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GAS COOLED POWER REACTOR
FEASIBILITY STUDY
OPTIMUM
PARTIALLY ENRICHED URANIUM NUCLEAR POWER PLANT
FOR
UNITED STATES ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE

KAISER ENGINEERS
Division of Henry J. Kaiser Company
OAKLAND, CALIFORNIA

And

NUCLEAR PRODUCTS-ERCO
Division of ACF Industries
WASHINGTON, D.C.

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REVISION I
April 1, 1958

FEASIBILITY STUDY
OPTIMUM
PARTIALLY ENRICHED URANIUM
GAS COOLED, GRAPHITE MODERATED
FOR
UNITED STATES ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
CONTRACT NO. AT(10-1)-925

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INTRODUCTION

The Idaho Operations Office of the Atomic Energy Commission awarded Contract Number AT(10-1)-925 to Kaiser Engineers Division of Henry J. Kaiser Company. Kaiser Engineers awarded a subcontract for nuclear design to ACF Industries, Inc.

The scope of work includes a preliminary design and feasibility studies of gas cooled, graphite moderated, nuclear power plants. More specifically, it includes the following major responsibilities:

- a. Title I design of a prototype natural uranium fueled plant. This plant is to have a net electrical output of 40,000 kw or such larger size as necessary to meet the prescribed parameters. It is to be a prototype of the optimum plant (Item b).
- b. Feasibility study of a full scale natural uranium fueled plant optimized for cost of power.
- c. Feasibility study of a prototype partially enriched uranium fueled plant, of approximately 40,000 kw net electrical capacity, to be the prototype of the optimum plant (Item d).
- d. Feasibility study of a full scale partially enriched uranium fueled plant optimized for cost of power.

This report describes the work on the optimum, partially enriched uranium plant listed in Item d above. The scope of this feasibility study includes establishment of the general concept, selection of design parameters, determination of operating characteristics, and preparation of estimates of capital and operating costs. The design is based on technology currently available to the Commission, with only such development work as may be undertaken on a schedule consistent with early initiation of construction.

Objectives of the feasibility study are (1) to obtain values of specific power, coolant temperature and pressure which will constitute a definite advancement in technology, (2) to determine whether fuel charging and discharging under load is economically feasible, and (3) to present a design incorporating the best features of similar plants now operating or under construction and, where possible, to improve those features to reflect the latest advances in technology. In connection with this and in consideration of the United Kingdom's prominence in pioneering of gas cooled, graphite moderated power reactors, the assistance and cooperation of the United Kingdom Atomic Energy Authority is hereby gratefully acknowledged.

A hypothetical site is used as a basis for design and for estimating construction costs. The assumed site characteristics are typical for many sections of the United States.

SUMMARY AND CONCLUSIONS

This report contains a discussion of the feasibility study and cost estimates for a gas cooled, partially enriched, uranium graphite moderated power plant optimized for minimum power cost. The plant, rated at 215 mwe (net electrical), is designed for location on the bank of a river in a section of the United States having an average winter and summer climate. The plant consists primarily of one multi-purpose building which houses the reactor plant, turbine generator, warehouses, and shop. Other facilities include an office building, river pump house substation and necessary utilities.

A general plant arrangement is shown on Dwg. No. 100-P301. The estimated design and construction cost of the 215 mwe optimum plant (253 mw gross electrical) is \$104,000,000, or \$410 per kw (gross electrical). This excludes design and construction costs of the switchyard and transmission lines which are omitted from utility power plant unit costs. The cost does not include the development program necessary to achieve the maximum capability of this plant. The plant can be built on a schedule which would complete construction within 51 months from the time preliminary design is initiated. This assumes that construction is initiated six months after the start of preliminary design. The estimated cost of power produced by this plant is 15 mills per kwhr of which fixed charges are 12 mills per kwhr, based upon a lifetime plant factor of 80%, annual fixed charges at 14%, and accepted utility plant accounting practices. The cost of \$410 per kw is considerably higher than that for a conventional steam power plant. In addition to the high costs expected to be associated with the nuclear portions of the plant, it should be noted that the steam generators for a gas cooled power reactor differ materially from standard boiler designs, primarily because the heat source is clean, high-pressure radioactive gas at relatively low temperature, the mass flow of gas is very large, and leak-tightness is vital.

Reactor Plant

The rating of 215 mwe (net), corresponding to a reactor thermal rating of 600 megawatts provides a reasonable value of plant output at the desired outlet temperature and at a power cost only slightly greater than that for much larger plants. The reactor is housed in a double-walled vessel consisting of a 26' diameter carbon steel cylinder with hemispherical heads and overall height of 57' 5". The vessel outer wall is 4" thick.

The inner wall (temperature barrier) is stainless steel 1-3/8" thick, and ensures the maintenance of the outer carbon steel wall at a maximum temperature of 650 F.

The cylindrical graphite core is approximately 24' in diameter by 26' high. Vertical channels (4" diameter with a lattice spacing of 7") contain a stack of nine 29" long fuel assemblies. Cooling gas passes up through the core. Control rods enter from above; the charge-discharge of fuel is accomplished from beneath the reactor, and general reactor

servicing is from above, these operations being performed while the reactor is operating.

The reactor has a single-region core consisting of preferred fuel assemblies, i.e., a cluster of seven identical columns of stacked 2.5% enriched uranium dioxide pellets. Each column is in a stainless steel tube. Each cluster is supported within a graphite sleeve. The use of the preferred fuel assemblies, as well as alternates discussed in Section 4, would depend upon the results of a development program. The 2.5% enrichment is selected to provide an average fuel element lifetime of 10,000 megawatt days per metric ton (mwd/M.T.).

The reactor physics calculations are based on conventional two-group theory, with the appropriate constants and modifications for partially enriched uranium, graphite moderated reactors.

The graphite structure which comprises both the moderator and reflector for the reactor is supported on a grid structure within the vessel. The design of the graphite stack allows for expansion due to temperature and radiation effects (Wigner growth). Under the operating conditions of this plant, no damage from Wigner energy release can occur, even if the coolant system should fail completely.

During the feasibility study, carbon dioxide was selected as the primary coolant. However, at such time as preliminary design is performed, other gases will be considered.

The reactor is located within a concrete biological shield approximately 41' above grade. The operating level of the fuel element charge-discharge area is beneath the reactor vessel, approximately 26' below grade. The CO₂ coolant gas enters the reactor vessel at 473 F and leaves at 1,000 F; it is pumped through six steam generators, each with its own coolant loop, located around the reactor pressure vessel. Blowers circulate the CO₂ back to the reactor vessel in a closed system. The steam generators provide steam at dual pressures, with reheat, to the 250 mw (net) turbine generator.

The control rods are of boron steel and are positioned from the top of the reactor by cable drives. Conventional reactor instrumentation is provided. In addition, a burst slug detection system permits the determination of the particular channel in which a fuel element failure has occurred.

The charge-discharge machine which operates beneath the reactor is capable of removing and replacing spent fuel elements or rearranging partially spent fuel elements from one position to another. A service machine located above the reactor can be used for replacing spent control rods, assisting in the removal of jammed fuel elements and in flux plotting. Both of these machines perform their operations without requiring a

reactor shutdown.

Steam Power Plant

The steam power plant consists of six steam generator units which supply steam at dual pressures to a single turbine generator. The turbine is nominally rated at 250,000 kw, and is directly connected to the hydrogen cooled generators, rated at a total of 320,000 kva. The equipment and control systems for the steam plant and electric installation are, in general, of conventional design except for the steam generators.

The turbine generator unit operates on a dual-pressure regenerative reheat cycle with high-pressure steam supplied from the steam generators to the turbine throttle at 2,400 psia and 950 F. Exhaust steam from the high-pressure turbine is reheated in the steam generators and mixed with the low-pressure superheated steam; then the combined steam flow re-enters the turbine at approximately 750 psia and 950 F, and passes to the condenser.

The condenser is at ground level in the turbine building, and the turbine generator is on the operating floor approximately 38' above ground floor. Controls for the reactor and power plant are centralized in one control room, located on the operating floor level at a point convenient to both the reactor and steam plant facilities.

A river pump house supplies the condenser cooling water. The condensing pressure is $1\frac{1}{2}$ " Hg abs with cooling water at 70 F. The estimated turbine heat rate is 8,420 Btu/kwhr.

It is necessary to go beyond feasibility design on certain unique components in order to ensure a reasonable degree of accuracy in performance and cost estimates. The items in this category are as follows:

Charge-discharge machine

Fuel elements

Control rod drive mechanism

Service machine

Steam generators

Design Criteria

Some of the more important design criteria for the prototype and optimum plants are tabulated below. Comparative data are also presented for the optimum and prototype natural uranium GCPR plants, Calder Hall and for three other United Kingdom plants (Berkeley, Hunterston and Bradwell) which are in various stages of design and construction.

COMPARATIVE PERFORMANCE DATA FOR GAS COOLED
GRAPHITE MODERATED POWER PLANTS

	Partially Enriched Uranium		Natural Uranium					
	Prototype GCP	Optimum GCP	Prototype GCP	Optimum GCP	(1) Calder Hall	(2) Berkeley	(2) Hunterston	(2) Bradwell
No. of Reactors	1	1	1	1	4	2	2	2
Net Electrical Output per Reactor (mw)	44	215	55	220	34.5	137.5	150	150
Net Plant Efficiency (%)	35.3	35.8	30.6	31.4	19.2	25	28.3	28.3
Specific Power Megawatts Thermal per Metric Ton	6.9	7.9	1.85	2.55	1.4	2.2	2.11	2.21
Outlet Gas Temperature (°F)	1,000	1,000	800	800	637	662	745	734
Outlet Gas Pressure (psia)	387	370	280	275	110	140	165	165
Fuel Charging & Discharging	During Operation	During Operation	During Operation	During Operation	Shut-down Rec'd	During Operation	During Operation	During Operation
HP Steam Pressure (psia)	2,400	2,400	1,450	1,450	210	320	590	765
HP Steam Temperature (°F)	950	950	750	750	590	612	700	700
LP Steam Pressure (psia)	750	750	300	300	63	77	160	209

(1) In Operation (2 Reactors)

(2) Under Construction

1.0 BASIS FOR OPTIMIZATION OF A PARTIALLY ENRICHED GCPR

The basis for optimization of a partially enriched GCPR is the minimum power cost for a single-reactor plant consistent with a design available for early initiation of construction.

There are many parameters involved in the design of a power plant; the object of this report is to select the combination yielding the optimum plant. The principal parameters considered in order of selection are:

- a. Vessel size, material and wall thickness.
- b. Most suitable fuel element.
- c. Temperature of reactor coolant.
- d. Pressure of reactor coolant.
- e. Core size.
- f. Coolant gas.

Vessel Size, Material and Wall Thickness

In general, unit power costs decrease as plant capacity increases. Thermal cycle efficiency increases with increasing temperature and pressure of the reactor coolant gas. Cylindrical vessels (with hemispherical heads) present the most economical design with smaller vessels permitting higher pressures for a given wall thickness. Larger vessels permit longer fuel channels with resulting higher gas temperatures for a given maximum fuel temperature. The vessels investigated were cylindrical with inside diameters of 13, 16, 20, 26 and 36 ft. The net effect of the various parameters is such that unit power costs decrease as vessel size increased. This relationship is quite pronounced for the smaller vessels and becomes less pronounced for larger vessels. The 26' diameter vessel with 4" thick walls was chosen for the optimum plant because it provides a reasonable value of plant output at the desired outlet temperature, and at a power cost only insignificantly greater than that for still larger vessels. The size of this vessel and its operating temperature establish the gas coolant pressure and maximum core diameter. The design is enhanced by selection of a double-walled vessel, with the inner wall serving as a temperature barrier. A higher pressure is then feasible within the reactor, since the outer wall is at a lower temperature than it would be with a single-walled vessel.

Most Suitable Fuel Element and Fuel Life

The fuel element selected for the optimum partially enriched reactor consists of a cluster of seven rods each composed of 2.5%

enriched uranium oxide clad in Type 304 stainless steel. On completion of a fuel element development program, this type of element would permit a coolant gas outlet temperature of approximately 1,000 F with a fuel life of 10,000 mwd/M.T.

Temperature of Reactor Coolant

The maximum allowable surface temperature of elements with Type 304 stainless steel cladding on uranium oxide fuel is approximately 1,300 F, corresponding to a coolant gas temperature of 1,000 F.

Pressure of the Coolant

The maximum operating pressure is 400 psia based on the vessel and gas temperatures described.

Core Size

For a vessel of given size, feasibility of support and sufficient access room determine the maximum core diameter of 24'.

Coolant Gas

Carbon dioxide was selected as the primary coolant gas during feasibility study. However, its reaction with graphite may preclude its economical use at the temperatures given above. At such time as preliminary design is performed, more extensive examination will be given to other gases.

2.0 BUILDINGS AND YARD FACILITIES

Three adjoining buildings house the major operational facilities: the reactor building, turbine building, warehouse-shop building. The office and cafeteria are in a separate building near the plant entrance. Service buildings consist of a fire pump house, service water pump house, chlorination building, well pump house, oil pump house and guardhouse.

Yard facilities are as follows: a river pump station, carbon dioxide, oil and water storage tanks; warm waste retention basin with a leaching pit; an Imhoff tank with sludge drying beds for sewage disposal; fire hose houses; and railroad spurs to the oil and CO_2 storage, and to the spent fuel basin.

2.1 Reactor-Turbine Building Complex

Reactor Building

The reactor building rises five floors above grade and extends two floors below grade. It has a concrete substructure and a steel framed superstructure. A concrete biological shield encloses the reactor vessel located in the center of the building. There are six steam generators, three on the north side and three on the south side of the reactor. Below the steam generators at ground floor level are the coolant gas blowers and auxiliaries, and the gas purification system rooms.

The five floors of the reactor building contain facilities for fuel handling, shield cooling, burst slug detection, health physics, and operational control. An enclosed spent fuel storage area is at ground level and is served by a 50/5 ton bridge crane. Operating level of the fuel charge-discharge machine is below the reactor vessel bottom biological shield. The control rod operation area is serviced by a 15/2 ton bridge crane above the reactor vessel.

Turbine Building

The turbine building rises three floors above grade. The superstructure is concrete and structural steel frame. The building contains the turbine generator, condenser, steam plant auxiliaries, electrical equipment, building heating and ventilating equipment, warehouse, and shop space. A 75/5 ton service crane is provided in the turbine bay. A 10/2 ton service crane is provided in the shop area.

Heating and Ventilating

All areas of the building complex except those described below are steam heated and ventilated by four systems. All indoor winter temperatures are thermostatically controlled. The control room, amplifier room, and instrument repair shop have an independent air

conditioning system. The charge-discharge area is steam heated and ventilated by an induced draft system that exhausts to the stack above the roof. The spent fuel storage area is steam heated by unit heaters and ventilated by gravity roof ventilators. The CO₂ blower motor houses are heated by unit heaters. The CO₂ blower motors are cooled by a central air cooling system in each blower house using water coils. The shops and warehouse are heated by unit heaters.

Power

The power system for the service facilities is supplied at 480 v, 3-phase, from normal and emergency power systems. Failure-free power is 120 v ac and 125 v dc. Normal, emergency and failure-free lighting is provided. The normal and emergency lighting systems are 480-120/208 v, 3-phase. The failure-free lighting system is 125 v dc.

2.2 Office Building

The office building is a one-story pumice block structure with a steel roof frame. Office space occupies one-half the floor area and the cafeteria the other half. Building service power is supplied from a motor control center in the office building. Normal and emergency lighting are provided. The building is steam heated and ventilated by a central unit.

2.3 Service Buildings

The six service buildings for the plant are of prefabricated steel frame and insulated metal panel construction on concrete floor slabs. Equipment contained in the service buildings is as follows:

Fire Pump House: Two fire pumps and control panels.

Service Water & Well #1 Pump House: One deep well pump; chlorination equipment for domestic system, and an electrical load center.

Chlorination Building: Chlorination equipment for condenser cooling water system.

Well #2 Pump House: One deep well pump.

Oil Pump House: Fuel oil and diesel fuel transfer pumps.

Guardhouse: Radio equipment and supervisory alarm panel.

Service buildings and supporting facilities are provided with electrical power systems consisting of 4.16 kv normal power and 480 v normal and emergency power. Indoor and outdoor lighting systems are supplied at 480-120/208 v, 3-phase, from normal and emergency systems.

All service buildings are heated by thermostatically controlled unit heaters, and ventilated by gravity roof ventilators.

3.0 REACTOR AND AUXILIARIES

3.1 Core

3.1.1 Core Arrangement

The core is essentially cylindrical, 25'9" high and 24'4" in diameter, consisting of a central graphite moderator surrounded by a graphite reflector. The assembly rests on a steel grid framework within the reactor vessel. The core is pierced by 4" vertical holes for fuel elements arranged on a 7" square lattice pattern and 3-1/4" vertical holes for control rods at the center of each group of 12 fuel channels.

There are nine fuel assemblies contained in each fuel channel, with a graphite dummy sleeve top and bottom. The fuel assemblies occupy the part of each fuel channel that passes through the moderator, and the dummy sleeves are in the reflector. The fuel element consists of a cluster of seven stainless steel clad uranium dioxide elements approximately 2' long. The cluster is supported at the center of the graphite sleeve by top and bottom stainless steel spiders. The degree of enrichment (2.5%) is selected to provide an average fuel element lifetime of 10,000 mwd/M.T.

3.1.2 Reactor Physics

The calculations of the infinite and effective multiplication factors are performed using the conventional two-group theory with the appropriate constants and modifications required to provide agreement between theoretical results and the results of previous exponential and critical experiments. The initial step in the calculations is to compute and tabulate the necessary nuclear constants. This is followed by the computation of the material volumes of the reactor. The four factors of the infinite multiplication factor are then evaluated for the particular enrichment and core configurations under consideration.

The k_{∞} and k_{eff} for various core configurations are calculated for various lattice spacings, fuel slug diameters, coolant channel diameters, enrichments, core volumes and number of slugs per cluster.

Because fuel lifetime is a prime consideration in choice of core configuration, a lifetime study is done for the more promising geometries. The following design data are determined by reactivity and engineering considerations:

TABLE 3.1.1

CORE DATA

Type of Fuel	7-slug UO ₂ cluster
Lattice Pitch	7.0"
Fuel Slug Diameter	0.705"
ID of Graphite Sleeve	3.0"
Enrichment	2.5%
Core Diameter	24.4"
Core Height	25.9"
Fuel Cladding	20 mils - stainless steel
UO ₂ Density	8.2 g per cm ³ (apparent density of cored slug)
Type of Graphite	AGOT

The nuclear characteristics of the above core configuration are listed in Table 3.1.2.

TABLE 3.1.2

NUCLEAR CHARACTERISTICS OF THE 215 MWE OPTIMUM

ENRICHED URANIUM REACTOR AT OPERATING TEMPERATURE

Infinite Multiplication Factor, k _∞	1.1728 clean 1.1370 poisoned
Initial Conversion Ratio	0.68
Effective Multiplication Factor, k _{eff}	1.1266 clean 1.0922 poisoned
Thermal Flux n/cm ² -sec	
Average in Total Core	1.7 x 10 ¹³
Maximum	4.3 x 10 ¹³
Estimated Fuel Lifetime	10,000 mwd/M.T.

3.1.3 Graphite Structure

The graphite stack consists of a graphite moderator, a graphite reflector, steel base plate, top grid plate, peripheral restraints and base radial keys. The graphite moderator and surrounding reflector together form a cylinder 24'-4" diameter and 25'-9" high. The design provides for Wigner growth, thermal expansion and seismic forces. The total weight of finished machined graphite is approximately 1,000,000 lbs.

The moderator and reflector graphite is grade AGOT, density 1.7 g per cm^3 . The graphite stack consists of 14 levels of bricks arranged in five concentric rings about a center circular brick. The effect of Wigner growth on fuel channel lattice pitch is made insignificant by (1) the extrusion lines of all bricks, except the center circular brick, running radially out from the active core center channel, and (2) the high bulk temperature of the core. Vertical and horizontal neutron streaming is made negligible by rotation of alternate levels of bricks.

The graphite stack has 980 vertical fuel channels of 4" diameter on a 7" lattice pitch. In addition there are 95 vertical control rod channels at 3-1/4" diameter on a 25.23" triangular pitch. The reflector graphite surrounds the moderator with a nominal thickness of 2'-0". Steel restraint straps, horizontally disposed around the graphite stack, limit radial movement of the core bricks. Movement of the whole core relative to reactor center-line datum is restricted by radial keys at the bottom level of the core interlocked with the core support ring girder. Steel leveling base plates maintain vertical alignment, and required flatness of the graphite core. A steel top grid plate provides protection for the graphite from impact loads imposed by falling control rods, and support for the burst slug detection piping. The base and top plates have holes primarily for access to the fuel and control rod channels, and passage of coolant gas.

3.1.4 Heat Transfer and Pressure Drop

A critical factor in the overall performance of the plant is the rate at which heat generated in the reactor core can be removed economically. Good heat transfer is necessary to achieve both high power output and as high a temperature of the coolant gas leaving the reactor as possible, with fixed temperature limits for the fuel and other core materials. High power output yields lower power cost by reducing unit fixed charges; and high gas outlet temperature also gives lower power cost by permitting better plant efficiencies. Although heat removed from the core increases with increased coolant flow, pressure drop also increases, and an economic limit is reached beyond which flow should not be increased. This limit depends on the balance between pumping power required and the net power available

for sale.

In order to promote good cooling of the core in spite of the relatively poor heat transfer properties of gases, a fuel design with large heat transfer area is required. Fuel heat transfer area may be obtained (1) by using a relatively large number of small fuel rods or (2) by adding extended surfaces, such as fins, to a relatively small number of large fuel rods. For uranium dioxide fuel clad with stainless steel, the first alternative is preferred because:

- a. Stainless steel has moderate neutron absorption, so minimizing its volume in the core avoids increased enrichment.
- b. Stainless steel has relatively low thermal conductivity and, therefore, is not a very effective fin material.
- c. Uranium dioxide has very low thermal conductivity, which makes it necessary to use small diameter fuel rods if high specific power is to be attained without reaching excessive temperatures in the interior of the oxide rod.

The selected fuel assembly, a cluster of seven 3/4" slugs, represents a compromise between the heat transfer requirement of large surface area and the neutron-absorbing effect of the larger volume of steel required to clad a larger number of slugs.

The thermal power output of the reactor is determined by the heat transfer and the temperature limitations of the fuel. For stainless steel clad uranium dioxide elements, the effective temperature limit is that of the cladding, approximately 1,300 F. It is desirable to run each channel in the reactor so that this temperature limit is approached at some point in the channel. This is achieved by a variable orificing system for all channels in the reactor, except the central channels, which require the most coolant flow. The flow through the unorificed channels both limits the power level of the reactor and sets the core pressure drop.

The most important assumptions in the thermal power output analysis are:

(1) The heat generation in the fuel is distributed:

(a) Horizontally constant across each fuel slug and across each fuel cluster.

- (b) Vertically as a chopped cosine function of the fuel height.
- (2) A negligible flow of gas bypasses the fuel sleeves by flowing around the outside of the sleeves, through the openings in the graphite stack, and around the temperature barrier.

The calculation of core pressure drop takes into account entrance and exit losses for the fuel channels, pressure losses due to flow around fuel supporting members, the loss due to acceleration effects, and the friction loss for the fuel elements. Friction factors for the fuel section are obtained from standard correlations.

Thermal Performance Values

The results of the reactor heat transfer and pressure drop calculations are given in Table 3.1.4.

TABLE 3.1.4
THERMAL PERFORMANCE DATA

Fuel Configuration: 7-rod cluster of 0.75" OD rods.

Plant

1. Net Plant Electric Power, mw	215
2. Pumping Power (Electrical), mw	28
3. Auxiliary Equipment Power (Electrical), mw	10
4. Gross Plant Electric Power, mw	253
5. Steam Cycle Heat Rate, Btu/kwhr	8420

Reactor

6. Reactor Thermal Power, mw	600
7. Average Reactor Specific Power, mw/M.T. Uranium	7.9
8. Inlet Gas Temperature, F	473
9. Mean Exit Gas Temperature, F	1000
10. Reactor Coolant Flow, lb/hr	14.4×10^6
11. Reactor Inlet Pressure, psia	400
12. Total Primary System Pressure Drop, psi	40
13. Pressure Drop in Core, psi	30
14. Pressure Drop in Steam Generator, psi	5
15. Pressure Drop in Primary System Piping, psi	5
16. Maximum Cladding Temperature, F	1300
17. Coolant Mass Velocity in Central Fuel Channel, lb/hr-sq ft	7.4×10^5

3.2 Reactor Vessel

The reactor vessel consists of an outer pressure shell and an inner temperature barrier. The outer pressure shell is a cylinder with hemispherical top and bottom heads; overall dimensions are 4" thick walls, 26'8" diameter and 57'5" height. The pressure shell is designed for an internal pressure of 400 psig at 650 F and is constructed of carbon steel plate. The inner temperature barrier is 1-3/8" thick and made of stainless steel, due to the 1,000 F gas temperature. To maintain the pressure vessel outer wall at a maximum temperature of 650 F, insulation is required on the inside of the temperature barrier and coolant gas at 473 F flows in the annulus between the barrier and the vessel outer wall. The exterior of the pressure vessel is insulated.

The coolant gas enters the vessel just below the top head, flows downward through the annulus mentioned above, and then upward through the core. After being heated in the reactor core fuel channels, the coolant gas leaves the reactor vessel through nozzles provided in the top head.

Pressure shell penetrations are as follows:

- a. Nozzles are located on the top head to accommodate control rods, thermocouples, burst slug detection system piping, and gas coolant piping.
- b. Gas coolant inlet nozzles are located around the upper portion of the cylindrical section.
- c. Nozzles are located on the bottom head for thermocouples and for access to the fuel element channels when required by the charge-discharge machine.

The vessel is supported on columns equally spaced around the bottom of the vessel. The total loading to be supported is the combined weight of the pressure shell, grid support structure, and graphite stack including reflector. The grid structure consists of eight stringers 1-1/2" thick and 36" deep spaced on 28" centers and eight intercostals 1-1/2" thick and 36" deep spaced on 28" centers. A 36" deep ring girder circumscribes the grid structure. The total estimated weight of the pressure vessel, supports, and internals, including the graphite stack, is approximately 1,000 tons.

3.3 Control Rod Mechanisms

Control rod mechanisms consist of control rod drive assemblies and shock absorber assemblies. The drive assemblies, located above the reactor, insert and withdraw control rods from the core. Shock absorbers attached to the top and bottom of the rod absorb the shock from a falling rod.

Each assembly consists of a control rod, shield plug, drive mechanism, connecting linkage, spring shock absorber, and broach arrestor. All the assemblies are identical and each may be operated as either a shim rod or a regulating rod as determined by an electrical interlock system. They are magnetically supported and act as safety rods.

The function of the shim rods is to bring the reactor up to the approximate desired power level during start-up and to compensate for variations in reactivity caused by temperature changes, fuel depletion and fission products poisoning. The function of the regulating rods is to counteract transient variations of reactivity.

The winding drums and gears for the rod suspension cables are located in the upper end of the plugs. The drum is driven through a pressure seal by a drive assembly located above the top biological shield. The drive assembly includes a synchronous motor, a gear-train, a magnetic particle clutch, a centrifugal velocity damper, a position transmitter, limit switches, and a brake. The entire unit can be removed from its mounting for maintenance without disturbing the control rod.

When power failure or a scram occurs, the clutch disengages, allows the rod to fall, and unwinds the suspension cable. When the rod reaches the last 2' of its fall, the brake contacts, stopping the rod at its travel limit. The rod comes to rest when the top shock absorber contacts the grid plate on top of the reactor core. Fallen rods can be recovered and the entire control rod-shield plug assembly can be removed and replaced by the service machine during reactor operation.

3.4 Control and Instrumentation

3.4.1 Nuclear Instrumentation

The neutron flux measuring instruments show the flux level and its rate of change from source level to full-power level, and provide signals to the safety system. The function of the nuclear instrumentation is as follows:

- a. Measure neutron flux at low, or source, levels.
- b. Measure neutron flux at intermediate levels.
- c. Maintain the reactor at the flux level required by power demand.
- d. Obtain information that permits calibration of control rods.
- e. Control the distribution of neutron flux throughout the core.

Neutron Flux Plotting System

Neutron flux distribution in the reactor core is determined by lowering 34 flux detectors into special vertical core channels and computing and recording the flux level of each at approximately 100 elevations.

3.4.2 Gas Cooling System Instrumentation

The gas flow is determined by measuring the pressure drop across each of six steam generators. Temperatures and pressures in each of the main gas loops are monitored at the reactor inlet and outlet, and at the steam generator outlets.

Pressure differential alarms across the main blowers indicate malfunction of the blowers. Motor-driven control valves are located in each of the reactor inlet and outlet lines to isolate the individual generators and blowers.

The total flow into each of the two purification systems is recorded and provided with low-flow alarm. The flows through both dryer systems are also recorded and provided with low-flow alarms. A dew point analyzer having an alarm for high moisture content is located on the outlet of each dryer system.

The inlet and outlet temperatures of each of the dryer systems are also measured. Pressure instruments are provided on each filter discharge, filter intake, purification system discharge, dryer intake and cyclone separator.

An integrating flowmeter measures the amount of CO₂ fed into the system. Two oxygen analyzers, each serving three gas loops, are provided in the CO₂ system. A dew point analyzer having an alarm for high moisture content is connected on the discharge side of each blower.

3.4.3 Plant Control System

The plant control system is designed to permit the station to operate as a load-following plant when required. It is anticipated, however, that the station will normally operate as a base loaded plant.

When operating in either manner, the control system:

- a. Positions the regulating rods and also adjusts the gas flow through the reactor to maintain a constant pressure in the high-pressure steam header.
- b. Maintains the temperature of the reactor inlet gas at a fixed value by throttling low-pressure steam flow to the turbine.

- c. Regulates the temperature of the reactor outlet gas at a predetermined value by resetting the flux controller.

Steam Pressure Control - An averaged flow signal from the two high-pressure steam headers senses load changes and acts as a reference signal for the gas flow control system and the flux controller. Also, a signal from a pressure sensing element in the same system trims the gas flow control. The speeds of the six gas blowers are controlled simultaneously to maintain constant steam pressure.

When the power demand of the turbine generator is reduced at a rate faster than the reactor can follow, the pressure rises in the high and low-pressure steam headers. At a preset pressure, valves discharge the excess steam to the dump condenser, where the condensate is collected and returned to the system. These valves can discharge up to 5% of the full load steam flow. If the steam pressure continues to rise, safety valves operate to release steam to the atmosphere.

Reactor Inlet Gas Temperature Control - A controller, that senses the average temperature of the gas in the six blower discharge lines, operates two control valves in the low-pressure steam headers leaving the steam generators. The pressure and the saturated temperature of the low-pressure steam, and consequently the amount of heat transferred, is thus modified to maintain the gas temperature constant.

Reactor Outlet Gas Temperature Control - Although the gas flow is varied to meet system demands, the exit gas temperature remains relatively constant. Final adjustment of the gas temperature is effected automatically by a temperature signal which trims the power level setting of the neutron flux controller.

3.4.4 Rod Control System

Excess neutron production is controlled by inserting boron steel control rods into the core. There are two types of control rods: 37 shim rods utilized to start up the reactor, establish a rough power base, and shut down the reactor; and three regulating rods used to regulate power level.

The shim rods are designed to operate in unison, in banks, or singly. Two drive speeds are available: fast speed for reactor "rundown", and slow speed for normal operation. Each rod is held by a brake to prevent any vertical movement when power is not being applied to the drive motor.

The regulating rods and shim rods are identical, except that the regulating rods are equipped with velocity feedback mechanisms for connection to the automatic control system.

The positions of all control rods are projected onto the face of a cathode ray tube. The rod position indicator incorporates visual warning when there is no tension on the rod drive cable.

Interlocks are provided for maximum safety for both personnel and equipment.

3.4.5 Reactor Safety System

The primary function of the reactor safety system is to reduce reactor power when critical parameters approach a dangerous value.

The safety system is as simple as possible without compromising plant or personnel safety. All off-normal conditions actuate audible and visual alarm signals in the main control room.

In the preliminary design of the plant the following safety actions are preselected on the basis of the severity of the condition:

Withdrawal Prohibit safety action prevents the withdrawal of any control rods. However, insertion of control rods is still possible.

Rundown safety action inserts the rods at their maximum motor-controlled rate as long as the emergency condition exists.

Scram safety action decouples the control rods from the driving mechanism and allows the control rods to fall into the reactor core, thereby providing the most rapid shut-down of the reactor.

The safety actions, their causes, and direction in which the cause has departed from its accepted value are listed below.

CAUSE OF ACTION	Departure From Normal	ACTION			
		Alarm	Withdrawal Prohibit	Run- down	Scram
Manual Control	Energized			X	X
Flux Level	High	X		X	X
Reactor Period	Short	X	X	X	X
Instrument Power	Failure	X			X
Blower	Failure	X			X
Reactor Power to Coolant Flow Ratio	High	X		X	X
Reactor Coolant Pressure	High	X		X	
	Low	X		X	X
Reactor Coolant Outlet Temperature	High	X		X	
Failure-Free Power System Battery	Disconnected	X			X
Control Rod Power Supply	Failure	X			X
Graphite Temperature	High	X			X
Fuel Element Surface Temperature	High	X		X	
Count Rate of Start-Up Instruments	Low	X		X	
Rod Clutch Bus	De-Energized	X		X	
Rod Insert Relay	Energized	X		X	
Fuel Element Can	Ruptured	X			
Reactor Shield Coolant Temperature	High	X			
Water Vapor in Reactor Coolant	High	X			
Water Level in Any Heat Exchanger Drum	High or Low				
Heat Exchange Auxiliaries	Failure	X			
Blower Auxiliaries	Failure	X			
Burst Slug Detector Auxiliaries	Failure	X			
Instrument Air Supply Pressure	Low	X			
Error in Flux Controller	High	X			
Water Vapor in Purification System	High	X			
Flow Through Dryers	Low	X			
Flow Through Filters	Low	X			
Period Scrams	Bypassed	X			
CO ₂ in Low-Pressure Steam	High	X			
Feedwater Pressure	Low	X			
Feedwater Flow	Low	X			

3.5 Burst Slug Detection System

System Function and Description

The burst slug detection system identifies a channel containing a burst slug within 30 minutes. The system consists of stainless steel piping within the reactor vessel which permits removal of gas samples from each fuel channel. The gas sampling piping is grouped, brought through the pressure vessel shell and terminated at five 40-port rotary selector valves located in the burst slug detection room. The gas samples leaving the selector valves are cooled, filtered, and introduced into the detection instruments. The gas samples are then returned to the main cooling system by use of gas compressors.

Detection System

The detection system consists of pressure chambers where particulate fission products are electrostatically precipitated onto a wire, and deposited activity is measured by a scintillation counter. Preamplifiers, amplifiers, readout scalers, decoders, and typewriters complete the required instrumentation.

3.6 Reactor Coolant System

The gas coolant is circulated and heated in the core, and then passes through the steam generators where the heat is transferred to the working fluid which drives the main turbine. The gas leaving the steam generators is recycled to the reactor. There are six gas coolant loops, each containing one steam generator and one variable speed blower. An automatic gas filtering and drying system of the continuous bypass type is also provided.

3.6.1 Gas Coolant

The high purity CO₂ required is specified below. Supplies for initial charging of the system and for make-up are delivered to the plant in liquid form and stored until required.

Carbon Dioxide	- 99.8% minimum, by weight
Hydrogen	- not to exceed 0.1%, by weight
Water	- not to exceed 0.1%, by weight
Nitrogen	- not to exceed 100 ppm
Oxygen	- not to exceed 100 ppm
Argon	- not to exceed 1 ppm
Boron	- not to exceed 0.1 ppm

3.6.2 Operating Conditions

The gas leaves the reactor at 1,000 F and, after passing through the steam generators where it is cooled to 450 F, is returned to

the reactor at 473 F (the heat input by the blowers produces a 23 F temperature rise at full load). The gas pressure at the reactor inlet is 400 psia and the total system pressure loss is 40 psi. The reactor power control system maintains the temperature of the gas leaving the reactor at 1,000 F. The gas temperature entering the reactor is maintained at 473 F by automatic controls at the steam generators.

3.6.3 Blowers and Drives

Each blower circulates one-sixth of the total coolant flow when pumping against the total system pressure differential. The pressure rise across the blowers is 40 psi, with gas entering at 450 F and 363 psia. Control of gas flow in proportion to power demand is effected by varying blower speed. The range of automatic control for blower speed is from 20% to 100% of gas flow for full load power output.

3.6.4 Piping Systems

Each of the gas coolant loops is fitted with motor-operated isolating valves to permit the servicing of individual loops without depressurizing the reactor. The main gas piping for each circuit is in a single plane and expansion joints are provided to protect the equipment and reactor vessel against thermal expansion thrusts. Safety valve and other gas system vents are discharged through high efficiency filters to remove radioactive particulate matter before gas is vented to the atmosphere through the stack.

A gas purification system, consisting of filters and dryers, is provided to maintain coolant gas purity and cleanliness.

3.6.5 Gas Storage and Make-Up System

Carbon dioxide is stored in liquid form at 450 psig and 20 F in two 30-ton capacity insulated and refrigerated storage tanks. The refrigeration system for the storage tanks includes a compressor and submerged cooling coils in each tank. Equipment for initial charging of the reactor coolant system with CO₂ is designed to fill the system completely in eight hours. An automatic gas make-up system with heat exchanger and all required controls is provided to maintain the average pressure in the coolant system. A gas recovery system is provided for the charge-discharge machine and for blower shaft seal leakage, if required.

3.6.6 Steam Generators

The steam generators are designed to operate on a dual-pressure reheat cycle. The steam pressures and temperatures are 2,430 psia, 950 F at the high-pressure superheater outlet, and 780 psia, 950 F

at the reheater and the low-pressure superheater outlets. The feedwater temperature is 307 F to the high and the low-pressure economizers.

The steam generators are installed outdoors to reduce building cost. The elevation of the steam generators, relative to the reactor, provides gravity circulation of coolant gas to dispel the reactor decay heat during shut-down periods. The steam pressures in the high and low-pressure circuits are normally greater than the gas pressure; leakage of radioactive gas into the steam circuits therefore is avoided in the event of tube failure. Instrumentation and alarms are provided to detect excess moisture in the gas resulting from steam or water leaks in the steam generators. During periods of low power operation, the pressure in the reheaters drops below that of the gas, and the steam flow from the reheaters is therefore monitored.

Steam generators for a gas cooled power reactor differ materially from standard boiler designs primarily because (1) the heat source is clean, high pressure radioactive gas at relatively low temperature, (2) the flow of gas is very large, and (3) the gas temperatures entering and leaving the steam generator are maintained at a constant level. These factors, coupled with the necessity for absolute cleanliness and leak-tightness, result in abnormally high equipment cost. The basic design criteria established for the steam generators are: gas flow in shell; water and steam flow through tubes; all tube sheets accessible from outside gas stream to permit plugging of leaking tubes; and assembly of shell in large sections, with internals already in place and tested, prior to shipment.

3.7 Reactor Auxiliaries Power Systems

Power for reactor auxiliaries is supplied at 4.16 kv ac, 480 v ac, 125 v dc, and 120 v ac. The 480 v power is supplied from both a normal and an emergency power system. The 125 v dc and 120 v ac power are supplied from failure-free power systems.

Power for nuclear instruments, recorders, and annunciators is supplied from the 120 v ac failure-free power system.

3.8 Radioactivity Monitoring

The instrument panel in the health physics room contains recorders, indicators, annunciators and switches to measure and/or record gamma radiation and particulate activity throughout the reactor and office buildings, the waste gas stack, and the warm sump tank.

Area radiation monitors are strategically placed throughout the reactor and office buildings.

The warm sump tank, located in a pit below the charge-discharge area floor of the reactor building, receives warm waste from the laboratory, all floor drainage in the reactor building, and drainage and purge from the spent fuel storage basin.

The waste water in the retention basin, located outside the reactor building, is tested periodically for radioactivity level. When the activity is less than the prescribed limits, it is safe to discharge the water to leaching beds to percolate into the earth.

Fixed and portable personnel monitors are distributed at key points throughout the reactor building.

The radioactivity level of waste gas is monitored by the stack gas monitoring system.

3.9 Shielding

The reactor pressure vessel is surrounded by a conventional reinforced concrete biological shield with vertical side walls, and flat slabs above and below the vessel. Due to the number of primary coolant duct penetrations through the biological shield, the shield is separated into an inner and outer shield ring. The primary coolant ducts make 90° bends in the region between the two shields to minimize radiation streaming. The inside face of the inner concrete shield is located a minimum distance of 4' from the outside face of the pressure vessel; this space is provided for working space during erection. A 6' high by 3' wide removable section is provided in the lower part of the shield for access to the reactor chamber.

There are penetrations through the shielding for shield ventilation, coolant gas ducts, bundles of burst slug detection tubes, ion chamber tubes, control rod servicing, and fuel loading. Radiation streaming is prevented by labyrinth and stepped openings which are designed to allow in-flow of cooling air, and to prevent radiation scattering into areas occupied by personnel. It is necessary to surround the reactor in the radial direction by at least 10' of conventional concrete to reduce the radiation level to 0.75 mrem per hr in areas occupied by personnel.

Shielding for the Primary Coolant

Conventional concrete provides shielding for the activated CO₂ in the primary cooling loop to insure acceptable radiation levels throughout the reactor building and adjacent areas. The blower rooms, shield cooling equipment room, steam generators and reactor building roofs are isolated areas requiring controlled personnel access for limited periods of time.

Biological Shield Cooling

The shield cooling system consists of cooling air circulating equipment, with two fans arranged in parallel and located in the reactor building on the top floor. This equipment filters the air coming into the reactor chamber from the outside. A distribution duct system in the reactor chamber directs the cooling air over the inside surface of the inner concrete shield and then to the fans that exhaust it to the stack.

Storage Basin - Shielding of Spent Fuel Elements

With a water depth of 20' the radiation level is approximately tolerance (2.5 mr/hr) at the surface of the water during the transfer of baskets of newly discharged elements. When no transfer is being made, the radiation level is less than 1/10 tolerance at the surface of the water.

3.10 Fuel Charge-Discharge Machine

The charge-discharge machine is a self-propelled, shielded pressure vessel beneath the reactor. It contains the mechanisms required for charging, discharging and programming fuel elements. The machine is designed to perform all its functions while the reactor is operating to minimize reactor shut-down time, and to effect improved fuel lifetime by programming fuel elements without loss of power production.

The charge-discharge machine rides on a carriage and bridge. This provides mobility for servicing the reactor core through the fuel charging nozzles and for positioning beneath the loading, unloading and rehearsal stations.

The mechanisms of the machine are operated remotely from a console in a control room. The operator can:

- a. Position the machine beneath any selected fuel charging nozzle.
- b. Purge, pressurize and seal machine to the fuel charging nozzle.
- c. Remove the shield plug from the fuel charging nozzle and stow it within the machine.
- d. Insert into the reactor the extensible tube through which fuel elements are transferred between the selected fuel channel and the magazine of the machine.
- e. Discharge, charge, or reposition fuel within any one of the 16 channels serviced from a single nozzle.

- f. Withdraw the transfer tube, reinsert the shield plugs, and seal the fuel charging nozzle.
- g. Transport the spent fuel to the unloading station for transfer to the storage basin.
- h. Return the machine to the loading station to receive a load of new fuel.

3.11 Service Machine

The service machine is a self-propelled, shielded pressure vessel mounted above the reactor and equipped with a control console, integral carriage and bridge to provide complete mobility over the control rod nozzles and the loading, unloading, warm storage and rehearsal areas.

It is designed to permit servicing and maintenance of reactor internals during reactor operation to minimize reactor shutdown time, and thus reduce operating costs and loss of power production.

Access to the reactor from above the biological shield is through the control rod nozzles. Operations such as dislodging and removing jammed fuel elements, control rods and debris are readily effected without shut-down. The machine also transports spent fuel to the conveyor in the unloading area and unloads control rods and shield plugs in the warm storage area.

3.12 Fuel Handling

Handling facilities are provided for the storage, preparation, and charging of new fuel assembly components, and for the shielded storage and shipping of spent fuel elements.

The storage area capacity is two months' supply of new fuel assembly components - fuel elements, graphite sleeves and end spiders. The fuel elements and components assembled for charging in a fuel preparation room, loaded into the charge-discharge machine and inserted in the reactor as required.

Spent fuel element assemblies are removed from the core by the charge-discharge machine, transferred to the storage basin where they are dismantled under water shielding, and stored in racks for a minimum decay period of 100 days. They are then loaded into shielded shipping casks for transport to the processing plant. The entire core loading of fuel elements can be stored in the canal in the event of an emergency unloading. Graphite sleeves are stored in the basin for a short decay period, then dried in an oven and stored for re-use.

4.C FUEL ASSEMBLIES

This section describes lifetime evaluation, selection of materials, and fabrication techniques for the partially enriched uranium GCPR fuel assemblies.

The preferred fuel assembly is a cluster of seven identical columns of stacked 2.5% enriched uranium dioxide pellets in individual stainless steel tubes supported within a graphite sleeve by top and bottom stainless steel spiders. The assembly is 29" long and 3.75" in diameter. Nine such assemblies are stacked on top of one another in each of the vertical fuel channels in the reactor core. They are designed for bottom charging and discharging. The predicted metallurgical life of the fuel assembly is greater than 10,000 mwd/M.T., provided that the surface temperature does not exceed 1,300 F (which corresponds to a coolant gas outlet temperature of 1,000 F). However, the selection of enrichment for the fuel elements is based on the degree necessary to maintain adequate core reactivity for an average fuel life of 10,000 mwd/M.T. In the optimum plant, the enrichment is 2.5%.

The fuel pellets are die-pressed from uranium dioxide mixed with a binder and sintered to 95% of their theoretical density. The stainless steel fuel cladding and spiders are produced by conventional means. The graphite sleeves are extruded, graphitized, machined to the required dimensions, and coated with silicon carbide.

Other designs that received consideration are fuel assemblies with a cluster of 19 instead of seven fuel elements, and fuel assemblies suitable for top charging and discharging.

4.1 Fuel Life

The extent of irradiation-induced fission gas release from uranium dioxide is dependent on the density and duration of exposure. Since fission gases released from the fuel material are confined within the fuel container, the internal pressure of the fuel container rises continuously throughout the fuel lifetime. The strength of the container, therefore, places an upper limit on fuel exposure. However, exposure of uranium dioxide fuel assemblies in the enriched GCPR is governed more by the inevitable decrease in reactivity as exposure continues than by metallurgical difficulties in containing or controlling the effects of long-term exposure. The tough, stainless steel can withstand fission gas pressures greater than those developed by exposure of the fuel to the limit set by reactivity requirements alone. This is true even if it is assumed that 50% of the fission gases are released from the fuel. Therefore, the expected lifetime of the enriched GCPR fuel assemblies is that determined by reactivity requirements: 10,000 mwd/M.T.

4.2 Selection of Materials

The greater reactivity of enriched uranium systems permits the use of refractory ceramic fuel pellets and the moderately neutron-absorbent structural materials such as stainless steel. Ceramic fuel pellets permit a higher fuel and gas coolant operating temperature with a corresponding increase in the thermal efficiency. Since they are more resistant to radiation damage, ceramic pellets allow longer fuel exposure with a corresponding decrease in fuel cycle costs. Tough, heat resistant structural materials, such as stainless steel, further simplify the problems of fuel containment and mechanical design.

4.2.1 Fuel Materials

Ceramic uranium dioxide in pellet form has been chosen as the fuel material for the enriched uranium GCPR. Its technology is more highly developed than that of other ceramic fuels, and irradiation-induced changes in its density and dimensions are very slight, although some cracking of pellets occurs. The crystal lattice of uranium dioxide retains most of the fission gases produced during irradiation. Even so, the extent of fission gas release can be reduced and available reactivity increased by increasing fuel density. Therefore, the fuel pellets are sintered to 95% of their theoretical density.

Although uranium nitride and uranium carbide appear to have advantages as fuel materials, the present lack of data on their irradiation behavior and of suitable fabrication techniques precludes their use.

4.2.2 Cladding Materials

Fuel element design depends on the cladding material to support the fuel pellets and retain fission products within the element. To meet these requirements the cladding material should have:

- a. Low, or moderate, neutron absorption.
- b. Resistance to coolant, or fission gas, penetration.
- c. Chemical compatibility with the coolant gas.
- d. High thermal conductivity.
- e. A coefficient of thermal expansion nearly equal to that of the fuel.
- f. Good fabricability.

g. Structural strength and low creep rates at high temperatures.

Type 304 austenitic stainless steel fulfills these requirements satisfactorily and is the least expensive. Several other materials, among which are beryllium, zirconium and its alloys, and molybdenum, offer advantages which may be made available by later developments, but the limitations of existing technology and cost preclude their consideration in this feasibility study.

4.2.3 Support Materials

To be acceptable, support materials should have negligible creep rates at the design temperature, low neutron absorption, and adaptability to mass production in the required shapes. Graphite was selected as the preferred material for the fuel assembly support sleeves, because other materials with acceptable mechanical and nuclear properties, such as beryllium and silicon carbide, are more expensive and more difficult to fabricate. Stainless steel was selected for the top and bottom fuel assembly support spiders, because graphite spiders of the required strength would restrict coolant flow excessively. The use of graphite support sleeves in a CO₂ atmosphere at 1,000 F is open to question, due to the possibility of a significant rate of reaction between CO₂ and graphite. One possible solution would be to coat the graphite with a layer of silicon carbide from 5 to 10 mils thick.

4.3 Mechanical Design

The sintered uranium dioxide fuel material is formed as a cored, right circular cylinder and enclosed in a thin-walled stainless steel can.

The mechanical design of the fuel assemblies must satisfy the extremely long periods of irradiation necessary to achieve high burnup, and the high probability that the associated thermal cycling will cause fragmentation of the fuel pellets. Because information on the extent of fission gas release under these conditions is incomplete, the assumption is made that 50% of the fission gases may be released into the voids within the can. The can is, therefore, designed as a pressure vessel with sufficient strength at maximum operating temperatures to contain the released gases. The open center fuel rod design was chosen to limit the pressure rise to values compatible with practical can wall thicknesses.

The fuel elements are loaded in tension by suspending them from the top spider. If they were loaded in compression (by allowing them to rest on the bottom spider) they might be subject to warping or bowing.

4.4 Fuel Assembly

The fuel assembly is a cluster of seven identical columns of stacked enriched uranium dioxide pellets in individual stainless steel cans supported within a graphite sleeve by top and bottom stainless steel spiders. The centers of six of the fuel rods are equally spaced on a 2" circle; the seventh fuel rod is at the center of the circle. The fuel assembly is designed for a coolant gas outlet temperature of 1,000 F and for bottom charging and discharging.

4.4.1 Fuel Rods

Each fuel rod consists of 35 cored, enriched uranium dioxide pellets. The rods are 26.6" long. Each pellet is 0.7" long, 0.7" OD and 0.32" ID. A magnesium oxide spacer pellet 0.19" long and 0.7" in diameter is placed at each end of each column of fuel pellets for thermal insulation of the end caps of the surrounding stainless steel can.

4.4.2 Steel Components: Cans, End Caps, and Spiders

The fuel and insulating pellets are placed in Type 304 stainless steel cans 27.5" long and 0.7" ID. The wall thickness is 20 mils. Top and bottom end caps of the same material, whose ends are shaped and slotted to fit into the spiders, are welded to the ends of the can to ensure gas-tight closures. The seven top end caps of the fuel rods in any one cluster are welded to the top support spider, from which they and the bottom spider are suspended. After the sub-assembly of fuel rods and top spider has been inserted into the graphite sleeve, alternate bottom end caps of the peripheral rods are welded to the bottom spider which laterally positions, but does not support, the lower ends of the rods. Only three of the rods are welded to the bottom spider to minimize stresses developed by differential thermal expansion.

The top spider is made of Type 304 stainless steel. It consists of a rim 3.28" OD, 0.09" thick, and 0.50" high, with a diametral bar crossed at right angles by two other bars 1" apart. The height and thickness of all bars are the same as those of the rim. The bottom spider is similar in all respects except that it is only 3/8" high, and it is notched to receive the lower rod end caps.

4.4.3 Graphite Support Sleeve

The fuel rods and spiders are surrounded by a graphite sleeve 29.0" long and 3.75" OD which may require silicon carbide or other suitable coating. The sleeve wall thickness is 0.375". Internal shoulders are machined at each end of the sleeve to receive the top and bottom spiders.

The entire load of the internal sub-assembly is carried by the internal shoulder at the top of the sleeve. The sleeves also protect the moderator graphite and transmit the load of the fuel assemblies to the lower ends of the fuel channels.

4.5 Alternate Assemblies

4.5.1 Nineteen Rod Cluster Fuel Assembly

A fuel assembly with a cluster of 19, rather than seven, fuel rods was considered as one of the variations in the optimization study of the enriched reactor. The thermal performance of the 19-rod assembly is superior to that of the 7-rod assembly, but this advantage is accompanied by the offsetting disadvantages of a lower conversion ratio and the necessity for higher enrichment.

4.5.2 Modified Fuel Assembly for Top Charging and Discharging

A fuel assembly suitable for insertion and removal from the top of the reactor was studied. For top insertion and removal, it is desirable to minimize the number of fuel assemblies per channel and, therefore, to increase fuel assembly length. The increase in length is limited by the restrictions thus imposed on fuel rearrangement for even burnup and by the space available above the reactor for handling fuel assemblies.

A suitable compromise between these opposite effects is a fuel assembly 65" long. This increased length would probably result in the following changes:

- a. Increased core fuel loading.
- b. Substitution of a more shock-resistant material for the graphite sleeve.
- c. Addition of a grappling tool receiver on the top spider.

If, during preliminary design, top charging and discharging are shown to be advantageous both economically and operationally, a fuel assembly of this type described above will be adopted.

4.6 Fabrication of the Fuel Assembly

Uranium dioxide, produced from uranium hexafluoride by the ammonium diuranate precipitation process, is mixed with small amounts of titanium dioxide and a binder. The mixture is die-pressed to form "green" pellets, which are then sintered to 95% of their theoretical density in a hydrogen atmosphere furnace at 1,700 C to form the

finished fuel pellets.

Powdered magnesium oxide and a suitable binder (if necessary) are mixed and die-pressed to form "green" pellets, which are then sintered to the required density to form the spacer pellets.

The fuel cans are made from Type 304 stainless steel seamless annealed tubing, suitably cut, inspected, and cleaned. The end caps are produced by automatic screw machines or by casting. The spiders are cast.

The graphite sleeve will be extruded with more than the required wall thickness, and then finished by boring, reaming, and outside turning. After the sleeves are cut to length, the inside shoulders are counterbored and the entire surface may be impregnated and coated with silicon carbide.

5.0 STEAM POWER PLANT

General

The steam power plant consists of six steam generator units which supply steam at dual pressures to the turbine generator. Equipment and control systems are of conventional design except for the steam generators and the primary cycle equipment. The plant is designed to operate on a dual-pressure, reheat steam cycle based on coolant gas temperatures of 1,000 F entering and 450 F leaving the steam generators. The improved performance of the dual-pressure reheat system over the single-pressure reheat cycle is shown below:

	<u>Turbine Heat Rate Btu/kwhr</u>	<u>Gross Power Output, kw</u>
Dual-Pressure Reheat Cycle	8,420	250,000
Single-Pressure Reheat Cycle	8,960	235,000

Turbine Generator

The turbine generator consists of a cross-compound, double flow, condensing, reheat turbine rated at 250,000 kw, 3600/1800 rpm, and direct-connected hydrogen cooled generators, 18,000 v, 3-phase, 60 cycle, 320,000 kva, 0.85 pf 0.64 short circuit ratio, 45 psig hydrogen pressure. There are three 375 v dc motor generator excitors. High-pressure steam is supplied at 2,400 psia and 950 F. The combined reheat and low-pressure steam re-enters the turbine at 750 psia and 950 F. Full load back pressure is 1-1/2" Hg abs.

Feedwater Cycle

Condensate is pumped from the main condenser through the hydrogen coolers, air ejector condensers and the low-pressure heaters to the deaerator and thence through the high-pressure heaters to the low-pressure and high-pressure economizers. Demineralized water is used for make-up to the system. Steam supply to the feedwater heaters is from extraction outlets on the turbine. The low-pressure heater drains cascade to the condenser, and high-pressure heater drains cascade to the deaerator.

Main Condenser and Auxiliaries

The main condenser is a single pass, divided water box type complete with two circulating water pumps, three condensate pumps, two air ejectors and all required accessories. Condensing pressure is 1-1/2" Hg abs with 70 F river water.

Dump Steam Condenser and Auxiliaries

A dump steam condenser, complete with all required auxiliaries, is designed to handle 5% of the total output from the steam generators and operates at atmospheric pressure. Electrical apparatus is connected to the emergency power system.

Boiler Feed Pumps

Three low-pressure and three high-pressure motor-driven boiler feed pumps are furnished, two normal and one standby. Two motor-driven boiler feed pumps, connected to the emergency power system, are provided for operation during reactor shut-down periods.

Steam Plant Controls

A conventional control system is provided for the steam power plant. A constant pressure is maintained at the turbine throttle and the control system permits base load and load following operation. A single control room is provided for the steam power plant and reactor.

Electrical Generation and Transmission Facilities

The electrical generation and transmission facilities include the generators, generator transformer, two normal auxiliary transformers, start-up transformer and 230 kv substation with two transmission circuits, all arranged to operate as a unit system. The generators, generator transformer and two normal auxiliary transformers are connected by isolated phase bus, without intervening circuit breakers, to form an integrated operating unit.

The transmission system facilities include a 230 kv substation with a 4-section ring bus and four oil circuit breakers to connect the unit generator system and a start-up transformer to two transmission system circuits.

The generator transformer is rated 280,000 kva, 18 kv split-winding, delta low voltage, 230 kv wye high voltage, with tap changing under load. The two normal auxiliary transformers are each rated 25,000 kva, 18 kv delta high voltage, 4.16 kv split-winding, wye low voltage. The start-up transformer is rated 25,000 kva, 230 kv wye high voltage, 4.16 kv split-winding, wye low voltage, with tap changing under load. Power for steam power plant auxiliaries is supplied at 4.16 kv ac, 480 v ac and 125 v dc.

6.0 UTILITY SYSTEMS

6.1 Water Systems

Because the optimized enriched uranium reactor is located at a hypothetical site, the following design assumptions are made:

- (1) condenser cooling water is assured because of location on the bank of a river carrying sufficient flow to prevent exceeding the allowable temperature rise in the river, and
- (2) there is adequate ground water supply available at a depth of 150'.

Domestic water is pumped from two wells, one of which is on stand-by service, and is chlorinated and stored in a tank.

Make-up water for the steam generator is supplied from the domestic water storage tank and demineralized by cation and anion de-ionizers.

Bearing cooling water is supplied from the domestic water storage tank.

Condenser cooling water is supplied by two pumps at the river pump station. Chlorinators apply intermittent treatment for slime and algae control. After passing through the condensers, the water is returned to the river downstream from the pumps. Two auxiliary pumps, one of which is standby, supply river water to the dump condenser, service cooling water heat exchangers, turbine lubricating oil cooler, and diesel generator heat exchanger. Upon power failure, automatic controls supply emergency power to one or the other of the auxiliary pumps.

Fire protection water is pumped from the river by two vertical turbine pumps, one on normal electric power and one driven by diesel engine.

A distribution loop supplies yard hydrants, building standpipes and sprinkler systems. The fire loop is pressurized from the domestic water system during periods of no usage.

6.2 Air Systems

Plant air is provided for general utility and for the service and refueling machines by a compressor, receiver, and distribution system. Instrument air is provided by a separate compressor with a dryer and receiver. The plant air system is piped to permit cross connection with the instrument air system in the event of instrument air compressor failure.

6.3 Sewage Disposal

Sanitary sewage drains from the building sanitary fixtures through the yard sewage system to a sewage lift station. Duplex sewage pumps lift the sewage to the Imhoff tank. Sludge is drained to a sludge drying bed. The effluent is chlorinated and piped to the river. Storm runoff is conducted to the river by surface ditches and by an underground storm sewage system.

6.4 Radioactive Waste Disposal

Gaseous wastes are exhausted to atmosphere from an 11' diameter stack 150' above the high point of the reactor building. Biological shield cooling air containing radioactive argon, and primary coolant containing CO₂, CO, and radioactive argon without fission products, can be released under all meteorological conditions. Primary coolant containing fission products from a burst fuel element will contain as much as 500 curies. Release of this gas is permissible only during favorable meteorological conditions in which case weekly tolerance may be exceeded in the plant area by 25%. This is acceptable because of infrequency.

"Warm" liquid wastes consist of those liquids which by dilution and temporary retention will have their radioactivity reduced to a level which will permit discharge either into the ground or the river. If the activity cannot be reduced to this level, the liquid is considered "hot", and will be shipped off-site for disposal. Solid wastes are sealed in suitable containers and buried.

6.5 Electrical Distribution Systems

Normal power for the station auxiliaries and station supporting facilities is supplied from 4.16 kv, 3-phase, 60 cycle, resistance grounded, and 480 v, 3-phase, 60 cycle, solidly grounded radial type power distribution systems.

Emergency power is provided for essential plant auxiliaries from an emergency diesel engine generator integrated with the 4.16 kv normal power distribution system.

Failure-free power is provided for critical plant and reactor auxiliaries from 125 v dc and 120 v ac failure-free power systems.

6.6 Communications and Alarms

Alarm systems for the area consist of a fire alarm system and a supervisory alarm system, each repeating signals into the district central fire station.

Telephone service for the area is by a commercial telephone system which includes paging signal chimes and horns.

Intercommunication networks consist of systems for nuclear and steam process operation and maintenance.

6.7 Steam Distribution for Building Heating

Steam for building heating, CO_2 vaporization and oil heating is supplied by a boiler in the turbine building.

6.8 Fuel Oil and Diesel Fuel System

Fuel oil is received, pumped to a storage tank, and recirculated from a suction heater in the tank to the boiler, with excess return to the tank.

Diesel oil for the emergency diesel generator and start-up of the boiler is received and pumped to a yard storage tank.

7.0 COST ESTIMATE AND SCHEDULE

7.1 Design and Construction Schedule and Preliminary Estimate of Cost

A design and construction schedule and an estimate of cost for the optimum plant are based on the assumption that the work is performed under cost-plus-fixed-fee contracts. The schedule and estimate assume that construction begins shortly after the start of detailed design and proceeds concurrently with engineering, significantly shortening the overall project. On this basis the facility would be completed in 51 months, and is estimated to cost approximately \$107,000,000; the cost is \$104,000,000 excluding the cost of design and construction of the switchyard and transmission lines.

It is assumed for estimating purposes that preliminary engineering (Title I) requires approximately three months to complete. Construction would begin approximately three months after the start of detailed design and would require approximately 45 months to complete. Costs include escalation commensurate with this time schedule.

Estimates of cost for the optimum plant assume construction at a hypothetical site with characteristics typical for many sections of the United States.

7.1.1 Design and Construction Schedule

Preliminary Engineering (Title I)	3 months
Detailed Design (Title II)	18 months
Construction, Inspection (Title III)	45 months
Combined Design and Construction Period	51 months

7.1.2 Preliminary Estimate of Cost

<u>Engineering</u>	\$10,000,000
<u>Construction</u>	
Site Work	2,850,000
Buildings	8,370,000
Reactor	17,990,000
Power Generation	48,880,000
Electrical Distribution	2,890,000
Sub-Total	<u>80,980,000</u>
Contingency	<u>16,020,000</u>
Total Construction Cost	<u>97,000,000</u>
Total Engineering and Construction Cost	\$107,000,000*

* \$104,000,000 excluding design and construction cost of switchyard and transmission line.

7.2 Estimated Cost of Operation

The cost of operation consists of:

- a. Annual fixed charges on the total capital cost.
- b. Operating and maintenance expenses, including general and administrative expense.
- c. Nuclear fuel costs.

Annual fixed charges and operating and maintenance costs are computed on the basis used by public utilities in the United States.

Nuclear fuel costs are estimated from net burnup of fissionable material, cost of fuel element fabrication, fuel rental, and cost of chemical processing of spent fuel elements. These are based on fuel life of 10,000 mwd/N.T. and a plant factor of 80%.

A lifetime operating plant factor of 80% was selected by the Commission for this report to allow unit cost comparisons with other proposed Commission reactor power plants. At this plant factor, the lifetime average production for the plant is 1.5×10^9 kwhr/year.

The 215 mwe power plant lifetime unit cost is estimated as follows:

Fixed Charges (14%)	11.6 mills/kwhr
Operating and Maintenance	1.3
Nuclear Fuel Costs	<u>2.4</u>
Total Cost	15.3 mills/kwhr

7.2.1. Annual Fixed Charges

These charges are taken as a percentage of the total capital cost. Included in capital costs are construction costs, interest charges during construction, plant start-up costs, operating spare parts inventory, working capital, and fuel fabrication cost of the initial reactor core.

The percentage used to compute the annual fixed charges is composed of depreciation allowances on the physical plant, ad valorem taxes on the physical plant, miscellaneous replacement costs, state and federal income taxes on the gross income and the net income allowed the average utility company after deduction of operating and other production expenses.

The assumptions used for computing the annual fixed charges are based on the following percentages of the total capital cost:

- a. Average utility company financing consists of 50% bonds, 20% preferred stock, and 30% common stock. Average allowable return is assumed at 6% of total investment.
- b. Depreciation is based on straight line rate of 2-1/2%.
- c. Ad valorem taxes are estimated to be 2%.
- d. The cost of miscellaneous parts which need replacement during the life expectancy of the plant is estimated to be 0.2% of the initial total capital cost.
- e. State income taxes are estimated to be 4%, and federal taxes 52% of the gross income.

The percentage of annual fixed charges based on the above assumptions is computed to be 14.5% of the initial total capital. This is in substantial agreement with the modern steam generating plant average of 14%. The initial total capitalization is reduced each year by the annual depreciation allowance; therefore, the percentage applied against the initial

capitalization can be reduced over the plant lifetime.

During the plant lifetime, an average annual fixed charge percentage of 9.2% would ordinarily be applied to the initial total capital cost established below. However, in an effort to maintain uniformity of fixed charge calculations among its various contractors working on nuclear power plant studies, the AEC requested that lifetime fixed charges (in mills/kwhr) be calculated based on 80% plant factor and 14% fixed charge on total capital cost. Therefore, the costs referred to on pages 3 and 45 reflect this method of calculation.

LIFETIME ANNUAL FIXED CHARGES

<u>Capital Cost Components</u>	<u>Capital Cost (Millions)</u>	<u>Rate</u>	<u>Annual Cost (Millions)</u>
Construction and Engineering (Excluding Substation and Transmission Lines)	\$107.0	9.6%	\$10.0
Fuel Fabrication Inventory	2.3	7.1%*	0.2
Interest During Construction (12.5% of Construction and Engineering)	9.6	7.6%**	0.7
Working Capital	3.3	5.1%***	0.2
Start-Up Costs	1.0	7.6%**	0.1
Equipment Stores	1.0	9.6	0.1
Total Costs	\$124.0	9.2% 14.0%	\$11.3 17.4

* 9.6% less 2.5% depreciation.

** 9.6% less 2.0% ad valorem taxes.

*** 9.6% less taxes and depreciation.

7.2.2 Operation and Maintenance Costs

Operating expenses of the power plant consist of plant personnel salaries and wages, payroll benefits, operating and maintenance supplies, liability insurance, and general administrative expenses.

It is estimated that a total of 108 employees is required to operate and maintain a nuclear power generating station of this type and size. The maximum number expected to be in the plant on a single shift is 54. The plant organization has 10 clerical and supervisory, 22 maintenance and 76 operating employees. The cost of operating and maintenance supplies is based on the costs of a comparable steam power generating

station, and adjusted to meet the requirements of a nuclear power generating station.

Summary of Operating and Maintenance Costs - The estimated annual operating and maintenance expenses are tabulated below:

	<u>Annual Costs</u>
Wages and Supervision	\$ 768,000
Payroll Taxes and Benefits @ 14%	108,000
Operating Supplies and Expenses	225,000
Maintenance Supplies and Expenses	150,000
Demand Charges	60,000
Insurance	480,000
General and Administrative Expense at 14% of Production Cost Less Fuel	249,000
 Total	 \$2,040,000
 Say	 \$2,000,000

7.2.3 Nuclear Fuel Costs

Fuel cost is computed from:

- a. Net burnup of fissionable material.
- b. Cost of fuel element fabrication.
- c. Fuel rental charges.
- d. Cost of processing spent fuel elements including shipping charges.
- e. Losses of fuel during fabrication and chemical processing.

The assumptions used in computing the cost of nuclear fuel are:

- a. The average fuel exposure is 10,000 mwd/M.T.
- b. The discharged fuel contains 1.4% of U²³⁵ and 8.5 grams plutonium per kg of fuel.

- c. In computing fuel rental charges, the 2.5% enriched uranium fuel is valued at \$296 per kg of uranium, and leased from the AEC at 4% per year.
- d. The reactor is loaded with 77 metric tons of 2.5% enriched uranium and operates at 80% plant factor and 600 mw thermal rating. The fuel discharged from the reactor is 0.048 metric ton per day.

Fuel Cycle Summary - Optimum Reactor

Annual Fuel Costs (18 metric tons per year):

	Cost Per kg Uranium	Annual Cost (Thousands)
Net Fuel Costs	\$86	\$1,500
Fabrication Cost	30	540
Fuel Inventory	56	1,000
Fuel Processing	36	<u>650</u>
Annual Total Cost of Fuel		\$3,690
	Say	\$3,700

8.0 SUMMARY OF STUDIES

General

The introductions and summaries from selected studies are included as follows:

CORE LIFETIME AS A FUNCTION OF IRRADIATION FOR ENRICHED URANIUM, GRAPHITE MODERATED, GAS COOLED REACTOR SYSTEM (ACF STUDY NO. 133)

Introduction - The purpose of this study is to determine the core lifetime for a partially enriched uranium, gas cooled, graphite moderated reactor.

Summary and Conclusion - The core lifetime is a prime consideration in selection of design parameters of a partially enriched reactor; consequently, a lifetime calculation is done for a number of core geometries. The core lifetime in megawatt days per metric ton (mwd/M.T.) is determined for:

- (1) A 7-slug cluster of 0.75" diameter rods with 2.0% enriched uranium and lattice spacings of 7.0", 8.0", and 9.0".
- (2) A 19-slug cluster of 0.5" diameter rod with 2.5% enriched uranium and lattice spacings of 7-3/4" and 9".

For each combination of fuel enrichment and lattice spacing, the k_{eff} for the poisoned core is plotted as a function of fuel life in megawatt days per metric ton of fuel in the core. The core lifetime is then taken as the irradiation value at which the k_{eff} for the particular case under consideration is reduced to unity. Table I summarizes the results:

Table I
Core Lifetimes

	2.0% Enrichment			2.5% Enrichment	
	0.75" Rod Diameter	7 Rods/Cluster		0.50" Rod Diameter	19 Rods/Cluster
Lattice Spacing	7"	8"	9"	7-3/4"	9"
Lifetime - mwd/M.T.	10,200	11,000	9,200	11,000	11,100

LATTICE STUDY FOR AN ENRICHED URANIUM, GRAPHITE MODERATED, GAS COOLED REACTOR (ACF STUDY NO. 120)

Introduction - The infinite multiplication (k_{∞}) and the critical buckling are two of the principal factors which determine the required enrichment, optimum spacing between fuel channels, fuel slug size and coolant channel dimensions.

This study determines the variation of k_{∞} and k_{eff} , with enrichment and the lattice dimensions of a gas cooled, partially enriched uranium, graphite moderated nuclear reactor with uranium oxide fuel.

Summary and Conclusion - The fuel assemblies considered are 7 and 19-slug clusters of uranium dioxide clad with stainless steel. The lattice parameters varied to determine the corresponding nuclear properties of the core are: fuel slug diameter, coolant channel diameter, lattice spacing, core size and fuel enrichment. For each specific set of geometric parameters, the thermal utilization factor, fast fission factor, resonance escape probability and neutron produced per thermal neutron absorbed in the fuel are calculated and are presented in tabular form. The infinite and effective multiplication factors are determined, and those of the critical systems are presented in graphical form as a function of lattice spacing.

The calculations are based on the hot-clean reactor.

STEAM CYCLE ANALYSIS FOR AN OPTIMUM ENRICHED, GRAPHITE MODERATED,
GAS COOLED REACTOR (KE STUDY NO. 13A)

Introduction - The purpose of this study is to analyze the steam cycles for the partially enriched uranium GCPR and to determine the most efficient cycles for various reactor coolant temperatures. The range of coolant temperatures considered is 350 F to 550 F leaving and 900 F to 1,200 F entering the steam generators. The cycles and heat rates developed in this study are used as parameters in the determination of optimum reactor operating temperatures.

Summary and Conclusion - Steam cycles analyzed include single and dual-pressure reheat systems. The critical variables are inlet and outlet temperatures for the primary fluid (CO_2), and the approach temperature, feedwater temperature, steam pressures and steam temperatures for the working fluid (steam). The ratio of high-pressure steam flow to total steam flow is a critical variable for the dual-pressure systems. Due primarily to the relatively low gas temperatures available from the enriched uranium GCPR, the dual-pressure reheat cycles have a considerably lower heat rate than comparable single-pressure reheat cycles. Optimum feedwater temperatures are lower than for normal regenerative feedwater heating cycles to permit the use of higher pressures in the steam cycle. The feedwater temperature, steam pressures and temperatures, and per cent of high-pressure steam are optimized to obtain the lowest cycle heat rate.

Dual-pressure reheat cycles are recommended for use in establishing the optimum plant design. A summary of the heat rates and design data for the most efficient steam cycles for various gas inlet and outlet temperatures is shown in Table II.

TABLE II
STEAM CYCLE SUMMARY

<u>Coolant Inlet Temp. F</u>	<u>Coolant Outlet Temp. F</u>	<u>Feedwater Temp. F</u>	<u>Superheater and Reheater Outlet Temp. F</u>	<u>HP Steam Pressure PSIA</u>	<u>LP Steam Pressure PSIA</u>	<u>Per cent of Steam Flow from HP Evaporator</u>	<u>Heat Rate Btu/kwhr</u>
900	350	165	850	2200	350	68	9420
900	450	325	850	2400	550	72	8730
900	550	440	850	2400	720	91	8390
950	450	320	900	2400	630	78	8520
1000	350	207	950	2400	320	78	8620
1000	450	308	950	2400	750	78	8440
1000*	450	205	950	1120	*	100	8960
1000	550	479	950	2400	1100	85	8010
1100	350	230	1000	2400	280	88	8380
1100	450	335	1000	2400	800	88	8210
1100	550	479	1000	2400	1380	95	7780
1200	350	265	1050/1000	2400	260	93	8320
1200	450	365	1050/1000	2400	850	93	8030
1200*	550	499	1050/1000	2400	*	100	7810

* Single pressure reheat cycle. All other values listed are for dual pressure reheat cycles.

9.0 SUMMARY OF DESIGN DATA

The principal design criteria and performance characteristics for the partially enriched uranium gas cooled power reactors are as follows:

	<u>Units</u>	<u>Prototype Plant</u>	<u>Optimum Plant</u>
Net Electric Power Output	mwe	144	215
Reactor Power	mwt	125	600
Power Added by Blowers	mwt	3.2	28
Power to Steam Generators	mwt	128.2	628
Gross Electric Power	mwe	51	253
Auxiliary Power, Including Blowers	mwe	7	38
Plant Efficiency - Gross	%	40.8	42.1
- Net	%	35.3	35.8
Coolant Gas	- Type		CO ₂
	- Flow	lbs/sec	805
	- Weight	tons	12
Reactor - Outlet Temperature	F	1000	1000
- Inlet Temperature	F	463	473
- Outlet Pressure	psia	387	370
- Inlet Pressure	psia	400	400
- Fuel Loading	M.T.	18	77
- Specific Power	mwt/M.T.	6.9	7.9
- Enrichment	%	3.0	2.5
Number of Fuel Channels		344	980
Fuel Channel Spacing, Sq Pitch	in.	7	7
Shim Safety Rods, Number		21	37
Regulating Rods, Number		3	3

	<u>Units</u>	<u>Prototype Plant</u>	<u>Optimum Plant</u>
Reactor Pressure Vessel - Shape		Cylinder with hemispherical heads	Cylinder with hemispherical heads
- Dimensions		18'-0" ID x 40'-0" overall	26'-0" ID x 57'-5" overall
Graphite Moderator and Reflector Weight	tons	170	510
High Pressure Steam - Pressure	psia	2400	2400
- Temperature	F	950	950
Low Pressure Steam - Pressure	psia	750	750
- Temperature	F	950	950
Condensing Pressure	in. Hg abs	1-3/4	1-1/2
Turbine Heat Rate	Btu/kwhr	8500	8420
Heat Cycle Efficiency	%	40.1	40.5

A B C D E F G H I J K L M N O P Q

215 MWE (NET) ENR. GCPR DRAWING INDEX

DWG. NO.	TITLE
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CIVIL

5714-GCPR-100-P 301	PLANT GENERAL ARRANGEMENT
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ARCHITECTURAL

5714-GCPR-601-A-P 301	REACTOR - TURBINE BUILDING PLAN AND ELEVATION
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601-A-P 302	REACTOR - TURBINE BUILDING PLAN AND SECTION A-A
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ELECTRICAL

5714-GCPR-602-E-P 301	ELECTRICAL ONE LINE DIAGRAM - SHEET NO.1
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602-E-P 302	ELECTRICAL ONE LINE DIAGRAM - SHEET NO.2
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MECHANICAL

5714-GCPR-601-M-P 800	REACTOR VESSEL ASSEMBLY
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601-M-P 803	GRAPHITE CORE - PLAN AND ELEVATION
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601-M-P 301	EXTERIOR FUEL HANDLING - FLOW SHEET
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PIPING

5714-GCPR-602-P-P 301	STEAM PIPING AND INSTRUMENT DIAGRAM
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602-P-P 302	CONDENSATE PIPING AND INSTRUMENT DIAGRAM
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602-P-P 303	BOILER FEED WATER PIPING AND INSTRUMENT DIAGRAM
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602-P-P 304	CIRCULATING AND SERVICE WATER PIPING AND INSTRUMENT DIAGRAM
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602-P-P 305	SIMPLIFIED HEAT BALANCE - 250,000 KW - 1½" Hg BACK PRESSURE - GAS IN = 1000 °F - GAS OUT = 450 °F
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601-P-P 800	MAIN GAS COOLING SYSTEM FLOW DIAGRAM
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601-P-P 801	GAS PURIFICATION SYSTEM FLOW DIAGRAM
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601-P-P 802	GAS VENT SYSTEM FLOW DIAGRAM
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601-P-P 804	BURST SLUG DETECTION SYSTEM FLOW DIAGRAM
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601-P-P 805	MAIN GAS COOLANT PIPING PLAN AND ELEVATION
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MECHANICAL SPECIALTIES

5714-GCPR-601-MS-P 828	CONTROL ROD SHIELD PLUG ASSEMBLY
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601-MS-P 841	CHARGE MACHINE GENERAL ASSEMBLY
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601-MS-P 871	SERVICE MACHINE GENERAL ASSEMBLY
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GCPR DRAWING INDEX

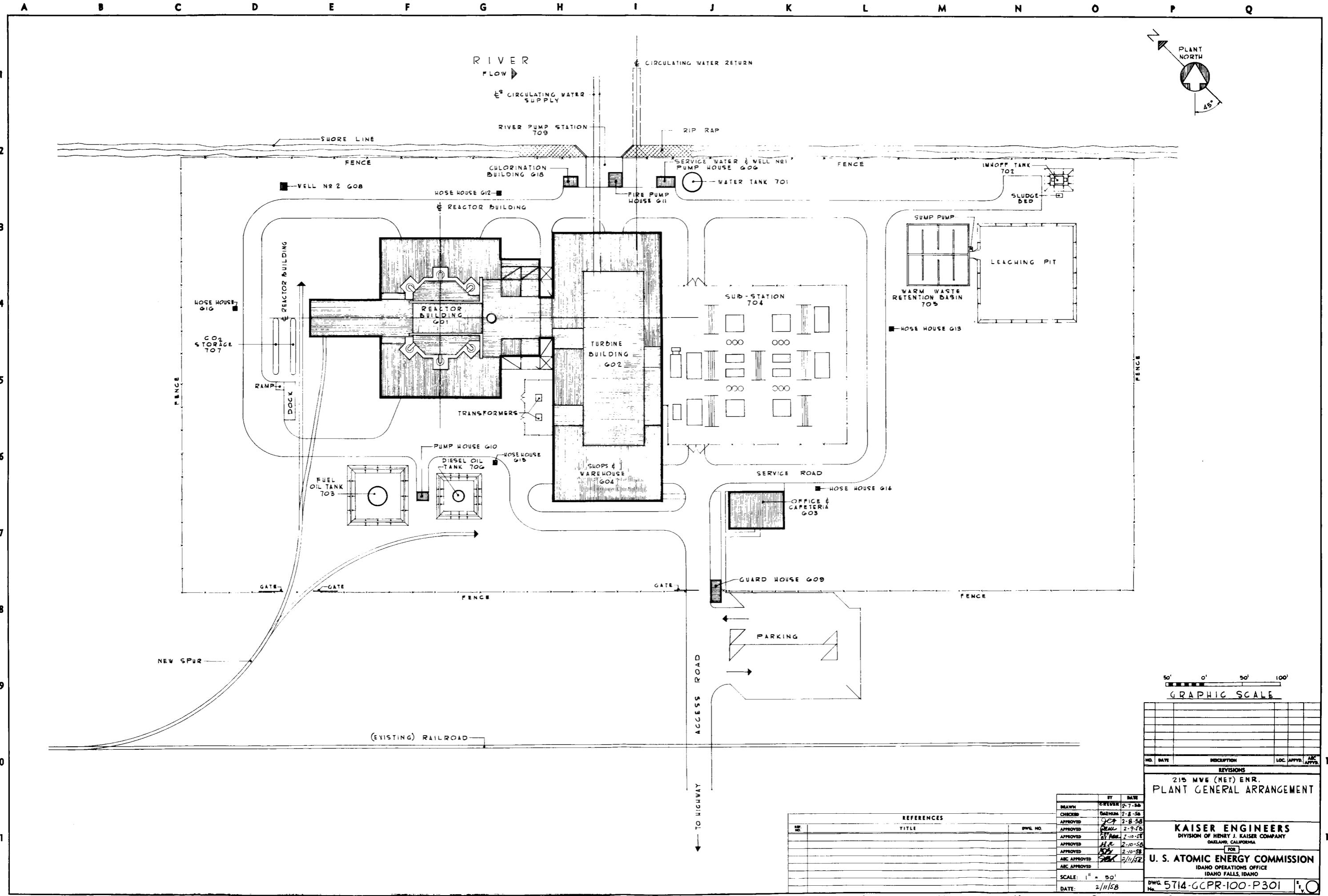
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DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA

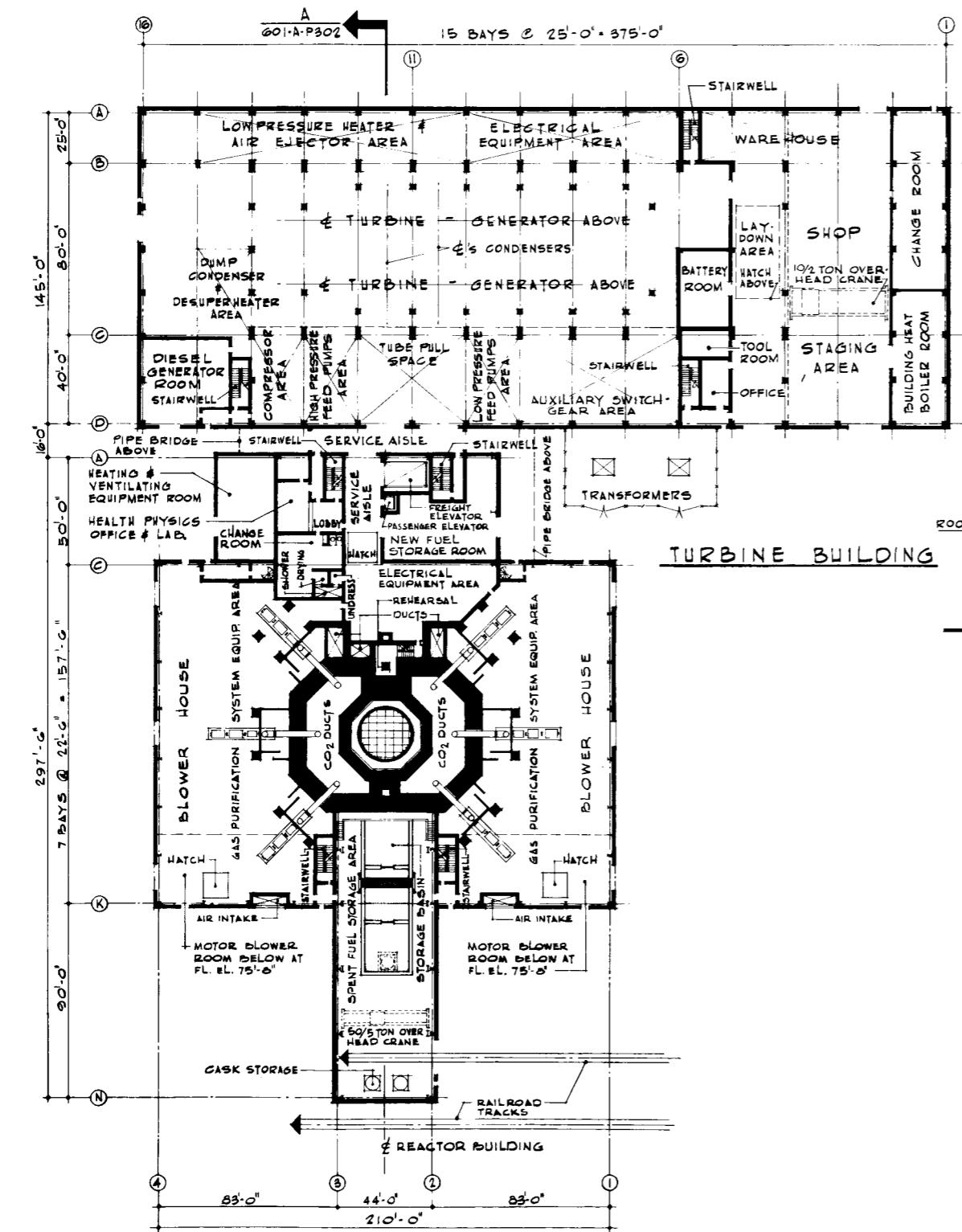
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U. S. ATOMIC ENERGY COMMISSION
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IDAHO FALLS, IDAHO

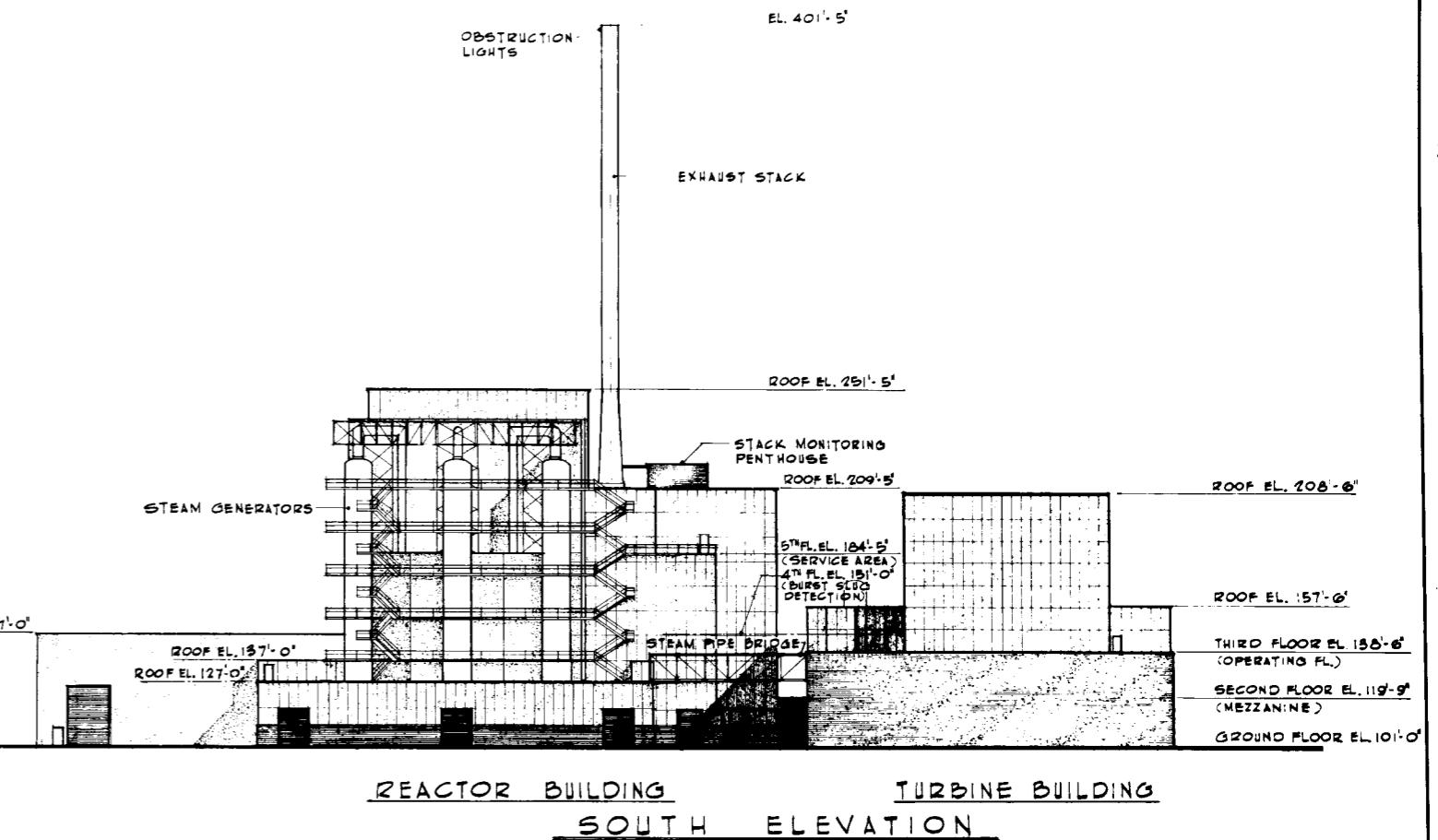
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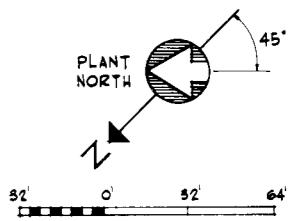




A 601-A-P302 REACTOR BUILDING
GROUND FLOOR PLAN

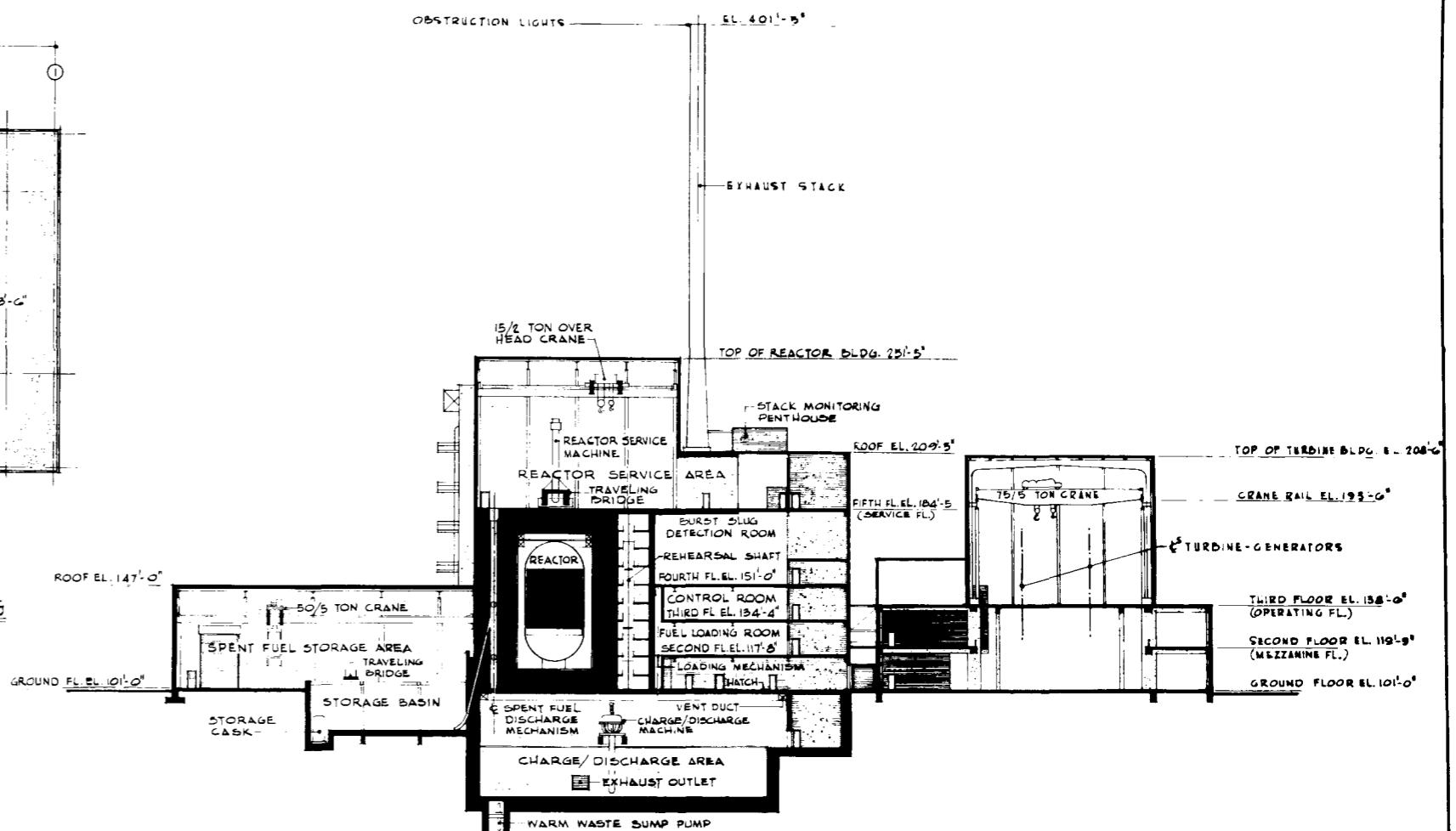
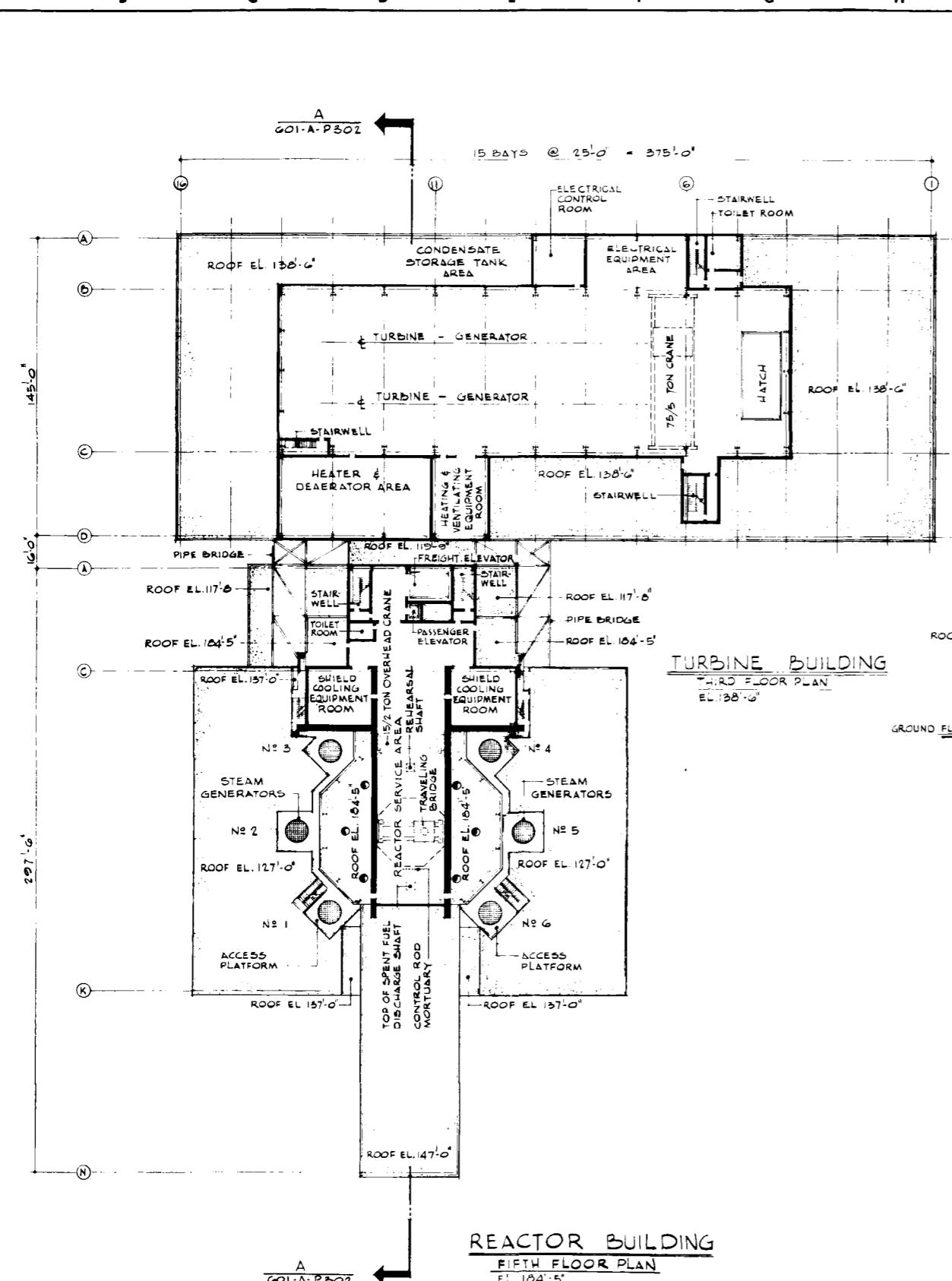


REACTOR BUILDING TURBINE BUILDING
SOUTH ELEVATION



GRAPHIC SCALE

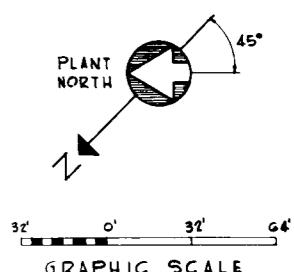
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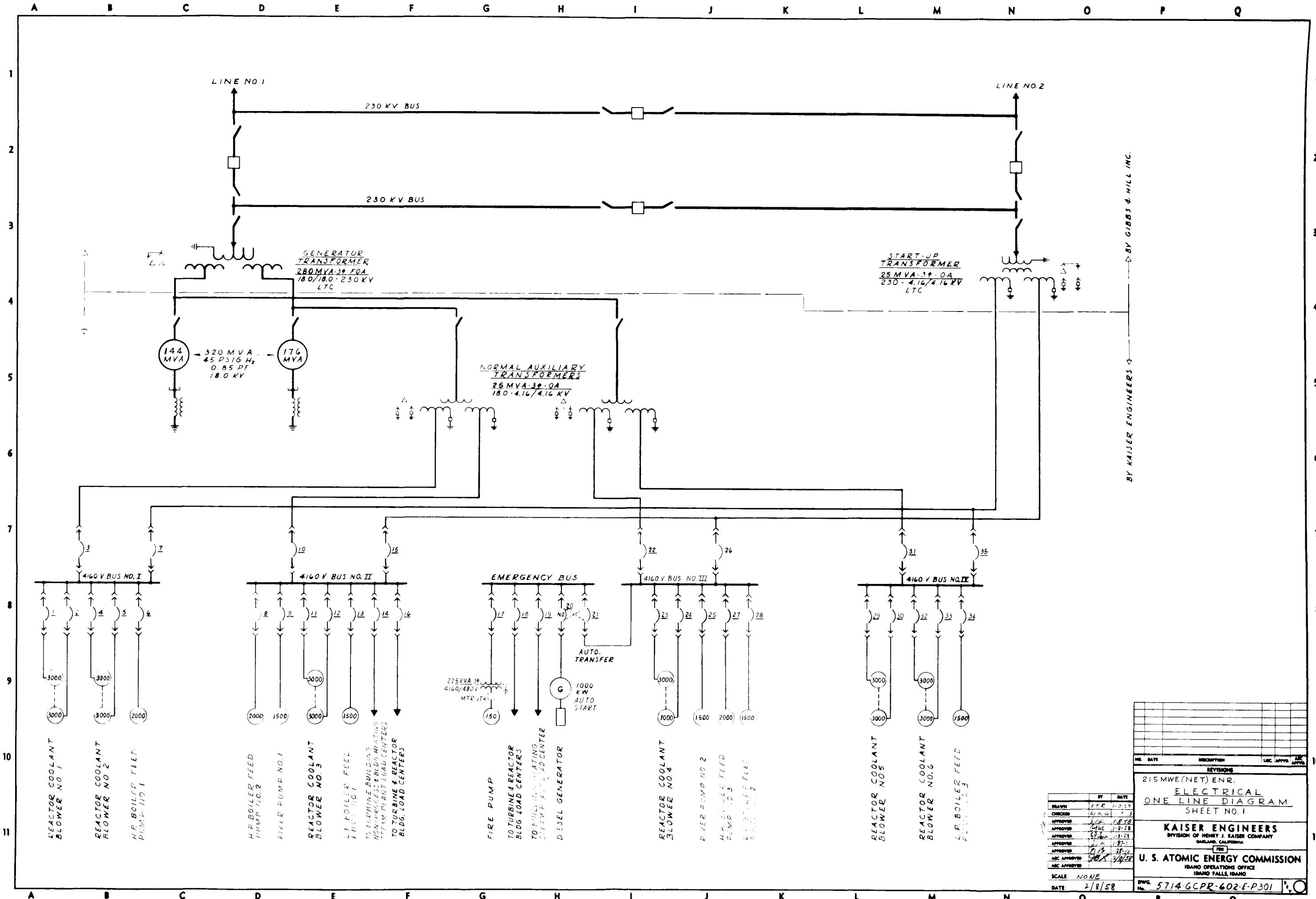
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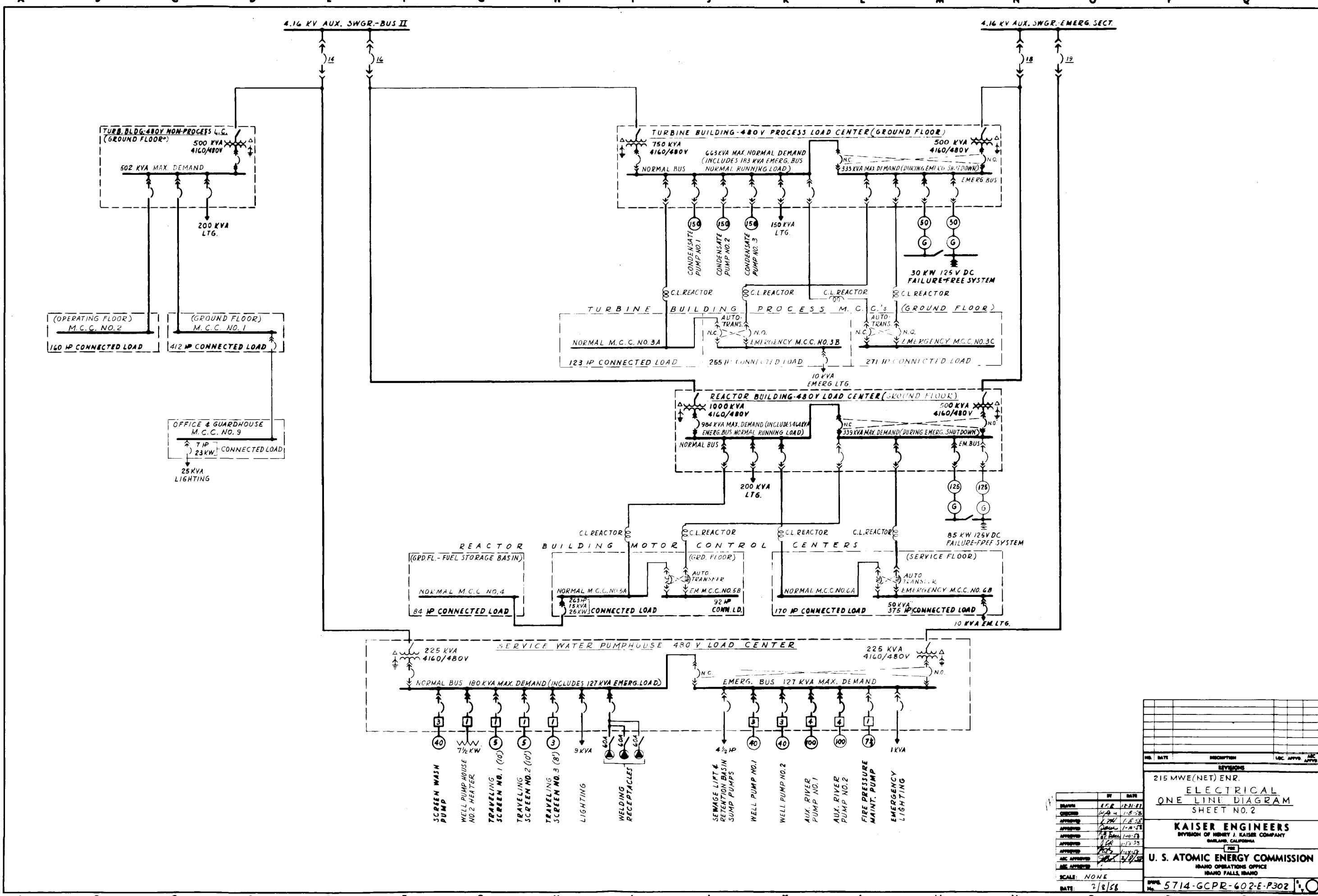
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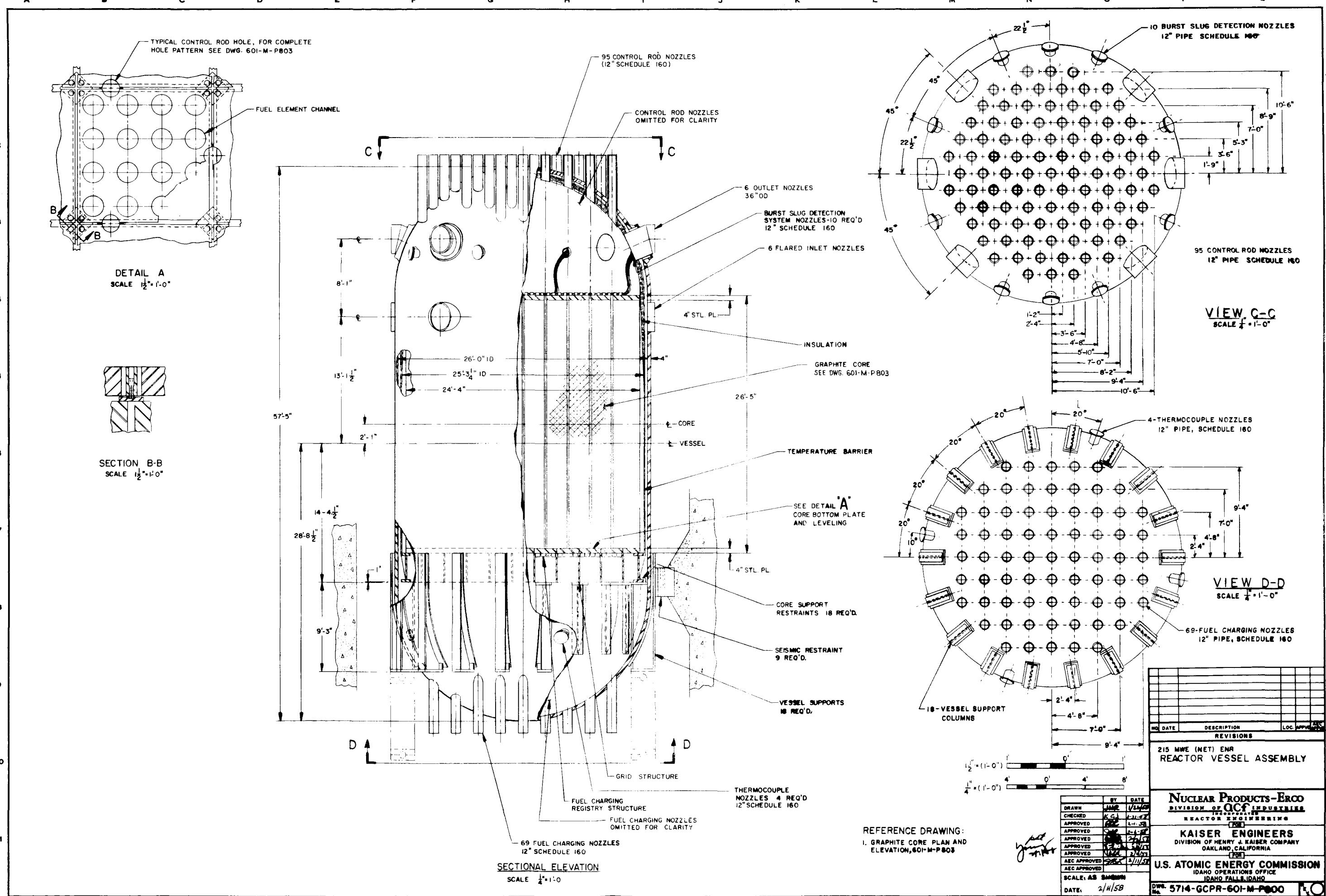
SECTION A-A

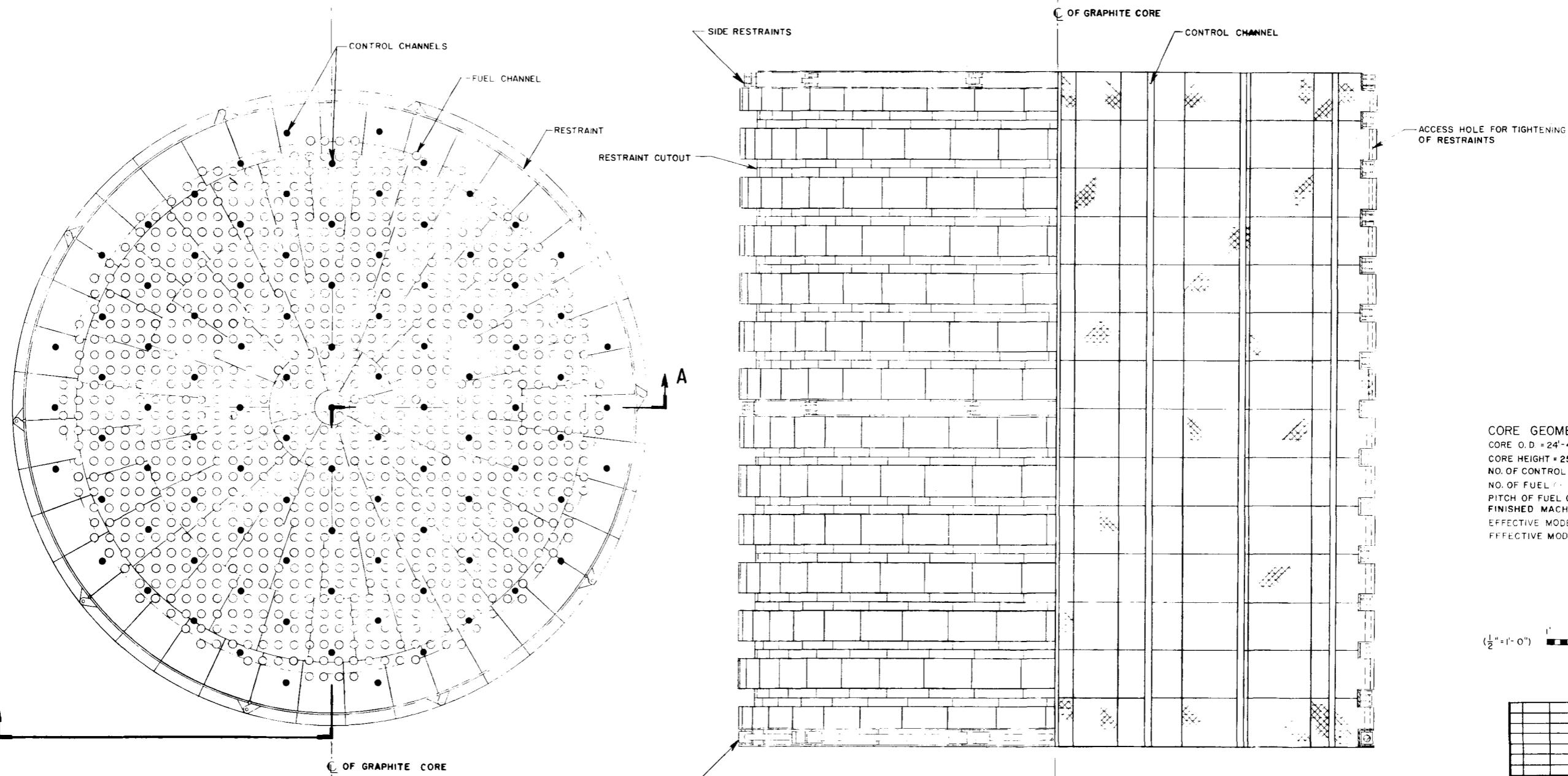


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DWG. 5714-GCPR-601-A-P302					
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PLAN VIEW OF GRAPHITE COR

SECTIONAL VIEW "A-A
SCALE - $\frac{1}{2}$ " = 1'-0"

CORE GEOMETRY
 CORE O. D. = 24'-4"
 CORE HEIGHT = 25'-9"
 NO. OF CONTROL ROD CHANNELS = 95 (3¹/₄" I.D.)
 NO. OF FUEL CHANNELS = 980 (4" I.D.)
 PITCH OF FUEL CHANNELS = 7"
 FINISHED MACHINED WT OF STACK = 1,021,400 LBS.
 EFFECTIVE MODERATOR DIA = 20'-7¹/₄"
 EFFECTIVE MODERATOR HEIGHT = 21'-9"

REVISIONS
MWE (NET) ENR
GRAPHITE CORE
PLAN AND ELEVATION

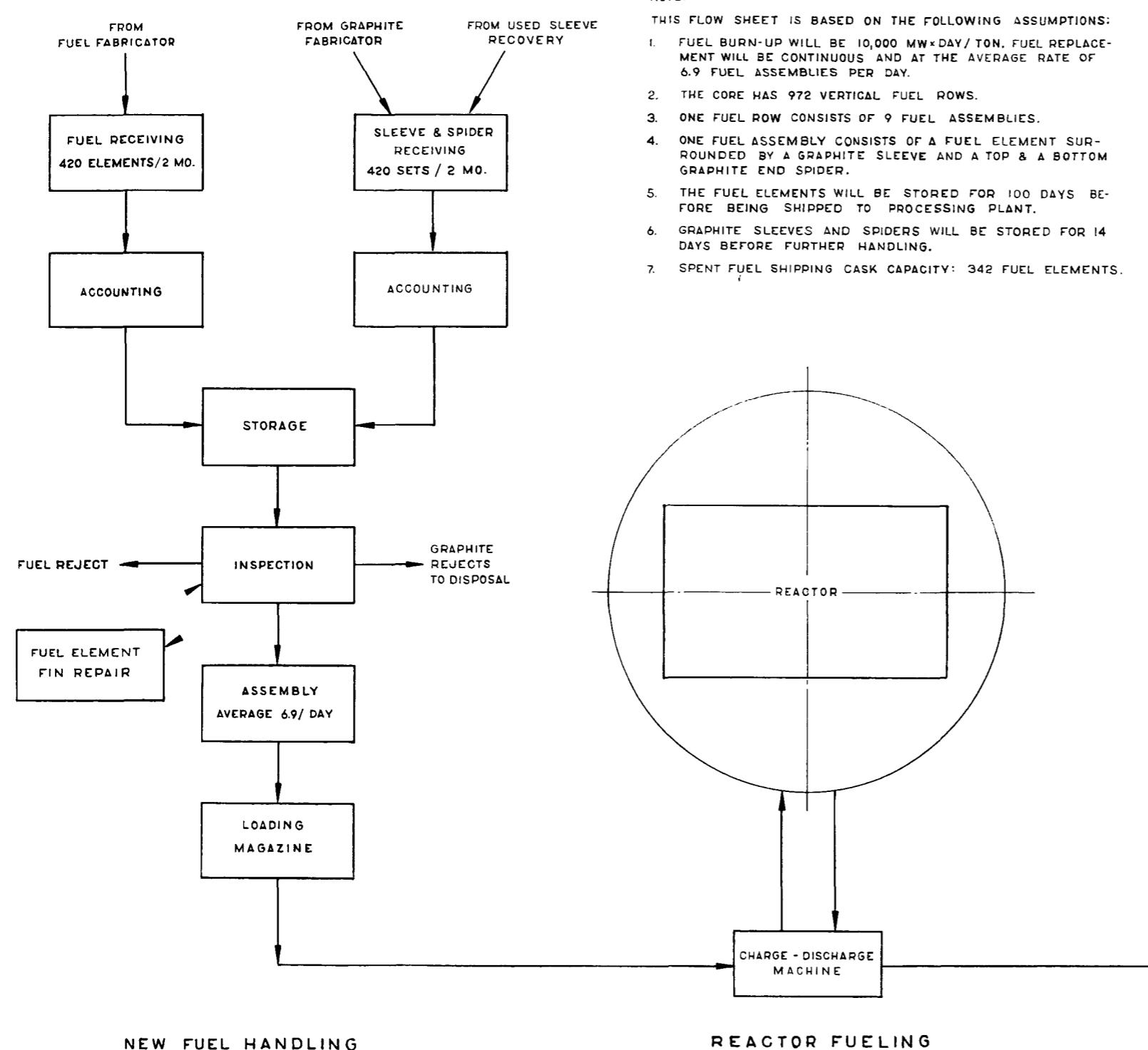
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DIVISION OF **QCF INDUSTRIES**
INCORPORATED
REACTOR ENGINEERING

KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

	BY
RAWN	V. MOORE
CHECKED	R.C.J.
APPROVED	R.C.J.
EC APPROVED	R.C.J.
EC APPROVED	R.C.J.

CALE: $\frac{1}{2}'' = 1' - 0''$

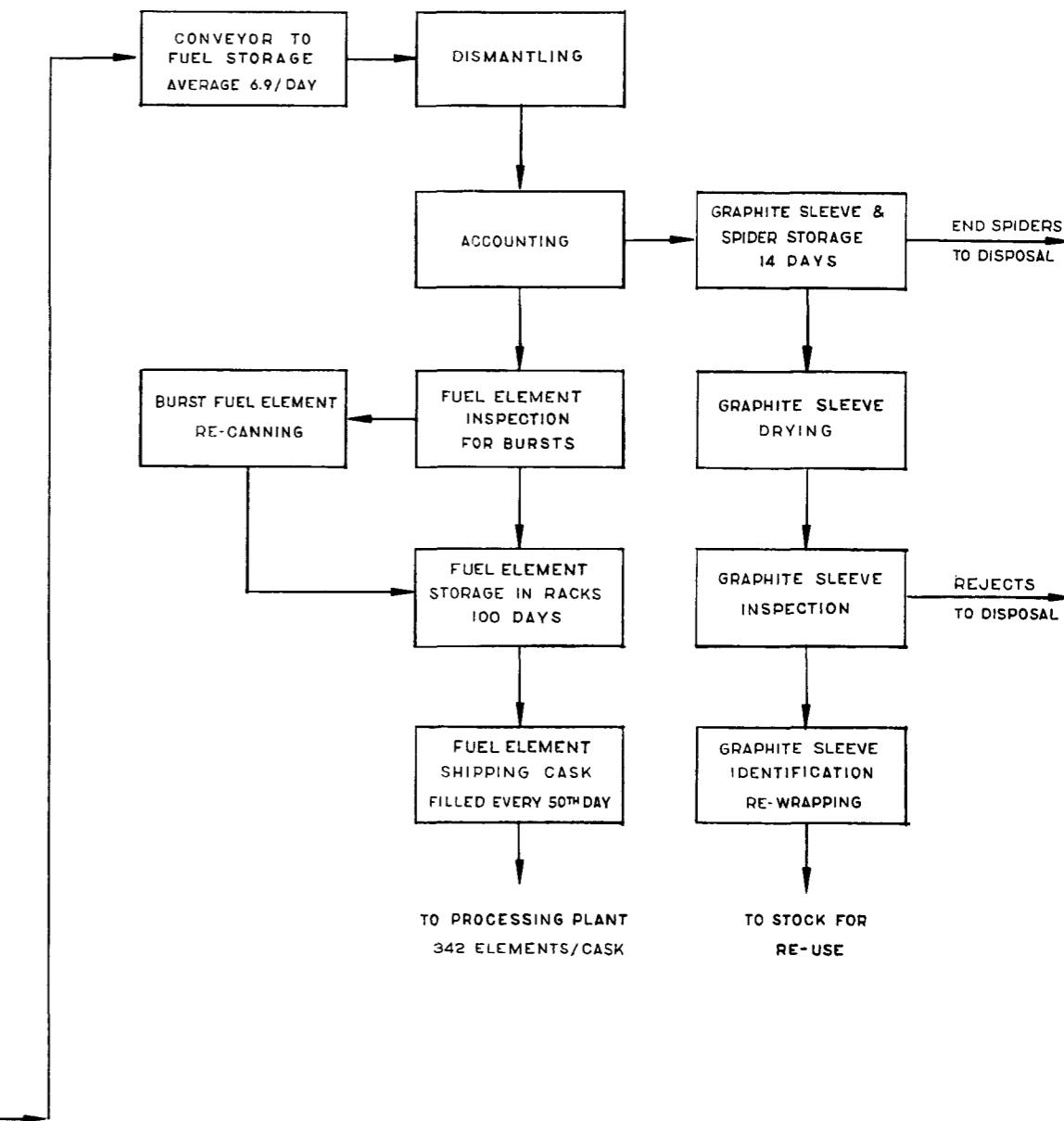
ATE: 2/10/58



11

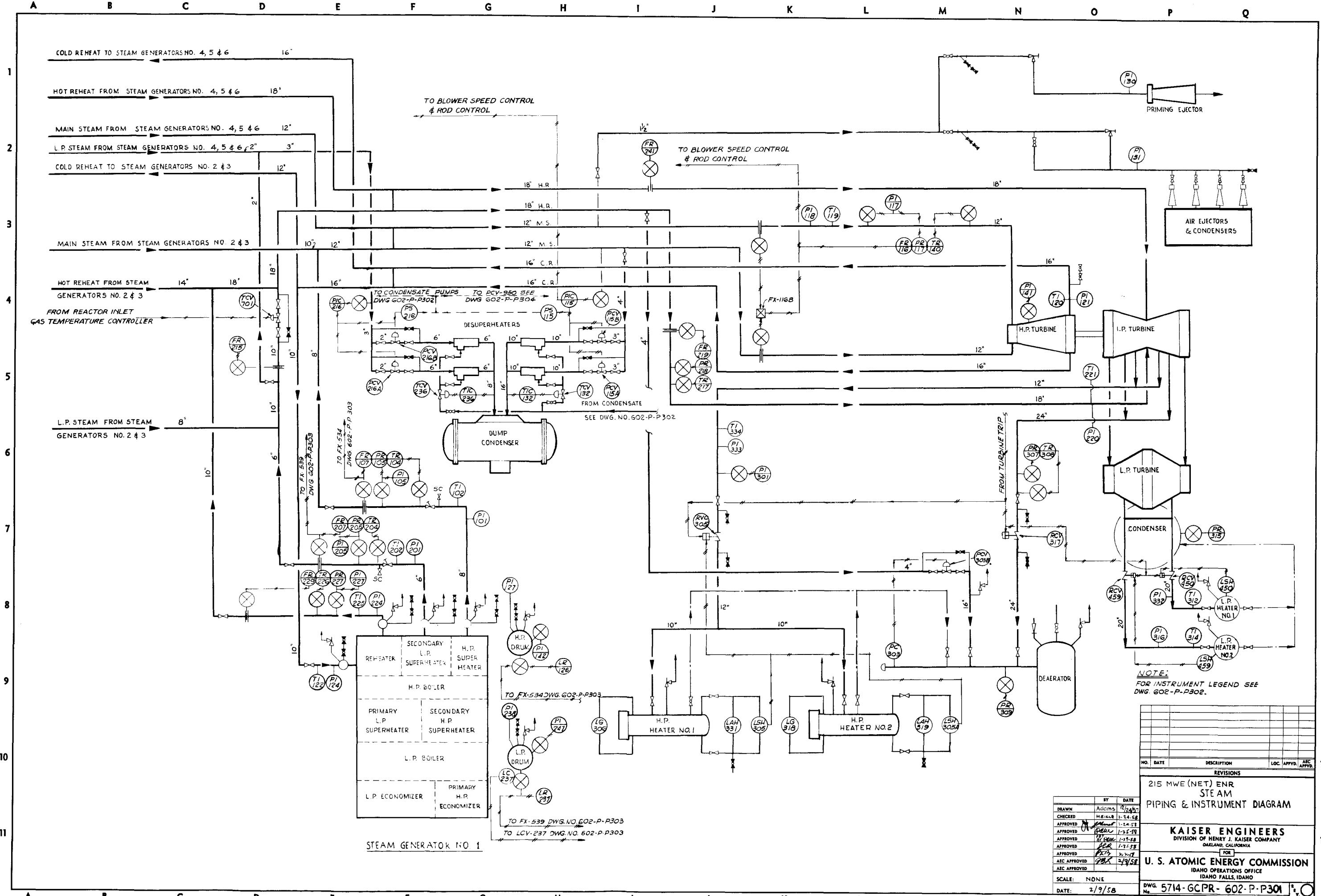
THIS FLOW SHEET IS BASED ON THE FOLLOWING ASSUMPTION

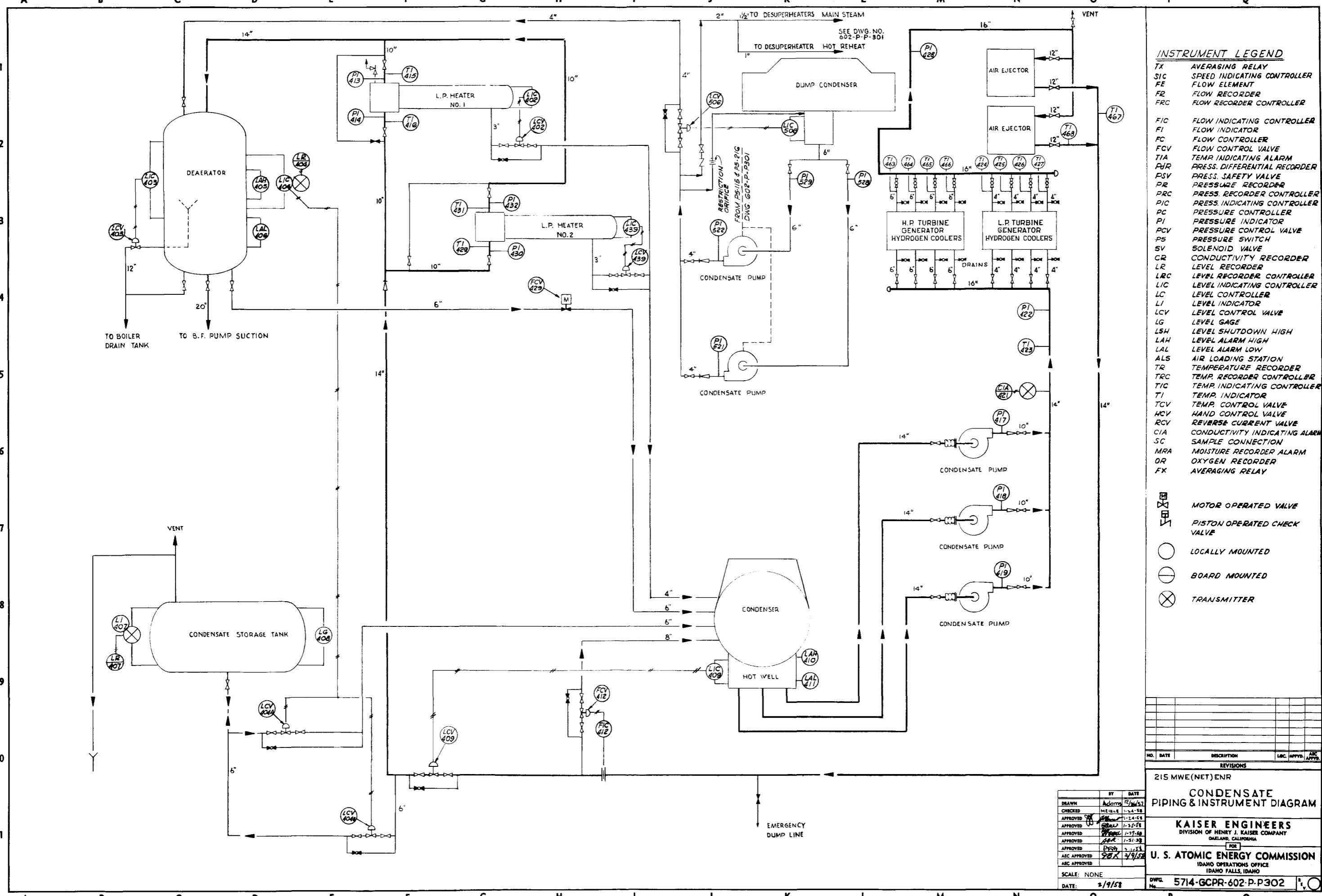
1. FUEL BURN-UP WILL BE 10,000 MW \times DAY/TON. FUEL REPLACEMENT WILL BE CONTINUOUS AND AT THE AVERAGE RATE OF 6.9 FUEL ASSEMBLIES PER DAY.
 2. THE CORE HAS 972 VERTICAL FUEL ROWS.
 3. ONE FUEL ROW CONSISTS OF 9 FUEL ASSEMBLIES.
 4. ONE FUEL ASSEMBLY CONSISTS OF A FUEL ELEMENT SURROUNDED BY A GRAPHITE SLEEVE AND A TOP & A BOTTOM GRAPHITE END SPIDER.
 5. THE FUEL ELEMENTS WILL BE STORED FOR 100 DAYS BEFORE BEING SHIPPED TO PROCESSING PLANT.
 6. GRAPHITE SLEEVES AND SPIDERS WILL BE STORED FOR DAYS BEFORE FURTHER HANDLING.
 7. SPENT FUEL SHIPPING CASK CAPACITY: 342 FUEL ELEMENTS.

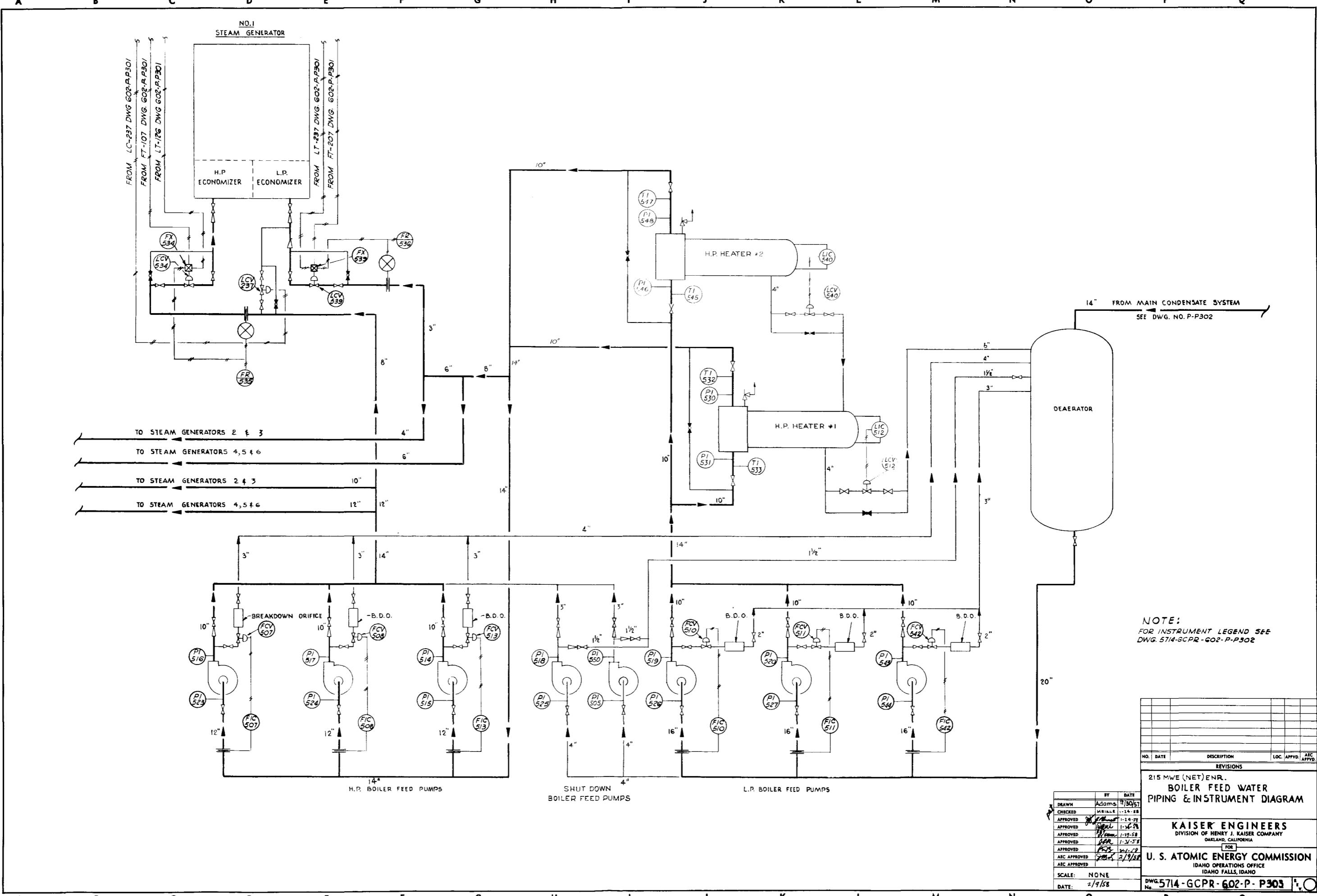


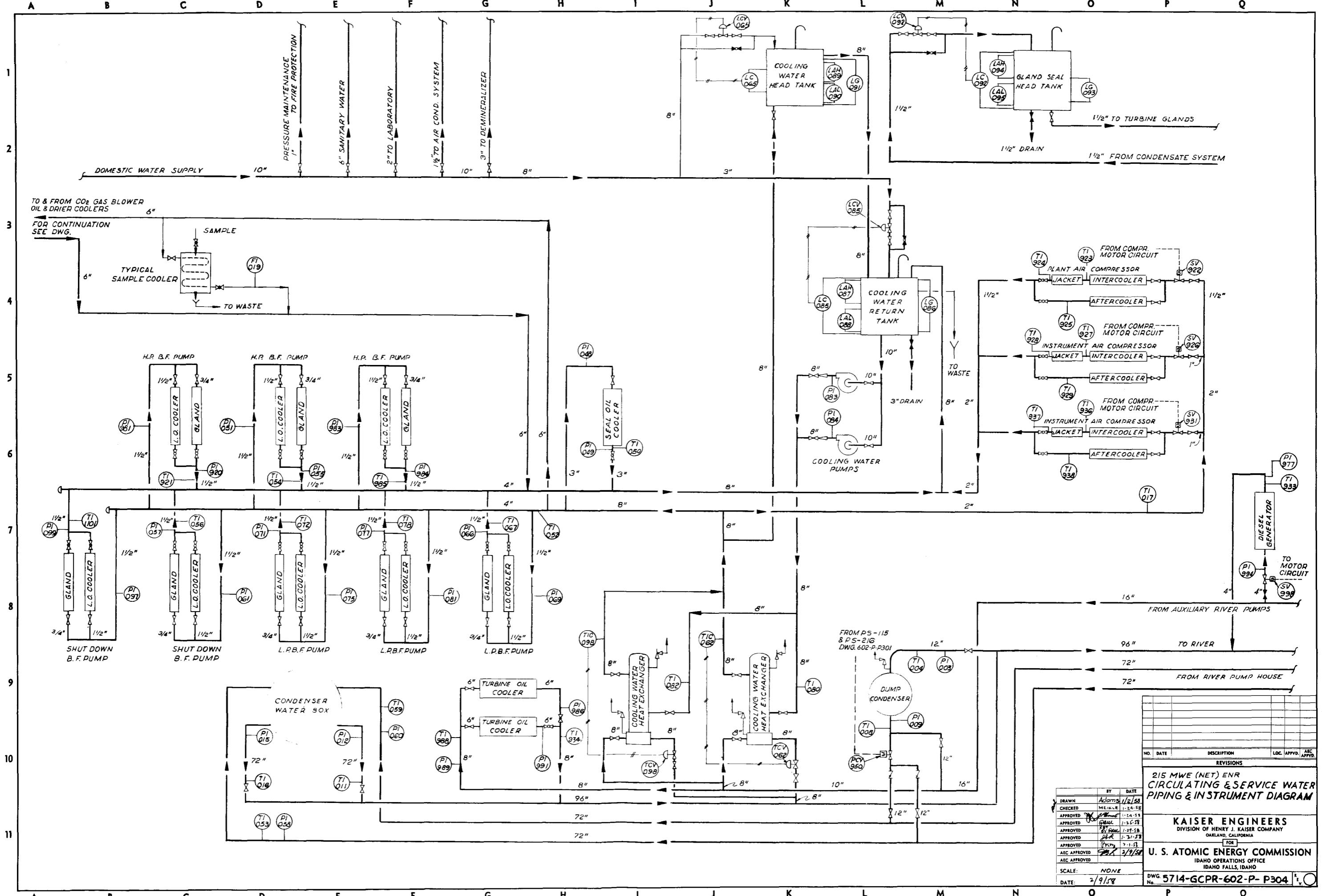
SPENT FUEL HANDLING

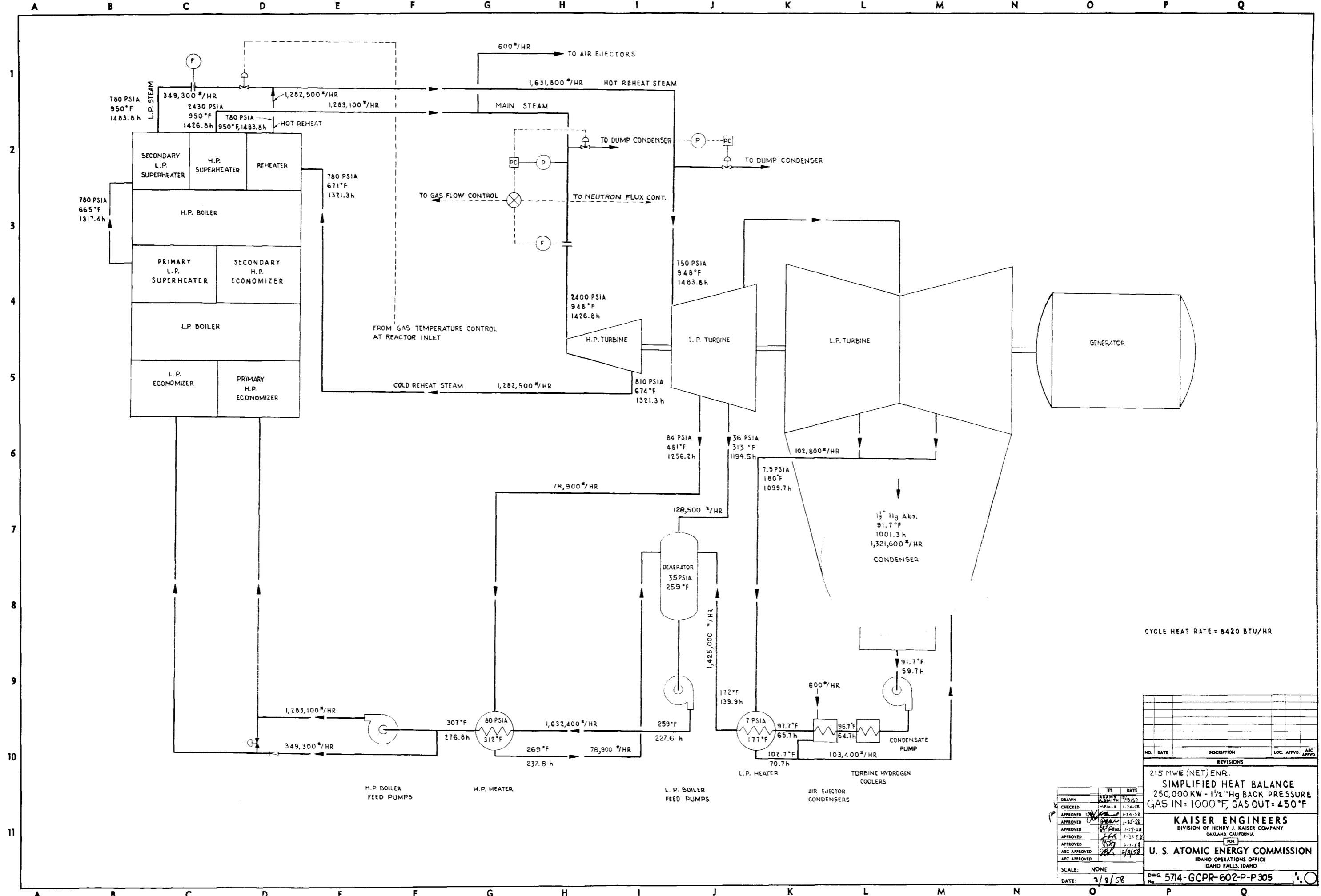
NO.	DATE	DESCRIPTION	LOC.	APPVD.	ABC APPVD.
REVISIONS					
215 MWE (NET) ENR.					
EXTERIOR FUEL HANDLING FLOW SHEET					
KAISER ENGINEERS DIVISION OF HENRY J. KAISER COMPANY OAKLAND, CALIFORNIA					
FOR					
U. S. ATOMIC ENERGY COMMISSION IDAHO OPERATIONS OFFICE IDAHO FALLS, IDAHO					
DWG. 5714-GCPR-601-M-P301					
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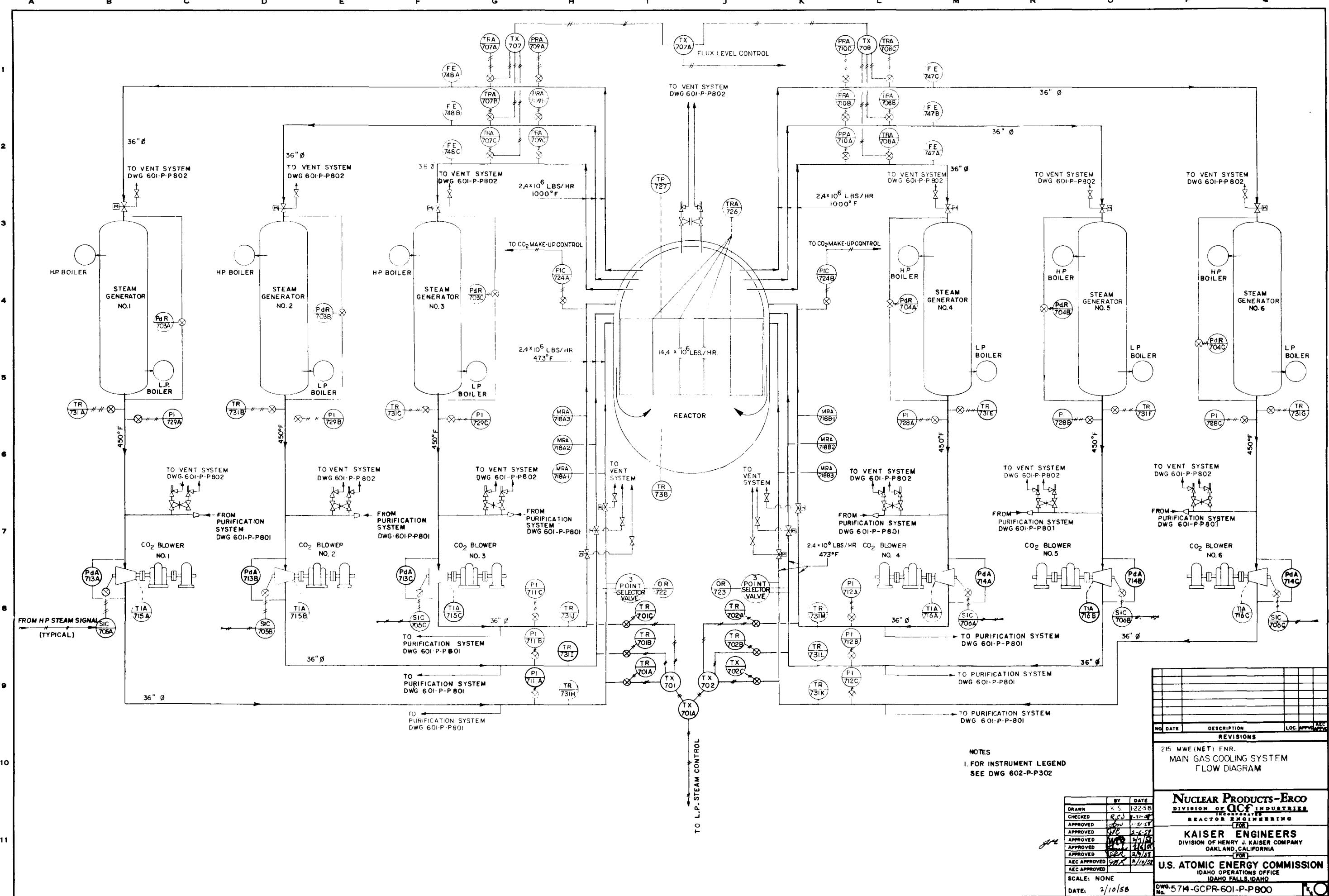


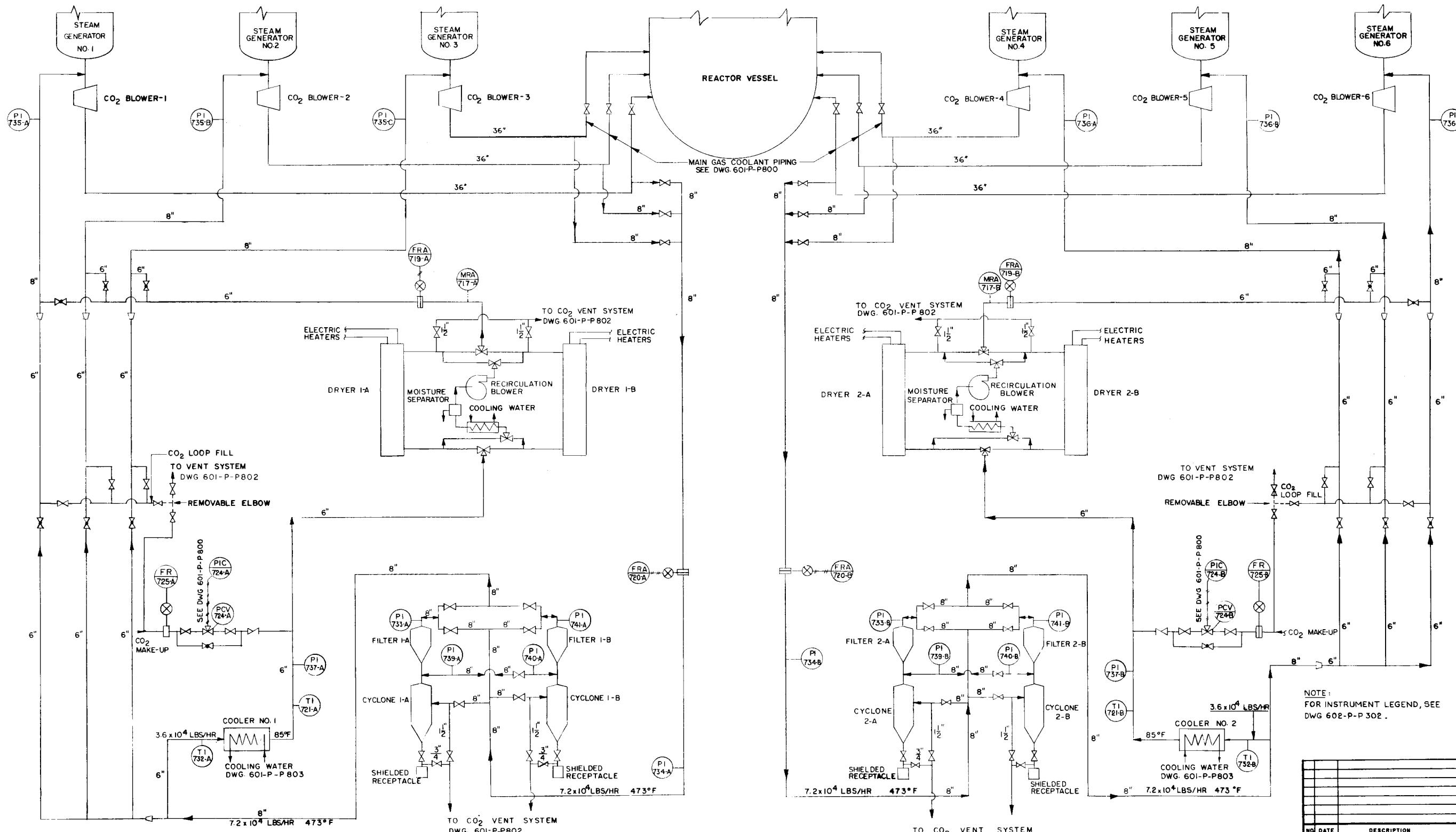












NOTE:
FOR INSTRUMENT LEGEND, SEE
DWG 608-P-P 302

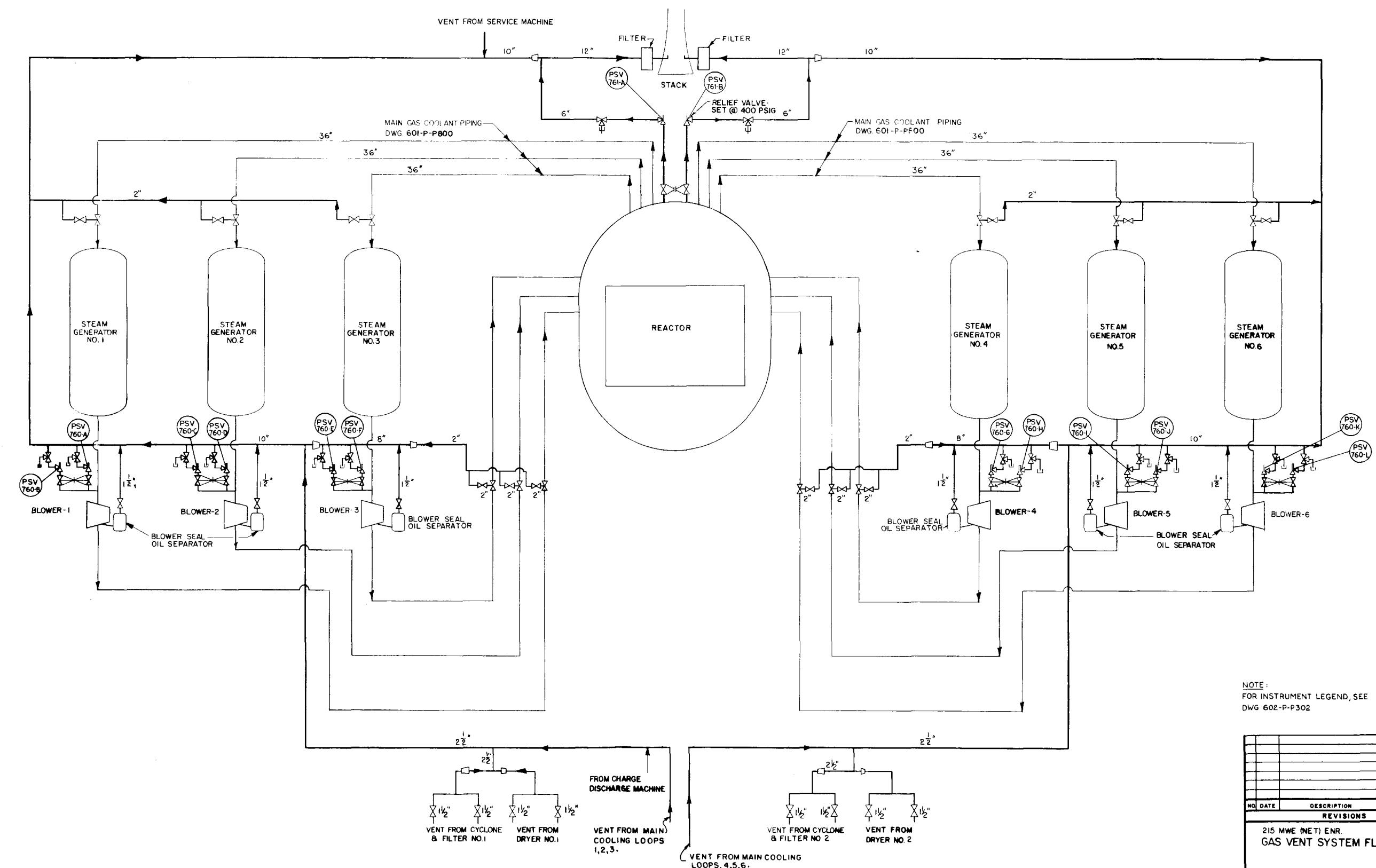
MWE (NET) ENR.
GAS PURIFICATION SYSTEM
FLOW DIAGRAM

NUCLEAR PRODUCTS-ERCO
DIVISION OF **QCF** INDUSTRIES
INCORPORATED
REACTOR ENGINEERING

FOR
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA

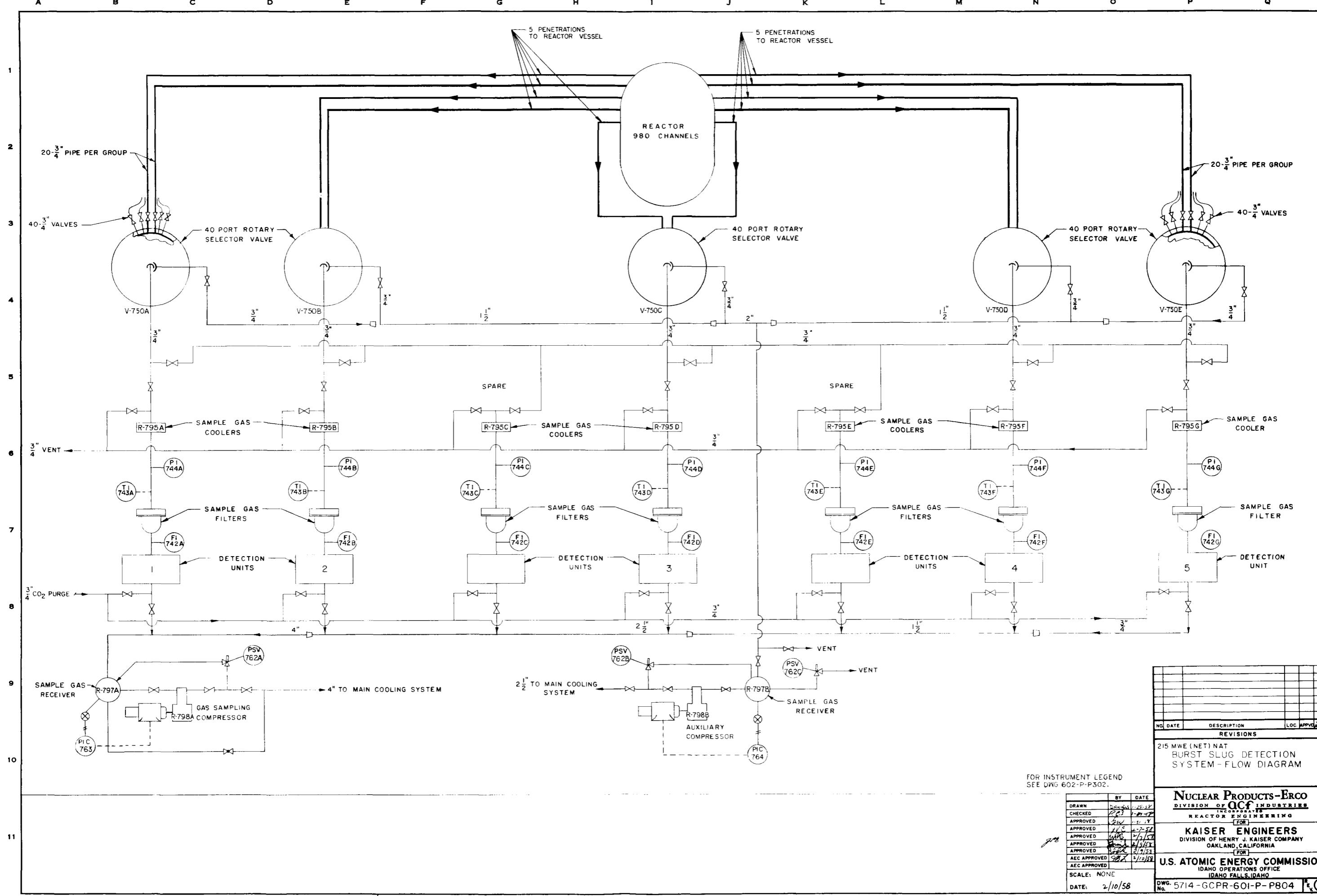
FOR
ATOMIC ENERGY COMMISSION
IDaho OPERATIONS OFFICE
IDaho FALLS, IDAHO

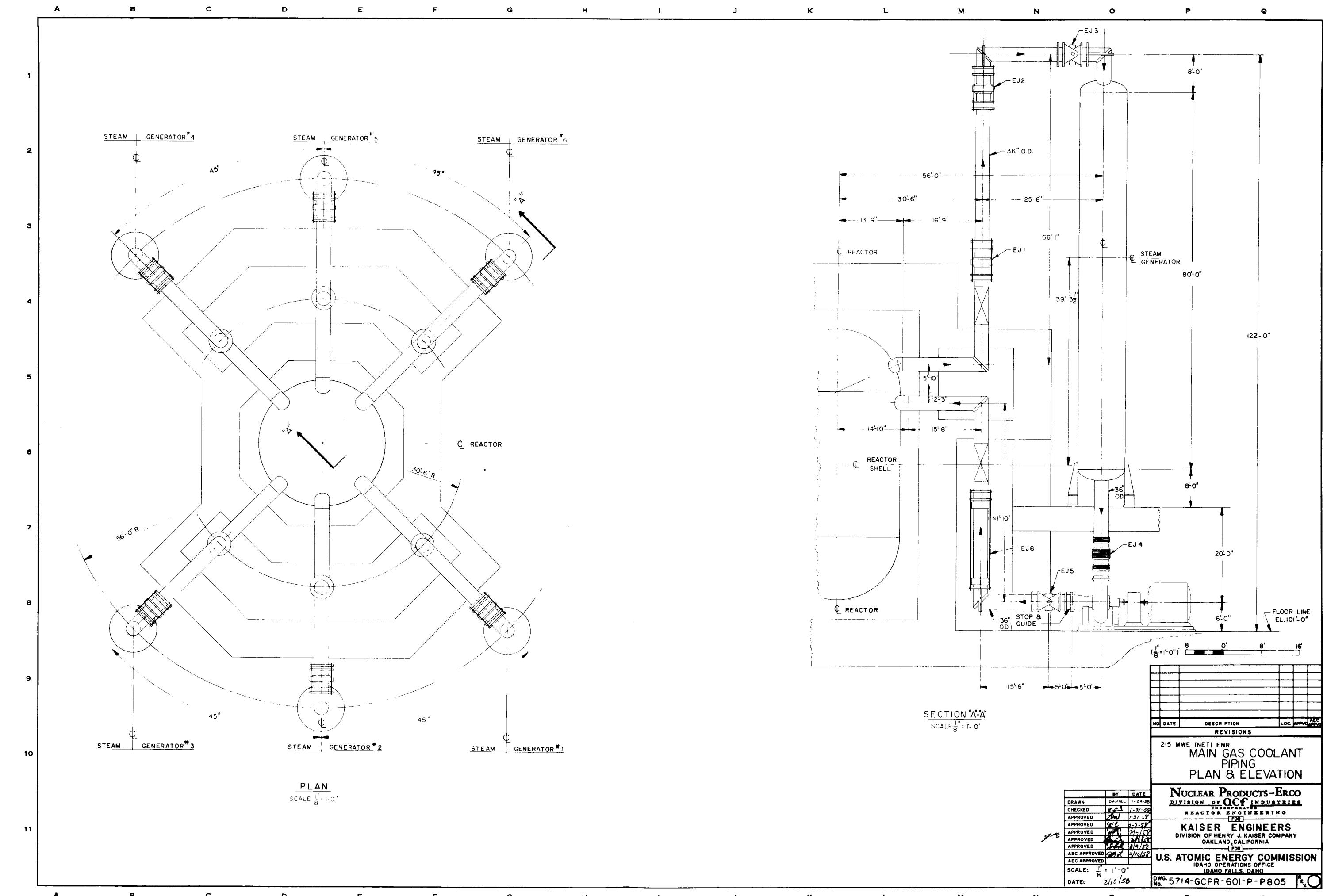
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RAWN	KS	1-2-58
HECKED	KL	1-3-58
APPROVED	JW	1-3-58
APPROVED	JPC	2-6-58
APPROVED	WLP	2-11-58
APPROVED	WLP	2-11-58
APPROVED	WLP	2-11-58
REC APPROVED	200	2/10/58
REC APPROVED		
CALE: NONE		
DATE:	2/10/58	
		DW NO.

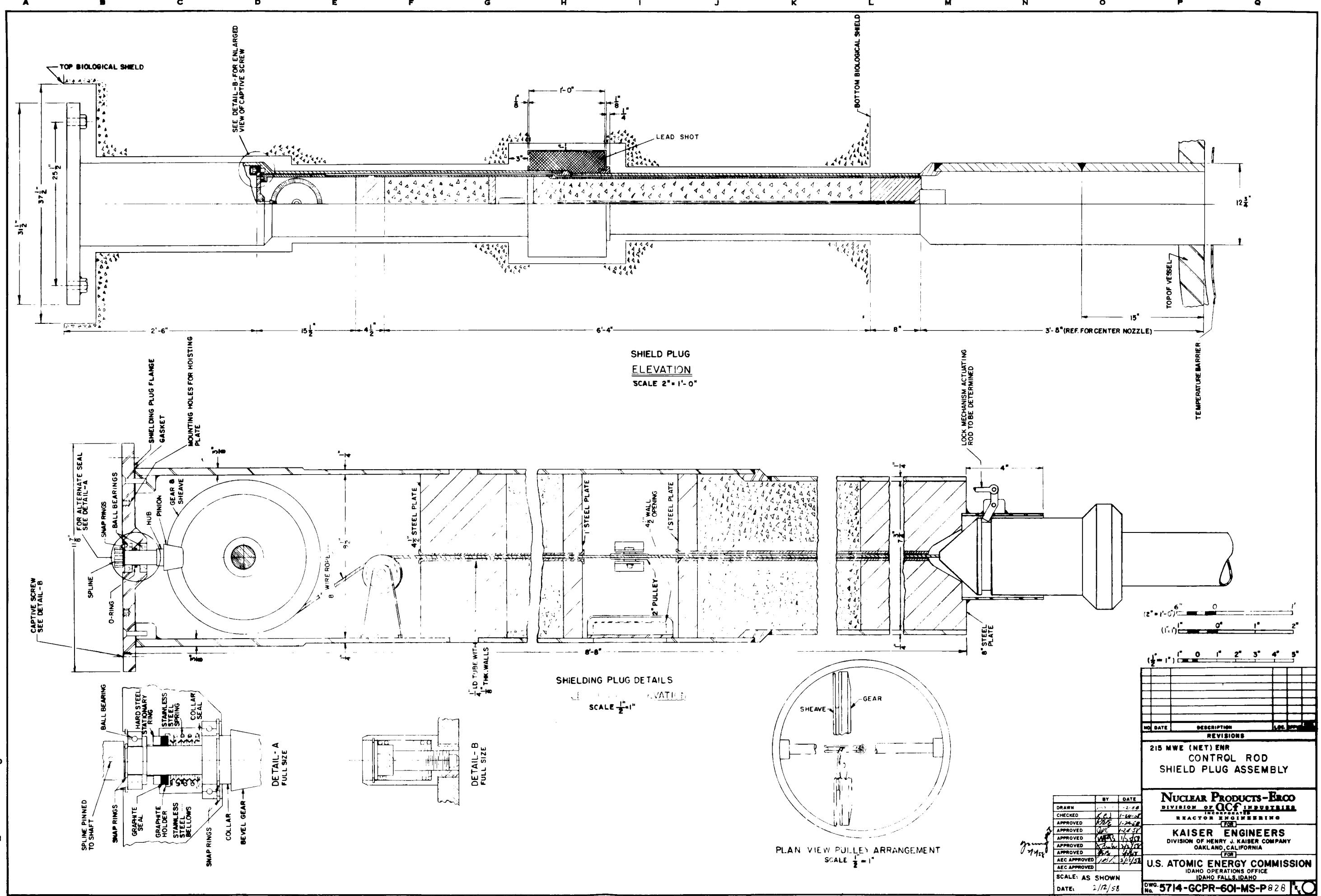


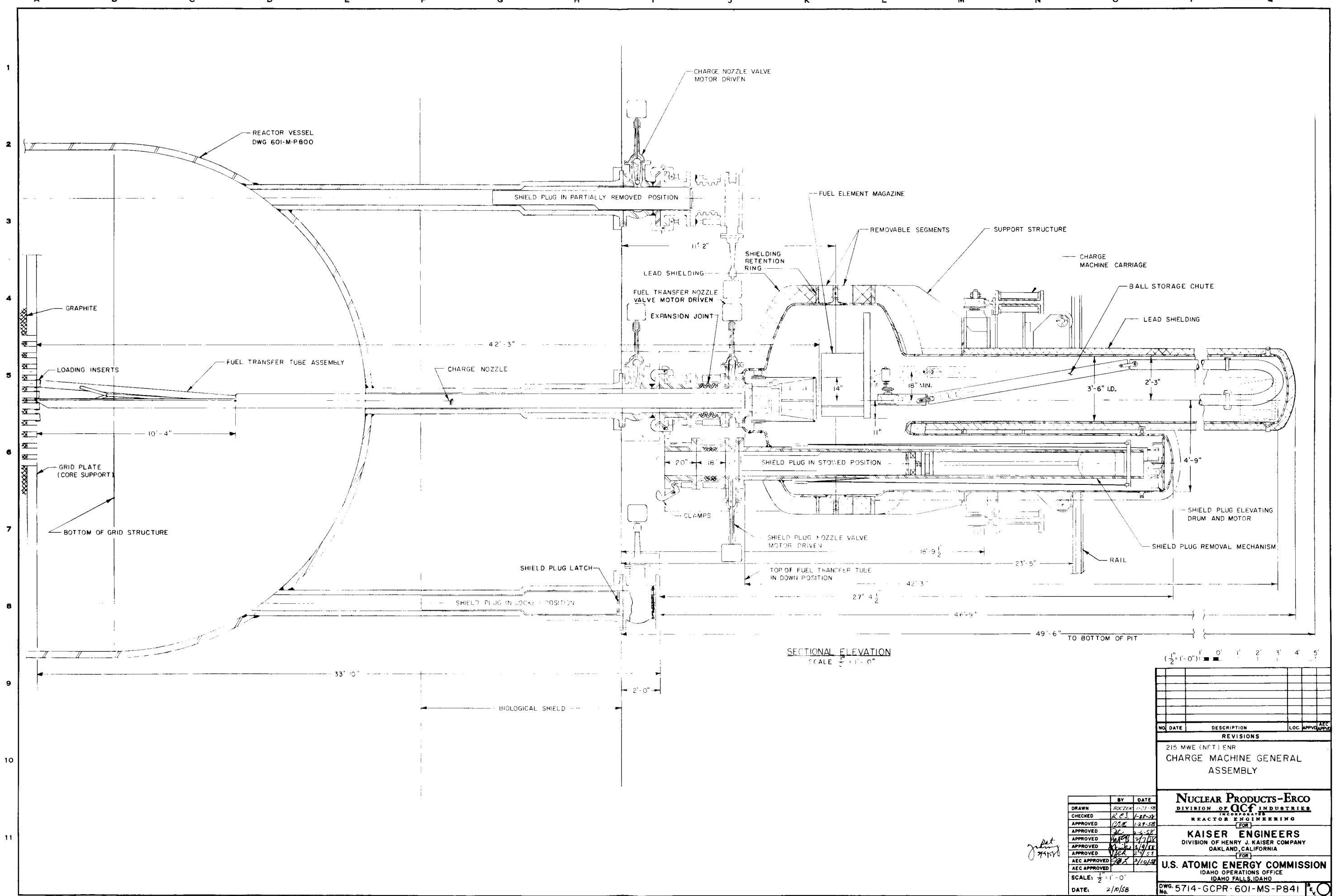
BY	DATE
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CHECKED	1-31-57
APPROVED	2-5-57
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APPROVED	2-11-57
AEC APPROVED	2/10/57
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SCALE: NONE	
DATE:	2/10/57
DWG. No.	5714-GCPR-601-P-P802

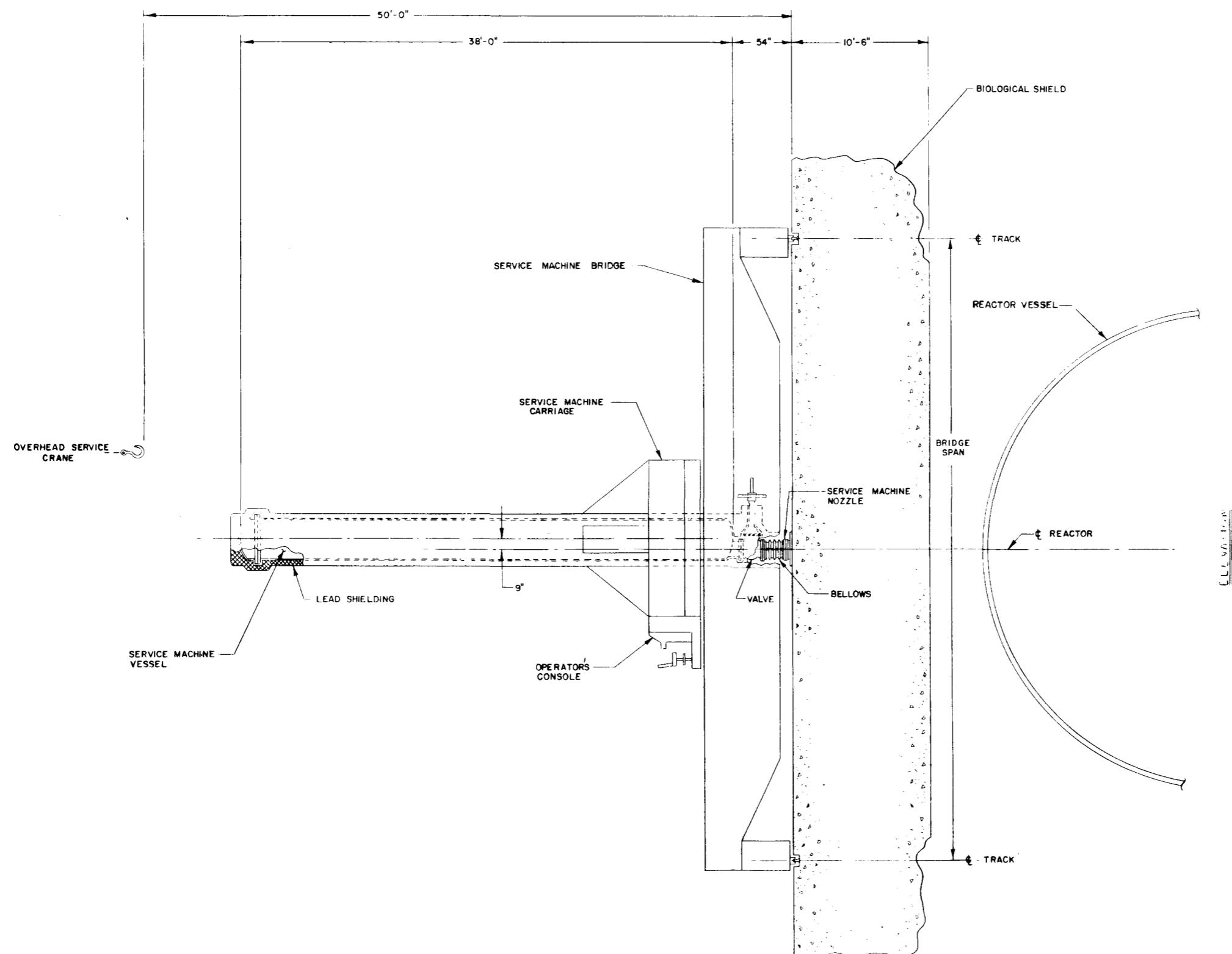
NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES
INCORPORATED
REACTOR ENGINEERING
FOR
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO











215 MWE (NET) ENR

REVISIONS

215 MWE (NET) ENR

GENERAL ASSEMBLY

NUCLEAR PRODUCTS-ERCO
DIVISION OF QCC INDUSTRIES
REACTOR ENGINEERING

ERCO

KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA

U.S. ATOMIC ENERGY COMMISSION
IDaho OPERATIONS OFFICE
IDAHO FALLS, IDAHO

SCALE: $\frac{1}{4}$ " = 1'-0"

DATE: 2/12/58

DWG. No. 5714-GCPR-601-MS-P871

BY	DATE
DRAWN	CARLE 12-20-57
CHECKED	RECE 1-20-58
APPROVED	1/25/58
AEC APPROVED	1/25/58
AEC APPROVED	1/25/58
SCALE: $\frac{1}{4}$ " = 1'-0"	
DATE:	2/12/58
DWG. No.	5714-GCPR-601-MS-P871