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Nuclear Materials & Propulsion Operation

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630A MARK V MARITIME NUCLEAR STEAM GENERATOR
STATUS REPORT

630A PROGRAM OPERATION

February 26, 1965

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"PRELIMINARY REPORT"

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ROUGH DRAFT

GEMP-342

630A MARK V MARITIME
NUCLEAR STEAM GENERATOR
STATUS REPORT

February 26, 1965

UNCLASSIFIED

FOREWORD

The primary purpose of this summary report is to set forth the current status of the 630A Mark V Nuclear Steam Generator, being developed by the Nuclear Materials and Propulsion Operation of the General Electric Company under a cooperative funding agreement between the U. S. Atomic Energy Commission and the General Electric Company.

The 630A Mark V Configuration incorporates several significant improvements resulting from the conceptual studies performed by the General Electric Company since 1961. Statements of the status and bibliographies of technical reports covering the progress of earlier 630A concepts are provided in Summary Reports GEMP-340* and GEMP-341**.

The most significant improvements incorporated into the 630A Mark V include:

1. The use of low enrichment fuel to provide substantial reduction in fuel cost.
2. The use of helium primary coolant which results in reduced operating temperatures for the same geometry, lower pumping power and smaller boiler size.
3. The use of enclosed gas-bearing circulators to eliminate shaft penetrations of the containment vessel.
4. The use of optimum steam conditions with about 400°F superheat (1500 psig, 1000°F) to obtain higher plant efficiency and reliability.

These improvements have been provided while maintaining the low capital cost of the nuclear steam generator.

Two concepts of the 630A Mark V Nuclear Steam Generator are described. The Mark V(A) uses a calandria-type reactor in which the fuel elements are contained within a reactor vessel suspended from the shield plug. The Mark V(B) uses a tube-type reactor with no calandria. The fuel elements are supported by the moderator tubes extending from the shield plug. Studies indicate that either of these concepts is feasible; however, the Mark V(B) appears most attractive and its features are emphasized in this report.

Following a description of the 630A Nuclear Steam Generator and system operation, succeeding sections of the report present in detail a description of the major components, accessories, waste handling and plant safety. The Appendix provides detailed specifications and operating conditions.

* "630A Mark III Maritime Nuclear Steam Generator Summary Report," GE-NMPO, GEMP-340, February 1965.

** "630A Mark IV Maritime Nuclear Steam Generator Summary Report," GE-NMPO, GEMP-341, February, 1965.

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1. DESCRIPTION OF THE 630A NUCLEAR STEAM GENERATOR AND SYSTEM OPERATION . . .

The 630A Nuclear Steam Generator consists of a reactor - shield-plug assembly, boiler, pressure vessel, side shield, two radial-flow gas bearing circulators, a containment vessel, controls, and auxiliary equipment with the major components arranged in a vertical assembly as shown in Figures 1.1 and 1.2. Figure 1.1 depicts the 630A Mark V(A) with the calandria-type reactor, and Figure 1.2 shows the 630A Mark V(B) with the tube-type reactor. The 630A Nuclear Steam Generator is designed to deliver superheated steam (1500 psig, 1000°F) for use with the most modern steam machinery; however, if required, reduced steam conditions can be delivered for a particular installation. The rating of the unit described in this document is 27,300 SHP; larger and smaller sizes are practical. Installation studies of such ships as the Mariner class (see Figures 1.3 and 1.4) showed that the assembly requires less space than the equivalent oil-fired boilers it would replace. These studies also showed that no major modifications would be required in the engine room, hull, or access openings. The existing stack uptake is sufficiently large to permit lowering the components through it during installation.

When installed in a ship, it is expected the nuclear steam generator will be located on the ship centerline with all of the nuclear components inboard of the side of the ship by 20 percent of the maximum beam and oriented so the circulators are parallel to the longitudinal axis of the ship. Approximately one foot of clearance beneath the lower head of the containment vessel is required to permit lowering this head for repair of the boiler. Consequently, the flat on which the 630A is mounted will be reinforced with girders to provide a suitable foundation for the nuclear steam generator. The upper portion of the 630A may be braced to the ship structure to add lateral restraint.

The once-through boiler is located in the lower part of the pressure vessel with the boiler header at the bottom. The boiler header also serves as the bottom closure for the pressure vessel. The reactor, shield plug, and reactor control elements are assembled as a unit called the reactor - shield-plug assembly. The calandria-type Mark V(A) reactor is suspended from the shield plug by six moderator-water exit pipes and the shim-rod guide tubes. The Mark V(B) reactor has fuel placed around the moderator tubes extending from the shield plug. The reflector is divided and is held in part by the shield plug and in part by the pressure vessel. The reactor - shield-plug assembly is located above the boiler with a combination shield and diffuser between the reactor and boiler. The diffuser reduces the activation of the boiler tubes and improves flow distribution of the reactor exit helium to the boiler region. A flange located about halfway up the shield plug closes the pressure vessel at the top. The radial-flow circulators are located within two equally-spaced nozzles that penetrate the pressure vessel near the bottom.

Shielding of the assembly is sufficient to permit access to the reactor compartment during full power operation. The containment vessel is shielded to assure acceptable radiation levels in the unlikely event of release of fission products into the containment vessel. The side shield, which surrounds the pressure vessel, is split into two annular sections by the containment vessel, so that the internal part of the side shield is located between the pressure vessel and the containment vessel and the external part surrounds the containment vessel.

Fig. 1.1 - 630A Nuclear Steam Generator Mark V(A) (221R442)

Fig. 1.1 (Cont'd) - 630A Nuclear Steam Generator Mark V(A) (221R442)

Fig. 1.2 - 630A Nuclear Steam Generator Mark V(B) (221R463)

INBOARD PROFILE

Overall Length: About 560 ft – 10 in.
Beam Molded: 76 ft – 0 in.
Displacement Molded: 18,610 tons

MAIN DECK PLAN

Fig. 1.3 - Profile and Plan View of the Mariner Class
showing the 630A Installation (221R112)

ELEVATION - Φ SHIP LOOKING TO PORT

Fig. 1.4 - 630A nuclear steam generator location in machinery space
of mariner class (221R462)

AUX. EQUIPMENT PACKAGE SIZE	
NO.	27,300 SHP
1	4'DIA. X 13'7"H
2	2'9" X 2'9" X 8'H
3	1'DIA. X 5'H
4	(1) TANK
5	2'9"DIA. X 3'6"L (2)
6	4'W X 6'L X 4'H
7	3'W X 4'L X 3'H
8	6'W X 8'L X 8'H
9	2'9" W X 1'6" L X 4'10" H
10	10'DIA. X 6'H (3)

PLAN-UPPER LEVEL 20'10 1/2"

Fig. 1.4 (Cont'd) - 630A nuclear steam generator location in machinery space of mariner class (221R462)

This results in the midsection region of the containment vessel being shielded by the part of the side shield outside the containment vessel. The upper section of the containment vessel is shielded with a layer of lead. Two nozzles located concentrically to the pressure vessel nozzles penetrate the containment vessel to maintain continuity of the containment around the gas-bearing gas circulators. The nozzles containing the circulators are closed by a flanged cylinder with a hemispherical head; thus, the circulators are completely enclosed within both the pressure and containment vessels. The entire moderator-water cooling system is also located within the upper head of the containment vessel.

The Mark V(B) configuration of the 630A Nuclear Steam Generator is 19 feet, 6 inches in diameter, 39 feet, 5 inches high, and weighs approximately 600 long tons. A summary of the component weights is shown in Table 1.1.

The primary loop working fluid (helium) is contained entirely within the pressure vessel and is circulated around the closed loop by the two gas-bearing circulators which take their suction from the plenum below the boiler. Helium discharged from the circulators is directed to the annular duct passages that surround the boiler and the reactor. After reaching the plenum formed by the lower surface of the shield plug and the upper face of the reactor, the helium flows into the entrances of the fuel elements and through the reactor. As the helium flows downward through the active core, it extracts heat from the fuel rods within the fuel elements by means of turbulent forced-flow convective heat transfer. The helium is then discharged into a plenum between the reactor and the boiler shield and diffuser. Instrumentation for measuring helium temperature is provided within this plenum. The helium then passes through the boiler shield and diffuser to the boiler. Heat is transferred in cross-counter flow from the helium to the steam and water in the boiler by first passing over the steam superheating surfaces which comprise about 22 percent of the total heat transfer area. It then passes successively over the evaporating and economizer surfaces which contain about 41 and 37 percent of the total heat transfer area, respectively. Finally, the helium discharged from the boiler, flows outward to the annular duct, and returns to the circulators, thus completing the cycle.

During startup or aftercooling operations, the primary coolant is circulated by use of the circulators receiving power from a 60-cycle Diesel electric generating unit. In the event of operation with one circulator, check valves are located at each circulator to prevent backflow through the circulator not in use.

1.1. PROPULSION SYSTEM

The propulsion equipment consists of the nuclear steam generator, propulsion turbines, auxiliary turbine generators, and other auxiliary equipment such as condensers, feedwater demineralizer, condensate pumps, boiler feed pumps, and desuperheaters. A typical arrangement is shown schematically in Figure 1.5. The main propulsion turbines and auxiliary equipment referred to as the secondary loop are conventional units in all respects. The steam cycle is a regenerative, non-reheat cycle utilizing five stages of feedwater heating.

Table 1.1

SUMMARY OF 630A MARK V(B) COMPONENT WEIGHTS

	<u>Steel & Misc.</u>	<u>Water</u>	<u>Lead</u>	<u>Total Wt. lbs.</u>	<u>Total Wt. Long Tons</u>
REACTOR SHIELD ASSEMBLY					
<u>Core-Plug Assembly</u>					
Shield plug	66,700	7,400	6,600	80,700	36.03
Reactor core and reflector	15,200	450		15,650	6.99
<u>Pressure Vessel Assembly</u>					
Pressure vessel shell	78,700			78,700	35.13
Thermal shields	91,800			91,800	40.98
Steam generator	76,500	900		77,400	34.55
Shielding, valves, structure	25,780			25,780	11.51
<u>Containment Vessel</u>					
Upper head	104,100		28,600	132,700	59.24
Main cylinder	167,300			167,300	74.69
Lower head	37,300			37,300	16.65
<u>Shielding</u>					
Inner side shield	20,700	30,700		51,400	22.95
Inner upper shield	6,100	9,400		15,500	6.92
Outer side shield	82,600	153,000	160,000	395,600	176.60
Outer upper shield	11,900	16,100		28,000	12.50
Circulator shields	9,000	7,000	2,600	18,600	8.30
<u>Moderator System</u>	14,000	6,000		20,000	8.93
<u>Gas Circulators</u>	8,000			8,000	3.57
<u>Containment Cooling & Misc. Systems</u>	900			900	.40
AUXILIARY EQUIPMENT					
Controls and instrumentation	6,700			6,700	2.99
Variable frequency generator (for circulator drives)	24,500			24,500	10.94
Helium storage and vacuum pump	5,000			5,000	2.23
Condensate demineralizer	10,400			10,400	4.64
Component cooling system	5,000	1,000		6,000	2.68
Waste gas system and filter	33,000			33,000	14.73
Waste water system	1,600			1,600	.71
Steam valves and piping	4,600			4,600	2.05
Moderator demineralizer and hydrogen system	2,300			<u>2,300</u>	<u>1.03</u>
TOTAL				1,339,400	597.94

Fig. 1.5 - Preliminary heat balance for 27,300 shp marine power plant with throttle steam conditions of 1015 psia pressure and 1000°F temperature

High-pressure, high-temperature steam is delivered to the propulsion turbine throttle from the nuclear steam generator. The propulsion turbine consists of a cross-compound, single-flow arrangement with six steam-extraction openings leading to the auxiliary equipment. Four extractions are taken from the high-pressure unit and two from the low-pressure unit.

The first steam extraction is taken from the first-stage shell of the propulsion turbine and is used to operate the auxiliary turbine generators. The second steam extraction is used to generate low-pressure steam for ship's service and to provide heat to the high-pressure, or fifth, closed feedwater heater. The third steam-extraction opening is used to bleed steam which is delivered to the fourth closed feedwater heater. The third feedwater heater receives heat from the fourth-extraction opening. The fifth extraction is for the deaerating heater. The sixth extraction from the main propulsion turbine is a low-pressure extraction used by the evaporator which provides fresh water for ship's service and make-up water for the steam power plant. Steam not extracted from the main propulsion turbine continues its expansion to the main condenser, condenses, and is pumped from the condenser hot well to be mixed with condensate from the auxiliary turbine-generator units. The condenser may be divided into two parts so that one section may be valved off for tube maintenance. The condensate next passes through the demineralizer and then to the first, or low-pressure, feedwater heater. Rather than using low-pressure extracted steam for its heat supply, this first heater, a shell-and-tube-type heat exchanger located within the containment vessel, receives its heat from the reactor-moderator water. A cycle performance gain is thereby achieved because moderator heat is not lost from the thermodynamic cycle of the power plant.

The temperature of the feedwater is elevated further as it is pumped through the remaining four feedwater heaters of this regenerative cycle. The motor-driven boiler feed pump elevates feedwater pressure and returns the feedwater to the economizer section of the steam generator. A second motor-driven boiler feed pump is available for startup, shutdown, and emergency operations.

The boiler is of once-through design, i.e., neither a water drum nor recirculating pump are required because feedwater flows continuously from the economizer to the evaporator and then to the superheater where it emerges as superheated steam.

A steam desuperheater is utilized to provide steam of reduced pressure and temperature directly to the auxiliary turbines during astern and reduced-load operations. Above about 70 percent load operation, the desuperheater provides steam to only the steam-jet air ejectors. A second desuperheater is used in the main steam bypass line to the main condenser during startup and also in the event of a required steam dump to the condenser if the load is suddenly dropped.

1.2 OPERATING PROCEDURES - SHIPBOARD

1.2.1 Startup

During startup, the power plant is operated by the auxiliary and standby equipment: the motor-driven boiler feed pump and standby power-generating units. The steam discharge is connected to the main condenser through a pressure-regulating valve and desuperheater. All steam turbine throttle valves are closed. The

boiler is filled with water to a predetermined level and then the feedwater supply is shut off. The system is charged with helium and pressurized for the proper operating pressure at the desired loop temperatures. The cooling helium is circulated by the two circulators. The moderator water is circulated through the reactor and the reactor is gradually brought up to about 10 percent of scheduled power. As steam is generated in the boiler, the steam pressure increases to a controlled level and steam is bled through the pressure-regulating valve and desuperheated to the main condenser. As the boiler water level decreases, and steam temperature increases to a scheduled value, the feedwater is set to supply feedwater at the rate required to maintain desired steam conditions. At this point, steam flow is delivered directly to the auxiliary turbine generator. The standby generators are then shut down.

1.2.2 Operation

With the reactor operating at 10 percent power, steam may be admitted to the propulsion turbines. Steam generation rate changes are obtained by operation of a power-demand lever which provides schedule demands for helium flow, feedwater flow and reactor power. Corresponding adjustment of the propulsion turbine throttle valves is required to regulate the propulsion turbine speed. Up to about 70 percent load, the auxiliary steam turbine continues to operate on desuperheated steam. To improve efficiency at greater loads, the desuperheated steam supply to the turbine is automatically cut off, and steam extracted from the main propulsion turbine is used to drive the auxiliaries. Thus, during operation at or near rated load, all steam generated is supplied to the propulsion turbine throttle, except for a small amount of desuperheated steam supplied to the steam-jet air ejectors. During maneuvering operations, a maximum time of 70 seconds is required for propulsion load changes of 30 to 100 percent or 100 to 30 percent.

Astern operation is accomplished in a conventional manner, i.e., the main propulsion turbine throttle is closed, desuperheated steam is automatically admitted to the auxiliary turbine and excess steam (desuperheated and reduced in pressure) is delivered to the main condenser. A guarding valve and an astern turbine throttle valve are then opened to admit steam to the astern turbine. Eighty percent normal ahead torque at 50 percent normal ahead rpm is obtained with design steam pressure and temperature.

1.2.3 Shutdown

The shutdown procedure is initiated by retarding the power-demand lever to the 10 percent position and the propulsion turbine throttle to the off position. Complete shutdown from this power level is then accomplished by:

1. Starting the auxiliary feedwater pump and shutting down the main feedwater pump.
2. Starting the standby generators and transferring all electrical loads to them.
3. Starting the secondary moderator cooling pump.
4. Inserting all reactor control elements.

5. Continuing operation of both circulators to accomplish aftercooling.
6. Reducing feedwater flow to minimum scheduled value.

1.3 NUCLEAR STEAM GENERATOR CONTROLS

The controls and instrumentation systems utilized in operations of the 630A Mark V are (1) the startup instrumentation, (2) the power range system, and (3) the safety system. The reactor startup instrumentation monitors reactor power between source level and 10 percent of full power and displays power level and rate of change of power to the operator who controls reactor power in this range by remotely adjusting the position of the control tubes in the reactor core. The power range system regulates reactor power, boiler feedwater flow rate and primary-loop helium flow rate to maintain demanded steam conditions of flow, temperature, and pressure. The safety system monitors nuclear steam generator parameters and, if any violate set limits, either provides a warning to the operator or acts directly to bring the parameters within limits.

1.3.1 Reactor Startup Range Instrumentation

Reactor power in the range between source level and approximately 10 percent of full power will be controlled manually by the operator as he remotely adjusts the positions of the shim tube drives. The condition of the reactor during the complete startup will be monitored and displayed for the operator by two different sets of instrumentation. Source range instrumentation will cover the range between source level and 5×10^{-6} percent full power, and intermediate range instrumentation will cover the range between 10^{-6} and 10 percent full power.

Source Range System - The source range system, shown in Figure 1.6, contains two identical channels each consisting of a fission chamber located in the shield plug near the reactor core, a pulse amplifier, a discriminator, a log count rate circuit, a log count rate amplifier, a log count rate meter, a period amplifier, a period meter, and a period trip circuit. Log count rate and period will be displayed for each channel for comparison by the operator. The operator manipulates the remotely-controlled shim actuators to obtain the desired log count rate level or rate of change. There will be a period warning if the measured period becomes less than a predetermined level (approximately 30 seconds). No other active safeties will be employed in the source range.

Interimmediate Range - The intermediate range system, shown in Figure 1.7, consists of two identical channels each made up of a compensated ion chamber, a log amplifier, a period computer, a log flux meter, a period meter, and safety trip circuits. Log flux and period will be displayed for each channel for comparison by the operator. As in the source range, the operator manipulates the remotely-controlled shim actuators to obtain desired log flux or rate of change.

An automatic period safety and a level safety are included in this range. A warning light will alert the operator if the reactor period becomes shorter than a set level. If the period is allowed to reach the next level of safety, a signal will call for automatic insertion of all tubes. Another function of the period safety is to serve as an active safety for

Fig. 1.6 - Startup instrumentation, source range preliminary block diagram

Fig. 1.7 - Startup instrumentation, intermediate range preliminary block diagram

the source range.

The level safety will also insert all shim tubes if the power measured by the intermediate range reaches a preset level with the power range demand still calling for minimum power (approximately 10 percent of full power). This safety will be disarmed when the operator moves the power demand lever to demand a power greater than the minimum demand.

The intermediate range will overlap the source range by about one-half a decade so that the operator can be sure that the intermediate range log flux meters are operating before using them for control.

1.3.2 Power Range System

The power range system, shown schematically in Figure 1.8, controls the nuclear steam generator in the range between 10 and 100 percent of full power. It is comprised of the reactor power range control system, the boiler feedwater control system and the gas circulator speed control system. The reactor control regulates reactor power to maintain constant helium temperature at the reactor exit; the feedwater control regulates feedwater flow rate to maintain demanded steam pressure; and the gas circulator control regulates circulator speed, and thus helium flow, to maintain steam temperature at the demanded value.

The three subsystems receive coordinated scheduled demands from a device called a power lever. It provides demands for reactor power level, feedwater flow rate and circulator speed. The magnitudes of the demands are functions of the power lever position.

The power lever is located adjacent to the propulsion turbine throttle control lever on the engine room control console. The positions of the propulsion turbine ahead and astern steam valves are controlled in response to commands from the throttle lever on the engine room or bridge control console. The engine room is the primary control center, but throttle control can be transferred to the bridge. The power lever and the throttle lever may be moved simultaneously for accomplishing load changes, or the power lever may be left in a position up to maximum power while the throttle lever is retarded to any position from full ahead to full astern. In the latter case, the excess steam is vented through a desuperheater to the main condenser.

The engine room control console will be designed to permit a single operator to monitor and control the complete propulsion system, including nuclear steam generator control, from a single station. It will be human-engineered for ease of operation and maximum plant safety.

Response of the nuclear steam generator to changes in load demand in the range between 30 and 100 percent will be at least one percent per second. Much faster response may be obtained by maintaining the nuclear steam generator power output high during maneuvering.

Reactor Power-Range Control - A block diagram of the reactor power-range control system is shown in Figure 1.8. The system's function is to regulate reactor power to maintain the temperature of the helium discharged from the reactor at the demanded level. This demanded level is changed

Fig. 1.8 - 630A Mark V controls block diagram

slowly during startup to limit thermal stress in structural components and is held at a constant 1200°F during normal power operation.

The system is essentially a flux control with a temperature trim. In operation, a helium flow signal either from the throttle or from a tachometer on the helium circulator will be fed into the flux error amplifier as a demand signal. If the resulting change in flux does not establish the controlled helium temperature, then the temperature trim loop will force a readjustment in flux level to reestablish the correct temperature.

The temperature trim loop employs proportional plus integral feedback in the error amplifier. The proportional portion assures good transient response as well as stable operation while the integral portion assures good reset action.

If a temperature error exists, the output of the temperature error amplifier trims the flux loop by forcing the regulating tubes to reposition until the temperature error is reduced to zero. These tubes are positioned in the core by two dynamic (proportional) actuators. Dynamic actuators are used to provide good accuracy and response over that attainable with slower shim tube actuators, although the reactor may be operated safely on shims alone. The dynamic actuators will be operated, within the velocity limits of the actuators, at a velocity proportional to flux error. Shim tube actuators will be controlled manually to keep the dynamic tubes approximately centered in the core during steady-state operation. The shim actuators will be in either of three states of operation; full insert rate, off, or full withdraw rate. They will be driven by synchronous motors, thus making their velocity proportional to line frequency and providing a method of "fixing" their reactivity rate.

The helium temperature used for control is sensed by multiple thermocouples arranged in parallel and located directly below the reactor core. Reactor power, used as feedback in the flux loop, is sensed by an ion chamber located in the side shield.

Feedwater Control System - In the once-through boiler used in the 630A, feedwater enters one end of the tubes and leaves the other end as superheated steam. Likewise, all of the helium which transfers heat from the reactor to the steam generator passes over the series arrangement of superheater, boiler and economizer sections. Since the temperature of the helium discharged from the reactor is held approximately constant, the steam conditions of pressure, temperature, and flow must be controlled by some combination of feedwater flow and primary loop helium flow regulation. The approximate feedwater flow and helium flow are scheduled as a function of load demand but accurate control of steam pressure and temperature requires some method of automatically adjusting these schedules.

The feedwater control system, shown schematically in Figure 1.8, schedules feedwater flow as a function of power lever position, compares this with a measured flow signal, and any difference or error is amplified and applied to the feedwater-flow control valve to regulate flow, thus forcing

the measured flow to equal the demanded flow. The steam pressure demand is compared with the measured pressure and the difference or error is used to adjust the feedwater schedule until the demanded steam pressure is obtained. A limiter is provided which stops feedwater flow if steam temperature drops below a minimum set value. An alternate method of control using steam temperature to trim feedwater flow was evaluated and found to be unstable at low flow or power conditions.

Primary-Loop Helium Flow Control - Since the reactor control regulates power to maintain the temperature of the helium discharged from the reactor at a constant level, the rate at which power is transferred from the reactor to the boiler is governed by the mass flow rate of helium circulating through the primary loop and the drop in helium temperature across the boiler. The helium flow rate is controlled by regulating the rotor speed of the helium circulator drive motors.

The helium flow control system, shown in block diagram form in Figure 1.8, schedules circulator motor speed as a function of power-lever position. Speed of the circulator is regulated by controlling the frequency of the driving power. The steam temperature demand is compared with the measured temperature and the difference or error is used to adjust the scheduled circulator speed until the steam temperature error is reduced to zero. This method of control has been evaluated using the once-through test boiler in conjunction with the described feedwater control. An alternate method of control using steam pressure to trim airflow was evaluated in an integrated control and found to be less stable at low flow and low power conditions.

1.3.3 Safety System

The safety system is designed to prevent damage resulting from a failure in the control system, operator negligence, and all foreseeable external perturbations. Five types of safety actions are provided:

1. Warning - To alert the operator to possible unsafe conditions.
2. Insert-all-shims - To insert automatically all shim negative reactivity at shim rate.
3. Scram - To insert automatically all safety negative reactivity at maximum rate with automatic followup insertion of all shim control elements.
4. Interlock - To enforce a specified sequence of events.
5. Override - To permit the operator to override the automatic control with the exception of safety responses.

Warning - Warning lights will indicate when certain power plant parameters exceed their normal maximum levels. This will warn the operator to be alert to the possibility of equipment malfunction. Depending on the circumstances, the operator may elect to reduce power or continue the operation. The lights will be automatically extinguished when the parameters return to their normal operating level. If they exceed still higher

preset limits, however, the reactor will be automatically scrambled.

Insert-all-Shims - Used in the intermediate range, the insert-all-shims action causes the opening of the shim-withdraw bus and closing of the shim-insert bus. This action, once initiated, cannot be reversed until the parameter that caused it has returned to limits. To withdraw the shim tubes again, the operator must manually close the shim-withdraw bus switches, which are of the spring-return type with a normally open position.

Scram - Scram will be initiated by removing power from the safety actuator latching solenoids which vents the pneumatic latch and permits mechanical springs to drive the poison into the core. Once initiated, scram must be completed. A reactor startup is required to bring the power plant back to operating power.

The warning, insert-all-shims, and scram parameters are listed in Appendix B.

Interlock - Three interlock safeties, which give added protection and backup to the reactor instrumentation, are the "all safeties latched", power demand, and a high-flux interlock. If all safety tubes are not withdrawn and latched, an insert-all-shims safety will be initiated which will prevent any shim tubes from being withdrawn. The power-demand safety insures that the automatic control system is available to "level-off" power at the bottom of the power range and thus reduce the possibility of a startup accident. Unless the power-demand control is placed in its minimum position, the reactor control will receive an insert-all-shims safety; thus preventing reactor startup. Only after the measured power reaches approximately 5 percent full power can the demand be increased above minimum. The high-flux interlock is employed to prevent a startup accident by inserting all shims when measured power exceeds some low-power level, approximately 10 megawatts, with the power demand calling for minimum power.

See Appendix B for the list of safety parameters and actions.

1.4 DESIGN POINT SELECTION

During the course of evaluating the 630A Nuclear Steam Generator concept a variety of cycle types, working fluids and operating conditions of pressure and temperature have been studied. An indirect cycle using helium as the primary coolant and a non-reheat steam propulsion plant is selected as a basis for the design point. Helium is an attractive coolant by virtue of its low pumping power requirement and attractive heat-transfer capability. Use of the gas reactor in the production of steam permits the use of nuclear power with modern steam propulsion equipment. Direct cycles using either steam, air, helium or other coolants appear to be feasible.

The recommended operating conditions for the 630A Mark V Nuclear Steam Generator at the design point of 27,300 SHP are shown in Table 1.2. The primary loop operating pressure at the circulator exit is 830 psia. This pressure was selected to minimize the circulator power requirement and to maintain acceptable

Zircaloy moderator tube operating temperatures below 450°F for the required tube thickness. The helium temperature at reactor exit is 1200°F and was selected to meet design criteria connected with boiler support, fuel element life and structural and boiler tube material requirements. The helium temperature at the boiler exit is 550°F, and was selected to minimize circulator power requirements and boiler size consistent with the optimum final feedwater temperature. The secondary loop consists of a non-reheat, regenerative cycle. The design point operating conditions for the propulsion plant are 1500 psig and 1000°F for a 27,300 SHP rating. Higher pressures may be desirable for significantly larger ratings. However, 1000°F throttle steam temperature is believed to be the optimum in marine propulsion at the present time for all ratings. Design back pressure is 1.5 in. Hg. absolute. The selected final feedwater temperature is 415°F, which is somewhat lower than the optimum considering only the steam plant. This was selected to minimize boiler area with an insignificant reduction in plant efficiency.

Table 1.2

DESIGN POINT CONDITIONS

Primary coolant	Helium
Rating, Shp	27,300
Circulator exit pressure, psia	830
Reactor exit temperature, °F	1200
Boiler exit temperature, °F	550
Turbine throttle pressure, psig	1500
Turbine throttle temperature, °F	1000
Condenser pressure, in. Hg. Abs.	1.5
Final feedwater temperature, °F	415

1.4.1 Cycle Performance

Preliminary overall power plant performance estimates are based on the stated design point conditions in Section 1.4, the NSG configuration shown in Figure 1.2, and the propulsion plant selection and arrangement shown in Figure 1.5.

The reactor is rated 60.4 MW_t for operation requiring 27,300 SHP plus ship and reactor auxiliary power and ship heating requirements. 4.2 MW of the total reactor power are delivered to the moderator. The preliminary specifications and heat balance for 27,300 SHP are shown in Figure 1.5. Overall plant efficiency is 33.7 percent, a net efficiency with no credit taken for the ship auxiliary heating and power loads.

Two gas-bearing circulators are used to circulate the primary coolant. They

are each rated at 250 BHP and are motor driven. The primary coolant, helium, is discharged from the two circulators at a pressure of 830 psia and a temperature of 553°F. The total flow rate is 66.2 lb/sec. In the reactor the helium is heated to an average exit temperature of 1200°F. The helium pressure drop is about 2.1 psi. Helium inlet and exit velocities are about 75 and 125 ft/sec, respectively. The maximum dynamic head is about 0.3 psi and the maximum Mach number is about 0.02.

Helium flows down through the boiler and is cooled from 1200°F to 550°F temperature while superheating, evaporating and preheating the secondary loop working fluid, water. The helium pressure loss is about 1.3 psi. The boiler total duty is 192.5×10^6 Btu/hr.

Ducting, reflector and circulator ducting pressure losses amount to about 2 psi and the total primary loop pressure loss is about 7 psi with allowance for margin and life effects, requiring a circulator pressure ratio of about 1.009. A summary of the most significant primary loop thermodynamic performance is shown in Table 1.3.

Table 1.3

<u>PRIMARY LOOP THERMODYNAMIC PERFORMANCE</u>	
<u>27,300 SHP</u>	<u>630A Mark V</u>
Reactor power, MW	60.4
Helium flow rate, lb/sec	66.2
Circulator exit pressure, psia	830
Circulator exit temperature, °F	553
Reactor exit temperature, °F	1200
Primary loop pressure loss, psi	7.2
Power to moderator, MW	4.2
Circulator total power, BHP	500

The secondary loop water flow rate at the boiler is 175,200 lb/hr and the final feedwater temperature resulting from five stages of feedwater heating is 415°F. Steam is delivered to the main propulsion turbines with operating conditions of 1500 psig, 1000°F. The plant auxiliary power is provided by a 1500 KW turbine generator and consists of 489 KW for the circulators, 442 KW for ship and reactor auxiliary power and 532 KW for the boiler feed pump drive motor. Main and auxiliary condenser back pressure is 1.5 in.Hg. absolute. The low pressure steam generator heat load is 0.1×10^6 Btu/hr and the ship's evaporator duty is 1.5×10^6 Btu/hr. The net overall plant efficiency is 33.7 percent. A summary of the secondary loop performance data is shown in Table 1.4.

Table 1.4

<u>SECONDARY LOOP THERMODYNAMIC PERFORMANCE</u>	
<u>27,300 SHP</u>	<u>630A MARK V</u>
Overall thermal efficiency, %	33.7
Steam flow rate, lb/hr	175,200
Throttle pressure, psig	1500
Throttle temperature, °F	1000
Final feedwater temperature, °F	415
Circulator power, KW	489
Auxiliary power, KW	442
Boiler feed pump power, KW	532
Condenser pressure, in. Hg. abs.	1.5

In summary, the cycle performance for the selected design point yields a high plant efficiency relative to efficiencies for conventional or other nuclear plants for marine application. The capability of the 630A Nuclear Steam Generator to provide steam with 400°F superheat, or more, depending on operating pressure, is utilized to obtain improved operations with regard to both efficiency and reliability. No turbine moisture removal devices are required. A large variety of plant sizes from 10,000 to 70,000 SHP or higher with throttle steam conditions as desired (up to 1000°F and 1500 psig or higher) are feasible.

1.4.2 Cycle Optimization

The thermal efficiency of 33.7 percent indicated on Figure 1.5 compares favorably with conventional maritime steam plant performance; however, it does not represent the maximum achievable efficiency. The basic cycle, the primary and secondary loop operating conditions and the feedwater heating cycle have been selected for consistency with the objectives of plant economy, reliability and use of readily available components requiring little or no development costs.

To meet these requirements, the indirect cycle with helium primary coolant and with a non-reheat, regenerative steam plant having five stages of feedwater heating was selected. During the course of the 630A conceptual studies the following parameters were investigated to determine their effects on cycle optimization.

- a. Primary coolant type
- b. Primary coolant operating pressure

- c. Primary coolant operating temperature
- d. Primary coolant circulator power
- e. Type of cycle - direct and indirect
- f. Type of steam plant - reheat and non-reheat
- g. Turbine throttle steam pressure
- h. Turbine throttle steam temperature
- i. Final feedwater temperature
- j. Number of feedwater heaters
- k. Moderator heat

A summary of the effects of varying these parameters is provided in Table 1.5, and a brief discussion of each follows:

Table 1.5

THE EFFECTS OF PARAMETER VARIATIONS ON PLANT EFFICIENCY^(a)

<u>Parameter</u>	<u>Base</u>	<u>Variation</u>	<u>Performance Point Change</u>
Coolant type	Helium	Air	- 0.6
Operating pressure, psia	830	- 100	- 0.16
Reactor exit helium temperature, °F	1200	+ 50	+ 0.1
Boiler exit helium temperature, °F	550	+ 50	- 0.19
Loop pressure loss, psi	7.0	+ 1.0	- 0.07
Circulator power, BHP	500	+ 100	- 0.11
Type of plant	non-reheat	reheat	+ 1.3
Throttle pressure, psig	1500	- 900	- 1.6
Throttle pressure, psig	1500	- 650	- 0.9
Throttle temperature, °F	1000	+ 50	+ 0.5
Final feedwater temperature, °F	415	+ 45	+ 0.091
	415	- 45	- 0.33
Moderator heat, %	7	+ 1	- 0.18
Number of heaters	5	- 1	- 0.10

(a) The base efficiency is 33.7 percent

Primary Coolant Type - With the assumptions of constant primary loop geometry, fixed operating temperatures of the coolant and small percent pressure loss systems, the circulator power requirement is approximately proportional to R^2/Cp^3 in which R is the gas constant and Cp the specific heat. This parameter and the relative pumping power for several working fluids at the operating conditions of the 630A is shown in Table 1.6.

Table 1.6

RELATIVE PUMPING POWER FOR SEVERAL WORKING FLUIDS

<u>Working Fluid</u>	<u>Gas Constant R Ft/°Rankine</u>	<u>Specific Heat Cp Btu/lb °F</u>	<u>$\frac{R^2}{Cp^3}$</u>	<u>Relative Pumping Power</u>
Helium	386	1.242	78,000	0.483
Steam	86	0.61	32,600	0.201
Air	53.3	0.26	161,600	Base
Carbon Dioxide	34.9	0.275	58,600	0.363
Hydrogen	767	3.42	14,700	0.091

Although hydrogen has the lowest pumping power requirement and also has high thermal conductivity, it is unattractive because of the associated explosion hazard. Helium, steam and carbon dioxide provide significant reduction in the necessary pumping power relative to that for air, and helium has the additional attractive characteristic of high thermal conductivity, about five times larger than that for the other three fluids. Also, helium has a negligible cross-section for neutron capture, is stable in the presence of heat and radiation, chemically inert, and is not hazardous.

Primary Coolant Operating Pressure - The pumping power requirement is approximately inversely proportional to the square of operating pressure. The 830 psia circulator exit pressure (900 psig design) was selected consistent with the Zircaloy moderator tube temperature and strength requirements.

Primary Coolant Operating Temperatures - The reactor exit helium temperature of 1200°F was selected as a compromise regarding reactor and boiler tube structural design, fuel element operating temperatures and boiler sizing. With low circulator power, the effect of varying exit temperature by 50°F, for instance, on plant efficiency is only 0.3 percent (0.1 points).

The boiler exit temperature of 550°F was selected to provide acceptable pumping power and boiler sizing consistent with near optimum final feed-water temperature. A 25°F increase in boiler exit helium temperature from 550°F to 575°F results in an increase in circulator power requirement of

about 90 BHP and less than 14 percent decrease in heat transfer area.

Primary Coolant Circulator Power - A reduction of circulator power improves plant efficiency and reduces the boiler duty and heat transfer area requirement. The parameters affecting circulator power are boiler, reactor, and ducting flow path geometry, operating pressure and temperatures and coolant type. The increase in plant efficiency is 0.33 percent (0.11 points) for a decrease in power of 100 BHP.

Type of Cycle - A direct cycle in which the reactor coolant is also the propulsion system working fluid offers the advantages of reduced weight and cost of the power plant by elimination of the boiler and by reduced height of pressure and containment vessels. Use of steam, air, helium or carbon dioxide as the working fluid would be possible but for the gases would require turbomachinery development. The indirect cycle, selected for the 630A nuclear steam generator to insure maximum safety, provides the advantage of retaining the reactor coolant within the pressure and containment vessels, thereby practically eliminating the possibility of releasing activated coolant to the engine room.

Type of Steam Plant - Brief investigations of the feasibility of using a reheat steam plant showed that reheating provides only marginal economic gains and furthermore the boiler arrangement is complicated and space requirements are increased. The performance gain for reheating to initial throttle temperature is about 4 percent (1.3 points) in plant efficiency.

Turbine Throttle Steam Pressure - The optimum throttle pressure for plant capacities near 30,000 SHP is about 1500 psig. The gain in plant efficiency resulting from an increase in pressure from 600 psig to 1500 psig is about 5 percent (1.6 points). For higher plant capacities, pressures higher than 1500 psig would yield additional performance improvements.

Turbine Throttle Steam Temperature - Thermodynamic performance is improved about 1.5 percent (0.5 points) for each 50°F incremental increase in throttle temperature. From considerations of turbine reliability in a marine application, it is felt that for the present 1000°F temperature is the highest desirable value.

Final Feedwater Temperature - The optimum final feedwater temperature for the selected plant size and throttle pressure is about 460°F, considering only the secondary loop performance. Reduction of this temperature from 460°F to 415°F provides increased logarithmic mean temperature differences in the steam generator and a resultant decrease in required heat transfer area of about 12 percent. The loss in cycle efficiency resulting from the temperature reduction is about 0.27 percent (0.091 points). An additional reduction to 370°F feedwater temperature would reduce performance by about 0.33 points.

Number of Feedwater Heaters - Availability of extraction openings in the propulsion turbine casing and economic dictates limit the number of heaters to about five. If four heaters rather than five were utilized, the performance loss would be about 0.33 percent (0.11 points). Eliminating one of the five indicated heaters may be justified economically.

Moderator Heat - The quantity of heat transferred from the moderator to the feedwater cycle depends on heat generation in the moderator and moderator-cooled reactor components and also on heat transferred from the primary coolant to the moderator. This moderator heating reduces cycle efficiency which is based on total reactor power. A one-percent increase in moderator heat reduces cycle efficiency by 0.54 percent (0.18 points).

Preliminary investigations of these parameters have provided a basis for the design point selection. It is felt that the selected cycle and operating conditions are close to optimum and provide a high degree of safety and reliability.

1.5 DESIGN SPECIFICATIONS AND STANDARDS

1.5.1 Applicable Codes and Requirements

The 630A is being designed to comply with all the codes and regulations of the cognizant regulator bodies including:

1. Atomic Energy Commission
2. United States Coast Guard, including CG-115, "Maritime Engineering Regulations and Material Specifications," and CG-259, "Electrical Engineering Regulations."
3. American Bureau of Shipping, including "Rules for Building and Classing Steel Vessels - 1964," and "Guide for the Classification of Nuclear Ships."
4. United States Public Health Service.
5. International Convention for Safety of Life at Sea, 1960.

1.5.2 Pressure Vessel Applications

In addition to the requirements of the above codes and regulations, the requirements of Section III (Nuclear Vessels) of the ASME Boiler and Pressure Vessel Code will be satisfied for those components falling within the scope of this Code.

The rules of the Code are intended to provide minimum safety requirements for new construction and for mechanical and thermal stresses due to cyclic operation. They do not cover all the details for provision against deterioration which may occur in service as a result of factors such as corrosion, erosion, radiation effects, instability of materials, mechanical shock, and vibratory loading. The materials, methods of construction, and inspection requirements will be carefully selected to minimize such effects. Requirements with respect to these effects will be established and incorporated into the design criteria to insure integrity throughout the 20-year design life of the vessel and structure.

1.5.3 Ship Motions

The 630A is designed to operate satisfactorily when subjected to the following

ship motions specified by the American Bureau of Shipping, "Guide for the Classification of Nuclear Ships," 1962:

1. Roll - 30 degrees single amplitude.
2. Pitch - 10 degrees single amplitude.
3. List - 15 degrees.
4. Trim - 5 degrees

Emergency devices and safety systems associated with the reactor are designed to operate satisfactorily when subjected to the following ship motions:

1. Roll - 45 degrees single amplitude.
2. Pitch - 12 degrees single amplitude.
3. List - 45 degrees.
4. Trim - 10 degrees

Maximum angles of roll and pitch will not be considered simultaneously with maximum angles of list and trim.

The containment vessel is designed to remain intact in any attitude assumed by the ship in the event of a collision or sinking. Provision will be made for flooding the containment vessel to equalize pressures and prevent collapse of the vessel if the ship should sink in deep water. The major components of the 630A will be designed to remain in place after a collision or sinking.

Forces Caused by Ship Motions - In the absence of a specific ship application, the 630A is designed to operate satisfactorily when subjected to acceleration forces of 1.5 G. For a specific ship application, the 630A will be designed to operate satisfactorily when subjected to the following forces caused by acceleration of the ship:

1. Vertical acceleration - The vertical acceleration is the vector sum of maximum accelerations caused by:

- a. Heave and pitch.

$$\left(0.3 + \frac{\text{distance to admidship}}{\text{length on LWL}} \right) G$$

- b. Roll. The maximum accelerations due to roll will be calculated on the basis of simple harmonic motion and the ship's natural period with roll angles between 0 and 30 degrees single amplitude. The natural period should be the shortest that can occur in any condition of ship's loading.

2. Athwartship accelerations - The athwartship acceleration is the

vector sum of maximum acceleration caused by:

a. Sway and yaw.

$$\left(0.1 + 0.4 \frac{\text{distance to amidship}}{\text{length on LWL}} \right) G$$

b. Roll. Same as 1 (b) but not less than 1.0 G.

3. Fore and aft accelerations - The fore and aft acceleration should be not less than 1.0 G.

Should the acceleration forces for a specific ship application prove more severe than the 1.5 G presently used in the design, they can be met without undue difficulty.

1.5.4 Radiation Protection Standards

The shielding and radioactive-contamination control will be designed to satisfy the requirements set forth in the "Code of Federal Regulations, Title 10 - Atomic Energy, Chapter 1 - Atomic Energy Commission, Part 20 - Standards for Protection Against Radiation," as required by the Atomic Energy Commission and the U. S. Coast Guard. The criteria set forth in the following documents will be used as guides in establishing the requirements for the overall design.

1. "Background Material for the Development of Radiation Protection Standards - Staff Report No. 1," Federal Radiation Council, May 1960.
2. "Background Material for the Development of Radiation Protection Standards - Staff Report No. 2," Federal Radiation Council, September 1961.
3. "Licensing of Production and Utilization Facilities," Code of Federal Regulations, Title 10, Part 50.
4. "Reactor Site Criteria," Code of Federal Regulations, Title 10, Part 100.
5. "Port Operation in the Matter of the N.S. SAVANNAH," AEC/MA Joint Group Revision of August 1, 1962.
6. "N.S. SAVANNAH Technical Specifications," Todd/SML-NSS 15, May 1963.

The Code of Federal Regulations is not directly applicable to shipboard reactors and references 5 and 6 have been prepared to provide interim regulatory standards of nuclear safety for a particular application. All of these references will be used in the formulation of the necessary design standards, safety procedures, and operating limits to govern the nuclear safety aspects of the 630A.

1.6 DESIGN PHILOSOPHY

The 630A is designed to delivery superheated steam throughout its 20-year design life. Although the particular configuration described is for 1500 psi, 1000°F steam, the power plant is easily adaptable to throttle steam conditions for

other modern marine propulsion turbines. The size is not to exceed 20 feet in diameter and 40 feet in height. The weight is not to exceed 600 long tons including shielding, containment vessel and accessory equipment.

A modular design concept is employed to facilitate installation and maintenance. The size and weight of the modules will be limited to that which can be shipped by commercial land transportation. This concept minimizes the cost of fabrication and installation and reduces the ship fabrication schedule as the power plant is being fabricated in parallel. The reliability and maintainability are also enhanced as the individual modules will be inspected and checked out prior to their installation on a ship.

There are three barriers in the 630A to prevent the release of fission products from the assembly: the cladding over the fuel, the pressure vessel which is fully shielded by the side shield and the shield plug, and the shielded containment vessel. A fourth level of containment is possible with a reactor compartment in the ship.

Reliability of components will be stressed; the objective is a nuclear steam generator that has a reliability consistent with the high reliability of modern steam turbine power plants.

The power plant is designed so that major maintenance operations such as refueling will normally be required only at intervals of two or more years. The refueling will be accomplished by removing the reactor - shield-plug unit complete with control rod actuators, control rods, and nuclear chambers and replacing it with a previously refueled and checked-out unit. Other modular units requiring major maintenance may be exchanged in a similar manner, thus minimizing the time required for such operations.

2. MARK V(A) REACTOR SHIELD-PLUG ASSEMBLY

2.1 DESCRIPTION AND FUNCTION

The 630A Mark V(A) reactor configuration employs the use of low enrichment fuel rods in a calandria-type reactor. This design does not represent an appreciable departure from earlier versions of the 630A reactor, except in fuel element design.

The reactor, shield plug, reactor control elements, and reactor instrumentation are assembled into an integral unit, called the reactor - shield-plug assembly. Figure 1.1 shows the reactor - shield-plug assembly positioned within the 630A Mark V(A) Maritime Nuclear Steam Generator. Reactor flux is measured by seven nuclear sensors located in seven of the 12 moderator-water exit pipes. Exit helium temperatures at selected cell locations are measured by thermocouples located in the instrumentation harness directly below the reactor.

Reactor - The helium-cooled, light-water-moderated reactor utilizes cylindrical fuel rods in the core. As shown in Figure 2.1, the calandria-type reactor vessel is suspended from the shield plug by the moderator exit tubes and control rod guide tubes. The normal design power is 60.4 thermal megawatts.

The reactor vessel is a 4.50-inch-thick Incoloy 800 cylindrical tank closed at each end by tube sheets. The vessel is 80.0 inches in diameter and 66.50 inches long. Moderator tube holes are drilled in each tube sheet on a two-pitch arrangement. The tubes are passed through the tank, rolled, and seal-welded to the tube sheets. The moderator tubes are 3.826 inches in outside diameter and consist of Zircaloy-2 in the active core region and Incoloy 800 at the ends for the tube sheet junctions.

The active core consists of an array of 109 fuel elements arranged in a hexagonal pattern with a total U^{235} inventory of 363 pounds. Figure 2.2 shows a typical fuel cell arrangement. The center 61 cells are spaced at a 4.4-inch pitch and the outer 48 cells at a 4.625-inch pitch. Each cartridge contains 55 low enrichment pin-type fuel elements. Details of the fuel element assembly are given in Section 2.2.

A 4-inch-thick beryllium reflector, which reduces reactor neutron leakage, surrounds the active core and is irregularly contoured to follow the outer row of fuel cartridges.

A Zircaloy-2 flow divider, 0.5-inch thick and 56 inches high, surrounds the reflector. Its contour follows that of the outer surface of the reflector. The 0.25-inch gap between the flow divider and the reflector is used as a water annulus for reflector cooling.

**Fig. 2.1 - Calandria-type reactor design for the 630A Mark V(A).
(221R441)**

221R438

VIEW B-B

VIEW C-C

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Fig. 2.2 - Fuel element and cell - Mark V(A) (221R438)

Two borated Type 304 stainless steel thermal shields are arranged concentrically with the core and provide gamma shielding between the active core and the reactor vessel. Thus, gamma heating (and thermal stresses) are reduced. Borated steel is used to reduce secondary gamma production. The inner shield is 2.0 inches thick and is located inside the calandria-type reactor vessel. A 0.25-inch-wide gap between this inner shield and the inside of the reactor vessel is also used as a water annulus. The outer thermal shield is 2.5 inches thick and is located in the helium flow path midway between the outside surface of the reactor vessel and the inside surface of the pressure vessel.

Shield Plug - The Mark V(A) shield plug provides shielding for the area directly above the reactor, serves as a moderator water inlet and outlet plenum, provides a mounting surface for the control rod actuators, and serves as the top head of the pressure vessel. The moderator water is the neutron shield and a lead slab is the primary gamma shield. The shield plug is divided into an upper and a lower plenum. The upper plenum serves as the water inlet plenum; the lower plenum acts as the water exit plenum and also contains the Incoloy-800-clad lead slab.

In addition to serving as an inlet water plenum, the upper plenum of the shield plug also provides space for movement of the control rod actuator plates. The bottom of the plenum is formed by a thick flanged plate and the top is closed by a flat cover plate. The control rod guide tubes penetrate and are welded to the flanged plate; thus, inlets for the moderator water are provided. The control rod actuators are mounted on the top cover plate; the extension rods pass through the cover plate and are connected to actuator plates which drive the control rods. The control rods, in turn, extend through the guide tubes into the core. The lower plenum contains a 7-inch-thick lead slab clad enclosed within, and bonded to, a 1-inch-thick Incoloy plate. Passages for the control rods are supplied by thin-wall Incoloy 800 tubes which go through the lead slab and are welded to the plates. The cylindrical shell and head serve as the top cap of the pressure vessel and as a header for the twelve moderator exit tubes.

The shield plug components, except for the head and the lead cladding, are SA-212B mild steel with an Incoloy 800 cladding on all inside surfaces exposed to the water. The head and lead cladding are Incoloy 800.

Moderator Circuit - The water in the shield plug is in series with the reactor moderator water. From the upper plenum of the shield plug, the moderator water flows down through the shim control rod guide tube and discharges into the plenum at the aft region of the reactor vessel. The water flows upward through the active core and reflector region toward the top of the reactor vessel. It then

passes from the core region into the lower plenum of the shield plug through the moderator exit tubes, which join the reactor vessel to the shield plug. The heated moderator water is discharged from the lower plenum by two tubes leading to the moderator heat exchanger. The moderator water is cooled in the heat exchanger by the boiler feedwater. The cooled moderator water then flows to the moderator pump inlet completing the flow circuit. A tube extending through the shield plug into the bottom of the reactor vessel through one of the 12 moderator exit tubes is provided to assure a positive means to pump the water from the reactor - shield-plug assembly.

2.1.1 Design Requirements

The reactor vessel and shield plug are designed for an external pressure of 900 psig and an internal pressure of 258 psig at a maximum operating temperature of 450°F. Both are designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, the American Bureau of Shipping, "Rules for Building and Cladding Steel Vessels - 1964," and "Guide for Classification of Nuclear Ships - 1962," and the U. S. Coast Guard, "Marine Engineering Regulations and Material Specifications - 1963."

Requirements with respect to corrosion, erosion, radiation effects, instability of materials, mechanical shock, and vibratory loading, will be established and incorporated into the design criteria to insure integrity throughout the 20-year design life of the vessels and structure. The materials, methods of construction, and inspection requirements will be carefully selected to minimize these effects.

2.1.2 Materials and Fabrication

All the metallic components of the reactor are Incoloy 800 with the exception of the reflector which is beryllium, the thermal shield which is borated Type 304 stainless steel, the moderator support tubes which are Zircaloy 2 - Incoloy 800 and the control rod guide tubes which are Zircaloy 2. Incoloy was selected because of its high resistance to radiation damage and corrosion. The reflector material is beryllium, selected because of its desirable nuclear characteristics. The two Type 304 stainless steel thermal shields have one percent boron added. The control rod poison matrix consists of vibration-compacted B₄C powder. All the control rods are clad with Incoloy 800.

The reactor pressure vessel and shield plug will be welded, fabricated and tested in accordance with the requirements of the U. S. Coast Guard, the ASME Boiler and Pressure Vessel Code, and the American Bureau of Shipping.

2.2 FUEL ELEMENTS

2.2.1 Description

The active core consists of an array of 109 fuel elements arranged in a hexagonal pattern. Figure 2.2 shows a typical fuel cell arrangement with the fuel

element in place. The center 61 cells are spaced at 4.4 inches pitch and the outer 48 cells at a pitch of 4.625 inches. Each fuel element contains 55 low enrichment fuel rods. The enrichment is a uniform 6 percent throughout the active core. There are 5995 total fuel rods in the core with a total U^{235} inventory of 363 pounds. The fuel elements are positioned inside the moderator support tubes.

Each fuel element consists of 43 standard 0.350-inch-diameter rods and 12 smaller 0.300-inch-diameter rods. Figure 2.2 shows a typical fuel element and cell. The rods are attached to a support plate at the top of the fuel element. A bottom support plate positions the rods and provides clearance for axial thermal expansion. The fuel rods are also positioned at the center of the element by a ferrule to limit vibration and deflection of the fuel rods. An 0.020-inch-thick Incoloy 800 or stainless steel cylindrical shell, around the entire rod bundle, provides the outer flow boundary for the primary helium coolant. The shell also serves to maintain an insulating helium gap, 0.040-inch thick, between the fuel element and moderator support tube. A seal located in the nose section of the fuel element prevents helium flow through the gap during operation. The aft section of the fuel element contains a latching device and supplies additional insulation beyond the active core region. A cross section through the fuel element is shown in Figure 2.3.

The cylindrical fuel rods are conventional in design concept. They are similar in configuration to the fuel rods currently in use in many large nuclear power generating stations. Additional design features have been incorporated which are unique to high-temperature gas-cooled and superheat fuel rods.

The fuel rods are identical in concept and design to the rods shown in Section 3.2. The only difference between the Mark V(A) and Mark V(B) fuel design is the fuel element assembly.

2.2.2 Design Requirements

The design requirements are identical with the design requirements discussed in Section 3.2.2. The burnup lifetime is 20,000 MWD/MT.

2.2.3 Materials and Fabrication

Materials and fabrication of the fuel elements in the Mark V(A) concept are similar to those in the Mark V(B) presented in Section 3.2.3.

2.3 MARK V(A) THERMODYNAMIC DESIGN

2.3.1 Operating Conditions and Performance

The operating conditions and performance of the 630A Mark V(A) and Mark V(B) are essentially the same. The major differences are that the Mark V(A) reactor has a slightly higher power level and increased moderator flow requirement because of higher moderator flow passage area and increased heat transfer. The heat transferred to the moderator is higher for the Mark V(A) because the

Fig. 2.3 - Fuel element cross section Mark V (A)-(221R439)

reflector is water cooled and because of increased heat transfer from the fuel elements to the moderator.

The average maximum fuel rod surface temperature for the Mark V(A) is about 1320°F, 15° lower than for the Mark V(B) because of reduced free flow area and resulting increased heat transfer coefficient. Fuel rod central temperatures are somewhat lower for the same reason.

Applicable detailed operating conditions are shown in Section 3.3.1 and also in the Appendix.

2.3.2 Analytical Expressions

The analytical expressions used for thermal analysis are described in Section 3.3.2 for both the Mark V(A) and Mark V(B) configurations.

2.3.3 Fuel Element Steady State Analyses

The Mark V(A) reactor heat deposition is similar to that shown in Section 3.3.3. However, the side reflectors are water cooled.

Fuel rod surface and central temperatures for both 2/1 sine and "rods in" power profiles are shown in Figure 2.4. These temperatures are based on average flow and power conditions. The average maximum surface temperature for a 2/1 sine power profile and a "rods in" power profile is 1320°F and 1370°F, respectively. The average maximum rod central temperatures are 1710°F and 1865°F for the 2/1 sine and "rods in" profiles, respectively.

2.3.4 Temperature Perturbations

Detailed temperature perturbations have not been calculated for the Mark V(A) reactor. They are not expected to be substantially different from those presented in Section 3.3.4 for the Mark V(B) reactor.

2.3.5 Fuel Element Transient Analyses

Normal and emergency aftercooling are discussed in Section 3.3.5. The node arrangement with selected temperatures and results of a Transient Heat Transfer Program calculation for the Mark V(A) hypothetical loss of coolant flow accident are shown in Figures 2.5 and 2.6, respectively. The maximum rod surface and central temperatures of an average cell with flat fine radial power are seen to be 2166°F and 2183°F, respectively. These temperatures are for the central rod and no other rods reach higher temperatures.

2.4 REACTOR PHYSICS

The change in design of the 630A to a low enrichment Mark V system brought about problems in the analysis methods which have not yet been fully resolved. The largest single problem has been in the determination of self-shielding factors for the U^{238} resonances; or stated differently, in the determination of the resonance integral of U^{238} in the Mark V system. Most of the available proven

Figure 2.4

630A MARK V(A)

Fuel Rod Surface and Central Temperatures

Note:

- (1) Letters indicate nodes whose temperatures are plotted on Figure 2.6.
- (2) Numbers are temperatures in $^{\circ}\text{F}$ for the nodes indicated at the time when the temperature of node a is a maximum (260 seconds).

Fig. 2.5

630A Mark V(A) Fuel Element Temperatures for Loss of Coolant Flow Analysis

data is for fuel pins in very large lattices, where the bulk of the neutron moderator occurs between the pins in the lattice. A few reports deal with clusters of fuel rods in fairly tight lattices, with large moderator regions between the clusters. The most relevant data unearthed so far are for clusters of natural uranium oxide rods, with heavy water as the moderator. These data are being used to check out the nuclear analysis methods which are being employed in the design of the 630A Mark V.

2.4.1 Analytical Techniques

The computer programs, used for nuclear design and analysis, have, for the most part, been described in earlier reports⁽¹⁾. Program 431 (Isotopic History Trace) which was written for use with reactors containing only one fissionable material (U^{235}), was extended to handle fissions in five uranium isotopes and three plutonium isotopes. The program was further improved by updating many fission product cross sections, and by improving the scheme for handling the Pm-Sm chain; Pm¹⁴⁷ and Pm¹⁴⁸ are important precursors of Sm¹⁴⁹. A new program, George III, was written which combines program C⁽¹⁾, a bare core, multigroup generator, with F-N⁽¹⁾, a one-dimensional, multigroup, multiregion diffusion code.

The analysis methods employ 19-energy-group models, which is fine enough to describe the high energy and resonance energy effects in U^{238} . Self-shielding factors for the U^{238} resonances, are obtained by computing an effective potential scattering cross section per U^{238} atom (σ^*) and employing this value in a NRIA calculation (Narrow Resonance, Infinite Absorber). The best technique for determining σ^* appears to be that described by Levine⁽²⁾. The self-shielding factors have been applied to groups 11, 12 and 13 which contain the resolved resonances. Levine's article infers that a small reduction should be made to (the infinite dilution value of) the adsorption cross section at group 10, which extends through most of the unresolved resonance range directly below the resolved region.

The method of obtaining the effective potential scattering per U^{238} atom, and the calculation of the self-shielding factors, is discussed in Section 2.4.4.

Two types of one-dimensional analysis techniques have been employed - single cell calculations and gross core calculations. As a rule, program George III is used for these calculations.

The single-cell calculation is employed primarily with an "average" fuel cell of the reactor, with its associated moderator. It is used for determining optimum fuel-moderator configurations (in terms of reactivity), moderator temperature coefficients, Doppler coefficients, flux spectrum in the fuel region, conversion ratios, and other quantities which are relatively independent of reflector and shield effects. Albedo conditions of 1.0 are usually

(1) "630A Maritime Nuclear Steam Generator Status Report No. 1," GE-NMPO, GEMP-231, September 12, 1963, pages 273-274.

(2) M. Levine, "Resonance Integral Calculations for U^{238} Lattices," Nuclear Science Engineering 1, 271 (1963).

applied to both boundaries in the direction of the calculation; neutron leakage in the other direction is handled by inclusion of a removal cross section (DB^2).

The gross core calculations are used to determine overall reactivity, gross power distributions, and control rod worths. These calculations require considerably more computer time, and far more complex input than the single-cell calculations. Therefore, these calculations are made infrequently until the design has reached a stable configuration and the techniques for determining the values for the resonance integrals have been checked against measurements. Preliminary values for in-core changes that result in reactivity changes are computed by comparing the output of a single-cell calculation and a gross-core calculation for a particular cell geometry, temperature and fuel loading, and then doing single-cell calculations with the parameter of interest being varied. Results of such analyses in the past have led to the conclusion that the technique is a good tool for survey work. For example, the moderator temperature coefficient for the 630A critical experiment (the fully enriched design) as predicted by single-cell methods and by gross-core methods, were in good agreement with the measured values and with each other.

2.4.2 Reactivity and Control

The following analyses results are dependent on the evaluation of the resonance integral of U^{238} in this system. This parameter, and the value of σ^* which it is based on, has varied considerably during the early design. The most recent, and complete series of analyses are based on $\sigma^* = 27.25$ barns, which results in a capture integral (Doppler broadened to $1500^\circ F$) of 20.956 barns. The capture integral is from the cadmium cutoff to 10 Mev; non fission capture in the Mev range is included. The experiment correlation work has caused the value of σ^* to be newly revised; an upper limit of 19.2 barns is indicated. The reduction in the capture integral is 2.817 barns when σ^* is reduced from 27.25 to 19.23 barns. This changes most of the contributors to the control requirements. In the remainder of this section, the reported values will be based on $\sigma^* = 27.25$ barns. These values which are affected by the revision of σ^* will be followed by a bracketed value which is the best estimate of the subject based on $\sigma^* = 19.23$ barns.

The "average" fuel element which is used for the single cell calculation consists of the 55-rod element within the Zircaloy pressure tube, surrounded by an annulus of water. The diameter of this cell is 11.7 cm. The calculations are based on an initial enrichment of 6 weight percent of U^{235} . The reactivity (K) of this cell at operating conditions but without any burnup is 1.20977, which corresponds to a reactivity of the whole core of about 1.131 [1.141].

A. Doppler and Moderator Temperature Coefficient

The change in reactivity in going from a cold configuration to a configuration in which the average fuel temperature is increased to $1500^\circ F$, is $-1.4531\% \Delta k/k$ [-1.35%]*. Therefore, the Doppler coefficient is -1.0 [0.9] $\times 10^{-3}\% \Delta k/k$ per $^\circ F$.

* As noted earlier, the bracketed value is the estimated value based on the lower σ^* .

The change in reactivity brought about by increasing the thermal base temperature up from 68°F to 212°F is +0.888% $\Delta k/k$. However, the reduction in density of the moderator that accompanies this change causes an increase in leakage, and an increase in the number of neutron captures in the U^{238} resonances, so that with this effect included, the overall effect of increasing the moderator temperature to 212°F is +0.361% $\Delta k/k$. Calculations with the moderator temperature at 248°F and 284°F show that the increase in neutron leakage and resonance captures are larger effects than the decrease in non-fission capture in the thermal region, so that the moderator temperature coefficient is negative in the operating range. At 248°F, the moderator temperature coefficient is $-4.25 \times 10^{-3}\%$ $\Delta k/k$ per °F.

The change in reactivity from the completely cold core at 68°F, to the hot (1500°F average fuel, 248°F average moderator) condition is -0.988 [-0.885%] $\Delta k/k$.

B. Lifetime and Enrichment

The fuel lifetime is computed by using the single cell calculation in conjunction with the isotopic history trace program. The first calculation uses the clean fuel element to obtain a neutron spectrum for the isotopic history trace program (Program 431). The latter code computes the fission product and plutonium buildup, and the uranium and plutonium depletion at various time intervals. These quantities are then inserted into the single cell calculation, yielding the change in neutron spectrum and the change in reactivity. The sequence is continued, using different spectra for different parts of the operational time, until the fission product and fuel inventory at a given time coincide with the spectrum used to obtain the inventory. With 6 percent initial enrichment, the Xe^{135} and Sm^{149} worths are found to be relatively constant; ΔK (single cell basis) was -.0235 and -.0104 for the two respective poisons.

The burnup corresponding to 0% excess reactivity was determined by multiplying the reduction in reactivity due to burnup and fission products, other than that due to Xe^{135} and Sm^{149} , by 1.1 (to account for longitudinal effects); subtracting the Xe and Sm worth from the excess reactivity of the hot, Doppler broadened system; and then subtracting the adjusted burnup worth from that value. There were two earlier series of calculations with 6 percent enrichment, which had initial conversion ratios of 0.359839 and 0.285445, respectively. The values for MWD/tonne per % K_{ex}^* were 1610 and 1357, respectively. K_{ex} is defined as the excess reactivity after Xenon 135 and Sm^{149} have been included in the hot, Doppler broadened reactor. It had been found that the higher conversion ratio does not mean longer life for a given inventory, for although the decrease in reactivity with burnup is smaller with a higher conversion ratio, the original K_{ex} is lower, as more neutrons are absorbed in the U^{238} and fewer are available for fission. The reduction in K_{ex} is a larger factor than the improvement in the BURN term, when the design point is 20,000 MWD/tonne.

The estimated lifetime with 6 percent enrichment and $\sigma^* = 27.25$ barns,

* This will be called the BURN term.

is 15,562 MWD/tonne. Increasing the fuel enrichment one percent adds .0357 to Kex. By increasing the enrichment to 6.9 percent, the lifetime is increased to the design value (20,000 MWD/tonne). This is a slight overestimate of the required enrichment, as the BURN term was assumed unchanged, when in fact, it will increase slightly with higher enrichments.

With the reduced value of σ^* , the enrichment is 6.7 percent. This is an upper limit as σ^* is probably lower than 19.25 barns, control rod effects (which have been neglected) will improve the conversion ratio, and again, the BURN term is based on 6 percent rather than 6.7 percent initial enrichment. The final enrichment required for the Mark V(A) will be between 6.0 percent and 6.7 percent.

C. Flooding Analysis

The reactivity increases if the core is flooded. The additional moderator reduces the overall neutron leakage from about the 10 percent that leak from the dry, cold core to 5 percent of the total neutrons. The increase in the resonance capture integral does not effectively reduce the reactivity as the neutrons are scattered through the resonance region more rapidly with the additional water. The reactivity of the cold, flooded reactor is estimated to be 1.2086, as compared to 1.141 for the cold, dry core.

D. Control Rod Worth

The worth of a single, unshadowed rod at the center of the reactor was determined for cold-dry, hot-dry and cold-flooded reactors. The fuel enrichment for these calculations was 5 percent, but this has a small effect on the rod worth. The rod, with a clad diameter of 0.5 inch, and 0.45-inch-diameter poison (65 percent dense natural B₄C) was found to be worth -0.312 percent, -0.304 percent and -0.268 percent $\Delta k/k$ in each of the respective cores. The worth of 171 of these rods were evaluated, using a Bessel function off-center weighting technique, and a flux perturbation technique for determining shadowing coefficients. In unflattened cores, consisting of 109 "average" cells, the overall worths are -22.64 percent, -23.45 percent, and -12.883 percent for the respective cores. The rods in the flooded core are worth less as the off-center worth decreases rapidly, and the large moderator fraction reduces the diffusion length, which reduces the area over which the rod is effective.

More analysis in this area is required, particularly on power flattened designs being flooded.

E. Operational Control Requirements

The 6.9 percent enriched core (with $\sigma^* = 27.25$ barns) will have a cold, clean reactivity of 1.173. If the moderator temperature is elevated while the fuel is cold, the reactivity will rise to a maximum of 1.178 [1.188] when the moderator is at 170°F. At operating conditions, after Xe¹³⁵ and Sm¹⁴⁹ have reached equilibrium, the reactivity will be about 1.131. Assuming

that an additional 0.7 percent is desired in the dynamic rods, and that two percent shutdown is desired, the necessary rod worth is

$$\frac{1 - (1.178 + .007 + .020)}{1.178 + .007 + .020} = \frac{-.205}{+1.205} = -17.0\% \Delta k/k.$$

As shown in the previous section, the worth of the rods fully inserted in the dry core are at least -22.64 percent $\Delta k/k$. Therefore, a 34-inch stroke should be sufficient for the control rods.

The flooding situation is not quite so simple. As previously mentioned, the reactivity increases upon flooding, while the rod worth decreases. At present, the best solution to the problem appears to be the addition of burnable poison, probably B_4C , to the fuel matrix. The quantity of natural B_4C that would be involved is in the order of 500 to 600 grams of contained boron. However, more complete analysis of the flooded, power-flattened core may show that the rod worths do not decrease in value; gross core analysis with all rods inserted is required.

F. Conversion Ratio and Fast Fissions

The initial conversion ratio is 0.334 [0.310]. These values are not calculated for each design iteration, as they are not used as input to the burnup calculation. The isotopic history trace program computes the conversion rate, fast fission in U^{238} and Pu burnup using the neutron spectrum and a built-in, 20-group cross section.

2.4.3 Power Shaping

Gross radial power shaping is accomplished by varying the fuel-to-moderator ratio in the radial direction. To minimize fabrication costs, the fuel element geometry (55 pins per element) is fixed for the entire core. The fuel-to-moderator ratio may be changed by (1) increasing the pitch between the fuel elements as the fuel elements distance from the center of the core increases, (2) increasing the enrichment in the fuel element as its distance from the center of the core increases, or (3) a combination of both. With the first method, the power density at the center of the core will increase with operating time, because the lower pitch there causes a higher production rate of plutonium per fission, so that the effective enrichment increases with operation. With the second method, the relative conversion rate is more nearly constant over the reactor so, as depletion occurs, the center of the core decreases in power. By combining the two methods, the gross radial power distribution can be maintained fairly constant over core lifetime.

The method of choosing the proper fuel-to-moderator ratio in a particular region is based on the assumption that the power generated in that region is partially dependent on the ratio of the reactivity of that fuel element (as determined in a single cell calculation) to the reactivity of the average fuel element, raised to approximately the 5th power. With a constant enrichment, the procedure used is as follows:

- A. Compute the reactivity of single cartridges with varying lattice sizes (varying fuel-to-moderator ratios.)
- B. Compute the gross radial power distribution, using the lattice spacing that yields the maximum reactivity.
- C. Change the fuel-moderator ratio in each region so that the initial power density in that region divided by the power density in the region closest to the reflector equals the reactivity of the maximum reactivity fuel element divided by the reactivity of the new fuel element, taken to the $1/5$ power.

For example, if the central fuel element is operating at 1.3 times the average power density, and the outermost fuel element is at 0.83 times the average density, the lattice spacing at the center of the core is reduced to the value that yields $(0.83/1.3)^{0.2} = 0.9142$ times the reactivity of the maximum reactivity fuel element on the basis of the single cell calculation.

At this point, the gross calculation is repeated using the varied lattice spacing. If the core is not quite power flattened (e.g., the central fuel element is operating at 1.06 times the average power density and it is desired to bring this value to 1.02), the procedure is repeated using a slightly different exponent in the reactivity to power correlation.

The fuel element spacing shown in Table 2.1 yields essentially a constant power per fuel element at the start of life, with 4 percent initial enrichment. The spacing is insensitive to uniform changes in enrichment.

Table 2.1

FUEL CELL PITCH REQUIREMENT FOR
FLAT INITIAL POWER

<u>No. of Fuel Elements</u>	<u>Effective Center-to-Center Spacing of Fuel Elements, in.</u>
1	4.202
6	4.208
12	4.228
18	4.265
24	4.334
30	4.475
18	4.574

The effect of fuel burnup and conversion (uranium to plutonium) on gross radial power distribution in this configuration have not been fully investigated. However, based on a constant fission density across the core, and starting with 4.5 percent initial fuel enrichment, after 11,250 Mw days/MT, the center fuel element would have an effective enrichment of 4.271 percent while the outermost cartridges would have an effective enrichment of 3.983 percent.

This small change in enrichment results in an increase in power density of approximately 9 percent at the center of the core; therefore, by starting with the outer cartridges loaded with 0.2 to 0.3 percent higher enrichment than the center cartridges, power flattening within ± 5 percent through core lifetime can be easily accomplished.

The mechanical design of the core is not as complicated as the many different pitches indicate. The center of the core is designed with 4.4-inch spacings, while the tubes near the reflector approach 4.65-inch spacing. Zircaloy spacers, in the form of heavy control rod guide tubes, shim out moderator water so that the desired volume of moderator water is surrounding each fuel element.

2.4.4 Resonance Integrals Determination and Critical Experiment Correlation

The nuclear cross sections listed on GE-NMPO library tapes and used in computer nuclear codes have infinite dilution values for isotopes such as U^{238} . The presently used values for U^{238} yield an infinite dilution resonance integral of 281.2 barns, and the nuclear cross sections are listed at 19 levels plus a thermal group. Thus, in order to apply the cross sections on the tapes, it was necessary to calculate resonance self-shielded and Doppler-broadened values of effective resonance integrals for the geometries and temperatures of interest.

The calculation of effective resonance integrals was accomplished by means of the narrow resonance infinite absorber (NRIA) approximation which for a homogeneous medium is given for each resonance by

$$I_{\text{eff}} = \sigma_m \frac{\Gamma}{E_0} J\left(\theta, \frac{\sigma_m \Gamma}{\sigma_0 \Gamma_\gamma}\right) \quad \theta = \left(\frac{\Gamma^2 A}{4kTE_0} \right)^{1/2}$$

where

σ_m = Σ_m/N_0 the ratio of the macroscopic potential scattering cross section (Σ_m) of admixed moderator to the number of absorber atoms N_0 .

Γ, Γ_γ = total and radiative level width of the resonance

E_0 = energy of the resonance σ_0 = peak resonance value

$J\left(\theta, \frac{\sigma_m \Gamma}{\sigma_0 \Gamma_\gamma}\right)$ = function accounting for Doppler broadening tabulated in ANL 5800. (1)

A = mass ratio of absorber to neutron

kT = energy equivalent at the absolute temperature of the absorber.

(1) ANL 5800, "Reactor Physics," p. 169.

The NRJA approximation applies when the absorbing resonance is narrow with respect to neutron energy loss on collisions with the moderator, but not with the absorber. This formulation was utilized to calculate the effective resonance integral for each of the 55 resolved resonances in U^{238} .⁽²⁾ The calculations were carried out for σ_m equal to 10, 25, 50 and 75 barns at temperatures of 68°F, 500°F, 1500°F and 4000°F. A plot of the effective resonance integral for 68°F and 1500°F values as a function of σ_m is shown in Figure 2.7. In the level formulation of GE-NMPO cross section tapes, the energy boundaries for levels 11, 12 and 13 are 1234-167 ev., 167-22.62 ev, 22.62-3.949 ev, respectively, and all of the resolved resonances are within these three levels. The contribution to the resonance absorption integral outside these levels, including $1/v$ absorption, is 5.03 barns with the infinite dilute resonance integrals within the three levels are 10.82, 69.97 and 193.97 barns for levels 11, 12 and 13, respectively. From the calculations of effective resonance integral a correction factor defined as

$$g_i = \frac{\sum_j I_{eff,i,j}}{I_{\infty,j}}$$

was determined for each of the levels containing resolved resonances, where j applies to the nuclear library level structure and i to the individual resonances within each level.

Application of Resonance Integral Calculations

The parameter which dictates the magnitude of the resonance integral in the previous formulation is the potential scattering per absorber nuclide, σ_m . Initial attempts to determine σ_m , and hence the effective resonance integral, utilized the methods discussed in ANL 5800, pp 277 ff. In essence, the assumption is made that systems having the same equivalence parameter, σ_e , have equal resonance integrals whether the fuel is homogeneously mixed with moderator, in an isolated lump, or in a lattice. σ_e is defined for lattices as

$$\sigma_e = \sigma_{pot} + \frac{1}{N_0} \left[\Sigma_1 + \frac{S(1-C)}{4V} \right]$$

where σ_{pot} is the potential scattering of the absorber atom, N_0 is the atom density of absorber atoms in the fuel region, Σ_1 is the macroscopic potential scattering of admixed scatters in the fuel, S is the fuel region surface area, V its volume, and C is a flux reduction or Dancoff correction factor for the lattice. Since a NRJA approximation was used in calculating resonance integrals, as discussed previously, σ_{pot} would be equal to 0.

Application of the above formula to a single isolated rod of UO_2 0.327 inch diameter (0.831 cm) of density 10 gms/cc gives

$$\sigma_e = 7.51 + \frac{1}{0.831} \cdot \frac{1}{0.021} = 7.51 + 57.30 = 64.81 \text{ barns}$$

where 7.51 represents the oxygen potential scattering per absorber nucleus

(2) Liikula, R. C. Updated R.B.U. Basic Library HW-75716 Vol. 3, May 20, 1963

Figure 2.7

Effective Resonance Integral U-238

I_{eff} Resolved + 5.03 Barn

(10 MEV to Cd Cut-Off)

$I_{\text{eff}} \sim \text{Barns}$

1500°F

68°F

$\sigma_m = \Sigma_m / N_o = \text{Potential Scattering Per Absorber Atom-Barns}$

13 X 10 TO T-HE CM. 355-1.4
KEUFFEL & ESSER CO. MADE IN U.S.A.

$S/4V = \frac{1}{D}$ for a rod. From Figure 2.7 the resonance integral corresponding to this value is approximately 27 barns. Measured values of resonance integrals for isolated UO_2 rods have been made by Hardy, et al,⁽³⁾ yielding a formula for UO_2 resonance integral

$$I = 6.09 + 24.8 \sqrt{\frac{S}{M_{UO_2}}}$$

For the case of interest, $S/M_{UO_2} = 0.482$

$$I = 6.09 + 24.8 (0.695) = 23.33 \text{ barns}$$

which from Figure 2.7 corresponds to $\sigma_m = 47$ barns.

The results are obviously not in good agreement, and the error is larger than can be attributed to treating oxygen as an effective moderator, since it contributes at most 7.5 barns to the total.

Experimental Correlation of Conversion Ratios

The initial conversion ratio correlation was not performed on the 630A Mark V insert because the measurements were not considered complete enough to warrant extensive analysis. This is due principally to the brevity of the testing period allowed and failure to obtain all the desired measurements. Instead the analysis was performed on a seven-pin natural UO_2 element. The measurements on this element were made in ZED-2, a low-power, heavy water-moderated reactor at Chalk River and have a quoted precision of about $\pm 1/2$ percent.⁽⁴⁾ The fuel element consists of 7 natural UO_2 fuel rods with an O.D. of .938 inch enclosed in aluminum tubes. The seven-pin cluster is enclosed in an aluminum pressure tube with an I.D. of 3.25 inches.

Analysis Technique

The 7-pin natural UO_2 fuel element was analyzed using program George III employing single cell type calculation. Resonance self-shielding factors for U^{238} were calculated for levels 11, 12 and 13 using the NR1A approximation described earlier. Initially the effective potential scattering per absorber nuclide (σ_e) was calculated using the relation

$$\sigma_e = \sigma_{pot} + \frac{1}{N_0} \left[\Sigma_1 + \frac{S(1-C)}{4V} \right]$$

Bell's approximation⁽⁵⁾ to the Dancoff correction (C) was used to calculate the flux reduction both between individual fuel rods and between fuel elements. George III results using the above technique are shown in Table 2.2.

As seen in Table 2.2, the results are not in good agreement, especially when one removes the sub-resonance part of the conversion (about 0.56 out of each value). The error in the resonance region is then made more apparent.

- (3) J. Hardy, Jr., G. G. Smith, D. Klein, "The Effective U^{238} Resonance Captive Integrals of Uranium Metal and UO_2 Rods," Nuclear Science & Engineering: 14, pp 358-365 (1962).
- (4) P. R. Tunncliffe, et. al, "A Method for the Accurate Determination of Relative Initial Conversion Ratios," Nucl. Science Engineering 15, 268-283 (1963)
- (5) ANL 5800, "Reactor Physics," page 280.

Table 2.2

INITIAL CONVERSION RATIOS (ICR) - 7-PIN ELEMENT

<u>Lattice</u>	<u>Air Coolant</u>		<u>D₂O Coolant</u>	
	<u>C</u>	<u>M</u>	<u>C</u>	<u>M</u>
19 cm	.913	.870	.939	.880
22 cm	.818	.793	.841	.810

C = calculated

M = measured

In all cases the calculated ICR is higher than that measured. This indicated that σ_e used in the NRIA approximation is much too large as was also concluded after the evaluation of a single isolated rod of UO₂ that was described earlier.

An improved method of determining σ_e in conjunction with the NRIA approximation has been suggested by Levine⁽²⁾. Levine defines σ_e as

$$\sigma^* = \frac{f\sigma_{\text{pot}}}{a_1} + \frac{S}{4NV} \cdot \frac{1-C}{1+(a_2-1)C}$$

where f is the fraction of the admixed moderator which is effective (f = .7 for Oxygen), a_1 is the escape probability parameter associated with the fuel rod chord length, and a_2 is the escape probability parameter associated with the lattice. Initial conversion ratios calculated using Levine's method for σ_e are shown in Table 2.3.

Table 2.3

INITIAL CONVERSION RATIO
(Levine Method)

<u>Lattice</u>	<u>Air Coolant</u>	
	<u>C</u>	<u>M</u>
19 cm	.868	.870
22 cm	.786	.793

As seen in Table 2.3, the results are in good agreement for the case of air coolant. In the seven-pin element analysis, the aluminum cladding on the fuel rod was neglected because of its negligible σ_p ; however, the Incoloy cladding must be included in the evaluation of the resonance integral in the Mark V(A).

2.5 CONTROL RODS

2.5.1 Description and Function

Reactivity control of the 630A Mark V(A) reactor is accomplished by two types of control rods and a burnable poison. The control rods are designated as shim-scam rods and dynamic rods. The shim-scam rods are used for rapid, positive shutdown of the reactor and also for gross reactivity control. The dynamic rods maintain fine control for normal operating transients. The burnable poison is used to extend the operating life of the reactor by furnishing gross reactivity control at the start of life, thus permitting a higher initial fuel inventory with reasonable control rod worth.

There are presently 168 shim-scam rods and 12 dynamic rods in the reactor. The rods are located in the interstitial positions between the cells. The rods are ganged into groups in the upper chamber of the shield-plug. The groups of control rods are driven by control rod actuators located on the top surface of the shield plug. The arrangement of the rods in the reactor and the grouping of the rods into frames in the upper chamber of the shield plug is shown in Figure 2.8. The position of the actuator is superimposed on each frame. There are seven shim-scam rod actuators and 2 dynamic rod actuators.

The shim-scam rods and the dynamic rods consist of a poison tip and an extension tube. These rods are identical except for length of the poison tip and the rod function. The poison tip consists of a cylindrical tube that is filled with the poison matrix (B_4C), with an outside diameter of 0.50 inch and a wall thickness of 0.025 inch, sealed at the bottom by an end piece. The length of the poison matrix is 34 and 30 inches for the shim-scam rods and dynamic rods, respectively. The extension tube has an outside diameter of 0.50 inch with a wall thickness of 0.06 inch. The extension tube and poison tip are joined by welding to a tubular center plug. The extension tube is closed at the top by an end fitting for attachment to the gang plate. The extension tube provides an expansion volume for the helium released from the poison matrix.

The burnable poison (natural B_4C) is added to the fuel matrix of the outermost pins in each fuel element. The total quantity of B_4C in the core will be in the order of 300-500 grams.

2.5.2 Control Rod Design Requirements

The control rods are designed to operate in a flowing water environment with a maximum temperature of 250°F. The shim-scam control rods must withstand the dynamic force exerted by the actuators during a scram shutdown. The control rods must also be able to operate in any attitude assumed by the ship.

2.5.3 Materials and Fabrication

The poison matrix material is natural boron carbide powder that is vibration compacted within the cladding to a density of 1.7 grams per cubic centimeter. Powdered B_4C is being used because of its low cost.

ACTUATORS		RODS	
TYPE	NO.	PER ACTUATOR	TOTAL
SHIM-SCRAM	4	26	104
SHIM-SCRAM	2	23	46
SHIM-SCRAM	1	18	18
DYNAMIC	2	6	12
TOTAL	9		180

Fig. 2.8 - Actuator Ganging Schematic Mark V(A) (849D322)

The material of the tubing and end fittings of the control rods is Incoloy. Incoloy was selected because of its resistance to radiation damage and to stress corrosion cracking.

2.6 CONTROL ROD ACTUATORS

Two types of control rod drive actuators are used for control and reactor safety. These consist of seven shim-scam actuators and two dynamic actuators. The shim-scam actuators have a 34-inch stroke and the dynamic actuators have a 30-inch stroke. One shim-scam actuator drives a gang of 18 rods, 2 shim-scam actuators drive 2 gangs of 23 rods each and 4 shim-scam actuators drive 4 gangs of 26 rods each. Two dynamic actuators drive 2 gangs of 6 rods each. The ganging arrangement is shown in Figure 2.8. Each actuator drive shaft is connected to the ganging frame through quick-disconnect couplings. Each actuator drive rod pierces the shield plug through a rod seal which is built into the actuator. The rod seal consists of a standard chevron-type seal which has proven highly successful in this type of application during recent tests.

2.6.1 Shim-Scram Rod Actuator

Figure 2.9 shows a combined shim-scam actuator which is spring driven in the "in" or shutdown direction and mechanically driven by a cable winding up on a drum in the "out" or power increase direction.

It consists of a cylindrical housing, piston and drive rod assembly, wind-up cable and drum, clutch, gearbox motor and limit switches.

In its normal off position the drive spring acts against the buffing piston to drive the shaft to its "in" position. To operate the unit, the clutch is activated which connects the cable drum to the drive motor so that the rod can be withdrawn by winding the cable onto the drum until the critical position is reached. It is held in this position by the non-backdriving gear box. From this position it can be either driven in or out. When it is driven out, it further compresses the scram spring and when it is driven in, the cable unwinds and the spring drives the rods in. This provides the shimming function of this actuator.

For scrambling, the clutch is de-energized and the spring drives the piston and shaft to its full insert position, at the same time unwinding the cable from the drum. The piston compresses the air contained in the cylinder ahead of it, thereby providing buffing for the actuator and the control rods. An additional spring buffer is provided for the drum to slow it down without causing damage to the drum or housing. Limit switches are provided at both ends of travel to stop the drive motor and to light indicating lights on the control console. Fail-safe operation of the scram function is provided because any loss of external power will cause the actuator to scram.

The scram spring has a preload of 215 pounds and a compressed load of 460 pounds which gives it the capability of driving a 100-pound rod load four inches in 100 milliseconds against 250 psi water pressure. It also has the capability of holding the control rods in the core in the event of capsizing.

Fig. 2.9 - 630A Shim-Scram Rod Actuator Assembly (849D323)

2.6.2 Dynamic Rod Actuator

The dynamic rod actuator, shown in Figure 2.10, consists of an electric-motor-driven, rotary-output powerhead, a ball lead screw, a drive rod, and a housing assembly. These actuators move a small amount of control very rapidly in the reactor core to provide good transient response and accurate power control. They operate, within their velocity limits, at a velocity proportional to flux error. They remain approximately centered in the reactor core during steady state operation; thus, they are in a position to provide regulation of power in either an increasing or decreasing direction.

The rotary-output powerhead, shown in Figure 2.11, consists of a two-phase 115-volt, 60-cycle, servomotor capable of a minimum of 10 watts of mechanical power which drives through a 5.03-to-1 gear reduction to the output shaft. A conductive plastic, position-indicating potentiometer is driven by the output shaft through gear reduction. The potentiometer turns 0.836 revolutions for 30 inches of actuator travel. The rotary actuator also contains a 28-volt d-c brake which is driven by the motor and holds the actuator in a fixed position whenever the 28 volts are not applied. This provides a lock to hold the actuator which is about equal to 100 pounds at the output shaft. During normal operations the d-c brake is energized and, therefore, out of the system. This brake is controlled by circuits external to the actuator.

In the actuator, movement of the output shaft is caused by turning the ball lead screw (with the rotary actuator) and restraining the ball nut from rotating, thus causing the ball nut to advance on the screw. The ball nut is restrained from rotating by three ball bushings housed in the ball nut assembly which slides on three guide rods and extend the entire length of the actuator.

At the ends of travel the actuator is driven into springs on the guide rod which act as buffers, thereby bringing the actuator to a smooth stop. At the end of travel, the servomotor is stalled and applies full torque to the ball lead screw.

The operating characteristics of the dynamic rod actuator are presented in Table 2.4

Table 2.4

OPERATING CHARACTERISTICS OF DYNAMIC ROD ACTUATORS

<u>Operating Condition</u>	<u>Motor Speed, rpm</u>	<u>Lead Screw Speed, rpm</u>	<u>Control Rod Speed in/sec</u>	<u>Force Output, lb.</u>
Free running	3400	675	5.6	0
Normal load	3035	600	5.0	10
Stalled	0	0	0	54

Fig. 2.10 - 630A dynamic rod actuator assembly (219R930)

Fig. 2.11 - 630A dynamic rod actuator powerhead assembly (692E628)

3. 630A MARK V(B) REACTOR - SHIELD-PLUG ASSEMBLY

3.1 DESCRIPTION AND FUNCTION

The 630A Mark V(B) reactor configuration employs the use of low enrichment fuel rods in a tube-type reactor.

The reactor, shield plug, reactor control elements and reactor instrumentation are assembled into an integral unit called the reactor-shield plug assembly. Figure 1.2 shows the reactor-shield plug assembly installed in the 630A Mark V(B) Maritime Nuclear Steam Generator. The reactor-shield plug assembly is delineated in Figure 3.1.

The reactor-shield plug assembly is removed as an integral unit for the refueling operation and replaced with a spare unit. This refueling concept minimizes the refueling time and cost because the spare unit can be loaded with fuel and thoroughly checked out at a dockside facility prior to installation.

Reactor - The helium-cooled, light-water-moderated reactor utilizes rod type fuel elements in the core. The normal design power is 60.4 thermal megawatts.

The active core of the reactor is formed by 216 fuel elements, each containing 27 fuel rods, dispersed around and supported from the 91 moderator tubes that extend downward from the shield plug. The moderator tubes are arranged in a hexagonal pattern and the fuel elements are located around each moderator tube in a hexagonal pattern as shown in Figure 3.1.

The active core is surrounded by a reflector and a thermal shield. The reflector is split into two sections, an inner reflector and an outer reflector. Splitting the reflector into two sections permits a smaller opening in the top of the pressure vessel and permits a more flexible reflector design in that different materials can be used. Beryllium oxide is used adjacent to the core because of its effectiveness as a reflector and its high temperature capability in either helium or, if required, air. Graphite, a less effective, less expensive material than BeO is used for the outer reflector. Since circulator discharge gas at 553°F is used to cool the graphite, satisfactory operation with air for limited periods is acceptable. The inner reflector is a part of the reactor-shield plug assembly and consists of Beryllium oxide supported within a structural framework. The Beryllium oxide is irregularly contoured to conform to the outer row of fuel elements in the hexagonal pattern. The average thickness of the beryllium oxide is 4.81 inches, its length is 56.75 inches. The structural framework consists of a top and bottom plate, with the same contour as the Beryllium oxide, connected by twelve 2 1/4 x 2 1/4 x 1/4 structural "T" sections. A .02 inch thick Incoloy 800 cylinder which is the length of the reflector is welded to the outside diameter of the top and bottom and plates to assure that the helium flow does not by-pass the core.

Fig. 3.1 - 630A Mark V(B) Reactor-Shield Plug Assembly (221R466)

The outer reflector consists of 8 inches of graphite, pierced with cooling holes, supported within a structural framework from the pressure vessel. The equivalent thickness of graphite is 6 inches. This reflector is split into 4 sections in order that it can be inserted into or removed from the pressure vessel from the top.

Four 2 1/2-inch-thick steel cylinders comprise the thermal shield. These cylinders are supported by the pressure vessel and reduce the integrated fast neutron dose (with energies greater than 1 MEV) to 10^{18} nvt at the inside surface of the pressure vessel in addition to reducing the heating rate in the pressure vessel.

The reactor inlet flow plenum located between the reactor and shield plug is formed by two perforated plates joined by tubes. This structure distributes the helium into the reactor. In addition, it insulates the bottom surface of the shield plug and the moderator tubes in the plenum region by providing static helium gaps in these areas. The fuel elements seal in the bottom perforated plate to insure the helium flow does not by-pass the core. A perspective view of the inlet flow plenum and the top of the reactor is shown in Figure 3.2.

Shield Plug - The shield plug supports the reactor and provides shielding in the area directly above the reactor. It also serves as a moderator water inlet and outlet plenum, provides a mounting surface for the control tube actuators, houses the nuclear sensors, and serves as the top head of the pressure vessel to which it is gasketed and bolted. The present configuration of the shield plug is shown in Figure 3.1. The shield plug is divided into an upper plenum and a lower plenum. The lower plenum serves as the outlet water plenum and contains a lead slab clad with Incoloy 800 as shielding. The upper section serves as the water inlet plenum and also provides space for the movement of the actuator gang plates. The contained moderator water serves as a neutron shield and the 7-inch-thick lead slab is the primary gamma shield. The plate which divides the shield plug into two sections provides the attachment surface for the moderator water inlet tubes. The flange ring is located approximately at the mid-point of the shield plug; hence, when installed in the pressure vessel, the lower portion fits within the pressure vessel while the upper section extends above it.

The upper section is formed by a cylindrical shell, which is welded to the flange ring, an upper flange ring and an actuator mounting plate. The actuator mounting plate is gasketed and bolted to the upper flange ring. The plate which divides the shield plug into two sections is bolted and gasketed to the flange ring. The control rod actuators are mounted on the mounting plate and the extension rods pass through the mounting plate and are connected to the actuator gang plates to which the driven control tubes are fastened. The control tubes extend down into the water inlet tube. The manual shim brackets are also mounted on the actuator mounting plate. The manual shim tube and rod are supported from the brackets and also extend down into the water inlet tube. The water inlet tubes are mechanically fastened to the divider plate. The control elements, actuated shim, dynamic and safety tubes, and the manual shim rod and tube, are described in Section 3.5.

Fig. 3.2 - Perspective view of inlet flow plenum and top of V(B) reactor.

The lower chamber of the shield plug is formed by a cylindrical shell and the lower flat head and contains a 7-inch-thick lead slab enclosed within and bonded to a 1-inch-thick Incoloy 800 plate on the periphery and on the faces. Passages through the slab for the moderator water inlet tubes are provided by thin-wall Incoloy 800 tubes which pass through the head and are welded to the top and bottom plates. The slab assembly is simply supported around the entire periphery. Radial alignment of the slab is provided by four keys which also permit differential thermal expansion in the radial direction. Four bolts at the key locations limit the vertical motion of the assembly. The bottom flat head is perforated to accommodate the 91 moderator tubes that extend into the reactor. The Zircaloy-2 moderator tubes are gasketed and bolted to the inside surface of the bottom head.

The moderator water flows into the upper chamber of the shield plug. The water flows downward through the annuli between the control elements within the water inlet tube to the bottom of the reactor and then upward through the annulus between the outside diameter of the water inlet tube and the inside diameter of the moderator tube to the lower chamber of the shield plug. It continues upward through the lead slab through the discharge pipe into the moderator heat exchanger. The moderator water is cooled in the heat exchanger by the boiler feedwater. Water leaving the heat exchanger enters one of the circulating pumps (the other is on standby) where it is pumped into the upper plenum of the shield plug. The moderator circuit is described in detail in Section 8.2.1.

Reactor Instrumentation - There are seven nuclear sensors to measure the neutron flux in the reactor. These sensors are located within the lower chamber of the shield plug just above the bottom head. They are supported from the actuator mounting plate.

The exit air thermocouples, located in the instrumentation harness directly below the reactor, measure exit gas temperatures at typical cell locations. The harness consists of a framework that supports the thermocouples and three tubular ducts which house the thermocouple leads. The three ducts pass through penetrations in the shield plug flange ring. The harness is supported from this flange ring. Since the mechanical connections and electrical disconnects are located outside the shield plug, remote handling is required only for inserting and withdrawing the harness into and from the reactor-shield plug assembly.

3.1.1 Design Requirements

The reactor-shield plug assembly is designed to comply with the codes listed in Section 1.5. The shield plug is designed for an external pressure of 900 psig and an internal pressure of 258 psig at a maximum operating temperature of 450°F. Requirements with respect to corrosion, erosion, radiation effects, instability of materials, mechanical shock, and vibratory loading, will be established and incorporated into the design criteria to insure integrity throughout the 20-year design life of the vessels and structure. The materials, methods of construction, and inspection requirements will be carefully selected to minimize these effects.

3.1.2 Materials and Fabrication

The bottom head and cylinder of the shield plug are Incoloy 800. The moderator tubes and water inlet tubes are Zircaloy-2. The remainder of the shield plug structure is SA212B clad with Incoloy on those surfaces exposed to the moderator water. The inner reflector material is Beryllium oxide. The inner reflector structure is stainless steel. The outer reflector is graphite with stainless steel as the structural material. The thermal shields will be borated steel. The inlet plenum is stainless steel.

Zircaloy-2 was chosen for the moderator tubes because of its low absorption cross section.

The reactor and shield plug assembly will be fabricated and tested in accordance with the requirements of the U. S. Coast Guard and the ASME Boiler and Pressure Vessel Code.

3.2 FUEL ELEMENTS

3.2.1 Description and Function

The Mark V(B) Fuel Element is shown in Figure 3.3. Each fuel element consists of 27 cylindrical fuel rods, front and rear support plates, a mid plane spacer and a thin wall container around the bundle. Each fuel rod is 0.357 inch outside diameter. The rods are arranged in a hexagonal array on a pitch of 0.433 inch. Each rod has a fueled length of 42.0 inches. A fastening device at the rear of the reactor core supports the fuel elements from the moderator support tube. The average free flow area per cartridge is 1.933 square inches. There are 5832 total fuel rods in the core with a total U^{235} inventory of 393 pounds at 6 percent enrichment.

The fuel rods are positioned and supported by the forward support plate. The rear plate provides position only; the rods are free to move for thermal expansion in an axial direction. The mid plane spacer, located at the center of the fuel rod, provides additional support against the possibility of vibration. The 0.020 inch thick Incoloy-800 or stainless steel shell around each fuel rod bundle provides a definite flow path for the primary coolant. A seal located at the forward end insures the coolant flow through the fuel rod bundle. A static helium gap, which acts as an insulator, is maintained between the shell and the moderator support tube.

The cylindrical fuel rods are conventional in design concept. They are similar in configuration to the fuel rods currently in use in many large nuclear power generating stations. Additional design features have been incorporated which are unique to high temperature gas cooled and superheat fuel rods.

A typical fuel rod design is shown in Figure 3.4. This design concept utilizes a thin wall (0.015) Incoloy-800 clad with a pellet type UO_2 fuel. Stainless steel is also being evaluated as a possible clad material. In order to maintain neutron economy, the wall thickness of the clad tube has been reduced to a minimum consistent with current technology. Thus, the clad is not self-supporting under long

Fig. 3.3 - Mark V(B) Fuel Element (692E667)

- NOTES:
3. EVACUATE AND BACK FILL WITH HELIUM TO 1 ATMOSPHERE
 2. PELLET DENSITY 91 % OF THEORETICAL.
 1. CLAD TO BE SWAGED TO FUEL AFTER ASSEMBLY.

Fig. 3.4 - Typical fuel rod design (139B9815)

term creep against external primary coolant pressure at reactor operating temperature. Under these conditions, the clad receives support from the fuel matrix. To insure uniform closure of the assembly clearance between the clad and pellets, the clad is collapsed under controlled conditions during the fabrication process. This cladding concept is known as a "non-freestanding" or "collapsible clad" design. This concept is currently being used in the following: Experimental Gas Cooled Reactor (EGCR), Mobile Low Power Reactor (ML-1), Advanced Gas Cooled Reactor (AGR), and the Boiling Nuclear Superheat Reactor (BONUS). A considerable research program is being conducted by the General Electric Company at the Vallecitos Atomic Laboratory to investigate various materials and the minimum clad thickness for high temperature superheat fuel rods. A good portion of this technology is directly applicable.

The matrix of the fuel rod is 6 percent enriched UO_2 in the form of pellets, pressed and sintered to 91 percent of theoretical density. A vibration compacted power is also being considered. The fuel rod design is identical regardless of whether pellets or power is used. The choice is mainly economic. However, from the test work at ORNL⁽²⁾ it appears that the vibration compacted powder may offer some design advantages. At high temperatures, the clad formed circumferential wrinkles at the interface between pellets. The "dished pellet" design was intended to relieve this situation. The use of powder may completely eliminate the interaction. A typical UO_2 pellet concept is shown in Figure 3.4. The diametral clearance between the cladding and the pellets is a minimum of 0.002 inch and a maximum of 0.006 inch. Studies at Oak Ridge National Laboratory⁽¹⁾ indicate a minimum clearance of 0.002 inch is required for each assembly of the pellets into the tube. The end of the pellet is dished to minimize the effect of thermal expansion and interaction between the fuel pellets and clad,⁽²⁾ which could lead to regions of high local plastic strain.

The plenum void region at the forward end of the fuel rod serves to collect the fission product gases. The plenum has been designed to accommodate 100 percent of the release of fission product gases using standard techniques.^(3,4) Recent data⁽⁵⁾ indicate this to be unduly conservative and the plenum length may be reduced as additional data become available. The plenum support tube provides support for the clad in the plenum region.

The fuel rod is evacuated and filled with helium to a pressure of one atmosphere to insure a good thermal bond between the pellets and the clad.

Alternate Rod Configuration - An alternate configuration under investigation is shown in Figure 3.5. The objective of this design was to achieve a freestanding clad whose thickness is minimized through control of the pressure differential across the cladding. The design does not depend upon the fuel matrix for support at anytime. There are indications that this design may extend the fuel element lifetime by minimizing the plastic strain

(1) Martin, W. and Weir, J., ORNL 3103, August 1961.

(2) Samuels, G., Nuclear Science and Engineering, Vol. 14 : 37-52 (1962)..

(3) Baroch, C., GEAP 4257, May 1963.

(4) Baroch, C., et al., GEAP 3653, March 1963.

(5) Pashds, T., and Naymark, S., APED 4265, October 1963.

Fig. 3.5 - Alternate fuel rod configuration (139B9813)

deformation in the fuel rod clad.

The concept uses a standard fuel rod that is internally pressurized during rod fabrication to a precalculated value established by the system operating pressure and the fission gas buildup. The fission gas plenum can be designed to minimize the increase in internal pressure due to fission gas buildup. The total internal pressure at operating conditions is designed to be approximately equal to the system operating pressure at the end of life.

3.2.2 Design Requirements

The cylindrical fuel rods are designed for a minimum burnup of 20,000 MWD/TU. The fission gas plenum will be capable of accommodating 100 percent release of the fission products. The total pressure from accumulated gases (helium, moisture, fission products) in the fuel rod at the end of life should be less than the normal system working pressure. The fission gas will be retained in the fuel element plenum throughout the fuel rod lifetime without the pressure exceeding the short time rupture strength of the clad.

The fuel rod must operate satisfactory in helium and for short periods of time in air and steam environment.

The clad will be designed to prevent wrinkle formation which could lead to shorter lifetime. The allowable plastic strain of the clad per temperature cycle will be established by a low cycle fatigue analysis.

3.2.3 Fuel Element Materials

It is anticipated that, initially, preparation of 630A fuel pins will follow more or less standard practices. Pellets of fuel will be cold pressed and sintered to approximately 95 percent of theoretical density and then encapsulated in tubular cladding. The cladding will be brought into contact with the pellets under controlled conditions. Other methods of processing fuel pins, such as vibratory packing of fuel powder in the cladding tube, probably followed by cold or hot swaging, will receive continuing attention and may ultimately be adopted.

The fuel material to be employed, at least in initial cores, is ceramic grade UO_2 .

An evaluation of the applicable data makes possible a tentative choice of potentially suitable cladding alloys. The ultimate choice for use in initial cores will probably be AISI Type 304 if further study indicates the chloride contamination in the primary loop can be controlled. If it cannot, depending upon the contamination level, Incoloy 800 or Inconel 600 will be the choice. The nuclear advantage associated with aluminum-bearing ferritic steels is tempting, and as additional pertinent data on them become available, selection of an alloy of this type will be considered.

3.2.4 In Pile Test Results of Related Designs

A considerable amount of related in-pile test data has been accumulated on fuel rods of similar design and operating environment. Although these data are not an

exact duplication of the proposed Mark V conditions, a close relationship exists, and trends may be indicated.

Mobile Low Power Reactor (ML-1) - An extensive in-pile test program was conducted and reported⁽¹⁾ for the ML-1 fuel rod design. The basic rod configuration is composed of enriched UO_2 fuel pellets with a 0.030 inch thick Hastelloy X clad. In-pile tests demonstrated the mechanical and structural integrity of the fuel clad for 10,000 hours with maximum hot spot temperatures of 1750°F. Several tests were conducted at a maximum average clad temperature of 1550°F with a maximum clad temperature of 1700°F. The results indicated negligible dimensional changes and again demonstrated the soundness of the design. The proposed 630A maximum average clad temperature is 1335°F. Thus, ML-1 tests were more severe than one would anticipate for the Mark V.

Experimental Gas Cooled Reactor (EGCR) - An irradiation program was initiated in the fall of 1958 for evaluating the EGCR fuel rod materials and design⁽²⁾. The EGCR fuel element configuration is a hollow cylinder clad on the outside with 0.020 inch of 304 stainless steel. Irradiation tests were designed to simulate the significant fuel element operating conditions⁽³⁾. Most of the experiments were conducted at cladding surface temperatures of 1300°F, but a substantial portion were conducted at temperatures of 1600°F. This simulated the EGCR "hot channel" and "hot spot factors". An external pressure of 315 psia was present during irradiation. Thermal cycling was specified for some of the ORR experiments to simulate power variations anticipated between startup, operation and shutdown. Specific burnups ranged from 2,000 to 13,000 MWD/MT UO_2 . The surface heat flux values ranged between 50,000 - 150,000 BTU/hr. ft^2 .

Only small dimensional changes were reported⁽⁴⁾ for the prototype fuel element tests. Some bowing was observed but generally limited to 0.025 inch or less. Some areas in the clad contained circumferential ridges at the cladding pellet interfaces. (The dished pellet design was intended to alleviate this problem). These pellets were run with clad temperatures between 1500°F - 1600°F. Post irradiation examination had shown the pellets to be cracked from thermal stress but had not powdered. Some of the long term burnup samples reported⁽⁵⁾ clad failures due to excess power levels or high cladding temperatures, well above design calculations.

These experiments confirmed the EGCR design philosophy, and demonstrated a successful fuel element rod design. The conditions are very similar to the Mark V design conditions except the EGCR pellet size is larger (0.500 inch OD vs 0.320 inch OD).

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- (1) G. L. Vivian, "A Summary Report on ML-1 Fuel Element Development In-Pile Test History", IDO-10040, June 1964.
 - (2) Nuclear Science and Engineering.
 - (3) Nuclear Science and Engineering.
 - (4) Nuclear Science and Engineering
 - (5) ORNL TM 832, "Performance of EGCR Fuel at Extreme Operating Conditions", May 1964.

3.3 MARK V(B) REACTOR THERMODYNAMIC DESIGN

3.3.1 Operating Conditions and Performance

The 630A reactor is sized to produce 27,300 SHP plus power for ship and reactor auxiliary requirements. The reactor preliminary design is based on a total power of 60.4 megawatts. About 93 percent of this power is transferred to the primary coolant. The balance is transferred by thermal and/or nuclear heating to the moderator water, the biological shield, and various parts of the nuclear steam generator structure.

As previously discussed, 630A operations will be performed with a reactor exit helium temperature of 1200°F and an operating pressure of 830 psia. Table 3.1 presents a summary for 1200°F temperature operation for the rated power condition of 27,300 SHP. The Mark V(B) tube type reactor has 216 fuel elements with 27 rods in each element. The performance shown here is based on fuel rods having no turbulence promoters. A parametric study to determine the effects of turbulence promoters is shown in Section 3.3.6.

TABLE 3.1

630A OPERATING CONDITIONS WITH AN EXIT HELIUM
TEMPERATURE OF 1200°F

Plant net output, SHP	27,300
Helium flow, lb/sec	66.2
Reactor inlet temperature, °F	553
Max. average fuel temperature, °F	1335
Reactor inlet pressure, psia	828
Reactor pressure loss, psi	2.08
Maximum dynamic head, psi	0.31
Maximum Mach number	0.021
Reactor power, MW	60.4

At rated power, helium enters the active core at a dynamic head of 0.186 psi and a Mach number of 0.016. As the helium is heated while passing through the core, the dynamic head and Mach number increase to 0.306 psi and 0.021, respectively. On the average, the forced convection heat-transfer coefficient is about 435 BTU/hr-ft²-°F. The average heat flux is 102,600 BTU/hr-ft². The pressure loss of about 0.25 percent is calculated using the conventional compressible flow relationships described below.

Fuel rod average surface temperatures vary from 700°F at the inlet to a maximum value of 1335°F at 0.95 fractional reactor length. A plot of the average fuel rod temperature, the bulk helium temperature and other operating conditions as a function of reactor fractional length are given in Figures 3.6 and 3.7. The maximum average surface temperature (1335°F) is defined as the temperature that would exist at the longitudinal position of maximum surface temperature with a 2/1 sine longitudinal power distribution if all local heat fluxes and mass velocities were equal

REACTOR OPERATING CONDITIONS
630A MARK V(B)
HEAT TRANSFER CHARACTERISTICS
FIGURE 3.6

FIGURE 3.7

REACTOR OPERATING CONDITIONS

630A MARK V(B)

FLOW CHARACTERISTICS

- NOTES:
1. 216 cans, 27 rods/can
 2. 417.7 in² A_{eff}
 3. 0.2015 in. D_h
 4. 66.2 lb/sec Helium Flow
 5. 2/1 Sine Power Profile
 6. 553°F Inlet He. Temp.
 7. 828 psia inlet pressure
 8. Overall loss in total pressure, 2.08 psia
 9. 60.4 MW total reactor power

to an average value. While the 630A reactor is designed to approach these conditions, certain perturbations will exist. To obtain the hot-spot temperature, these perturbations must be added to the average-maximum temperature. See Section 3.3.4 for a discussion of temperature perturbations. Fuel rod central temperatures are discussed in Section 3.3.3.

The moderator circuit will be designed to prevent boiling of the moderator water. The pressure and flow will be maintained at 250 psia and 1000 gpm, respectively. At these conditions, the water temperature will be 210°F at the inlet and 240°F at the exit. Since the saturation temperature is 401°F, and the film temperature rise is less than 25°F, bulk boiling will be prevented.

3.3.2 Analytical Expressions

Reactor helium pressure loss is predicted by integration of the expression

$$dP_o = \frac{-kM^2 P_o}{2} \left[\frac{dT_o}{T_o} + 4f \frac{dx}{D} \right]$$

where

P_o = initial total pressure, psia
 k = specific heat ratio
 M = average Mach number
 T_o = initial total temperature, °R
 f = friction factor
 dx = length in flow direction
 D = hydraulic diameter

The friction factor is equal to $(0.046) Re^{-0.2}$ where Re is Reynolds Number and the friction factor multiplier applied for surface effects and discontinuities in the flow path with no turbulence promoters has been assumed equal to 1.0.

Bulk helium temperatures are calculated on the basis of average power for finite increments of fuel rod length. Heat transfer coefficients and fuel rod clad temperatures are predicted by use of the two following equations:

$$h = \frac{(c)(k)}{D_h} (0.023) Re^{0.8} Pr^{0.33}$$

$$\Delta T_f = \frac{Q \left(\frac{P_x}{P_{avg}} \right)}{A h}$$

where

h = heat transfer coefficient, BTU/hr-ft²-°F
 Pr = Prandtl Number (based on film temperature)
 Re = Reynolds Number (based on film temperature)
 D_h = hydraulic diameter, ft (based on wetted perimeter)
 k = thermal conductivity (based on film temperature)
 Q/A = average heat flux, BTU/hr-ft²
 P_x/P_{avg} = ratio of the local power to the average power
 c = heat transfer coefficient multiplier, $c = 1.0$ with no turbulence promoters.

For heat transfer to water in turbulent flow, as in the moderator loop, the following expression is used to determine the heat transfer coefficient:

$$h = \frac{C_p G (0.023) \left[1 + \left(\frac{D}{L} \right)^{0.7} \right]}{\left(\frac{\mu C_p}{k} \right)_b^{0.667} \left(\frac{\mu_w}{\mu_b} \right)^{0.14} \left(\frac{DG}{\mu_b} \right)^{0.2}}$$

where

- h = heat transfer coefficient, BTU/hr-ft²-°F
C_p = specific heat, BTU/lb-°F
G = mass velocity, lb/hr-ft²
D = equivalent diameter, ft
L = length, ft
μ = dynamic viscosity, lb/hr-ft
k = thermal conductivity, BTU/hr-ft-°F
b = subscript indicating property values evaluated at bulk temperature
w = subscript indicating property values evaluated at wall temperature

3.3.3 Fuel Element Steady State Analyses

The 630A Mark V(B) reactor heat deposition for an average condition over the life of the core is shown in Table 3.2. At normal power the average heating rate for unfueled components within the active core is about 1.2 x 10⁻² watts/gm-MW. On this basis the Transient Heat Transfer Program (THTC) was utilized to predict steady state, rated power operating temperatures of fuel rods, fuel can and moderator tube for a 30 degree sector of a fuel element. Heat losses to the moderator tube by radiation and conduction from the fuel can were also calculated. About 7 percent of reactor total power is delivered to the moderator including approximately three percent transferred by radiation and conduction from the fuel cans. Thus, about 93 percent of reactor power is transferred to the primary coolant.

TABLE 3.2
REACTOR HEAT DEPOSITION

<u>Component</u>	<u>Percent</u>
Fuel rods	94.75
Moderator tubes and control elements	1.53
Fuel element can	0.15
Core water	1.62
Shield plug	0.47
Boiler	0.48
Side reflectors (helium cooled)	1.00
	<u>100.00</u>

Fuel rod surface and central temperatures for both 2/1 sine and "rods in" power profiles are shown in Figure 3.8. These temperatures are based on average flow and power conditions. The average maximum surface temperature for a 2/1 sine power profile and for a "rods in" power profile is 1335°F and 1395°F, respectively and occurs near the end of the active core. For the 2/1 sine profile and "rods in" profile the average maximum rod central temperature is 1760°F and 1925°F, respectively and occurs at about 0.7 core fractional length.

In order to obtain estimates of the effects of gross and fine radial power profiles and flow distribution within an element the General Flow Passage Program was used to calculate bulk coolant and rod surface temperatures in a centrally located fuel element. Thus, these calculations are based on a gross radial power 5 percent higher than average and a "rods in" axial power profile with a peak to average value of 1.45. The fuel element was arbitrarily divided into 3 regions as shown in Figure 3.9. The assumed fine radial power for regions 1, 2 and 3 was 0.85, 0.90 and 1.09 respectively, relative to average power. A power scallop of 1.3 was applied at the outer edge of rods in region 3. Calculations were made to determine the relative flow in each region based on the same coolant pressure drop through each region and the particular hydraulic diameter and free flow area for each region. The stated fine radial power distribution is based on 6 percent fuel enrichment.

The axial temperature profiles for the coolant and the corresponding surface temperature profiles are shown in Figure 3.10 for each region. Relative mass flows for regions 1, 2 and 3 are 0.102, 0.299 and 0.599 respectively. The maximum surface temperature near the hot streak is about 1530°F and is used as the basis for preliminary estimates of hot spot temperatures in the following section.

3.3.4 Temperature Perturbations

The average maximum fuel rod clad temperature was previously shown to be about 1335°F (Section 3.3.1) for the Mark V(B) reactor operating at rated power of 60.4 MW. This temperature is based on the assumption of average coolant flow and average power for a rod having nominal dimensions. The longitudinal power profile is assumed to be a 2/1 sine curve.

Two groups of temperature perturbations effecting clad temperature have been considered. The first group includes the effects of nominal power variations in the longitudinal profile, gross radial profile across the core, fine radial profile across an element and circumferential scallop around a rod. The second group of temperature perturbations includes the effects of variations from nominal power profiles, coolant flow, dimensions and local heat generation. The combined effect of these perturbations produces a calculated hot spot temperature of about 1610°F for the fuel rod clad.

The perturbed temperature based on the effects of power profile variations is estimated to be about 1530°F (Section 3.3.3). This preliminary estimate is based on a longitudinal profile peaking factor of 1.45, a gross radial factor of 1.05, a fine radial factor of 1.09 and an outer rod circumferential scallop factor of 1.3. The effects of coolant flow distribution within a fuel element are included. The result is conservative because temperature alleviation by conduction circumferentially through the rod and clad and coolant mixing effects are not included.

FIGURE 3.8

630A MARK V(B)

Fuel Rod Surface and Central Temperatures

FIGURE 3.9
630A MARK V(B) STEADY STATE OPERATION
AT 60.4 MW

NOTES:

- (1) Regions 1, 2 and 3 are delineated on the figure.
- (2) A centrally located fuel element is considered ($1.05 P_{\ell}/P_{avg}$).
- (3) The "rods in" axial power profile is used (1.45 Peaking factor)
- (4) A power scallop of $1.3 P_{\ell}/P_{avg}$ is applied to the outside of rods in region 3.
- (5) Temperatures for region 3 are hot streak temperatures.

Region	No. of Rods	P_{ℓ}/P_{avg}	Relative Flow	Exit Helium Temp., °F	Exit Surface Temp., °F
1	3	0.85	.102	1203	1378
2	9	0.90	.299	1255	1444
3	15	1.09	.599	1248	1530

FIGURE 3.10

To obtain the hot spot temperature the combined effects of variations from nominal conditions are added to the calculated perturbed temperature of 1530°F . Random variables with normal distribution are assumed. The combined effects were obtained by a statistical summation of the individual deviations of bulk coolant temperature and film temperature drop at the hot spot. This hot spot temperature deviation is estimated to be $\pm 80^{\circ}\text{F}$ and the hot spot temperature is consequently about 1610°F .

Table 3.3 shows the various causes of hot spot deviation, the 3 sigma temperature deviations on coolant temperature rise and film temperature drop and the total hot spot deviation obtained by calculating the square root of the sum of the squares of individual surface temperature deviations. The significance of this calculation is that for the assumed confidence limit of 99.87 percent (the hot spot temperature will not be exceeded in 1.3 chances out of 1000) the frequency of realizing temperatures exceeding the calculated hot spot can be estimated. Of the 216 fuel elements in a Mark V(B) reactor only about 10 percent of them experience the 5 percent gross radial power perturbation. Only 15 rods in each element experience the most severe circumferential scallop. Thus, with 330 rods in each reactor operating near the maximum power level only 1 rod in every 2.3 reactors is expected to operate at a temperature in excess of the calculated hot spot temperature. However, exceeding the hot spot temperature does not automatically result in failure of the fuel rod.

The 80°F temperature deviation is a 3 sigma value. The hot spot temperature using a 4 sigma deviation is about 1640°F and this hot spot temperature would be exceeded only once in 31,500 chances or once in every 95 reactors.

The average maximum rod central temperature was shown to be about 1760°F based on a 2/1 sine power profile and about 1925°F for a "rods in" power profile (Section 3.3.3). Preliminary estimates, allowing for the gross radial profile, fine radial profile and thermal conductivity variation effects, indicate the maximum perturbed fuel rod central temperature at rated power will be about 2300°F .

It is recognized that at the present stage of the 630A design there may be larger uncertainties in power profiles than those assumed here. However, several conservative assumptions for other factors of the hot spot calculation tend to make these predictions realistic. It is essential, however, that a hot spot temperature be determined with the best input available to establish reasonable mechanical design limits.

3.3.5 Fuel Element Transient Analyses

Normal Aftercooling - In the event of a scheduled reactor shutdown, no fuel rod overtemperature occurs. Primary coolant is circulated by both circulators and the afterheat generated is transferred by the primary coolant to the boiler and secondary loop working fluid.

Emergency Aftercooling - The most severe postulated aftercooling condition has been analyzed with use of the Transient Heat Transfer Program to determine fuel rod, clad, fuel can and moderator tube temperatures in the event of multiple component failures. Specifically, complete and instantaneous loss of primary coolant flow was assumed to occur during full power operation. That is, both circulators become inoperative (failure of 2 circulators or their drives or failure of all turbine and diesel generators) with no coast-down flow and no natural circulation through the core. Afterheat is stored in the fuel elements and also transferred by radiation and conduction to the

TABLE 3.3

TEMPERATURE DEVIATIONS
630A MARK V(B)

Factor	3 Sigma Deviation	Coolant Temperature Rise, °F	Film Temperature Rise, °F	Surface Temperature Rise, °F	Surface Temperature Variance, °F
Flow Maldistribution	$\pm 1\%$	6.95	2.3	9.25	85.6
Inlet Temperature	$\pm 10^{\circ}\text{F}$	10.0	---	10.0	100.0
Axial Power	$\pm 1\%$	---	2.8	2.8	7.8
Gross Radial Power	$\pm 1\%$	6.95	2.83	9.78	95.6
Fine Radial Power	$\pm 2\%$	13.9	5.66	19.56	382.6
Fuel Density	$\pm 2\%$	13.9	5.66	19.56	382.6
Fuel Enrichment	$\pm 0.07\%$	7.6	3.11	10.71	114.5
Rod Pitch	± 0.003 in.	43.1	18.1	61.2	3745.0
Rod Diameter	± 0.0015 in.	22.6	9.28	31.88	1016.0
Manufacturing Bow	± 0.01 in.	---	10	10	100
Thermal Bow	± 0.015 in.	---	15	15	225

$$\Sigma \text{ Variance} = 6254.7^{\circ}\text{F}$$

$$3\sigma \text{ Deviation} = 79.1^{\circ}\text{F}$$

$$\text{Surface Hot Spot Temperature } (1530 + 80) = 1610^{\circ}\text{F}$$

Notes:

- (1) The hot spot location is near $X/L = 1.0$
- (2) Uncertainty in heat transfer coefficient is omitted because the correlation used in the prediction of average maximum temperature is about 20 percent conservative.
- (3) Maximum surface temperature at the hot spot is 1530°F .
- (4) Film temperature rise at the hot spot is 282°F .
- (5) Hot streak coolant temperature rise to the hot spot is 695°F .

moderator. The assumed scram delay time is 0.1 seconds. For conservatism, a flat fine radial power distribution for the rods in an average fuel element was assumed.

For this hypothetical accident, with the stated assumptions, it was found that the maximum clad temperature of the hottest fuel rod for the Mark V(B) is ~~20~~ 2090 °F and is reached 150 seconds after the incident. The maximum rod central temperature is 2100 °F. The afterheat generation used for the analyses is shown in Figure 3.11. The geometry and significant temperatures for the Mark V(B) configuration are shown in Figure 3.12. The temperature profiles for several of the nodes are shown in Figure 3.13. The effects of coastdown flow and radial variation in power from rod to rod, increasing with proximity of the moderator tube, would be to diminish the maximum temperatures predicted.

The node arrangement used for the THTC program analysis consists of 123 nodes as shown in Figure 3.12 for a 60 degree sector of the fuel element. To obtain accurate rod central temperatures considerable detail (16 nodes for one-half rod) was used for two of the rods whereas the remaining rods have 6 nodes per half rod. Gas nodes were provided around the fuel rods and between the fuel element can and the moderator tube. Normal moderator water flow was maintained in the moderator tubes.

To provide conservatism to the predicted temperatures a flat fine radial power profile was used. An average cell having no gross radial power perturbation was used as this gross radial effect is more than compensated for with use of a fine flat radial profile. (The center rod would have received about 10 percent less power if actual gross and fine radial profiles had been used.) The longitudinal position used for the calculations was 0.73 X/L, the expected location of peak temperature.

The radiation view factors between fuel rods were computed using the method presented in GAMD 3779⁽¹⁾ and applied to solid nodes representing the clad. The view factors between the fuel rods and surrounding fuel element can were determined by reciprocity relations. The emissivity factors used were the products of the radiating surface emissivities which were assumed constant and equal to 0.70 for all surfaces.

The thermal conductivity of the UO₂ matrix was used according to the equation $k = 68.4 T^{-.496}$ according to data in the ANS Transactions⁽²⁾.

By virtue of the conservative assumptions used in this loss of coolant flow analysis it is believed that the predicted temperatures are maximum values. More precise values can be obtained with use of finally established power profiles and the effects of uncertainties in thermal conductivity can be evaluated. It is emphasized that (1) although clad temperatures increase to

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- (1) J. A. Landoni, "View Factors for Heat Radiation in HTGR Core Configuration," GAMD 3379, December 10, 1962.
 - (2) ANS Transactions, Vol. 6, No. 1, Lyons, et al., June 1963.

Notes:

- (1) Letters indicate nodes whose temperatures are plotted on Figure 3.13.
- (2) Numbers are temperatures in $^{\circ}\text{F}$ for the nodes indicated at the time when the temperature of node a is a maximum (150 seconds).

Figure 3.12

630A Mark V(B) Fuel Element Temperatures
Loss of Coolant Flow Analysis

relatively high levels for this postulated accident, melting temperature (about 2550°F) is not reached and (2) the likelihood of occurrence of the postulated accident is extremely remote.

3.3.6 Alternate Configurations - Turbulence Promoters

Turbulence promoters can generally be used to obtain increased forced convection heat transfer coefficients from heat dissipating surfaces at the expense of increased pressure drop and pumping power. Data reported in the literature indicate that by keeping the turbulence promoters sufficiently small, to minimize form drag, high efficiency heat transfer surfaces can be obtained in configurations suitable for nuclear reactor fuel elements. The results of analysis of the effect of turbulence promoters on fuel element design variables are summarized on Figures 3.14, 3.15, 3.16 and 3.17. The figures show number of fuel rods, core free flow area, clad and turbulence promoter areas, fuel rod internal temperature rise and several other characteristics as a function of turbulence promoter pitch to height ratio for fixed conditions of core pressure drop, maximum average clad surface temperature and fuel loading.

The values shown are for a core pressure drop of 3.0 psi and maximum average clad surface temperatures of 1300°F and 1350°F based on a "2/1 sine" longitudinal power profile and Mark V operating conditions. The core configuration used for this parametric study was similar to but not precisely the same as the cores described in this report. A helically wrapped spacer wire of large pitch was assumed here in addition to the turbulence promoters but is not currently included in the design. The turbulence promoter configuration considered consists of a 10 mil diameter wire wrapped around the fuel rod. By varying the wire pitch to diameter ratio (L/D), different values of heat transfer and friction factor multipliers are obtained. The multipliers used in the analysis are based on experimental data reported in ASME Paper No. 63HT-1(1).

It may be noted that by use of suitable turbulence promoters the number of fuel rods required for the reference configuration could be reduced by as much as 40 percent. Additional evaluation of turbulence promoters is required to establish the magnitude of savings in fuel manufacturing costs and the effect on fuel element reliability resulting from their use.

3.3.7 Aerothermal Computer Programs

The two computer programs described below are programs that have been used in the aerothermal design of the 630A Mark V Nuclear Steam Generator. These programs can be used for any coolant gas without revision.

General Flow Passage Program (GE-ANP 663)⁽²⁾ computes the steady-state coolant and fuel surface temperatures as well as major flow parameters such as pressure loss, Reynolds number, Mach number, and dynamic head for a single flow passage of arbitrary cross section with a compressible gas coolant. The flow can be specified in a variety of ways, as by total flow plus inlet conditions

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- (1) N. Kattchee and W. V. Mackewicz, "Heat-Transfer and Fluid Friction Characteristics of Tube Clusters with Boundary Layer Turbulence Promoters, ASME Paper No. 63HT-1.
 - (2) Skirvin, S.C., "General Flow Passages; A Computer Program to Calculate Aero-thermal Performance of an Arbitrary Flow Passage of Fixed Geometry," GE-NMPO, APEX-664, October 1961.

Figure 3.11

The Effects of Turbulence Promoters

Notes:

- (1) Core pressure drop is 3.0 psi
- (2) Average maximum surface temperature is 1300°F

Figure 3.15

The Effects of Turbulence Promoters

Notes:

- (1) Core pressure drop is 3.0 psi
- (2) Average maximum surface temperature is 1300°F

Figure 3.16

The Effects of Turbulence Promoters

Notes:

- (1) Core pressure drop is 3.0 psi
- (2) Average maximum surface temperature is 1350°F

Figure 3.17

The Effects of Turbulence Promoters

Notes:

- (1) Core pressure drop is 3.0 psi
- (2) Average maximum surface temperature is 1350°F

or maximum surface temperatures plus inlet conditions. Both laminar and turbulent flow are treated together with entrance length effects. Input heat transfer relationships are conventional Nusselt number correlations in terms of exponentials of Prandtl number and Reynolds number. Input fluid flow friction factors consist of exponential functions of Reynolds number and appropriate friction factor multipliers.

Thermal properties of coolant gases are expressed as temperature-dependent polynomials. This program can carry out calculations for air, neon, helium, and hydrogen. Polynomial property functions for additional gases can be readily added to this program.

Transient Heat Transfer Programs - THTB and THTC (GE-ANP 662)⁽¹⁾ solve steady state and transient problems by computing nodal point temperatures for any stated geometry. The programs account for all three modes of heat transfer: conduction, convection, and radiation. In addition; provision is made to handle the isothermal energy absorption or release associated with phase change. Boundary conditions are defined as arbitrary functions of time and thermal property data are defined as arbitrary functions of temperature; all functions are expressed in tabular form. The method of problem formulation and solution is by finite differences, the latter employing the implicit form wherein the heat balance is made on the temperature distribution at the end of the time step. This is done because of the consequent freedom in choosing the length of the time steps for which the transient is run and the ability to proceed directly to a steady state solution at any point during the transient solution.

These two programs represent the third and fourth, respectively, in a series of transient heat transfer programs developed within the General Electric Company.

The basic method of carrying out the computations has remained essentially the same, but substantial improvements have been made in successive versions in terms of increases in the size of the problems which can be handled, in the speed and accuracy of solutions, and in the flexibility of input and output. THTC performs convective heat transfer coefficient calculations.

The THTB program does not perform any flow calculations. Consequently, coolant flows must be prescribed as functions of time and convective heat transfer coefficients must also be defined as functions of time. Only the THTB program is currently documented; the THTC program is undergoing active development to provide an increasingly powerful tool for the solution of thermal design problems encountered by the General Electric Nuclear Materials and Propulsion Operation.

(1) Stephens, G. L. and Campbell, D. J., "Program THTB for Analysis of General Heat Transfer Systems," GE-FPD, R6OFFD647, April 21, 1961.

3.4 REACTOR PHYSICS MARK V(B)

The core configuration of the Mark V(B) is unusual for gas cooled, liquid moderated reactors in that the fuel surrounds the moderator rather than moderator surrounding fuel as in the usual case, of which Mark V(A) is an example. This configuration thus allows some changes in philosophy for operation of the reactor for the required lifetime in that excess reactivity can be held in displacement of water moderator with a material of poor moderating characteristics relative to water. The use of water displacers, referred to as shims, results in an increase of mean energy of the neutron spectrum with resultant greater absorption in the fertile U-238 in the fuel. This in effect allows the excess reactivity to be held in conversion ratio of fertile to fissile material rather than in burnable poison which would be required were this feature not available. Also, since the water displacement, with resultant undermoderation of the core, is capable of changing reactivity by large amounts, the necessary control for the usual operating transients-moderator temperature coefficients, doppler temperature coefficients and burning of fissile material can be effected with low absorption cross section material such as Zircaloy. Since each moderator tube is shimmed individually, this also provides a method of gross radial power flattening for the core.

3.4.1 Analytical Technique

As mentioned previously, Section 2.4, the calculation of effective resonance integral for the cell geometry has given some problems. The approach which was finally taken to determine this value for the Mark V(B) geometry was to assume an annular ring of fuel surrounding the moderator tube. In a fueled ring with moderator both inside and outside, the effective resonance integral is given by an equation of the form⁽¹⁾

$$I_{\text{eff}} = a + b(S/M)^{1/2}$$

where S/M is the surface to mass ratio of the fueled piece, and for moderation both internally and externally,

$$S/M = (S_o + \gamma S_i)/M$$

where S_o is the outer surface area, S_i is the inner surface area, and γ is a weighting factor given by

$$\gamma = 2 \Sigma_1 r [1 - P_2(\Sigma_1 r)]$$

in which Σ_1 is the macroscopic scattering cross section of the moderator which fills the center region, r is the radius of the hole, and $P_2(\Sigma_1 r)$ is the probability that a neutron which suffered a collision in the hole will suffer a next collision in the hole prior to entering the fuel. If $\Sigma_1 r$ is large,

$$P_2 \approx 1 - S_w/4 \Sigma_1 V_w = \frac{1}{2r\Sigma_1}$$

where S_w and V_w are the surface and volume of the water region, respectively.

Therefore $\gamma = 1$. For purposes of initial calculations, it was assumed that the outer surface $S_o = 0$. These expressions were used to estimate the resonance integral for Mark V(B) geometry, and using the data of Hardy⁽²⁾ et al., I_{eff} was

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- (1) Weinberg, Alvin M. and Eugene P. Wigner, "The Physical Theory of Neutron Chain Reactors," University of Chicago Press, 1958.
 (2) Hardy, et al. loc. cit.

TABLE 3.4

BASE CELL PARAMETERS - MARK V(B)

Parameter	Hexagonal Array Across Flats Dimension	Equivalent Cylinder Radius in Centimeters
Cell pitch	11.612 cm	6.097
Fuel inside	7.823 cm	4.107
Core and Void inside		3.734
Moderator tube inside		3.219
Water return annulus inside		3.066
Water inlet - outside dimension		3.016
UO ₂ volume/cell	29.259 cc	
UO ₂ density	10 gm/cc	
H ₂ O/UO ₂ volume ratio-base cell	1.079	
No. fuel rods per cell	54	
Fuel rod O.D.	0.831 cm	
Fuel enrichment	5%	
Moderator tube	Zircaloy	
Water flow divider	Zircaloy	
Clad and can material	Incoloy-800	

calculated to be 13.83 barns. This corresponds to an effective potential scattering cross section of approximately 12 barns in Figure 2.7. The Dancoff correction for this lattice was calculated to be 0.0093 by the method of Velarde.⁽¹⁾ Table 3.4 lists other cell characteristics required for the calculations.

3.4.2 Reactivity and Control

Nuclear calculations have been carried out for single cell configurations of the Mark V(B) to demonstrate the effects of water displacement on reactivity and conversion ratio and to determine the temperature effects on reactivity. Calculations were carried out for 6 percent enriched fuel with an effective resonance integral of 20 barns, which corresponds to an equivalent potential scattering of 25.3 barns in Figure 2.7, as well as for both 5 and 6 percent enrichment for the resonance integral of 13.83 barns previously discussed. The calculations for the higher value of I_{eff} were made with both 304 stainless steel and Incoloy-800 as clad and can material, and all calculations were made to show the effects of water displacement by means of Zircaloy and Beryllium shims. The calculations were made for water displacements of 15, 35, and 55 percent by means of a solid rod of either Zircaloy or Beryllium at the center of the moderator tube. The results of these calculations are shown in Figures 3.18, 3.19, 3.20, and 3.24, and for low I_{eff} values the following cases are included.

<u>Case</u>	<u>Fuel Temperature</u>	<u>Moderator Temperature = Neutron Temperature</u>
1	68°F	68°F
2	1500°F	68°F
3	1500°F	250°F
4	68°F - flooded	68°F

Case 1 is indicative of the cold-clean reactivity of the cell; case 2 gives the effect of Doppler broadening to 1500°F in the U-238 with the neutron temperature held at 68°F. A fuel temperature of 1500°F has been assumed for the mean Doppler effective temperature of the fuel, and no attempt has been made to correlate this value with an average fuel temperature from thermodynamic analyses. In detailed analysis, the two temperatures do not necessarily equal each other. The indication from the single cell reactivity is, however, a negative fuel temperature coefficient of approximately $1.3 - 2.0 \times 10^{-5} \Delta K / ^\circ F$. This value is relatively constant with enrichment, and increases somewhat for higher values of water displacement. Comparison of case 3 with case 1 yields the combined effects of Doppler broadening and moderator temperature coefficients. For the hot moderator, the density of H₂O was reduced to 0.944 g/cc, a density change which increases longitudinal leakage somewhat. The total temperature coefficient depends upon the amount of water in the cell; for little water displacement, the overall temperature coefficient is positive but it becomes negative for about 10 percent of the water displaced. Since the startup point for the reactor would be for significant fractions of the water removed, the overall temperature coefficient is negative for water fractions of interest.

Inspection of Figures 3.18 and 3.19 shows that there are relatively minor differences in the hot clean reactivity of the base cell for 5 percent enriched fuel

(1) Velarde, G., "Dancoff Correction for Several Infinitely Long Cylindrical Rings", Nuclear Science and Engineering: Vol. 15, January 1963, p. 99.

* Doppler broadening increases the resonance integral to 14.87 barns for hot fuel cases; admixed water increases it to 20.3 barns for flooded cases.

with low I_{eff} and 6 percent enriched fuel with higher I_{eff} . There are, however rather significant differences in the effects due to water displacement since the higher resonance integral for 6 percent fuel results in resonance captures increasing more quickly as water is displaced. Because of this, it is very important to obtain precise values of the resonance integral for these calculations.

Figures 3.19 and 3.20 compare the results when enrichment is changed maintaining all other parameters constant. In general, the curves have the same shape with the 6 percent enrichment resulting in nearly 0.04 Δk increase in reactivity.

All of the cases calculated show a striking difference between the reactivity resulting from displacement of water with Beryllium and with Zircaloy. The large difference occurs because Beryllium is a much better moderating material than Zircaloy, has a lower absorption cross section, and also contributes neutrons through the $(n,2n)$ reaction. For cases with large amounts of water displacement the latter effect contributes approximately 2 percent to total reactivity.

The conversion ratio of fertile to fissile material for the case of hot fuel and elevated moderator temperature is plotted as a function of water fraction in Figure 3.21 for the 6 percent high I_{eff} , Figures 3.22 and 3.23 for low I_{eff} , respectively. The important criterion for comparison of Zircaloy and Beryllium is the conversion ratio that results for equivalent single cell reactivity. Values are marked on the curves for k single cell equal to 1.07, and it can be seen that once again Beryllium gives superior results. Unfortunately, Beryllium is not attractive from the standpoint of mechanical stability, and some compromise will probably be required.

The important result of these calculations is that a substantial increase in conversion ratio can be obtained by utilizing water displacement as a means of holding excess reactivity. To be sure, if burnable poison were added to the fuel to reduce the reactivity of the base cell, the conversion ratio would increase slightly due to a small amount of spectral hardening. In some cases which have been investigated, equivalent reactivity reduction with burnable poison in the form of boron increased conversion ratio by about 7 to 8 percent of the base ratio.

It is interesting also to compare the conversion ratios which result for the three cases discussed where reactivity of the cell is reduced to a common value. As stated previously, values for k single cell equal to 1.07 are marked on the curves. The values are compared in Table 3.5.

TABLE 3.5

CONVERSION RATIOS AT SINGLE CELL REACTIVITY EQUAL TO 1.07

Shim Material	6% High I_{eff}	6% Low I_{eff}	5% Low I_{eff}
Beryllium	0.494	0.468	0.472
Zircaloy	0.456	0.412	0.420

The interesting result is that for the 20 percent increase in enrichment, from 5 to 6 percent, the conversion ratio for equal reactivity is only slightly affected. A more significant increase (~6%) results when I_{eff} is increased.

Fast effects have been calculated for the various configurations by calculating the ratio of U-238 fissions to U-235 fissions in the spectra obtained from the

computer programs and applying this ratio δ to the formula

$$\epsilon = 1 + 0.59838 \delta \left[\frac{1 - 7.0176 \delta}{1 - 5.2023 \delta} \right]$$

due to Tunncliffe, et al.⁽¹⁾ The values of δ obtained for the hot-clean cases at 1.07 k single cell utilized previously are approximately 0.06 yielding ϵ approximately equal to 1.03.

Flooding - The geometry of the Mark V(B) core is such that the core gains reactivity if the gas passages and dead spaces are filled with water. This is illustrated in Figure 3.24. The configuration considered for control of the flooding situation is by means of poison sleeves pushed into the moderator tube just outside the water flow divider. From mechanical considerations it appears necessary to insert the poison sleeve inside the flow divider tube, however, the results of the calculations are indicative of control sleeve effects. (See Section 3.5). Although calculations of the effects of various poisons have not been completed for the geometry which has been discussed until now, calculations were completed for a very similar cell, as shown in Table 3.6.

TABLE 3.6
GEOMETRY FOR FLOODING SAFETY ANALYSIS

Region	Material	Outside Radius (cm)
1	Water	1.0
2	Be	1.5, 1.9, 2.119 ^(a)
3	Water	2.2492
4	Zircaloy	2.3
5	Incoloy 800 clad	2.3508
6	Poison	2.5286
7	Incoloy 800 clad	2.5794
8	Water	2.68567
9	Zircaloy	3.09883
10	Can and Void	3.68906
11	Fuel	5.21332

(a) Three radii analyzed, corresponding to 82, 62, and 50 percent of base cell water, respectively, without safety rod inserted.

The single cell reactivity flooded was 1.2543, 1.2730, and 1.2807 for the 1.5, 1.9, and 2.119 cm radius Beryllium shims, respectively. The calculated reactivity with poison inserted is summarized in Table 3.7.

(1) Tunncliffe, et al., loc. cit.

TABLE 3.7

REACTIVITY OF SINGLE CELLS WITH POISON SLEEVES

Shim Radius	Poison Material		
	Boron Incoloy ^(a)	B ₄ C ^(b)	Cadmium
1.5	0.7625	0.5431	0.7414
1.9	0.8074	0.5781	0.7955
2.1	0.8327	0.5984	0.8281

(a) 2 wt. percent natural boron in Incoloy 800

(b) 65 volume percent B₄C - natural boron

The large reduction in reactivity resulting from insertion of poison sleeves results from the very effective isolation of the flooded fuel region from the major volume of moderator contained in the moderator tube. Although the water in the fuel region produces a more thermal spectrum and enables more neutrons to escape resonance capture in the U²³⁸, the major portion of neutron thermalization occurs in the moderator tube.

Control Requirements, Lifetime, and Enrichment - The enrichment required to obtain the design point fuel lifetime of 20,000 MWD/MT has not been established for the Mark V(B) geometry. The compromise of shim material between Zircaloy and Beryllium must be established before meaningful calculations can be made to determine initial enrichment. At present it is anticipated that the permanent shim, required for gross radial power flattening at the end of life, would be Zircaloy as part of the flow divider tube. The portion of the shim which is moved by the control rod actuator would probably also be Zircaloy for mechanical stability and the amount of reactivity in this shim would need to be that required for the temperature transients ($\sim 0.03 \Delta k$), for Xenon transients ($\sim 0.02 \Delta k$) a shutdown margin ($\sim 0.02 \Delta k$) and for initial stage burndown ($\sim 0.03 - 0.04 \Delta k$) totaling $\sim 0.10 - 0.11 \Delta k$. The burndown can be accomplished in three approximately equal increments, utilizing Beryllium for the two manually removeable pieces, adjusted in size so that a flat gross radial power distribution is maintained through life. The values of the conversion ratio listed in Table 3.5 indicate that value of approximately 0.45 or greater can be attained. Comparison of 5 percent and 6 percent enriched fuel values with other parameters being equal indicates that the conversion ratio will be maintained at roughly the initial value. For a conversion ratio of 0.45, the estimated burnup value is approximately 1900 MWD/MT per 0.01 Δk ; i.e., 0.105 Δk required.

3.4.3 Power Shaping

Figure 3.25 is an illustration of the fine radial power distributions obtained from the calculations for 6 percent enriched fuel. The power distributions are plotted as a function of fuel volume, and the vertical lines indicate the amount of fuel associated with each of the three positions which the fuel rods occupy, rod facing the moderator directly, rods partially shadowed by the rods facing the

moderator, and the central fuel rod. Curves have been plotted for the base cell, for 55 percent moderator displaced by Beryllium and 35 percent moderator displaced by Zircaloy. Once again the effects of a more energetic mean neutron spectrum are apparent in the reduction of fine radial power distribution as water is displaced. The plotted curves represent the power distribution through a homogeneous fuel ring of reduced density compared to that of an actual fuel rod; the power distribution within individual rods would require 2 dimensional analyses which have not been performed.

Gross radial power shaping has not been completed for a full core configuration. The method of attack for this problem has been to adjust the amount of shim material in the cells to produce a flat power profile in the gross radial direction where 91 cells are put together for the full core. This results in more water displacement in the center cells of the core than the average for the core. The most difficult area is at the edge of the core where relatively more fuel is associated with each of the moderator tubes, since each of the outside fuel cans is a full can containing 27 fuel rods. For this region a pseudo-cell has been set up consisting of the additional fuel added to the outside of the cell listed in Table 3.4 and with BeO as an annular ring on the outside of the fuel. Single cell calculations with an albedo of 1 at the boundary yields an indication of the flux distribution through this pseudo-cell and enables cell corrections for the various regions to be determined. In the full core calculation this region is then homogenized with appropriate cell corrections for each of the materials in the particular cell subdivisions (BeO, fuel, water, etc.). From the full core calculations with reflectors, the albedos at the outer and inner boundaries of this region are determined. The new albedos are then utilized in single cell calculations to determine new flux distribution factors and the process iterated. This iterative procedure has not been completed, however there appears to be no difficulty in attaining the design point of ± 1.05 local to average gross radial power distribution.

100X140 7/100 Inch Divisions

GENERAL ELECTRIC COMPANY, SCHENECTADY, N. Y., U.S.A.

FN-521-B (8-50)

Figure 3.18

100X140 7/100 Inch Divisions

GENERAL ELECTRIC COMPANY, SCHENECTADY, N. Y., U.S.A.

FN-521-B (8-50)

Figure 3.20

Conversion Ratio
Versus Water Displacement
6% Enriched, $I_{eff} = 20$ barns
Incoloy-800 clad

100X140 7/100 Inch Divisions

GENERAL ELECTRIC COMPANY, SCHENECTADY, N. Y., U.S.A.

FN-521-B (2-50)

Conversion Ratio
Versus Water Displacement
6% Enriched, $I_{eff} = 14.87$ barns

Conversion Ratio

0.7
0.6
0.5
0.4
0.3
0.2

Fraction H_2O in Cell

Single Cell Reactivity
Equal to 1.07

Zircaloy Shims

Be Shims

Figure 3.23

0 0.2 0.4 0.6 0.8 1.0

100X140 7/100 Inch Divisions

GENERAL ELECTRIC COMPANY, SCHENECTADY, N. Y., U.S.A.

FN-521-B (8-50)

Figure 3.24

Fig 3.25

100X140 7/100 Inch Divisions

GENERAL ELECTRIC COMPANY, SCHENECTADY, N. Y., U.S.A.

FN-521-B (8-50)

Mark V(B)
 Fine Radial Power Distribution
 6% Enrichment
 27 Rods/Element
 Inner R(Equiv) = 4.10745 cm
 Outer R = 6.09674 cm

Figure 3.25

3.5 CONTROL TUBES

3.5.1 Description and Function

Reactivity control of the 630A Mark V(B) reactor is accomplished by three types of actuated control tubes and a manual shim tube and rod.

The actuated control tubes are designated as safety tubes, shim tubes, and dynamic tubes. The safety tubes are used for rapid, positive shutdown of the reactor; the shim tubes provide gross reactivity control; the dynamic tubes maintain fine control for normal operating transients.

The manual shim tube and rod are used to provide gross reactivity control at the start of life, thereby resulting in a smaller actuated shim worth to control reactivity over a portion of the operating lifetime. Because the manual shims can be removed from the active core in finite steps throughout the lifetime of the core, optimization of the moderator to fuel ratio for conversion can be achieved. The use of these manual shims also lessens the longitudinal power peaking at the bottom of the core that occurs if all the reactivity is held in actuated shims.

There are 48 actuated safety tubes, 84 actuated shim tubes, and 7 actuated dynamic tubes in the active core located within the 91 moderator tubes. Each of the 91 moderator tubes also contain a manual shim tube and rod. This results in 7 moderator tubes that contain an actuated dynamic tube and the manual shim tube and rod; 36 moderator tubes that contain an actuated shim tube and the manual shim tube and rod; and 48 moderator tubes that contain an actuated safety tube, an actuated shim tube and the manual shim tube and rod. The actuated control tubes are ganged into groups in the upper chamber of the shield plug. The groups of the control tubes are driven by the control tube actuators located on the top surface of the shield plug. The arrangement of the control tubes in the reactor and the grouping of the tubes into frames in the upper chamber of the shield plug is shown in Figure 3.26. The position of the actuator is superimposed on each frame. There are 6 safety tube actuators, 6 shim tube actuators and 2 dynamic tube actuators.

The manual shims consist of a tube and nested rod located in the center of each moderator tube. They are positioned manually, either in or out of the active core, from the top head of the shield plug. It is intended to remove the tube and rod in turn at intervals of approximately one-third of the fuel life. Manual shim adjustments are scheduled to take place during ship inspection and maintenance periods.

Radial cross sections of the moderator tubes showing the location of the actuated control tubes and the manual shims within the moderator tube are shown in Figure 3.27. Figure 3.28 is an isometric drawing of a moderator tube depicting the method of driving the shim tube and safety tube concentrically. In this figure, the safety tube is fully withdrawn from the active core, the actuated shim tube is in an intermediate position, the manual shim tube and rod are fully inserted in the core.

The safety tubes are cylindrical in shape and consist of a 30-inch-long poison section and an extension tube. The poison section consists of a poison such as cadmium or borated stainless steel clad with Incoloy. The extension tube is Incoloy. The safety tubes are fully withdrawn from the active core during normal operation and fully inserted during reactor shutdown. Because these tubes are fully withdrawn from the active core during normal power operation, burnup of the poison does not present a problem. Hence, the safety rods may be used for more than one fuel cycle.

ACTUATORS		CONTROL TUBES	
TYPE	NO.	PER ACTUATOR	TOTAL
SHIM	6	14	84
SAFETY	6	8	48*
DYNAMIC	1	3	3
DYNAMIC	1	4	4
TOTAL	14		139

* THE 48 SAFETY TUBE HOLES CONTAIN BOTH A SHIM TUBE AND A SAFETY TUBE.

Fig. 3.26 - Control tube and actuator pattern (849D320)

Fig. 3.27 - Radial cross section of the two typical types of control cells
(118C9770)

**Fig. 3.28 - Isometric sketch of control cell containing safety tube
(849D321)**

The shim and dynamic actuated tubes and the manual shim tube and rod are cylindrical in shape. They are fabricated from a low absorption cross section material such as Zircaloy or Be and provide control by displacing the moderator water. Since there is no burnup involved in these materials, the tubes are good for the life of the core.

3.5.2 Control Tube Design Requirements

The control tubes are designed to operate in a flowing water environment at a maximum temperature of 300°F. The safety control tubes must withstand the dynamic force exerted by the safety actuators during a scram shutdown. The control tubes also must be able to operate in any attitude assumed by the ship. All control tubes are cooled below 250°F by the moderator water as it enters the core.

The control tube actuators are readily accessible for maintenance and replacement, and can be manually repaired or replaced without disturbing any of the other system components. The control tubes are also accessible by removing the top plate of the shield plug.

The manual shim tubes are accessible from the top of the reactor and can be moved after depressurizing the moderator loop and draining water below the level of the actuator mounting face.

3.6 CONTROL ROD ACTUATORS

Three types of control tube drive actuators are used for control and reactor safety. These consist of 6 shim, 2 dynamic and 6 safety tube actuators. The safety actuators have a 34-inch stroke, the shim actuators have a 42-inch stroke and the dynamic actuators have a 30-inch stroke. Each actuator drives from 3 to 17 control tubes which are ganged together within the upper plenum of the shield plug. (See Figure 3.29). Each actuator drive shaft is connected to the ganging frame through a quick-disconnect coupling. The actuators are mounted on the shield-plug upper-plenum cover plate through quick-disconnect couplings. Each actuator drive rod pierces the shield plug through a rod seal which is built into the actuator. The rod seal consists of a standard chevron type seal ring made of cotton duck impregnated with rubber. Recent tests showed this type of seal to be extremely reliable, to have long life and low leakage.

3.6.1 Safety Tube Actuator

Figure 3.30 shows the pneumatically-cocked, spring-driven, linear safety tube actuator. It provides a very simple and reliable means for rapidly shutting down the reactor if unsafe operating conditions are too closely approached. It consists of a cylinder assembly, cocking piston and drive rod, compression drive spring, rod latch assembly, and associated solenoid valves and limit switches.

In its normal power off position, the drive spring acting against the cocking piston holds the drive rod in its fully inserted position. The actuator is driven to its cocked position by applying electrical power to the retract solenoid valve. This valve directs air at 150 psi to the lower face of the retract piston and causes the actuator to move in a retracting direction compressing the scram spring from its preload force of 212 pounds to its cocked position where it has a force of 410 pounds. When the piston reaches the retract position, it actuates the retract limit switch which controls a light on the operator's console. The operator then energizes the latch solenoid which directs air to the latching piston and

3

ACTUATORS		CONTROL TUBES	
TYPE	NO.	PER ACTUATOR	TOTAL
SHIM	6	14	84
SAFETY	6	8	48*
DYNAMIC	1	3	3
DYNAMIC	1	4	4
TOTAL	14		139

* THE 48 SAFETY TUBE HOLES CONTAIN BOTH A SHIM TUBE AND A SAFETY TUBE.

Fig. 3.29 - Control tube and actuator pattern (849D320)

de-energizes the cocking solenoid thereby venting the air on the lower face of the cocking piston to the atmosphere. The latch piston restrains the fingers on the scram latch, thus holding the rod in the retract position. The scram latch fingers are designed to be fail safe, i.e., in the event the air pressure behind the latch piston is removed, the latch will release and the actuator will scram due to the scram spring force. No external source of power is required.

The actuator is held in a latched position as long as the power is applied to the latch solenoid. When a scram is required, the electric power to the latch solenoid is interrupted; this causes the air to vent from behind the latch piston releasing the latch, thus allowing the spring to insert the rod. As the piston drives in the insert direction, it compresses air ahead of it which provides a buffer to slow down the piston. This air is allowed to leak off through a retract valve until the piston comes to rest at the end of travel, thereby providing a smooth stop for the piston. When the piston reaches a fully inserted position, it actuates a limit switch which operates a light on the control console.

The spring has the capability of driving a 100-pound mass 5 inches in water against 235 psig water pressure in 100 milliseconds. The actuator will continue to accelerate until it has travelled approximately 24 inches, at which time the buffer pressure will have built up sufficiently to start to slow it down. The final 10 inches are spent in slowing down and stopping.

3.6.2 Shim Tube Actuator

The shim tube actuator, shown in Figure 3.31, consists of an electric motor-driven, rotary-output powerhead, a ball lead screw, a drive rod, and a housing assembly. These actuators position shim tubes in the reactor core to compensate for slow changes such as fuel depletion and poisoning, and provide followup of the dynamic tubes causing them to return to a neutral position. They may also be used for control without the dynamic tube actuators but with some sacrifice in accuracy and response. They operate at either the full-insert rate or the full-withdraw rate. The power head consists of a three-phase hysteresis synchronous motor operating on single-phase, 60 cycle, 115-volt power by the use of a capacitor across two phase windings. This capacitor alters the phase relationship of the windings so that a three-phase current relationship occurs in the windings. Withdraw or insert action is obtained by applying the single-phase power to one winding or the other. This simplifies the power requirements and the switching problems of both the insert - withdraw switch and the limit switches. Power consumption is approximately 18 watts.

The motor operates at 3600 rpm. This speed is reduced to 36 rpm at the output shaft by a 100-to-1 planetary gear reduction. This provides a 4.5-inch-per-minute linear speed for the shim actuator. The rotary actuator has a torque capacity of 5.0 inch-pounds which provides approximately 225 pounds of linear force at the output shaft of the actuator.

During operation, whenever the a-c voltage is removed from the drive motor, a d-c voltage is applied to the motor winding to provide dynamic braking to minimize coasting. The d-c voltage remains on the motor winding whenever the actuator is not driving and also provides a static brake to hold the actuator in a stationary position. This holding force is equal to or greater than the maximum output of the actuator, 225 pounds.

The motor is also equipped with a 28-volt d-c brake which holds the actuator in a fixed position whenever the driving power is not applied. This provides a lock

to hold the actuator in position in the event of a power failure. The brake has a holding force equivalent to about 1400 pounds on the actuator shaft. During normal operation the d-c brake is energized, and, therefore, out of the system. It is controlled by circuits external to the actuator.

The rotary actuator contains a ten-turn position-indicating potentiometer driven by the output shaft through a planetary gear reduction to provide 8.93 turns of the potentiometer for 42 inches of actuator travel. The same shaft which turns the potentiometer also actuates limit switches at the end of its travel to prevent the actuator from driving into the mechanical stops.

In the actuator itself, movement of the output shaft is caused by turning the ball lead screw and restraining the ball nut from rotating. The ball nut is restrained from rotating by three ball bushings housed in the ball nut assembly which slide on three guide rods that extend the entire length of the actuator.

3.6.3 Dynamic Tube Actuators

The dynamic tube actuator is identical to the one used in Mark V(A) and is described in paragraph 2.6.2.

Fig. 3.30 - 630A safety tube actuator including wiring diagram (219R882 Sk 1)

Fig. 3.31 - 630A shim rod actuator assembly (219R894)

4. BOILER

4.1 GENERAL DESCRIPTION AND FUNCTION

The 630A boiler is a once-through water tube heat exchanger capable of producing the rated flow of approximately 175,200 pounds per hour of superheated steam at 1005°F and 1550 psia. The superheated steam is supplied to the propulsion turbines at 1000°F and 1515 psia. The once-through boiler was chosen because of the low number of penetrations of the pressure vessel, smaller size, and less complex mechanical design than a recirculating type boiler with a steam drum. The 830 psia primary gas coolant pressure permits a compact tube arrangement. Tube fouling will not be a problem since there are no combustion products in the gas. Since the maximum average temperature of the primary coolant is 1200°F, there is minimal danger of burning out of the tubes during water-level changes or loss of secondary coolant.

As shown in Figure 1.2, the boiler assembly is located below the reactor in the lower section of the pressure vessel. The bottom dished head of the boiler assembly closes the bottom of the pressure vessel. The boiler shield and diffuser is mounted on top of the boiler assembly directly beneath the reactor. The boiler shield and diffuser serves a dual function: it improves the flow distribution of the primary gas into the boiler region, and it reduces the fast neutron dose to the boiler.

The boiler assembly, shown in Figure 4.1, consists of a tube bundle, a shell that houses the tube bundle, a flow divider, the bottom head, and an exit steam manifold.

Based on preliminary data prepared by different boiler manufacturers, there are several suitable methods of meeting the requirements for the boiler used in the 630A. One of the most attractive of these methods utilizes a tube bundle similar to that used in the Enrico Fermi atomic power plant. In the 630A concept, 320 tube circuits consisting of 64 clips of 5 tubes each are used. The tubes and the shell housing the tubes are supported from the flow divider. The tubes, formed into a serpentine configuration in the longitudinal plane, make 19 horizontal runs in the annular space between the outside diameter of a central thermal shield and the inside diameter of the outer shell. Thus, each clip presents 95 horizontal tube surfaces. Primary gas passes over the tube surfaces as it flows downward through the boiler. The 64 clips are arranged in radial position around the circumference of the unit. In order to maintain an even spacing of 0.70-inch transverse pitch between the tubes for uniform flow distribution and heat extraction, each clip of tubes is formed into an involute curve. The 16 BWG tubes are 0.50 inch in outside diameter and approximately 146 feet long, exposing a total of 5380 square feet of effective surface to the helium flowing in the annular region.

Feedwater is admitted to the outer annular headers in the bottom head and flows upward through the tubing in the annular region. The feedwater is then heated, evaporated, superheated and returned downward through the cylindrical

Fig. 4.1 - Boiler Assembly Drawing (221R465)

Fig. 4.1 (Cont'd) - Boiler Assembly Drawing (221R465)

center section inside the thermal shield to the inner annular header as superheated steam. There are two independent feedwater inlet headers connected by the tubes to two independent steam outlet headers. In a given clip of 5 tubes, 2 tubes are connected to one set of headers and 3 tubes to the other set of headers. By alternating the header to which 3 tubes of a clip are connected, the number of tubes connected to each header is made equal, thus dividing the boiler into two sections of equal heat transfer area. Should a tube fail during operation, the section containing the failed tube could be valved off, permitting continued operation at reduced power. With either 2 or 3 tubes in each clip inoperative, the temperature at any radial cross section through the boiler will be essentially constant, i.e., no hot streaks will result.

Reactor discharge gas, at a temperature of 1200°F, flows downward through the boiler in the annulus formed by the outside diameter of the central thermal shield and the inside diameter of the shell that houses the tube bundle. The gas passes over the steam superheater surfaces, which comprises about 22 percent of the total heat transfer area. The gas then passes successively over the evaporator and economizer surfaces, which contain about 41 and 37 percent of the total heat transfer area, respectively. Thus, heat is transferred in cross-counter-flow from the primary loop to the secondary steam-and-water loop in the boiler.

The flow divider, attached to the boiler header, acts as a separator for the circulator inlet and exit gas. Gas exiting from the boiler, passes upward in the annulus between the boiler shell and the flow divider, enters the inlet duct to the circulator openings in the flow divider, and is discharged from the circulators into the annular area between the flow divider and the pressure vessel.

The bottom of the boiler is a dished head which is bolted to a flange on the bottom of the pressure vessel. The header consists of two separate components, joined in manufacture, to form an integral unit. The basic form is a 3.25-inch-thick dished head with an open crown. This head has two annular inlet water plenum chambers which provide feedwater to the two individual sets of boiler tubes. The other component, which fits into the open crown of the basic dished head, is an exit steam header with one circular and one annular steam plenum chamber to receive the superheated steam from the individual boiler tubes. The dished head and the exit steam header are provided with longitudinal walls to facilitate the joining of the two components and to provide a longer flow path for heat conduction. A radial gap is provided between the walls except where the joint is made. This permits both radial and longitudinal expansion due to the relative temperature differences between the water and steam plenum chambers.

An exit steam manifold casting is bolted to the two exit steam plenums and provides flange connections for the two steam pipes.

Both the inlet and exit manifolds can be removed for tube plugging operations in the event of a tube failure.

The boiler and diffuser is located between the reactor and the boiler. Although

not a part of the boiler assembly, it is supported from the top of the boiler shell. The boiler shield and diffuser are shown in Figure 4.2.

The boiler shield and diffuser is a 13-inch-high circular, stainless steel structure consisting of six circulator 1-inch-thick vanes, interstested to form a series of 1-inch-wide flow passages, that are supported by six equally-spaced radial ribs that extend from the center section to the outer ring. The boiler shield and diffuser is fastened to a mounting ring which is bolted to the boiler shell. The method of fastening the boiler shield and diffuser into mounting ring will accommodate the differential thermal expansion between the two components. It will also be capable of being unfastened through the top of the pressure vessel. This design feature will permit the diffuser to be removed through the top of the pressure vessel after the reactor - shield-plug is removed. A laminated circular seal mounted on the bottom surface of the outer ring of the instrumentation harness bears against the top surface of the diffuser outer ring, thus effecting a seal that prevents the primary loop coolant from bypassing the reactor. The instrument harness is sealed against the bottom surface of the beryllium oxide bottom support plate of the reactor.

4.2 DESIGN REQUIREMENTS AND CONDITIONS

The boiler will be designed to comply with the Codes listed in Section 1.5. The objective design life is 20 years at the rated conditions. Ample margin will be provided for erosion and corrosion losses throughout the operating life.

The design conditions for operation at 27,300 SHP consist of a heat load of 1.925×10^8 Btu/hr with inlet and exit helium temperatures of 1200°F and 550°F, respectively. Inlet and outlet helium pressures are 825.9 and 824.6 psia, respectively. Feedwater is circulated through the boiler at a rate of about 175,200 pounds per hour, entering at a temperature of 415°F and a pressure of 1700 psia. Superheated steam is delivered from the boiler at 1550 psia and 1005°F. Other details of the boiler design are provided in Appendix A.

Preliminary design requirements have been established based on heat transfer requirements, operating pressures and temperatures, and desired envelope dimensions. The requirements for normal design, and off-design operating conditions have been specified. Off-design conditions include those for maximum and partial power operation, atmospheric pressure operation, and aftercooling. Possible growth version requirements have also been defined.

4.3 MECHANICAL DESIGN

4.3.1 Materials

The materials to be used will be determined by the boiler manufacturer during the engineering study. The boiler tubes will probably be made from a low alloy steel such as 9Cr-1Mo or Inconel. The average fast neutron flux (greater than 0.8 Mev) to the superheater (assumed to be the upper 17.5 inches of the boiler) is 2.6×10^{19} nvt per fuel cycle of 1×10^6 megawatt hours. Although the fast neutron dose is at a potentially troublesome level for the low alloy steels at room temperature, the operating temperatures involved would afford a good deal

Fig. 4.2 - Boiler shield and diffuser (849D324)

of self-annealing of any radiation damage. Inconel could be used if the radiation damage of the low alloy steel appears to be questionable. Although somewhat more costly, Inconel has good stress corrosion resistance and low transition temperature characteristics and would be well suited for this application.

The bottom dished head will be fabricated from SA-212 B to match the material of its mating component, the pressure vessel.

4.3.2 Fabrication and Assembly

The boiler will be fabricated, tested and certified to the requirements of the U. S. Coast Guard and ASME Boiler and Pressure Vessel Code by a boiler manufacturer.

4.3.3 Handling Requirements

The boiler assembly will be fabricated and handled as a unit. The boiler assembly is inserted into the lower portion of the pressure vessel and the bottom head is bolted in place.

4.4 THERMODYNAMIC DESIGN

The major thermodynamic design requirements are (1) provide for each region of the boiler an adequate heat transfer area for the established heat load, the overall heat transfer coefficient, and logarithmic mean temperature difference, and (2) minimize primary coolant pressure losses for the envelope provided. Satisfying both requirements involves careful selection of parameters such as tube size and spacing, number of tubes per clip, length-to-diameter ratio of the tube bundle, coolant pressure level, and temperatures. Optimization studies involving the various parameters mentioned have been carried out in conjunction with performance estimates provided by the boiler manufacturers. These preliminary sizing estimates will be verified during the course of the boiler engineering study.

In the present preliminary configuration, the boiler meets the thermodynamic requirements and is compact and easily maintained. At full power the heat transfer rate per unit of tube bundle volume is about 0.97×10^6 Btu/hr-ft³ and the pressure loss is about 0.2 percent of the inlet primary coolant pressure. Detailed operating conditions are shown in Figure 4.3 for five regions; one for the economizer, three for the evaporator where the mean temperature difference and tube side heat transfer coefficient change appreciably, and one for the superheater. The primary coolant pressure level of 830 psia, the maximum temperature of 1200°F and the steam conditions of 1500 psig, 1000°F were selected by virtue of reliability, performance and structural design considerations. The helium exit temperature of 550°F was selected as a value low enough to permit use of two circulators each rated about 250 horsepower but high enough to provide an adequate temperature difference between helium and water at the top of the economizer. This temperature difference at the knee of the water temperature plot, Figure 4.3, affects boiler heat transfer area to a large degree.

Figure 4.4 shows the effects of varying exit helium temperature on circulator power

Boiler Operating Conditions

27,300 SHP 630A NSG

Figure 4.3

BOILER STUDY

EFFECTS OF BOILER EXIT HELIUM TEMPERATURE VARIATION

NOTES:

1. 1200°F Inlet Temperature
2. 460°F Final Feedwater Temp.
3. Plant Performance is about 0.33 percent (0.11 points) poorer for each 100 BHP increase in circulator power.

Figure 4.4

and boiler heat transfer area. For this study 460°F feedwater temperature, the optimum considering only the steam plant was selected. The circulator power increases nearly 100 BHP with a 25°F helium temperature rise from 550 to 575°F; heat transfer area decreases about 14 percent. With 415°F final feedwater temperature, as selected, the reduction in heat transfer area is less than 14 percent for the same helium temperature increase.

Figure 4.5 shows the effects of varying final feedwater temperature on plant efficiency and boiler heat transfer area. Other parameters of pressure and temperature were held fixed for this boiler study. With a reference feedwater temperature of 460°F, heat transfer area is reduced by 12 percent and plant efficiency is reduced only 0.091 points with 415°F feedwater temperature. With an additional reduction of 45°F to 370°F feedwater temperature, plant efficiency is reduced about 0.327 additional points and the reduction in area is only 6 percent. A tentative selection of 415°F final feedwater temperature has been made.

The tube bundle described in the preceding Section 4.1 was selected following a parametric study in which bundle diameter, tube diameter, tube pitch and number of tubes per clip were varied. The selected bundle provides the required heat transfer area and acceptable pressure loss and permits use of a constant diameter pressure vessel around the reactor and boiler. Characteristics of the various tube bundles studied are shown in Table 4.1.

Alternate number 8 is the tentatively selected configuration. The cost of this tube bundle is less than 6 percent higher than the minimum bundle cost as calculated for case 7.

BOILER STUDY

EFFECTS OF FINAL FEEDWATER TEMPERATURE VARIATION

Notes:

1. There is no significant variation in circulator power.
2. Helium exit temperature is 550°F.
3. Helium inlet temperature is 1200°F.
4. A plant efficiency change of 1 percent is about 0.34 points.

FINAL FEEDWATER TEMPERATURE, °F

Figure 4.5 ~

Table 4.1

BOILER STUDY SUMMARY

<u>Alternate Number</u>	<u>Tube OD, in.</u>	<u>Tube⁽¹⁾ Transverse Pitch in.</u>	<u>Bundle OD, in.</u>	<u>No. Tubes per Clip</u>	<u>No. Clips</u>	<u>Area AHT ft²</u>	<u>Bundle Height, in.</u>	<u>Helium Pressure Drop psi</u>
1	5/8	0.781	91.5	4	50	5428	87.7	2.07
2	5/8	0.781	91.5	3	66	5850	94.6	1.55
3	5/8	0.781	80.	4	50	5250	107.5	2.91
4	5/8	0.781	80.	3	66	5300	115.5	3.04
5	7/8	1.031	80.	2	48	5410	177	4.93
6	1/2	0.656	80.	4	80	4990	85.1	1.95
7	1/2	0.656	80.	5	64	4930	79.8	1.92
8	1/2	0.700	80.	5	64	5380	87.0	1.314

Note:

- (1) Longitudinal pitch is equal to transverse pitch except for alternate number 8 which has a longitudinal pitch of 0.625 in.

5. PRESSURE VESSEL

5.1 DESCRIPTION AND FUNCTION

5.1.1 Mark V(A) Pressure Vessel

The pressure vessel is a carbon steel cylindrical shell that is flanged on each end, as shown in Figure 5.1. It is 16 feet, 9 inches high and consists of a lower section, an upper section, a conical transition section, and two equally-spaced nozzles that accommodate the gas circulators.

As shown in Figure 1.1, the top of the pressure vessel is closed by the reactor - shield-plug assembly and the bottom is closed by the boiler header. The gas circulators provide closures for the nozzles. The pressure vessel contains the reactor primary coolant (helium) and supports the reactor - shield-plug assembly, the boiler, the inner side shield water tank assembly, and the gas circulators.

The lower section of the pressure vessel is a cylinder 76.5 inches high with an outside diameter of 121.0 inches. The wall thickness is 3.0 inches except for the flange where the thickness is 10 inches. The nozzles penetrate the lower section 44 inches above the face of the bottom flange and have an inside diameter of 30 inches.

The annulus formed by the outside surface of the flow divider and the inside surface of the lower section of the pressure vessel provides a flow path for the primary coolant discharged from the gas circulators.

The upper section accommodates the reactor - shield-plug assembly and provides a flow path for the primary coolant between its inside surface and the outside surface of the reactor shell. The upper section is a 2.5-inch-thick cylinder, 104 inches high with an inside diameter of 92.0 inches. Four equally-spaced keys provide alignment for the calandria vessel of the reactor - shield-plug assembly. If the core were to break away from the shield plug and, in effect, partially withdraw control rods, these keys would also limit the amount of drop of the core. The top of the vessel is closed by bolting the reactor - shield-plug assembly to the top flange of the pressure vessel with fifty-six 2.75-inch studs.

The 20.0-inch-high, 3-inch-thick conical transition section joins the upper and lower sections of the pressure vessel and supports the inner portion of the inner side shield. Four equally-spaced keys provide alignment for the boiler assembly, and provide lateral restraint of the upper section of the boiler.

The bottom of the vessel is closed by bolting the boiler header to the bottom flange of the pressure vessel with sixty 3.0-inch studs. The bottom of the boiler is a 3.5-inch-thick dished head with a 10-inch-wide, 10-inch-deep flange. The mating flange on the pressure vessel is also 10 inches wide and 10 inches deep.

Fig. 5.1 - 630A Mark V(A) Pressure Vessel (221R457)

The gas circulator nozzles are flanged close to the pressure vessel shell to permit the pressure vessel to be installed within the containment vessel. After the pressure vessel is positioned within the containment vessel, a short, cylindrical section, flanged on each end, is bolted to each pressure vessel nozzle flange.

The pressure vessel is supported at the gas circulator nozzle centerline by eight brackets that rest on eight mating brackets on the containment vessel. The pressure vessel is bolted to a support ring on the containment vessel with two 2.0-inch studs per bracket, as shown in Section B-B of Figure 1.1. These studs are approximately 65 inches long and are capable of withstanding the differential radial expansion between these components. Radial keys on the brackets maintain concentricity and permit radial thermal expansion.

5.1.2 Mark V(B) Pressure Vessel

The pressure vessel is a carbon-steel cylindrical shell that is flanged on each end as shown in Figure 5.2. It is 16 feet, 3-1/2 inches high and consists of a lower flanged cylindrical section, a conical transition section, an upper flange ring, and two equally-spaced nozzles that accommodate the gas circulators.

As shown in Figure 1.2, the top of the pressure vessel is closed by the reactor-shield-plug assembly and the bottom is closed by the boiler header. The gas circulators provide closures for the nozzles. The pressure vessel contains the reactor primary coolant (helium) and supports the reactor - shield-plug assembly, the boiler, the outer reflector and thermal shield, and the gas circulators.

The lower section of the pressure vessel is a cylinder 175.5 inches high with an outside diameter of 107.5 inches. The wall thickness is 2.75 inches except for the bottom flange which is 10 inches wide and 10 inches deep. The nozzles penetrate the lower section 41.0 inches above the face of the bottom flange and have an inside diameter of 30 inches.

The bottom of the pressure vessel is closed by bolting the boiler header to the bottom flange of the pressure vessel with sixty 3.0-inch studs.

The 10-inch-high, 2.75-inch-thick conical transition section joins the lower cylindrical section to the upper flange ring. The top of the vessel is closed by bolting the reactor - shield-plug assembly to the top flange of the pressure vessel with forty-four 2.75-inch studs.

The 10-inch-thick thermal shield that surrounds the outer reflector is comprised of four 2.5-inch-thick concentric cylinders fabricated from borated (1 w/o natural boron) carbon steel. These cylinders are inserted into the pressure vessel through the bottom flange and are permanently attached to the lower cylindrical section of the pressure vessel. One-half-inch wide cooling annuli are provided between the cylinders.

The outer reflector consists of an 8-inch-wide annulus of graphite perforated with coolant channels and contained within appropriate structure. The outer reflector is supported from the top flange ring and is segmented so that it can

Fig. 5.2 - 630A Mark V(B) Pressure Vessel (221R464)

be removed through the top of the pressure vessel if required.

The annulus formed by the outside surface of the flow divider and the inside surface of the lower section of the pressure vessel, provides the flow path for the primary coolant immediately after it is discharged from the gas circulators. The coolant then flows upward in parallel paths through the thermal shield and outer reflector to the reactor inlet plenum.

5.2 DESIGN REQUIREMENTS

The pressure vessel is designed for a pressure of 900 psig (pressure relief valves will be set at 900 psig), at a temperature of 700°F and meets the requirements of the United States Coast Guard, the American Bureau of Shipping and the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels". The design life of the pressure vessel is 20 years.

All bolted joints will be gasketed and seal-welded where practical. Metallic O-rings are being considered for the material to effect the seals. The studs will be pre-loaded with hydraulic stud-tensioners, instead of torquing the nuts.

5.3 MATERIALS AND FABRICATION METHODS

The pressure vessel is fabricated of SA-212, Type B carbon steel except for the flanges on each end which are forged from SA-105, Type II carbon steel and the nozzle flanges which are forged from SA-182, Type F22 low-alloy steel.

Bids on construction of the pressure vessel will be solicited from the major fabricators of pressure vessels.

5.4 NUCLEAR ANALYSIS

Preliminary calculations indicate that the pressure vessel will be exposed to an integrated flux of 1.0×10^{18} nvt, with energies greater than 1.0 Mev, during the 20-year life of the vessel.

5.5 THERMODYNAMIC DESIGN

The inside surface of the pressure vessel is exposed to the primary coolant having a temperature of about 550°F. The outer surface contacts the air within the containment vessel; this air maintains a temperature range from 120° to 150°F. The nuclear heating in the pressure vessel wall is in the order of 0.001 watt per gram at full power for the 630A Mark V(A) and about 0.01 watt per gram for the Mark V(B). Preliminary estimates show that pressure vessel temperatures for either configuration will be less than 580°F.

6.0 CONTAINMENT

6.1 DESCRIPTION

The containment vessel as shown in Figure 6.1 is a low-carbon steel pressure vessel enclosing the entire reactor-shield plug, boiler, pressure vessel, gas circulators, moderator cooling system and inner shield assembly. The containment system serves to isolate the primary loop, boiler and the moderator cooling system. The function of the containment system is to contain any leakage from these systems due to single and credible multiple failures. The containment vessel is shielded to reduce the external dose rates to acceptable levels in the event that the primary coolant loop is breached after a major fuel element cladding failure. The containment vessel interior is cooled to maintain a suitable environment for instrument and control components. The containment vessel also contains a fire-detection system.

The containment vessel is essentially a 12-1/2-foot diameter cylindrical low-carbon steel (SA-201B) shell with elliptical ends. The overall height, including a quick-opening, breech-lock-type access hatch in the top head, is 39 feet, 5 inches. The overall outside envelope does not exceed 14.0 feet in diameter except at the circulator nozzles. The void spaces within the containment vessel above and below the pressure vessel and shield tanks are interconnected by annuli at the ID and OD of the side shield. These void spaces provide a total expansion volume of approximately 1400 cubic feet.

The containment vessel consists of five components: (1) a cylindrical midsection, (2) two flanged cylinders and hemispherical heads that enclose the gas circulator drives, (3) a shielded upper head assembly, and (4) a lower dished head.

The cylindrical midsection is 21 feet, 11 inches long and is flanged on each end. Two nozzles located concentrically to the pressure vessel nozzles penetrate the midsection of the containment vessel to maintain continuity of containment around the gas circulators. The midsection wall thickness is 3.00 inches except in the nozzle region where it is reinforced.

The removable shielded upper head assembly is 13 feet, 10 inches high and is flanged at the bottom end. The walls are 3.00-inch thick low-carbon steel covered with 1.00 inch of lead shielding material and clad with .25 inch of low-carbon steel.

The 3-inch thick upper head is equipped with a quick-opening, breech-lock-type closure which permits the head to be opened for inspection and maintenance of components such as the control rod actuators and the moderator water circulating system. Penetrations for moderator water, feedwater, cooling water, controls and instrumentation are provided in this section.

The bottom is closed by a 3-inch thick dished head which is bolted to the lower containment vessel flange. A manhole is provided in the head for access for

inspection and maintenance. Penetrations for boiler feedwater and steam lines and a containment vessel drain are provided in this section.

The containment vessel is supported by the ship structure from a flange on the cylindrical midsection. Internal flanges on the cylindrical midsection support the pressure vessel and inner side shield tanks.

All flanged joints will be designed with a double-gasket seal with leakage collection plenums. These plenums and the electrical penetration boxes will be connected to containment leakage collection piping which discharges to the waste-gas handling system. This system will maintain a slight negative pressure to drain off, monitor and filter any leakage prior to release to atmosphere.

6.2 REQUIREMENTS

The containment vessel is designed to the specifications for a class "B" vessel in Section III, "Nuclear Vessels," of the ASME Boiler and Pressure Vessel Code. In order to establish overall dimensions and to determine the possibility of mechanical interferences, a maximum design pressure of 550 psi was selected. This value is conservative and will be used until the safety analyses have been carried out in sufficient detail to establish a more realistic design pressure.

The vessel will withstand an external pressure corresponding to a water depth of 150 feet. If the ship should sink in deep water, an external relief valve is provided to allow sea water to enter the vessel and equalize pressures to prevent vessel collapse. The design life of the containment vessel is 20 years.

The allowable leak rate depends upon the assumed accident conditions, provisions for filtering, holdup and mixing prior to discharge, and other details of the safety analyses. Since this work has not been completed, the criterion for the N.S. SAVANNAH (1.2 percent of the contained volume per day at 60 psig) is being used as a guide. For the upper limit 630A condition of 550 psi and using helium as the contained gas, the corresponding allowable leakage would be 5.25 percent per day of the contained gas. A test pressure of 100 psig and a leakage rate of 1.5 percent has been established as being sufficient to guarantee this rate once initial proof testing has been completed.

6.3 LEAK RATE PREDICTIONS

A preliminary estimate of helium leakage through the containment vessel flange connections and electrical penetrations indicates that the leakage should not exceed .12 percent of the contained volume per day at the design pressure of 550 psi.

The flange leak predictions are based on the results from an experimental program carried out at General Electric Advanced Technology Laboratories under a NASA contract*. In these tests, helium leakage across various types of gaskets was measured for various gasket loadings and internal pressures.

* 63GL41 through 63GL46, "Design Criteria for Zero Leakage Connectors for Launch Vehicles," Vols. 1-6, W. J. Harwick, et al, March 15, 1963.

The prediction of leakage through the electrical penetrations was based on actual test data presented by the manufacturer of Burndy Type BEH connectors. These connectors satisfy the leakage requirements specified in MIL-C-26482C. When the manufacturer's data is extrapolated to the 630A containment conditions, the leakage through all the electrical connectors is found to be less than 0.0008 percent per day of the contained volume.

7. SHIELD

7.1 INTRODUCTION

Successful operation of the 630A nuclear steam generator is dependent upon the maintenance of low environmental levels of nuclear radiation. This can be accomplished by the planned arrangement of the major power plant components and the placement of lead, borated water, and steel shielding around the reactor. The objectives of the shield design are to attenuate the normal operating radiation to meet the regulatory requirements, allow access for power plant maintenance and inspection after shutdown, and to limit the exposures that would result from postulated accident conditions. The regulatory requirements are considered as an upper limit and, in general, radiation levels are reduced as much as possible consistent with the weight and economic considerations of a practical installation.

The configuration of the shielding is shown in Figure 1.2. The sides of the reactor are shielded by a lead cylinder, annular compartments of borated water, and thicknesses of mild steel. Above the reactor and attached to it is the shield plug consisting of iron slabs, lead, and layers of moderator water. Shielding below the reactor consists of a diffuser - shield, boiler structure and tubes, and a shield plug in the center of the boiler. An annular skirt of lead and water extends from the ship's inner bottom to the lower end of the outer side shield.

7.2 MECHANICAL DESIGN

7.2.1 Side Shield

The side shield has a total radial thickness of 73.5 inches of which 49.25 inches is borated water, 5 inches is lead, and 19.25 inches is steel.

The inner portion of the shield is made up of 2-inch cylinders of mild steel placed between the reflector and the pressure vessel. An annular tank of borated water occupies the region between the pressure vessel and the containment vessel and extends from the upper pressure vessel flange to a support flange at the level of the blower openings. The tank will be split horizontally at about the blower centerline level. The lower part will be made of four segments to allow installation. The tank is constructed of 0.5 inch thick low carbon steel plate and contains a 15.25 inch thickness of borated water at the core midplane. One inch radial clearances are provided on both sides of the vessel to connect the upper and lower containment volumes and to reduce heat transfer from the pressure vessel into the shield water. The tanks are designed for 10 psig internal pressure, 150°F water, and 20 year life. The shield water is circulated and cooled by the shield water cooling system mounted outside of the containment. (A more detailed explanation of this system is given in Section 8). The tank is also connected to relief valves in the containment vessel to allow the pressure in the tank to be equalized with the containment pressure at all times.

The remainder of the required side shield thickness is located outside of the containment vessel in a second annular tank assembly. This tank is 41.0

inches thick at the reactor midplane and consists of an inner one-inch wall of low carbon steel, 5.0 inches of lead, 0.5 inch wall of low carbon steel, 34.0 inches of borated water and an outer wall of 0.50 inch low carbon steel. The assembly is divided into four segments to facilitate handling and assembly. Primary support is provided by the power plant support structure at the tank bottom.

The tank walls adjacent to the pressure vessel and the steel cladding on the OD of the lead shields have been calculated to be a significant source of secondary gammas. To reduce the secondary effect, these walls will be fabricated of steel containing approximately one weight percent of natural boron. Boron will be added to the shield water in the form of potassium tetraborate with a minimum concentration of 0.6 weight percent of natural boron.

7.2.2 Upper Shield

The shielding above the reactor is made up of the shield plug, an annular water tank between the shield plug and containment vessel, an annular water tank outside of the containment vessel above the side shield, the upper containment head, and the lead shielding on the upper containment head.

A detailed description of the shield plug is contained in Section 3.

An annular shield tank about 40 inches high fills the void above the inner side shield and between the shield plug and containment vessel. The tank is filled with borated water and interconnected to the side shield tank inside the containment vessel. It is mounted on the upper containment head and is removed and replaced with the head during refueling operations. It is fabricated of 1/2 inch mild steel and the bottom and inner walls are borated.

At the same level an annular shield of the same height is placed outside of the containment vessel. This is a continuation of the outer side shield and must be removed during refueling to provide an unobstructed plane for positioning of the refueling fixture. This tank is fabricated of 1/2 inch low carbon steel and also contains borated water. The 3-inch thick containment vessel head is supplemented with a 1-inch layer of lead held in place on the outside of the vessel wall with a 1/4-inch cover of carbon steel.

7.2.3 Lower Shield

There is less shielding below the reactor than radially because of the higher allowable dose rates below the plant. The arrangement of the plant allows the utilization of the steel and water in the boiler as shielding material to reduce the overall shield weight. Directly below the core, a diffuser-shield prevents direct radiation from impinging on the boiler. The center region of the boiler will be filled with aluminum oxide in the form of pellets. This will leave a void volume fraction of about 25% for the region. It is also planned to put boron in the aluminum oxide to reduce neutron captures in the boiler material.

There are two concentric annular voids outside of the boiler. The inner one forms the inlet plenum for the gas circulators and the outer one forms the outlet plenum and passage to the reactor. The upper portion of the inlet

plenum and the lower portion of the outlet plenum are not required for gas flow, and both of these spaces are filled with shielding material.

There are voids in the side shield around the gas circulators for installation of the circulators and the containment enclosures. These voids may be filled with blocks of borated graphite or borated polyethylene. A 2-inch layer of lead and 12 inches of water cover the containment cap that encloses the gas circulators.

Current calculations indicate that the dose rate in the region beneath the power plant will be troublesome. Radiation directly out of the bottom can be handled by utilizing the ship's double bottom as tankage for borated water. Radiation in a radial plane from this region will probably require the utilization of an annular shield skirt that extends between the double bottom and the lower part of the side shield. This skirt may be an integral part of the nuclear steam generator foundation. Material composition and thicknesses of the skirt are yet to be established.

7.2.4 Shield Cooling System

Because of the thermal and nuclear heat load in the shield inside of the containment vessel a shield water cooling system will be supplied. Aside from the cooling functions, it provides the means for filling and draining the shield tanks, sensing and adjusting boron concentration, controlling level, and adding make-up water. A more complete description of the system may be found in Section 8.

7.2.5 Design Requirements

All components will be designed for a 20 year expected life and 10 psig internal pressure. The design temperature of the borated water is 150°F. All tanks will be designed to meet the applicable specifications, standards and ship motions outlined in Section 1.

7.2.6 Assembly and Installation

All components are prefabricated prior to shipment to the site for assembly into the power plant with the exception of the lower skirt shield. Should that be required the lower skirt shield could be made a part of the 630A's foundation or the ship's structure.

The containment vessel cylinder is the first major component installed in the ship. The lower segments of the side shield tank may be installed in the containment either before or after this operation, however, it must be done before the pressure vessel assembly is lowered into place.

The diffuser shield, boiler center shield, and the boiler duct shields are installed as the boiler is fabricated. After the pressure vessel is in place, the inner annular side shield tank is installed and the shield water piping system connected.

The upper portion of the inner side shield tank is preassembled to the upper containment head and is positioned in place as the head is installed. Piping connections for the shield water systems are made so that all internal tanks are interconnected.

The outer four side shield segments are installed at any time after the cylindrical portion of the containment vessel is in place. These segments are bolted to the foundation and to each other to form a continuous annular structure.

The upper outer annular tank is installed after the upper containment head is in place. This tank is in four pieces for ease of handling. The segments are bolted together after they are in place and are supported by and bolted to the outer side shield.

7.3 NUCLEAR SHIELDING DESIGN

7.3.1 Nuclear Shielding Criteria

The radiation protection specifications for the 630A have been established to meet the criteria set forth in the Code of Federal Regulations, Title 10, part 20 - "Standards for Protection Against Radiation."

The primary requirements in shielding design are:

A. For individuals with access to restricted areas:

- (1) During any calendar quarter the dose to the whole body shall not exceed 3 rem.
- (2) The dose to the whole body, when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rem where N equals the individual's age in years at his last birthday (it is assumed that the accumulated occupational dose of each individual will be known.)

B. The levels in unrestricted areas shall be limited to values that are not likely to cause any individual to receive a dose to the whole body in excess of 0.5 rem in one calendar year; in excess of 100 mrem in any 7 consecutive days of continuous occupancy; or in excess of 2 mrem in any one hour of continuous occupancy.

Guidance and recommendations from the following documents have also been used in the shield design:

- A. Federal Radiation Council Staff Report No. 1, "Background Material for the Development of Radiation Protection Standards," May 13, 1960.
- B. N.B.S. Handbook 59 - "Permissible Dose from External Sources of Ionizing Radiation."
- C. American Bureau of Shipping - "Guide for the Classification of Nuclear Ships."
- D. U. S. Coast Guard - "Maritime Engineering Regulations and Material Specifications," CG-115, March, 1963.

7.3.2 Nuclear Shield Design Specifications

In order to meet and interpret the general regulatory criteria, nuclear shielding specifications have been formulated for the 630A configuration and are outlined in this section.

A. Definitions

Unrestricted Area - Space accessible to passengers or members of the crew while off duty, including areas outside the ship's hull, above the light ship waterline.

Restricted Area - Any area which has controlled access for the purpose of protection of individuals from exposure to radiation.

Occupational Area - Space which personnel may be required to occupy in the performance of normal occupational duties.

Limited Access Area - Space to which only intermittent access is permitted during normal operation.

Exclusion Area - Space to which access is permitted only after specific approval by the Radiological Safety Officer.

B. Radiation Zones

Four zones, shown in Figure 7.1, have been identified around the 630A for the purpose of specifying the radiation constraints for a typical installation. These constraints are based on a 70 percent utilization factor which is equivalent to operating at full power for 70 percent of the year.

Zone 1 - The region within a 20-foot radius of the center of the core (approximates the size of the reactor compartment). There are areas within this region where the dose rate may exceed the 56 hr/wk occupational limit and limited access will be permitted during operation.

Zone 2 - The region above the midplane of the reactor outside of the 20-foot radius. This region will be restricted to radiation workers but continuous occupancy 24 hours per day, 365 days per year will not result in exceeding 5 rem/year. The outer boundaries of the restricted area will depend on the ship installation and construction.

Zone 3 - The region below the midplane of the reactor where the radiation will be entering the engine room and other occupational areas. This region will be a restricted area for occupational radiation workers. Dose rates will be limited to 2.3 mrem per hour based on individual work periods of 56 hours per week for 52 continuous weeks.

Without taking any special precautions, the total yearly dose for radiation workers under these assumptions would not exceed 5 rem per year. Under actual operating conditions, it is expected that the doses received during normal operations would be much less than 5 rem per year and would allow a safe margin for periodic maintenance work in higher radiation fields.

Fig. 7.1 - Definition of Radiation Zones

Zone 4 - A conical region directly below the reactor where access should not be required during operation. The dose rates in this region will be maintained at a level that does not cause a significant, scattered contribution in the occupational areas.

C. Accident Considerations

In addition to the constraints for each zone for normal operation, no person should receive an integrated whole body dose exceeding 25 rem following a maximum credible accident. Under postulated accident conditions, the crew should have adequate time to secure the plant in a safe and orderly manner before leaving their work stations.

7.3.3 Shielding Calculations and Results

Detailed shielding calculations have not been made for the specific Mark V(B) configuration shown in the general arrangement drawing. Previous calculations for similar configurations were used to establish dose rates, material thicknesses and other shielding information. On this basis it is expected that there would be minor changes in the quoted values and modifications of the shield geometry in the final design. The shield outside diameter, however, should remain approximately constant as it is determined mainly from the fast neutron dose rate, which is relatively insensitive to trades between iron, steel, and lead thicknesses in the radial shield. The secondary gamma dose is dependent upon the location and thicknesses of materials, particularly iron, water, and boron, and extrapolations from one configuration to another are less accurate. If a more detailed analysis indicates a variation in the secondary gamma dose at the shield surface adjustments in the lead shield thickness would be made.

a. Operating Primary Dose Rates in Radial Shield

Direct beam dose rates at the surface of the radial shield in the reactor midplane have been estimated using attenuation factors calculated for a similar geometry. The fast neutron and gamma dose are given as a function of radius from the core centerline in Figure 7.2. The results are based on a reactor power of 60.4 MW.

b. Thermal Flux in Radial Shield

The thermal flux in the radial shield shown in Figure 7.3 has been estimated using computer results from a 7-group, one-dimensional diffusion calculation. The absolute values near the active core were determined by normalizing results to core power. Absolute values in the shield region were determined by normalizing the diffusion program results to the fast neutron dose rate shown in Figure 7.2.

c. Secondary Gamma Dose Rate

The gamma dose rate resulting from neutron capture in the power plant structural and shielding components is a major part of the dose rate outside of the shield. In order to estimate the magnitude of the secondary contribution on the core midplane the radial components 22 inches above and 22 inches below the core midplane were evaluated as sources.

Fig. 7.2 - 630A Mark V(B) primary dose rates on core midplane 60.4 mw

Fig. 7.3 - 630A Mark V (B) thermal flux on core midplane 60.4 mw

The neutron capture rate and resulting gamma source in each component were first determined and the attenuation between the source and shield surface was estimated. The contribution is tabulated for each in Table 7.1. There will also be a significant contribution from the boiler-diffuser shield and the lower head of the upper plug, however, the evaluation of these components has not been completed.

TABLE 7.1
ESTIMATED SECONDARY GAMMA DOSE RATES
ON SHIELD SURFACE AT REACTOR MIDPLANE.
(See Point 3 on Figure 7.4.)

<u>Component & Material</u>	<u>Thickness, inches</u>	<u>Dose Rate mrem/hr.</u>
Graphite retainer structure, 304 SS	0.50	.215
Thermal shields - borated steel	10.0	1.209
Pressure vessel - steel	2.75	1.808
Inner shield tank inner wall - borated steel	0.50	.666
Borated water in inner shield tank	15.25	neg.
Inner shield tank outer wall - steel	0.50	.001
Containment vessel wall - steel	3.00	.004
Tank wall at ID of lead slab - steel	1.00	neg.
Lead shield	5.0	neg.
Tank wall at OD of lead slab - borated steel	0.50	.144
Borated water in outer shield tank	34.0	.030
Tank wall at OD of side shield - steel	<u>0.50</u>	<u>neg.</u>
TOTAL		4.077

Although the boiler contains a large number of passages for the primary gas and secondary steam, it does provide partial shielding for the region below the boiler. Direct beam dose rates have been estimated at several points using computer calculations for a similar configuration.

From these results it is evident that there will be a significant thermal flux with secondary gamma sources being generated in and below the boiler region. A lower shield skirt has been proposed to attenuate these secondary gammas as well as scattered gamma and neutron radiation from the core.

Contributions from each of the three boiler regions to secondary gamma dose rates outside of the shield are shown in Table 7.2. The evaporator is the largest source and therefore the major contributor to the dose rate at receiver point 4 as indicated on Figure 7.4. However, at point 6 the economizer becomes the major contributor due to self attenuations in the boiler.

TABLE 7.2

BOILER CAPTURE GAMMA DOSE RATES

<u>Secondary Source</u>	<u>Volume Fraction of Water</u>	<u>Dose Rate, mrem/hr.</u>	
		<u>Point 4</u>	<u>Point 6</u>
Superheater	0	0.423	0.001
Evaporator	.112	3.450	0.013
Economizer	.142	<u>0.007</u>	<u>0.096</u>
TOTAL		3.880	0.110

d. Shield Surface Dose Rate Distribution

The results of the primary gamma, secondary gamma, and direct beam neutron estimates have been tabulated at several locations on the shield surface and are shown on Figure 7.4. When extrapolated to 20 feet the doses are approximately within the specifications outlined in Section 7.3.2.

e. Heating Rates in Radial Shield

Estimated heating rates in the radial shield on the reactor midplane are given in Figure 7.5. Primary gamma, secondary gamma and n- α heating were considered to be the major sources. Kinetic energy heating was neglected.

f. Integrated Fast Neutron Dose

The steel cylinders between the reactor core and the pressure vessel are very effective in reducing the heating rate and the integrated fast neutron dose at the pressure vessel. Assuming 120,000 hours of operation at rated power over the 20 year power plant life, the maximum nvt (greater than 1 mev) in the pressure vessel at the core midplane is 2.3×10^{17} .

The void regions below the core allow the fast neutron dose to penetrate more deeply in the boiler region. Assuming homogenized compositions for the boiler and boiler diffusers, the nvt was estimated at various distances below the core. The results are plotted in Figure 7.6. No attempt was made to determine a radial distribution.

g. Dose Rates from Fission Products in the Containment Vessel

Dose rates outside of the containment vessel were calculated for the hypothetical conditions of fuel element melting followed by rupture of the pressure vessel and release of fission products into the containment void. The calculation was based on infinite operation prior to the accident. The assumed net fission product release to the volume inside the pressure vessel following a failure was assumed to consist of 100 percent

FIGURE 7.4
SHIELD SURFACE DOSE RATE
(221R467)

(1)	A	.017
	B	.068
(2)	A	.0087
	B	1.4
(3)	A	.45
	B	1.4
	C	4.08
	D	.5*
	E	6.4
(4)	A	.0077
	B	.63
	C	-
	D	3.9
(5)	A	.018
	B	.0016
(6)	A	.0000029
	B	.00015
	C	-
	D	.11
(7)	A	.000002
	B	.0000027
(8)	A	830.0
	B	3.7
(9)	A	.43
	B	.12
(10)	A	.02
	B	.00073

A Fast neutron dose rate in mrem/hr
 B Primary gamma dose rate in mrem/hr
 C Secondary gamma dose rate from radial components in mrem/hr
 D Secondary gamma dose rate from the boiler only in mrem/hr
 E Total dose rate in mrem/hr

* Estimated upper limit

Fig. 7.5 - Nuclear heating rates radial shield 60.4 mw

FIGURE 7.6
INTEGRATED FAST
NEUTRON DOSE IN BOILER
630A MARK II B

of the noble gases, 50 percent of the halogens, 50 percent of the volatile solids, and 1 percent of the non-volatile solids. It was further assumed that if the pressure vessel ruptured, allowing the containment vessel to become contaminated, 50 percent of the released halogens and 70 percent of the released solids would remain inside of the pressure vessel. The net release for these conditions is given in Table 7.3.

Data on the important fission products was obtained from "Decay of Fission Product Gammas," NPA 27-39 by F. H. Clark. Although this is not the latest data available, it is in the very useful form of source strength in $\text{Mev sec}^{-1} \text{ watt}^{-1}$ for each isotope and seven gamma energy groups as a function of time after shutdown. Source terms are given from 1/2 hour after shutdown to 100 days after shutdown. The energy group structure is shown in Table 7.4.

Classification of the fission products was based on the characteristics of the oxides of the fission products with consideration given to the parent-daughter relationship. The fuel element cladding will melt at about 2500°F , which is well below the melting temperature of UO_2 . It was assumed that the UO_2 would oxidize forming U_3O_8 and resulting in a volume increase which would contribute to the release of fission products even though the oxidized uranium did not melt. Table 7.5 lists the fission products by class together with the parent isotope, where applicable, the half lives of both, and the energy groups to which the isotope contributes.

It should be noted that Rb^{88} and Cs^{138} have both been classified as noble gases because the parents are Kr and Xe, respectively, and that Rh^{106} is classified as a volatile solid because the parent is Ru.

As will be shown in the next section, energy groups IV and V are the major contributors to the dose rates during the first 12 hours after shutdown. For this reason, Figures 7.7 and 7.8 have been prepared to show the isotopes which are important in these energy groups. It will be noted that Kr^{88} , Rb^{88} and Cs^{138} are responsible for nearly all the dose rates during the first several hours after shutdown.

Dose Rates Outside the Containment

Calculations of dose rates at 50 feet from the center of the upper containment were made assuming sources based on the instantaneous melting of part of the reactor and uniform distribution of the released fission products to the combined volumes of the pressure and containment vessels. The total containment void volume of 1400 ft^3 is a little more than three times the primary loop volume of 450 ft^3 and the upper containment void is about half the total containment void volume. Therefore, about 40 percent of the total void volume or total source in the aerosol is represented by the upper containment volume. The upper containment head is assumed to consist of 3.0 inches of steel backed by 1.0 inches of lead.

Machine calculations were made only for the one hour after shutdown case, and factors were hand calculated for converting to other times after shutdown. The calculations were made on a per MW basis assuming infinite operation. Results of dose rate vs. time after shutdown are plotted in Figure 7.9. These dose rates are those which could be expected from the

melting of 1/60.5 of a reactor that had been operating continuously at 60.5 MW.

Figure 7.9 also shows the contributions to the dose rate from the most important energy groups. This indicates that groups IV and V contribute the major portion of the dose rate during the first 12 hours after shutdown. A breakdown of the sources within these groups has been shown in Figures 7.7 and 7.8.

The integrated dose per MW vs. time after shutdown is shown in Figure 7.10. Obviously the dose builds up rapidly in the first few hours and then increases at a much reduced rate. Figure 7.11 is a plot of estimated lead thickness required versus 24 hour integrated dose at 50 feet.

Effect of Plate-Out Within the Pressure Vessel

To determine the effect of the plate-out assumption on dose rates, the analysis was repeated assuming no plate-out within the pressure vessel. In other words, the percent released from the meltdown in column 1 of Table 7.3 was considered to be the net percent released to the combined pressure vessel and containment vessel volume.

Again, energy groups IV and V are major contributors as shown in Figure 7.14. The source terms for these groups are detailed in Figures 7.12 and 7.13. Cs^{138} , Kr^{88} and Rb^{88} are still predominant for the first few hours but the halogens become important sooner after shutdown than when plate-out is considered.

The resulting dose rates shown in Figure 7.14 are very little more shortly after shutdown, but dose rates at 24 hours are increased by a factor of 4 when no plate-out is considered.

In Figure 7.15 the cumulative dose for the no plate-out case is shown along with the cumulative dose for the case with plate-out. It is apparent that plate-out has little effect during the first few hours but that it becomes increasingly important as the time after shutdown increases. Initially, the noble gases are the major contributors but after these decay, the categories which plate-out predominate.

Comments on the Assumptions

As mentioned previously the latest data on fission products was not used because it would mean redoing many of the calculations made in the preparation of NDA 27-39. It is not expected that the new data would materially affect the results. Also, no detailed analysis of the gamma ray sources for the first half hour after shutdown was performed. The curves of NDA 27-39 were extrapolated back to zero time after shutdown which is not felt to introduce serious error.

On the other hand, the assumption that Rb^{88} and Cs^{138} are noble gases is conservative. Actually the accumulated inventories of these isotopes at the time of shutdown should be treated as volatile solids. Only that portion which results from decay after shutdown should be put in the noble gas category.

The assumption that the fission products which do not plate out will be uniformly distributed throughout the combined volume of the primary loop and the containment vessel is also considered conservative. If the relief valves function properly, a higher pressure will be maintained in the pressure vessel than in the containment and fewer fission products will get into the containment.

The assumption of an instant melt is also unrealistic. While it may be possible that a portion of the reactor would melt almost immediately in case of an excursion, the reactor would shut down and further melting would not occur. This points up the fact that this analysis is not based on any safety analysis results but only on assumptions which establish what is believed to be an upper limit for dose rates from fission products in the containment vessel.

TABLE 7.3

RELEASE OF FISSION PRODUCTS FROM THE FUEL

<u>Class of Fission Products</u>	<u>% Released from Melt</u>	<u>% Not Plated Out on Cool Surfaces</u>	<u>Net % Released</u>
Noble gases	100	100	100
Halogens	50	50	25
Volatile Solids	50	30	15
Other Solids	1	30	0.3

TABLE 7.4

GAMMA ENERGY GROUP STRUCTURE

<u>Energy Group Number</u>	<u>Group Boundaries, Mev.</u>
I	0.1 - 0.4
II	0.4 - 0.9
III	0.9 - 1.35
IV	1.35 - 1.8
V	1.8 - 2.2
VI	2.2 - 2.6
VII	above 2.6

TABLE 7.5

CLASSIFICATION OF FISSION PRODUCTS

NUCLIDE		HALF LIFE	PARENT	HALF LIFE	GAMMA ENERGY GROUP						
					I	II	III	IV	V	VI	VII
<u>Noble Gases</u>											
Xenon	Xe 135	9.13 h	I	6.68 h	x						
Krypton	Kr 85m	4.36 h	-		x						
	88	2.77 h			x				x		
Rubidium	Rb 88	17.7 m	Kr	2.77 h			x		x		x
Cesium	Cs 138	32 m	Xe	17 m		x	x	x			
<u>Halogens</u>											
Bromine	Br 84	32 m	-			x			x		
Iodine	I 131	8.14 d	Te	24.8 m	x	x					
	132	2.4 h	Te	77.7 h		x		x	x		
	133	20.8 h	-			x		x			
	134	52.5 m	Te	44 m			x			x	
	135	6.68 h	-				x		x	x	
<u>Volatile Solids</u>											
Ruthenium	Ru 103	39.8 d	-			x					
	105	4.5 h	-			x					
Rhodium	Rh 106	30 s	Ru	1 y		x	x	x		x	
Tellurium	Te 129m	33.5 d	Sb	4.2 h	x						
	129	72 m	Sb	4.2 h	x						
	131m	30 h	-		x						
	131	24.8 m	-		x	x					
	132	77.7 h	-		x						
Molybdenum	Mo 99	68.3 h	-		x	x					
Cesium	Cs 136	13.7 d	-					x ¹			
<u>Other Solids</u>											
Strontium	Sr 91	9.7 h	-			x	x	x			
Yttrium	Y 91m	51 m	Sr	9.7 h		x					
	91	61 d	Sr	9.7 h	x		x				
	93	10 h	Sr	7 m		x					
	94	16.5	-					x			
Zirconium	Zr 95	65 d	Y	10.5 m	x	x					
Niobium	Nb 95m	90 h	Zr	65 d	x						
	95	35 d	Zr	65 d		x					
	97m	60 s	Zr	17 h		x					
	97	72.1 m	Zr	17 h		x					
Barium	Ba 139	85 m	Cs	9.5 m	x						
	140	12.8 d	-		x	x					
Lanthanum	La 140	40.2 h	Ba	12.8 d	x	x		x		x	
Cerium	Ce 141	32.5 d	La	3.7 h	x						
	143	34 h	La	19 m	x	x					
Praseodymium	Pr 144	17.5 m	Ce	275 d		x		x	x		
Neodymium	Nd 147	11.6 d	-		x	x					
Samarium	Sm 153	47 h	-		x	x					
Europium	Eu 156	15.4 d	Sm	10 h					x		

FIGURE 7.7
GROUP IV SOURCES
1.4 - 1.8 Mev

FIGURE 7.8
GROUP V SOURCES
1.8 - 2.2 Mev

FIGURE 7.9
DOSE RATE AT 50 FEET

Gamma Dose Rate, Rads(carbon)/hr

2.0
1.6
1.2
0.8
0.4
0°

359-11
MADE IN U.S.A.
: X 10 TO THE 1/2 INCH
HUFFEL & ESSER CO.
K₁

Hours after Shutdown

FIGURE 7.10
INTEGRATED DOSE AT 50 FEET

Gamma Dose, Rads (Carbon)/MW

K
O X 10 TO 1/2 INCH 45 1323
1 X 10 INCHES
MADE IN U.S.A.
KEUFFEL & ESSER CO.

FIGURE 7.11
24 HOUR INTEGRATED DOSE
AT 50 FEET

Gamma Dose, Rads (Carbon)/MW

Reference Design

Lead Thickness, Inches

FFE-6585F 2D, 4, Y, NO. 359-61
mls. 2 1/2" to 1 1/2" to the inch, 5th lines associated
JAN 1961

FIGURE 7.12
GROUP IV SOURCES
1.4 - 1.8 Mev
NO PLATE OUT

OXFORD INSTRUMENTS INC. 259.11
CALIFORNIA 95800

Mev sec⁻¹ wall⁻¹ x 10⁻⁸

Noble
Gases
(Cs¹³⁸)

Halogens
(I¹³¹, I¹³³)

Total

0 2 4 6 8 10 12

Hours After Shutdown

FIGURE 7.13
GROUP V SOURCES
1.8 - 2.2 Mev
NO PLATE OUT

Total

Noble Gases
(Kr⁸⁸, Rb⁸⁸)

Halogens (Br⁸⁴, I¹³², I¹³⁵)

Hours after Shutdown

10 X 10 FC 1/2 INCH 46 1323
7 X 10 INCHES
KEUFFEL & ESSER CO.
MADE IN U.S.A.

Mev sec⁻¹ watt⁻¹ x 10⁻⁵

20
15
10
5
0

0 2 4 6 8 10 12

FIGURE 7.14
DOSE RATES AT 50 FEET
NO PLATE OUT

FIGURE 7.15

INTEGRATED DOSE AT 50 FEET

Gamma Dose, Rads(carbon)/M

10 x 10 TO 1/2 INCH 26 1323
1 x 10 INCHES MADE IN U.S.A.
GEUFEL & ESSER CO.

Time after Shutdown, Hours

No Plate Out

With Plate Out

7.3.4 Material Selection and Design Optimization

In general, lead and water form the best combination of materials for attaining a light weight and low cost shield for a maritime nuclear power plant. The heavy lead gamma shielding may be placed near the reactor where the weight will be the smallest for a given thickness. The lighter water neutron shield would then occupy the larger volume at greater distances from the core. In a bulk unit shield where attenuation in the order of 10^{13} takes place, the problem of secondary gamma production from neutron capture becomes of significant and sometimes primary importance. This is particularly true when steel structural or shielding materials are interspersed throughout the shield. Steel does act as a gamma shield but, depending upon the incident neutron flux, it may become a new source of secondary gammas that overshadows the core gamma sources. The use of lead in place of iron as a gamma shield reduces this tendency to produce a secondary gamma source because of its thermal neutron absorption cross section of only 0.170 barn compared to 2.43 barns for iron. Thus, secondary gamma production in the lead would be about one-fortieth of the production of iron of equal thickness. The mass absorption coefficient for lead is approximately 50 percent higher than that of iron for the predominant primary gamma rays. Therefore, properly locating the lead with its better mass attenuation coefficient for core-produced gammas and its decreased secondary gamma production results in the minimum shielding weight.

Other factors such as temperature limitations and mechanical design constraints influence the material selection. In the MARK V(B) configuration, steel was placed around the reflector in a region of high neutron flux. Temperatures in this region preclude the use of lead and the steel provides good, high energy neutron shielding to reduce the fast flux (and nvt) at the pressure vessel to allowable limits for carbon steel. The steel in this region also acts as a thermal shield to reduce nuclear heating in the pressure vessel wall. The high neutron capture rate in the steel outside of the reflector quickly depresses the thermal flux from the core and reflector, so that most of the captures occur in the inner rings of the steel shielding. The outer rings near the pressure vessel provide shielding for the secondary gamma generated in the inner rings. During passage through the steel, inelastic scattering of the high energy neutrons has caused an accumulation of neutrons below about 0.5 mev that are not slowed down and are not captured. As this energy group enters the shield water outside of the pressure vessel, they are immediately slowed down by the water to an energy level that is easily captured by the walls of the water tank. To reduce the very high secondary gamma sources that would be produced, boron containing steel will be used in the tank wall and boron will be introduced into the shield water in the form of potassium tetraborate ($K_2B_4O_7 \cdot 4H_2O$). The high capture cross section of boron will cause the thermal neutrons to be eliminated by the n- α reaction with boron 10 with a subsequent release of only 2.79 mev/capture, compared to about 8 mev/capture for steels. The gammas emitted from the boron are at about 1/2 mev and are easily attenuated whereas the high energy 6 to 7 mev gammas from the steel are very penetrating.

The same accumulation of low energy neutrons occurs in the containment vessel wall. The effect here is not as pronounced, however, the secondary gammas produced at the larger diameter are still significant in producing gamma dose at the shield surface. For this reason, as well as mechanical design requirements

and fabrication economics, the lead portion of the gamma shield has been placed outside of the containment vessel. Again, borated water and boron containing tank walls will reduce secondary gamma production. One other advantage of placing the lead at this diameter is that the gamma shielding effectiveness of the water between the containment vessel and pressure vessel is fully utilized. Compton scattering is the mechanism by which the water attenuates the gamma flux and the softer gammas that result from this process would travel a comparatively long distance in water. Lead with its very high cross section in the low energy range stops the low energy gammas very quickly.

To optimize both weight and cost, the allowable surface dose is composed of about 90% gamma dose and 10% neutron dose. With the neutron dose very low and with boron in the outer shield water, the addition of boron to outer walls of the outer shield tank is not required. The boron in the water in the outer tank is required to reduce the production of 2 mev capture gammas in the hydrogen.

7.3.5 Analysis Methods

Radiation dose rates were calculated using a computer program* based on a point kernel approach. In this program the core is simulated by a large number of individual source points distributed throughout the geometry of the core. Each source point represents a small region of the core and is weighted according to the core power distribution. For each source point the distance to the receiver point and the distance through each material in the shield is calculated. These distances and the material attenuation functions are used to compute the attenuation from the source point to the receiver point. The program then integrates over the source points in the reactor to yield the total dose rate at the receiver point.

The attenuation function for the neutron dose rate is an Albert Welton kernel. This kernel is a mathematical fit to the experimental neutron measurements in water from a reactor source. The non-water or non-hydrogenous part of the attenuation is calculated using an exponential function and a removal cross section. The removal cross sections were experimentally determined.

The attenuation function for gamma rays is an exponential function using linear absorption coefficients and buildup factors. For the gamma ray calculation, the gamma ray energy spectrum of the reactor is broken down into 14 energy groups and the attenuation calculations are carried out for each group. The gamma ray sources in the reactor used in this calculation included prompt-fission gammas, fission-product decay gammas, and gamma rays resulting from non-fission neutron captures in the reactor. It should be noted that a shielding analysis based strictly on this type of calculation for fission process and fission product decay gammas may be low. The contribution by capture gammas outside of the active core can be of the same magnitude as the direct doses as indicated by the values listed in Table 7.1.

* J. T. Martin, J. P. Yalch, and W. E. Edwards, "Shielding Computer Programs 14-0 and 14-1, Reactor Shield Analysis," GE-ANPD, XDC 59-2-16, January 1959.

The gamma ray dose rate resulting from neutron capture is determined by first, calculating the number of radiative captures per second in the shield and structural materials outside of the core. The capture rate is determined by computing the neutron energy spectrum at the point of interest using a one-dimensional multi-group diffusion program. For points near the core, absolute values of the flux are obtained by normalizing the diffusion program results to the operating reactor power. At larger distances where the absolute values from the diffusion program become less accurate, the neutron spectrum is normalized with the aid of the 14-0 shield program results. The fast neutron dose is calculated with 14-0 and compared to the FND that the diffusion program would give after converting flux to dose. A ratio of these two values becomes the normalizing factor that can be used to adjust any part of the neutron spectrum to an absolute value. When this is done, the product of the absolute fluxes and material capture cross sections yield the capture rate at the point of interest. The capture rate can be translated to gamma energy release and used as the secondary gamma ray source input in the shielding computer code to obtain dose rates outside the shield.

This program combination for calculating normalized neutron fluxes may also be used to obtain other quantities such as activation, heating rates, and integrated fast neutron dose.

7.4 THERMODYNAMIC DESIGN

About 0.1 percent of the total reactor power is generated in the side shield. The temperature of the circulating water that cools the inner tanks of the shield increases about 15°F during maximum power operation. The cooling water flow rate is 30 gpm. There is no need to cool the outer shields because the heat deposition is sufficiently low that self cooling is sufficient. Internal heat generation in the side shield is dissipated to the cooling water primarily by free convective heat transfer. The film coefficient is about 80 Btu/hr-ft²-°F.

No problems are envisioned in keeping the low boron concentration in solution at the minimum temperatures contemplated in the shield water.

8. NUCLEAR STEAM GENERATOR ACCESSORIES

The 630A Mark V accessory systems and components, with the exception of the condensate demineralizer, are shown schematically in Figure 8.1. They perform such functions as circulating primary-loop helium, circulating and cooling moderator water, cooling and filtering helium bled from the primary loop, pressure relieving both primary and secondary loops, and cooling miscellaneous components.

8.1 PRIMARY COOLANT SYSTEM

The primary coolant system consists of two gas-bearing circulators for circulating helium through the primary loop during both normal and emergency operation, a variable frequency power supply for the circulators, helium makeup storage, and a vacuum pump for initially evacuating air from the primary loop.

8.1.1 Gas Circulators

The 630A utilizes two electrically-driven, gas-bearing supported radial flow gas circulators arranged in parallel. The drive motor and supporting bearings are completely contained within the primary loop, thus eliminating the need for a shaft seal. The journal bearings are of the tilting pad, self-acting type capable of supporting the shaft over the full speed range without external pressurization. The thrust bearings are of the multiple-pad, self-acting type designed to carry thrust loads in either direction.

The circulators are designed to operate with either air or helium over the pressure range of 14.7 psia to 915 psia. They will normally be operated in parallel at the same speed and will receive their power from the same source. Valves are provided in the inlet duct of each circulator to permit operation with a single circulator should it become necessary.

During normal operation electrical power for driving the circulators will be furnished by a variable frequency supply, which provides variable circulator speed, consisting of a motor driving an alternator through an eddy-current coupling. For operation in the event of failure of the variable speed drive power supply, provisions are made for switching to the 60-cycle bus which will provide sufficient circulator speed for lower power operation and for aftercooling.

8.1.2 Helium Supply System

Helium stored in high-pressure bottles as shown schematically in Figure 8.1 will be used as makeup for normal leakage.

8.2 AUXILIARY COOLING SYSTEM

The auxiliary cooling system consists of loops which cool the moderator water, and the various components such as eddy-current coupling, side shield, waste gas filter and gas circulators.

Fig. 8.1 - Accessory system 630A Mark V Nuclear Steam Generator (221R426)

GE-NMPO

8.2.1 Moderator System

The moderator water cooling and circulating system consists of the components for circulating, cooling and maintaining the required conditions of the reactor moderator water. The system components required to operate after an emergency shutdown are enclosed within the upper containment head in order to provide shielding and containment. This system is shown schematically in Figure 8.1.

The major components of the moderator system are moderator heat exchanger, expansion tank, circulating pumps, relief valves, deionizer, and associated piping and valves. The heat exchanger transfers heat from the moderator water to the boiler feedwater, thus acting as a first-stage boiler feedwater heater. Three pumps are provided. Two of the pumps are each sized to deliver the required flow and thus only one is operated at a time. The third, smaller pump is sized to provide flow required after shutdown.

A small percent of the water delivered by the operating pump is passed through a cooler and deionizer to the expansion tank. From there it is returned to the moderator heat exchanger inlet. The remainder of the water is delivered to the upper plenum of the shield plug. From there it passes through the reactor core and returns to the lower plenum of the shield plug. It then is returned through the moderator heat exchanger to the pump inlet thus completing the loop. Check valves in the pump discharge lines prevent back flow through the standby pumps. A small vent pipe connects the upper plenum of the shield plug to the expansion tank.

Positive pressure is maintained in the expansion tank by the use of hydrogen gas. Water level in the tank is sensed with detectors located on both the fore and aft sides of the tank. A low-water-level signal causes makeup water to be added automatically. A check valve is provided in the expansion tank which allows gas to flow from the containment vessel into the expansion tank in the event containment vessel pressure exceeds expansion tank pressure.

8.2.2 Containment Vessel Cooling System

As shown schematically in Figure 8.1, the upper part of the containment vessel is cooled by circulating the entrained gas over a heat exchanger where heat is transferred from the gas to component cooling-system water. The system is sized to maintain the gas temperature at or below a maximum of 150°F.

8.2.3 Components Cooling System

The heat generated in the auxiliary components is removed as shown schematically in Figure 8.1. The system handles the excess heat from the side shield water, eddy-current coupling, waste-gas handling system, the gas circulators, and from the moderator water when the power plant is shut down. This is accomplished by first transferring the heat from each system through its own heat exchanger to an intermediate fresh-water cooling loop and then transferring the total heat through a heat exchanger to sea water or to air. This intermediate loop is provided to prevent a single leak from contaminating all the systems with sea water. The intermediate-loop water is continually monitored to detect leakage of sea water into it.

The flow of sea water through the heat exchanger will be held constant. Water pressure in the intermediate loop will be held at a constant value in the heat-exchanger inlet header. The flow of this water through the eddy-current coupling and shield-water heat exchangers will be set manually at the maximum predicted flow rates required, while flow through the circulators will be regulated automatically to provide the desired temperature.

This system also supplies sea water to the waste-gas handling system to cool the charcoal absorption beds.

8.3 WASTE HANDLING

8.3.1 Waste-Gas Handling System

The waste-gas handling system shown in Figure 8.1 is designed to allow discharge of the primary loop helium to the atmosphere whenever it is desired to depressurize the primary loop, and to filter gas from the containment leakage collection piping before it is discharged to atmosphere. This system consists of a heat exchanger, water trap, gas compressor, charcoal adsorption bed and several flow control valves.

The charcoal adsorption bed is designed to take advantage of the fact that noble gases in a helium carrier stream are delayed in their passage through a charcoal bed. This delay is caused by the repeated adsorption and desorption of the noble gas molecules by the charcoal. Migration of the noble gas molecules occurs only during the time they are not absorbed and in the direction of the carrier gas flow. This phenomenon has been well investigated and many reports covering the investigations issued.*

The charcoal adsorption bed will also act as an excellent filter for removing the halogens and volatile solids from the helium. Although an exact removal efficiency has not been determined, it has been shown to be better than 99.99 percent for halogens through a 6-inch-long charcoal filter.** The 120-inch length of the proposed adsorption bed should have a much greater efficiency than the 6-inch bed.

* R. D. Ackley, R. E. Adams, and W. E. Browning, Jr., "The Disposal of Radioactive Fission Gases by Adsorption," Sixth AEC Air Cleaning Conference, TID-7593, July 7-9, 1959.

R. E. Adams and W. E. Browning, Jr., "Evaluation of Activated Charcoal Fission Gas Adsorbers Designed for the GC-ORR Loop Experiment No. 1," Oak Ridge National Lab., CF 60-1-24, March 1960.

R. E. Adams and W. E. Browning, Jr., "Re-evaluation of the Activated Charcoal Fission Gas Adsorber for Leak Detection System of the GC-ORR Loop No. 1," Oak Ridge National Lab., CF 61-4-38, April 1961.

** R. E. Adams and W. E. Browning, Jr., "Removal of Radioiodine from Air Streams by Activated Charcoal," Oak Ridge National Lab., ORNL 2872, April 1960.

For depressurization of the primary loop the helium will be bled through the heat exchanger and adsorption bed. The delay of the noble gases will be sufficient to allow complete depressurization of the primary loop before elution of the noble gases from the adsorption bed occurs. This will allow the adsorption bed to be valved off following depressurization of the primary loop so that all of the noble gases may be held until they decay and the bed can be purged.

Since the retention time for an adsorption bed of a given size operating at a given temperature is inversely proportional to the volume flow of carrier gas through the bed, the flow control valve is located on the outlet of each adsorption bed to maintain maximum pressure within the filter. This throttling valve will be controlled to hold the pressure drop across the charcoal to less than or equal to a set value.

The heat exchanger will cool the gas to 100°F before it enters the filter system. Since the gas from the primary loop may contain water as a result of a boiler leak, a moisture separator will be provided on the outlet of the heat exchanger. This will insure that the maximum amount of water carried into the adsorption beds will be that due to the helium being saturated with water at 100°F. This amount of moisture will have an insignificant effect on the capacity of the adsorption beds.

A study layout of an adsorption bed is shown in Figure 8.2. The 30-inch diameter by 12-foot length of the charcoal-filled portion of the pressure vessel provides an active bed volume of approximately 57 cubic feet. The shielded unit weighs about 15 tons.

Decay heat from the captured fission products is conducted from the charcoal, which is a rather poor thermal conductor, to aluminum cooling fins which are arranged in a rectangular pattern. The aluminum fins are cooled by water supplied from the component cooling system and flowing through tubes to which the fins are attached. The number of cooling fins in the first 6 inches of the bed is double that in the final 11.5 feet. This is to remove the power generated by the halogens, essentially all of which will be trapped within the first 6 inches of the bed.

The adsorption bed shield is a steel-clad lead shield fabricated as an integral unit on the pressure vessel. The shield thickness is increased from two inches to seven inches on the upstream end of the filter to provide adequate shielding for the high concentration of halogens in the first 6 inches of the adsorption bed. This shield configuration is capable of reducing the surface radiation levels to values which are compatible with the prescribed radiation standards for nuclear ships, provided about 20 hours decay time is allowed before attempting a bleed-down following a postulated reactor melt.

8.3.2 Waste-Water Handling System

The waste-water handling system provides for draining water from the primary loop in the event of a leak. If water accumulates in the bottom of the primary loop pressure vessel, the system senses it and automatically drains it to the lower head of the containment vessel. From there it may be drained by gravity to a waste-water storage tank. The level sensors are the vibrating-paddle type.

Fig. 8.2 - Waste-gas filter

The frequency at which they vibrate in gas is different from the frequency at which they vibrate in liquid. This change in frequency causes actuation of the drain valves.

8.4 CONDENSATE DEMINERALIZER SYSTEM

A condensate demineralizer system will minimize scaling problems in the boiler and provide a source for moderator makeup water. The demineralizer consists of two full-flow units arranged in parallel and located in the secondary loop between the main condensate pump and the moderator heat exchanger. Each unit consists of a high-flow-rate, polished-condensate, mixed-bed deionizer capable of processing feedwater at the rate of 300 gpm and maintaining effluent at an average of 0.10 ppm total dissolved solids. The capacity of each unit is 252,000 grains. Regeneration will require about 4 hours. Utilizing one unit on the line while the other unit is being regenerated will permit continuous operation at the equivalent of a continuous sea-water leak into the condensate of about 30 gallons per hour.

The purpose of the demineralizer is to remove dissolved solids (1) existing in the system components at startup, (2) introduced by the system components and piping during operation, (3) introduced with makeup water fed into the system upstream of the demineralizer, and (4) introduced into the system as a result of a condenser tube leak. It was assumed that sea water will contain 35,000 ppm total dissolved solids and the 0.5 percent makeup water will be delivered from evaporators with an average of 4 ppm total dissolved solids. With only this addition of solids from the makeup, regeneration would be required only once every 12,000 hours.

Conductivity of both inlet and outlet water will be monitored. The signals will be used to provide indication and warning.

The only major maintenance required by the demineralizer will be regeneration. The materials required for regeneration are 150 pounds of either sulphuric or hydrochloric acid, 105 pounds of sodium hydroxide, and about 3000 gallons of condensate for backwashing. Except for the condensate, these can be stored in bulk storage tanks and pipes to the demineralizer. As planned, the regeneration system is fully automatic and requires only manual initiation.

9.0 HANDLING AND SERVICE

The overall design of the 630A has been developed with major considerations for service and maintenance operations. High reliability components and a system designed for rapid repair or replacement of components requiring service is intended to produce a high level of system dependability. Routine and planned down-time operations at specific intervals are planned to replace those components with an expected service life less than that of the entire steam generator.

Some components such as fuel elements would be replaced only at a ship service facility while other components such as pumps, control rod actuators or even primary coolant circulators could be changed aboard ship at other locations. Spare units for most of these components would be carried on board, along with fixtures necessary for their installation. Components removed from service could be repaired aboard ship or exchanged for factory rebuilt units.

9.1 GENERAL HANDLING AND SERVICE

The handling and service techniques for the 630A Mark V are basically unchanged from those of previous concepts. The modular design permits rapid reactor core removal and replacement with another core for minimizing refueling time. Primary coolant circulators are also designed for minimum exchange time, as are other smaller components.

Those components, such as the boiler, which are not suited for rapid exchange, are designed for service in place. Hatches, cover plates or other provisions are made to allow access to service these components. Shielding is provided in most locations to maintain radiation levels sufficiently low for contact maintenance.

Due to the difference in fuel element configuration, the fuel changing operation is slightly different but follows the same procedure as that planned for the earlier configurations.

9.1.1 Gas Circulators

Primary coolant circulator exchange utilizes the same procedure as in earlier systems, however, since the circulators are electric motor driven, it is practical to exchange both motor and circulator as a unit.

Auxiliary shield rings and the containment shell can be removed manually using counter-weighted slings. Other augmenting shield rings inside the containment enclosure can be removed by the same method, exposing the circulator flange fasteners. A circulator handling fixture, similar in nature to the blower handling fixture for previous configurations (GEMP 231, Figure 8.1) is positioned and secured to the flange and to the circulator. This circulator handling fixture is approximately the same length, but about 14 inches smaller in diameter than the original blower handling equipment. Dual sets of ball screws enclosed within the fixture provide the forces to withdraw the motor and circulator into the sealed fixture housing and also provide the moment to support the cantilevered load. The attaching yoke is equipped with wheels to carry the load weight on guide rails within the fixture housing.

As with previously planned blower handling operations, this exchange operation is performed with the primary coolant system depressurized to very slightly below ambient pressure. A closure can be provided to seal off the vessel opening until the circulator is replaced using the same or a similar fixture. The handling fixture provides shielding as required for the circulator when it is separated from the steam generator.

9.1.2 Boiler Service

Boiler service operations are unchanged from previously devised methods. The inspection and service operations which are expected to be necessary during the 20-year lifetime are performed manually while the boiler is in operating position. Anticipated maintenance operations are: isolation of a boiler section at the time of a tube failure, plugging of a ruptured tube, periodic inspection of the tube sheet, and the flushing of the boiler tubes with scale dissolving solvent as required.

Access to the tube sheets is required for header inspection and tube plugging and cleaning operations. The boiler tube sheet, which also serves as the lower head of the pressure vessel, is exposed by removing the lower dished head of the containment vessel. Steam and feedwater piping and manifold covers are then removed, exposing the tube ends in the tube sheet. Manual access is available to both ends of all tubes of the boiler in this arrangement for pressure testing and permanent sealing of any tubes which may develop leaks. Statistically, about 4 percent of the tubes can be expected to fail from various causes over the 20-year design life of the boiler. Tube flushing operations at approximately 1-year intervals can be expected with conventional water treatment. Flushing operations that would not require access to the individual tube ends could be performed by connecting to the steam and feedwater lines outside the boiler.

9.1.3 Control Rods and Actuators

Replacement of control rod actuators is accomplished manually by the same established and proven procedure planned for earlier configurations. (GEMP 231, Section 8.3.1).

Replacement of control rods, tips and gang plate assemblies is possible in the Mark V design as in previous systems. However, with the proposed displacement shimming, replacement of control rods would probably be quite infrequent. Should shim or scram rod replacement be necessary, they can be changed manually aboard ship or at a service facility.

Each control rod actuator is removed and the control rod assembly lowered back into the fully inserted position where it is locked in place by a latch on each assembly. Procedurally only one gang of rods is removed from the fully inserted position at any time, maintaining the reactor in a sub-critical condition. A single special tool is used to lower the rods into latched position. When all actuators have been removed the actuator mounting plate is removed allowing individual rods or a gang of rods to be withdrawn into a shielded container. Replacements are installed immediately following removal of any rods.

When the desired rod replacements have been completed, the actuator mounting plate is replaced and the actuators installed. A single special tool is used to lift the rod-ganged assembly to an elevation such that the actuator can be attached to the rod assembly, then the actuator and rod assembly is lowered to its seated location where the actuator is latched to the actuator mounting plate.

9.2 REFUELING

After completion of power operations to a degree requiring refueling, the modular replacement concept is utilized for rapid completion of the fuel changing operation. The reactor-shield plug assembly is removed from the ship as a unit, transferred to the refueling hot cell and a refueled and tested reactor assembly is replaced in the ship's steam generator. The same core-shield plug assembly can be refueled and returned to the same ship if necessary, requiring several extra days for the fuel element exchange operations. Where necessary, rebuilt controls and instrumentation components can be exchanged at the same time. Primary coolant circulators, filters, boilers and other ship's components could receive routine service at this time to make best use of this down-time.

At all times during reactor handling and service operations the core is suspended in its normal manner and attitude from the shield plug. Handling connections are made on the manually accessible exposed portion of the shield plug structure. Normal aftercooling is provided by the steam generator system while the reactor is in its normal operating location. Circulation and cooling units in the transfer cask assembly, as well as moderator loop coolers, provide redundant systems for removal of after-heat while the reactor is within the transfer cask. Moderator loop cooling is transferred to the cask system prior to removal of the reactor from the steam generator.

Elevation of the reactor shield plug assembly into the transfer cask is by a pair of quad-screw jack systems providing 100% backup lift capability. Once the reactor assembly is seated and secured in the cask the shield doors are closed and direct air cooling of the fuel is provided by the cask heat dump system. The cask is now transferred from the ship to the refueling service cell. Backup power sources are provided for the multiple aftercooling systems built into the transfer cask assembly.

In location over the refueling cell port, the cask is lowered onto locating and bearing pads on the cell, cask doors are then opened and the reactor shield plug assembly lowered onto an adapter ring which supports the reactor in position for refueling. When facility aftercooling is established, the transfer cask can be disconnected from the expended core and moved to an adjacent cell port where a refueled and tested core-shield plug assembly is located. This ready core shield-plug is loaded into the transfer cask, secured and then transferred to the steam generator in the ship. The replacement operation is essentially the reverse of the removal procedure except that aftercooling is not required.

9.2.1 Fuel Element Changing

Fuel elements of the 630A reactor are removed from the core by a remotely controlled refueling machine. The core is positioned in the overhead port of the refueling cell such that the lower face is exposed to the refueling machine in the cell. Instrumentation which may be on the lower face is removed by use of hot cell manipulating equipment and then the individual fuel elements are released from the core and lowered into handling racks for cool-down and decay in a hot fuel storage area. Underwater storage could be used.

After fuel removal is complete, control rods may be replaced if necessary by manual contact operations on the exposed upper portion of the shield plug which remains outside the hot cell. When all rods have been replaced and are secured, fuel loading may commence.

New fuel elements are loaded into the reactor core by means of the refueling machine, which also operates the latch mechanism and provides a test load to assure that the fuel element has been properly secured. Direction of the refueling machine is by remote control and by automatic sensing and centering systems on the machine. Criticality measurements can be performed and instrumentation tested to assure suitability of the refueled assembly before its installation in the next ship.

Fuel elements are held in the hot fuel storage until they have decayed to a condition suitable for loading and shipping to a reprocessing facility. If necessary, unfueled portions of the fuel elements can be cut away in the refueling hot cell as an aid to handling and shipping operations.

9.3 HANDLING EQUIPMENT AND FACILITIES

Safe and rapid service operations require the utilization of the proper fixtures and handling equipment and facilities. Many of the smaller items can be handled in conventional fashion with common slings and fittings. However, where necessary for quick and safe performance of an operation, suitable handling equipment would be provided.

9.3.1 Handling Equipment

Table 9.1 lists the general areas of handling equipment planned for support of 630A steam generators. Other specialized and general purpose equipment is necessary at the refueling hot cell facility.

The core and shield plug transfer cask and its transport system are key items in the reactor exchange services concept. The cask system provides physical support and protection, radiation shielding and afterheat dissipation for the reactor assembly during its transfer from the operating location in the steam generator aboard ship to the refueling hot cell.

Attachment to the reactor shield plug assembly is by means of the multiple handling pads on the upper exposed portion of the shield plug. After the core-plug assembly is seated in place in the transfer cask, additional locks engage the shield plug flange providing a completely separate support system in addition to the normal handling attachments.

Shielding is provided by the steel encased lead shield, cylindrical in shape with power driven biparting doors at the bottom. The open top is plugged by the reactor shield plug seated against a stop ring at the cask top. Adequate shielding is provided to allow personnel access to the cask.

Table 9.1

Major Items of Handling Equipment - 630A Service	
1.	Transfer Cask Assembly
2.	Pressure Vessel Cover and Inspection Equipment
3.	Refueling Machine and Fuel Handling Equipment
4.	Diffuser - Shield Handling Equipment and Shields
5.	Primary Coolant Circulator and Drive Handling Equipment
6.	Boiler Inspection and Header Handling Equipment
7.	Filter Handling Equipment
8.	Control Rod and Actuator Handling Equipment
9.	Sensor and Instrumentation Handling Equipment
10.	Containment Cover Handling Equipment
11.	Piping Systems Handling Equipment
12.	Shielding Handling Equipment
13.	Tooling, Tensioners and Other Special Equipment

Dual afterheat removal systems circulate the air within the cask and cool it with finned tube air to water heat exchanger completely enclosed within the cask shield. Hot water from this system is circulated to air cooled heat exchangers located outside the cask.

Figure 9.1 illustrates arrangement of the heat exchangers and blowers inside the shield and the blower drive motors outside the shield.

Air ducting seals, located on the cask doors, make a seal against the core structure and cask air ducts when the doors are closed.

The transfer cask is suitable for handling with a heavy crane or by means of a 4 wheeled transport vehicle. A crane bail with a load indicating device can be provided at the top of the cask for handling with conventional cranes. Crane capacity required varies with the size of cask and core plug assembly being handled, as shown in Figure 9.2.

Figure 9.1

REACTOR EXCHANGE OPERATION
CRANE OR VEHICLE LIFT REQUIRED

Figure 9.2

Power for aftercooling systems can be provided by a trailing cable and an engine-driven generator on the cask for normal and backup supply. A system of motor-driven screw jacks are used to load and unload the reactor shield plug. Dual systems divide the load under normal operation with either system designed to handle the full load independently. (Figures 13.1 and 13.2 of GEMP 304 illustrate this type of system.)

The transfer cask used with the vehicle system is essentially the same as the crane handled cask except that the crane bail and screw jacks are replaced with a dual purpose jack system. The double capacity screw jacks are located on the vehicle frame and serve to lift the core-shield plug assembly in and out of the cask as well as to raise and lower the cask to traveling and loading positions. Shield plug attachment is by means of an extension frame which extends down through the cask. Multiple latches on the cask sides and the vehicle frame allow the cask to be lowered the necessary distance into the ship. Suitable structure is necessary at the deck level for vehicle support.

Commercial electric drive wheels are suitable for this system, providing low bearing pressure, compliance and excellent control. Wheels with double the required capacity are currently used on earthmoving machines and steel mill forklifts as well as military vehicles.

Vehicle drive generators could provide power for cask systems. Auxiliary generator and heat dump equipment could also be located on the vehicle and provided with flexible connections to the cask.

9.3.2 Ship Service Facility

A ship service facility, providing a refueling hot cell, fuel storage and other capabilities is required in support of 630A powered ships. Steam generator size, zone of operations, fleet size and a large number of other factors influence the exact nature of the facility.

In general, the ship service facility provides those special services necessary for refueling and exchange operations for components of the 630A steam generator which require special handling equipment or procedures. Table 9.2 itemizes the major requirements of this nature for which the ship service facility would be equipped.

Table 9.2

Ship Service Facility Requirements

1. Ship mooring capability
 2. Refueling and remote work hot cell and manipulating equipment
 3. Transfer cask and transfer system
 4. Handling equipment and storage facilities for new and used:
 - a) Fuel cartridges
 - b) Control rods and actuators
 - c) Primary coolant circulators
 - d) Boiler shield-diffuser
 - e) Nuclear sensors and instrumentation
 - f) Other replacement components
 5. Pressure and Containment vessel testing equipment
 6. Fuel cartridge shipping cask loading station
 7. Hot waste handling facilities
 8. Decontamination and survey facilities and equipment
 9. Nuclear measurements equipment
-

The refueling hot cell requirements are defined primarily by the type and scope of reactor service planned. Considerable reactor size variation can be accommodated by cell port adapters, which maintain the reactor at a fixed refueling position in the cell. Provisions are also made to lower the core plug assembly into the cell for inspection or limited remote work on the core or lower plug region.

The hot cell also provides the capability for other remote work such as replacement of core instrumentation, filter media replacement, or inspection and repair of the boiler shield and diffuser. General purpose manipulators, remote control power tooling, and special purpose inspection equipment in this cell would provide the flexibility of operation necessary for this type reactor service.

The ship service facility is adaptable to a number of possible arrangements. A shore facility or a barge or ship hull installation is possible, with various advantages and disadvantages for each system. Closely related to the facility location is the means by which the transfer cask is transferred from the ship to the service cell. Both of the methods which appear feasible, the crane system and the vehicle system, can effectively serve shore or floating service cell installations.

Figure 9.3 illustrates the maximum of mobility concept, whereby the hot cell and all related ship service facilities are built into a ship hull or hull section. This floating facility could possibly be towed to any location and perform its service function quite independent of other shore facilities. A small crane would be used to handle shipping casks and other supplies. Surplus ship hulls are being used advantageously today for barges, canneries and other uses.

Also mobile in nature is the barge mounted crane and refueling cell shown schematically in Figures 9.4 & 9.5. A semi-conventional barge-mounted crane serves the hot cell, using a crane handled transfer cask. The hot cell and other facilities replace a portion of the normal ballast on this type barge. Attachment of the barge to the ship hull could stabilize the system during the heavy lift if necessary.

A similar barge-mounted crane could serve a hot cell on shore which was located adjacent to a dock within reach of the crane.

A hot cell on shore, but located some distance from the dock or pier, could be served by a walking gantry crane with rails connecting the ship and hot cell lift locations. The vehicle transported cask could also be used with a shore hot cell as illustrated in Figure 9.6.

The top of the cell is below roadway level to allow the cask and vehicle to move away with the core shield plug assembly in place in the cell port. The hot cell could also be located at grade level and a captive guiding ramp provided for vehicle elevation over the cell.

When the number of ships being served by this type facility increased to where additional refueling stations were required, consideration could be given to a much simpler refueling cell. This concept as shown in Figure 9.7. utilizes a transfer cask with slightly heavier shielding and possibly with viewing ports in the side wall at advantageous locations. The reactor-shield plug assembly remains in the cask during the refueling operation. Transfer cask doors are opened with the cask located over the refueling machine and shield assembly. A double eccentric rotating shield locates the refueling machine at the proper location and the hot fuel element is removed and immediately discharged into a storage pool or cooling cask below. Very

Vessel Hull Facility
Figure 9.3

Barge Facility Plan View
Figure 9.4

Barge Facility
Figure 9.5

Vehicle Served Shore Facility -
Figure 9.6

In Cask Refueling Facility
Figure 9.7

limited instrumentation or remote accessory work would be possible with this arrangement and a pool facility might be necessary for additional fuel handling. This type facility would represent a considerably lower investment where its lack of flexibility could be tolerated.

10. SAFETY ANALYSIS

Since the 630A Mark V Nuclear Steam Generator is in the early stages of design, a comprehensive safety analysis has not been performed. However, preliminary investigations indicate that one of the most attractive aspects of the 630A Mark V power plant is its inherent safety. Some preliminary discussion of the subject in this report is therefore appropriate.

As discussed in the foregoing sections, two alternate versions of the 630A Mark V power plant have been considered, the Mark V(A) and Mark V(B). In most respects (i.e., with the exception of the configuration of the nuclear core) the two versions can be considered essentially identical for the purposes of this section. The general discussion to follow may be considered applicable to both the Mark V(A) and Mark V(B) power plants unless specific limitations are pointed out.

The system parameters of primary importance to safety have been assumed to be as presented in Table 10.1. Since time was not available to locate and resolve all possible conflicts, these assumed data may differ in some cases from those contained elsewhere in this report. It is believed, however, that any discrepancies which may exist are relatively unimportant to the evaluation of this section and that the broad conclusions reached herein are valid within the limits of accuracy it was possible to achieve during the short period of study.

Table 10.1

Assumed Operating Conditions and Characteristics
of a 60.4 Mw, 630A Mark V Power Plant

Primary Coolant (Helium)

Volume, ft ³	450
Mass, lb	120
Average temperature, °F	680°
Average pressure, psia	826
Maximum temperature, °F	1200
Maximum pressure, psia	830
Flow rate, lb/sec	66.2

Secondary Coolant

Feedwater temperature, °F	415
Feedwater pressure, psia	1700
Steam temperature, °F	1000
Steam pressure, psia	1500
Steam flow rate, lb/hr	175,200
Water mass in boiler, lb	890

Containment Air

Total volume, ft ³	1400
Mass, lb	95
Temperature, °F	125
Pressure, psia	14.7

Moderator Water

Total volume, ft ³	226 (V-A)	124 (V-B)
Mass, lb	13350 (V-A)	7330 (V-B)
Average temperature, °F		235
Pressure, psia		250
Saturation temperature, °F		401

10.1 RADIOLOGICAL ASPECTS OF NORMAL POWER PLANT OPERATION

For the most part, the radiological aspects of normal power plant operation are essentially unchanged from earlier versions of the 630A system. One area of obvious difference from the previous analysis is that the rod type fuel of the Mark V will not show the same type of progressive fission product release during operation as would the clad metallic - UO_2 dispersion fuel used in prior versions. The reason for this is that the Mark V fuel elements will have a thicker cladding (0.015 inch vs 0.004 inch) and are designed so that the buildup of fission gas pressure is quite small. Thus the probability of cladding failure of a single fuel pin is quite low and significant fission product leakage is not expected. Failure of the cladding would, of course, result in the dispersion of a small portion of the radioactive fission products and, in the event of gross rupture, some UO_2 fuel throughout the primary loop. The capability exists, as with earlier 630A power plants, to continue normal operation with considerable leakage of fission products into the coolant. Although it is believed that this condition will not be as important a consideration with the Mark V as it was for these earlier designs, the large number of pins in the core creates a reasonable probability that a certain number may become defective. Therefore, continued operation of the reactor with a number of defective pins is an established design condition and the requirements of a bypass gas treatment system will be thoroughly investigated.

The Mark V system has a completely sealed primary loop, using gas-bearing circulators so that there will be no continuous seal bleed leakage problems comparable to those of the Mark I, II and III versions. However, some minor leakage and diffusion of helium from the primary system will probably occur so that a small containment gas cleanup system probably will be required. Little, if any, continuous atmospheric discharge of contaminated gases will be involved, however, and no problems are foreseen in achieving extremely low levels of atmospheric pollution. Depressurization of the primary loop may impose more stringent gas cleaning requirements, but the technology is well in hand to meet any conceivable problems in this area. Cooled charcoal adsorption units of the type discussed in prior 630A status reports * can be sized to accomodate all foreseen effluent cleaning requirements.

Radioactive liquid waste and direct radiation problems associated with the Mark V will be minimal and essentially identical to those of earlier designs.

* General Electric, NM&PO, "630A Maritime Nuclear Steam Generator Status Report No. 1", GEMP-231, Sept. 12, 1963.

10.2 RADIOLOGICAL ASPECTS OF SERVICING AND REFUELING

In nearly all respects, the radiological aspects of servicing and refueling the 630A Mark V Nuclear Steam Generator will be virtually identical to those of the Mark III and IV versions. The one aspect of this work that will be made considerably easier is the fact that contamination of the primary coolant gas and internal components of the primary loop is expected to be much less in the case of the latest design so that the control of contamination, during periods when the primary loop is breached, will be much less of a problem. Nevertheless, some contamination of the primary loop is probable and provisions must be made for the purging of the primary loop and the maintenance of a slight negative pressure therein during periods when it is opened for maintenance. This can be accomplished via the depressurization adsorption beds in the same manner as that planned for earlier models.

Most power plant components, which will require servicing at intervals more frequent than the regular refueling outages, will be designed for safe accessibility while the ship is in a port of call or at sea. All of the necessary handling and protective devices may be carried aboard ship so that properly trained personnel may carry out these routine servicing functions with minimal radiological hazards being encountered. Such functions as the exchange of control rod actuators or nuclear sensors and the plugging of leaky boiler tubes can be accomplished safely within the capabilities of the ships crew and equipment.

The only presently foreseen servicing operation, other than refueling, which will require breaching of both the containment and primary loop is the replacement of a complete gas circulator assembly. Dose rates to be encountered in changing a circulator have not been investigated but, since the use of a shielded removal fixture is quite feasible, no real difficulty is expected. The question that must be answered is: "Should shipboard capability be provided for gas circulator exchange without external support or should this capability be limited to one or more specially equipped service ports?" No real barriers are seen to providing full shipboard capability for this operation.

Removal of the reactor core-shield plug assembly for refueling and the exchange of gas circulators; i. e., operations involving breaching the primary loop and, temporarily, the containment, will require the control of radioactive effluent. This requirement must be met by temporary closures for the opened ports and by maintaining both the interior of the loop and the transfer cask cavities at a negative pressure with respect to atmosphere. Providing adequate closures are maintained, relatively small blowers will suffice for contamination control and the effluent can be discharged through either individual or centralized gas cleaning systems. The core-plug transfer cask, of course, must also be equipped to provide in-transit afterheat removal. Systems to meet these requirements have

been investigated in connection with servicing studies of earlier 630A designs, and no significant changes are believed necessary for the Mark V plant.

10.3 POWER PLANT ACCIDENTS

A cursory investigation has been made of a number of different events and sequences to obtain information relative to the safety performance of the 630A Mark V power plant in abnormal situations. In all cases, these events would normally be prevented, or automatic safety action taken to prevent hazardous results, by built-in features of the control and safety system. Thus, multiple equipment failures, operator error, or both must be assumed before a hazardous accident can occur.

A marine power plant can, of course, operate over many different trade routes and enter a great many different populated seaport areas. On the other hand, such a power plant will probably be refueled in only a few selected locations. Therefore, it will be necessary to postulate a maximum credible accident (MCA) applicable to the operating configuration for use in evaluating the overall safety of the system in a wide range of environments and a separate MCA for refueling when the probability and potential severity of an accident appear to be greatest. In this preliminary stage of conceptual design, however, it is not possible to thoroughly assess the safety characteristics of the Mark V power plant nor to define the maximum credible accidents for any plant condition. Therefore, the reader is cautioned not to assume that any of the following events is intended to represent a postulated maximum credible accident.

10.3.1 Startup Accident

A startup accident was postulated in which an uncontrolled power excursion was initiated from a shutdown condition with a fresh reactor core. It was assumed that a normal reactor startup was in progress when, through some undefined series of control system malfunctions and human errors, rod withdrawal was not halted and all safety devices failed. Reactivity was assumed to be added as a continuous ramp, at the design rate of 1% $\Delta k/k$ per minute, until all rods were completely withdrawn.

The resulting transient was analyzed using the IREKIN* computer program, together with suitable subroutines, on an IBM 7040 digital computer. The calculated fuel temperature (Doppler) coefficient was introduced as a second order polynomial fit of the calculated reactivity-energy curve. No heat removal by the primary coolant circulating at a reduced rate during startup was included in these initial computations since it was assumed that the transient would be rapid enough so that this error would be small. The sole

* Wagner, R. J., "IREKIN - Program for the Numerical Solution of the Reactor Kinetic Equations", IDO-16928, Phillips Petroleum Co., March 1, 1963.

mechanism for finally quenching the excursion was assumed to be the physical rearrangement of nuclear fuel due to melting. Figure 10.1 shows the computed power profile and integrated power obtained in these preliminary investigations. As indicated, the negative reactivity input due to the fuel temperature (Doppler) coefficient temporarily overrides the positive ramp, terminating the initial rapid power transient at about 70 megawatts. After a brief period of decrease, the power slowly rises for about 2-1/2 minutes until enough energy is stored in the core to cause melting of the UO_2 in the central transverse plane of the reactor (at about this point the cladding in this region also starts to vaporize). A deliberate time delay is imposed on the shutdown function by allowing a three-inch-length in the center of all fuel rods to become molten before any fuel translocation occurs. The reactivity effect of melting as a function of time was estimated by assuming that the molten fuel would progressively move downward under the force of gravity redistributing itself over the next successive unmolten increments. To allow for the viscosity of the molten material, the acceleration of gravity was arbitrarily degraded by a factor of ten. This oversimplified shutdown model is believed to be conservative. The total energy generated in the excursion was about 3100 megawatt-seconds and about 8% of the UO_2 fuel was estimated to have melted.

It is quite obvious that, since the reactor is at significant power levels for several minutes, it was not valid to neglect heat removal by the circulating coolant. Time was not available for more definitive computer calculations but it is obvious that the effect of heat transfer from the fuel would be to stretch out the excursion over a longer time interval and increase the power at which melting would occur, resulting in a greater energy release. The initial power spike would probably not be significantly affected. While these initial calculations are not regarded as truly definitive, it is believed that they provide a reasonable first approximation of the results of this type of accident assuming that all automatic safety devices fail and that the excursion is allowed to continue to its natural termination. Adequate time is available for operator intervention to prevent significant fuel melting, however.

The total excursion energy of this type of event is sufficient to raise the primary loop gas pressure from its approximate pre-start level of 400 psia above the relief point (900 psig). Therefore, pressurization of the containment vessel and eventual release of a fraction of the fission products to atmosphere would be expected. The peak containment pressure has not been calculated for this accident, but it is expected to be only slightly lower than that predicted in the full power excursion discussed in the following section.

Figure 10.1 - Start-up Accident - Power & Energy vs. Time

10.3.2 Excursion from Full Power

A second excursion was postulated assuming that the reactor was operating at full power (60 megawatts) when, through a large number of malfunctions and/or operator errors, an uncontrollable rod withdrawal commenced. Again the reactivity insertion rate was assumed to be 1% per minute and all safety systems were assumed to fail. In this case it was assumed that heat removal from the system remained constant at 60 megawatts throughout the excursion. Figure 10.2 shows the calculated power profile and integrated energy. As indicated, a gradual power rise was predicted until the peak power of about 97 megawatts was reached at roughly 2.3 minutes after the start of the excursion. At this time enough energy was stored in the core to initiate melting and the ensuing rapid shutdown was based upon the same simplified shutdown model as postulated for the starting accident. The total energy generated during the excursion was calculated to be about 10,600 megawatt seconds of which about 8,400 megawatt seconds were removed via the normal heat extraction path. The total fractional core melt was estimated to be 11 percent.

For these conditions, the peak pressure in the primary loop would exceed the relief valve set point of 900 psig, so that helium and entrained fission products would escape to the containment vessel. Detailed calculations to predict the resulting pressure transient have not been made. However, it was arbitrarily assumed that one-half of the excess energy generated in the excursion (i. e., above 8,400 mw-sec) was removed by the steam generator during the interval required for the gas coolant and reactor core to reach temperature equilibrium, and that complete mixing of the helium and containment air would take place. For these conditions, the peak containment pressure would be about 350 psia. In the actual case, of course, further heat losses to the shield and the containment cooling system would reduce the pressure from this peak value very rapidly. The capability may also exist to bleed off containment pressure through the depressurization gas treatment system, if desired.

10.3.3 Core Flooding

There has not yet been an opportunity to investigate experimentally the reactivity characteristics of the 630A Mark V core under a condition of flooding of the gas coolant passages. This is, of course, a very important consideration for a gas-cooled, water-moderated system - particularly when intended for marine use. Initial analytical work indicated that it was possible to design the Mark V-A version of the reactor with a negative flooding coefficient. In this case, the gain in reactivity due to reduced neutron leakage from the flooded core would be more than balanced by the increased resonance capture in U-238. Subsequent work on actual reference designs has indicated that the flooding coefficient is more likely to be positive - particularly the Mark V-B version that currently appears superior in most other respects. Critical experiment data is required to establish

the flooding characteristics with reasonable certainty, but current indications are that at worst the worth of flooding may be about $+10\% \Delta k/k$. This is only about half of the positive worth of flooding determined for earlier versions of the 630A, and control rod capabilities can easily be increased to provide an adequate shutdown and control margin for total flooding or for any conceivable transient flooding mechanism.

10.3.4 Loss of Primary Coolant Circulation

Loss of primary coolant flow could conceivably occur as a result of total loss of electrical power to the circulator drive motors. Because of the redundant power supplies to be provided, however, such a failure is highly unlikely. Total loss of flow through mechanical failure of the circulators is considered incredible since two completely independent and widely separated circulators are used. Failure of one unit could cause the reactor to scram and the remaining circulator would assume the aftercooling load.

In the case of the former concentric ring fueled reactor, heat transfer calculations showed that, even in the event of total loss of primary gas flow, the heat transfer from the fuel elements to the moderator loop would be sufficient to prevent significant fuel melting due to afterheat. Indications are that this significant safety feature is even more effective for the current rod-fueled design. Initial computations have been made, using computer code THTC*, of the fuel element temperatures in the Mark V(A) reactor following a total loss of primary coolant flow and with a subsequent reactor scram. In this preliminary work, flat fine radial and gross radial power profiles were assumed but this approximation is not believed to have resulted in significantly pessimistic results. The predicted cladding temperature as a function of time after scram is shown in Table 10.2. At the time of this writing, similar computations are being performed for the Mark V(B) reactor.

It can generally be concluded that fuel cladding temperatures following a loss of coolant flow will remain below the melting temperature of 2550°F . Even in the event that power to the moderator circulating pumps is lost, boil-off of the moderator water would provide sufficient aftercooling to prevent clad melting for a period of several hours so that corrective action could be taken.

10.3.5 Partial Blockage of Primary Coolant Flow

A maldistribution of coolant flow through the core could occur as the result of material carried by the primary gas stream depositing at the inlet to one or more coolant passages at the upper end of the reactor. Introduction of foreign material into this closed-cycle power plant could occur only at the time of initial assembly or during maintenance. It is conceivable, however, that pieces of internal structure from some of the parts in the system might come loose during operation and interfere to some degree with flow.

Blockage of coolant flow through a single tube bundle, to the extent that cladding failure occurs can be postulated. The resulting increase in primary loop fission gas activity would be detectable and if the change were significant, the reactor would be deliberately shut down for repair. An incident of this type would not cause an increase in primary loop pressure, and no release to containment would occur. Radiological consequences would be insignificant.

* Transient Heat Transfer Program, series C, undocumented. The preceding series THTB is documented in: Stephens, G. L. and Campbell, D. J., "Program THTB for Analysis of General Heat Transfer Systems," GE-RPD, R 60 RPD 647, April 21, 1961.

Table 10.2

PREDICTED FUEL CLADDING TEMPERATURE RISE FOLLOWING
INSTANTANEOUS LOSS OF COOLANT FLOW - 630A MARK V REACTOR

<u>Time After Scram (sec)</u>	<u>Mark V(A) Clad Temperature (°F)</u>	<u>Mark V(B) Clad Temperature (°F)</u>
10	1697	1752
100	2028	2072
250	2165	2085
500	2098	2025
1000	1972	1872

10.3.6 Depressurization of the Primary Loop

Credible mechanisms of rapid depressurization of the primary loop have not been defined at this time. If a sudden loss of pressure occurred, the reactor would be scrammed by an indication of low circulator outlet gas pressure. After-cooling flow would continue to be supplied by the circulators and no core damage would result. The resulting peak pressure in the containment vessel, assuming no net heat or flow losses, would be about 220 psia. No significant release of radioactivity would result from an accident of this type.

10.3.7 Loss of Secondary Coolant

The effects of single and multiple steam line and boiler tube breaches were investigated for earlier versions of the 630A power plant, but have not been studied in as great detail for the Mark V concept. Some of the basic conclusions of this prior work should apply as well to the system and additional calculations have been made in order to provide a preliminary estimate of the most important results.

For the case of a single boiler tube breach within the pressure vessel, the reactor would be scrammed and afterheat would be absorbed in the remaining unfailed boiler sector. No fuel overheating would occur. The primary loop pressure would increase to the point of gas relief to the containment vessel. Pessimistically assuming failure (closed) of the steam back pressure control valve on the damaged sector, so that all steam in this sector would be discharged to the primary loop, the peak containment pressure was estimated to be about 200 psia. For the case of a single steam line failure outside of the primary loop, a peak containment pressure of less than 70 psia was estimated. In the event of rupture of both steam lines (an extremely improbable occurrence) the reactor would scram and all feedwater valves would close. For the case of rupture of the steam lines, outside of the pressure vessel but within the containment vessel, evaporation of water in the boiler would result in a containment pressure of about 130 psia (both back pressure valves failed closed). At this pressure the containment vessel also would contain 288 pounds of feedwater (assuming twice the normal feedwater flow during the 3 seconds allowed for isolation valve closure).

Multiple boiler tube rupture within the pressure vessel would overpressure the primary loop with a resultant release to the containment vessel. For the worst case the following assumptions were made:

1. One-third of the boiler tubes in each section are ruptured at the inlet to the pressure vessel.
2. Pressure vessel relief valves fail to close.
3. Isolation valves close 3 seconds after rupture.
4. Reactor scrams after one second.
5. Reactor heat during first 3 seconds goes to the helium.
6. No heat is transferred to structural components.

The peak containment pressure under these conditions with no gaseous mixing would be 350 psia. At equilibrium temperatures (complete mixing of gases) the containment vessel pressure would be only 250 psia. Since the helium is being circulated very rapidly, the initial peak pressure will probably be very close to the lower figure of 250 psia. After mixing of the gases, the reactor would gradually reach an equilibrium temperature of 540 degrees F with the gases in the primary loop and containment vessel. At this temperature the pressure would be about 470 psia.

These calculations assume that the reactor afterheat, after 3 seconds, is exactly balanced by heat removal from the system in the moderator water, while, in fact, the heat transferred to the moderator will reduce the final temperature and pressure below the quoted values of 540°F and 470 psia. The total cooling effect has not been fully assessed for the stated conditions but it seems obvious that, in a loss of secondary coolant accident, reactor temperatures will be reduced rather than increased and that no overheating of fuel will take place.

10.3.8 Loss of Moderator Cooling

If moderator circulation is stopped, the reactor would normally be scrammed by a flow monitoring device; if not, the reactor would be scrammed on moderator-loop overpressure after the saturation temperature was exceeded in the moderator loop. Steam would be released to the containment through relief-valve action. If moderator flow could not be re-established, boil-off of the moderator water would eventually occur and overheating of the core structure might result. However, if primary-loop circulation were maintained, no fuel melting would occur. If no gas circulation is assumed, so that all afterheat is dissipated to the moderator, some fuel melting might ultimately occur, but a period of several hours is available in which to re-establish moderator flow and prevent such damage.

10.3.9 Separation of Reactor and Shield Plug

A potentially hazardous incident can be postulated if it is considered credible for the shield plug and reactor to become separated during removal from the nuclear steam generator and transfer to the refueling facility. Since the control rod actuators are mounted on the shield plug, total separation of these major components in the Mark V(A)* would withdraw all rods, possibly causing a power excursion. However, several means of preventing such an excursion are available. The most desirable measure, however, is to design the transfer cask so that, even under the worst conceivable impact, the plug and reactor could not be separated.

The critical points of the transfer will occur during removal or insertion of the assembly, at the refueling facility or in the pressure vessel, when the cask bottom doors are open. If the assembly were to be dropped at this time, due to failure of the lifting mechanism, the shield plug would be stopped by its mating flange and the interconnecting tubes might fail in tension.

If the incident occurred upon withdrawal or insertion into the pressure vessel, the reactor would drop only slightly before coming to rest on restraining keys or upon the diffuser plate. The control rods would not be withdrawn a significant distance, and subcriticality would be maintained. Similar safety features can be designed into the refueling fixtures. It appears certain that the probability of a nuclear excursion from this type of accident can be reduced to an extremely low value.

10.3.10 Control Rod Removal

The prevention of a criticality accident during removal of control rods, for inspection or maintenance, will depend upon the application of proper procedural controls unless the moderator is poisoned. However, because of the large number of rods that can be completely withdrawn without attaining a critical condition, it is believed that the probability of an accident is exceedingly low even without poison in the loop.

10.3.11 Initial Loading and Refueling

During initial loading or refueling, the subcritical multiplication data will be directly compared with prior data and appropriate procedural control information obtained. Again, procedural controls must be relied upon to prevent criticality, but the probability of an accident is very low and the radiological consequences also would be very low because of the fresh reactor fuel.

* In the Mark V(B) design the moderator is in individual tubes rather than in a calandria and gross tube failure is even less likely.

10.3.12 Loss of Aftercooling During Core Removal

During transfer of the reactor from the NSG to service facility, aftercooling will be maintained by circulation of both the moderator loop and air in the transfer cask cavity, through self-contained heat exchangers. If one of these cooling modes is lost, no hazard would result since fuel afterheat can be dissipated by either system, alone. If the event both coolant systems fail, however, clad melting could eventually occur; the rate of temperature rise being dependent upon the power history and decay time after the last shutdown. Probably the most severe incident of this type would occur if the cask were dropped from a height sufficient to render the cooling systems inoperative and to rupture the moderator loop so that the loss of cooling water is quite rapid. However, an important design goal for the transfer cask handling system is to provide positive assurance that the loaded cask cannot fall free. This can be accomplished by means of screw-type lifting devices or means other than crane hoists, as described elsewhere in this report.

A preliminary investigation has been made for instantaneous loss of all cooling 24 hours after shutdown from equilibrium full power conditions. Highly pessimistic calculations assuming complete isolation of the fuel rods show a temperature rise of only 20°F/min and a time lapse of about two hours before clad melting temperatures would be reached. Assuming more realistically that the moderator water is not lost when both cooling systems become inoperative, no fuel damage is probable. Vaporization of the moderator water, which would serve as a heat sink, would require 12 hours for the Mark V(B) (22 hours for the Mark V(A)) assuming complete absorption of all afterheat in the moderator. Thus, it is obvious that adequate time would be available to re-establish cooling in some form and thus prevent significant fuel damage. If fuel damage did occur, its most probable form would be the loss of cladding strength followed by cladding rupture due to internal gas pressure. It is relatively certain that the UO₂ fuel would never reach its melting point of 5000°F due to radiant heat losses to the cask walls.

10.4 RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

In this section are presented the results of a pair of postulated reference accidents. The reader is again cautioned not to assume that either of these events, as postulated, is intended to represent the so-called maximum credible accident.

10.4.1 Reference Accident - Power Plant in Operating Configuration

The full power rod induced excursion, discussed in Section 10.3.2, is selected for partial evaluation of the associated hazards. The estimated

fractional core melt at termination of the excursion is about eleven percent. The percent release of various fission product groups to the primary loop gas was estimated as shown in Table 10.3. The fission product inventory in the core was assumed to be that associated with 300 days of full power operation. Fifty percent of the fission products released from the primary loop were assumed to be plated out within the containment vessel and, therefore, not available for further leakage to the atmosphere. The estimated peak containment pressure of 350 psia was assumed to decrease rapidly to between 150 and 200 psia, through the normal containment cooler and heat losses to the shield, but not to be further reduced through the use of the effluent treatment system. The leak rate from the containment vessel was assumed to remain constant at a level of 2% of the contained volume per day. No further source depletion by plateout within the ship or by processing through a reactor compartment ventilation system was included. The release to atmosphere was assumed to be at a height of 27 meters (90 feet) above the receptor.

Table 10.3

Estimation of Weighted Percent Fission Product Release to Primary Loop

<u>Class of Fission Product</u>	<u>Noble Gases</u>	<u>Halogens</u>	<u>Volatile Solids</u>	<u>All Others</u>
% F.P. at Equilibrium	10	10	11	69
% Released from Molten Fuel	100	50	50	1
% Not Plated Out on Cool Surfaces	100	50	30	30
% of Core Melt	11	11	11	11
Weighted % Release	1.1	0.275	0.182	0.023

From these source assumptions, downwind dose calculations were made using the Gaussian plume model for poor atmospheric diffusion conditions (i. e., a strong inversion with $n = 0.50$, $C_y = 0.40$, $C_z = 0.07$ and a low wind speed of one meter per second). The diffusion conditions, wind speed and wind direction were assumed to hold constant over the entire effective period of release, and the receptor was assumed to remain fixed and unprotected throughout the period. These assumptions are clearly unreasonably pessimistic and, in an actual accident, the long term radiological hazards would be expected to be at least an order of magnitude lower.

Because the release is weighted heavily toward the noble gases and iodines, the immersion dose to the whole body and the inhalation dose to the thyroid will be of primary importance. Of these two, the thyroid dose will be the limiting consideration for the postulated release conditions. The calculated dose to the thyroid, as a function of distance downwind is shown in Figure 10.3. The peak dose, occurring at a distance of 2.7×10^3 meters for the above conditions, is about 54 rads. This is lower than the AEC's suggested siting criteria emergency dose of 300 rads* by a factor of 5.5.

10.4.2 Reference Accident - Power Plant Refueling

The maintenance accident selected for further evaluation is the total coolant loss event described in section 10.3.12. As previously stated, the design goals of the transfer cask and handling mechanism are such that a catastrophic failure of cooling mechanisms would be precluded. In view of the long time period available in which to take corrective action it is most probable that no cladding rupture or melting would ever occur even if all normal coolant circulation modes were lost. Therefore, the credibility of this type of accident is considered extremely low. Nevertheless, it is assumed in this case that a total loss of coolant circulation has occurred during removal of the core to a refueling facility and that, through some unexplained circumstance, restorative efforts have failed. Twelve or more hours after the initial loss of flow, the moderator water would be boiled away and fuel temperatures eventually would rise above the point where the cladding would be seriously weakened. At some undefined temperature, probably above 2000°F, cladding ruptures would begin to take place releasing fission products to the cask cavity. Since it is relatively certain that the UO_2 melting temperature of 5000°F would never be reached, due to radiant heat losses to the cask walls, the release of non-volatile fission products would not be significant and the release of volatile solids and halogens would be much less than in a melting accident.

The total number of fuel pins that would suffer cladding failure has not been determined, but it has been assumed arbitrarily that radiant heat transfer to the reflector and thence to the cask walls would be sufficient to keep an outer annulus, containing 50% of the fuel pins, below the failure point. The estimation of total fission products available for release from the cask cavity is as shown in Table 10.4.

* Code of Federal Regulations, Title 10, Part 100, "Reactor Site Criteria"

FIGURE 10.3 - INHALATION DOSE - 630A NSG
OPERATING ACCIDENT

Table 10.4

Estimation of Weighted Fission Product Release -
630A Mark V Core Transfer Accident

<u>Class of Fission Product</u>	<u>Noble Gases</u>	<u>Halogens</u>	<u>Volatile Solids</u>	<u>All Others</u>
% F.P. at Equilibrium *	10	10	11	69
% Fuel Pins Ruptured	50	50	50	50
% Released from Fuel	25	5	1	0
% Not Plated Out Within Cask	100	50	30	30
Weighted % Release	1.25	.125	.0165	0

* Refers to relative contributions before decay from equilibrium reactor operation.

Although the closed cask is not truly an air-tight vessel, the expulsion of contaminated air would occur quite slowly. This is because the heating rate is low and the fuel pin failures would take place progressively over a substantial time period. Therefore, no significant pressure differential would exist. It was assumed, in this preliminary work, that the release rate from the cask was 10% per day beginning 36 hours after shutdown. This time lapse, of course, corresponds to the minimum time after reactor shutdown that the core could be removed from the steam generator, lose coolant and heat to the failure point. The effluent, in this case, was assumed to be released at ground level.

Using the foregoing source assumptions and the same ultra-conservative atmospheric diffusion conditions of the previous section, downwind dose calculations were made for both the thyroid inhalation and whole body immersion doses. The results are shown in Figures 10.4 and 10.5. As indicated, the infinite thyroid inhalation dose is below the AEC's suggested siting criterion of 300 rem beyond 650 meters. The whole body dose goes below the 25 rem criterion at a distance of 450 meters.

FIGURE 10.4 - THYROID DOSE VS. DISTANCE -
CORE TRANSFER ACCIDENT

FIGURE 10.5 - WHOLE BODY DOSE VS. DISTANCE -
CORE TRANSFER ACCIDENT

Due to the extremely low probability of this accident occurring, the highly pessimistic atmospheric diffusion assumptions used, and the relatively small resulting biological insult, it appears that the selection of a safe refueling site will be considerably easier than was predicted earlier in the 630A program.

APPENDIX A

SPECIFICATIONS AND OPERATING CONDITIONS

630A MARK V(B) NUCLEAR STEAM GENERATOR*

GENERAL SYSTEM CHARACTERISTICS

Power Rating

Shaft horsepower	27,300
Megawatts	60.43

Primary Loop Characteristics

Primary working fluid	Helium
Type of primary cycle	Closed
Thermal efficiency of overall system, %	33.7
Helium inlet temperature to reactor, °F	553.
Helium discharge temperature from reactor, °F	1200
Mass flow through reactor, lb/sec	66.2
Heat transfer area (total in active core), ft ²	1907.7
Free flow area (total in active core), in ²	417.7
Primary gas loop volume, ft ³	450
Gas circulator outlet pressure, psia	830
Gas circulator pressure ratio	1.009

Secondary Loop Characteristics

Throttle pressure, psig	1500
Throttle temperature, °F	1000
Back pressure, inches of Hg abs.	1.5
Feedwater flow, lb/hr	175,370
Final feedwater temperature, °F	415
Number of heaters	5
Feed pump power (motor driven), bhp	663
Plant thermal efficiency, %	33.7

Steam Conditions (From Boiler)

Temperature, °F	1005
Pressure, psig	1535

* Reactor specifications are provided for both the MARK V(A) and MARK V(B) reactors.

System Size

Height, ft-in.	39 5
Diameter, ft-in.	19-6
Weight, long tons	598

Operating Life Requirements

Overall system (excluding fuel and C/R), yr	20
Fuel burnup, Mw days/MT	20,000

Nuclear Characteristics

Type fuel	Low enrichment
Fuel form	UO ₂ , 6.0%
Moderator	UO ₂
Fuel inventory, pounds U ²³⁵	Light water
Fuel inventory, pounds UO ₂	393
	7,432

Shield Characteristics

Radiation specifications	
Maximum general populous dose per calendar year, rem	0.5
Maximum radiation workers dose per calendar year, rem	5.0
Shield materials	Lead, mild steel and borated water

REACTOR SPECIFICATIONS

	<u>MARK V(A)</u>	<u>MARK V(B)</u>
<u>Moderator</u>		
Material	Light water	Light water
Inlet temperature, °F	225	210
Outlet temperature, °F	240	240
Working pressure, psia	250	250
Safety valve setting, psig	258	258
Pressure drop, psi	20	15
Flow rate, gpm	2600	1000
Moderator tube maximum temperature, °F	450	450
Moderator tube wall thickness, in.	0.183	0.200

	MARK V(A)	MARK V(B)
<u>Thermodynamic Characteristics</u>		
Primary coolant	Helium	Helium
Inlet temperature, °F	553	553
Outlet temperature, °F	1200	1200
Inlet pressure, psia	828	828
Outlet pressure, psia	825.3	825.9
Safety valve setting, psig	900	900
Pressure drop, psi	2.7	2.1
Dynamic head (inlet), psi	0.25	0.19
Dynamic head (maximum), psi	0.41	0.31
Mach number (inlet)	0.019	0.016
Mach number (maximum)	0.024	0.021
Mass flow, lb/sec	65.5	66.2
Friction factor multiplier	1.0	1.0
Free flow area, in ²	362	417.7
<u>Fuel Parameters</u>		
Average maximum fuel rod temperature, °F	1320	1335
Average heat flux fuel, Btu/hr-ft ²	105,000	102,600
Average heat transfer coefficient, Btu/hr-ft ² -°F	500	435
Heat transfer coefficient multiplier (smooth tube)	1.0	1.0
Heat transfer area, ft ²	1862.7	1907.8
<u>Nuclear Characteristics</u>		
Active core length, in.	42	42
Active core equivalent diameter, in.	48.25	48.012
Forward reflector length, in.	10	10
Forward reflector material	Light water	Light water
Rear reflector length, in.	10	10
Rear reflector material	Light water	Steel
Side reflector thickness (average), in.	5.0	4.81BeO; 6.0C
Side reflector material	Beryllium	Beryllium oxide and graphite
Fuel loading, lb U ²³⁵	363	393
Fuel loading, lb UO ₂	6,856	7,432
Number of cells	109	91
Cell spacing, in.	4.4 and 4.625	4.571
Cell tube outside diameter, in.	3.826	2.940
Cell tube thickness, in.	0.183	0.200

<u>Reactor Vessel</u>	<u>MARK V(A)</u>	<u>MARK V(B)</u>
Design temperature, °F	450	Not
External operating pressure, psig	815	applicable
External design pressure, psig	900	for (B)
Material	Incoloy	
Outside diameter, in.	80.0	
Overall length, in.	66.5	
Outer shell thickness, in.	4.5	
Front tube sheet thickness, in.	1.5 center; 3.0 outer rim	
Rear tube sheet thickness, in.	1.5 center; 3.0 outer rim	
Number of cells	109	
Cell spacing, in.	4.4 and 4.625	
Cell tube outside diameter, in.	3.826	
Cell tube thickness in active core, in.	0.183	
Volume of moderator water, ft ³	226	

<u>Fuel Cartridge</u>		
Number of fuel elements	109	216
Number of fuel rods per element	55	27
Fuel matrix material	UO ₂	UO ₂
Total UO ₂ inventory, lbs	6,856	7,432
Total U ²³⁵ inventory, lbs	363	393
Enrichment, wt %	6	6
Pitch between fuel rods, in.	0.42	0.433
Active core length, in.	42	42
Cladding material	Incoloy	Incoloy
Cladding thickness, in.	0.015	0.015
Fuel rod outside diameter, in.	0.350 and 0.300	0.357

Control Elements: MARK V(B)

Dynamic tubes, actuated and manual shim tubes, type	Moderator water displacement
Safety tubes, type	Poison
Dynamic and manual shim tubes	7 cells
Actuated shim tubes, manual shim tubes, safety tubes	48 cells
Actuated shim tubes and manual shim tubes	36 cells
Dynamic tube material	Zircaloy or Be
Actuated and manual shim tube material	Zircaloy or Be
Safety tube material	Cadmium 0.050 in. thick clad with Incoloy 0.025 in. thk.
Number of actuated shim tube actuators	6
Number of dynamic tube actuators	2
Number of safety tube actuators	6

Control Rods: MARK V(A)

Number of dynamic rods	12
Number of dynamic actuators	2
(2 actuators drive 12 rods ganged in groups of 6)	
Number of shim scram rods	168
Number of shim scram actuators	7
(4 actuators drive 104 rods ganged in groups of 26)	
(2 actuators drive 46 rods ganged in groups of 23)	
(1 actuator drives 18 rods)	
Dynamic rod poison length, in.	30
Shim scram rod poison length, in.	34
Dynamic and shim scram rod poison diameter, in.	0.450
Poison material	B ₄ C powder
Poison material density, gm/cc	1.70
Dynamic and shim scram rod clad thickness, in.	0.025
Dynamic and shim scram rod material	Incoloy

SHIELD PLUG

General

Maximum design temperature, °F	450
Internal operating pressure, psig	235
Internal design pressure, psig	258
External operating pressure, psig	815
External design pressure, psig	900
Shield materials	Lead, water, steel
Volume of moderator water, ft ³	139

BOILER

General

Type	Once through
Overall height, in.	102.6
Overall diameter (bottom header flange), in.	121.5
Effective height over tubes, in.	87
Effective outside diameter over tubes, in.	80

Design Data for Normal Power

Helium flow, lb/sec	66.2
Water flow, lb/hr	175,200
Duty, 10 ⁶ Btu/hr	192.5
Inlet helium temperature, °F	1200
Exit helium temperature, °F	550
Inlet helium pressure, psia	825.9
Exit helium pressure, psia	824.6
Steam pressure, psia	1550

Tubes

Configuration	Involute
Radial plane	Serpentine
Longitudinal plane	
Number of tubes	320
Average tube length, ft	146
Tube pitch, in. (transverse/longitudinal)	0.70/0.625
Tube outside diameter, in.	0.50
Wall thickness, in.	0.065

Condensate Demineralizer

Exchangers	
Number required	2
Type	Mixed bed
Maximum inlet temperature, °F	100
Maximum inlet pressure, psig	100
Maximum pressure drop, psi	35
Capacity (each), gpm	300
Inlet water total solids, ppm	5
Maximum outlet water total solids, ppm	0.10
Weight, lbs	10,400
Regeneration equipment	Semi automatic

PRESSURE VESSEL

General

Normal operating temperature, °F	600
Design temperature, °F	700
Normal operating pressure, psia	830
Design pressure, psig	900
Safety valve setting, psig	900
Material	SA 212 B carbon steel

SIDE SHIELD

General

Shield materials	Steel, lead, water 0.6 wt % boron
Water inlet temperature, °F	115
Water outlet temperature, °F	130
Shield maximum temperature, °F	150
Water pressure	Atmospheric
Water flow rate, gpm	30

CONTAINMENT VESSEL

Design temperature, °F	< 650
Design pressure, psig	< 535
Material	SA 201 B (A300) carbon steel, lead
Overall height, ft-in.	39-5
Net volume, ft ³	1400
Inside diameter, in.	144
Diameter across circulators, ft-in.	24-1

PRIMARY COOLANT SYSTEM

Circulators

Number of units	2
Unit rating, BHP	250
Medium	Helium
Normal inlet temperature, °F	550
Maximum pressure, psig	900
Normal minimum inlet pressure, psia	822.8
Normal maximum exit pressure, psia	830
Maximum flow, lb/sec (total, 2 units)	66.2
Rated speed, rpm	8500
Design life, hr	150,000
Overhaul life, hr	15,000

WASTE HANDLING

Helium Filter

Type	Carbon
Maximum temperature, °F	100
Maximum pressure, psig	900
Average flow rate, scfm	185

Heat Exchanger

Helium inlet temperature, °F	550
Helium exit temperature, °F	100
Water inlet temperature, °F	95
Maximum helium inlet pressure, psig	900
Maximum helium flow, scfm	185

Water Level Sensor

Temperature, °F	550
Inlet pressure, psig	900

High Pressure Valve

Temperature, °F	550
Inlet pressure, psig	900

AUXILIARY COOLING SYSTEMS

MODERATOR LOOP COOLING

Circulating Pump and Motor

Number required	2
Type	Centrifugal
Medium	Water
Inlet temperature, °F	210
Maximum internal pressure, psig	258
Normal inlet pressure, psig	220
Normal discharge pressure, psig	235
Flow per pump, gpm	1000

Heat Exchanger

Normal inlet temperature, moderator side, °F	240
Normal outlet temperature, moderator side, °F	210
Maximum cooling water inlet temperature, °F	100
Normal inlet pressure, moderator side, psig	225
Normal inlet pressure, cooling water side, psig	50
Maximum inlet pressure, cooling water side, psig	100
Allowable pressure drop, moderator side, psi	5
Moderator water flow, gpm	1000
Normal cooling water flow, gpm	285

Expansion Tank

Maximum temperature, °F	200
Maximum pressure, psig	260
Normal pressure, psig	230
Capacity, ft ³	28

COMPONENT COOLING SYSTEM

Fresh Water to Sea Water Heat Exchanger

Maximum sea water inlet temperature, °F	95
Sea water outlet temperature, °F	110
Fresh water inlet temperature, °F	120
Fresh water outlet temperature, °F	105
Maximum pressure sea water side, psig	10
Maximum pressure fresh water side, psig	15
Sea water flow, gpm	90
Fresh water flow, gpm	90

Side Shield Water Heat Exchanger

Maximum cooling water inlet temperature, °F	105
Maximum cooling water outlet temperature, °F	120
Maximum shield water inlet temperature, °F	130
Maximum shield water outlet temperature, °F	115
Maximum pressure cooling water, psig	15
Cooling water flow, gpm	30
Shield water flow, gpm	30
Heat load, Btu/min	5120

Intermediate Loop Circulating Pump and Motor

Type pump	Centrifugal
Discharge pressure, psig	15
Water flow, gpm	90

Sea Water Pump and Motor

Type	Centrifugal
Discharge pressure, psig	10
Flow, gpm	90

AUXILIARY GENERATOR REQUIREMENTS AT NORMAL SEA LOAD (All power requirements shown are in kilowatts.)

SHIP SERVICE AUXILIARIES

360

PROPULSION AND REACTOR AUXILIARIES

Main condensate pump	9
Condensate transfer pump	2
Lube oil system	35
Instrument air compressor	5
Demineralizer	2
Reactor controls and instrumentation	10.0
Auxiliary cooling system pumps	4.0
Moderator circulating pumps	15.0

Total propulsion and reactor auxiliaries 82 82

Total auxiliary generator requirements 442

APPENDIX B

OPERATING AND SAFETY PARAMETERS AND ACTIONS

PARAMETERS	DATA LOGGER	CONSOLE INDICATION	ALARM	INSERT ALL SHIMS	SCRAM	QUANTITY
<u>PRIMARY LOOP</u>						
Reactor outlet gas temperature	x	x	high		high	3
Circulator outlet gas temperature	x	x				2
Circulator outlet gas pressure	x	x	high low		high low	2
Circulator speed, one			low			
Circulator speed, two	x	x	low		low	2
Circulator variable frequency power supply frequency	x	x	low			1
Circulator bearing temperature	x		high			4
Circulator stator winding temperature	x					2
Circulator cooling gas temperature	x					2
Circulator vibration	x	x	high			4
Circulator bearing touch indicator		x				4
Circulator Δp	x	x				2
Water vapor detector			high			1
Circulator cooling water flow			low			2
<u>SECONDARY LOOP</u>						
Boiler inlet water temperature	x	x ^{D*}				1
Boiler inlet water pressure	x	x				1
Boiler inlet water flow	x	x				1
Boiler outlet steam temperature	x	x ^D	high low			1

* D - point indicated on demand.

PARAMETERS	DATA LOGGER	CONSOLE INDICATION	ALARM	INSERT ALL SHIMS	SCRAM	QUANTITY
Boiler outlet steam pressure	x	x	high			1
Boiler outlet steam flow	x	x				1
Boiler feedwater conductivity	x	x	high			
<u>MODERATOR LOOP (WATER)</u>						
Moderator water tank level	x		high low			1
Moderator water tank pressure	x	x	high low			1
Moderator water plug outlet pressure	x		high low		high low	3
Moderator water flow ΔP	x	x	low			1
Moderator water heat exchanger inlet temperature	x	x	high			1
Moderator water heat exchanger outlet temperature	x	x				1
Moderator heat exchanger coolant inlet temperature	x	xD				
Moderator heat exchanger coolant outlet temperature	x	xD				
Moderator heat exchanger coolant flow	x	xD				
Moderator water pH	x		high low			
Moderator water conductivity	x		high			
Moderator water flow switch			low		low	3
<u>SHIELD WATER</u>						
Shield water tank level			high low			1
Shield water heat exchanger inlet temperature	x	xD				
Shield water heat exchanger outlet temperature	x	xD				

PARAMETER	DATA LOGGER	CONSOLE INDICATION	ALARM	INSERT ALL SHIMS	SCRAMS	QUANTITY
Shield water flow			x			1
<u>HELIUM SUPPLY</u>						
Helium supply pressure	x	x				
<u>ELECTRICAL POWER</u>						
440V, 3 ϕ , 60 cycle normal volts	x		low		low	
440V, 3 ϕ , 60 cycle emergency volts	x		low			
24V DC battery source volts	x		low		low	
DC to 120V, 1 ϕ , 60 cycle vital volts	x		low		low	
Ion chamber voltage - Positive & Negative	x		low		low	
Fission chamber voltage - Positive	x		low			
<u>REACTOR</u>						
Control element actuators						
6 shim actuators						
1 position indicator and 1 selector switch		x				1
6 full insert lights		x				6
6 full withdraw lights		x				6
2 dynamic actuators						
2 position indicators		x				2
2 full insert lights		x				2
2 full withdraw lights		x				2
6 safety actuators						
6 full insert lights		x				6
6 full withdraw lights		x				6
Source range flux	x	x				2
Source range period	x	x	short	short		2
Intermediate range flux	x	x				2

PARAMETER	DATA LOGGER	CONSOLE INDICATION	ALARM	INSERT ALL SHIMS	SCRAM	QUANTITY
Intermediate range period	x	x	short	short		2
Power range flux	x	x	high		high	3
High flux safety	x	x	high		high	2
Manual scram button					x	1
All safeties latched interlock			x	x		
Minimum power demand interlock			x	x		
Control system continuity			x			
<u>RADIATION MONITOR</u>						
Ventilation system exhaust area monitor	x	x	x			1
Main steam line area monitor	x	x	x			1
Waste gas filter area monitor	x	x	x			2
Feedwater demineralizer area monitor	x	x	x			1
Waste water storage area monitor	x	x	x			1
Overboard vent particulate monitor	x	x	x			1
Crew and passenger ventilation filter particulate monitor	x	x	x			1
Primary loop fission product detector	x	x				1
<u>COMPONENT COOLING SYSTEM</u>						
Sea water flow			low			1
Fresh water flow			low			1
Fresh water heat exchanger outlet temperature	x		high			1
Containment vessel cooler outlet air temperature	x	x	high			1
Fresh water tank level			low			

PARAMETER	DATA LOGGER	CONSOLE INDICATION	ALARM	INSERT ALL SHIMS	SCRAM	QUANTITY
<u>CONTAINMENT VESSEL</u>						
Containment vessel pressure	x	x	high			1
Fire warning			x			1
<u>OTHER</u>						
Collision detector			x		x	2

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