSG II is assumed to be a rupture in the primary system. In comparison with other water reactors, the probability of this accident is greatly reduced by providing a primary system that is essentially completely within the reactor vessel, the most reliable pressure containing component in a reactor system. Should a primary system rupture occur, such as by failure of one of the vessel penetrations, the energy released will be absorbed by the water surrounding the reactor vessel. In this manner the pressure buildup in the containment vessel is kept to a minimum. This pressure suppression arrangement is designed to function regardless of the ship's attitude.

4. OPERATIONAL PROCEDURES

4.1. General Characteristics

When comparing the CNSG II to other nuclear reactor plants, its outstanding characteristic is simplicity. Experience with the NS Savannah has not only shown the problems inherent in operating a complex power plant, but it has also shown that great simplification can be effected. At the time the Savannah was built, many of the characteristics of the nuclear reactor were not known; therefore, provisions were made for a great many problems that have been proved nonexistent.

A second characteristic of the CNSG II that will simplify its operation is the inherent stability of the system. As was explained in the section on safety analysis, inherent characteristics tend to oppose any tendency of the system to get out of control. This tendency greatly simplifies the functions of the operators and automatic controls.

4.2. Normal Operation

To simplify operation of the entire plant and to make the adaptation of automation as simple as possible, the steam plant heat balance, as shown in Figure 4-1, was adapted. The flow cycle has been simplified even to the extent of eliminating high-pressure feed heaters.

The power plant can easily adapt itself to changes in power levels. The once-through steam generator, which is unique to shipboard power plants, is fitted with an automatic control that is basically similar to the three-element feedwater controls now employed in shipboard power plants. The only difference is that one of the inputs is steam pressure rather than steam generator water level. Control studies have established that this control will accommodate the fast power changes required in merchant marine service. The ability to manually control the steam generator has been retained.

The nuclear reactor plant is inherently self-regulating and tends to follow changes imposed by the steam plant. Owing to the thermal

capacity of the system, the primary system can meet any power change demanded by the system. In practice, the plant will be controlled by the turbine throttle. The rest of the plant will adjust itself to meet the demands imposed by the throttle. This power plant will require close control over primary water and feedwater quality. Experience with other reactors and power plants on board has shown that such water quality control is feasible and practical.

4.3. Reactor Plant Warmup and Cooldown

The reactor vessel is filled with high purity water from the secondary water storage; the vessel is vented at the pumps and nozzles to bleed out air. Hydrogen is added (Section 5.4.2) as the water level is lowered in the reactor vessel to the hot water level. When the pressure in the vessel reaches the net positive suction head of the primary pumps, the pumps are turned on. The hydrogen system is shut off and disconnected after the predetermined amount has been added.

The system is heated to a temperature corresponding to zero power by nuclear heat. As the water expands it is bled off to storage. To cool the system, the decay heat removal system functions as described in Section 5.3. Water must be added to the reactor vessel during cooldown to maintain pump suction and to assure coverage of the core.

4.4. Decay Heat Removal

The power plant will be provided with a decay heat removal system to remove heat generated after the reactor is shut down. The system shown in Figure 5-4 will also be used for emergency cooling when all electrical power supplies fail. Decay heat is normally removed by continued operation of the steam generator and primary pump(s), the heat being dissipated in the condenser. For emergency decay heat removal, the system takes advantage of the steam generator being a cooling coil. The secondary system becomes an intermediate system under emergency conditions. Steam or hot water flows through the emergency cooler when the turbine throttle is closed and is condensed and cooled by sea water before entering the deaerating feedwater heater. The deaerating feedwater cooler is used because it normally contains a large volume of

water. The cooled water is pumped back to the steam generator by the emergency cooling pump to complete the cycle.

For normal decay heat removal when the plant is shut down and electrical power is available, one or more pumps will circulate the primary fluid and the secondary system will remove the heat using the main condenser as the heat sink. Power supplied to the secondary equipment will permit the completion of the cycle. Having no power available to the primary pumps and emergency power available to the emergency cooler pumps, heat will be removed by the steam generator by natural convection in the uppermost water and condensation plus some natural circulation in the steam space. To aid natural circulation, ports are provided between the upper water and the shell side of the steam generator.

Emergency cooling engine-driven pumps to be connected in parallel with the motor driven pumps are provided to ensure coolant circulation if no power is available. The salt water pump takes its suction from an overboard line.

4.5. Effect of Ship's Attitude and Motion

The power plant has been designed to meet the conditions usually imposed by the Maritime Administration with regard to pitch, roll, and heaving. None of these motions will have any harmful effects on the reactor core or on heat removal from the primary system. Having a gas overpressure, the water in the core is subcooled, which gives less voids in the core than if the water was at saturated pressure. The ship's extreme motion in rough seas will have little or no effect on the reactor stability or change its reactivity by gravity variations on the voids.

4.6. Refueling

The refueling time of a nuclear marine plant extends from the time the power is shut off to the time that power can be generated with a new core. This refueling cycle includes cooling the reactor plant down, opening the containment vessel, disconnecting any instrumentation or equipment that must be removed before access to the reactor vessel can be accomplished, unbolting of the reactor vessel head, removing any internals that are in the way of the removal of the fuel

elements, erecting any temporary shielding, locating any positioning devices used in the fuel handling procedure, and the actual fuel handling (unloading and loading of the core). After the new core is installed, the reverse procedure must be followed (disassembly of the fuel handling mechanism, temporary shielding, installation of any internals, assembly of the pressure vessel head, assembly and connection of all equipment that had been previously disassembled, and installation of the containment cover. At this point, all equipment that was disconnected or reconnected during the refueling cycle must undergo a thorough testing to ensure that each reconnected piece of equipment is functioning as it should. Subsequent to the equipment check, the containment tightness must be established. After the precritical testing, the nuclear critical testing can begin; after its completion, the reactor vessel can then be warmed up to its operating temperatures. At this point, the refueling cycle is complete and the reactor is ready to go on the line.

For most reactors to date, the long sequence of events occurring during a refueling cycle has extended from weeks to months. A major purpose of the CNSG II design has been to eliminate events or reduce the time spent on them by changes in the major components. A reduction in the refueling cycle has been accomplished in the following ways:

1. The large bolted closures usually associated with pressurized water reactors have been eliminated, resulting in the reduction of the reactor temperature cooldown time.
2. Except for a few instruments, disassembly of equipment prior to opening the reactor vessel has been eliminated. This has been accomplished mainly by locating the control rod drives in the lower half of the reactor vessel and not in the top head.
3. The refueling cover of the CNSG II is relatively small and has relatively small bolts, which make the removal operation simpler and faster.
4. There are no internal structures above the fuel elements to be removed prior to refueling. This has been accomplished primarily by utilizing a bottom holddown latch for each of the fuel elements in place of the top holddown grid usually associated with pressurized water reactors.

5. The precritical testing of equipment that has been disassembled and reassembled during the refueling process has been greatly reduced. Other than the main pressure systems, precritical testing involves only a few instruments that have to be checked out prior to criticality.

An analysis of the refueling cycle of the CNSG II discloses that as a consequence of the design improvements made in this reactor, the refueling cycle has been reduced to an estimated 10 working days or less.

4.7. Maintenance

One of the major objectives of this study was to design a reactor having the major components arranged to allow easy access to any component requiring maintenance. The primary pumps, the control rod drives, and the steam generator are readily accessible for maintenance.

The primary pumps located in the top head of the pressure vessel may easily be removed for maintenance or replacement after access is gained to the containment. After disconnecting the electrical leads the pump may be unbolted, lifted directly off of the reactor vessel head, and removed to the maintenance area.

Provision has been made to maintain or remove any part of the rotating elements of the control rod drive system without having to drain the water out of the reactor vessel or the containment shell. This is accomplished by canning the control rod drives in the containment and providing a temporary sealing device for removing the pinion gear and shaft. The operation will be accomplished by personnel standing outside of the containment shell.

The steam generator, consisting of approximately 844 separate helical coils, is anticipated to be essentially trouble-free during its entire lifetime. Unlike fossil-fuel-fired generators, the steam generator in a pressurized water reactor environment can never experience overheating of any tube, and since Inconel tubing is used in this steam generator, corrosion problems are practically nonexistent. The possibility that one or more of the 844 tubes may develop a leak due to faulty erection or some material problem cannot be overlooked, and for this reason the CNSG II design affords ready access to the tube sheets for tube plugging if a leak does develop in a tube. As can be seen in Figure 1-1, both the inlet and outlet secondary steam generator headers and tube sheets

are located in the top head of the reactor where the tube sheet is accessible after removing a small portion of the flange-connected piping. The inlet and outlet secondary steam generator piping is arranged to allow for thermal and pressure expansion without the use of expansion joints.

4.8. Responses to Accidents

Owing to the self-limiting characteristics of the power plant, all anticipated accidents due to failure of equipment can occur without damage to other parts of the plant. Complete failure of all normal electrical power will cause the reactor to scram and the primary circulating pumps to shut down. In this case, the decay heat removal system will automatically come into operation preventing any damage to the reactor plant. Control rod withdrawal accidents, whether caused by improper operation or by a failure of the control system, will be limited by inherent characteristics of the reactor and by a safety system that will prevent damage to the reactor plant.

Simultaneous failure of all primary circulating pumps will result in a reactor scram, and no plant damage will occur during the pump system coastdown.

Figure 4-1. Heat Balance and Flow Diagram

Legend

Superheated Steam ——— Desuperheated Steam ———
 Low Pressure and Bleed Steam ——— Feed and Condensate ———
 Drain - - - - - Gland Leakoff and Vent - - - - -
 Stop V ; Check V ; Orifice ; Control V ; Back Pressure V
 PG = psig; PA = psia; Q = lb/hr flow (flows are total per ship); H = Btu/lb;
 F = temp in Fahrenheit

Data

Steam and Feed Conditions: Superheater Outlet, 397 psig and 523 F
 Main Turbine Throttle, 385 psig and 518 F
 Main Condenser Vacuum, 28.5" Hg at 75 F inj.
 Feedwater Temp. to Boiler, 251 F
 Calculated Heat Rate: 9522 Btu/shp hr, based on main turbine non-extr steam rate of 7.98 lb/shp hr
 Distilling Plant Load: 13,600 gpd
 Turbo-Generator Load: 1455 kw
 Injection: Pump

5. DESIGN

5.1. General Description

The CNSG II shown in Figures 1-1 and 5-1 consists of a pressurized water reactor, a steam generator, and a pressurizer combined in a single pressure vessel. These components are arranged within the pressure vessel in such a way that the nuclear core in the center of the vessel is completely surrounded by the steam generator tubes, which are in turn arranged in a series of helical coils. The section of the pressure vessel above the water line is the pressurizer portion of the reactor system.

The primary coolant circuit and the secondary boiler are also contained in the same pressure vessel. The primary coolant flow path is indicated by arrows in Figure 5-1. The primary coolant is heated in the core area in the center of the vessel; it then rises through baffles into the outer annulus of the pump stalk and up to the impeller of the pump, which forces the water down through the central section of the pump stalk and into the torus-shaped steam generator space immediately above the helical-coiled steam generator. The flow proceeds downward through the steam generator and back into the core area through openings in the core support cylinder to complete its cycle.

The steam generator is part of the secondary system, which begins outside of the reactor when feedwater enters from a header. The feedwater enters the steam generator, through two separate tube sheets and then flows through the vertical tubes, which convey the water down the side of the vessel wall to the bottom of the steam generator. From here, the tubes spiral upward. The economizer section of the steam generator extends from the point where the downcomer tubes begin transferring heat to the point where nucleate boiling begins. The secondary fluid rising in the spiraling coil passes through the steam generation section to the superheater. Four steam-outlet headers

collect the superheated steam (400 psig and 523 F). This steam, piped out of the containment vessel in four steam lines, drives the turbines. The primary system is designed for a ship's roll of 30 degrees. At 30-degree list, no joints between the pump suction plenum and suction pipe are exposed. Maximum assurance applies since the pumps are located less than 45 degrees from the fore and aft centerline of the ship. The inlet holes in the pump inlet plenum are designed so that a roll of 45 degrees will not uncover them. There are no holes located 90 degrees from the fore and aft centerline.

5.2. Containment

The reactor plant containment consists of a water-filled vessel approximately 22 feet in diameter and 37 feet high. The containment is divided into two areas; the central area approximately 12 feet 7 inches in diameter extends the full length of the containment and houses the reactor pressure vessel. This is the dry section of the containment. The annular section between the central diameter portion and the outer shell of the containment is partially filled with water. The water in the outer section of the containment serves as a vapor suppressor and as a shield water tank for biological shielding. The proportion of water to void volume in the outer portion of the containment was established on the basis that all of the steam released from a maximum credible accident in the reactor vessel would be condensed in this water without raising the pressure above the design pressure of the containment, approximately 100 psi. Pipes or ducts convey the released vapor from the dry portion of the containment to the water. No pipes are located on the sides so that they do not become uncovered during a ship's roll. If the ship is laying on its side at 90 degrees, the suppression ducts are still submerged. Check valves or covers prevent a back flow into the dry containment.

5.3. Reactor Vessel Arrangement

5.3.1. General Description

The design and fabrication techniques for the CNSG II reactor pressure vessel are well known and have been proved in service. The reactor core, steam generator, and circulating pumps are contained within the pressure vessel, which is largely filled with water to

transport heat and moderate neutrons. The space at the top of the vessel provides a surge volume to maintain a constant pressure when changes in water volume occur.

The core, centrally located in the reactor vessel, consists of 16 fuel elements assembled from Zircaloy tubes containing low-enriched UO_2 . The steam generator, which is located concentrically around the core, has inlets and outlets at the top of the vessel so that tubes can be easily plugged if necessary. Vertical positioning of the steam generator relative to the core was selected so that N^{16} activity will be low in the steam to the propulsion turbine.

Specifically, the reactor pressure vessel is designed to be fabricated from carbon steel that has been clad on the inside surfaces with stainless steel. The top head of the reactor vessel is penetrated by four primary pump nozzles, two secondary system feed nozzles, four secondary steam outlet nozzles, and a relatively small opening for refueling (approximately 50-inch ID). The refueling port closure is a simple flanged and bolted design that is sufficiently small to assure a reliable seal with conventional flexitallic gasket or metallic O-rings. The reactor vessel is supported from the bottom by a cylindrical skirt, which extends downward to a ring flange that is bolted to the inner containment bottom head.

The most significant feature of this reactor vessel is the elimination of the large bolted closure flanges usually found in pressurized water reactor vessels of this diameter. The use of the small diameter refueling port having small closure flanges rather than the usual full-diameter opening head having large closure flanges results in a savings in weight and space as well as a reduction in the cooling and heating time, which is a significant factor in judging the effectiveness of the design.

5.3.2. Reactor Core

5.3.2.1. Core Physical Design

Basically, the core consists of 16 fuel elements; 12 have a square cross section and 4 have a triangular cross section. Each of the square cross-sectional fuel elements has incorporated in it four T-shaped channels through which control rods pass. The

fuel elements are designed to provide a combination of simple and economic fabrication along with structural ability to withstand handling and operational acceleration loads. Essentially, the fuel elements consist of Zircaloy-clad fuel rods held together in a uniform pitch lattice by grid spacers located at intervals along the length of the element. The grid spacers, which are brazed assemblies of stainless steel, are connected structurally by means of axially disposed tie-rods. These tie-rods also connect to the upper and lower adapters, thus providing the necessary structure for fuel handling location and operational rods. Each of the square fuel elements contain approximately 470 to 490 fuel rods.

The T-shaped control rod configuration was selected for several reasons. From a structural standpoint this arrangement permits a much sturdier fuel element configuration, since in this design all portions of the element grid spacers can be connected through the center of the element. In addition, this configuration offers a significantly improved nuclear effectiveness as compared with the more conventional centrally located cruciform rod. Each group of four rods is connected together at the lower end by means of a coupling that also connects to the control rod drive line.

The fuel elements are located within an octagonal-shaped core container. This container, which is fabricated from stainless-steel castings, is sized on the interior contour to provide the necessary control of coolant flow along the outer periphery of the core. The container not only serves as a flow control baffle but also as a main support member for the core being supported by means of a flange from the central permanent structure within the vessel. At the lower end of the core container is a grid plate that includes accurately located receptacle holes which receive the lower ends of the fuel elements. In addition, T-shaped slots through which the control rods penetrate are machined into this grid plate. Each fuel element is latched to the grid plate through its lower adapter by means of a holddown latch. This holddown latch having the ability to restrain the fuel element from upward movement due to any anticipated condition of hydraulic forces, ship's attitude, and accelerative forces is designed as a highly reliable and simply actuated mechanism. The latch is released by a simple

vertical motion of an actuating shaft passing upward through the center of the fuel element. At the upper end of this shaft is a handling knob that can be engaged for fueling operations by a simple, long-handled tool. This latch is of a type developed and tested for other reactor applications by B&W. It has a self-energizing characteristic that prevents inadvertent release of the element unless the actuating rod is pulled upward.

5.3.2.2. Core Nuclear Design

The two means of reactivity control in the CNSG are control rods and burnable poison. There are 12 control rods, as shown in Figure 5-2. The burnable poison, which consists of ZrB_2 dispersed in ZrO_2 , is contained in rods identical externally to fuel rods. These poison rods (616) are dispersed throughout the core. Burning out the poison in these rods partially offsets the reduction in reactivity due to burnout of uranium and to the accumulation of fission products. Burnout of the poison maintains the reactivity more nearly constant, thus reducing the number of control rods required and permitting longer core life.

To provide an even distribution of power, four enrichment zones are provided. The average enrichment at the beginning of core life is about 5.3%. For initial analysis, it has been assumed that the use of these four enrichments gives a peak-to-average power of 3.0. This peak-to-average power is a conservative assumption based on analysis of a similar core.

5.3.2.3. Core Thermal Design

Preliminary thermal design of the CNSG II was based on the requirement that the core not be damaged by operation of 130% full power. Damage was defined as central melting of the fuel or clad burnout (departure from nucleate boiling or DNB). Because the overpower condition of 130% might be reached by a rapid transit from low power, it was further required that the overpower condition be satisfied using the higher core inlet temperature conditions corresponding to normal operation at 25% power. These requirements may be modified as detailed design proceeds. Additional requirements will be made as additional operational conditions (flow coastdown, ramp

additions of reactivity, and load changes) are investigated. At the present time, thermal design must be sufficiently descriptive to show that the general characteristics of the core fit the plant design.

At 100% power the central fuel temperature is 4250 F (manufacturing tolerance factors included) at the core hot spot. The central temperature at the same location for 130% power is 4850 F, which is still below the melting temperature of 5000 F. In reality overpower conditions only result when a transient and a safety action occurs before the temperature is stabilized to the above maximums. Hence, the temperatures given, which were based on steady state operation, are somewhat pessimistic.

The heat transfer design limits for the core were based on boiling burnout curves presented in APED-3892.² The thermal design analysis was based on the heat released in the hottest channel in the core. This heat release is approximately 1.9 times that of the average channel and is due to the radial and local peak-to-average power factors, plus a hot channel factor based on manufacturing tolerances.

The hot channel will be located near a control rod, since the local peaking factor is based on improved neutron moderation in the water gap between the control rod and the nearest fuel pin. The presence of this water gap also induces additional water flow through the hot channel, which improves heat transfer in the hot channel. This effect, while substantial, could not be evaluated in the preliminary analysis. Therefore, it was ignored, thus providing a degree of conservatism in the core design.

Based on the analysis indicated above, the peak heat flux is 459,000 Btu/ft²-hr at normal power and 598,000 Btu/ft²-hr at 130% power. Figure 5-3 shows the design limit heat flux curves (ϕ_{DL}) and the surface heat flux curves (ϕ_S) for 150% power plotted against exit quality for the hot channel. It is apparent that even at 150% power there is ample margin to burnout for the condition corresponding to a step increase in reactivity from a steady state power level of 25%. At this stage of design some conservatism has been purposely included to cover coastdown accidents, various rod positions, and different modes of operation. More detailed design will probably show that these margins are unnecessarily high.

5.3.3. Steam Generator

The CNSG II steam generator is of the forced-circulation, once-through type that is widely used in fossil-fueled stationary power plants. It offers the advantage of not requiring large, bulky steam separation equipment that would make the reactor vessel for this particular design concept considerably larger. Since the steam leaving the coils is superheated, the generator ensures that the steam is dry. The steam generator is a wound coil of continuous tubes. This design allows more square feet of heating surface per cubic feet of steam generator space than a more conventional design. The tubes are made of Inconel, which have thinner walls, better heat transfer, and lighter weight than stainless steel. This generator permits the steam turbine to operate at constant steam pressure, permitting the steam system to have a lower design pressure. Studies have been made to determine the lowest steam pressure for the most desirable steam temperature. Superheated steam may be produced with the loss of one primary pump with only a loss of about 30 degrees of superheat.

The once-through steam generator is located within the reactor vessel in the space between the core support cylinder and the reactor vessel wall. It is divided into two sections; each has a feedwater inlet and two steam outlets. Each section may operate independently. It is possible to operate with one section if leaks or other difficulties make it desirable.

Each of the feedwater lines connect to a tube sheet as they enter the top portion of the reactor. The tubes pass down through the steam void and the primary water in the outer annulus to the bottom of the tube bank. At this point the tubes start to spiral upward around the core support cylinder to the top of the core, emerge from the primary water, and again pass through the steam zone. At the top of the reactor the tubes for each steam generator section connect the two tube sheets as they pass through the vessel head. Four steam outlets are provided to produce a more compact arrangement. Steam leaving the tube sheets is channeled into four steam lines routed outside the containment. The tube sheets are accessible from outside of the reactor vessel for plugging operations. Detailed data on the steam generators are listed in the Appendix to this report.

5.3.4. Primary Pumps

The primary system contains four primary pumps located on the reactor vessel top head and spaced nearly 90 degrees apart. The number of pumps selected for this study was based on the space available in this particular arrangement. Pumps located in the top head can easily be maintained. The suction of all four pumps are taken from a torus-shaped plenum through the outer pipe of two concentric pipes. When the primary water reaches the pump, it makes a 180-degree turn inside the pump casing and then is forced down by the propeller-type canned motor pump. The water is discharged down the central duct to another torus-shaped plenum before being discharged to the steam generator.

Each pump is driven by a canned electric motor cooled by intermediate cooling water. A total developed head of 28 feet is produced at a capacity of 18×10^6 lb/hr. During cold startup, the required net positive suction head for the pump is supplied by pressurizing the reactor vessel with hydrogen. The pressurization fills the pump and pipes with water, and by venting the pump, the prime is complete.

5.3.5. Control Rod Drives

Twelve control rod drive mechanisms are located below the reactor core and enclosed within the containment vessel. The configuration of the drive mechanism is similar to the rack and pinion drive mechanism being developed for the Advanced Test Reactor (ATR), located at Idaho Falls, Idaho. The upper end of the rack component of the drive mechanism is connected to the lower end of the vertical control rod element, providing for withdrawal of the rods in an upward (out of the top of the core) direction and insertion (scram) in a downward direction.

The canned drive motor mechanism and pinion shafting is mounted in a horizontal position and made accessible for maintenance purposes through watertight nozzle closures located in the side of the containment vessel. Removal and replacement of these components without draining the reactor vessel is done by utilizing a service tool and procedure similar to that designed for the ATR.

If power failure or any other malfunction should occur, the control rod drive system will fail safely by gravity scrambling the reactor. Fast insertion of the control rods at the same speed as the normal rod withdrawal rate may be accomplished with the ship in a 90-degree position. Positive interlocks hold the rods in the fully inserted position after a scram, regardless of the ship's position.

The scram delay time from the receipt of the scram signal to two-thirds completion of rod travel does not exceed 2 seconds. The design of the mechanism and scram circuit enables the drive mechanism to withdraw, insert, or scram the rods individually, simultaneously, or in groups. A remote position indication system provides for positive indication of individual rod positions.

5.4. Auxiliary Systems

5.4.1. General

To consolidate and simplify the design of the CNSG II, some auxiliary systems superseded other systems and some were eliminated.

5.4.2. Primary Fill and Makeup System

Initial fill of the primary system is accomplished by using the makeup pump (see Figure 5-4) and a temporary line through the open containment and reactor vessel hatches. The water is demineralized secondary water and except for the oxygen content, the quality of the fill water and the makeup water are the same. An oxygen scavenger, such as hydrazine, is added to the initial fill to help remove any dissolved oxygen. To vent the primary system of air, the initial fill of cold water should completely fill the reactor vessel. Primary water is pumped or bled from the system to the hot water level. As the reactor is brought up to zero power and water heated, expanded water is bled out of the system through the makeup line.

Makeup required after the reactor is placed into operation is supplied from the secondary system makeup tanks. Reciprocating pump(s) located outside the containment vessel are the makeup pumps. Primary water level is indicated on the console so the operator may cut the makeup pumps on and off. The level of the primary water will be allowed to drop 1 foot before beginning the makeup. Makeup

will continue until the level is 1 foot above normal water level. Water conditions for initial fill and makeup are given in Section 5 of the Plant Information Sheets in the Appendix to this report.

5.4.3. Hydrogen Addition

Hydrogen is added at initial plant startup. The primary reason for using hydrogen is to provide an excess of hydrogen to suppress the radiolytic decomposition of the reactor water. An additional amount of hydrogen is added to provide an overpressure of 200 psia in the steam space so that local boiling at the surface of the fuel pins will be avoided. Hydrogen is also maintained in the reactor steam space to provide net positive suction head requirements for the primary pumps.

At operating conditions, about 460 cubic feet of hydrogen at standard temperature and pressure (STP) is located in the steam space while approximately 1620 cubic feet at STP is dissolved in the water. Hydrogen is added from cylinders located outside the containment vessel during heatup of the primary water. It is anticipated that supply cylinders will not be carried on board ship during normal cruises since the large quantity of hydrogen in the primary water will supply the steam space should hydrogen be lost through leakage.

5.4.4. Containment Cooling

The containment water and the stagnant air space between the top portion of the reactor vessel and the shielding are cooled by sea water flowing through coils located in these respective areas. Sea water is also used to cool intermediate water, which in turn, cools the four primary pumps and the twelve control rod drive motors. Two pumps that take suction from a sea chest pump sea water through the intermediate cooler or the containment cooling coils. Three containment water cooling coils, equally spaced around the reactor vessel, cool the water by natural circulation. A coil also cools the air space by natural circulation. Two pumps provide flow for the closed intermediate loop to cool the four primary pumps and the twelve control rod drive motors. Valves are located in the circuits for isolation and control purposes.

5.4.5. Primary System and Secondary System Relief Valve Venting

The primary system utilizes two methods to prevent over-pressurization of the reactor vessel: (1) a diaphragm-operated safety valve operating on a signal from pressure instrumentation and (2) two code self-actuated safety valves. The diaphragm valve has a remote operated valve located upstream for backup or isolation purposes should it leak. The code safety valves have a remote operated three-way valve located up stream for switching purposes in case a valve should leak. For the secondary system, a code self-actuated safety valve is located on each of the feedwater inlet lines to protect the steam generator if the system becomes full of water. All of the safety valves for both the primary system and the secondary system discharge directly to the containment water through sparging nozzles where the relief steam is quenched and collected.

5.4.6. Purification

The design of the CNSG II does not include a purification system. However, it is assumed that the customer will have a purification system at his dockside facility for cleaning up the primary water after power operation prior to entering the containment vessel for refueling or maintenance. For this reason blind flange connections are provided outside the containment for connecting to dockside and recirculating the primary water.

The philosophy in operating a CNSG is that no purification equipment is normally needed for operating the ship at sea, but is only needed in port to clean up the primary system prior to opening the reactor vessel for maintenance or refueling. Connections to and from the reactor are such that breaking off or rupture of a connection will not drain the reactor vessel below the top of the core and at the same time will release primary pressure below the containment water level.

5.4.7. Sampling

Since the quality of the primary water is not being controlled after the reactor vessel is initially filled, a normal sampling system is not needed. It is anticipated that a sampling system is

provided in association with the dockside purification facility so that cleanup of the primary water can be monitored prior to refueling or maintenance work.

5.4.8. Decay Heat Removal

No special equipment is required for the normal removal of decay heat following a shutdown. It is accomplished by the continued generation of steam in the once-through steam generators; the steam flows to the condensers where the decay heat is dissipated. If the primary water must be cooled down beyond the point where steam can no longer be generated, then feedwater flowing through the steam generator coils is used to remove the heat.

An emergency cooling system is provided when steam can no longer be dumped into the main condenser or when the main feedwater pump is inoperative. Steam or hot water is routed to the emergency cooler when the turbine throttle and bypass valve are closed. The secondary fluid is cooled and dumped into the deaerating feedwater heater. The emergency cooling pump takes suction on the large supply of cooled water stored in the deaerating feedwater heater and returns the feedwater to the steam generator. The emergency cooler dissipates heat by using sea water. Having no emergency power or no sea water suction available, internal combustion engine driven pumps are provided on either sides of the ship as backup emergency pumps.

5.4.9. Waste Handling and Venting

Because of the design philosophy of the CNSG II, a design for storing radioactive gaseous and liquid wastes is not required. When at sea, gases collected in the containment may be diluted with outside air and disposed of through the ship's stack. When in port and entry to the containment is required, the containment vessel can be purged and ventilated by the ventilating fan. Air from the containment is filtered for suspended matter and monitored before discharged up the stack. Gases formed in the reactor vessel may be disposed of through a line connected to the dockside disposal facility.

Any wastes resulting from reactor vessel leakage collects in the containment. Should this leakage raise the containment water level, a line will be provided above normal water level to bleed

excess water for disposal. Final removal of reactor and containment water will be done by dropping suction lines into the vessel and pumping out.

Vents are provided on each nozzle and pump on the reactor top head and on the top hatch. Water may be vented directly to containment during initial fill or gases may be vented to gaseous waste disposal after shutdown.

5.5. Shielding

Of primary interest in the CNSG II 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, placement of secondary heat exchangers, and 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 1.0 mrem/hr or less when operating at a reactor power level of 184 MW-t.

This dose rate was reduced to permit essentially unrestricted access to the outside of the containment when the reactor is operating at full power. (This was the same approach as used in the design of the NS Savannah.) Although conservative, this design is considered the most rational. The access requirement may be relaxed with an attendant reduction in shielding weight when more knowledge becomes available on the ship's arrangement. The containment may be placed in a reactor compartment, in which case, limited access can be guaranteed and the dose rate at the edge of the containment raised accordingly. Also, it may be possible to group equipment around the containment to afford some additional shielding. Finally, it may be practical to lay out the ship so that limited access is necessary only in the immediate vicinity of the containment.

The compact CNSG II shield (Figures 1-1 and 5-1) combines the functions of a primary and a biological shield. This shield consists of primary coolant water and thermal shielding inside the reactor vessel and lead and containment water outside the reactor vessel except on the top and bottom. The bottom of the core is shielded with 10 feet of

primary water and the 7.5 inches of steel that forms the pressure and containment vessels. In addition, double-bottom water tanks may be filled to afford more water shielding in this area. The top of the core is shielded by 4 feet of primary water in the reactor vessel and a 14-inch-thick removable lead shield above the top of the pressure vessel. Most of the shielding in this region is due to the primary gammas of the core rather than the secondary gammas or the N^{16} in the coolant pool. This shielding is adequate to reduce the dose rate to 1 mrem/hr or less at the top of the containment. A hollow cylindrical lead shield having a 4-inch wall thickness has been provided immediately below this top shield to maintain adequate protection in the region of the piping penetrations. The lead shielding exterior to the pressure vessel at the radial centerline of the core is sized at 8 inches and is backed by a 1-inch-thick steel support. At elevations above the core top and bottom, the shielding is reduced owing to increased water shielding and the slant thicknesses afforded by the lead. In the region near the bottom of the pressure vessel, the lead shield is 4 inches thick. Near the top of the vessel, however, 6 inches of lead thickness is required because of the termination of the containment water. Using this shield design, the gamma dose rate at the containment vessel exterior is 1 mrem/hr or less. The neutron dose rate contribution at the exterior of the containment is negligible in comparison with the gamma dose rates.

The thermal shield, primary water, secondary water, and steam generator metal are sufficient to reduce the fast neutron flux on the inside surface of the reactor vessel to a 20-year nvt of approximately 10^{17} . The arrangement and design of the steam generator with its high secondary water and steam velocities hold the N^{16} in the secondary system to such a level that a dose rate of 3 mrem/hr or less can be expected at the surface of a 6-inch Schedule 40 steam line upon its exit from the top of the reactor vessel. Depending on the steam transit times and the line accessibility in the final layout some additional local shielding may be required on these lines.

5.6. Instrumentation and Control

The majority of the instrumentation and control of the reactor and its auxiliary systems is entirely conventional with special emphasis placed on design in accordance with U. S. Coast Guard Codes and ABS Rules. The simplicity of the CNSG II and the limited number of auxiliary systems eliminate most of the instrumentation and control normally associated with a nuclear propulsion plant.

Special attention must be paid to the selection and placement of neutron detectors, because the steam generators are located between the reactor core and the more traditional locations for nuclear detectors outside the reactor vessel. The alternate locations considered for nuclear detectors are as follows:

1. Inside the core - Space here is very limited, but detectors for this service are commercially available. Reactor startup with these detectors, however, has not been demonstrated at the present time.

2. Between the core and the steam generator - Space limitations here are less stringent, but commercially acceptable detectors for this location must be developed.

3. Outside the reactor vessel - The distance from the core to the vessel wall in all directions will require the development of a suitable neutron window.

5.7. Erection

The simplicity and consolidation of the CNSG II nuclear plant design is directly reflected in the small portion of the plant erected in the shipyard. The following discussion relates to the areas where shipyard erection is affected; a brief explanation of how the plant is erected is also given. The number of auxiliary systems used in the CNSG II plant has been kept to a minimum. A comparison of the flow diagram shown in Figure 5-4 with a flow diagram of a nuclear plant, such as the NS Savannah, gives an indication of the relatively small number of auxiliary systems that have to be erected aboard ship.

One very significant item regarding erection of the CNSG II is that the reactor vessel containing the internals and the primary system

and the steam generator will be completely erected in the shop prior to shipment to the shipyard. Thus, excepting for the four primary pumps, steam piping and the control rod drives, the entire primary and secondary systems may be installed in the ship in one piece at one time, resulting in savings of cost and time over that experienced in the erection of the nuclear installation aboard the NS Savannah. Another benefit of a shop fabrication of the consolidated reactor and heat exchanger is that the original cleanliness established in the shop can be easily maintained since no further work on the reactor is done after leaving the shop.

A tentative erection sequence that might be followed is described below:

1. The bottom-head portion of the containment shell is received and installed in its cradle aboard ship and bolted securely in place.
2. The reactor vessel is installed in one piece on the support ring provided in the bottom-half of the containment shell.
3. The inner shell of the containment vessel to which the biological lead shielding has been attached is positioned around the reactor vessel and built up in sections to its full height.
4. The upper outer-shell section of the containment is lowered over the reactor vessel and inner containment shell to a position immediately above the previously installed lower head of the containment vessel.
5. The lower head of the containment vessel and the upper outer-shell section of the containment vessel are welded together. This welded joint will be in a section of the containment vessel having a thickness not greater than that for which stress relieving is exempt.

The preceding erection sequence outline neglects items, such as small pipes, feedwater inlet and steam outlet connections, and cooling coil connections, which in the erection would be entirely dependent upon specific shipyard practices and, therefore, are not covered in this report.

Figure 5-1. CNSG II Reactor and Containment Arrangement

Figure 5-2. Core Plan and Control Rod Positions

Figure 5-3. CNSG II Hot Channel Design
(66,000 shaft horsepower)

Figure 5-4. Preliminary Systems Design

LEGEND

	GATE VALVE NORMALLY OPEN
	GATE VALVE NORMALLY CLOSED
	GLOBE VALVE NORMALLY OPEN
	GLOBE VALVE NORMALLY THROTTLED
	CHECK VALVE
	3700 CHECK VALVE
	CYLINDER CONNECTING COIL
	FLOW METER
	PRESSURE REGULATOR
	ELECTRIC MOTOR OPERATOR
	SOLENOID MOTOR OPERATOR
	RELIEF VALVE
	3 WAY VALVE
	REMOTE OPERATED DIAPHRAGM VALVE
	FILTER
	FLANGE CONNECTION
	LOCKED CLOSED

INSTRUMENTATION
(SYMBOL CIRCLE)

- ☐ INSTRUMENT LOCALLY MOUNTED
- ☒ INSTRUMENT CONTROL ROOM MOUNTED

HI - HIGH ALARM
LO - LOW ALARM

FIRST LETTER

P = PRESSURE
F = FLOW
H₂ = HYDROGEN
L = LEVEL
I = AMPERES
T = TEMPERATURE
R = RADIATION

SECOND, THIRD & FOURTH

A - ALARM
I - INDICATOR
C - CONTROL

6. DESIGN HISTORY

6.1. Introduction

The CNSG II design is the result of a series of design efforts starting with that of the NS Savannah. At each stage of development, new information from research programs, from experience in the design, fabrication, and operation of the NS Savannah, and from similar design studies has been incorporated. Therefore, based on experience and experimental knowledge, great advancements in reactor design have been made since the NS Savannah was built.

This section will describe the immediate design history that has been incorporated into the CNSG design. The first three parts of Section 6 describe projects which were not a part of the CNSG, but which contributed important concepts to the CNSG design. The last four parts describe the history of the CNSG design.

6.2. Pressurized Water Reactor Design

The overall plant arrangement of this pressurized water reactor³ is shown in Figure 6-1. This design has the heat exchange equipment and the reactor vessel arranged in separate pressure vessels, as in the NS Savannah. The design differs from the Savannah since new designs for components were considered and several new concepts, as discussed below, were included in the design. Using this approach, it was possible to make a reactor plant that was much smaller, lighter, and cheaper than the NS Savannah.

A considerable portion of the effort expended in this design was used in determining what the temperature and pressure level of the primary and secondary systems should be. It was realized that the trend at that time for shore based nuclear power plants was toward ever increasing pressures and temperatures. At the same time, the trend in conventional shore based power plants lay in the same direction. However, for shipboard power plants the temperature and pressure levels have been stabilized for several years at levels considerably below those encountered on shore. B&W, therefore,

considered that it would be important to discover whether the same kind of trends could be found in shipboard reactor plants. Two sets of design conditions for the primary system were investigated. To be representative of shore based practices, one condition was set at a pressure of 2000 psi and a temperature of 636 F. To provide an adequate spread and to discover trends, a second condition was chosen at 812 psi and 520 F. This second condition was chosen at the lower end of the pressure-temperature range to provide as wide a spread as possible within the limited scope of the design study. Component sizes and costs and fuel costs were estimated for each condition. An optimum set of secondary steam conditions were also determined for each set of primary systems. As a result of these studies the following trends were discovered. These trends were considered to hold valid if the chosen parameters were in the range where the reactor would be a pressurized water reactor (the pressures were below the critical pressure of water).

1. Fuel costs remained constant. This was true within the accuracy limitations of the calculations performed. The higher pressure and temperature plant had better thermal efficiencies but this trend was offset by a decrease in the nuclear efficiency of the core. As the temperature of the core increased, it was necessary to increase the amount of U^{235} in the core, causing an increase in fuel costs. In addition, the high-temperature core must be made larger because heat transfer is poorer at the higher temperatures. The increased core size increased the cost of fabricating the core, causing another addition to the fuel costs. Finally, owing to the higher pressures, the cladding on the fuel elements had to be made thicker. This means that there was more parasitic material in the core, which again tended to increase the fuel costs. The combination of these three factors and the opposite trend in thermal efficiency counter-balanced the fuel costs.

2. Capital costs of power plant was higher for the high-pressure condition. This was primarily due to the thicker construction required to withstand the higher pressures. The difference in costs between the two conditions studied was approximately \$800,000.

3. It was considered that lower pressures would decrease the problems of designing, fabricating, and operating a nuclear power plant.

Based on these reasons, it was concluded that the reference design for the power plant should be the 812-psi condition. It was concluded that there was a range of conditions in which the costs would be relatively constant, and that so far as this optimization study had determined it would be satisfactory to design a future power plant anywhere within the range of from approximately 700 to 1000 psi.

This was the first B&W design to incorporate self-pressurization. It was found that self-pressurization would make the power plant inherently more simple by eliminating the pressurizer and its associated controls in piping. This had a direct effect upon capital costs and operating costs. In addition, a self-pressurized plant had the lowest operating pressure for a given temperature. This meant that the lightest and cheapest components could be used for a given temperature condition. The use of self-pressurization entailed a core penalty that seemed worthwhile at the time. This penalty was due to the poorer flow distribution that resulted from the voids in the core and to the effect of the voids in the core upon reactivity.

This power plant produced superheated steam through the use of a small superheater located parallel to the boiler in the primary system circuit. By the use of this parallel configuration the size of the superheater was considerably reduced. The superheated steam had a small thermodynamic advantage obtained at a small price for the cost of the superheater. It also will improve the quality of steam going to the turbines and reduce the maintenance and capital costs of the turbines.

6.3. Integral Boiling Reactor

The B&W developed Integral Boiling Reactor (IBR) design⁴, shown in Figure 6-2, was the first B&W design to have the entire primary system, excepting for the pressurizer, consolidated within one pressure vessel. The pressurizer was located in a separate vessel, similarly to the Savannah. Since this design used natural-circulation steam generators, steam drums had to be located on the outside near the primary system. This design also used natural circulation in the primary system.

6.4. Wet Containment

In November 1960 the AEC suggested the idea of a wet, that is flooded, containment vessel to B&W and requested the Company to study this concept.

The initial step in the investigations was a paper study⁵ of the problems inherent in wet containment. It was concluded that the concept of a wet containment would permit consolidation of several systems, because the containment could be used as a heat sink for components that required cooling and also as a tank for storage of waste water. It was also found that the water in the containment would serve very effectively as shielding. A potential advantage was found in that the water in the containment would suppress the pressure imposed on the containment by primary system rupture. The basic function of the containment is to protect personnel in case of such a rupture. This potential advantage would have to be explored by means of research. The major potential problem discovered by this study was the effect of the water in the containment upon the insulation around the primary system and upon the hot components of the primary system if the insulation failed. One possible way of protecting this insulation would be to install a watertight metal can around the entire primary system. This solution looked unattractive because it would require a complicated system having many joints that would have to be made watertight. It also apparently required a system for determining when and where leaks occurred. This system would in itself be complicated. It was, therefore, concluded that the major efforts should be devoted to discovering what would happen to the insulation if water came in contact with it, and to find an insulation that could function effectively regardless of whether it was wet or dry.

A research program was initiated at the Alliance Research Center to explore the pressure suppression aspect of wet containment. The initial phase of this research was reported in B&W Report BAW-1258.⁶ This research is still continuing. The first phase of the research established that rupture of a simulated primary system would be absorbed by water in the containment with a moderate pressure pulse. These conclusions were based on simulated ruptures up to 6 inches in diameter under rather restricted geometry. The characteristics of the ruptures were not determined completely, because only a limited amount of instrumentation was available on these tests. The second phase of the test now being carried out by Alliance Research Center will include more extensive instrumentation and, therefore, will show the effects of primary system ruptures in greater detail.

Research was also initiated on the possibility of using a wet thermal insulation. Two types of insulation were tested in these initial tests. One group consisted of conventional insulation such as Kaowool or magnesia surrounded by a jacket of epoxy resin or a similar material. The other group of insulations included stainless-steel wools packed to various densities. In some cases, epoxy resin covering was used; in some cases, it was not. During these tests no insulation was found that would be satisfactory for long-term operation. Therefore, a new set of experiments are being run at the Alliance Research Center to consider insulations subjected to less severe contact with water. This approach is made feasible by the simplified primary system due to the CNSG concept.

6.5. CNSG I, Phase I

In November 1961 B&W presented the IBR design to the AEC. The AEC decided that they would like to have design work performed on a reactor concept that would include the best features of the IBR and be adapted to merchant ship application. The new concept was to be called the CNSG concept. The CNSG was to incorporate the best, not only of the IBR, but also of the previous design studies and research and development that had been conducted by B&W.

The first phase of this new study was to determine what features would be incorporated into the CNSG. The previous design studies and the background developed in the steam generation field by B&W provided a broad spectrum of concepts that could be included in this concept. The major items included in this phase of work are listed as follows:

1. Forced- or natural-circulation primary system.
2. Forced- or natural-circulation secondary system.
3. Wet containment.
4. Self-pressurization.

In order to make the choices indicated in 1, 2, and 3 above, several conceptual designs incorporating various of these features were drawn up for the CNSG. Three of the conceptual designs drawn up in this phase are shown in Figures 6-3, 6-4, and 6-5. The reactor shown in Figure 6-3 utilizes natural circulation in both the primary and secondary systems. It was rejected because it was larger, heavier, and more expensive than the forced-circulation concepts. In addition, the design shown has only one section in its steam generator; if a tube failure occurred the entire unit

must be shut down until that tube is plugged. The reactor shown in Figure 6-4 utilizes a modified once-through steam generator that incorporates drums. The once-through steam generator is a platen configuration. It was found from control studies that the steam drums were unnecessary with this type reactor. It was also learned that the platen geometry for the once-through steam generator resulted in poor utilization within the reactor vessel. In addition, this design is still larger and more bulky than one using forced circulation in both the primary and secondary systems.

The configuration adapted as the reference design in this phase of the study is shown in Figure 6-5. It utilizes forced circulation in both the primary and secondary systems. It also uses a helical coil configuration for the once-through steam generator and it eliminates steam drums.

From studies of various kinds of containment made at the same time as the above studies were made, it was found that the wet containment when combined with the consolidated system promised new economies in the efficient use of material, weight, and space. This feature was, therefore, incorporated into the CNSG.

Also, the choice of self-pressurization was reviewed. This study influenced the choice of natural versus forced circulation for the primary system. It was concluded that the combination of natural circulation and self-pressurization introduced too many unknowns into the design. Under these circumstances the flow through the systems could not be predicted when the ship was subjected to pitch, roll, and heave. Since forced circulation made the unit smaller, it was concluded that the best combination would be made by eliminating the separate pressurizer vessel and its associated heaters and controls.

6.6. CNSG I, Phase II

The reference design at the initiation of Phase I is shown in Figure 6-5. Although this design presented a great advantage over previous reactors, it was considered that a detailed investigation of the design in the various parts of this system would make the entire reactor plant much more compact. This sort of work constituted Phase II of the CNSG I program. The reference design as it existed at the end of this phase is shown in Figure 6-6. By comparing Figures 6-5 and 6-6, it can be seen that the size of the reactor and its containment was reduced markedly.

In refining the design, attention was also given to accessibility and maintainability. It was found that the pumps and the lower tube sheets of the steam generator were located too close to the core for easy maintenance. Therefore, the pumps were shifted to stalks projecting from the bottom of the reactor vessel. These stalks extend above the level of the core so that pump maintenance can be carried out while the core is covered with water, which is necessary for removing decay heat. The steam generator inlet tube sheets were also shifted to this vessel.

To make the reactor vessel more compact, all of the steam generator connections were moved to the reactor vessel head. This permitted the use of a central chimney whose size was dictated by the space required by the control rod drives, and therefore, permitted making the reactor vessel about 30% smaller than it would have otherwise been. This also meant that the steam generator would have to be removed from the reactor vessel during refueling.

6.7. Burnable Poison Core Design

The CNSG I design utilized a soluble poison core (reactivity was controlled by a neutron poison, boric acid, dissolved in the primary water). This concept had many important advantages: it allowed a great deal of flexibility in the amount of reactivity which could be controlled; it permitted the use of a longer life core than would be possible based on control rods; and it improved the power distribution in the core, therefore, permitting the use of a smaller core and decreased fuel costs. However, as the work progressed it became obvious that the use of a soluble poison would require a large development program to determine its characteristics. Also, the use of soluble poison introduced important questions regarding the safety of the ship, particularly after sinking when it could be postulated that the boric acid solution could be replaced by sea water. The effects of the sea water would be an increase in reactivity and possibly the reactor going critical after it was on the bottom of the ocean. Therefore, B&W initiated a search for an alternate means of control that would maximize the advantages of soluble poison and not incorporate its objectionable features.

The concept selected was a lumped burnable poison concept. For this concept a neutron poison is placed in the core in the shape of pins externally identical to the fuel pins themselves. The poison pins contain

ZrB₂ dispersed in ZrO₂. By arranging the proper spacing and concentration of poison in the core, it is possible to keep the reactivity of the core essentially constant within their limits. This concept was explored under an AEC contract and reported in B&W Report SUP-37-1.⁷ This study established that an arrangement of poison pins that met the requirements and did not take up excessive space in the core could be developed. It appeared that the reactivity control would be accomplished by a concentration of pins that was equal to about 10% of the total number of pins in the core.

6.8. CNSG II Design

As was explained in the introduction to this report, the AEC requested B&W to re-examine the CNSG I design having a number of factors in mind. The two major factors involved were compactness and ease of refueling. Again a number of conceptual designs were drawn and a choice made. The reference design is shown in the early portion of this report. The design that came closest to being accepted is shown in Figure 6-7. It differed from the reference design in that it has a full-diameter closure flange, so that the steam generator could be replaced if considered necessary. The presence of this flange made the reactor vessel heavier, more expensive, and larger. Since the complete steam generator is considered a very reliable component and since means are provided for plugging any tubes that become faulty, it was concluded that this flange should be eliminated, resulting in the reference design.

The CNSG II design also incorporated the burnable poison core that was developed and described in SUP-37-1.⁷

To reduce steam generation in the core, which affects thermal performance, the amount of hydrogen gas in the primary system was increased. This provided an overpressure on the primary system, thus suppressing boiling.

The ease of refueling was increased by geometrical rearrangement. All obstructions over the core were eliminated by enlarging the chimney over the core to the full core diameter, and by placing the control rod drives below the core. The steam generator was lowered to take advantage of the space around the core and the space required below the core for the control rod drives. This position also minimized N¹⁶ activity in the secondary system, since the portion of the steam generator nearest the core contained low density steam.

Due to the gas overpressure, it was possible to move the pumps to the top head, where they could be more easily maintained. The control rod drives were rearranged so that they could be maintained by personnel outside the containment. Special maintenance tools will be required to keep water from draining out of the vessel (similar tools have been developed for the Advanced Test Reactor).

Figure 6-1. PWR Plant Arrangement (30,000 shaft horsepower)

Figure 6-2. IBR Plant Arrangement

SPACER BARS

BAFFLES

3030 I.O.D. RISER TUBES
ON $\frac{1}{4}$ TRIANGULAR SPACING

GENERAL ELEVATION SECTION

Figure 6-4. IBR General Arrangement

Figure 6-5. Dual Purpose Plant (82.5/400 MW-t)

LEGEND

- CARBON STEEL
- STAINLESS STEEL
- LEAD
- INSULATION

Figure 6-6. CNSG I Reactor and Containment Arrangement

Figure 6-7. Advanced CNSG Scheme I

APPENDIX
Plant Information Sheets

Plant Information Sheets

1. Heat Balance

Primary System Heat Generation

Reactor input, 183.7 MW-t, Btu/hr	627,766,000
Primary pumps input, Btu/hr	1,003,000
Total	628,769,000

Primary System Heat Losses

To containment air space, Btu/hr	35,000
To containment water space, Btu/hr	70,000
Total	105,000

Heat Transferred

At steam generator, Btu/hr	628,664,000

2. Primary System Design Information

Design Conditions

Number of pumps	4
Pressure, psia	1,400
Temperature, F	600

Operating Conditions at 183.6 MW-t

Pressure, steam dome, psia	1,200
Core outlet temperature, F	544.6
Core inlet temperature, F	517
Reactor flow $\times 10^6$, lb/hr	18
Flow per pump $\times 10^6$, lb/hr	4.5

Volumes at 183.6 MW-t

Primary water	--
Core, ft ³	59.3
Upper section, ft ³	351.0
Steam generator and annulus, ft ³	477.0
Lower section, ft ³	1,200.5

Sub-Total 2,087.8

Steam dome, ft³ 320.3

Total 2,408.1

Transit Times

Reactor core to steam generator, sec	1.13
Steam generator, sec	0.80
Steam generator to turn, sec	2.76
Turn to reactor core, sec	3.22
Reactor core, sec	0.55
Total	8.46

Pressure Drops

Core, psi	1.75
Steam generator, psi	3.0
Other, psi	1.95
Total	6.70

3. Secondary System Design Information

Design Conditions

Number of steam generators	1
Number of steam generator sections	2
External pressure (steam generator), psia	1,400
Internal pressure (outside reactor vessel), psia	1,400
Temperature, F	600

Operating Conditions at 183.7 MW-t

Pressure, (superheater outlet), psia	412
Temperature, (superheater outlet), F	523
Degrees superheat, F	75.5
Feedwater temperature, F	251
Feedwater pressure, economizer inlet, psia	453
Steam flow, lb/hr	605,300

4. Components

Steam Generator

General

Type	Once-through
Number required	1

Functional Data

Heat load, Btu/hr	628,664,000
Primary flow rate, lb/hr	18,000,000
Steam flow rate, lb/hr	605,300
Steam pressure, psia	412

Steam temperature, F	523
Amount of steam superheat, F	75.5
Feedwater temperature, F	251
Primary side pressure drop, psi	3.0
Tube side pressure drop, psia	41.0

Physical Data

Tube material	Inconel
Number of tubes	844
Tube OD, in.	3/4
Minimum wall thickness, in.	0.060
Average effective length, ft	111.1
Heat transfer surface, ft ²	18,400

Primary Pumps

Type	Canned motor, propeller
Number of pumps	4
Capacity at 544.6 F, gpm	12,000
Total developed head, ft	22

Relief Valves

Primary Valves

Pilot-Actuated, Non-Code Valve

Number	1
Capacity, lb/hr	33,200
Setpoint, psig	1,300
Size, in.	2
Operator	Pneumatic

Self-Actuated, Code Valve

Number	2
Capacity, lb/hr	33,200
Setpoint, psig	1,400
Orifice, in. ²	0.503
Inlet size, in.	1.5
Outlet size, in.	2.5

Steam Generator Self-Actuated Water Relief

Number	2
Setpoint, psig	1,400

These valves are to provide relief of water pressure when the isolation valves are closed and the steam generator is solid water.

Containment

Design Data

Pressure, psig	100
Temperature, F	280

Heat Loss to Containment During Operation

To air space, Btu/hr	35,000
To water space, Btu/hr	70,000
Total	105,000

Physical Data

Diameter of cylindrical section, ft	22
Height of overall system, ft	37

Volumes at 110 F

Water volume, ft ³	4,730
Air volume, ft ³	2,404
Total	7,134

Reactor Core

General Data

Power, MW	184 Normal
Coolant flow rate $\times 10^6$, lb/hr	18
Operating pressure, psig	1,300
Average outlet temperature, F	544.6

Core

Number of fuel elements	16
Number of standard elements	12
Number of corner-partial elements	4
Number of fuel pins	7,008
Number of poison pins	616
Active fuel length, in.	62
Fuel cladding material	Zircaloy-2
Fuel	UO ₂
Fuel rod diameter, in.	0.43
Fuel clad thickness, in.	0.017
Metal-to-water ratio (unit cell)	0.778
Fuel rod pitch, in.	0.576
Fuel diameter, in.	0.394
Type of fuel fabrication	Vibratory compacted
Total uranium (UO ₂), tonnes	8.466
Total uranium metal U, tonnes	7.462
Enrichment (average), %	5.3

Theoretical density of fuel, %	88
Core life (184 MW-t), full power days	1,036
Average fuel burnup, MWD/T	25,000
Maximum fuel burnup, MWD/T	48,000 (approx.)
Number of control rods	12
Absorber material	Boron-stainless steel
Fabrication	
Stainless steel clad, in.	1/16
Boron-stainless steel center, in.	1/8
Stroke length, in.	50
Absorber length/rod, in.	40
Absorber length, in.	50
Follower length, in.	62
Zirconium	
Fuel pin average heat rate, kw/foot	4.85
Fuel pin maximum heat rate, kw/foot	15.2
Average surface flux, Btu/hr-ft ²	147,000
Maximum surface flux, Btu/hr-ft	459,000
Hot channel surface flux (FOP 1.3) - Btu/hr-ft ²	598,000

5. Water Conditions

Primary Fluid (initial Fill)

Total solids, ppm	0.05
pH	Neutral
Oxygen (hydrazine controlled), ppm	0.01
Chlorides (as chloride ion), maximum, ppm	0.01

Makeup (Primary and Secondary)

Total solids, ppm	0.01
pH	Neutral
Oxygen, ppm	0.01
Chlorides (as chloride ion), maximum, ppm	0.01

Feedwater

Total dissolved solids, ppm exclusive of NH ₃	0.05 - 0.50
Suspended solids, ppm	0.01 - 0.04
Hardness, ppm	0
Organic, ppm	0
Free, caustic, ppm	0
Dissolved oxygen, maximum, ppm	(Preferably 0) 0.007
CO ₂ , ppm	(Preferably 0) Minimum
Chlorides (as chloride ions), maximum, ppm	0.01
Total silica (as SiO ₂), maximum, ppm	0.02
Total iron, maximum, ppm	0.01
Total copper, maximum, ppm	0.002

The pH value should be adjusted to obtain 0.01 ppm of iron maximum. This will normally require a pH value within the range of from 8.8 to 9.2 between the temperature range of 75-125 F.

6. Materials of Construction

Materials Contacting Primary Fluid

Reactor shell	Carbon steel clad with Type 304 stainless steel
Reactor heads	Carbon steel clad with Type 304 stainless steel
Internals	Type 304 stainless steel
Nozzles	Carbon steel clad with Type 304 stainless steel
Tubing	Inconel
Tube sheets	Carbon steel clad with Inconel
Piping	Type 304 stainless steel

Materials Contacting Secondary Fluid

Tubing	Inconel
Tube sheets	Carbon steel clad with Inconel
Piping	Carbon steel

REFERENCES

- ¹ Consolidated Nuclear Steam Generator - Final Report, The Babcock & Wilcox Company, BAW-1243, Lynchburg, Virginia, August 1962.
- ² Janssen, E. and Levy, S., Burnout Limit Curves for Boiling Water Reactors, APED-3892, April 14, 1962.
- ³ NS Savannah Upgrading Program - Advanced Pressurized Water Reactor Concepts, The Babcock & Wilcox Company, BAW-1219, Lynchburg, Virginia, March 1961.
- ⁴ MacMillan, J. H., Graham, D., and Kulynych, G., Natural Circulation Reactors for Marine Propulsion, The Babcock & Wilcox Company, BAW-192, Lynchburg, Virginia.
- ⁵ A Study of the Flooded Containment Concept and Its Application to the NS Savannah (Official Use Only), The Babcock & Wilcox Company, SUP-26, Lynchburg, Virginia, July 1961.
- ⁶ Luken, R. C. and Leeman, C. A., Savannah Upgrading Program - Vapor Suppression Test Program Report, The Babcock & Wilcox Company, BAW-1258, Lynchburg, Virginia, August 1962.
- ⁷ Consolidated Nuclear Steam Generator - Design Review, The Babcock & Wilcox Company, SUP-37-1, Lynchburg, Virginia, October 1962.

DISTRIBUTION

1. United States Atomic Energy Commission

Botsford, CW, Maritime Reactors Branch, Germantown, Md. (50)
Robb, JE, Washington 25, D. C.

2. The Babcock & Wilcox Company

Central Files (10)
Craven, JP/Barberton Library
Deuster, RW (2)
Dobel, HF
Gumprich, WC
Library, AED (2)
Littrell, LW (2)
Mumm, JF
Pletke, LR
Purdy, DC
Rock, HR

Sankovich, MF
Sawyer, GA
Schoessow, EE
Schomer, RT
Taber, AP/Alliance Library
Technical Report Group
Travis, CC
Wascher, RE
Webb, RA
Wilson, CW, New York Sales (20)