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REVISION 1
APRIL 1, 1958

GAS COOLED POWER REACTOR

FEASIBILITY STUDY

44,000 KW PROTOTYPE

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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44,000 KW PROTOTYPE

PARTIALLY ENRICHED URANIUM

GAS COOLED, GRAPHITE MODERATED

NUCLEAR POWER PLANT

(PROTOTYPE FOR AN OPTIMUM POWER PLANT)

FOR

UNITED STATES ATOMIC ENERGY COMMISSION

IDAHO OPERATIONS OFFICE

CONTRACT NO. AT(10-1)-925

Report No. 58-3-RE
March, 1958

KAISER ENGINEERS
Division of Henry J. Kaiser Company
Oakland, California

and

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

TABLE OF CONTENTS

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 prototype, partially enriched uranium plant listed in Item c above. The purpose of building a prototype would be to simulate insofar as possible the design conditions of the optimum plant on a reduced scale.

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 tentative site was selected at the National Reactor Testing Station to form a basis for estimating construction costs. The design is based on the interconnection of the prototype plant with the present power distribution system at NRTS.

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.

The plant is designed as a 44 mwe (net) prototype of an optimum plant rated at 215 mwe (net). The plant consists primarily of four adjoining buildings which house the reactor plant, turbine generator, warehouses, and offices. Other facilities include the induced draft cooling tower, a substation and necessary utilities.

A general plant arrangement is shown on Dwg. No. 100-P201. The estimated design and construction cost of the 44 mwe prototype (51 mwe gross) at the NRTS is $49,000,000 or $960 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 support design to achieve the maximum capability of this plant. The plant can be built on a schedule which would complete construction within 42 months from the time preliminary design is initiated. This assumes that construction is initiated three months after the start of detailed design. The estimated cost of power produced by this plant is 32 mills per kwhr of which fixed charges are 25 mills per kwhr, based on a lifetime plant factor of 80%, annual fixed charges at 14%, and accepted utility plant accounting practices. The cost of $960 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.

For comparison, the estimated cost of the optimum power plant is $104,000,000 or $410 per kw (gross electrical). The estimated cost of power is 15 mills per kwhr of which fixed charges are 12 mills per kwhr.

Reactor Plant

The major design parameters of the 44 mwe prototype plant follow closely those of a 215 mwe (net) power plant optimized for minimum power cost. The optimum reactor is housed in a double-walled vessel consisting of a 26' diameter carbon steel cylinder with hemispherical heads and overall height of 57'. The vessel outer wall is 4" thick. The optimum plant 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 degree of enrichment is selected to provide an average fuel element lifetime of 10,000 megawatt days per metric ton (mwd/M.T.): for the optimum plant it is 2.5%; for the prototype plant it is 3.0%.

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

The prototype reactor is designed to operate at the same temperature, pressure, specific power and fuel life as the optimum plant. The prototype reactor double-walled pressure vessel is an 18' ID carbon steel cylinder with hemispherical heads and overall height of 40'; the vessel outer wall is 3" thick. The inner wall temperature barrier is stainless steel 3/4" thick, and ensures the maintenance of the outer carbon steel wall at a maximum temperature of 650 F. The cylindrical graphite core is approximately 16' in diameter by 18' high. Vertical channels (4" diameter with a lattice spacing of 7") contain a stack of six 29" long fuel assemblies. Cooling gas passes up through the core. Control rods enter from above; 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 cylindrical pressure vessel containing the prototype core is the most economical geometry for this reactor. The vessel requires fabrication, erection, and stress-relieving techniques similar to those which would be required for the vessel of the optimum plant.

The graphite structure which comprises both the moderator and reflector for the reactor is supported on a grid structure within the vessel. The graphite stack is composed of rectangular blocks keyed together in a manner which 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 prototype reactor is located within a concrete biological shield approximately 10' thick. The horizontal centerline of the reactor is approximately 10' above grade. The operating level of the fuel element charge-discharge area is beneath the reactor vessel, approximately 53' below grade. The CO_2 coolant gas enters the reactor vessel at 463 F and leaves at 1,000 F; it is pumped through two steam generators, each with its own coolant loop, located on opposite sides of the reactor pressure vessel. Blowers circulate the CO_2 back to the reactor vessel in a closed system. The steam generators provide steam at dual pressures, with reheat, to the 50 mwe (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 shut-down.

Prototype Steam Power Plant

The steam power plant consists of two steam generators which supply steam at dual pressures to a single turbine generator. The turbine is nominally rated at 50,000 kw, and is directly connected to a hydrogen cooled generator rated at 64,000 kva. The equipment and control systems for the steam plant and electrical 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 two 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 generator and mixed with the low-pressure superheated steam; then the combined steam flow at approximately 750 psia and 950 F re-enters the turbine and passes to the condenser.

The condenser is at ground floor level in the turbine building and the turbine generator is on the operating floor approximately 27' above the 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.

An induced draft cooling tower cools the condenser circulating water. The condenser and cooling tower combination maintain a back pressure of 1-3/4" Hg abs. The estimated turbine heat rate is 8,500 Btu/kwhr.

It was necessary to go beyond feasibility design on certain unique components in order to ensure a reasonable degree of accurancy 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

The following table tabulates some of the more important design criteria for the prototype and optimum plants. Also, comparative data are presented for Calder Hall and 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 GCPR	Optimum GCPR	Prototype GCPR	Optimum GCPR	(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	1.8 5	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	185
Fuel Charging & Discharging	During Operation	During Operation	During Operation	During Operation	Shut-down Req'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 increases. 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 (3.0% for the prototype) 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 the given size, feasibility of support and sufficient access room determine the maximum core diameter of about 24' (the prototype core is 16' in diameter).

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

Four adjoining buildings house the major operational facilities of the GCPR plant: the reactor building, turbine building, warehouse and shop building, and office building. Plant service buildings consist of two well pump houses, an oil pump house, service water pump house, and a guardhouse.

Yard facilities are as follows: a cooling tower; carbon dioxide, sulfuric acid, oil and water storage tanks; warm waste retention basin with a leaching pit; an Imhoff tank with sludge drying beds for sewage disposal, and a leaching pit; and fire hose houses.

2.1 Reactor-Turbine Building Complex

Reactor Building

The reactor building has three floors above grade and two below. It has a concrete substructure and a steel superstructure. The building contains about 33,000 sq ft of floor area and adjoins the turbine building on the east and office building on the south. A concrete biological shield encloses the reactor vessel located in the center of the building. There are two steam generators located outside the building, one to the north and one to the south of the reactor vessel. Below and west of the steam generators at the ground floor level are located gas coolant equipment rooms and an ion chamber room. On the west side of the reactor vessel is an enclosed spent fuel storage area containing a storage basin with traveling bridge, and an overhead 20/5 ton bridge crane. Three above-grade working levels on the east side of the reactor vessel are as follows:

Ground Floor - Fuel preparation and loading rooms, charge-discharge and control rod rehearsal shaft.

Second Floor - Burst slug detection equipment.

Third Floor - Control rod service machine and traveling bridge, a 15/2 ton bridge crane, a storage room and biological shield cooling equipment.

Two Below-Grade Service Floor Areas - Fuel charge-discharge machine areas.

All floor levels are served by passenger and freight elevators and two enclosed stairwells which exit to the outside of the building at ground floor level.

Turbine Building

The turbine building has three floors above grade and adjoins the east wall of the reactor building and north and east walls of the office building. It has a steel frame superstructure. The total floor area is about 41,000 sq ft and contains the turbine generator, condenser, steam plant auxiliaries, electrical equipment, building heating and ventilating equipment, reactor and turbine control room, and a 25/5 ton service crane.

Heating and Ventilating

All areas except those described below are steam heated and ventilated by two separate systems. The control room, amplifier room and instrument repair shop have an independent air conditioning system. The charge-discharge area is ventilated by an induced draft system that exhausts to the stack above the roof. The spent fuel storage area and the CO_2 filter and blower oil cooling equipment areas are heated by steam unit heaters and ventilated by gravity roof ventilators. The CO_2 blower motors are cooled by evaporative coolers on each unit. The shops and warehouse are heated by unit heaters. The office building is steam heated and ventilated by a central unit.

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.

2.2 Warehouse and Shop Building

The warehouse and shop building is an L-shaped structure adjoining the north and east walls of the turbine building. It has a steel frame superstructure. The warehouse ground floor has about 3,000 sq ft for heavy storage. The second floor has about 3,000 sq ft for light storage and an enclosed cooling tower electrical load center. The shop ground floor has about 3,000 sq ft of area which includes a machine shop, pipe and welding shop, electric shop, instrument shop, staging area, and tool room. A 10/2 ton bridge crane serves the shop area.

2.3 Office Building

The office building has two floors and adjoins the south wall of the reactor and turbine buildings. It has a steel frame superstructure. The ground floor has about 3,000 sq ft of area which includes the entrance lobby, change rooms, health physics office and laboratory. The second floor has about 3,500 sq ft of area which includes offices and a cafeteria.

. 2.4 Service Buildings

The five service buildings for the plant are of prefabricated steel frame and insulated metal panel construction on concrete floor slabs. Service buildings and supporting facilities are provided with electrical power systems consisting of 4,160 v 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. Major equipment contained in the buildings is as follows:

Service Water Pump House: Domestic water pumps and chlorination equipment; fire pumps; motor control center for fire and domestic water, oil and diesel fuel pumps.

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

Two Well Pump Houses: One deep-well pump in each.

Guardhouse: Radio equipment, supervisory alarm panel.

3.0 REACTOR AND AUXILIARIES

3.1 Core

3.1.1 Core Arrangement

The core is essentially cylindrical, 18'-6" high and 16'-2" 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 six 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 (3.0%) 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 configuration under consideration. The geometric parameters for this study are the same as the ones chosen for the 215 mwe enriched reactor, with the exception of core volume. Both the volume and the enrichment are governed by the heat transfer and metallurgical considerations for duplicating the operating conditions of the 215 mwe enriched reactor.

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	0.75"
ID of Graphite Sleeve	3.0"
Enrichment	3.0%
Core Diameter	16'-2"
Core Height	18'-6"
Slug 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 given in Table 3.1.2.

TABLE 3.1.2

NUCLEAR CHARACTERISTICS OF THE 44 MWE PROTOTYPE ENRICHED URANIUM REACTOR AT OPERATING TEMPERATURE

Infinite Multiplication Factor, k_∞	1.2345 clean 1.2011 poisoned
Initial Conversion Ratio	0.635
Effective Multiplication Factor, k_{eff}	1.1365 clean 1.1057 poisoned
Thermal Neutron Flux, n/cm^2-sec	
Average in Total Core	0.9×10^{13}
Maximum	1.7×10^{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 16'-2" diameter and 18'-6" high. The design provides for Wigner growth, thermal expansion and seismic forces. The total weight of finished machined graphite is approximately 340,000 lbs.

The moderator and reflector graphite is grade AGOT, density 1.7 g per cm^3. The graphite structure consists of ten levels of bricks arranged in four concentric rings around 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 temperatures of the core. Vertical and horizontal neutron streaming is made negligible by rotation of alternate levels of bricks.

The graphite structure has 344 vertical fuel channels of 4" diameter on a 7" lattice pitch. In addition there are 35 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 centerline 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 a large heat transfer area is required. Fuel heat transfer area may be obtained either (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) Horizontally, as a chopped zero order Bessel function of core radius.

c) 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 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 on the following page.

Table 3.1.4

THERMAL PERFORMANCE DATA

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

Plant

1.	Net plant electric power, mwe	44
2.	Pumping power, mwe	3.2
3.	Auxiliary equipment power, mwe	3.8
4.	Gross plant electric power, mwe	51.0
5.	Steam cycle heat rate, Btu/kwhr	8500

Reactor

6.	Reactor thermal power, mw	125
7.	Average reactor specific power, mw/M.T. of uranium	6.9
8.	Inlet gas temperature, F	463
9.	Mean exit gas temperature, F	1000
10.	Reactor coolant flow, lb/hr	2.9×10^6
11.	Reactor inlet pressure, psia	400
12.	Pressure drop in core, psi	13
13.	Pressure drop in steam generator, psi	5
14.	Pressure drop in primary system piping, psi	5
15.	Total primary system pressure drop, psi	23
16.	Maximum fuel cladding temperature, F	1300
17.	Coolant mass velocity in central fuel channel, lb/hr-sq ft	5.6×10^5

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 3" thick walls, 18' diameter and 40'-0" height. The pressure shell is designed for an internal pressure of 418 psig at 650 F and is constructed of carbon steel plate. The inner temperature barrier is 3/4" thick and made of stainless steel due to the 1,000 F gas temperature. In order 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 463 F flows in the annulus between the barrier and the vessel outer wall. The exterior of the outer wall of the pressure vessel is insulated.

The coolant gas enters the vessel just below the top head, flows downward through the annulus mentioned above, 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 access to fuel element channels when required by charge-discharge machine.

The vessel is supported on columns equally spaced around the periphery of the bottom of the vessel. The total loading to be supported is the combined weight of the pressure shell, grid support structure, and the graphite stack including reflector. The grid structure consists of six stringers 1" thick and 24" deep spaced on 28" centers and six intercostals 1" thick and 24" deep spaced on 28" centers. A 24" deep ring girder circumscribes the grid structure. The total estimated weight of the pressure vessel, supports, and internals, including the graphite stack, is 350 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 geartrain, 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 eight 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 steam generator. Temperatures and pressures in each of the main gas loops are recorded at the reactor inlet and outlet, and at the steam generator outlet. 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 steam generators and blowers.

The total flow into the purifiation system is recorded and provided with a low-flow alarm. The flow through the dryer is also recorded and has a low-flow alarm. A dew point analyzer having an alarm for high moisture content is located on the outlet of the dryers.

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

An integrating flowmeter measures the amount of CO_2 fed into the system. Two oxygen analyzers are provided in the CO_2 system. A dew point analyzer having an alarm for high moisture content is connected on the discharge side of each of the blowers.

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 performs the following functions:

a. Positions the regulating rods and also adjusts the gas flow through the reactor to maintain a constant pressure in the high-pressure steam line.

b. Maintains the temperature of the reactor inlet gas at a fixed value by throttling low-pressure steam flow from the generator.

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

Steam Pressure Control

An average flow signal from the high-pressure steam line 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 both 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 20% 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 two blower discharge lines operates a control valve in the low-pressure steam header 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: 21 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 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 shutdown of the reactor.

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

CAUSE OF ACTION	Departure from Normal	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
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	X			
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. Gas samples are withdrawn from each fuel channel into two sampling headers. There are 720 such headers arranged in thirty-six 10 x 10 piping matrices. This piping is grouped, brought out of the pressure vessel, and terminated at two 45-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. To avoid loss of coolant gas, the samples are returned to the main coolant system by means 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 two 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 CO_2 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

Each coolant loop is designed to handle half the total gas flow. The gas leaves the reactor at 1,000 F and, after passing through the steam generator where it is cooled to 450 F, is returned to the reactor at 463 F (the heat input by the blowers produces a 13 F temperature rise at full load). The gas pressure at the reactor inlet is 400 psia and the total system pressure loss is 23 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 463 F by automatic controls at the steam generators.

3.6.3 Blowers and Drives

Two variable speed blowers are provided with each blower circulating half the total coolant flow. The pressure rise across the blowers is 23 psi, with gas entering at 450 F and 377 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 either loop 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 a 25-ton capacity insulated and refrigerated storage tank. The refrigeration system for the storage tank includes a compressor and submerged cooling coils in the tank.

Equipment for initial charging of the reactor coolant system with CO_2 is designed to fill the system completely in four 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, 750 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 shutdown periods. The steam pressures in the high and low-pressure circuits are normally greater than the gas pressure; hence, leakage of radioactive gas into the steam circuits 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 the heat source is clean, high-pressure radioactive gas at relatively low temperature, the flow of gas is very large, and 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.

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

Retention Basin Monitoring - 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.

Monitoring of Operating Personnel - Fixed and portable personnel monitors are distributed at key points through the reactor building.

Stack Gas Monitoring - 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. The side walls are octagonal and the total thickness is at least 9'6"; the top and bottom slabs are 10'6" thick. The inside face of the concrete shield is located 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 side 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 9'6" 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 CO_2 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 System

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 inner shield surface 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 the 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 shut-down 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 one year's supply of new fuel assembly components - fuel elements, graphite sleeves and end spiders. The fuel elements and components are prepared and 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.

4.0 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 3.0% 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. Six 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 prototype plant the enrichment is 3.0%.

The fuel pellets are die-pressed from uranium dioxide mixed with a binder and sintered to 95% of their maximum 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 released 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 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 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 upon 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 CO_2 atmosphere at 1,000 F is open to question, due to the possibility of a significant rate of reaction between CO_2 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 support 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 finished sleeve is then impregnated and coated with silicon carbide to obtain a protective surface coating.

5.0 STEAM POWER PLANT

General

The steam power plant consists of two steam generator units which supply steam at dual pressures to a single turbine generator. Equipment and control systems are of conventional design except for the steam generators and the primary cycle equipment.

Selection of Steam Cycle

The steam power 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:

	Optimum Plant Turbine Heat Rate Btu/kwhr	Gross Power Output, kw	
		Prototype	Optimum
Dual-Pressure Reheat Cycle	8,420	50,000	250,000
Single-Pressure Reheat Cycle	8,960	47,000	235,000

Turbine Generator

One steam turbine generator is provided. The turbine is a tandem compound, condensing, reheat unit rated at 50,000 kw, 3,600 rpm, with a maximum capability of 53,000 kw at the design steam conditions. 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- 3/4" Hg abs. The generator is a 13.8 kv, 3-phase, 60 cycle, 3,600 rpm hydrogen cooled unit, nominally rated 64,000 kva, 0.85 pf, 0.64 short circuit ratio at 30 psig hydrogen pressure with a direct connected exciter.

Feedwater Cycle

Condensate is pumped from the main condenser through the air ejector condenser and the low-pressure heater to the deaerator and thence to the high-pressure heater and 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 two-pass, divided water box type complete with two circulating water pumps, two condensate pumps, and air ejector, and all required accessories. A multiple-section induced draft cooling tower is furnished. Condensing pressure is 1-3/4" Hg abs with 72 F water.

Dump Steam Condenser and Auxiliaries

A dump steam condenser complete with all required auxiliaries is provided. It is designed to handle 20% of the output from the steam generators and operates at atmospheric pressure. Electrical apparatus is connected to the emergency power system.

Boiler Feed Pumps

Two low-pressure and two high-pressure motor driven boiler feed pumps are furnished. 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 generator, generator transformer, normal auxiliary transformer, start-up transformer and a 132 kv substation with two transmission circuits. They are arranged to operate as a unit system. The generator, generator transformer and normal auxiliary transformer are connected by isolated phase bus, without intervening circuit breakers, to form an integrated operating unit.

The transmission system facilities include a 132 kv substation with a 4-section ring bus and four oil circuit breakers to connect the generator system and the start-up transformer to two transmission circuits. The generator transformer is rated 57,000 kva, 13.8 kv delta low-voltage, 132 kv wye high-voltage, with tap changing under load. The normal auxiliary transformer is rated

7,500 kva, 13.8 kv delta high-voltage, 4.16 kv wye low voltage. The start-up transformer is rated 7,500 kva, 132 kv wye high-voltage, 4.16 kv 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

Raw water is supplied by one of two deep well pumps. Emergency peak demands are met by operating both pumps. The raw water is piped to the domestic water system, fire protection system, cooling tower make-up system and steam generator makeup.

Domestic water is obtained by chlorinating raw water and accumulating it in a storage tank for distribution.

Fire protection water can be supplied either from the wells or the emergency storage tank to the suction of two fire pumps. Automatic sprinkler systems, standpipes with hose stations, and yard hydrants are on a looped distribution piping system.

Make-up water for the condenser cooling system enters the turbine building by gravity from the upper portion of the fire protection storage tank. The pH control, corrosion inhibitors, and algae control are supplied at the sump. Make-up water for the steam generator and the storage basin is supplied from a branch of the same gravity line from the fire protection storage tank and is treated in a cation-anion regeneration type de-ionizer.

6.2 Air Systems

Plant air is provided for general utility and for the service and charge-discharge machine by a motordriven compressor, receiver and distribution piping. 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 building fixtures through yard sewer mains to a sewage lift station. From the lift station, the sewage is pumped to an Imhoff tank. A sludge drying bed and a leaching pit are provided.

6.4 Radioactive Waste Disposal

Gaseous wastes are disposed of by venting through a 6' diameter stack 100' above the high point of the reactor building. The sources of these wastes are: the biological shield cooling air with radioactive argon; primary coolant (released during normal depressurizing) with carbon monoxide, carbon dioxide, and radioactive argon; and primary coolant (released on depressurizing after a fuel element leak) with an estimated 500 curies of gaseous fission products. This last source can be released only during favorable meteorological conditions in which case weekly tolerance may be exceeded by 25% in the plant area. This is acceptable because of the infrequency of this condition.

Liquid wastes are considered "warm" if, by dilution and temporary retention, their activity can be reduced to a level which will permit discharge into the ground. "Hot" liquid wastes are those with higher activities and will be shipped offsite for disposal.

Solid radioactive wastes, such as contaminated filters, tools, and reactor parts, are packaged 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 power distribution systems.

Emergency power is provided for essential plant auxiliaries from an emergency diesel-electric generator integrated with the 480 v 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 NRTS central station.

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

Intercommunication is provided by five 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 Systems

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

Diesel oil for the emergency diesel-electric generator and start-up of the boiler for building heating is received by tank truck 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 were prepared for the 44 mwe prototype plant based on the assumption that the work is performed under cost-plus-fixed-fee contracts. Construction would start approximately three months after the start of detailed design and would require three years to complete. The over-all program would be completed within 42 months after the start of preliminary engineering and is estimated to cost approximately $51,000,000; the cost is $49,000,000 excluding the design and construction of the switchyard and transmission lines.

The estimate of cost for the prototype enriched reactor assumes construction at the NRTS. Costs include escalation commensurate with the time schedules for completing the project.

A summary of schedules and a preliminary estimate of cost are included in the following sections.

7.1.1 Design and Construction Schedule

Preliminary Engineering (Title I)	3 months
Detailed Design (Title II)	12 months
Construction, Inspection (Title III)	36 months*
Combined Design & Construction Period	42 months

*Construction to begin 3 months after start of detailed design.

7.1.2 Preliminary Estimate of Cost

Engineering		$ 5,300,000
Construction		
Site Work	$ 1,010,000	
Buildings	4,440,000	
Reactor	12,990,000	
Power Generation	15,360,000	
Electrical Distribution	1,730,000	
Total Construction		35,530,000
Total Engineering & Construction		40,830,000
Contingency @ 25%		10,170,000
Grand Total		$51,000,000*

*$49,000,000 excluding design and construction cost of switchyard and transmission lines.

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/M.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 3.1×10^8 kwhr/yr.

The 44 mwe power plant lifetime unit cost is estimated as follows:

Fixed Charges (14%)	25.2	mills/kwhr
Operating, Maintenance, General and Administrative	3.5	
Nuclear Fuel Costs	3.5	
Total Cost	32.2	mills/kwhr

7.2.1 Annual Fixed Charges

These charges are taken as a percentage of the total capital costs. 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 upon 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 upon 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 of 9.4% would ordinarily be applied to the initial total capital cost as tabulated on page 50.

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 48 reflect this method of calculation.

LIFETIME ANNUAL FIXED CHARGES

Capital Cost Components	Capital Cost (Millions)	Rate	Annual Cost (Millions)
Construction and Engineering (Excluding Switchyard and Transmission Lines)	49.0	9.6%	4.7
Fuel Fabrication Inventory	0.54	7.1%*	0.04
Interest During Construction (7.5% of Const. & Eng.)	3.7	7.6%**	0.28
Working Capital	1.4	5.1%***	0.07
Start-Up Costs	0.55	7.6%**	0.04
Equipment Stores	0.2	9.6%	0.02
TOTAL CAPITAL COST	55.4	9.4% 14.0%	5.2 7.8

*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 74 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 38. The plant organization has eight clerical and supervisory, 15 maintenance, 47 operating employees, and four guards. The cost of operating and maintenance supplies is based on the cost 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:

	Annual Costs
Wages and Salaries	$527,000
Payroll Taxes and Benefits @ 14%	73,000
Operating Supplies and Expenses	93,000
Maintenance Supplies and Expenses	62,000
Demand Charges	60,000
Insurance	140,000
General and Administrative Expense at 14% of Production Cost Less Fuel	135,000
Total	$1,090,000
Say	$1,100,000

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 1.9% of U^{235} and 6.9 grams plutonium per kg of fuel.

c. In computing fuel rental charges, the 3.0% enriched uranium fuel is valued at $375 per kg of uranium, and leased from the AEC at 4% per year.

d. The reactor is loaded with 18 metric tons of 3.0% enriched uranium and operates at 80% plant factor and 125 mw thermal rating. The fuel discharged from the reactor is 0.01 metric ton per day.

Fuel Cycle Summary - Prototype Reactor

Annual Fuel Costs (3.65 metric tons per year):

	Cost Per kg Uranium	Annual Cost (Thousands)
Net Fuel Costs	$110	$400
Fabrication Cost	30	110
Fuel Inventory	7	320
Fuel Processing	65	240
Annual Total Cost of Fuel		$1,070
	Say	$1,100

8.0 SUMMARY OF KE STUDY NO. 13A

STEAM CYCLE ANALYSIS - 44 MWE (NET) ENRICHED URANIUM REACTOR

Introduction - The purpose of this study is to analyze the steam cycles for the 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 Conclusions - 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 pressures 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 I on the following page.

TABLE I
STEAM CYCLE SUMMARY

Coolant Inlet Temp. F	Coolant Outlet Temp. F	Feedwater Temp. F	Superheater and Reheater Outlet Temp. F	HP Steam Pressure PSIA	LP Steam Pressure PSIA	Percent of Steam Flow from HP Evaporator	Heat Rate Btu/kwhr
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.

** This became 8420 in final cycle analysis.

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:

		Units	Prototype Plant	Optimum Plant
Net Electric Power Output		mwe	44	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	CO_2
	- Flow	lbs/sec	805	3900
	- Weight	tons	12	39
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

	Units	Prototype Plant	Optimum Plant
Reactor Pressure Vessel - Shape		Cylinder with hemi-spherical heads	Cylinder with hemi-spherical 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

44 MWE (NET) ENR. GCPR DRAWING INDEX

DWG. NO.	TITLE
CIVIL	
5714-GCPR-101-P 201	NRTS LOCATION MAP
100-P 201	PLANT GENERAL ARRANGEMENT
ARCHITECTURAL	
5714-GCPR-601-A-P201	REACTOR-TURBINE BUILDING - PLAN AND ELEVATION
601-A-P202	REACTOR-TURBINE BUILDING - PLAN AND SECTION
ELECTRICAL	
5714-GCPR-602-E-P201	ELECTRICAL ONE LINE DIAGRAM SHEET NO. 1
602-E-P202	ELECTRICAL ONE LINE DIAGRAM SHEET NO. 2
MECHANICAL	
5714-GCPR-601-M-P700	REACTOR VESSEL ASSEMBLY
601-M-P703	CORE PLAN AND ELEVATION
601-M-P201	EXTERIOR FUEL HANDLING FLOW SHEET
PIPING	
5714-GCPR-602-P-P 201	STEAM PIPING AND INSTRUMENT DIAGRAM
602-P-P 202	CONDENSATE PIPING AND INSTRUMENT DIAGRAM
602-P-P 203	BOILER FEED WATER PIPING AND INSTRUMENT DIAGRAM
602-P-P 204	CIRCULATING AND SERVICE WATER PIPING AND INSTRUMENT DIAGRAM
602-P-P 205	SIMPLIFIED HEAT BALANCE - 50,000 KW - 1 3/4" Hg BACK PRESSURE - GAS TEMP. IN = 1000 °F, OUT = 450 °F
601-P-P700	MAIN GAS COOLING SYSTEM FLOW DIAGRAM
601-P-P701	GAS PURIFICATION SYSTEM FLOW DIAGRAM
601-P-P702	GAS VENT SYSTEM
601-P-P704	BURST SLUG DETECTION SYSTEM FLOW DIAGRAM
601-P-P705	MAIN GAS COOLANT PIPING PLAN AND ELEVATION
MECHANICAL SPECIALTIES	
5714-GCPR-601-MS-P728	CONTROL ROD SHIELD PLUG ASSEMBLY
601-MS-P741	CHARGE MACHINE GENERAL ASSEMBLY
601-MS-P771	SERVICE MACHINE GENERAL ASSEMBLY

REFERENCES

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44 MWE (NET) ENR.

GCPR DRAWING INDEX

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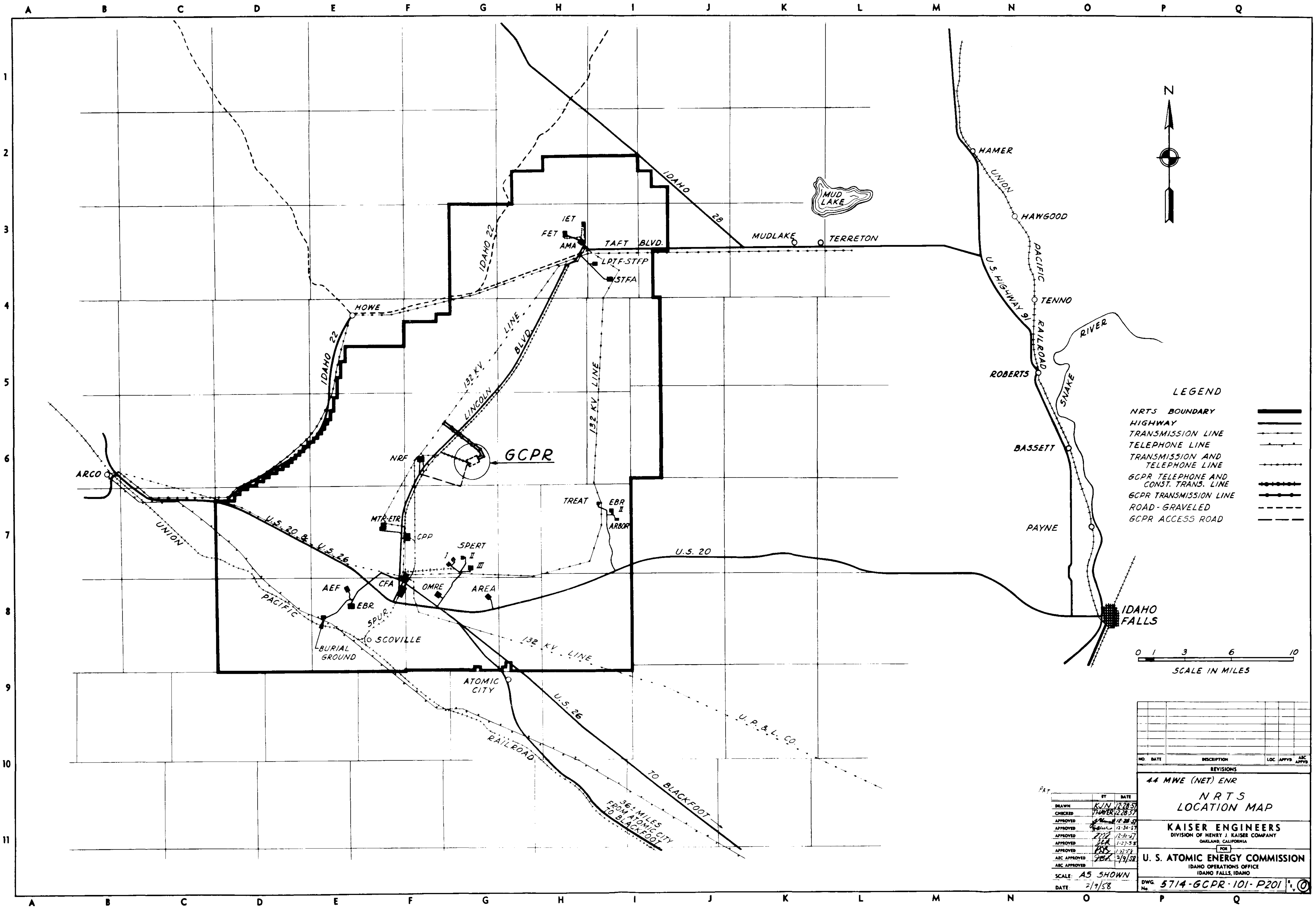
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BURIAL GROUND
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ATOMIC CITY
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U.S. 20
U.S. 26
TO BLACKFOOT
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UNION PACIFIC RAILROAD
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MUDLAKE
TERRETON
HAMER
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TENNO
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BASSETT
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IDAHO FALLS
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SNAKE RIVER
N
LEGEND
NRTS BOUNDARY
HIGHWAY
TRANSMISSION LINE
TELEPHONE LINE
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44 MWE (NET) ENR
NRTS LOCATION MAP
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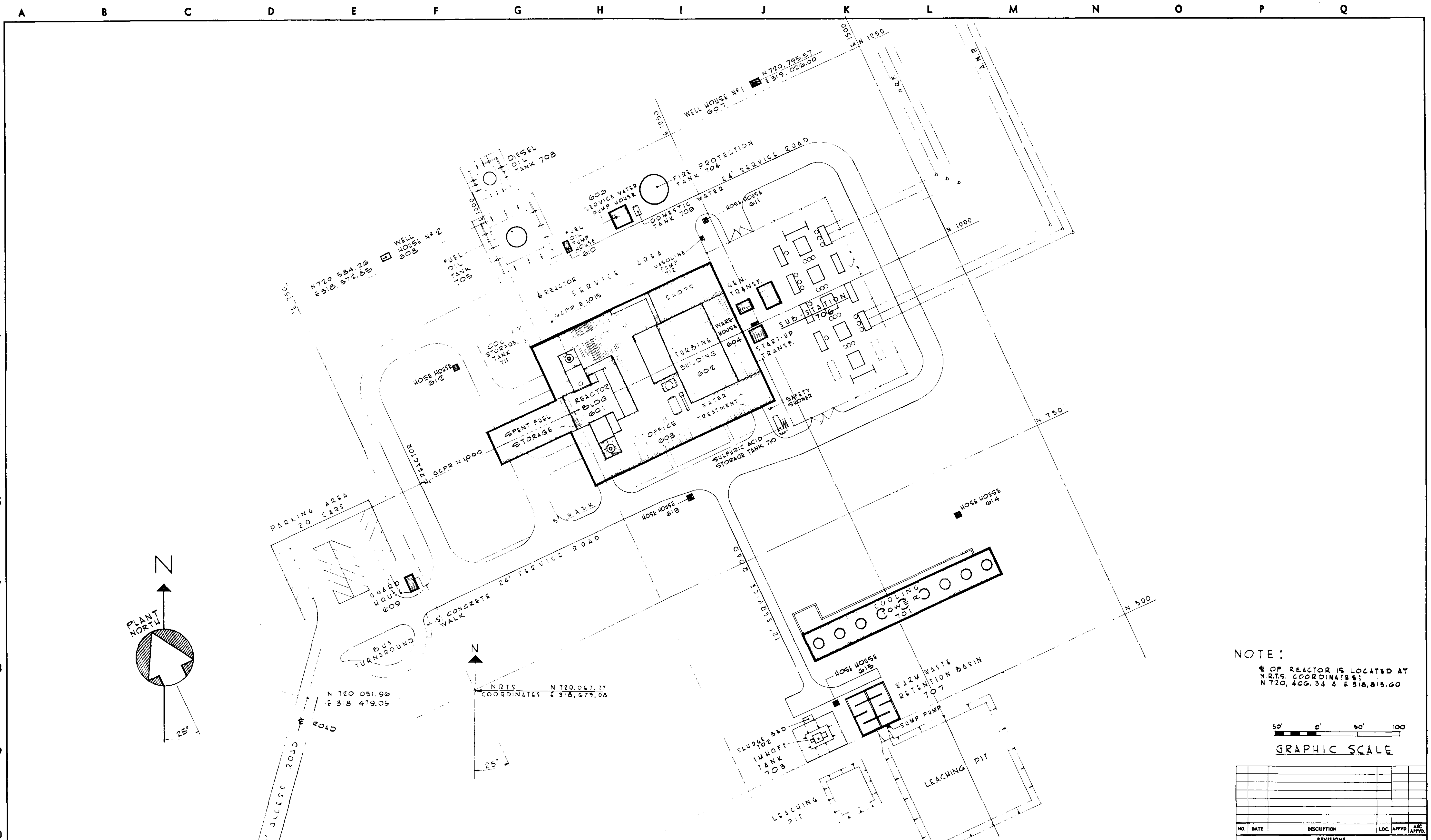

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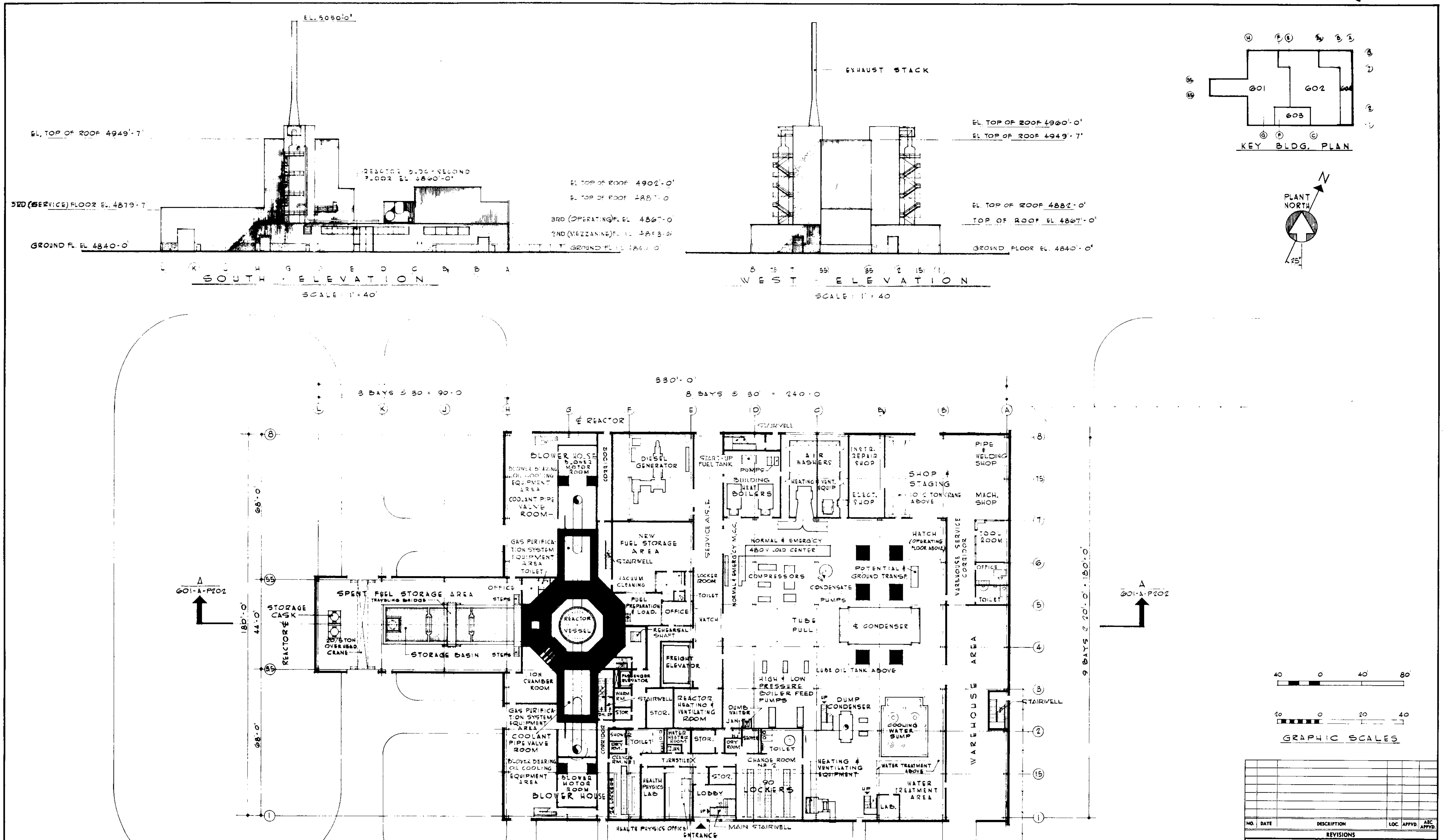

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44 MWE (NET) ENR
REACTOR - TURBINE BLDG
PLAN AND ELEVATION

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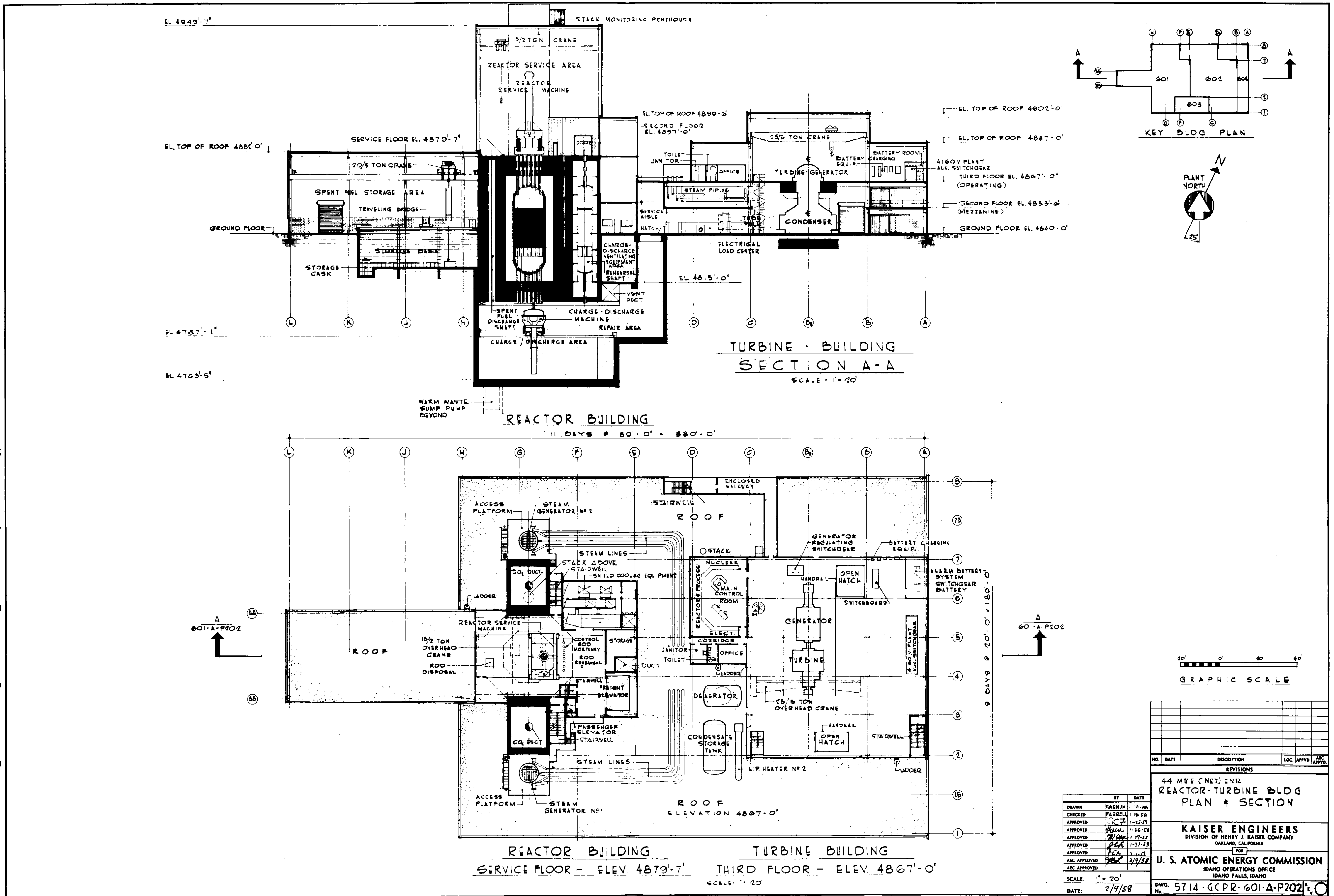

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REACTOR-TURBINE BLDG
PLAN & SECTION

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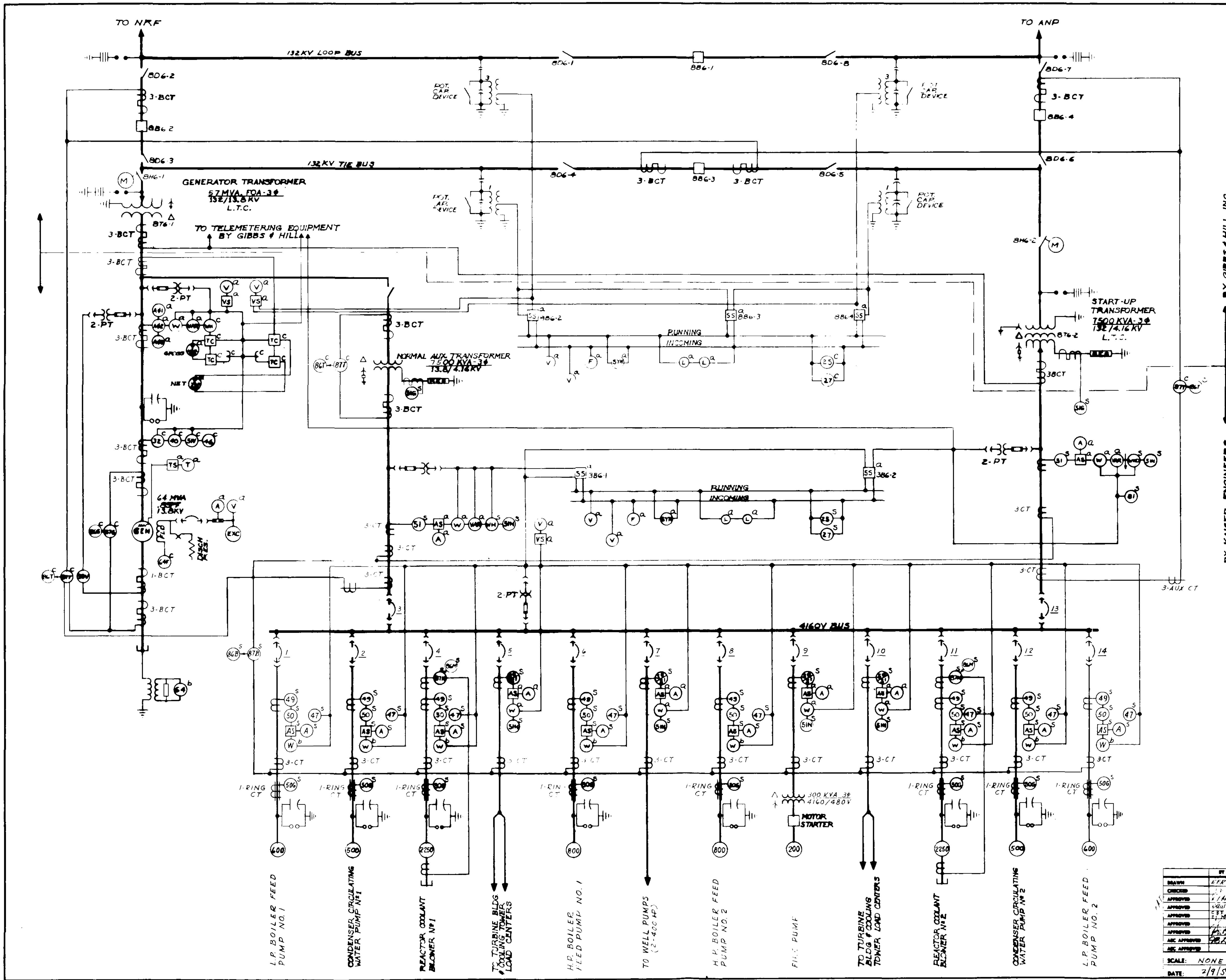

RELAY DESIGNATIONS

- (25) SYNCHRONIZING RELAY.
- (27) UNDERVOLTAGE RELAY (FOR USE WITH (25)).
- (32) REVERSE POWER (ANTI-MOTORING) RELAY FOR GENERATOR.
- (40) LOSS-OF-FIELD RELAY FOR GENERATOR FIELD.
- (46) CURRENT-BALANCE RELAY FOR GENERATOR.
- (47) 3 PHASE UNDERVOLTAGE AND PHASE SEQUENCE RELAY.
- (49) THERMAL OVERLOAD RELAY.
- (50) INSTANTANEOUS OVERCURRENT PHASE-FAULT RELAY.
- (50G) INSTANTANEOUS OVERCURRENT GROUND-FAULT RELAY.
- (51) TIME-DELAY OVERCURRENT PHASE-FAULT RELAY.
- (51G) TIME-DELAY OVERCURRENT NEUTRAL GROUND-FAULT RELAY.
- (51N) TIME-DELAY OVERCURRENT RESIDUAL GROUND-FAULT RELAY.
- (51V) TIME-DELAY OVERCURRENT WITH VOLTAGE RESTRAINT RELAY FOR GENERATOR FOR BACK-UP PROTECTION.
- (64) GROUND-FAULT DETECTION RELAY FOR UNIT-OPERATED GENERATOR AND TRANSFORMERS.
- (64F) GROUND-FAULT DETECTION RELAY FOR GENERATOR FIELD.
- (81) UNDERFREQUENCY RELAY.
- (87B) DIFFERENTIAL CURRENT RELAY FOR BUS.
- (87G) DIFFERENTIAL CURRENT RELAY FOR GENERATOR.
- (87M) DIFFERENTIAL CURRENT RELAY FOR MOTOR.
- (87T) DIFFERENTIAL CURRENT RELAY FOR TRANSFORMER.
- (90V) GENERATOR VOLTAGE REGULATOR.
- (86M) AUXILIARY LOCK-OUT RELAY FOR 87M.
- (86B) AUXILIARY LOCK-OUT RELAY FOR 87B.
- (86T) AUXILIARY LOCK-OUT RELAY FOR 87T.
- [TC] THERMAL CONVERTER
- [TS] TEMPERATURE INDICATOR SELECTOR SWITCH
- (T) TEMPERATURE INDICATOR
- [SS] SYNCHRONIZING SELECTOR SWITCH

METER & RELAY LOCATION DESIGNATION

a. LOCATED ON ELECTRICAL CONTROL PANELS.
b. LOCATED ON REACTOR AND PROCESS CONTROL PANELS.
c. LOCATED ON DUPLEX SWITCHBOARD.
s. LOCATED ON METALCLAD SWITCHGEAR.

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44 MWE(NET) ENR.
ELECTRICAL ONE LINE DIAGRAM
SHEET Nº 1

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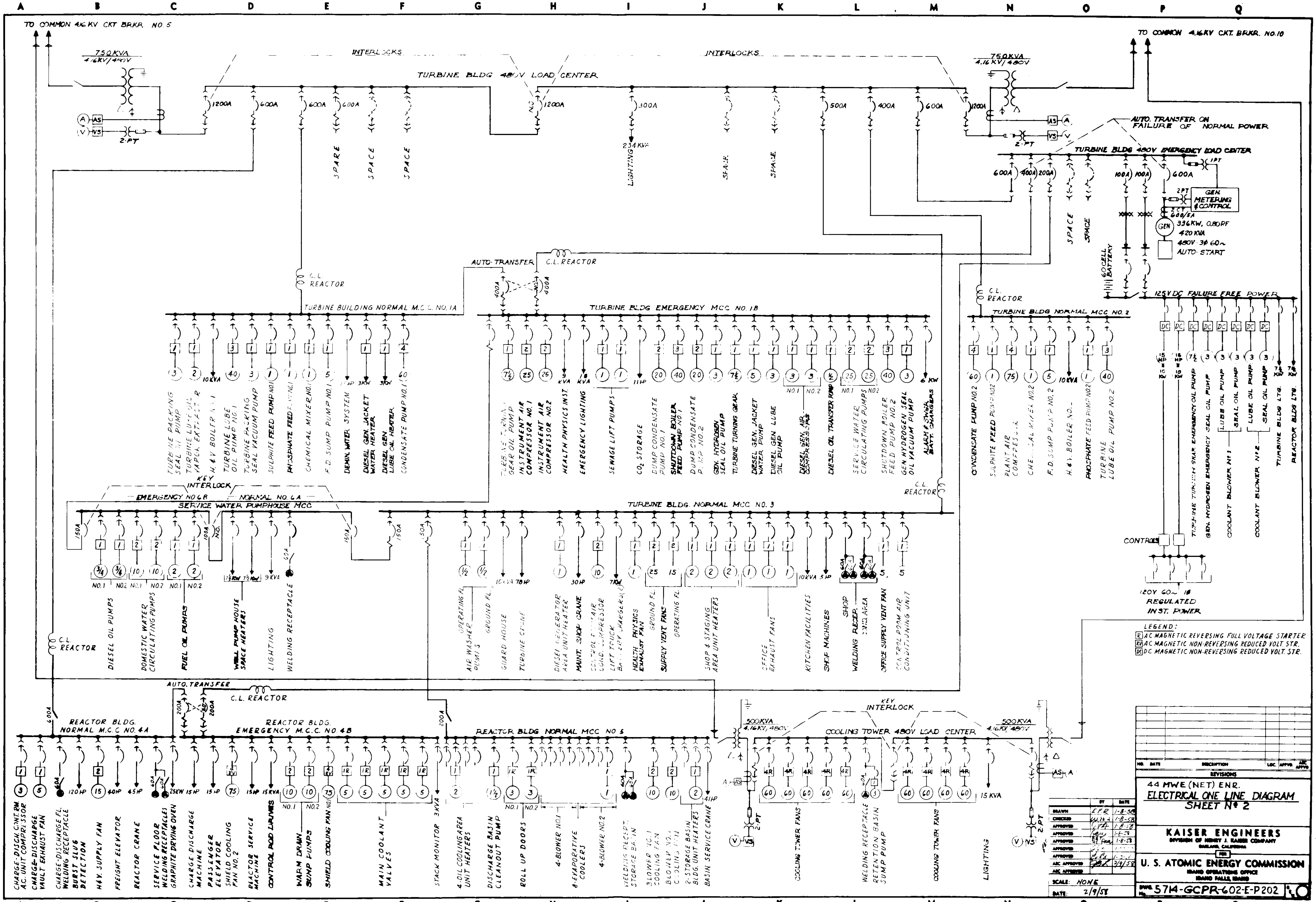
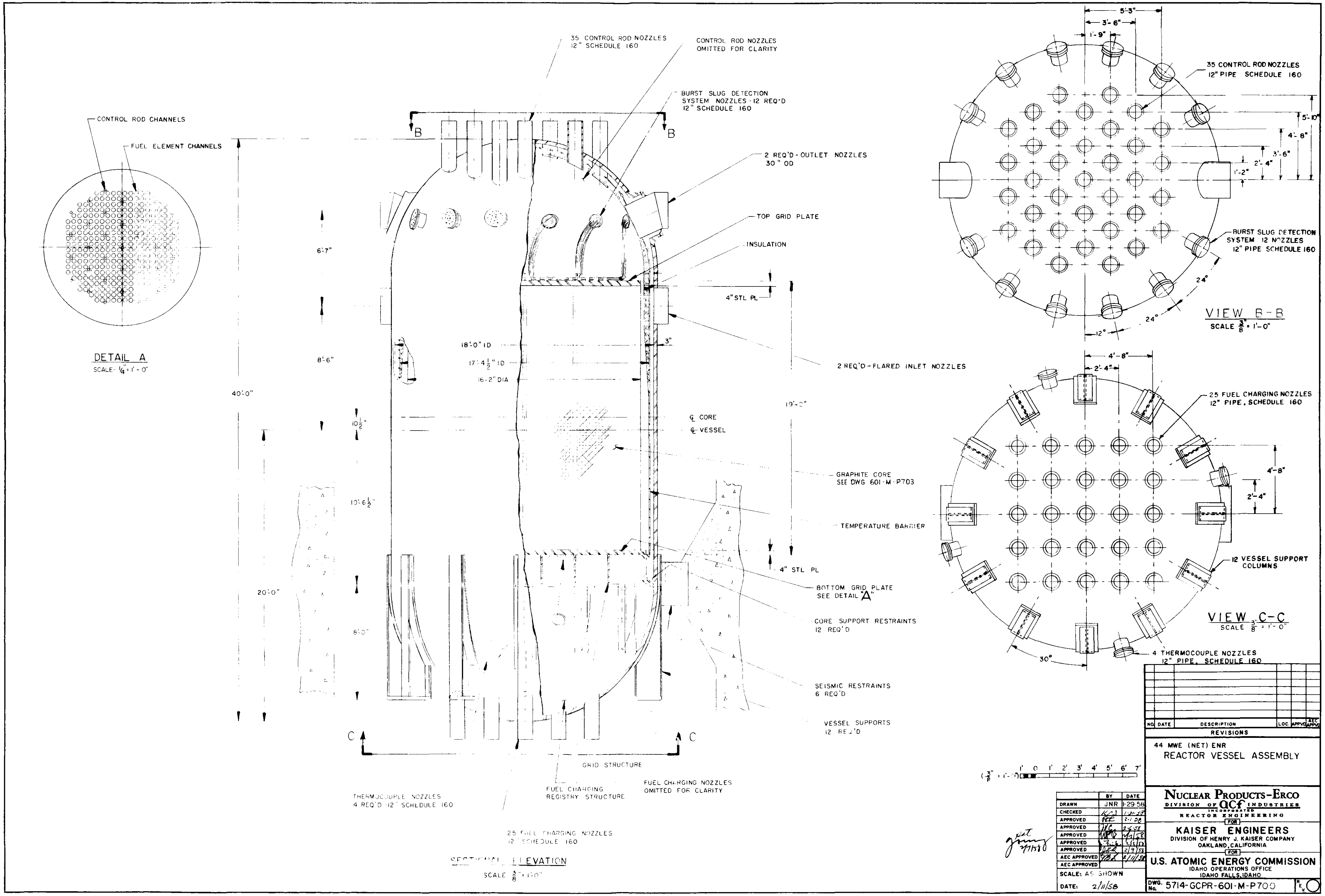
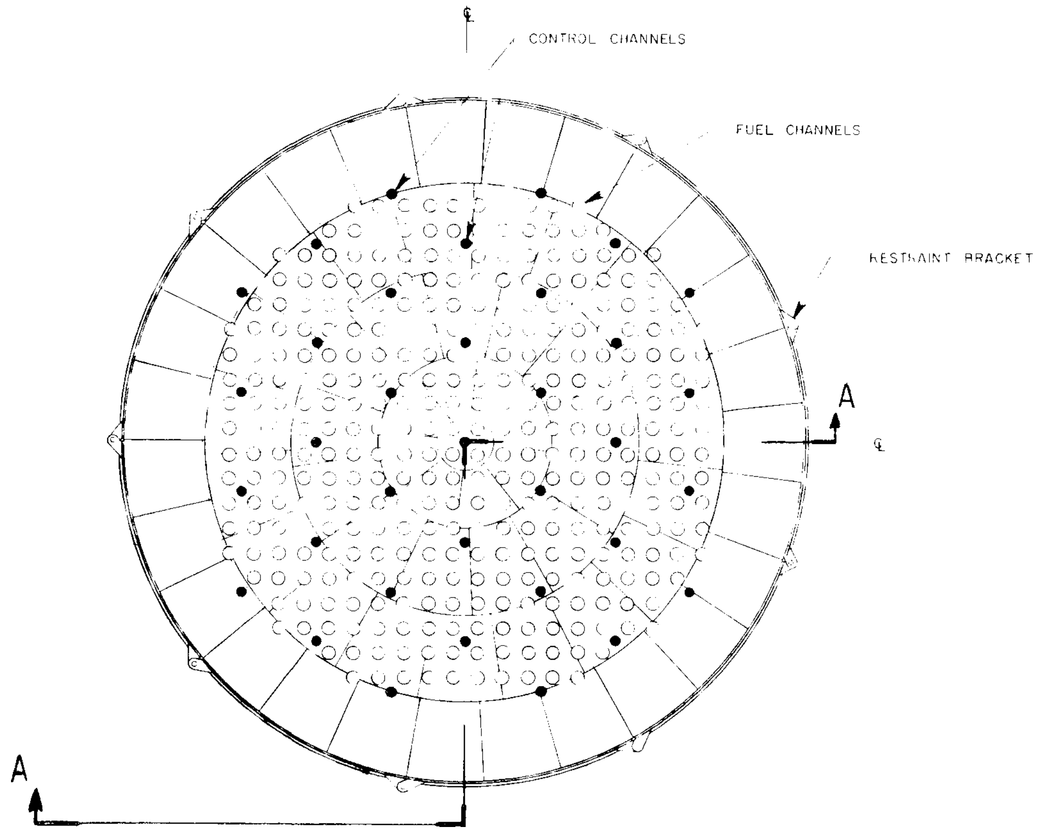
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-602-E-P201

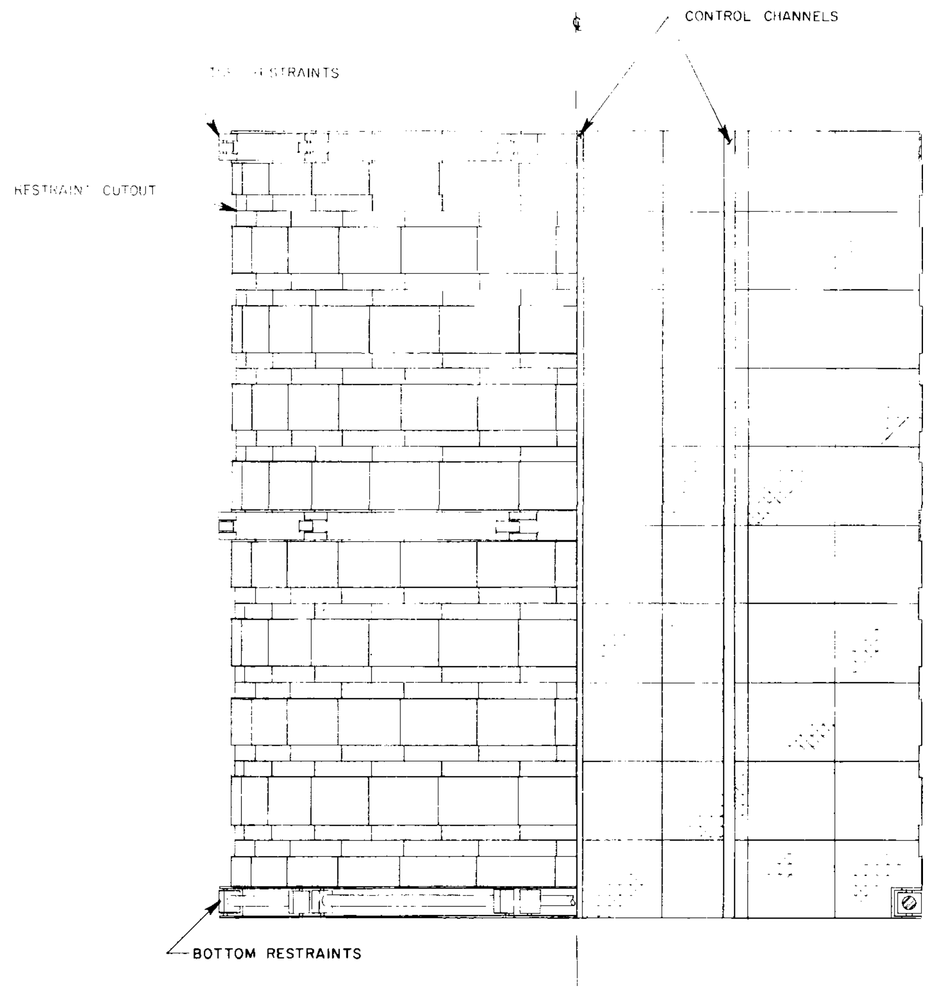


TO COMMON 4.16 KV CKT BRKR. NO. 5
750 KVA 4.16KV/480V
INTERLOCKS
TURBINE BLDG 480V LOAD CENTER
TO COMMON 4.16KV CKT. BRKR. NO. 10
750 KVA 4.16KV/480V
AUTO. TRANSFER ON FAILURE OF NORMAL POWER
TURBINE BLDG 480V EMERGENCY LOAD CENTER
GEN. METERING & CONTROL
336KW, 0.80PF
420 KVA
480V - 3φ 60~
AUTO. START
60 CELL BATTERY
125V DC FAILURE FREE POWER
C.L. REACTOR
AUTO-TRANSFER
TURBINE BUILDING NORMAL M.C.C. NO. 1A
TURBINE BLDG EMERGENCY MCC NO. 1B
TURBINE BLDG NORMAL MCC NO. 2
KEY INTERLOCK
EMERGENCY NO. 6B
NORMAL NO. 6A
SERVICE WATER PUMPHOUSE MCC
TURBINE BLDG. NORMAL MCC NO. 3
CONTROLS
120V 60~ 1φ REGULATED INST. POWER
LEGEND:
R AC MAGNETIC REVERSING FULL VOLTAGE STARTER
RV AC MAGNETIC NON-REVERSING REDUCED VOLT. STR.
DC DC MAGNETIC NON-REVERSING REDUCED VOLT. STR.
REACTOR BLDG. NORMAL M.C.C. NO. 4A
REACTOR BLDG. EMERGENCY M.C.C. NO. 4B
REACTOR BLDG NORMAL MCC NO. 5
COOLING TOWER 480V LOAD CENTER
500 KVA 4.16KV/480V
SPARE
SPACE
LIGHTING
DIESEL OIL PUMPS
DOMESTIC WATER CIRCULATING PUMPS
FUEL OIL PUMPS
WELL PUMP HOUSE SPACE HEATERS
WELDING RECEPTACLE
COOLING TOWER FANS
MAIN COOLANT VALVES
STACK MONITOR
44 MWE (NET) ENR.
ELECTRICAL ONE LINE DIAGRAM
SHEET Nº 2
KAISER ENGINEERS
U. S. ATOMIC ENERGY COMMISSION
SCALE: NONE
DATE: 2/9/58
5714-GCPR-602-E-P202


CONTROL ROD CHANNELS
FUEL ELEMENT CHANNELS
DETAIL A
SCALE- 1/4" = 1'-0"
35 CONTROL ROD NOZZLES
12" SCHEDULE 160
CONTROL ROD NOZZLES
OMITTED FOR CLARITY
BURST SLUG DETECTION
SYSTEM NOZZLES - 12 REQ'D
12" SCHEDULE 160
2 REQ'D - OUTLET NOZZLES
30" OD
TOP GRID PLATE
INSULATION
4" STL PL
3"
2 REQ'D - FLARED INLET NOZZLES
18'-0" ID
17'-4 1/2" ID
16'-2" DIA
6'-7"
8'-6"
40'-0"
10 1/2"
10'-6 1/2"
20'-0"
8'-0"
19'-2"
℄ CORE
℄ VESSEL
GRAPHITE CORE
SEE DWG 601-M-P703
TEMPERATURE BARRIER
4" STL PL
BOTTOM GRID PLATE
SEE DETAIL "A"
CORE SUPPORT RESTRAINTS
12 REQ'D
SEISMIC RESTRAINTS
6 REQ'D
VESSEL SUPPORTS
12 REQ'D
B
B
C
C
GRID STRUCTURE
FUEL CHARGING
REGISTRY STRUCTURE
FUEL CHARGING NOZZLES
OMITTED FOR CLARITY
THERMOCOUPLE NOZZLES
4 REQ'D - 12" SCHEDULE 160
25 FUEL CHARGING NOZZLES
12" SCHEDULE 160
SECTIONAL ELEVATION
SCALE 3/8" = 1'-0"
5'-3"
3'-6"
1'-9"
35 CONTROL ROD NOZZLES
12" PIPE SCHEDULE 160
5'-10"
4'-8"
3'-6"
2'-4"
1'-2"
BURST SLUG DETECTION
SYSTEM 12 NOZZLES
12" PIPE SCHEDULE 160
24°
24°
12°
VIEW B-B
SCALE 3/8" = 1'-0"
4'-8"
2'-4"
25 FUEL CHARGING NOZZLES
12" PIPE, SCHEDULE 160
4'-8"
2'-4"
12 VESSEL SUPPORT
COLUMNS
VIEW C-C
SCALE 3/8" = 1'-0"
30°
4 THERMOCOUPLE NOZZLES
12" PIPE, SCHEDULE 160
(3/8" = 1'-0") 1' 0 1' 2' 3' 4' 5' 6' 7'
NO. DATE DESCRIPTION LOC. APPVD AEC APPVD
REVISIONS
44 MWE (NET) ENR
REACTOR VESSEL ASSEMBLY
BY DATE
DRAWN JNR 1-29-58
CHECKED 1-31-58
APPROVED 2-1-58
APPROVED 2-4-58
APPROVED
APPROVED
APPROVED 2/9/58
AEC APPROVED 2/11/58
AEC APPROVED
SCALE: AS SHOWN
DATE: 2/11/58
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
DWG. No. 5714-GCPR-601-M-P700
REV. 0



PLAN VIEW OF GRAPHITE CORE
SCALE - $\frac{1}{2}$" = 1'-0"



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

CORE GEOMETRY:

CORE O.D. = 16'-2"
CORE HEIGHT = 18'-6"
NO. OF CONTROL ROD CHANNELS = 35 ($3\frac{1}{4}$" I.D.)
NO. OF FUEL CHANNELS = 344 (4" I.D.)
PITCH OF FUEL CHANNELS = 7"
FINISHED MACHINED WEIGHT OF STACK = 340,152 LBS.
EFFECTIVE MODERATOR DIA. = 12'-2$\frac{1}{8}$"
EFFECTIVE MODERATOR HEIGHT = 14'-6"

($\frac{1}{2}$" = 1'-0") 1' 0' 1' 2' 3' 4' 5'

NO.	DATE	DESCRIPTION	LOC.	APPVD	AEC APPVD

REVISIONS

44 MWE (NET) ENR

CORE PLAN AND ELEVATION

	BY	DATE
DRAWN	V. MOORE	1-29-58
CHECKED	K.C.J.	1-31-58
APPROVED	[illegible]	1-31-58
APPROVED	[illegible]	2-6-58
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	2-6-58
AEC APPROVED	[illegible]	2/10/58
AEC APPROVED		

SCALE: $\frac{1}{2}$" = 1'-0"
DATE: 2/10/58

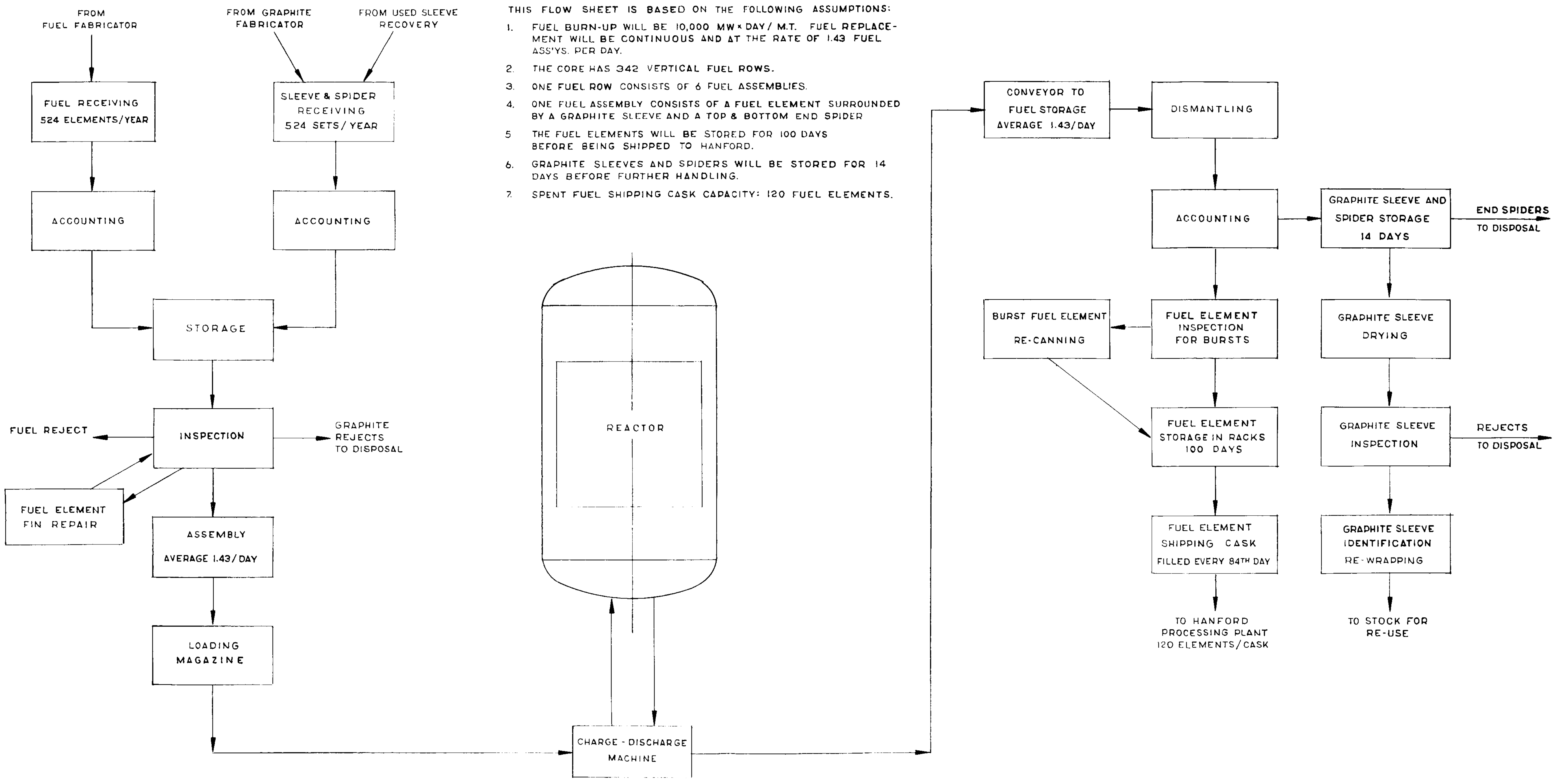
NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
REACTOR ENGINEERING
FOR
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-601-M-P703 REV. O

NOTE:

THIS FLOW SHEET IS BASED ON THE FOLLOWING ASSUMPTIONS:

1. FUEL BURN-UP WILL BE 10,000 MW × DAY / M.T. FUEL REPLACEMENT WILL BE CONTINUOUS AND AT THE RATE OF 1.43 FUEL ASS'YS. PER DAY.
2. THE CORE HAS 342 VERTICAL FUEL ROWS.
3. ONE FUEL ROW CONSISTS OF 6 FUEL ASSEMBLIES.
4. ONE FUEL ASSEMBLY CONSISTS OF A FUEL ELEMENT SURROUNDED BY A GRAPHITE SLEEVE AND A TOP & BOTTOM END SPIDER
5. THE FUEL ELEMENTS WILL BE STORED FOR 100 DAYS BEFORE BEING SHIPPED TO HANFORD.
6. GRAPHITE SLEEVES AND SPIDERS WILL BE STORED FOR 14 DAYS BEFORE FURTHER HANDLING.
7. SPENT FUEL SHIPPING CASK CAPACITY: 120 FUEL ELEMENTS.



NEW FUEL HANDLING

REACTOR FUELING

SPENT FUEL HANDLING

REFERENCES

REF NO	TITLE	DWG. NO.

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

	BY	DATE
DRAWN	E.J. E.R.	12-2-57
CHECKED	Naygandt	12-23-57
APPROVED	Young	12-27-57
APPROVED	Pams	12-30-57
APPROVED	MJ	12-30-57
APPROVED	JLR	1-2-58
APPROVED	KB	1-27-58
AEC APPROVED	TBL	2/9/58
AEC APPROVED		

SCALE: NONE

DATE: 2/9/58

44 MWE (NET) ENR.

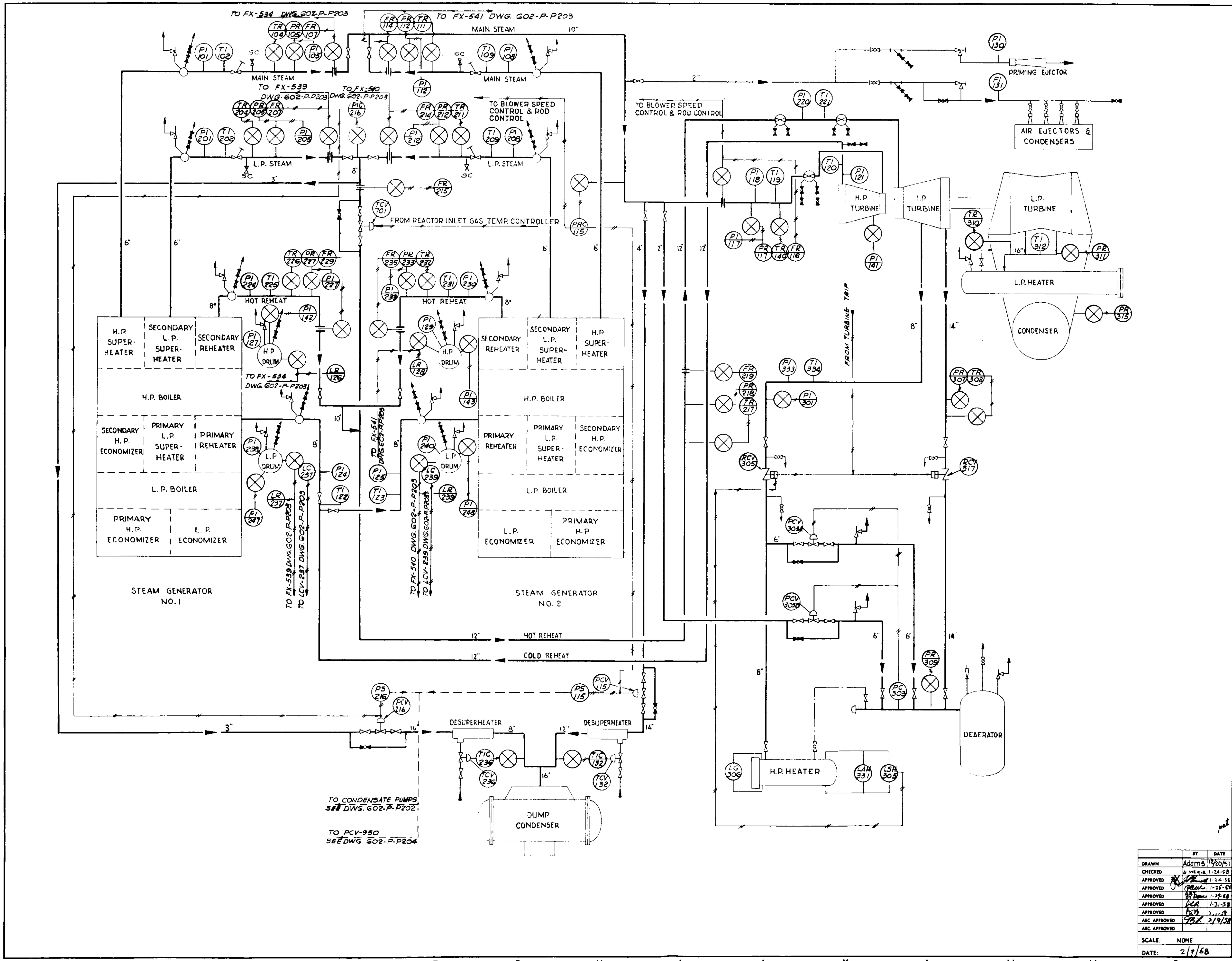
EXTERIOR FUEL HANDLING FLOW SHEET

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

FOR

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

DWG. No. 5714-GCPR-601-M-P201 REV. 0



INSTRUMENT LEGEND

FR	FLOW RECORDER
FRC	FLOW RECORDER CONTROLLER
FX	AVERAGING RELAY
FIC	FLOW INDICATING CONTROLLER
FI	FLOW INDICATOR
FC	FLOW CONTROLLER
FCV	FLOW CONTROL VALVE
MRA	MOISTURE RECORDER ALARM
SIC	SPEED INDICATING CONTROLLER
PdR	PRESS. DIFFERENTIAL RECORDER
PR	PRESSURE RECORDER
PRC	PRESS. RECORDER CONTROLLER
PIC	PRESS. INDICATING CONTROLLER
PC	PRESS. CONTROLLER
PI	PRESS. INDICATOR
PCV	PRESS. CONTROL VALVE
PS	PRESS. SWITCH
PX	AVERAGING RELAY (PRESSURE)
SC	SAMPLE CONNECTION
SV	SOLENOID VALVE
OR	OXYGEN RECORDER
LIA	LEVEL INDICATING ALARM
LR	LEVEL RECORDER
LRC	LEVEL RECORDER CONTROLLER
LIC	LEVEL INDICATING CONTROLLER
LC	LEVEL CONTROLLER
LI	LEVEL INDICATOR
LCV	LEVEL CONTROL VALVE
LG	LEVEL GAGE
LSH	LEVEL SHUTDOWN HIGH
LAH	LEVEL ALARM HIGH
LAL	LEVEL ALARM LOW
ALS	AIR LOADING STATION
TR	TEMP. RECORDER
TRC	TEMP. RECORDER CONTROLLER
TIC	TEMP. INDICATING CONTROLLER
TI	TEMP. INDICATOR
TCV	TEMP. CONTROL VALVE
HCV	HAND CONTROL VALVE
RCV	REVERSE CURRENT VALVE
CIA	CONDUCTIVITY INDICATING ALARM
PIA	PRESS. INDICATING ALARM
PdA	PRESS. DIFFERENTIAL ALARM
TX	AVERAGING RELAY

LOCALLY MOUNTED

BOARD MOUNTED

MOTOR OPERATED VALVE

PISTON OPERATED CHECK VALVE

TRANSMITTER

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

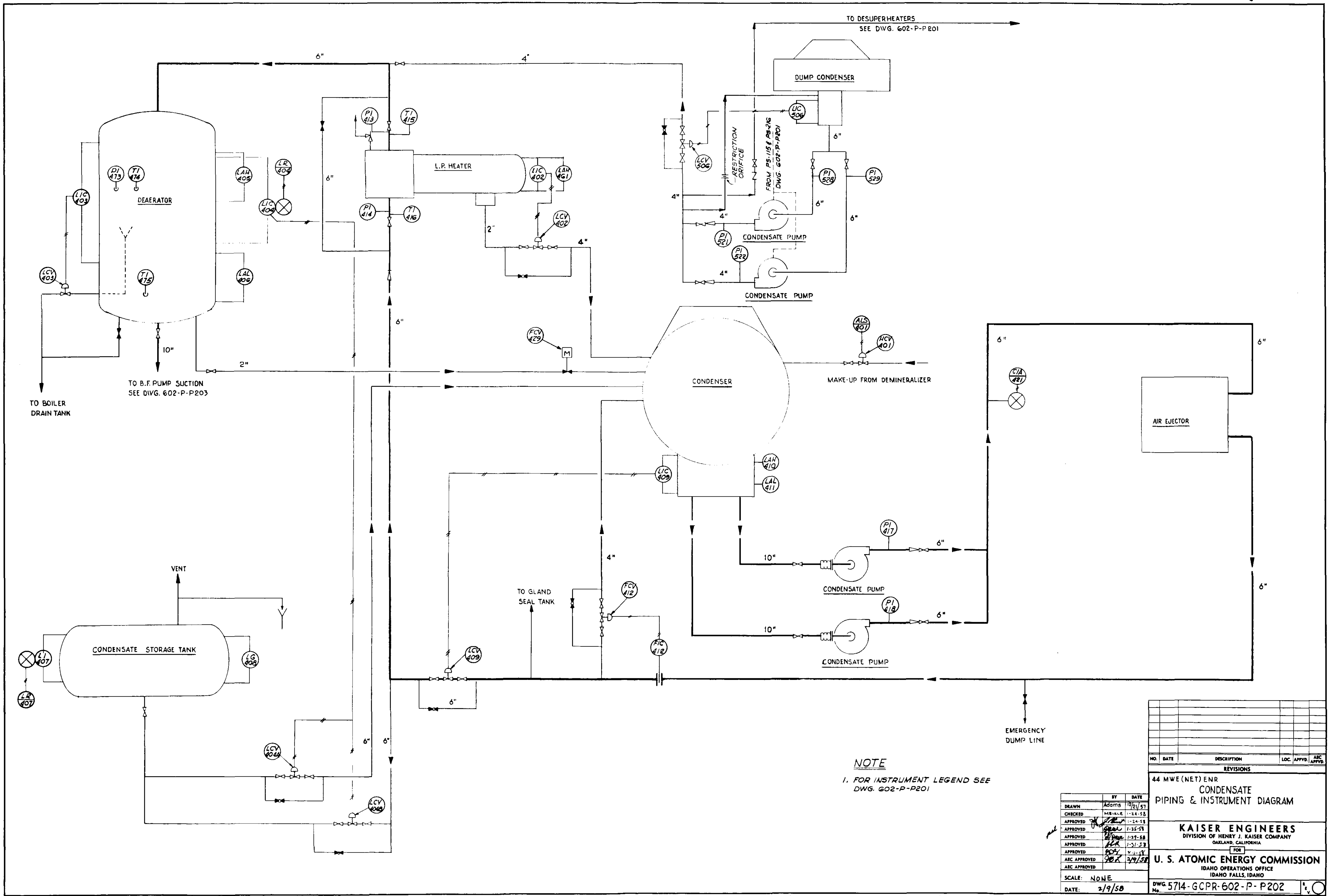
44 MWE (NET) ENR

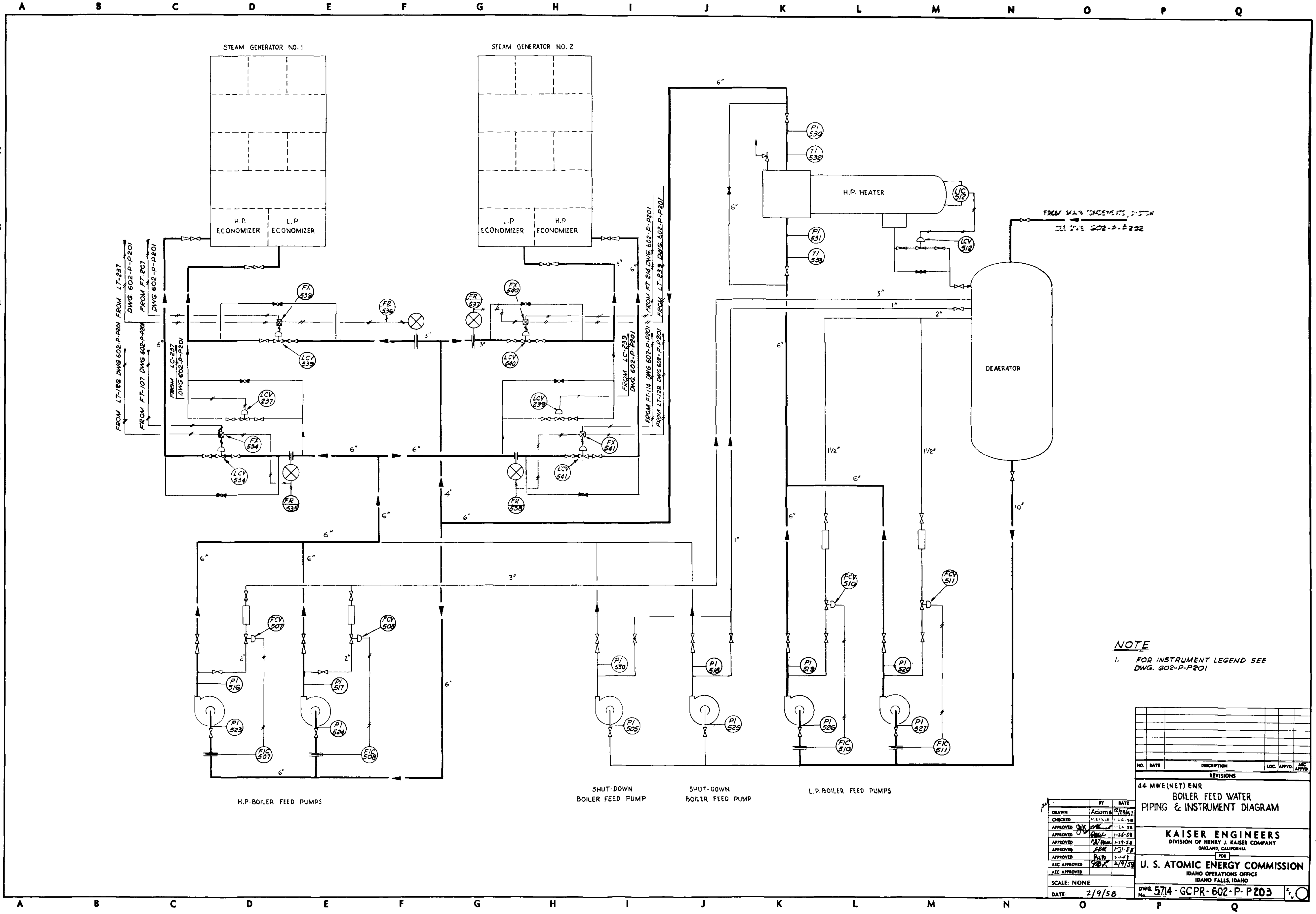
STEAM
PIPING & INSTRUMENT DIAGRAM

	BY	DATE
DRAWN	Adams	12/20/57
CHECKED		1-24-58
APPROVED		1-24-58
APPROVED		1-16-58
APPROVED		1-19-58
APPROVED		1-31-58
APPROVED		2-1-58
AEC APPROVED		2/9/58
AEC APPROVED		
SCALE:	NONE	
DATE:	2/9/58	

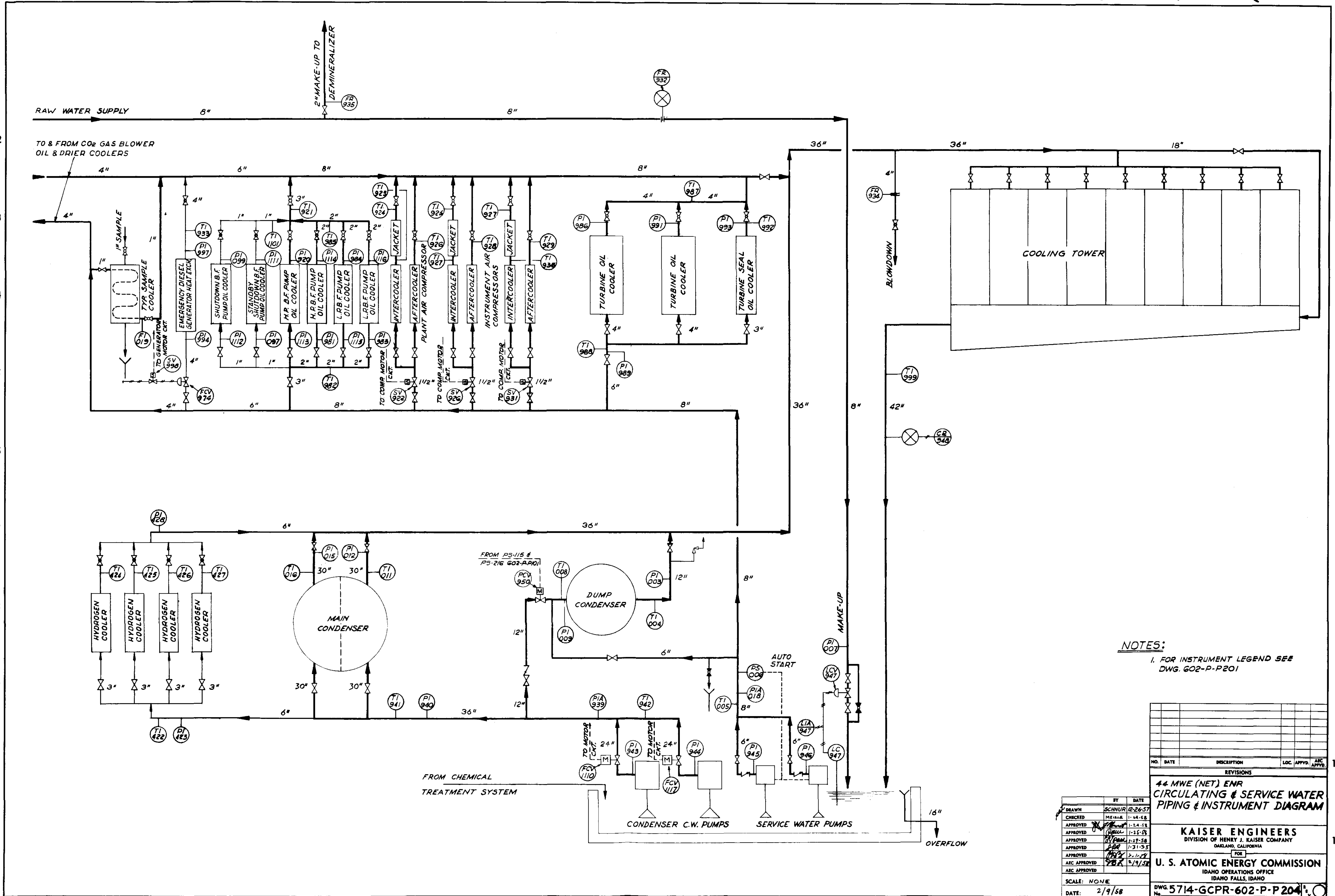
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-602-P-P201 REV. 0

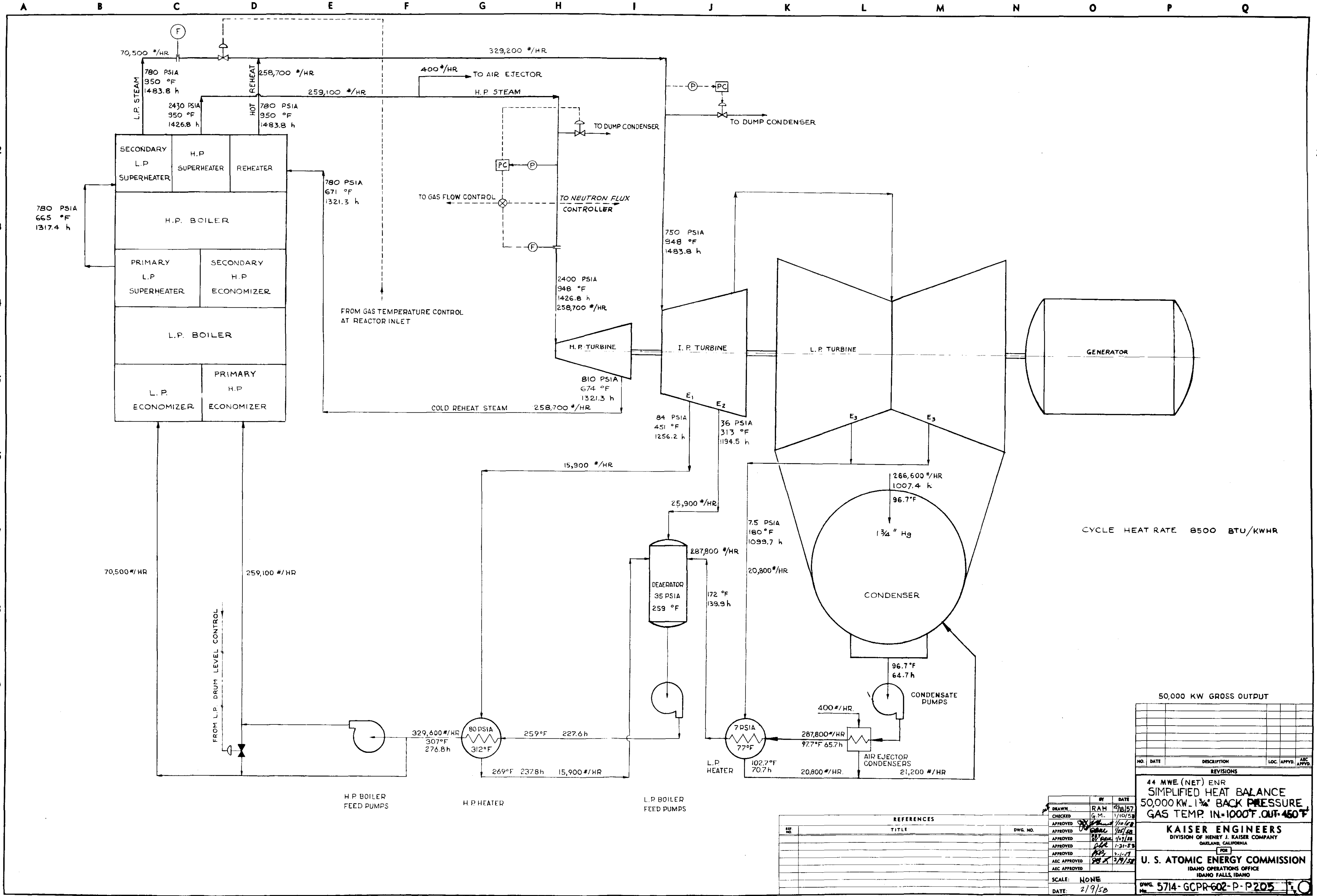

TO DESUPERHEATERS
SEE DWG. 602-P-P201
DUMP CONDENSER
LIC 506
LCV 506
RESTRICTION ORIFICE
FROM PS-115 & PS-216 DWG. 602-P-P201
PI 528
PI 529
PI 521
PI 522
CONDENSATE PUMP
CONDENSATE PUMP
6"
4"
PI 413
TI 415
L.P. HEATER
LIC 402
LAH 461
LCV 402
PI 414
TI 416
2"
DEAERATOR
PI 473
TI 474
LIC 403
LCV 403
TI 475
LAH 405
LAL 406
LR 404
LIC 404
10"
TO B.F. PUMP SUCTION
SEE DWG. 602-P-P203
TO BOILER DRAIN TANK
FCV 429
M
CONDENSER
ALS 401
HCV 401
MAKE-UP FROM DEMINERALIZER
CIA 421
AIR EJECTOR
LIC 409
LAH 410
LAL 411
PI 417
PI 418
CONDENSATE PUMP
CONDENSATE PUMP
TO GLAND SEAL TANK
FCV 412
FIC 412
LCV 409
VENT
CONDENSATE STORAGE TANK
LI 407
LR 407
LG 408
LCV 404A
LCV 404B
EMERGENCY DUMP LINE
NOTE
1. FOR INSTRUMENT LEGEND SEE DWG. 602-P-P201
NO. DATE DESCRIPTION LOC. APPVD. AEC APPVD.
REVISIONS
44 MWE (NET) ENR
CONDENSATE
PIPING & INSTRUMENT DIAGRAM
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DRAWN Adams 12/21/57
CHECKED
APPROVED
AEC APPROVED
SCALE: NONE
DATE: 2/9/58
DWG. No. 5714-GCPR-602-P-P202



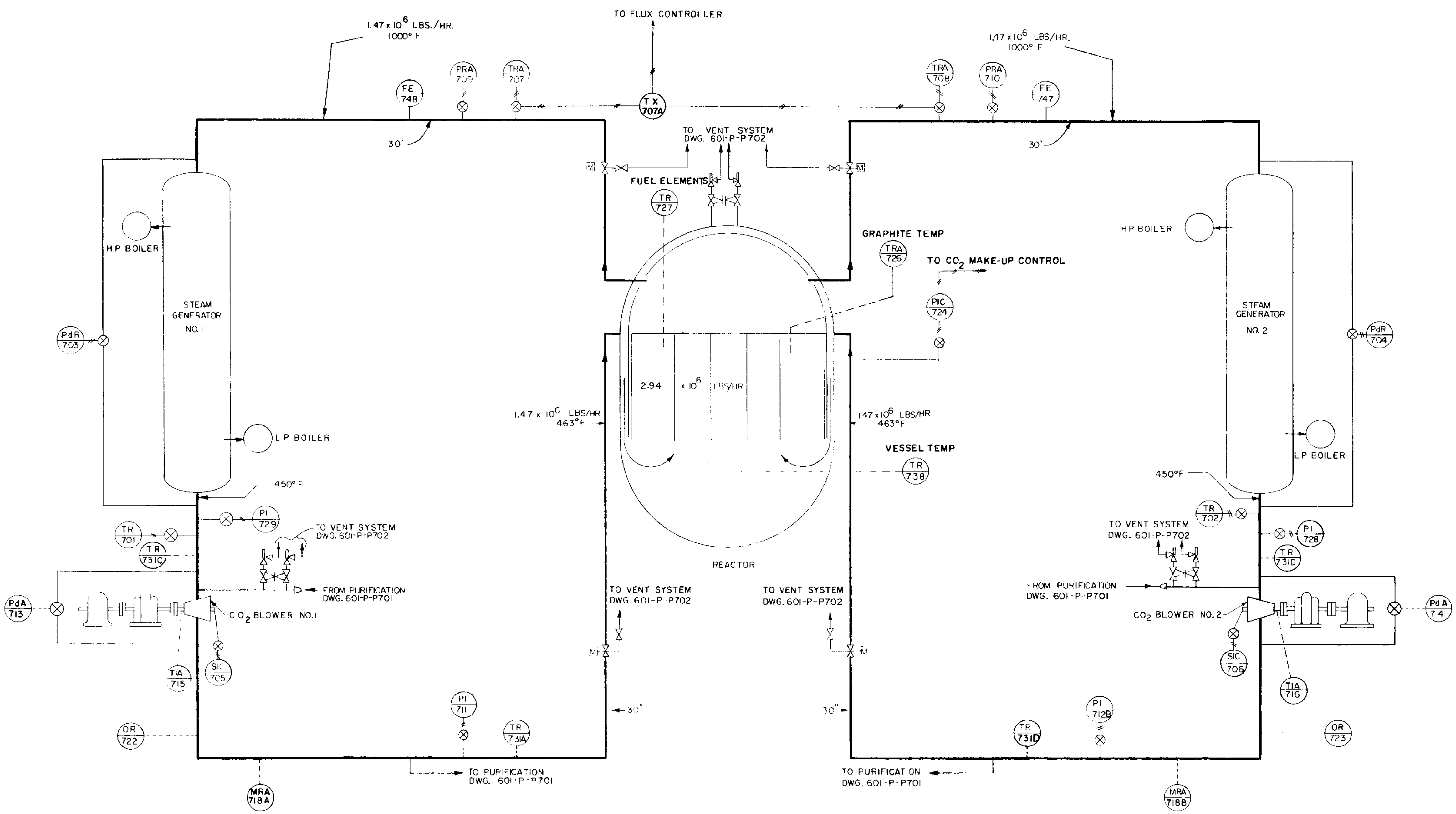
STEAM GENERATOR NO. 1
STEAM GENERATOR NO. 2
H.P. ECONOMIZER
L.P. ECONOMIZER
H.P. HEATER
DEAERATOR
H.P. BOILER FEED PUMPS
SHUT-DOWN BOILER FEED PUMP
SHUT-DOWN BOILER FEED PUMP
L.P. BOILER FEED PUMPS
FROM LT-237 DWG. 602-P-P201
FROM FT-207 DWG 602-P-P201
FROM LT-126 DWG 602-P-P201
FROM FT-107 DWG 602-P-P201
FROM LC-237 DWG 602-P-P201
FROM LC-239 DWG. 602-P-P201
FROM FT 214 DWG 602-P-P201
FROM FT-116 DWG 602-P-P201
FROM LT-128 DWG 602-P-P201
NOTE
1. FOR INSTRUMENT LEGEND SEE DWG. 602-P-P201
NO. | DATE | DESCRIPTION | LOC. | APPVD. | AEC APPVD.
REVISIONS
44 MWE(NET) ENR
BOILER FEED WATER
PIPING & INSTRUMENT DIAGRAM
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DRAWN Adams 12/23/57
CHECKED MEIKLE 1-24-58
APPROVED 1-24-58
APPROVED 1-25-58
APPROVED 1-29-58
APPROVED 1-31-58
APPROVED 2-1-58
AEC APPROVED 2/9/58
AEC APPROVED
SCALE: NONE
DATE: 2/9/58
DWG. No. 5714-GCPR-602-P-P203
REV. 0



RAW WATER SUPPLY
2" MAKE-UP TO DEMINERALIZER
TO & FROM CO2 GAS BLOWER OIL & DRIER COOLERS
1" SAMPLE
TYP. SAMPLE COOLER
EMERGENCY DIESEL GENERATOR HEAT EXCH.
SHUTDOWN B.F. PUMP OIL COOLER
STANDBY SHUTDOWN B.F. PUMP OIL COOLER
H.P. B.F. PUMP OIL COOLER
H.P.B.F. PUMP OIL COOLER
L.P.B.F. PUMP OIL COOLER
L.P.B.F. PUMP OIL COOLER
JACKET
INTERCOOLER
AFTERCOOLER
PLANT AIR COMPRESSOR
INSTRUMENT AIR COMPRESSORS
TO GENERATOR MOTOR CKT.
TO COMP. MOTOR CKT.
TURBINE OIL COOLER
TURBINE OIL COOLER
TURBINE SEAL OIL COOLER
BLOWDOWN
COOLING TOWER
HYDROGEN COOLER
MAIN CONDENSER
FROM PS-115 & PS-216 602-P-P101
DUMP CONDENSER
AUTO START
MAKE-UP
FROM CHEMICAL TREATMENT SYSTEM
TO MOTOR CKT.
CONDENSER C.W. PUMPS
SERVICE WATER PUMPS
OVERFLOW
NOTES:
1. FOR INSTRUMENT LEGEND SEE DWG. 602-P-P201
NO. DATE DESCRIPTION LOC. APPVD. AEC APPVD.
REVISIONS
44 MWE (NET) ENR
CIRCULATING & SERVICE WATER PIPING & INSTRUMENT DIAGRAM
DRAWN SCHNUR 12-26-57
CHECKED
APPROVED
AEC APPROVED
SCALE: NONE
DATE: 2/9/58
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DWG. No. 5714-GCPR-602-P-P204



70,500 #/HR
329,200 #/HR
780 PSIA
950 °F
1483.8 h
L.P. STEAM
HOT REHEAT
258,700 #/HR
400 #/HR
TO AIR EJECTOR
259,100 #/HR
H.P. STEAM
2430 PSIA
950 °F
1426.8 h
780 PSIA
950 °F
1483.8 h
TO DUMP CONDENSER
TO DUMP CONDENSER
SECONDARY L.P SUPERHEATER
H.P SUPERHEATER
REHEATER
780 PSIA
671 °F
1321.3 h
780 PSIA
665 °F
1317.4 h
H.P. BOILER
TO GAS FLOW CONTROL
TO NEUTRON FLUX CONTROLLER
750 PSIA
948 °F
1483.8 h
PRIMARY L.P SUPERHEATER
SECONDARY H.P ECONOMIZER
2400 PSIA
948 °F
1426.8 h
258,700 #/HR
FROM GAS TEMPERATURE CONTROL AT REACTOR INLET
L.P. BOILER
H.P. TURBINE
I. P. TURBINE
L. P. TURBINE
GENERATOR
L. P. ECONOMIZER
PRIMARY H.P ECONOMIZER
810 PSIA
674 °F
1321.3 h
COLD REHEAT STEAM
258,700 #/HR
84 PSIA
451 °F
1256.2 h
36 PSIA
313 °F
1194.5 h
15,900 #/HR
266,600 #/HR
1007.4 h
96.7°F
25,900 #/HR
7.5 PSIA
180 °F
1099.7 h
1 3/4" Hg
CYCLE HEAT RATE 8500 BTU/KWHR
70,500#/HR
259,100 #/HR
287,800 #/HR
DEAERATOR
35 PSIA
259 °F
20,800#/HR
172 °F
139.9 h
CONDENSER
FROM L.P. DRUM LEVEL CONTROL
96.7°F
64.7 h
CONDENSATE PUMPS
50,000 KW GROSS OUTPUT
400 #/HR.
329,600 #/HR
307°F
276.8h
80 PSIA
312°F
259°F
227.6h
7 PSIA
77°F
287,800 #/HR
97.7°F 65.7h
AIR EJECTOR CONDENSERS
269°F 237.8h
15,900 #/HR
L.P. HEATER
102.7°F
70.7h
20,800 #/HR
21,200 #/HR
H.P BOILER FEED PUMPS
H.P HEATER
L.P BOILER FEED PUMPS
REFERENCES
TITLE
DWG. NO.
REVISIONS
44 MWE (NET) ENR
SIMPLIFIED HEAT BALANCE
50,000 KW - 1 3/4" BACK PRESSURE
GAS TEMP. IN - 1000°F OUT - 450°F
DRAWN RAH 12/18/57
CHECKED G.M. 1/10/58
SCALE: NONE
DATE: 2/9/58
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U. S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DWG. No. 5714-GCPR-602-P-P205



NOTE

1 FOR INSTRUMENT LEGEND SEE DWG 602-P-P201

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

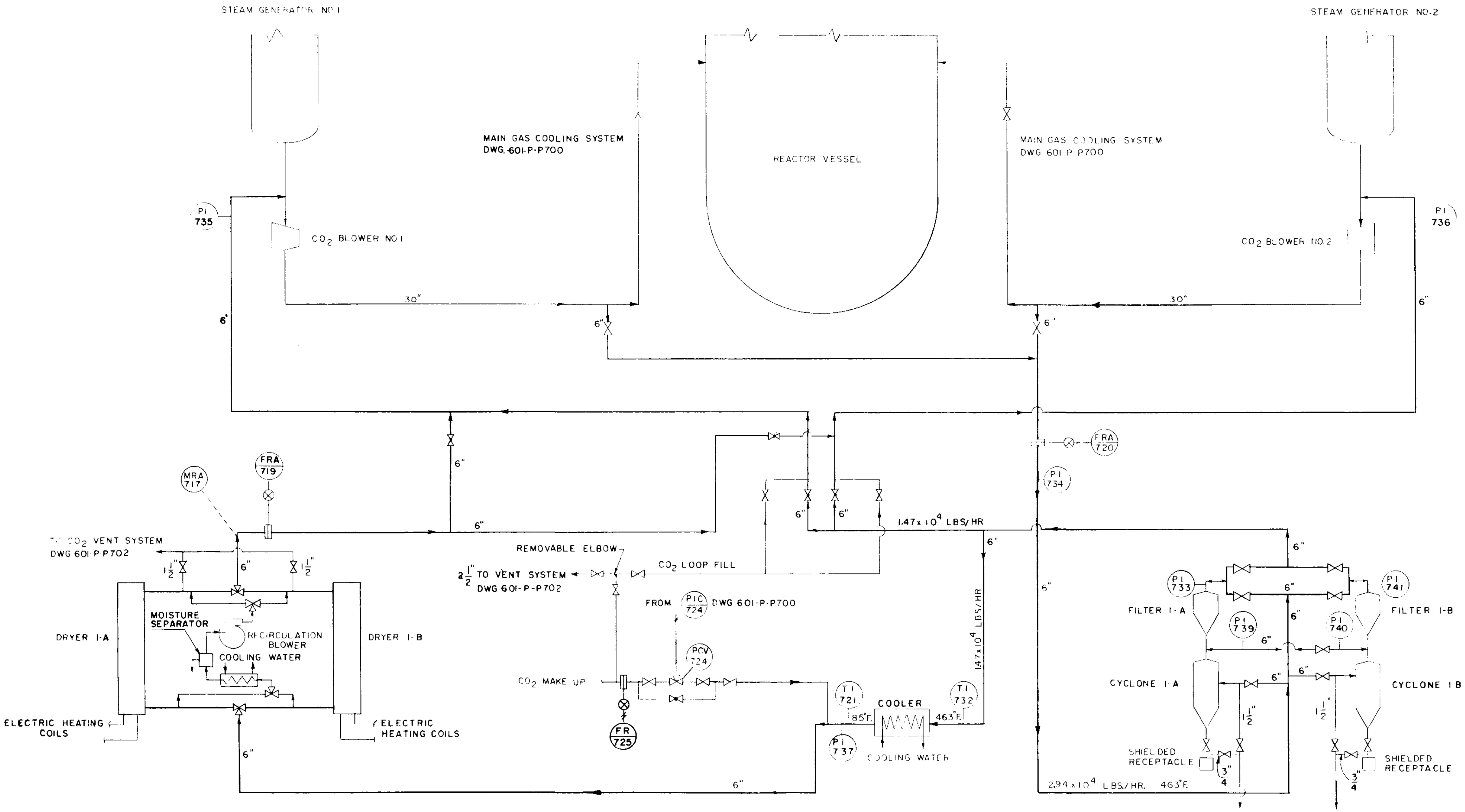
44 MWE (NET) ENR
MAIN GAS COOLING SYSTEM
FLOW DIAGRAM

	BY	DATE
DRAWN	KS	1-22-58
CHECKED		
APPROVED		
APPROVED		
APPROVED		
APPROVED		2/9/58
APPROVED		2/9/58
AEC APPROVED		2/9/58
AEC APPROVED		

SCALE: NONE
DATE: 2/9/58

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

DWG. No. 5714-GCPR-601-P-P700



NOTE

1. FOR INSTRUMENT LEGEND SEE DWG. 602-P-P201

NO.	DATE	DESCRIPTION	LOC.	APPVD.	AEC APPVD.

REVISIONS

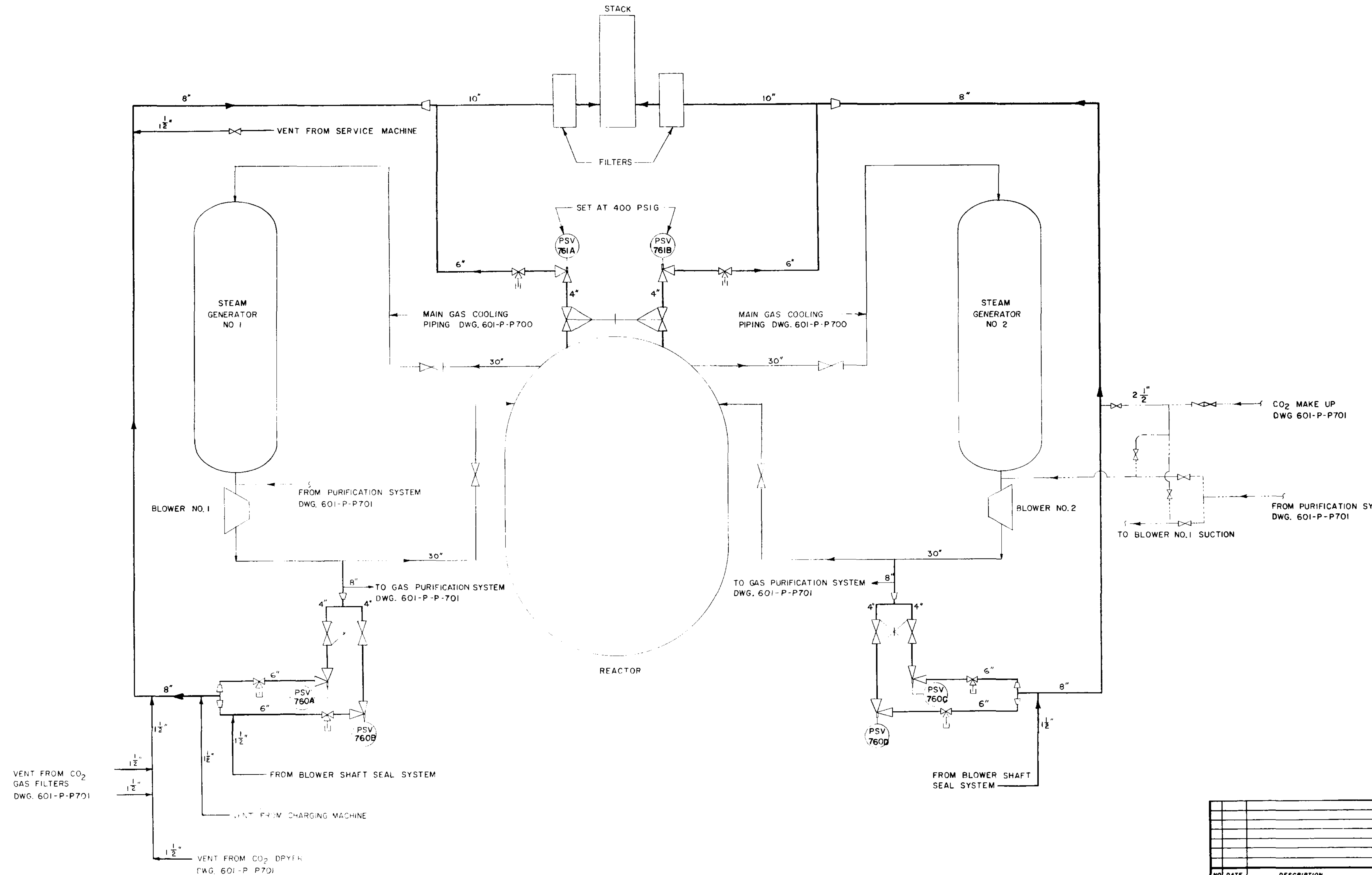
44 MWE (NET) ENR
GAS PURIFICATION SYSTEM
FLOW DIAGRAM

	BY	DATE
DRAWN	KS	1-23-58
CHECKED	RGS	1-31-58
APPROVED		1-31-58
APPROVED		2-6-58
APPROVED	WFB	2/7/58
APPROVED		2/9/58
APPROVED		2/9/58
AEC APPROVED		2/9/58
AEC APPROVED		

SCALE: NONE
DATE: 2/9/58

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

DWG. No. 5714-GCPR-601-P-P701 REV.



NOTE

1 FOR INSTRUMENT LEGEND SEE DWG 602-P-P201

NO	DATE	DESCRIPTION	LOC.	APPVD	AEC APPVD

REVISIONS

44 MWE (NET) ENR

GAS VENT SYSTEM

	BY	DATE
DRAWN	KS	1-24-58
CHECKED	[illegible]	1-31-58
APPROVED	[illegible]	1-4-58
APPROVED	[illegible]	1-6-58
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	2/9/58
AEC APPROVED	[illegible]	2/9/58
AEC APPROVED		
SCALE:	NONE	
DATE:	2/9/58	

NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
REACTOR ENGINEERING
FOR
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
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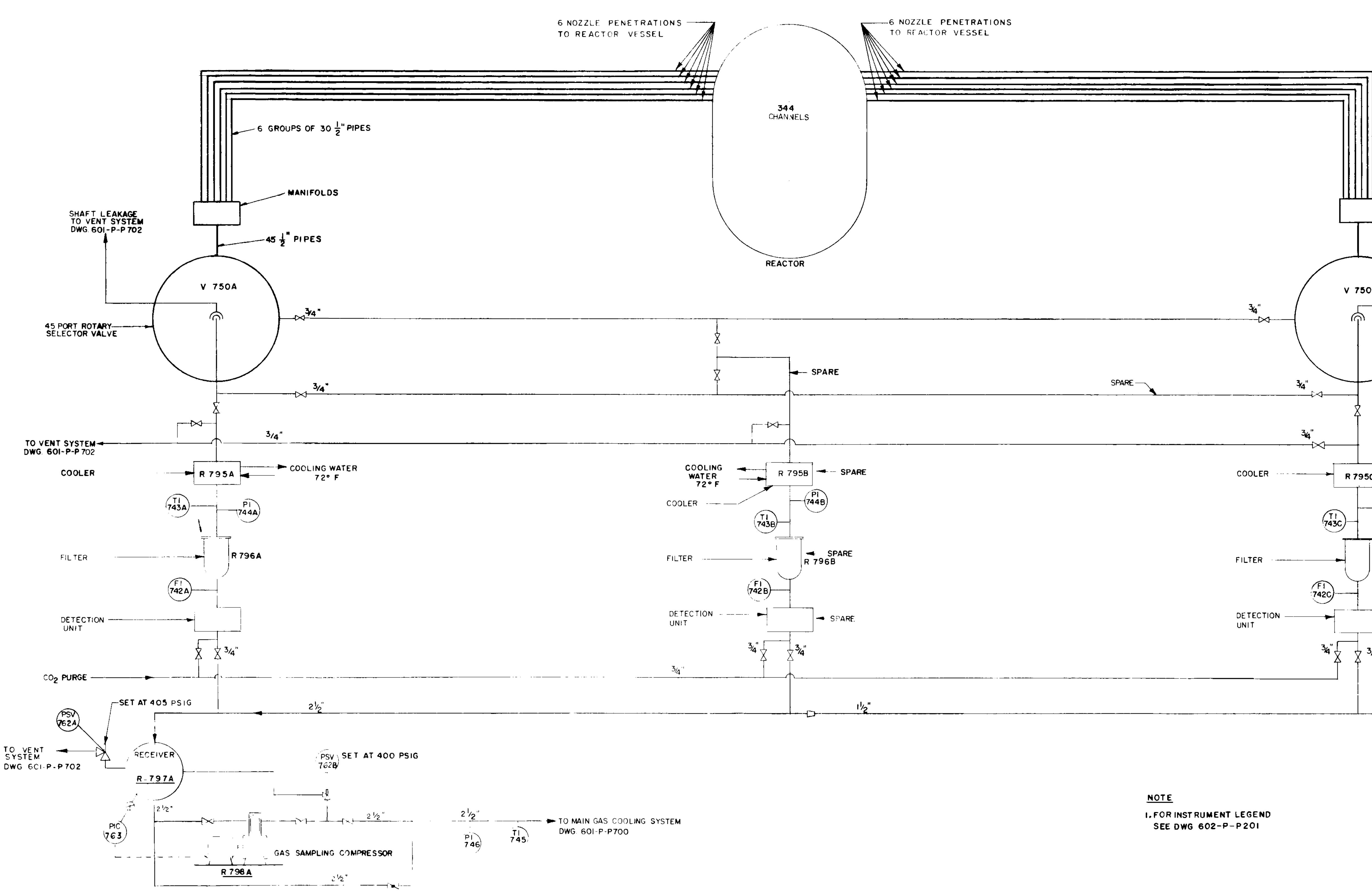
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-601-P-P702



NOTE

1. FOR INSTRUMENT LEGEND SEE DWG 602-P-P201

NO.	DATE	DESCRIPTION	LOC.	APPVD	AEC APPVD

REVISIONS

44 MWE (NET) ENR
BURST SLUG DETECTION SYSTEM
FLOW DIAGRAM

	BY	DATE
DRAWN	KS	1-26-58
CHECKED	RE.J	1-31-58
APPROVED	[illegible]	1-31-58
APPROVED	JWC	2-6-58
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	2-9-58
AEC APPROVED	[illegible]	2/9/58
AEC APPROVED		

SCALE: NONE
DATE: 2/9/58

NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
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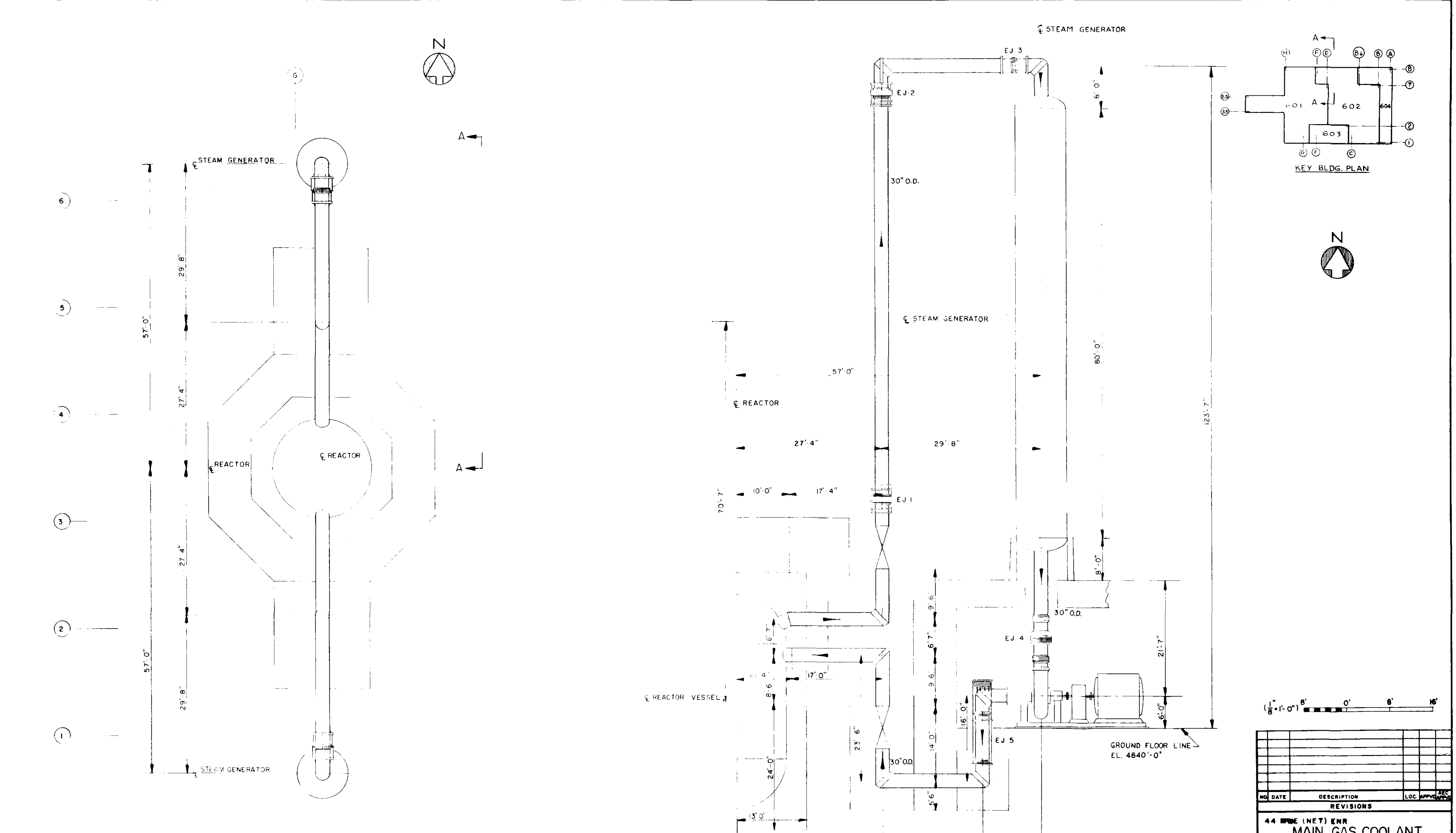
DWG. No. 5714-GCPR-601-P-P704 REV.



PLAN
SCALE $\frac{1}{8}$"=1'-0"

ELEVATION A-A
SCALE $\frac{1}{8}$"=1'-0"

NO.	DATE	DESCRIPTION	LOC.	APPVD	AEC APPVD

REVISIONS

44 MWE (NET) ENR
MAIN GAS COOLANT
PIPING
PLAN & ELEVATION

	BY	DATE
DRAWN	KS	1-29-58
CHECKED	RCJ	1-31-58
APPROVED		1-31-58
APPROVED	JJC	2-7-58
APPROVED		
APPROVED		
APPROVED		2-9-58
AEC APPROVED		2/9/58
AEC APPROVED		

SCALE: $\frac{1}{8}$"=1'-0"
DATE: 2/9/58

NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
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FOR
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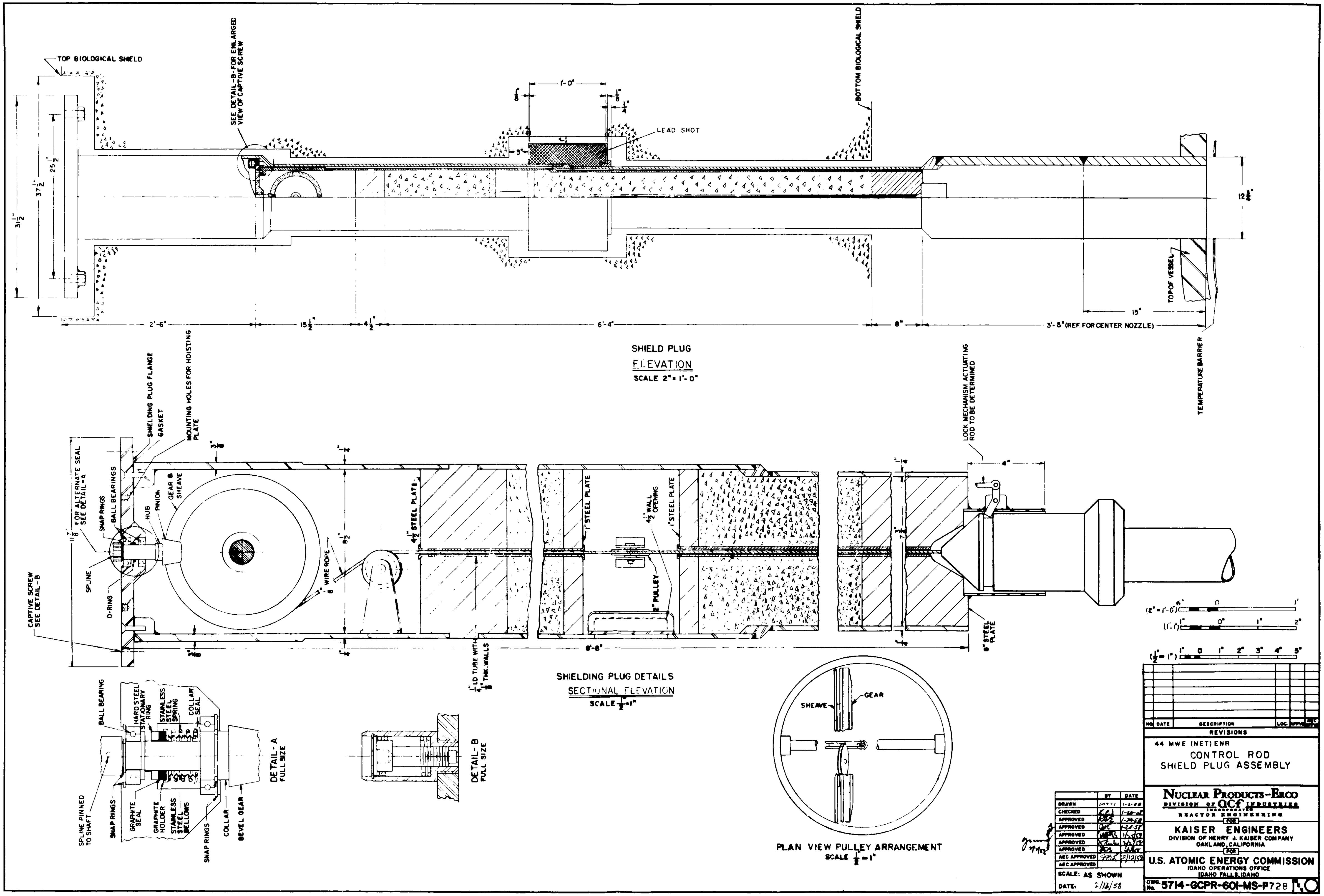
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-601-P-P705 REV. 0



TOP BIOLOGICAL SHIELD
SEE DETAIL-B FOR ENLARGED VIEW OF CAPTIVE SCREW
BOTTOM BIOLOGICAL SHIELD
LEAD SHOT
1'-0"
TOP OF VESSEL
TEMPERATURE BARRIER
2'-6"
6'-4"
8"
3'-8"(REF. FOR CENTER NOZZLE)
15"
SHIELD PLUG
ELEVATION
SCALE 2" = 1'-0"
SHIELDING PLUG FLANGE
GASKET
MOUNTING HOLES FOR HOISTING PLATE
FOR ALTERNATE SEAL SEE DETAIL-A
SNAP RINGS
BALL BEARINGS
HUB
PINION
GEAR & SHEAVE
SPLINE
O-RING
CAPTIVE SCREW SEE DETAIL-B
WIRE ROPE
STEEL PLATE
WALL OPENING
PULLEY
LOCK MECHANISM ACTUATING ROD TO BE DETERMINED
STEEL PLATE
I.D. TUBE WITH THK. WALLS
8'-8"
SHIELDING PLUG DETAILS
SECTIONAL ELEVATION
BALL BEARING
HARD STEEL STATIONARY RING
STAINLESS STEEL SPRING
COLLAR SEAL
SPLINE PINNED TO SHAFT
SNAP RINGS
GRAPHITE SEAL
GRAPHITE HOLDER
STAINLESS STEEL BELLOWS
SNAP RINGS
COLLAR
BEVEL GEAR
DETAIL-A FULL SIZE
DETAIL-B FULL SIZE
SHEAVE
GEAR
PLAN VIEW PULLEY ARRANGEMENT
NO. DATE DESCRIPTION LOC. APPD.
REVISIONS
44 MWE (NET) ENR
CONTROL ROD
SHIELD PLUG ASSEMBLY
DRAWN
CHECKED
APPROVED
AEC APPROVED
SCALE: AS SHOWN
DATE: 2/12/58
NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
REACTOR ENGINEERING
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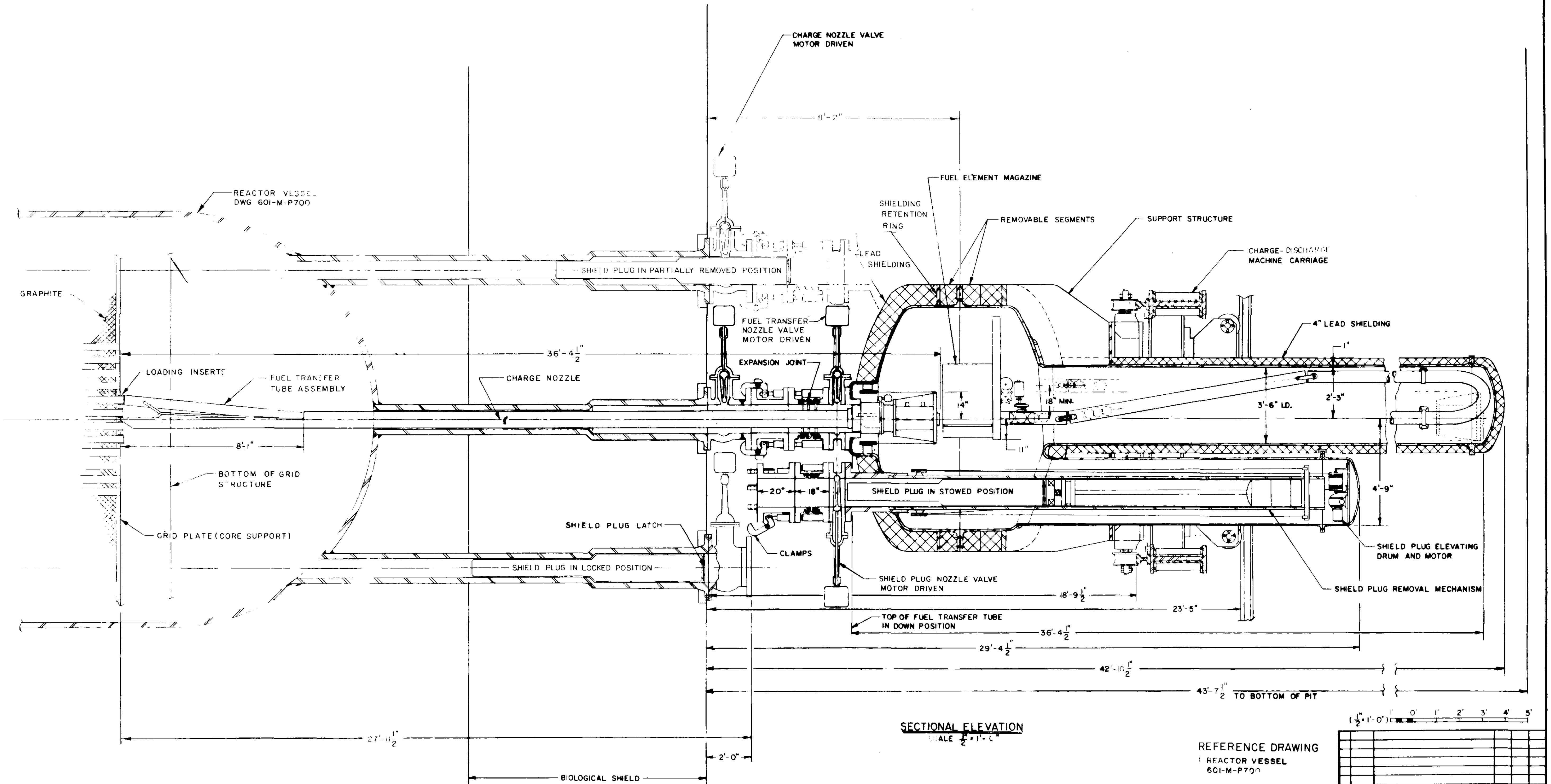
KAISER ENGINEERS
DIVISION OF HENRY J. KAISER COMPANY
OAKLAND, CALIFORNIA
FOR
U.S. ATOMIC ENERGY COMMISSION
IDAHO OPERATIONS OFFICE
IDAHO FALLS, IDAHO
DWG. No. 5714-GCPR-601-MS-P728



SECTIONAL ELEVATION
SCALE 1/2" = 1'-0"

REFERENCE DRAWING
1 REACTOR VESSEL
601-M-P700

NO.	DATE	DESCRIPTION	LOC.	APPD.	AEC APPD.

REVISIONS

44 MWE (NET) ENR
CHARGE MACHINE GENERAL ASSEMBLY

	BY	DATE
DRAWN	ROCZEN	1-23-58
CHECKED	R.C.S.	1-28-58
APPROVED	RAE	1-29-58
APPROVED	JWC	2-4-58
APPROVED	[illegible]	2/7/58
APPROVED	[illegible]	[illegible]
APPROVED	JER	2-9-58
AEC APPROVED	JMK	2/10/58
AEC APPROVED		

Young 2/19/58

SCALE: 1/2" = 1'-0"
DATE: 2/10/58

NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
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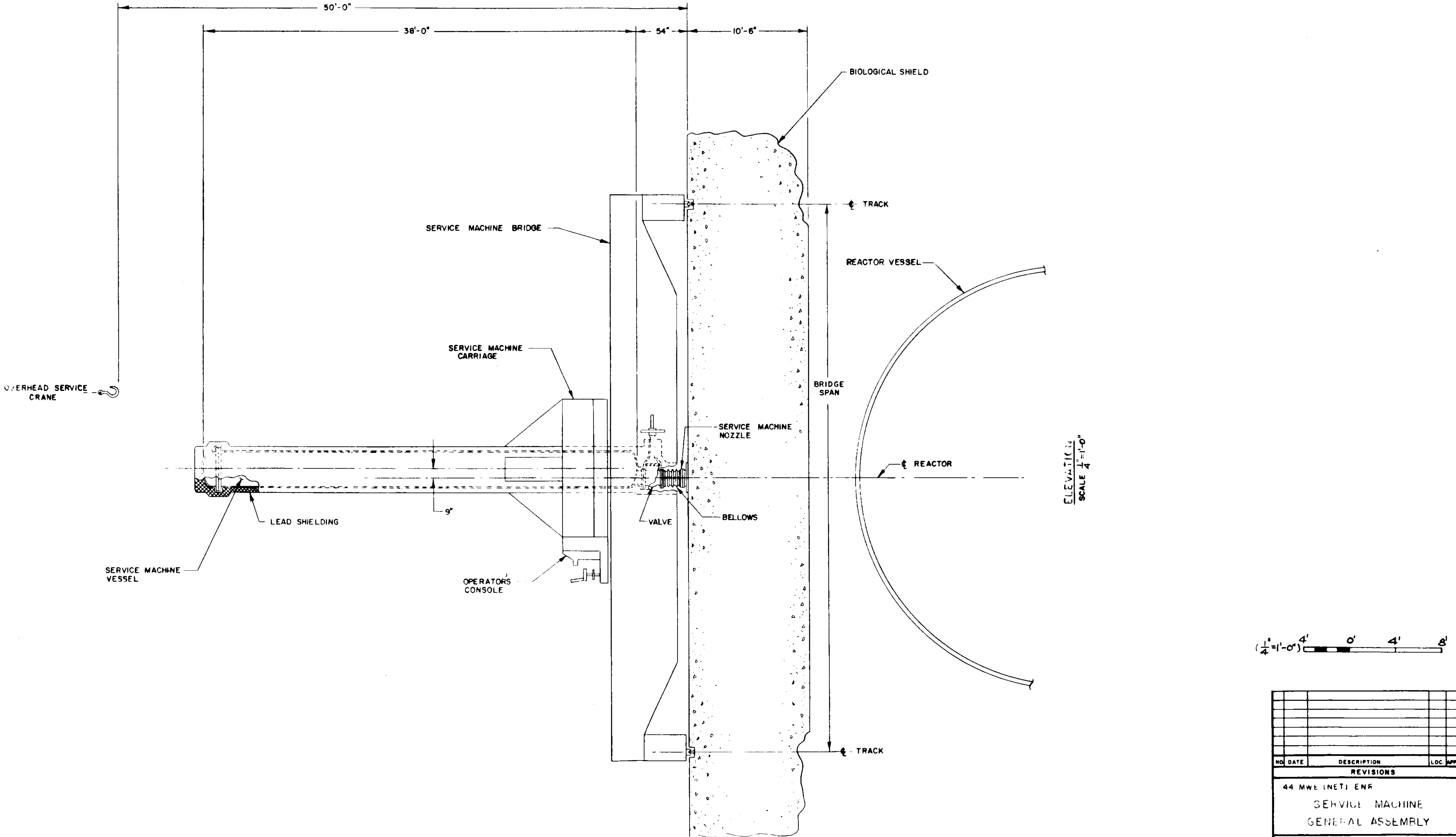
IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-601-MS-P741



NO.	DATE	DESCRIPTION	LOC.	APPVD	AEC APPVD

REVISIONS

44 MWE (NET) ENR

SERVICE MACHINE
GENERAL ASSEMBLY

	BY	DATE
DRAWN	CARLE	12-20-57
CHECKED	R.C.[illegible]	[illegible]
APPROVED	[illegible]	1-25-58
APPROVED	[illegible]	1-25-58
APPROVED	[illegible]	1/25/58
APPROVED	[illegible]	[illegible]
APPROVED	[illegible]	2/3/58
AEC APPROVED	[illegible]	[illegible]
AEC APPROVED		

SCALE: 1/4" = 1'-0"
DATE: 2/12/58

NUCLEAR PRODUCTS-ERCO
DIVISION OF ACF INDUSTRIES INCORPORATED
REACTOR ENGINEERING
FOR
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IDAHO FALLS, IDAHO

DWG. No. 5714-GCPR-601-MS-P771 REV. O