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DESCRIPTION AND SAFETY ANALYSIS
FOR A CONCEPTUAL UNIT
AT INDIAN POINT

VOLUME II

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KARL R. KENNISON
CIVIL AND HYDRAULIC ENGINEER
361 CLINTON AVE., BROOKLYN, N. Y.

Mr. G. R. Milne
Mechanical Engineer
Cons. Edison Co. of N. Y.
4 Irving Place
New York 3, N. Y.

Nov. 18, 1955

Dear Sir :

You have described to me the general features of the atomic-energy power plant which you are planning to construct on the east bank of the Hudson River below Peekskill. I understand that you wish me to report on such hydrologic features of the site as may affect your plans.

From the information that you have made available to me I conclude that the most useful information I can give you is that which relates to the amount and character of the flow in the river. At the proposed site the river has a width of about 4500 to 5000 feet, a maximum depth of 55 to 75 feet at less than 1000 feet off shore, and a cross-sectional area of about 165,000 to 170,000 square feet. Sheet 1 shows a number of cross sections of the river, plotted from the U.S.C.&G.S. charts, at intervals of 1500 feet, from 3750 feet upstream to 5250 feet downstream from the proposed plant.

At this site the effect of the tides is all important and so far outweighs any other consideration that, at least for present purposes, the information already available on the day-by-day variation of the runoff from the tributary watershed is adequate.

On Sheet 2 I have plotted an approximate flow-duration curve from data I had already calculated covering a period of

Gem 11/22

17 years.

An average rate of about 26000 cfs may be expected to be exceeded 20 % of the time
" " " " " 15250 " " " 40 % " " "
" " " " " 10500 " " " 60 % " " "
" " " " " 7000 " " " 80 % " " "
For say 2 % of the time the rate may be as low as 4000 cfs

However as above indicated the ebb and flow of the tide is the all important consideration. The river is tidal to as far upstream as Troy. Its hourly behavior in the tidal range varies throughout its length. The U. S. Coast & Geodetic Survey has tabulated a great deal of information from which a general picture of conditions off the shore at the proposed site can be obtained.

On Sheet 3 I have plotted the data, as they are applicable to this particular site. This indicates that the elevation of the water surface is so responsive to the tidal cycle that the average rate of flow, or runoff from the tributary watershed, has relatively little effect on the velocity past the site. I conclude that it is this velocity and the resulting volume of flow available for mixing and dilution in which you are primarily interested. In the limited time at my disposal I can only draw general conclusions. These may be adequate for present purposes. You could obtain better information by running a series of tests on surface and sub-surface floats, at varying distances off shore, throughout the tidal cycle.

The velocity recorded by the U.S.C.&G.S. is that in midstream at or near the surface. In order to be on the safe side in drawing conclusions, I have assumed that 80 % of this velocity represents the average vertically from surface to bottom, and that 80 % also represents the average horizontally from side to

2.4 Land Usage and Population

The land usage within a 5 and 15 mile radius of the Indian Point Station is shown on four maps included in this Section. Areas are designated which are served by public utilities and sewage systems, showing location of water supplies, and the land usage.

The population within a 5 and 15 mile radius of the plant site is shown in Tables I, II, and III. These data are based on the 1960 census and are projected to 1985 based on increases proportional to those estimated by the Regional Plan Association in Bulletin 100, September, 1962.

Regional Plan is a non-profit civic organization supported by 1500 businesses, governments and individuals. Special research projects are sponsored by Regional and National Foundations. The region served by the Association consists of 22 counties in New Jersey, New York and Connecticut within roughly a 50 mile arc from Times Square, encompassing more than 1400 governmental units. Through Research and Information Programs, the Association provides governments and profit organizations in the region with data to make decisions on the way land is to be used and on the transportation system to serve it. A development plan on New York and Environs, fostered by the Association since 1929, has now been achieved in most of its fundamentals - in the network of radial and circumferential expressways of the Region, in regional parks, metropolitan airport and port development, and in the substantial growth of effective municipal and county planning. The Association is constructing new guidelines for metropolitan development to 1985, based upon several years of intensive research and with the advice and consultation of regional leaders.

The 100th Bulletin of the Regional Plan Association sketches the way the New York Metropolitan Region will look in 1985 if present economic trends, popular taste and public and corporate policies continue. The economic trends are based on the New York Metropolitan Region Study ("Harvard Study") which was conducted for the Regional Plan by the Harvard Graduate School of Public Administration under the direction of Raymond Vernon.

TABLE I

POPULATION DATA
FROM 1960 CENSUS REPORTS

Civil Subdivision	Population within 5 miles	Population within 15 miles
Westchester County		
Bedford (Town)		8,050
Mt. Kisco (Village)		6,805
Cortlandt (Town)	14,587	17,505
Buchanan (Village)	2,019	2,019
Croton-on-Hudson (Village)	592	6,812
Greenburgh (Town)		1,267
Tarrytown (Village)		7,510
Lewisboro (Town)		417
Mt. Pleasant (Town)		16,670
Briarcliff Manor (Village)		5,105
No. Tarrytown (Village)		8,818
Pleasantville (Village)		5,877
New Castle (Town)		10,163
Mt. Kisco (Village)		6,805
North Castle (Town)		680
Ossining (Town)		2,967
Briarcliff Manor (Village)		5,105
Ossining (Village)		18,662
Peekskill (City)	18,737	18,737
Somers (Town)		4,921
Yorktown (Town)		16,453
TOTAL	35,935	171,348
Putnam County		
Carmel (Town)		5,468
Mahopac (Unincorporated)		1,337
Kent (Town)		654
Philipstown (Town)	986	4,932
Cold Spring (Village)		2,083
Nelsonville (Village)		555
Putnam Valley (Town)		3,070
TOTAL	986	18,099
Dutchess County		
Fishkill (Town)		1,181
TOTAL		1,181

TABLE I (Continued)

POPULATION DATA
FROM 1960 CENSUS REPORTS

Civil Subdivision	Population within 5 miles	Population within 15 miles
Rockland County		
Clarkstown (Town)		32,487
Upper Nyack (Village)		1,833
Haverstraw (Town)	6,769	16,632
Orangetown (Town)		15,649
Nyack (Village)		6,062
So. Nyack (Village)		3,113
Ramapo (Town)		19,797
Hillburn (Village)		279
Sloatsburg (Village)		1,923
Sp. Valley (Village)		6,538
Suffern (Village)		3,821
Stony Point (Town)	5,482	8,739
TOTAL	12,251	116,873
Orange County		
Blooming Grove (Town)		2,518
Washingtonville (Village)		393
Chester (Town)		349
Cornwall (Town)		8,094
Cornwall (Village)		2,785
Cornwall Southwest (Unincorporated)		2,824
Highlands (Town)	1,865	11,990
Highland Falls (Village)		4,469
Monroe (Town)		5,965
Harriman (Village)		718
Monroe (Village)		3,323
New Windsor (Town)		1,985
New Windsor (Unincorporated)		4,041
Tuxedo (Town)		2,227
Tuxedo Park (Village)		723
Woodbury (Town)		2,887
Harriman (Village)		34
TOTAL	1,865	55,325

TABLE II

PREDICTED POPULATION

County	Population within 5 miles		Population within 15 miles	
	1965	1980	1965	1980
Westchester	39,533	51,970	188,504	247,809
Putnam	1,233	3,389	22,624	62,215
Dutchess			1,342	2,047
Rockland	15,201	30,404	145,025	290,050
Orange	2,078	3,142	61,639	93,211
Bergen			106	127
TOTAL	58,045	88,905	419,240	695,459

TABLE III

POPULATION DATA AND PREDICTED POPULATION

Population within 5 miles

County	1960	1965	1970	1975	1980	1985
Westchester	35,935	39,533	44,197	48,195	51,970	54,857
Putnam	986	1,233	1,849	2,465	3,389	4,160
Rockland	12,251	15,201	19,087	27,721	30,404	31,298
Orange	1,865	2,078	2,382	2,737	3,142	4,004
TOTAL	51,037	58,045	67,515	81,118	88,905	94,319

Population within 15 miles

County	1960	1965	1970	1975	1980	1985
Westchester	171,348	188,504	210,743	229,805	247,809	261,576
Putnam	18,099	22,624	33,936	45,247	62,215	76,355
Dutchess	1,181	1,342	1,543	1,778	2,047	2,349
Rockland	116,873	145,025	209,006	264,457	290,050	298,581
Orange	55,325	61,639	70,660	81,183	93,211	118,768
Bergen	92	106	116	124	127	131
TOTAL	362,918	419,240	526,004	622,594	695,459	757,760

1960 figures for Westchester, Putnam, Dutchess, Rockland, and Orange Counties are from 1960 Census Reports.

1960 figures for Bergen County are based on a count of 23 residences within the area concerned, as shown on U. S. Geological Survey Park Ridge 7½' quadrangle, revised 1955, and multiplying 23 by 4 to obtain 92.

Figures for later years are based on increases proportional to those estimated for each county in Regional Plan Association Bulletin 100, September 1962, "Spread City". No distinction is made between increases in presently urban areas and increases in presently rural areas.

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2.5 Hydrology

The study on hydrology of Mr. Karl R. Kennison is included. The engineering staff of Consolidated Edison has reviewed its content and has found it still appropriate in 1965, with the exception of the second paragraph on Page 4. In this paragraph, Mr. Kennison states that "there is no likelihood that in the future any nearer municipality (than Poughkeepsie, 30 miles or more upstream) will take its domestic water supply from the Hudson." This paragraph becomes obsolete since in August 1965, New York City authorities have decided to reactivate the Chelsea pumping station, located 25 miles upstream from Indian Point.

Maps included in this application show the location of water supplies within a radius of 15 miles from the site. Further description of water supplies within a 20 mile radius can be found in the report "Environmental Factors to be Considered After an Accidental Release of Radioactivity From the Consolidated Edison Thorium Reactor" of the Division of Environmental Health Services of the New York State Department of Health, April, 1962.

In addition, Consolidated Edison has contracted the consultant firm of Metcalf and Eddy to prepare an extensive hydrological study of the site.

2.6 Environmental Radioactivity

Environmental surveys of radioactivity in the vicinity of Indian Point Station were begun in 1958, about four years before Unit #1 began operation, and have continued as a regular testing program which is reported semi-annually to the AEC, Docket 50-3. The reports contain tabulations and charts of radioactivity in soil, water, vegetation, marine life, and air particulate samples collected during the period of almost eight years. The 1958 report describes the sampling equipment, sample preparations, and counting methods employed in the program.

Each report lists the sampling points for the reporting period and locates these on a map. The present environmental program includes continuous sampling of air particulate and fallout, Hudson River water, and gross gamma background on the Indian Point site. Surface water from a small lake on the site and drinking water from the Indian Point tap and nearby reservoirs is sampled weekly. Monthly samples are taken from the Hudson River up and downstream from the site, vegetation on the site, marine life from the river, and water from Indian Point well. Vegetation and soil samples are taken seasonably and a gross gamma background survey on roads in the station vicinity is made annually.

The measurements have shown that operation of Indian Point Unit No. 1 for more than three years has had no perceptible affect on the plant environment. During the past few years there have been minor changes in the radioactive background, but these have been due to fallout from nuclear weapons tests, and not from Unit #1.

side, hence that roughly 64 % represents the average over the entire cross section. I have also assumed that 15 % of the total cross section, or a stretch about five or six hundred feet wide off shore, is all that should be used in considering the initial mixing or diluting effect. In making this assumption I am governed to some extent by Hazen's studies relative to the off-shore distance of Poughkeepsie's water intake to avoid direct contamination by its sewage. I have further assumed that the velocity in this off-shore stretch is only 60% of the midstream velocity, hence that roughly 48% represents the average over the cross section of this off-shore stretch.

On Sheet 4 I have shown the result of these assumptions, which, as above stated, are believed to be on the safe side in considering the direct effect of mixing or dilution of your wastes. This emphasizes the all-important effect of the tides, the quantity available for dilution varying in about three hours from a maximum of eight or ten million gallons per minute to nothing.

Although you will have to put up with this variation as far as your continuous cooling water circulation is concerned, it does point to the desirability of incorporating in your design a method of controlling the time for the discharge into the cooling water outlet of any and all waste that is to any extent radioactive. I would say that this should be done in any event for the drainage from your routine and emergency demineralizers, and it might well be done also for drainage from all areas liable to accidental contamination.

From your estimate of the extent of dilution already

3.0 FACILITY DESCRIPTION

3.1 Nuclear Steam Supply System

The Indian Point No. 2 Plant is designed to produce 2,758 MWt of heat and 916 MW of gross electrical power. Components of the turbine-generator plant, however, have been designed to permit generation of larger amounts of electrical power, if the nuclear steam supply system, in the future, should prove to have adequate margin for such operation. The nuclear steam supply system, consisting of a pressurized light-water cooled and moderated reactor and its attendant heat transfer equipment, is housed within the reactor containment and supported on the foundation mat.

The system contains four parallel coolant loops connected to the reactor vessel, each loop containing a steam generator and a reactor coolant pump. Steam from the four steam generators is admitted to the double flow, high pressure turbine cylinder and exhausts through a moisture separator-reheater, at the exit of which the steam contains approximately 120°F of superheat. The reheated steam is then admitted to the double flow, low pressure turbine cylinders, after which the steam is exhausted to three condensers, one for each low pressure section. Condensate is returned to the steam generators through six stages of regenerative feed-water heating.

The reactor is fueled with slightly enriched uranium dioxide fuel in the form of pressed and sintered pellets 0.366 in. in diameter. The pellets are stacked in 0.422 in. OD full length Zircaloy tubes to form an active core, 12 ft. in height. Fuel rods are assembled into a square 15 x 15 lattice with a center to center distance of 0.563 in. to make a fuel assembly. The fuel assembly design has a total of 225 fuel rod positions but 21 selected fuel rods are omitted from this pattern to permit the installation of 20 control rod guide thimbles which accept the movable control rod cluster and one instrumentation location. Lateral support and spacing for the fuel rods and control rod guide thimbles are obtained by use of spring clip grid assemblies welded to the control rod guide thimbles and spaced approximately

one foot apart. Fuel rods are inserted through holes in the grids, and then end plates and nozzles are secured to the structure to form the fuel assembly.

One hundred and ninty-three fuel assemblies are loaded vertically upon the core support structure to form a uniformly spaced, cylindrical rod lattice core with an equivalent diameter of 11.06 ft. The total uranium dioxide mass is approximately 189,000 lbs. Use of three U-235 fuel enrichments is planned.

The fuel elements are arranged in a three region, symmetrical pattern with the lowest enrichment in the center region and the highest enrichment in the outer region, thereby limiting fuel shuffling at refueling periods which are expected to occur every 12 to 14 months. Use of an inward loading arrangement results in calculated average equilibrium fuel burnups of approximately 27,000 MWD per MTU at the time of fuel removal from the core.

The core is controlled by a combination of neutron absorbing control rod clusters and a soluble neutron poison. Each cluster consists of 20 cylindrical control rods fastened together as a unit by means of a spider-like bracket. The control rods are fabricated of a silver-indium-cadmium alloy and sealed in stainless steel tubes. Each control rod cluster is actuated by a separate drive mechanism mounted on the reactor vessel head. Exact numbers and distribution of control rod clusters will be established during later design. It is planned that long-term regulation of core reactivity will be accomplished by adjustment of soluble poison concentration in the reactor coolant system, with short-term control for power changes or trip provided for by movement of control rod clusters. The main thermal-hydraulic criterion for core design is the specifying of a safety margin between the conditions which cause departure from nucleate boiling and the normal operating condition. This safety margin is expressed as the DNB ratio and the reactor control and protection system is designed to provide actuation of automatic reactor trip, to prevent routine plant transient from producing a DNB ratio lower than 1.30. (W-3 correlation)

The Reactor Coolant System consists of the reactor vessel and four closed loops connected in parallel to the reactor vessel. The principal components of each loop are a steam generator, a circulating pump, loop piping and instrumentation. A pressurizer is connected to one of the loops. Auxiliary system piping connections are provided as necessary. The Reactor Coolant System flow diagram is shown in Figure 3.1-1.

During full power operation, coolant enters the reactor vessel at a temperature of 548°F and leaves at a temperature of 600.4°F. Coolant flow is approximately 136.4 million lb/hr.

Steam is produced in the steam generators at a pressure of 725 psia, dry and saturated. Maximum moisture at the outlet is 1/4 of 1 per cent.

The Reactor Coolant System and its auxiliaries are designed to accommodate 10 per cent step changes in plant load and 5 per cent per minute ramp changes over the range from 15 to 100 per cent of full power subject to Xenon limitations. Based on experience however, the greater capabilities are to be expected as the operating conditions would probably not be as pesimistic as those used for the design basis. In addition, the turbine by-pass system will accept an instantaneous 40 per cent loss of rated load without reactor trip by steam dump to the condenser. The Reactor Coolant System will accept a complete loss of load from full power without reactor trip and without damage to the plant.

In designing Reactor Coolant System components, code requirements are considered minimum design objectives. Design pressures and temperatures exceed operating pressures and temperatures by an ample margin. The entire system is hydrostatically tested before initial operation.

The reactor vessel is cylindrical in shape with a hemispherical bottom and a hemispherical removable head. A view of a typical vessel and arrangement of the internals is shown in Figure 3.1-2. The body of the vessel is low alloy steel and the initial NDTT of less than 40°F is the designed target for the reactor vessel.

The inside surfaces in contact with coolant are clad with approximately 5/32 inch Type 304 austenitic stainless steel. The vessel is designed and manufactured in accordance with the ASME Nuclear Vessel Code Section III.

Reactor Coolant System piping is 27½ and 29 in. ID stainless steel pipe with 33 in. ID sections between the steam generators and the pumps. Exact dimensions of the pipe and mating nozzles on the components will be established in the detailed design.

Piping is designed and fabricated in accordance with ASA B31.1 Code for Pressure Piping, Section 1.

Instrument taps and auxiliary system connections are located for accessibility. They are provided with thermal sleeves, where necessary, to limit thermal stress and shock. The piping and other Reactor Coolant System components are insulated to minimize heat losses.

Each loop contains a vertical single stage centrifugal pump of the controlled leakage type.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, and a secondary seal which directs the controlled leakage out of the pump. A vapor seal minimizes the leakage of vapor from the pump into the containment atmosphere.

The controlled leakage seal is free-running, not a friction type, and some small leakage through the primary seal is accepted. This leakage is, however, collected and removed from the pump with very little escape of either liquid or vapor to the reactor containment. The water passing through the seal is supplied by an auxiliary system and is comparatively cool, crud free and uncontaminated.

High pressure water is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a

buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the secondary seal is also collected and removed from the pump.

Each loop contains a vertical shell and U-tube steam generator. The steam generators are designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The U-tubes are Inconel and all other surfaces in contact with reactor coolant are clad with stainless steel.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell, then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly. The vanes impart a centrifugal motion to the mixture and separate the water particles from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another passage through the steam generator.

Two primary components, a pressurizer and a pressurizer relief tank, are used to maintain the required reactor coolant pressure during steady state operation, to limit the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevent the pressure in the Reactor Coolant System from exceeding the design pressure.

Under normal operating conditions, about one-half of the pressurizer volume is occupied by water and the other half by steam. The pressurizer also contains replaceable direct immersion heaters, safety and relief valves, a spray nozzle, and the interconnecting piping, valves, and instrumentation. The electric heaters located in the lower section of the vessel pressurize the Reactor Coolant System by

keeping the water and steam in the pressurizer at saturation temperature. The heaters are capable of raising the temperature of the pressurizer and contents at the desired rate during start-up of the reactor plant.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line to the pressurizer comes from the hot leg of a reactor coolant loop. During a positive surge caused by a decrease in plant load, the spray nozzle, which is fed from the cold leg of a coolant loop, condenses steam in the vessel so that the pressure will not increase to a value that would actuate the power operated relief valves. Power operated spray valves on the pressurizer limit the pressure during most load transients. In addition, the spray valves can be operated manually by a switch in the main control room. A small continuous purge spray is provided into the pressurizer to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray piping.

During a negative surge caused by an increase in plant load, flashing of water to steam and generation of steam by automatic actuation of the heaters keep the pressure above the minimum allowable. Heaters will also be energized on high level during positive surges to heat the subcooled surge water entering the pressurizer.

Two power operated relief valves and three code safety valves are provided to protect against large pressure surges which are beyond the pressure limiting capacity of the pressurizer spray. The relief valves operate to limit coolant pressure to 2350 psia. The safety valves prevent system pressure from exceeding code limitations. The relief valves are set at a lower pressure in order to minimize operation of the safety valves. The relief valves may also be operated remotely from the central control room. Each relief valve has a remotely operated isolation valve.

Steam and water discharged from the safety and relief valves pass to the pressurizer relief tank which is partially filled with water at or near ambient containment vessel temperature. The cool water condenses the discharged steam and the condensate is drained to the Waste Disposal System.

The pressurizer relief tank also provides a means for removing any non-condensable gases from the Reactor Coolant System which collect in the pressurizer vessel. A remotely operated vent valve is provided to periodically vent gases to the relief tank and thence to the Waste Disposal System.

The principal design parameters of the reactor core and coolant system are given in Table 3.1-1.

TABLE 3.1-1

Principal Core & Coolant System Parameters

REACTOR CORE	
General	
Total Heat Output, Megawatts	2,758
Heat Generated in Fuel, per cent	97.4
Heat Transfer	
Active Heat Transfer Surface Area, sq.ft.	52,200
Average Heat Flux, Btu/hr-sq.ft.	175,600
Maximum Heat Flux, Btu/hr-sq.ft.	570,800
Maximum Thermal Output, Steady State kw/ft	18.5
Nominal Conditions DNB Ratio W-3 Correlation	1.81
Fuel Rod (Cold Dimensions)	
Outside Diameter, in	0.422
Clad Thickness, in	0.0243
Clad Material	Zircaloy
Diametral Gap, in	0.0074
Pellet Diameter, in	0.366
Fuel Length, (Pellets Only) in	144
Pitch, in	0.563
Rod Array in Assembly	15 x 15
Rods per assemblies	204
Total Number Fuel Rods	39,372
Equivalent Core Diameter, ft	11.06
Core Length, Between Fuel Ends, ft	12.0
Fuel Weight, Uranium, lb	189,800

TABLE 3.1-1 Continued

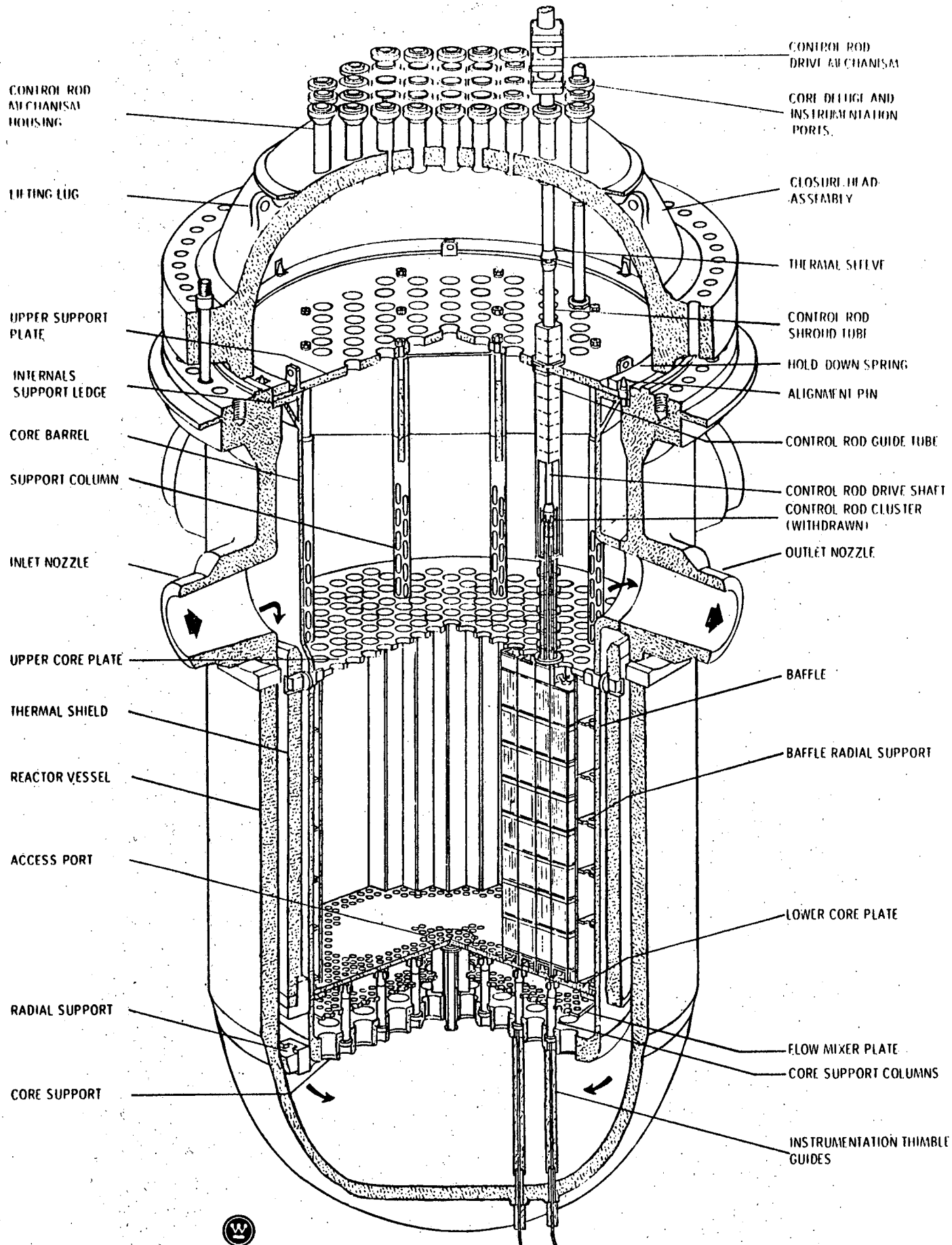
Reactor Coolant System	
Number of loops	4
Nominal operating pressure, psia	2,250
Design pressure, psia	2,500
Design temperature, °F	650
Reactor inlet temperature, °F	548
Reactor outlet temperature, °F	600.4
Total reactor coolant flow, lb. per hr.	136.4×10^6
Total reactor coolant system coolant volume (including pressurizer), cu.ft.	12,209
Expected rate of heat-up and cooldown, °F per hr.	50-100
Reactor Vessel	
Material ASE 302 GRB	Low alloy steel, 304 stainless steel clad internally
Overall length of vessel and enclosure head (including mechanism adapter) ft.-in	42-4
Inside diameter of shell, in	173
Diameter across nozzles, in	245
Nominal clad thickness, in	5/32
Weights, lb. Closure head including mechanism adapters	175,000
Vessel shell, including bottom head	700,000
Reactor Coolant Pumps	
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge
Design capacity, each gpm	90,500
Design head, ft.	280
Operating pressure, psia	2,250
Suction temperature, °F	548
Motor type	AC, induction, single speed

TABLE 3.1-1 Continued

Steam Generators	
Type	Vertical U-tube with integral steam drum
Operating pressure, tube side, psia	2,250
Design pressure, tube side, psia	2,500
Tube material	Inconel
Coolant flow, each, lb. per hr.	34.1×10^6
Coolant inlet temperature, °F	600.4
Coolant outlet temperature, °F	548.0
Shell side full load pressure at outlet, psia	725
Full load steam temperature at outlet, °F	507
Full load steam flow each, lbs. per hr.	2.94×10^6
Maximum moisture at outlet (full load, per cent)	1/4
Shell side design pressure, psia	1,100
Pressure Control Components	
Pressurizer	
Operating pressure, psia	2,250
Design pressure, psia	2,500
Design temperature, °F	650
Material of construction	Carbon Steel, stainless steel clad internally
Overall height, ft.	45
Inside diameter, in.	83
Total internal volume, cu.ft.	1,600
Water volume, normal operation, cu.ft.	800
Steam volume, normal operation, cu.ft.	800
Type of heaters	Direct immersion
Pressurizer Relief Tank	
Total volume, cu.ft.	1,500
Normal water volume, cu.ft.	700

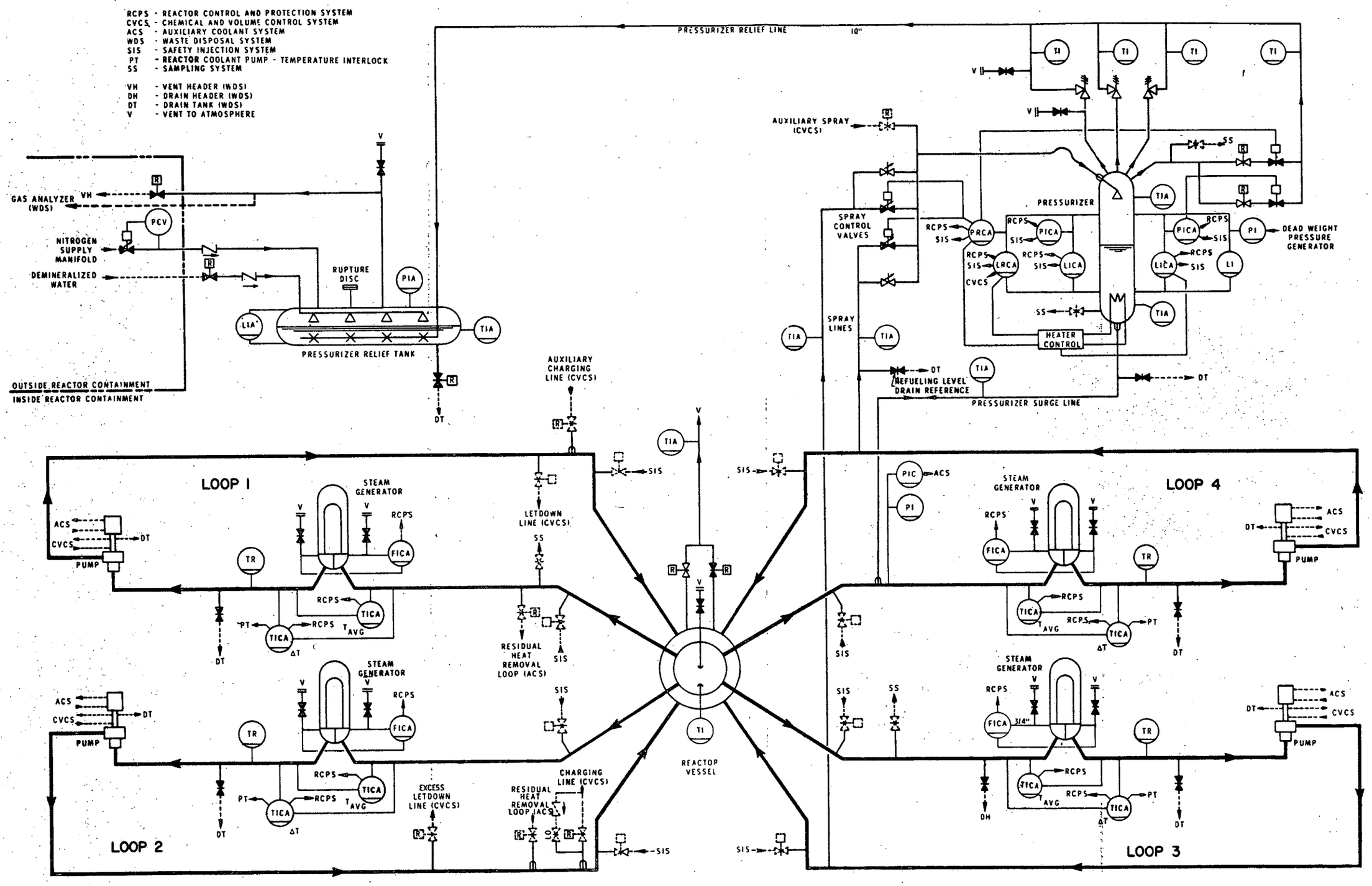
TABLE 3.1-1 Continued

Normal water temperature, °F	120
Design pressure, psia	115



REACTOR VESSEL AND INTERNALS

FIG. 3.1-2



Reactor Coolant System -- FIG.3.1-1
E.D.SK. 319386-F

3.2 AUXILIARY SYSTEMS

In addition to the reactor coolant system described in Section 3.1, auxiliary systems are provided for the operation of the plant. Description of these systems and their functions are given below.

3.2.1 Chemical and Volume Control System

The chemical and volume control system maintains the proper water inventory in the reactor coolant system, adjusts the concentration of chemical neutron absorber, reduces the concentration of fission products and corrosion products in the reactor coolant, maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and provides the seal water circuit for the reactor coolant pumps. The system is also used to fill and hydrostatically test the reactor coolant system.

The chemical and volume control system flow diagram is shown in Figure 3.2-1. The regenerative heat exchanger, the excess letdown heat exchanger, and the charging and letdown line isolation valves are located within the reactor containment. The mixed bed demineralizers, the spent fuel pit demineralizer and the reactor coolant filter are located in the ion exchanger pit. The letdown orifices are located in the pipe tunnel immediately outside the reactor containment. All other system equipment is located in the primary auxiliary building. This system also utilizes the deborating demineralizers of the waste disposal system.

During plant operation, reactor coolant is drawn from the cold leg of loop 1 of the reactor coolant system and returned to the cold leg of loop 2. Each of these connections to the reactor coolant system is provided with an isolation valve located close to the loop piping. In addition, a check valve is provided upstream of the charging line isolation valve.

Reactor coolant entering the chemical and volume control system flows through the shell side of the regenerative heat exchanger where its temperature

is reduced and then flows out of the reactor containment into the pipe tunnel where its pressure is reduced by the letdown orifices. The cooled, low pressure water then enters the primary auxiliary building where it undergoes a second temperature reduction in the tube side of the non-regenerative heat exchanger followed by a second pressure reduction by a low pressure letdown valve. After passing through one of the mixed bed demineralizers, the coolant flows through the reactor coolant filter and enters the volume control tank through a spray nozzle. The vapor space in the volume control tank contains hydrogen which dissolves in the coolant. Fission gases present are stripped from the coolant and removed from the system by venting the volume control tank to the waste disposal system prior to a cold or refueling shutdown.

From the volume control tank, coolant flows to the charging pumps which raise its pressure above that in the reactor coolant system. Coolant then flows from the primary auxiliary building to the reactor containment along two parallel paths. One path returns to the reactor coolant system through the tube side of the regenerative heat exchanger. Coolant following the other path either returns to the reactor coolant system through the reactor coolant pump labyrinth seals, or it returns to the volume control tank through the reactor coolant pump controlled leakage seals, the seal water filter and the seal water heat exchanger.

During normal operation, letdown flow through the regenerative heat exchanger exceeds charging flow by the quantity of coolant injected through the reactor coolant pump labyrinth seals. The low pressure letdown valve automatically maintains sufficient pressure in the letdown piping to prevent flashing of coolant downstream of the letdown orifices. If for some reason, the normal letdown and charging path through the regenerative heat exchanger is not operable, water injected through the reactor coolant pump labyrinth seals is returned to the volume control tank through the excess letdown heat exchanger.

Makeup to the reactor coolant system is provided by the chemical and volume control system from the following sources:

a. The demineralized water supply provides demineralized water when the concentration of boron is to be reduced.

b. A boric acid tank supplies a concentrated boric acid solution when an increase is required in the concentration of boron or for shutdown operations.

c. Demineralized water and concentrated boric acid are blended to match the reactor coolant boron concentration for normal plant makeup.

d. The refueling water storage tank supplies borated water for emergency makeup.

e. The chemical addition tank supplies small quantities of hydrazine or pH control solutions. Solutions are flushed out of the tank with demineralized water.

Concentrated boric acid solution used for leakage makeup or for increasing the reactor coolant boron concentration can be pumped from the boric acid tank into the volume control tank or charging pump suction header by either or both boric acid pumps.

3.2.2 Auxiliary Coolant System

The auxiliary coolant system provides cooling for the following heat sources:

1. The reactor coolant pump thermal barriers and bearings (reactor coolant system)
2. The sample heat exchangers (sampling system)
3. The reactor coolant letdown flow (chemical and volume control system)

4. The seal water heat exchanger (chemical and volume control system)
5. The reactor coolant system (residual and sensible heat during plant shutdown)
6. The spent fuel pit (residual heat)
7. The recirculation flow (residual and sensible heat in the event of a loss-of-coolant accident)

The system consists of three loops: 1) the component cooling loop; 2) the residual heat removal loop; and 3) the spent fuel pit cooling loop. The process flow diagrams for the system are shown in Figures 3.2-2 and 3.2-3.

Component Cooling Loop

The component cooling loop is an intermediate barrier between the reactor coolant and the river water. The component cooling water flows through the reactor plant heat sources and then through two component cooling heat exchangers where the heat is transferred to the river water. This double barrier arrangement reduces the probability of any leakage of radioactivity into the river cooling water.

Component cooling water removes heat from the following:

1. Reactor coolant pump bearings and thermal barriers
2. The sample heat exchangers
3. The excess letdown and non-regenerative heat exchangers
4. The reactor coolant pump seal water heat exchanger
5. The spent fuel pit heat exchanger
6. The residual heat exchangers
7. Residual heat removal pumps

The component coolant flows from the component cooling pumps, through the shell side of the component cooling heat exchangers, through the components being cooled, and back to the pumps. River water flows through the tubes. The surge tank is connected to the suction side of the component cooling pumps. Makeup

water is supplied to the loop near the surge tank. All components in the component cooling loop are located within the primary auxiliary building with the exception of the cooling lines to the excess letdown heat exchanger and the reactor coolant pumps.

During normal full power operation, one component cooling pump and one component cooling heat exchanger accommodate the heat removal loads. The standby heat exchanger provides 100 per cent backup during normal operation. Operation of both component cooling heat exchangers is required for removing residual and sensible heat during a normal plant shutdown. Failure of one of these components increases the time required to shut down the plant but does not affect the safe operation of the plant. The surge tank accommodates surges resulting from component coolant thermal expansion and contraction and accommodates water which may leak into the system from components which are being cooled. The surge tank also contains sufficient water to insure continuous component cooling water supply until a leaking cooling line can be isolated.

Residual Heat Removal Loop

The normal functions of the residual heat removal loop are to: 1) remove residual heat from the reactor core during shutdown; 2) reduce the temperature of the reactor coolant system during the latter stages of plant cooldown; and 3) transfer water from the refueling canal back to the refueling water storage tank after a refueling operation. Additional functions of the loop after a loss-of-coolant accident are: 1) automatically supply water to the safety injection system (core deluge); and 2) cool and recirculate through the residual heat exchangers and back to the reactor coolant system, water which has spilled into the reactor pump.

During plant shutdown, reactor coolant flows from the reactor coolant system to the residual heat removal pumps, through the tube side of the residual

heat exchangers and back to the reactor coolant system. The inlet line to the residual heat removal loop starts at the hot leg of loop 1 and the return line connects to the cold leg of loop 2. Both heat exchangers are used to cool the water circulated during the latter phase of safety injection system operation. The heat loads are transferred by the residual heat exchangers to the component cooling loop. The heat exchangers and pumps are located in the primary auxiliary building.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel stored in the spent fuel pit. The pump draws water from the pit, circulates it through the heat exchanger, and returns it to the pit. The heat exchanger is cooled by the component cooling loop. A portion of the flow passes through a demineralizer and filter as required to maintain the spent fuel pit water purity.

To maintain a clean water surface, a spent fuel pit skimmer loop consisting of a skimmer, a strainer, a pump, and a filter is provided. The pump suction is at the water surface. Demineralization or cooling is not accomplished in this loop.

3.2.3 Sampling System

The sampling system is designed to provide fluid samples for laboratory analysis to evaluate the reactor coolant chemistry and radiochemistry conditions.

Evaluation of the analysis provides guidance in the operation of the reactor coolant system and the chemical and volume control system. Typical of the analysis performed on such samples are reactor coolant boron concentrations, fission product radioactivity levels, dissolved gas content, and corrosion product concentration. The sampling system is operated manually, on an intermittent basis.

The sampling system shown in Figure 3.2-4 includes the following equipment:

1. One delay coil
2. Two sample heat exchangers
3. One sample vessel station
4. One sampling sink
5. Associated piping, valves and instrumentation

The delay coil and several sample lines with valves are located in the reactor containment. Other system equipment is located in the sampling room, and in the radioactive valve room.

Two types of samples are obtained: high temperature, high pressure reactor coolant system samples from inside the reactor containment; and low temperature, low pressure samples from the chemical and volume control system. The reactor coolant system samples are:

1. Hot legs of two loops
2. Pressurizer steam
3. Pressurizer water

In addition hot samples can be taken from the residual heat removal loop in the auxiliary coolant system.

Chemical and volume control system samples and their points of origin are:

a. A demineralizer inlet sample taken just downstream of the low pressure letdown valve.
b. A demineralizer outlet sample taken just downstream of the demineralizers.
c. A make-up solution sample taken just downstream of the boric acid blender.
d. A gas sample taken from the volume control tank gas space.

All sample lines terminate in a hooded sampling sink located in the hot laboratory in the primary auxiliary building.

3.2.4 Disposal System

The waste disposal system consists of equipment to receive, process, store and dispose of radioactive wastes from the reactor systems. This equipment is located in the auxiliary building with the exception of the large storage tanks for waste and reclaimed liquid which are located outside.

Figure 3.2-5 shows the provisions made for collecting and treating radioactive wastes.

The principal radioactive wastes are the fission products from the reactor core and the corrosion products in the reactor coolant water. Boron dilution liquid is also handled by the waste disposal facilities.

Liquid Waste

The major source of radioactive liquid waste evolves during the process of decreasing the reactor coolant boron concentration during plant startup and during full power operation. At high boron concentrations, waste liquid is handled by concentration and reuse and at low boron concentrations the boron is removed by ion exchange. Radioactive water discharged from the reactor coolant system and its auxiliary systems flows to one of the stainless steel waste holdup tanks. A high tank level starts one of the gas stripper feed pumps which transfer the liquid to the stripper evaporator train.

Provisions are made to transfer spent ion exchange resin flush water as well as water from the spent fuel pit and the refueling cavity water storage tank to the waste holdup tank. The quantity of water disposed of from these sources is small.

Floor drainage, samples exposed to air, and drainage generally not considered to be radioactive are delivered from the sump tank to the spent regenerant chemicals holdup tank for sampling and analysis. Liquids having a suitably low activity are pumped from this tank and released to the Hudson River by mixing with

the condenser circulating water discharge. After mixing, the condenser circulating water discharge activity levels will be below 10CFR20 limits.

Radioactive liquid wastes are processed in an evaporator gas stripper train which concentrates the fission and corrosion products in the bottoms and strips the hydrogen and radioactive fission product gases from the liquid. The evaporate concentrate is pumped through ion exchange beds and a filter and stored for reuse in the chemical and volume control system.

Spent resins are stored for decay in the spent resin storage tanks prior to drumming for shipment. When necessary, drums are stored on site until decay has reduced the activity to meet ICC requirements for common carrier shipment. Water recovered from the evaporator stripper train is pumped to the monitor tanks where it is sampled and analyzed to determine acceptability for re-use as reactor make-up or discharged to the condenser circulating water discharge.

The evaporator capacity is sufficient to process all liquid expected during normal power transients. The waste holdup tanks, together with the deborating ion exchangers, provide sufficient liquid storage capacity for two consecutive startups and shutdowns during most of plant life.

Volatile Waste

Hydrogen gas and radioactive fission product gases, such as xenon and krypton, are removed from the reactor coolant in the stripper evaporator train during normal operation as a result of feed and bleed operations. Smaller quantities of hydrogen and fission product gases are accumulated from radioactive system and waste tank vents.

The evaporator waste holdup and other wast disposal tanks are connected to gas compressors which pump into gas decay tanks. After an appropriate decay period, and when meteorological conditions are suitable, gases are diluted with air monitored fro activity level and discharged to the atmosphere through the

gas vent pipe.

Solid Wastes

Combustible solids, such as removable floor covering cloths used for decontamination and contaminated paper, are compressed by a hydraulic baler. The bales are subsequently shipped off-site for disposal. If shielding of the baled waste is necessary, the bale is placed in a 55 gal. steel drum and concrete poured around the bale. The concrete is allowed to solidify before the drum is shipped.

Spent ion exchanger resins are enclosed in concrete shipping cases for disposal.

Gaseous Discharges

Discharges of radioactive gaseous wastes from the plant to the environment are made via a vent system located on top of the reactor containment building. A preliminary evaluation has been performed to estimate the maximum off-site exposures which might result from the expected maximum releases from the plant.

The evaluation assumed an elevated release but a conservative model was used to incorporate the effects of down draft in the building wake in reducing the large dispersion normally associated with elevated stack releases. The results of the evaluation have shown that even under the most adverse conditions, the maximum expected off-site concentrations are below the limits of 10CFR20.

Fuel Handling System

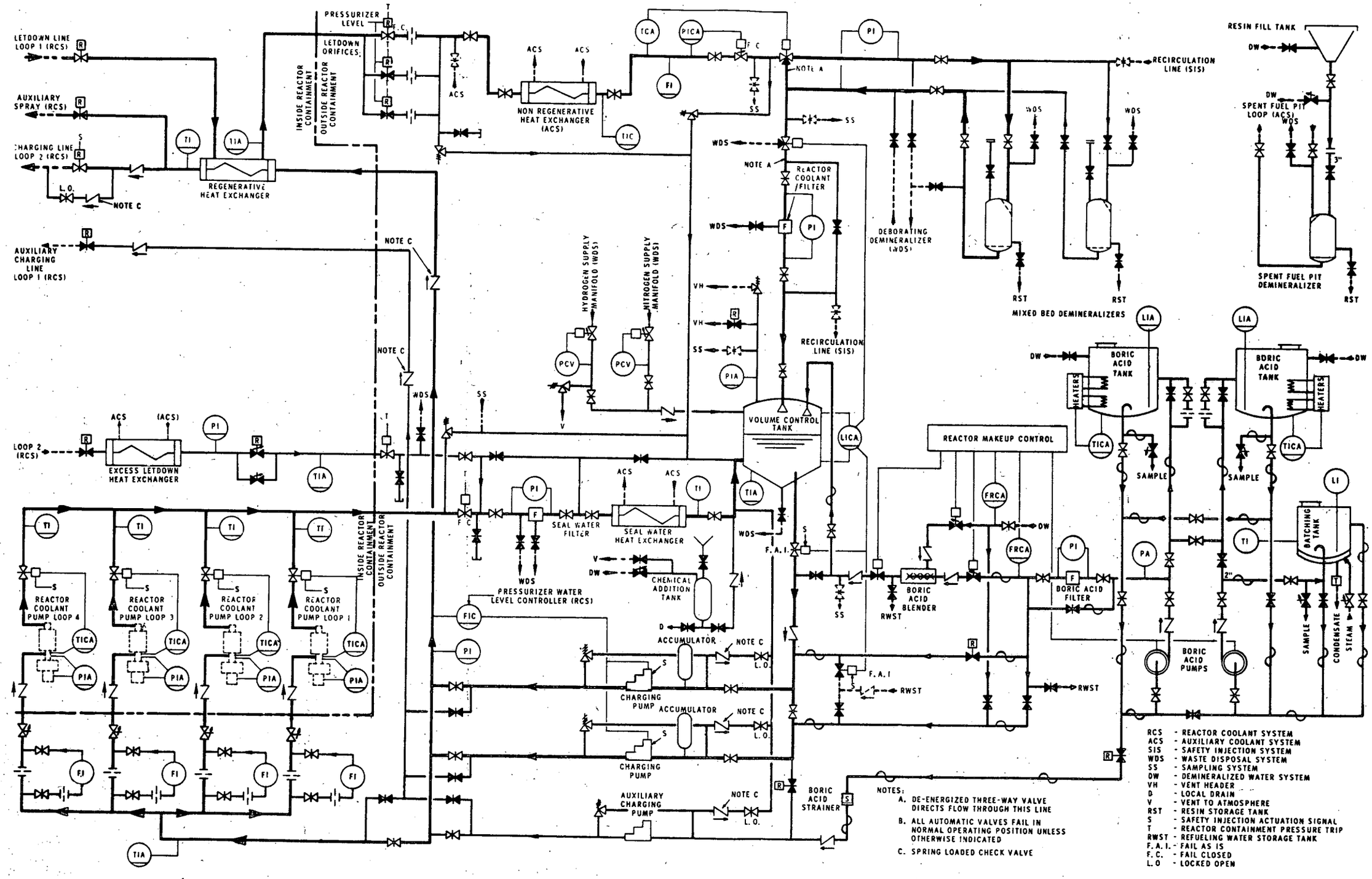
The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an effective, economic and optically transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Boric acid is added to the water to insure sub-critical conditions during refueling.

The Fuel Handling System may be generally divided into two areas: the reactor cavity which is flooded only during plant shutdown for refueling and the spent fuel pit which is kept full of water and is always accessible to operating personnel. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the plant containment.

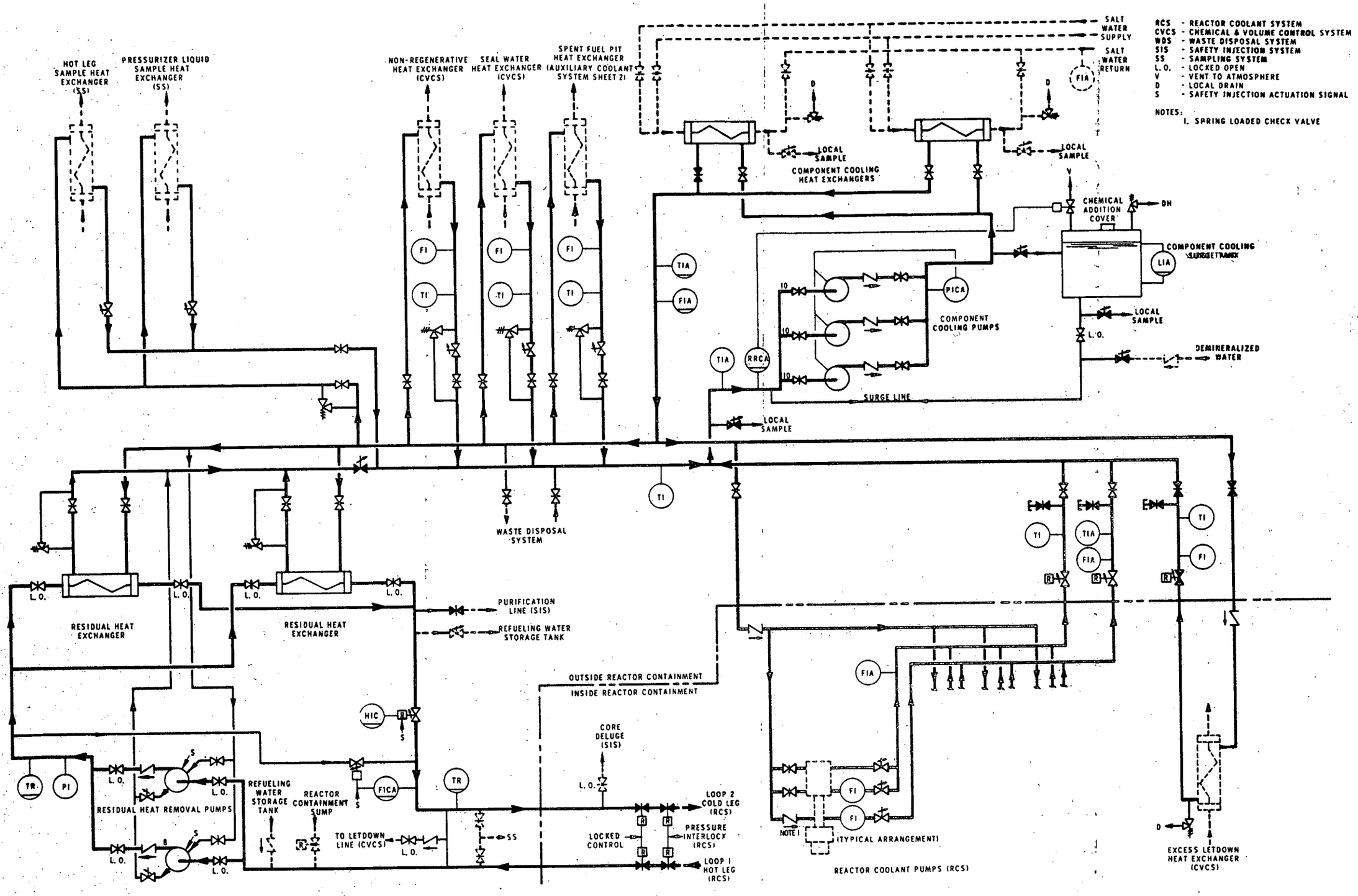
In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pit, the fuel is removed from the transfer system and placed in storage racks with long manual tools suspended from an overhead crane. The pit is designed to accommodate a total of 1 and 1/3 cores; it has space for the normal storage, a spen 1/3 core during its decay period and the forced unloading of a complete operating core. After a sufficient decay period, the fuel is removed from storage and loaded into a shipping cask for removal from the site.

New fuel assemblies are received by rail or truck and stored in the new fuel storage building. New fuel is delivered to the reactor by lowering it into the spent fuel pit and taking it through the transfer system. Alternatively the fuel may be taken through the reactor containment equipment hatch and lowered directly into the reactor cavity. The new fuel storage building is sized for storage of the fuel assemblies and control rods normally associated with the replacement of one-third of a core.

A general view of the spent fuel storage pit is shown on Figure 3.2-6 and a schematic diagram of the fuel transfer system is shown on Figure 3.2-7.

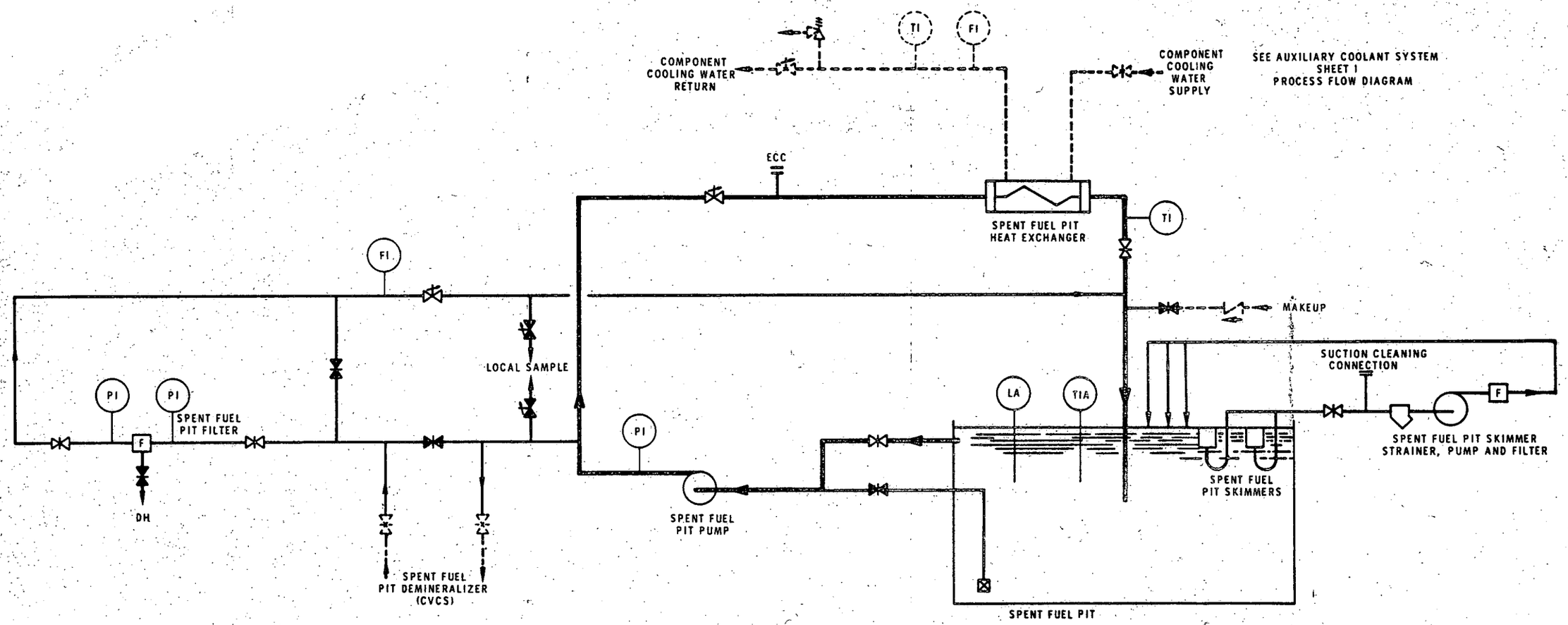


Chemical and Volume Control System -- FIG. 3.2-1
E.D.SK. 319387-F

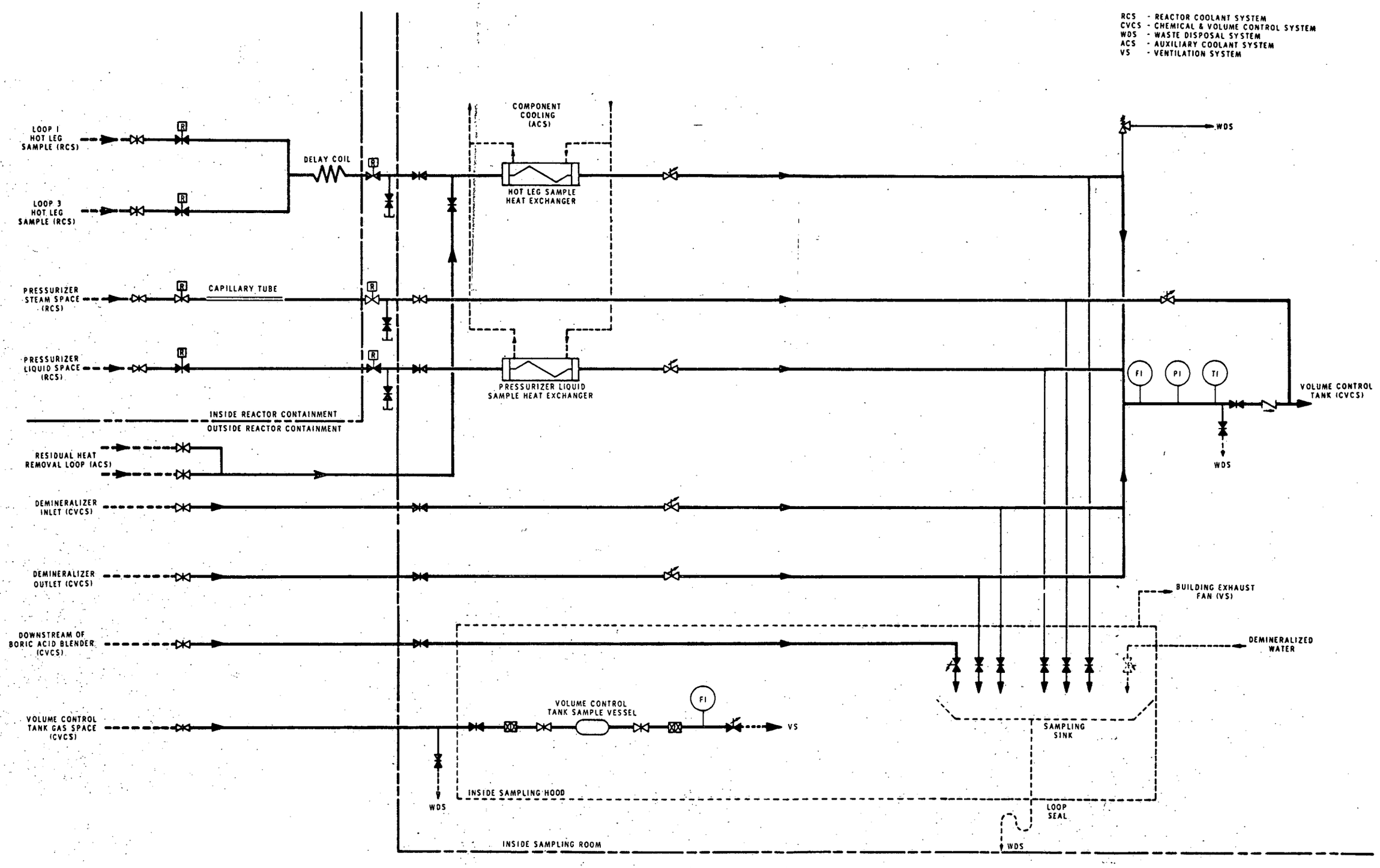


Auxiliary Coolant System - Sheet 1 of 2 -- FIG. 3.2-2
E.O.SK. 319388-F

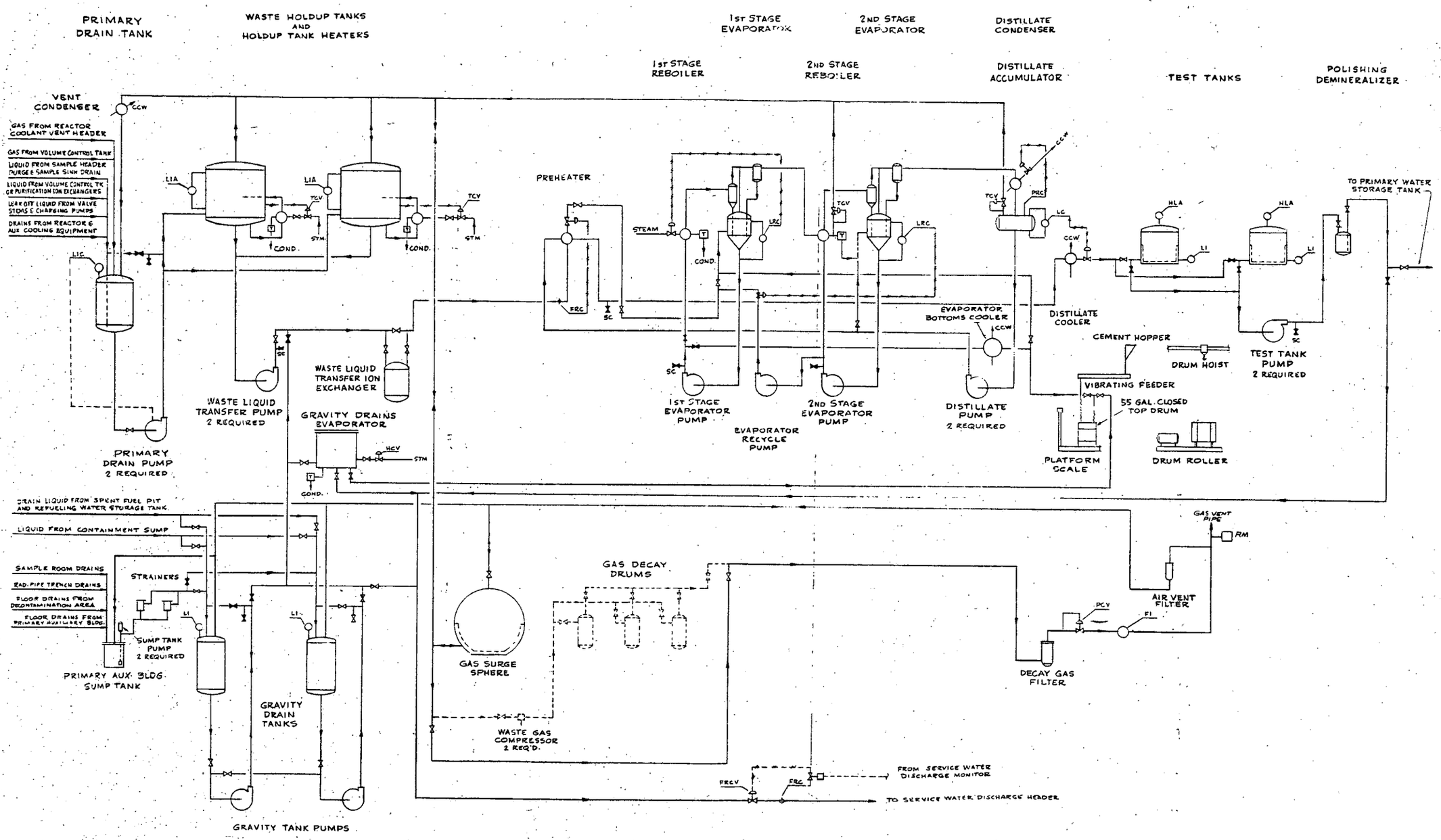
CVCS - CHEMICAL AND VOLUME CONTROL SYSTEM
ECC - EMERGENCY COOLING CONNECTION
DH - DRAIN HEADER (WASTE DISPOSAL SYSTEM)



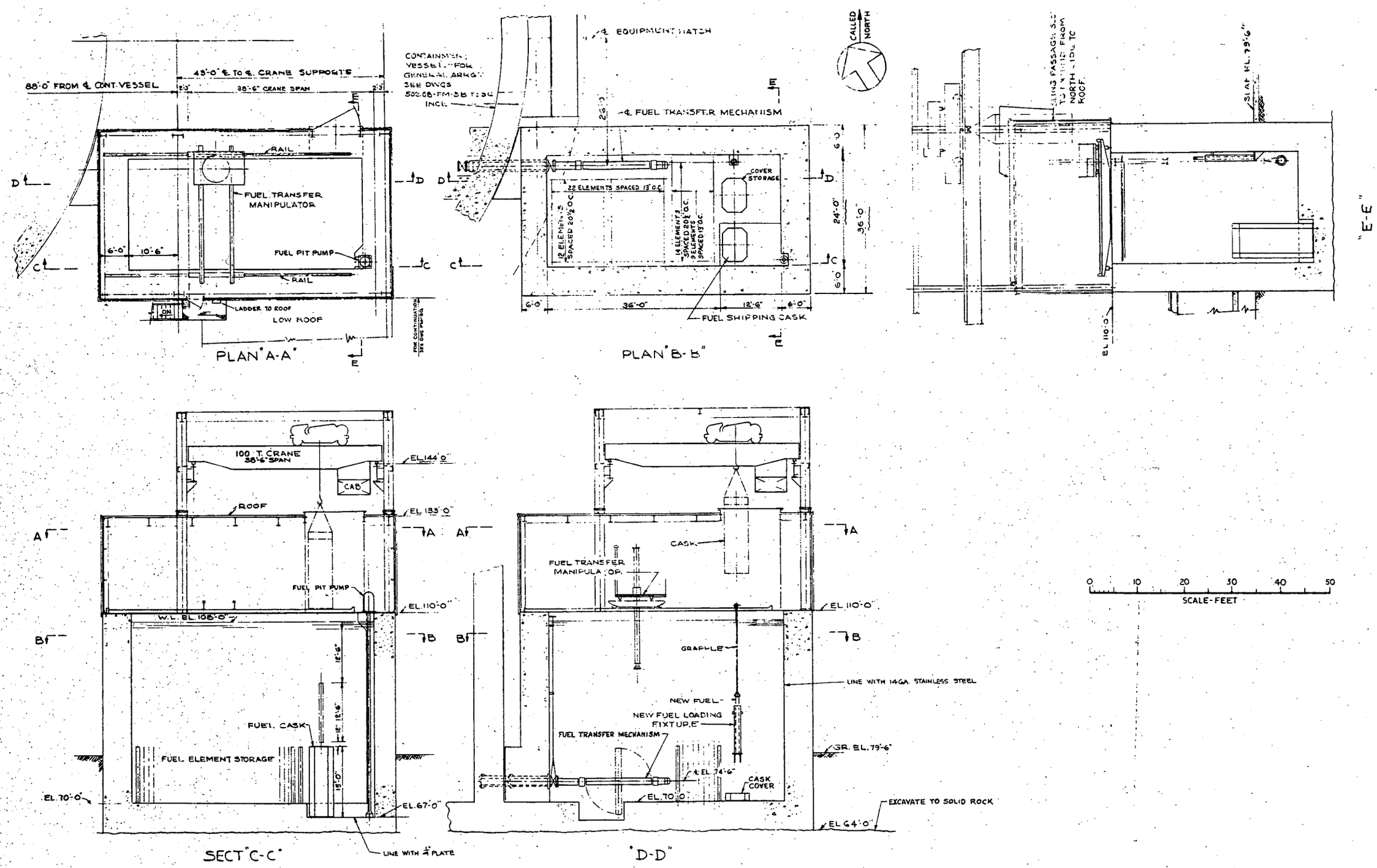
Auxiliary Coolant System - Sheet 2 of 2 -- FIG. 3.2-3
E.D.SK. 319389-F



Sampling System -- FIG. 3.2-4
E.D.SK. 319390-F

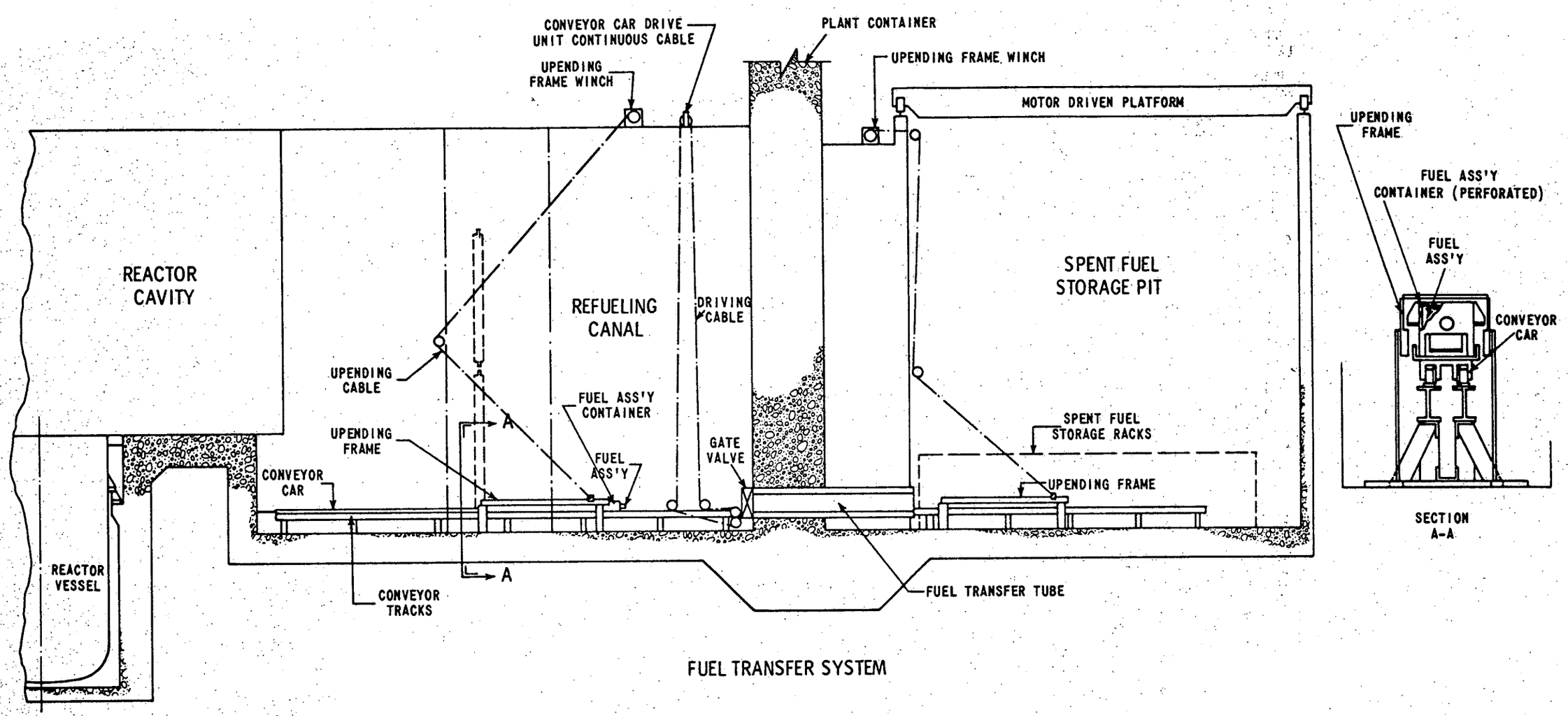


Typical Waste Disposal System -- FIG. 3.2-5



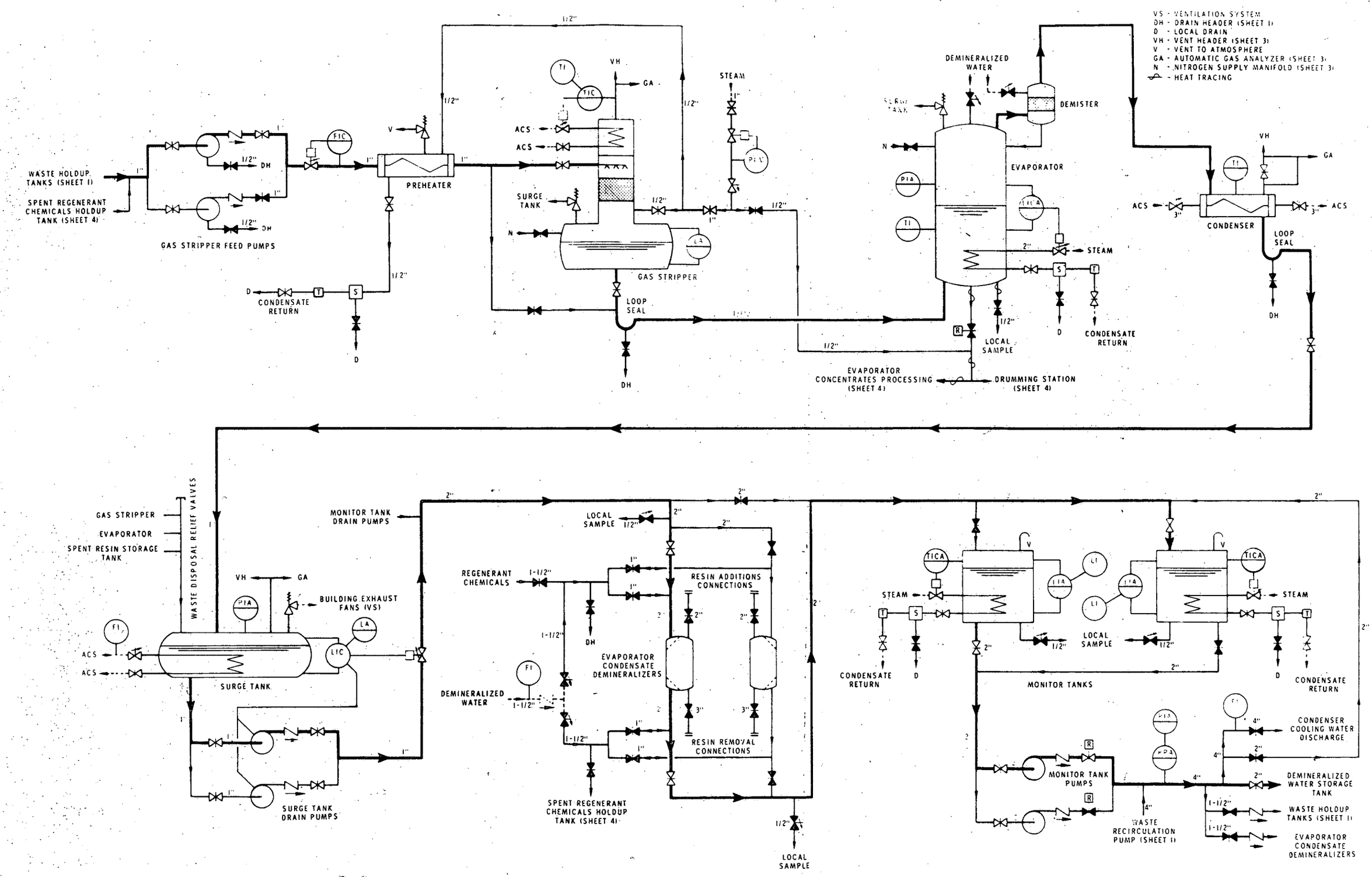
ISSUED FOR ESTIMATE ONLY

General Arrangement Spent Fuel Pit -- FIG. 3.2-6



FUEL TRANSFER SYSTEM

Fuel Transfer System -- FIG. 3.2-7



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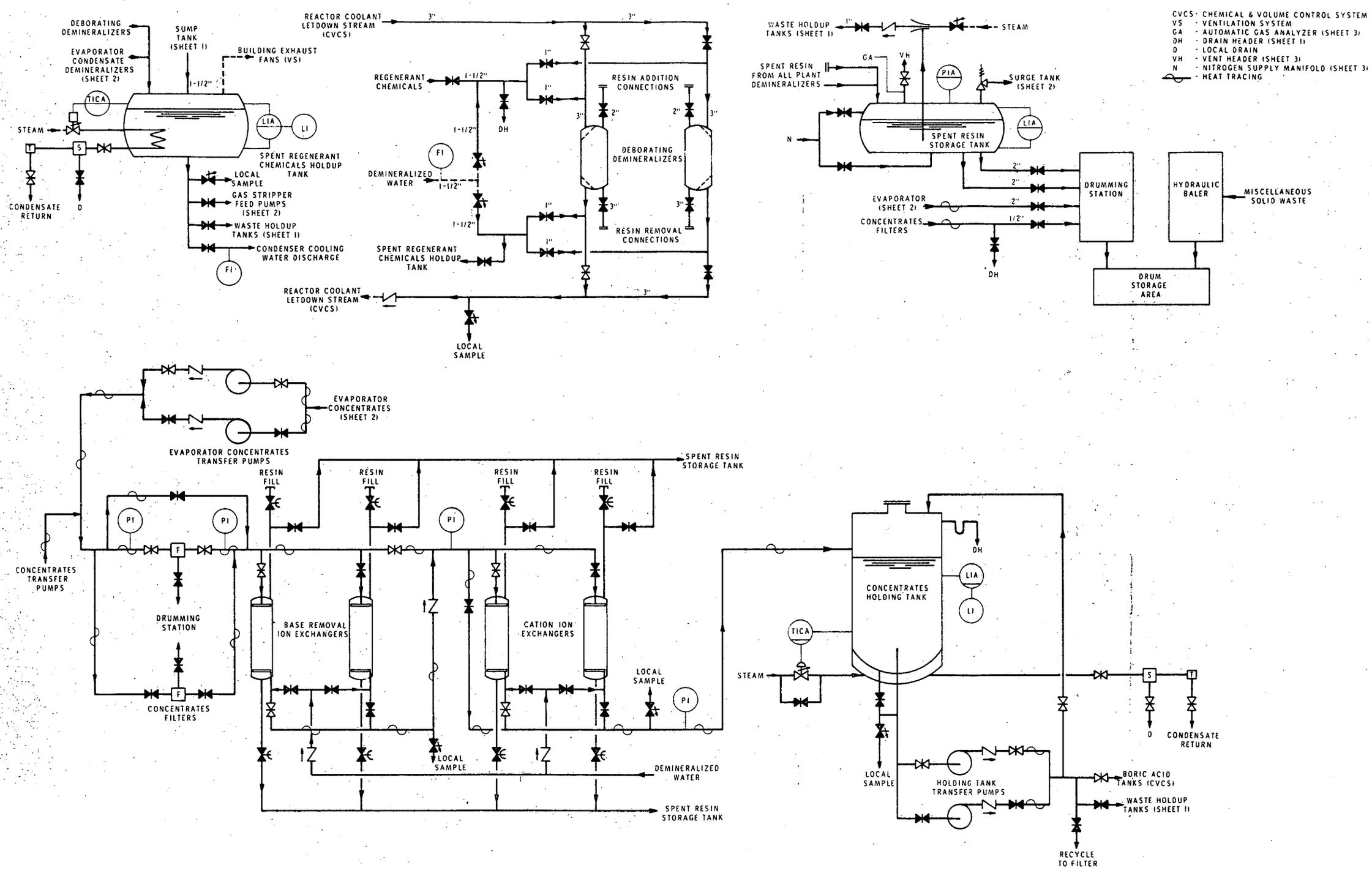
Waste Disposal System - Sheet 2 of 4 -- FIG.3.2-5b



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Waste Disposal System - Sheet 4 of 4 -- FIG. 3.2-5d

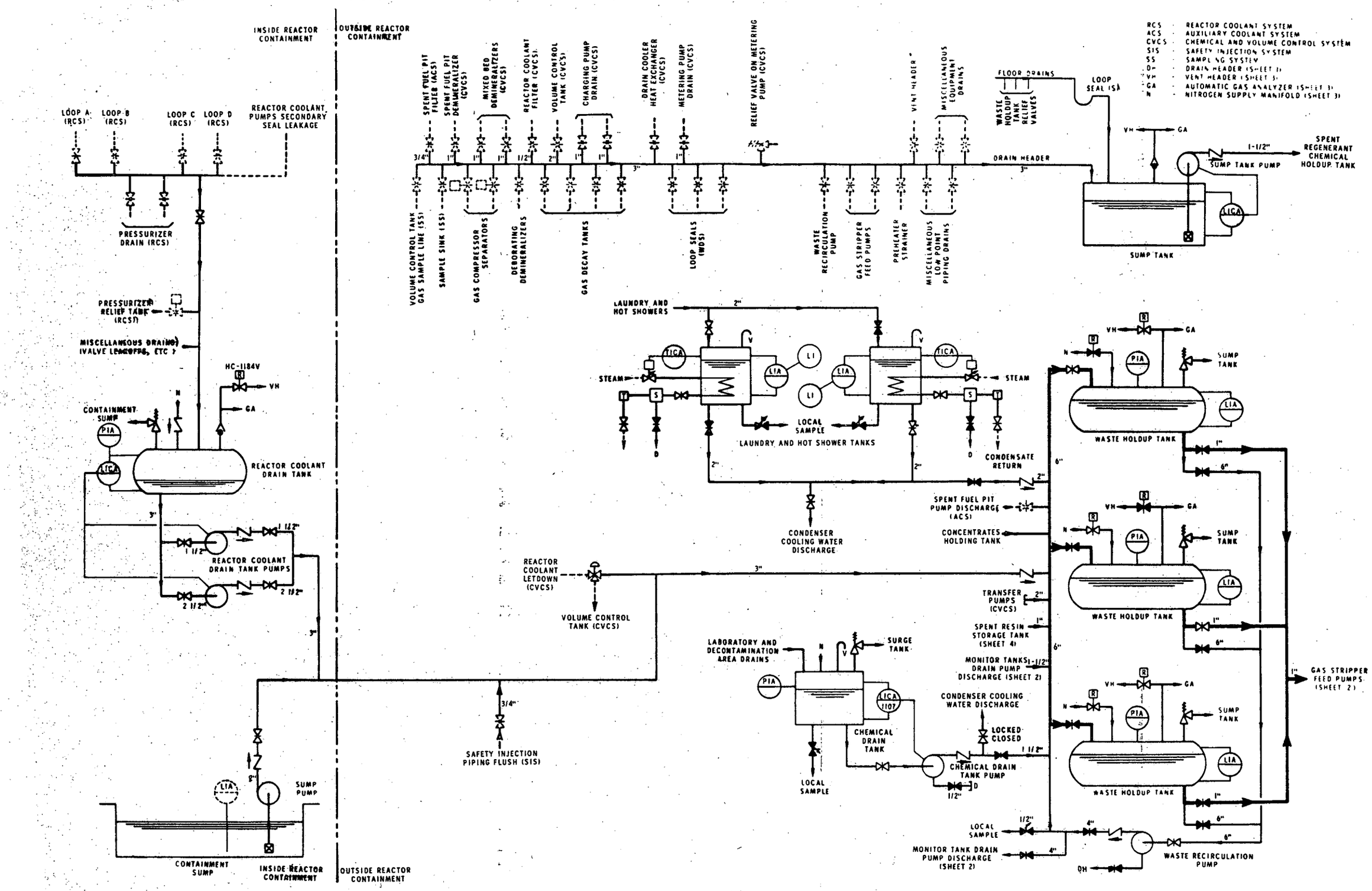
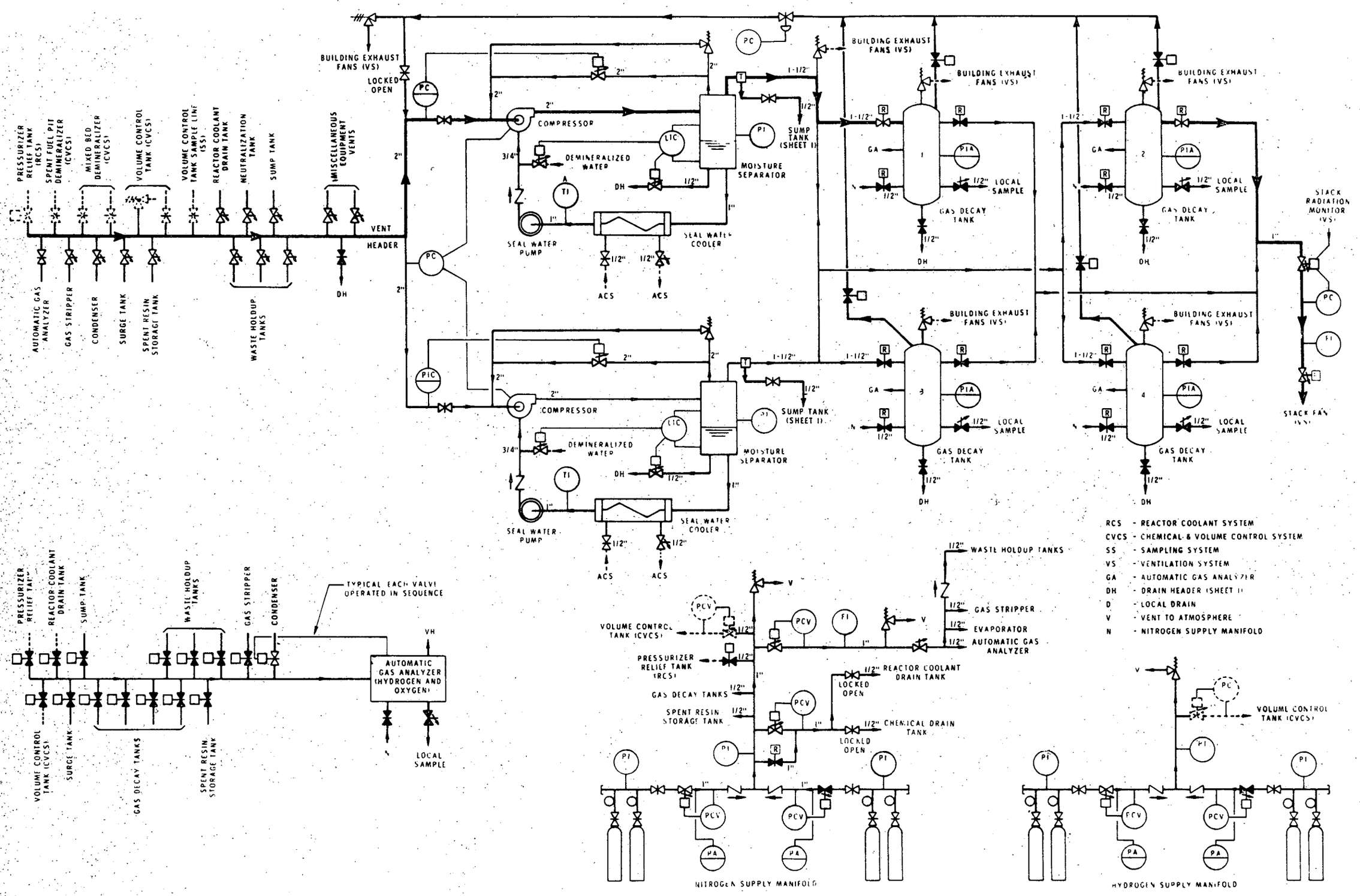


FIGURE 3.2-52 Waste Disposal System - Sheet 1
E.D.SK. 319392F



E.D.SK. 319394F

Waste Disposal System - Sheet 3 of 4 -- FIG. 3.2-5c

3.3 Station Electrical

The auxiliary power distribution system for the Indian Point Unit No. 2 is essentially the same as that used in present day conventional plants. Two large auxiliary transformers supply power at 6.9 KV. One transformer, designated the unit auxiliary, is fed directly from the generator output. The other transformer, designated the station auxiliary, is supplied from the 138 KV Buchanan Substation.

6900 Volt System

During startup, shutdown and hot standby, auxiliary power is supplied from the Station Auxiliary Transformer. When the generator is synchronized to the 345 KV system, auxiliary power is supplied primarily by the Unit Auxiliary Transformer. A small portion of the auxiliary load, the motor driven boiler feed water pump and two 6900/480 auxiliary transformers, remain on the Station Auxiliary Transformer at all times. The Station Auxiliary Transformer supplies two 6.9 KV bus sections (Nos. 5 and 6) through 2000 ampere, 500 MVA air circuit breakers. The Unit Auxiliary Transformer supplies 6.9 KV bus sections Nos. 1, 2, 3 and 4 through 2000 ampere, 500 MVA air circuit breakers.

The 6.9 KV bus section No. 5 can be connected to either 6.9 KV bus section No. 1 or No. 2 by a 2000 ampere, 500 MVA air circuit breaker. In a similar manner, the 6.9 KV bus section No. 6 can be connected to either 6.9 KV bus section No. 3 or No. 4. All major auxiliaries above approximately 300 HP are supplied from the 6.9 KV switchgear by 1200 ampere, 500 MVA air circuit breakers. These auxiliaries are supplied primarily from 6.9 KV bus sections Nos. 1, 2, 3 and 4 when the generator is synchronized to the 345 KV system.

Each of these bus sections supplies one reactor coolant pump and other auxiliaries as shown on the main one line diagram, Figure 3.3-1. In addition to the segregation of the four reactor coolant pumps, the remaining auxiliaries are

also divided between the four 6.9 KV buses to provide a high degree of diversity and to minimize the possibility of plant trip due to loss of one 6.9 KV bus. After a reactor or turbine generator trip, the necessary auxiliaries on 6.9 KV bus sections 1, 2, 3 and 4 are transferred automatically to 6.9 KV bus sections 5 and 6 by a dead transfer monitored by voltage relays which allow transfer after the residual bus voltage has decayed to about 33% of normal. This transfer time is on the order of two seconds so that the transfer will be completed before the reactor coolant pumps have begun to slow down.

480 Volt Systems

Motors between 100 and 300 HP are supplied by 480 volt switchgear. This 480 volt system consists of four bus sections designated as Nos. 2, 3, 5 and 6. Each section is normally supplied by a 1500/2000 KVA ventilated, dry type 6900/480 volt transformer. The transformers are numbered to correspond with the bus section they supply.

Two of these 6900/480 volt transformers (Nos. 5 and 6) are normally supplied by the 138/6.9 KV Station Auxiliary Transformers Nos. 5 and 6. These bus sections normally supply most of the engineered safeguards equipment and are left on the outside source of power to avoid subjecting them to a transfer at the time that they may be called upon to operate. Complete duality of supply to engineered safeguards equipment is maintained, however, additional equipment which may be called upon to operate is supplied from 480 volt bus sections Nos. 2 and 3 which can be connected to bus sections Nos. 5 and 6 if required.

The 6900/480 volt transformers Nos. 2 and 3 are fed from bus sections Nos. 2 and 3 of the 6900 volt bus systems, and normally supply 480 volt bus sections Nos. 2 and 3. The buses can be interconnected by electrically operated tie breakers in case one of the normal 6900/480 KV transformers is out of service. Interlocks prevent energizing a bus section from more than one source of power

to stay within switchgear rating.

Three of the 480 volt bus sections (Nos. 3, 5 and 6) can each also be supplied by an automatic starting diesel generator which will act as a back up to the normal source of supply to the bus section. If the normal AC supply to 480 volt bus section 3, 5 or 6 is lost, a diesel connected to that section will automatically start and energize the bus. If the supply of 480 volt bus section No. 2 is lost it can be interconnected to one of the other 480 volt bus sections.

For loss of all normal AC power to the 480 volt bus sections, the tie breakers between the sections allow any of the four to be supplied by diesel generators. The diesels will be sized so that any two of the three diesels will have sufficient capacity to supply the engineered safeguards load required for an acceptable post-blowdown containment pressure transient.

Summary

The four 480 volt bus sections which supply power to the engineered safeguards equipment have alternate sources of power to provide assurance of operation under all conditions.

1. All four buses can be supplied from the 138 KV Buchanan Substation
2. Three buses can be supplied directly by automatically starting diesel generators
3. All buses could be interconnected by tie breakers

accomplished in the demineralizer waste overflow, I trust you can get an approximate figure for the dilution that may result in the river off shore, and can compare this with what you may find necessary or desirable for adequate protection of fish life or of the fish eating public.

As far as the effect on public water supplies is concerned, the use of the Hudson River for water supply, other than condenser cooling, is very limited. The nearest municipality involved is Poughkeepsie, 30 miles or more upstream, and even at that distance threatened at times with the problem of salinity. There is no likelihood that in the future any nearer municipality will take its domestic water supply from the Hudson. In fact the tendency is the other way, and the more remote municipalities of Catskill and Hudson have abandoned earlier supplies taken from the river.

As far as the effect on ground water is concerned, you have acquired an ample area of surrounding land. I can see no possibility of any deleterious effect.

I trust that this information which I have assembled in the limited time available will be helpful to you. If from these approximate figures there appears to be any question as to the adequacy of the safety factor in dilution, you may, as above stated, require additional information from float tests.

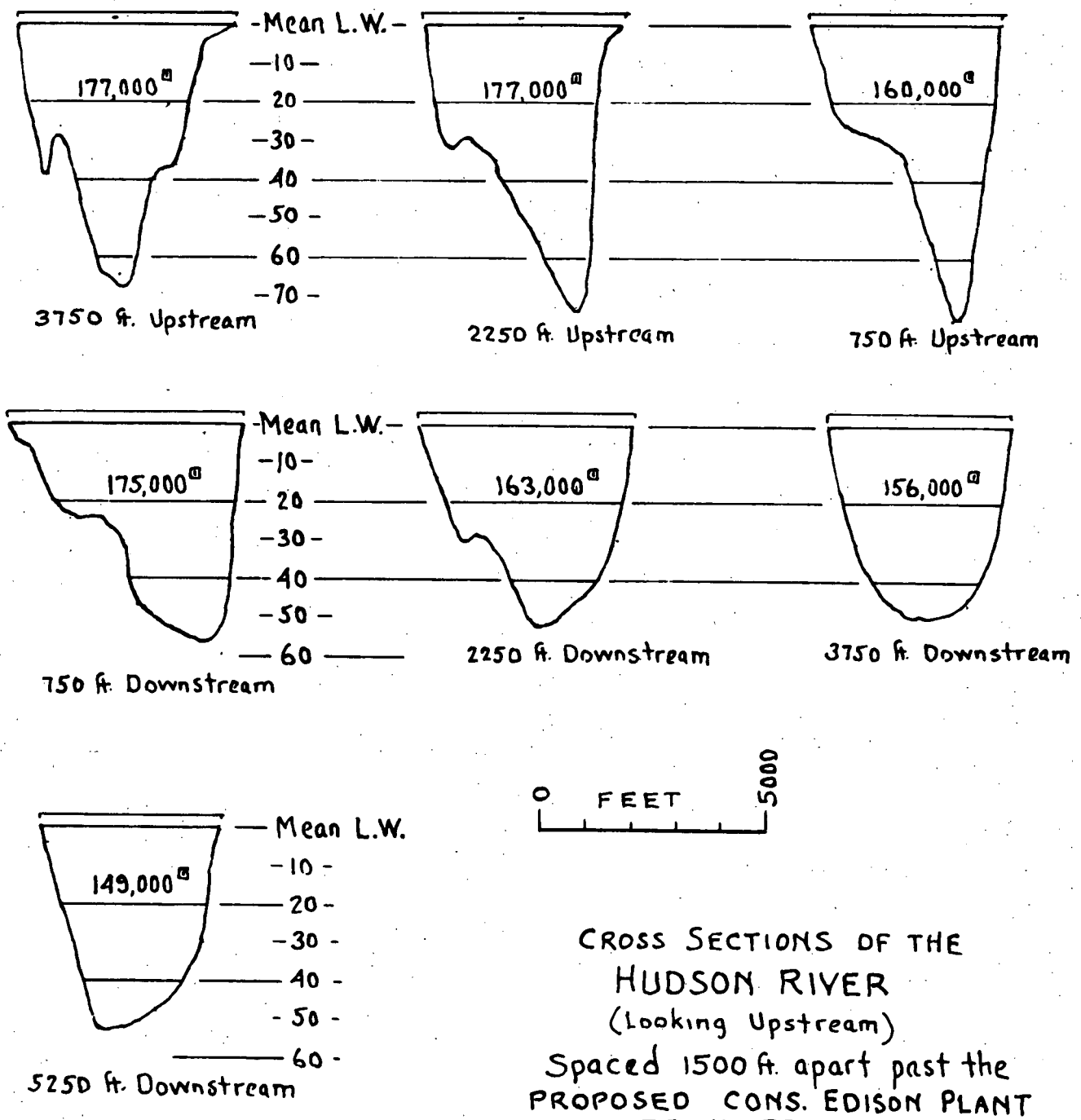
From what you have told me about your proposed designs and methods of operation, I suspect that there is no real question of safety but only one of public relations - the avoidance of even the appearance of danger.

Very truly yours,

Karl R. Kennison
Consulting Engineer

Sheet 1
9-13-'55

KARL R. KENNISON
CIVIL AND HYDRAULIC ENGINEER
361 CLINTON AVE., BROOKLYN, N. Y.



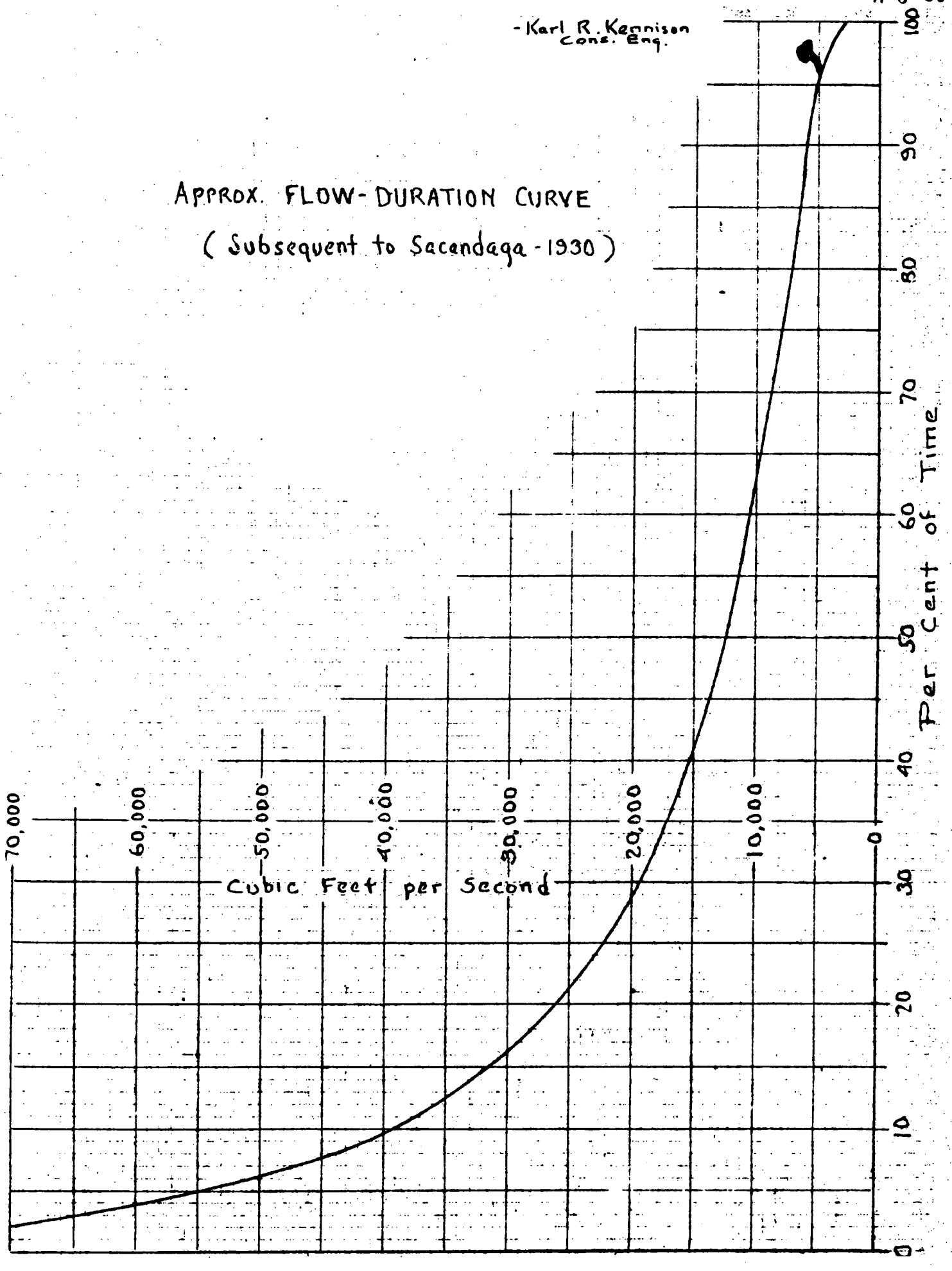
CROSS SECTIONS OF THE
HUDSON RIVER
(Looking Upstream)
Spaced 1500 ft. apart past the
PROPOSED CONS. EDISON PLANT
BELOW PEEKSKILL

R.K.

Hudson River at Location of Proposed Cons. Edison Plant below Peekskill

11-8-55

- Karl R. Kennison
Cons. Eng.



K.R.K.

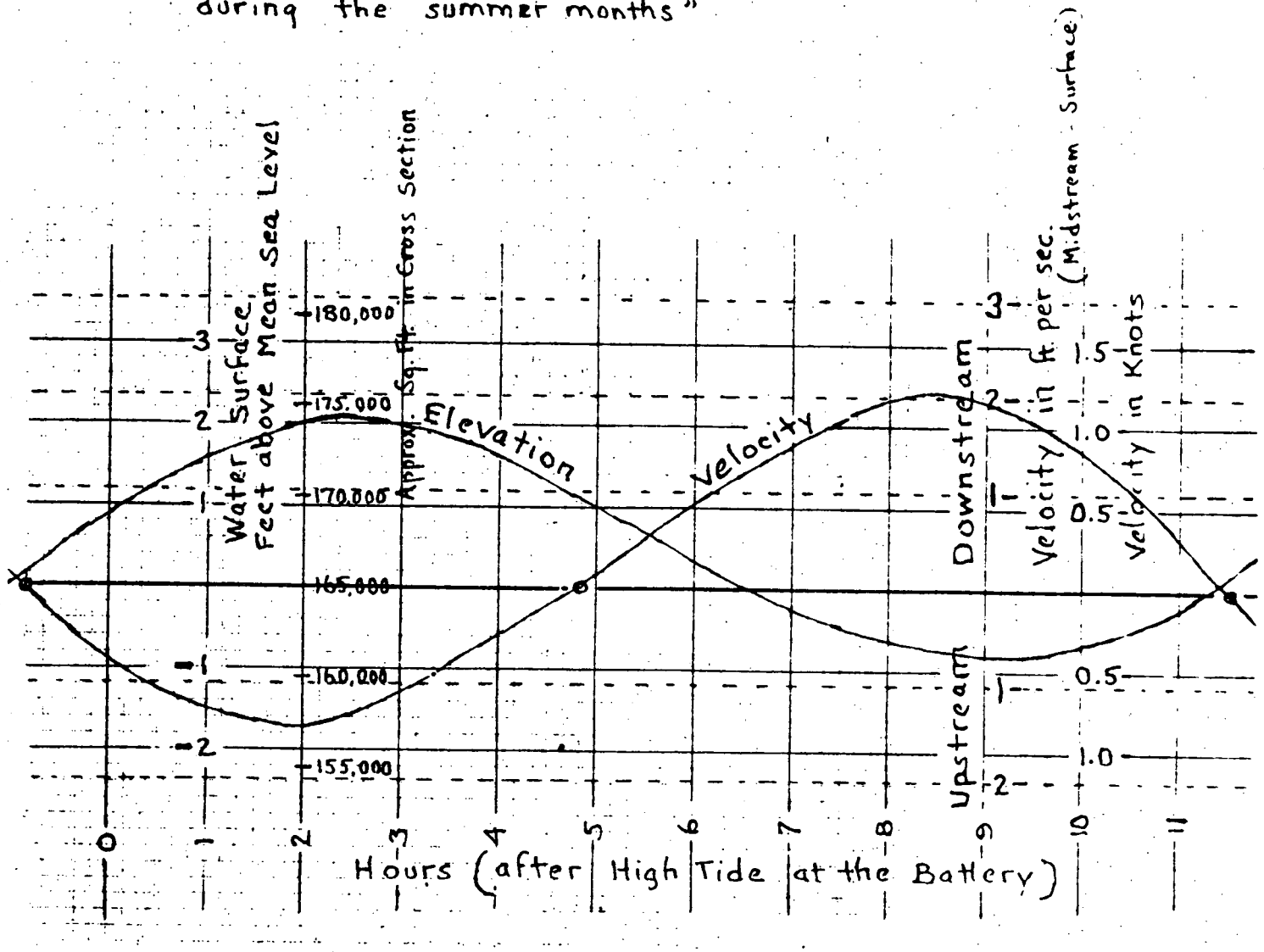
Hudson River at Location of Proposed
Cons. Edison Plant below Peekskill

9-13 5

- Karl R. Kennison
Cons. Eng.

Data from Coast & Geodetic Survey

"Average conditions in the axis of the main channel during the summer months"

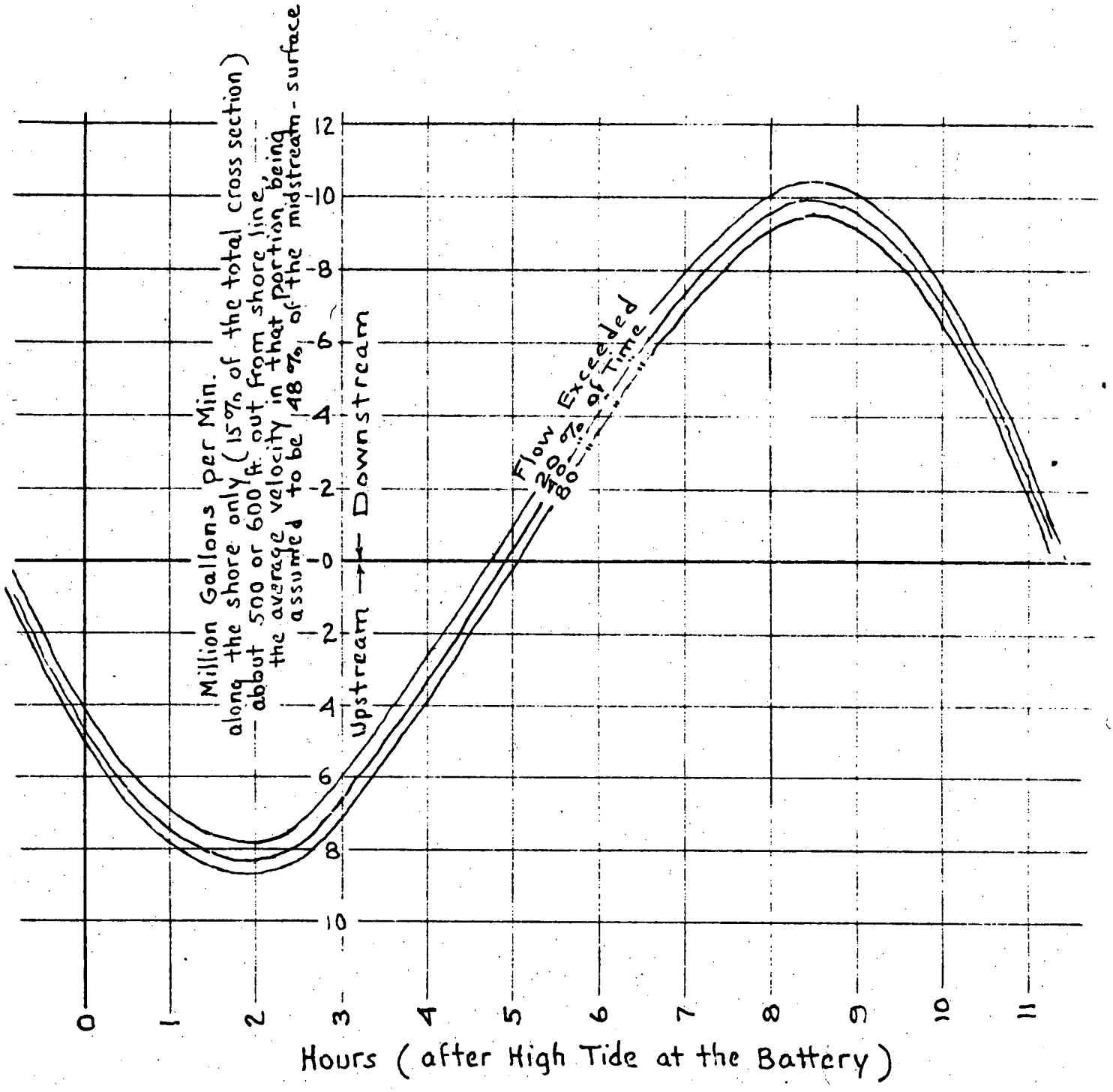


R.K.

Hudson River at Location of Proposed Cons. Edison Plant below Peekskill

11-8 55

- Karl R. Kennison
Cons. Eng.



4. ENGINEERED SAFEGUARDS

4.0 General

Five types of engineered safeguards are included in the design of this facility. These are:

1. The safety injection system which injects water into both the hot leg and cold leg of each reactor coolant loop. This system limits damage to the core and limits the energy released into the containment following a loss of coolant accident.
2. The containment spray system which is used to reduce containment pressure and remove elemental iodine from the containment atmosphere by washing action.
3. A steel-lined, reinforced concrete reactor containment vessel with pressurized penetrations and testable welds which forms a virtually leak-tight barrier to the escape of fission products should a loss of coolant occur.
4. The air recirculation filters which provide for rapid removal of iodine from the containment atmosphere if fission products are released from the reactor.
5. The air recirculation coolers which reduce containment pressure following the loss of coolant. Air is recirculated through these coolers by the same fans which circulate air through the air recirculation filters.

4.1 Safety Injection System

4.1.1 Performance Objectives

The primary purpose of the safety injection system is to supply borated water to the reactor coolant system to limit fuel rod cladding temperatures in the unlikely event of a loss of reactor coolant. The system also provides a means of manually introducing borated spray water in the containment as a backup for the air recirculation cooling system, described later in Section 4.4

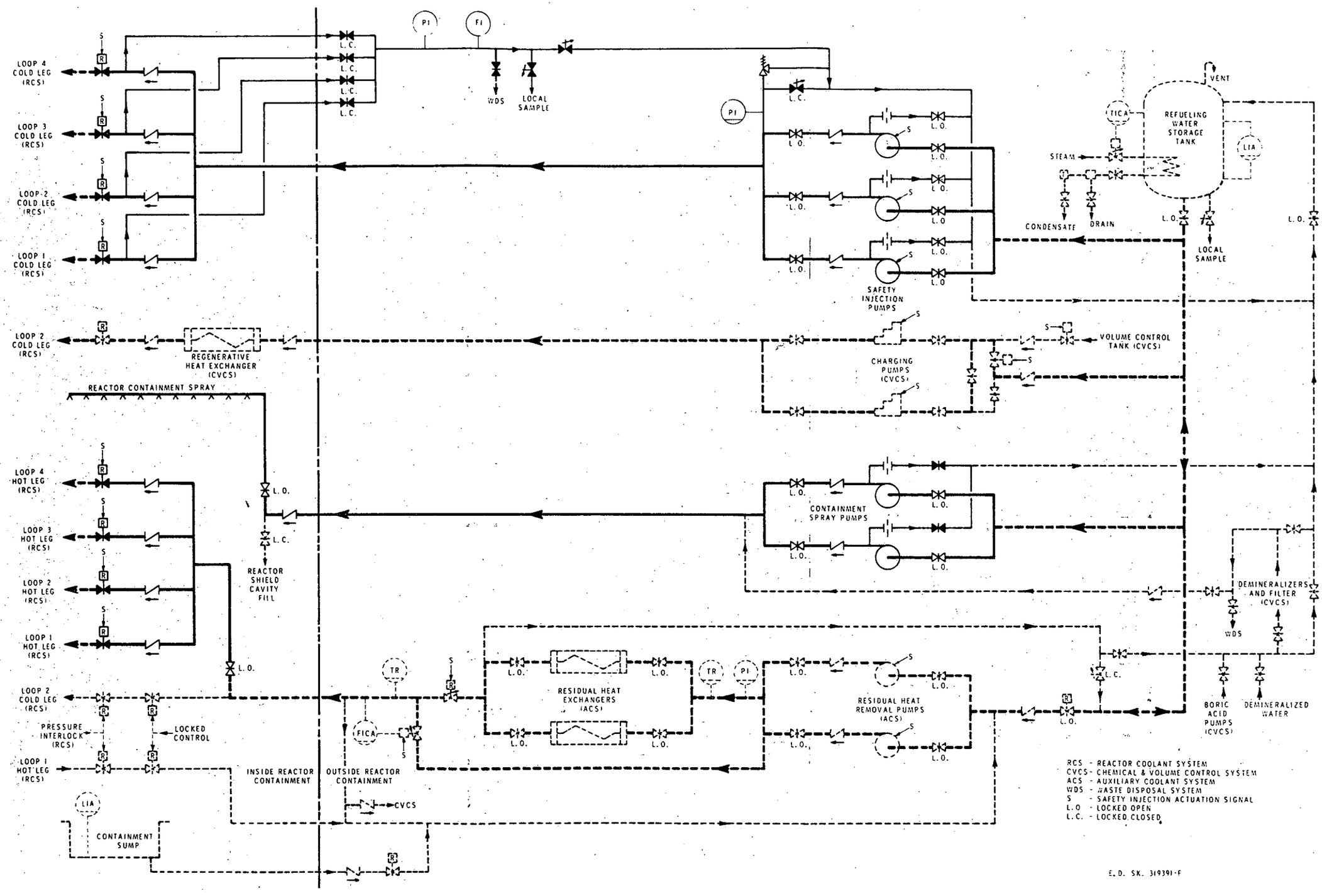
With normal AC power available, operation of the entire safety injection system prevents any melting of fuel cladding and limits metal-water reaction to essentially zero for reactor coolant piping ruptures up to and including the diameter of the largest pipe connected to the coolant loops (the pressurizer surge line or the residual heat removal lines) or the rupture of an injection line.

For the situation when normal AC power is not available, on-site emergency power is provided. The pumping capacity available with emergency power will ensure long-term removal of core residual heat and any metal-water reaction energy. An acceptable, post blowdown, containment pressure transient is assured by the combined operation of the safety injection system and the containment air recirculation coolers.

4.1.2 Description

The principal components of the safety injection system which provide emergency core cooling are three high head safety injection pumps, the two charging pumps of the chemical and volume control system and the two low head residual heat removal pumps of the auxiliary coolant system. These pumps are all located in the primary auxiliary building and take suction directly from the refueling water storage tank located adjacent to the primary auxiliary building.

The safety injection pumps refill the reactor vessel by way of the reactor inlet piping, assuring core cooling by restoring water level to a point above the top of the core. The safety injection pumps discharge into each of the four cold legs of the reactor coolant piping. Flow from these pumps is augmented



Safety Injection System -- FIG. 4.1-1

by the charging pumps which discharge into one of the four loops.

The residual heat removal pumps deliver borated water to the reactor vessel above the core, providing an immediate cooling effect. These pumps discharge into each of the four hot legs of the reactor coolant piping.

The residual heat removal pumps can also take suction directly from the containment sump. The safety injection and charging pumps can take suction from the containment sump via the discharge of the residual heat removal pumps.

Two separate pumps, manually activated, are provided for containment spray. These pumps are also located in the primary auxiliary building and can take their suction directly from the refueling water storage tank or from the containment sump via the discharge of the residual heat removal pumps.

The safety injection system arrangement, including both emergency core cooling and containment spray, is shown on Figure 4.1-1.

Safety injection (emergency core cooling) is actuated by coincidence of low pressurizer pressure and low pressurizer water level. These coincident signals open the safety injection system isolation valves and start the high head safety injection pumps and low head residual heat removal pumps. Under conditions of low pressurizer pressure and water level, the charging pumps would already be running. In addition to opening the isolation valves and starting the pumps, the safety injection signal shifts the suction of the charging pumps from the volume control tank to the refueling water storage tank. Suction for the safety injection pumps and the residual heat removal pumps is already from the refueling water storage tank when the reactor is in operation.

After the reactor coolant system has been cooled and depressurized, coolant spilled from the break is cooled and returned to the reactor coolant system by the auxiliary coolant system.

In this mode of operation, the residual heat removal pumps take suction from the reactor containment sump, circulate the spilled coolant through the residual heat exchangers and return the coolant to the reactor. The safety injection pumps may be used to augment the head capacity of the residual heat removal pumps in re-

turning the spilled coolant to the reactor.

The emergency core cooling portion of the safety injection system may be tested for proper operation at shut-off head any time when the reactor is hot and pressurized. Lines connecting each of the safety injection branch lines to the refueling water storage tank are used periodically to recirculate the safety injection piping contents to ensure that water in the safety injection piping contains the same boron concentration as the water in the refueling water storage tank.

The containment spray pumps can be tested at any time during reactor operation. The valves at the discharge of the spray pumps will be closed for this test, and refueling water will be recirculated from pump discharge back to the refueling water storage tank.

4.2 Reactor Containment

4.2.1 Performance Objectives

The reactor containment completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the reactor coolant system were to occur. The structure provides adequate biological shielding for both normal and accident situations.

The design pressure of the containment is at least equal to the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to the hypothetical severence of a reactor coolant pipe. In addition, the design pressure will not be exceeded during any subsequent long term pressure transient determined by the combined effects of heat sources such as residual heat and metal water reactions, structural heat sinks, and the operation of other engineered safeguards utilizing only the emergency on-site electric power supply.

The supports for the reactor coolant system are designed to withstand blowdown forces associated with complete severence of the reactor coolant piping.

The containment is designed to withstand an earthquake having a horizontal acceleration of .1 g in combination with failure of the reactor coolant system.

Stress criteria for reinforcing steel in the containment design are as follows:

LOAD CONDITION	PRIMARY STRESSES*	PRIMARY PLUS SECONDARY STRESSES**
	% Yield	% Yield
Operating plus incident	53.3	66.7
Operating plus 0.1 g horizontal earthquake	50.0	53.3
Operating plus incident plus 0.03 g horizontal earthquake	66.7	66.7
Operating plus incident plus 0.10 g horizontal earthquake	80.0	80.0

* Primary stresses do not include temperature stresses.
**Secondary stresses are considered as follows:

1. Stresses resulting from normal operating temperature gradient from inside the containment to the outside atmosphere.
2. Stresses resulting from incident temperature effect.

The steel liner is placed in compression by the temperature effects of the accident. The combination of compressive stresses due to temperature and the tangential shearing stresses due to the horizontal earthquake loading will not cause buckling or the exceeding of the yield strength of the linear material.

Missile protection is afforded by the three foot thickness of the polar crane support wall and the two foot thickness of the refueling floor. These structures surround the reactor coolant system and the other high pressure piping and components, failure of which might cause or be caused by a loss-of-coolant accident. A missile shield is located above the reactor vessel to stop any missile that might be propelled as the result of a rupture of a control rod mechanism housing.

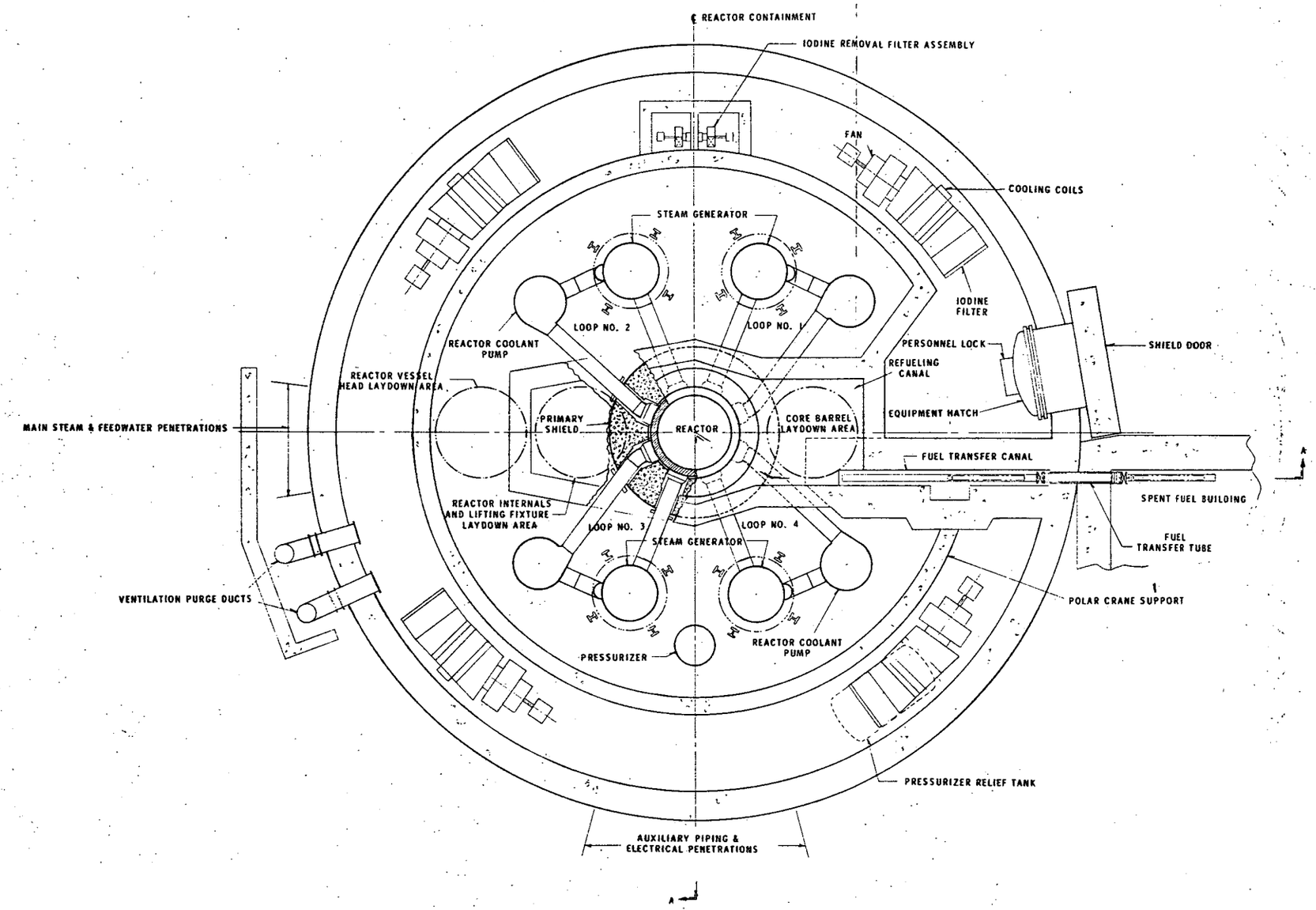
The leakage of the containment under accident conditions will not exceed the value necessary to meet the requirements of 10 CFR 100 with proper credit taken for the other engineered safeguards assuming complete core meltdown.

The containment design will include adequate means of testing the containment for leak tightness.

4.2.2 General Description

The reactor containment consists of a containment structure and several auxiliary systems which are commonly referred to as "engineered safeguards." The containment is designed to limit the consequences of any release of radioactive material from the reactor coolant system.

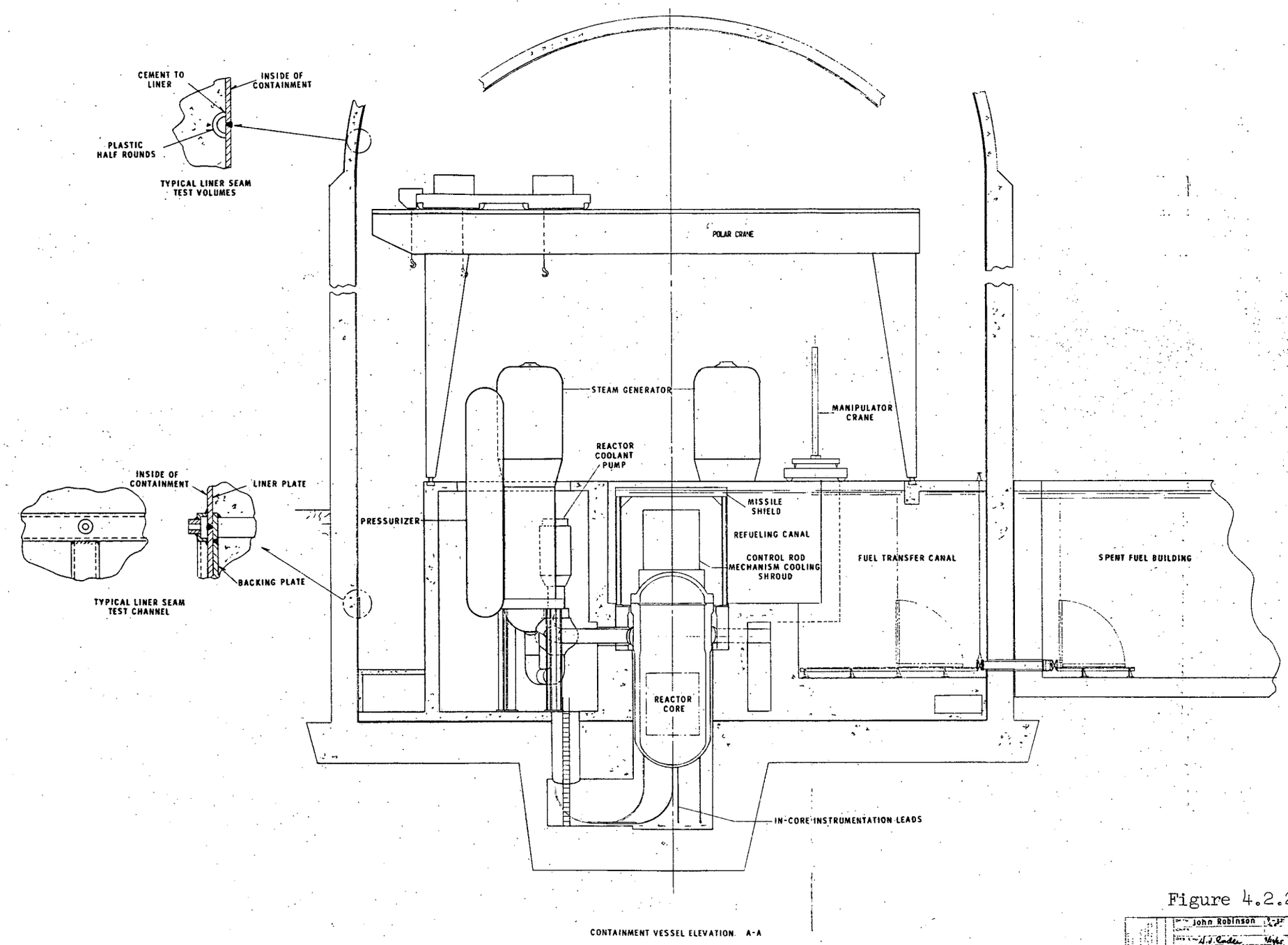
The containment structure is basically a right circular cylinder with a hemispherical dome and a flat base. The inside diameter is 135 feet and the straight height is 147 feet. The gross volume is approximately 2,742,000 cubic feet and the net volume is approximately 2,610,000 cubic feet.



TYPICAL CONTAINMENT VESSEL PLAN
BELOW REFUELING FLOOR

Figure 4.2.2-1

DRAWN John Robinson	WESTINGHOUSE ELECTRIC CORPORATION	
	ATOMIC POWER DIV. PITTSBURGH PA., U.S.A.	
	TYPICAL CONTAINMENT VESSEL PLAN BELOW REFUELING FLOOR	
	FD SK 319441	



CONTAINMENT VESSEL ELEVATION. A-A

Figure 4.2.2-2

John Robinson	WESTINGHOUSE ELECTRIC CORPORATION ATOMIC POWER DIV. PITTSBURGH, PA. U.S.A.
	INDIAN POINT PROJECT – CONTAINMENT VESSEL ELEVATION A-A
	ED SK 319442
	DO NOT SCALE

The construction is reinforced concrete with an interior steel liner which acts as a leakage barrier. The liner is 1/4" thick on the bottom, 3/8" thick on the cylindrical walls, and 1/2" thick on the dome. The flat concrete base mat is 9' thick with an additional 2' thick concrete floor slab over the bottom liner. The cylindrical wall is 5-1/2' thick, and the dome is 4-1/2' thick. The general arrangement and typical liner details are shown on Figures 4.2.2-1 and 4.2.2-2.

The internal pressure used in the design is 47 psig and the ground acceleration in the postulated earthquake is 0.10 g at zero period. Additional information on design loads is provided in Section 4.2.3.

The containment structure is inherently safe with regard to common hazards such as fire, flood and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, present in the containment. A system of lightning rods is installed on the containment dome as protection against electrical storm damage. The containment structure is located at a high elevation with respect to the river level, so that flooding of the river presents no hazard to the containment.

Internal Structures

Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are support by the mat.

A 3-foot thick concrete ring wall surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-foot thick concrete charging floor covers the reactor coolant system compartments. Removable concrete slabs are provided to permit crane access to the reactor coolant pumps. The four steam generators, pressurizer and various piping penetrate the charging floor. Spiral stairs provide access to the areas below the charging floor.

Three major radiation shields are also provided external to the ring wall. A 3-foot thick concrete wall is located in front of the main steam and feedwater piping penetrations. A 3-1/2 foot thick concrete wall with supplementary wings surrounds the two containment ventilation duct penetrations. The main access hatch is shielded by a labyrinth of precast concrete slabs.

Most of the high pressure piping runs within the ring wall. The annulus between the ring wall and the containment outside wall serves as a pipe chase. A 12-inch thick floor shields the piping from the areas accessible for inspection at power. Certain additional equipment, such as the regenerative heat exchanger, is individually shielded.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel building. The floor and walls of the canal are concrete, with the walls and shielding water providing the equivalent of 6 feet of concrete. The floor is 4 foot thick. The concrete walls are lined with 3/8 inch carbon steel plate and the floor with 1/4 inch stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operations.

Containment Isolation

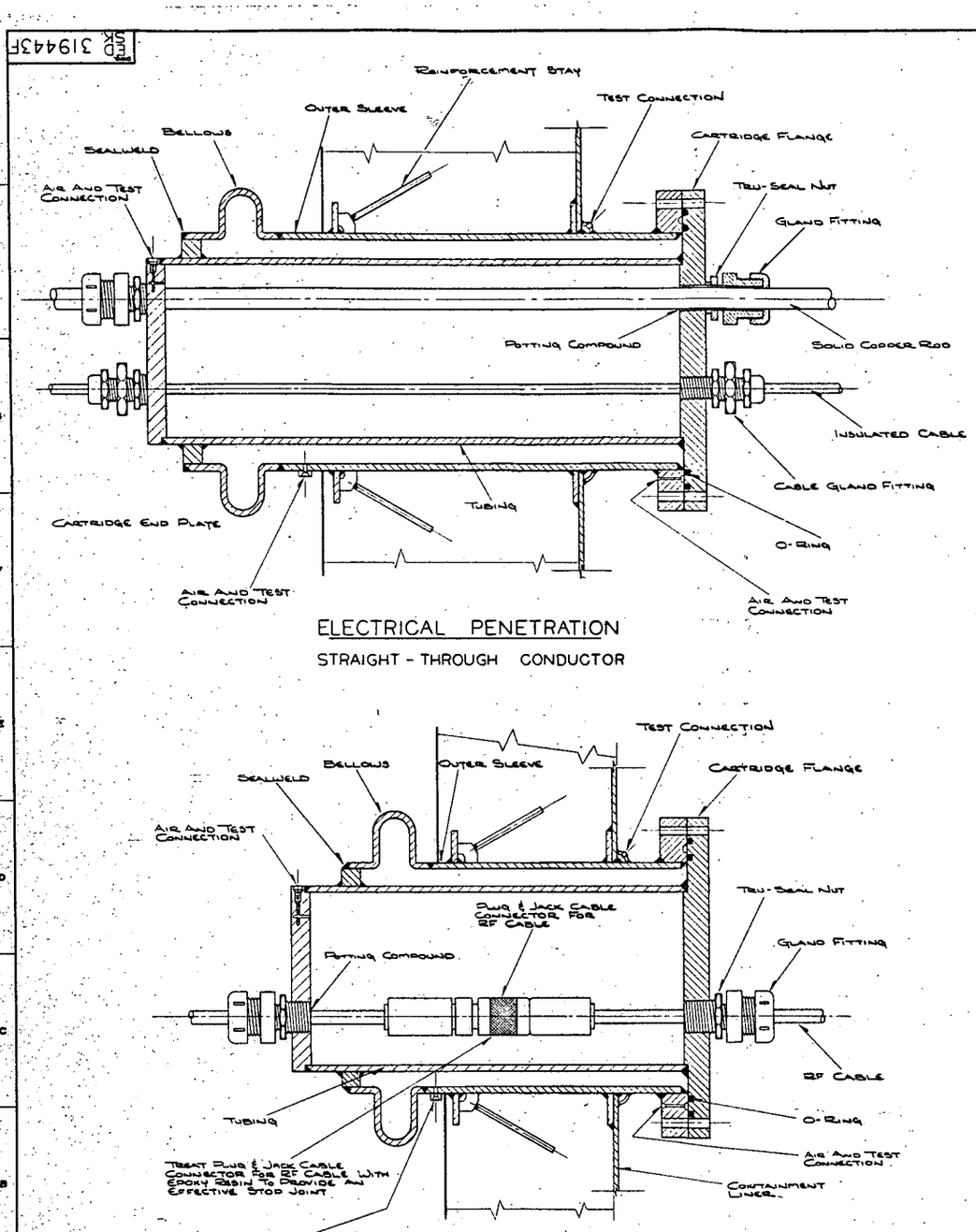
Containment isolation valves are provided as necessary in lines penetrating the containment liner to assure that no unrestricted release of radioactivity can occur. Such releases might possibly be caused by the rupture of a line within the containment concurrent with a loss of coolant accident or by the rupture of a line outside the containment which connects to a source of radioactive fluid within the containment.

In general, isolation of a line outside the containment can protect against a rupture of a line inside concurrent with a loss of coolant accident, or will close off a line which communicates with the containment atmosphere in the event of a loss of coolant accident.

Isolation of a line inside the containment will prevent flow from the reactor coolant system or any other large source of radioactive fluid in the event that a piping rupture outside the containment occurs. A piping rupture outside the containment at the same time as a loss of coolant accident is not considered credible.

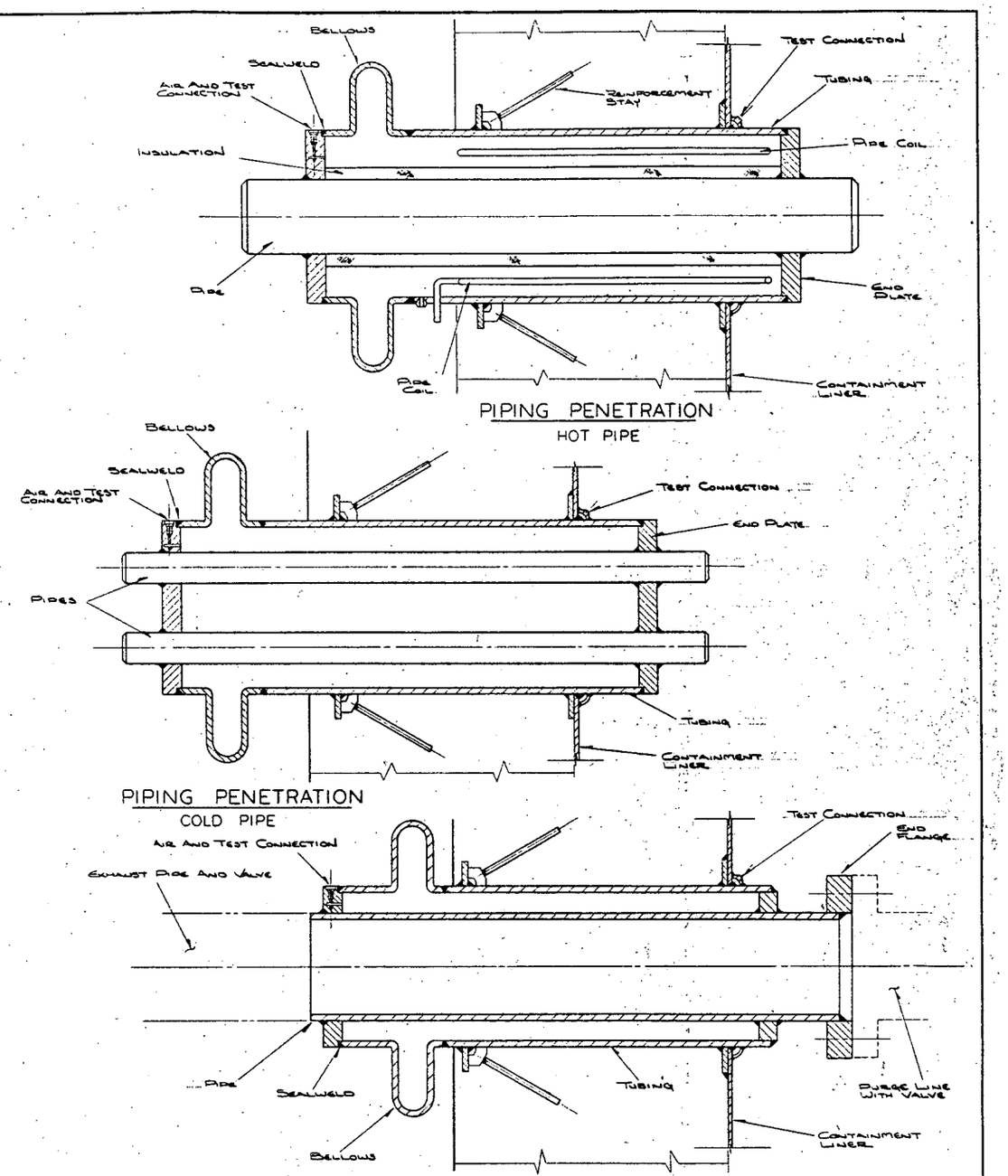
The following measures are provided to assure containment integrity in the event of a loss of coolant accident with the conservative assumption of a concurrent rupture of a pipe inside the containment.

1. No valve is required if the pipe penetrating the containment is part of a closed piping system outside the containment and that piping system is designed for pressures higher than the containment design pressure.
2. A valve will be installed outside the containment in lines which are not part of a closed piping system as in 1) above.
   a. The valve will be remotely operated from the central control room in lines where flow direction is normally out from the containment.
   b. A check valve will be used in the line outside the containment where flow direction is into the containment.
   c. If the line is always isolated when the containment is sealed, a manual valve will be installed in the line outside the containment. This valve will always be closed when the containment is sealed.
   d. As a special case, because of size, the large ventilation purge lines will have a tight-sealing butterfly valve outside the containment which can be closed either manually or automatically upon a signal of high internal pressure or radiation level in the containment.



ELECTRICAL PENETRATION
STRAIGHT - THROUGH CONDUCTOR

ELECTRICAL PENETRATION
JOINTED CONDUCTOR



PIPING PENETRATION
HOT PIPE

PIPING PENETRATION
COLD PIPE

PIPING PENETRATION
PURGE PIPING

FIGURE 4.2.5-1

UNLESS OTHERWISE SPECIFIED THE FOLLOWING TOLERANCES APPLY

SEE PROCESS SPECIFICATION NO CAP595128-1 FOR SUPPLEMENTARY MANUFACTURING INFORMATION

1	10-1-60	Revised O-Ring Sealing Of Electrical Penetrations. Deleted Test Port Connection Cable Gland Fitting Cable Gland Nut
2	10-1-69	

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WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIV. PITTSBURGH, PA. U.S.A.

TITLE: TYPICAL CONTAINMENT PENETRATIONS

SCALE N.T.S. | DIMENSIONS IN INCHES DO NOT SCALE | SUB 2 | ED SK 319443-F

The following measures assure no unrestricted release of radioactive fluid in the event of a line rupture outside the containment.

1. No valve is required for lines which are always closed to the containment atmosphere and which do not connect to a source of radioactive fluid within the containment.
2. For lines connecting to the reactor coolant system or any significant source of radioactive fluid within the containment, a valve (or in special cases where operations permit, a blind flange) will be installed on the line inside the containment.
   a. The valve will be remotely operated from the central control room in lines where the flow direction is normally out from the containment.
   b. A check valve will be installed within the containment in lines where the flow direction is into the containment.
3. As special cases to insure penetration leak-tightness, the large ventilation purge lines will have a second tight-closing butterfly valve inside the containment which can be closed either manually or automatically upon a signal of high pressure or radiation level within the containment. Provisions will be made to pressurize the space between valves during normal operation.

In addition, a valve or blind flange will be installed on the spent fuel transfer tube opening inside the containment to allow pressurization of the tube to prevent containment out leakage.

Auxiliary Facilities

Intraplant communications utilize Bell system equipment with several special features. Station to station calls can originate from any location. In addition, loud speakers and unitized amplifiers couple to the telephone system so that paged messages may be broadcast from any telephone. Loud speakers installed

throughout the containment as well as the rest of the plant ensure the hearing of messages and alarms in all locations.

Two other independent communications systems provide for uninterrupted direction of fuel handling and for instrument calibration. The fuel handling communication system connects the central control room, the containment at the charging floor level and the fuel handling building. The other system permits communication between control points in the containment and the control room for instrument calibration and testing.

The usual maintenance facilities are installed in the containment. These include a reactor containment crane for handling major pieces of equipment. Monorails are located over vital equipment. A fuel manipulator and associated tools are provided for underwater refueling and maintenance of the reactor vessel internals. Underwater inspection facilities are also included.

Service water, demineralized water and electricity are available in the containment at all times. Compressed air and steam are supplied during shutdown.

Access and Occupancy

The containment and the equipment within it are designed for convenient access, refueling flexibility, easy maintenance and personnel safety.

The containment will be completely closed whenever the core is critical or whenever the reactor coolant system temperature is above 200°F and the pressure above 300 psig with the fuel in the core. The containment vessel is normally operated at a slight positive pressure for the purpose of continuously monitoring leakage. No ventilation valves have to be closed in an accident to insure containment integrity.

Limited access to the containment is possible with the reactor at power. Such access is through the double-door, personnel hatch, and personnel are restricted to the annular area between the crane support wall and the containment shell. This area contains the air recirculation system equipment and instrumentation equipment. Shielding and viewing ports are provided so that other important areas can be inspected.

Integrity During Shutdown

Refueling is carried out with reactor coolant in the refueling cavity. The coolant system is heavily borated during refueling such that the core is always in a substantially subcritical condition. Operations which may change core reactivity, such as replacement of fuel or control rods, are performed only when the equipment hatch is in place and when containment integrity can be quickly re-established by closing the personnel hatch, the butterfly valves, and the fuel transfer penetration valve. Only major maintenance operations, not involving changes in core reactivity, are carried out with the equipment hatch open.

4.2.3 Design Criteria

Internal Pressure

The principal design load on the containment structure is the internal pressure created by a loss of coolant accident. The reactor coolant system contains 507,080 lbs. of water at a weighted average enthalpy of 602.1 Btu/lb., for a total energy content of 305,290,000 Btu. In the hypothetical accident this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment.

Additional energy is available for release from the following sources:

1. Stored heat in the core
2. Stored heat in the reactor vessel, piping and other reactor coolant system components
3. Residual heat production
4. Metal-water reaction energy
5. Hydrogen-oxygen recombination reaction

There are several heat sinks available in the containment to absorb a part of this energy and limit the pressure rise. They are as follows:

1. Containment structure
2. Equipment inside the containment
3. The safety injection system (hot leg and cold leg injection)

4. The air recirculation system

5. The containment spray system

All of these sources and sinks are considered in the calculation of the containment pressure transient. The results of the preliminary calculations of transient pressures are given in Section 5.1.

From the results presented in that section, it can be seen that the maximum calculated pressure is less than 45 psig. The design pressure selected for the containment is 47 psig.

No energy contribution from the secondary coolant system is included in the pressure transient calculation. The steam generator supports are so designed that the reaction caused by complete severance of any reactor coolant line will not result in failure of steam or feedwater piping.

Leakage from the containment will be demonstrated to be no more than 0.1 per cent of the containment volume per 24 hour period with design pressure inside the containment.

Stress Criteria

Reinforcing steel used in the reactor containment structure conforms to ASTM A432 with a minimum guaranteed yield strength of 60,000 psi. The following table lists the maximum allowable values of primary and primary plus secondary stresses used for the reinforcing steel.

Load Conditions	Primary*		Primary plus Secondary**	
	Stress	% of Yield	Stress	% of Yield
Operating plus incident	32,000 psi	53-1/3 %	40,000 psi	66-2/3%
Operating plus 0.03 g horizontal earthquake	30,000 psi	50 %	32,000 psi	53-1/3%
Operating plus incident plus 0.03 g horizontal earthquake	40,000 psi	66-2/3 %	40,000 psi	66-2/3%
Operating plus incident plus .10 g horizontal earthquake	48,000 psi	80 %	48,000 psi	80%

*Primary stresses are not considered to include temperature stresses.
**Secondary stresses are considered as follows:

1. Stresses resulting from normal operating temperature gradient inside of containment to outside atmosphere
2. Stresses resulting from the incident temperature effect

The 3/8" steel liner conforms to ASTM A442 with a minimum guaranteed yield strength of 32,000 psi. Under the combination of incident plus earthquake loadings, the internal pressure plus temperature effect causes tension stresses in the concrete wall. For design purposes, it is conservatively assumed that the wall has little or no capacity to resist the tangential shearing forces resulting from the horizontal earthquake loads. The liner alone is designed to resist this tangential shear.

Insulation is provided on the lower portion of the liner so that the combination of compressive stresses resulting from incident temperature and tangential shearing stresses due to horizontal earthquake loading will not result in liner stresses exceeding the yield strength of the material.

Thermal Loads

In the hypothetical loss of coolant accident, the containment structure is exposed to steam at approximately 272°F, or less, within a matter of seconds. The liner reaches the full accident ambient temperature within a few minutes, while the concrete structure, due to its large mass, is still at near-normal temperatures. The concrete then gradually heats up until a thermal gradient exists across the wall. Containment pressure is reduced by the engineered safeguards during this heatup.

The initial temperature rise in the liner causes it to expand relative to the concrete. The liner is studded to the concrete on 24 in. centers to prevent elastic buckling under these stresses.

At the juncture of the cylindrical containment wall with the bottom mat, the reinforced concrete is almost rigid. This prevents expansion of the liner under incident plus earthquake loading which would cause stresses in the liner to exceed the yield point unless special precautions are taken. To maintain stresses in the liner at design levels, the lower portion of the liner is fully insulated, thereby eliminating most of the temperature differential.

The innermost and outermost circumferential reinforcing bars in the thick walled concrete reactor containment are separated by several feet. Temperature gradients occur between these two sets of reinforcing bars during both normal operation and accident conditions. To calculate the stress levels induced by these gradients, temperature profiles will be determined for both normal atmospheric conditions and transient conditions induced by accidents.

Impact Loads

The only impact loads that might act on the containment are those created by a flying missile. Generation of such a missile is extremely unlikely. Most high pressure equipment is located within the 3 ft. thick crane support wall below the 2 ft. thick charging floor, or beneath the refueling cavity missile shield. These reinforced concrete structures will terminate the flight of any conceivable missile. Openings in the charging floor required for ventilation or access are covered by heavy steel grating, which provides adequate missile protection. However, since it is impossible to completely eliminate the possibility of some form of missile striking the containment structure, an analysis of the missile hazard shows that conceivable missiles will not rupture the liner.

Two types of missiles might be generated -- concrete and steel. Because of its lower density and lower strength, a concrete missile must be an order of magnitude heavier than a steel missile of comparable diameter and velocity for it to cause the same impact damage to a steel shell. In the event of the hypothetical accident, however, steel missiles are more likely. Therefore, only steel missiles have been evaluated.

The most hazardous missile for a given mass and velocity is rod shaped, impacting end-on. Rods of various diameters and weights have been investigated.

Missile velocities as high as 100 fps might be generated by rupture of a high pressure coolant loop. This conservative value (see "Penetration of Reactor Containment Shells," Donald E. Davenport; Nuclear Safety, 12/60, Volume 2, Number 2) for missile velocity has been used in the analysis. Formulations and penetration equations developed by the U.S. Army Ballistic Research Laboratories form the basis

for the studies.

Table 4.2.3-1 summarizes the results of the analysis. Inspection indicates that, even at 100 fps, the required weight and dimensions for penetration of the plate thicknesses of interest are absurdly high for missile sizes which can logically be postulated in the reactor containment. The plate thicknesses shown in table 4.2.3-1 bracket the thicknesses of interest for the containment liner and piping systems. Major components, such as the steam generator, have greater shell thicknesses and, therefore, are immune to penetration by the postulated missiles.

Seismic Loads

The nuclear plant is designed on the basis of an earthquake having a maximum horizontal ground acceleration of 0.03 g at zero period. For earthquakes having a horizontal acceleration of 0.1 g at zero period, there will be no loss of function of systems important to safeguards,although in some cases, the stresses may reach or slightly exceed yields. Only horizontal accelerations are assumed to be acting. For the reinforced concrete containment structure, a value of 7 per cent of critical damping is used as the damping factor.

The procedure for calculating the reactor containment earthquake response is based on the Rayleigh method. (See "Design of Multistory Reinforced Concrete Buildings for Earthquake Motions" - Portland Cement Association, page 21.) The containment structure is designed for the equivalent static loads resulting from the above dynamic analysis.

Miscellaneous Loads

The containment structure is capable of withstanding wind loads up to 150 miles per hour, during normal operation.

A load of 30 psf is considered for snow and ice. However, no snow or ice loads are included in combination with internal pressure and seismic loads, because the snow and ice would reduce the stresses in the walls.

The containment will withstand a negative pressure approximately 7.5 psi below the outside atmosphere. However, it is inconceivable that such a pressure

differential could occur. The containment is normally at a positive pressure of 32 to 35 in. Hg abs, for the purposes of continuous leak monitoring. Diurnal temperature fluctuations will not materially affect the containment atmosphere because of the thick concrete walls.

TABLE 4.2.3-1

MISSILE DIMENSIONS AND WEIGHTS REQUIRED
TO PENETRATE PLATE OF VARYING THICKNESSES
AT 100 FPS VELOCITY

Material to be Penetrated						
Reactor Containment Plate, 3/8 in.	Weight, lbs.	25.8	73	134	206	288
	Length, in.	116	82	67	58	52
4 in. Sch 160 Pipe or 0.531 Wall Thickness	Weight, lbs.	43.3	123	226	346	485
	Length, in.	194	138	113	97	87
6 in. Sch 160 Pipe or 0.718 Wall Thickness	Weight, lbs.	68.2	193	354	545	762
	Length, in.	308	216	177	153	136
8 in. Sch 160 Pipe or 0.906 Wall Thickness	Weight, lbs.	96.5	273	502	772	1080
	Length, in.	432	306	250	216	193
10 in. Sch 160 Pipe or 1.125 Wall Thickness	Weight, lbs.	133.7	378	695	1070	1495
	Length, in.	600	424	346	300	268

4.2.4 Structural Design

General

There are no codes directly applicable to the design of the reinforced concrete containment structure. Accepted ACI building and ASME pressure vessel codes, as well as state and national codes, are used as a guide in arriving at a safe criteria for design purposes.

Materials

Table 4.2.4-1 lists the materials used in the construction of the reactor containment.

Reinforcing Steel

The reinforcing bars have a guaranteed yield point of 60,000 psi and the minimum elongation per two inches is 9 per cent. Check analysis is performed for each heat. Physical property tests are also made on each heat, one at the mill and

one just prior to erection. Certified test reports are retained on all reinforcing bars. In addition, each bar is visually checked for size, surface condition and other fault prior to erection.

Connections between reinforcing bars and made either by arc welding or by the use of the Cadweld process. This process is based on a special sleeve which is filled with a molten bronze alloy. When cooled, the connection will develop a minimum of 125% of the yield strength of the reinforcing bar, because of the shear strength of the filler metal.

Concrete

All concrete materials conform to ACI and ASTM specifications. Low alkali Portland cement and a water reducing agent are used.

Liner

With the exception of the equipment and personnel hatches, none of the liner components are subject to low ambient temperatures. Normal internal temperatures will be maintained between a minimum of 50°F and a maximum of 120°F. The liner material is ASTM A442, Gr. 60, which has a yield point of 32,000 psi and is a low carbon/high manganese steel made with a fine grain structure that exhibits an NDTT lower than -20°F without heat treatment in thicknesses less than one inch. This material is specified primarily to insure ductility during winter fabrication and during the freestanding pressure and leak tests that are performed on the liner before the reinforced concrete structure is poured around it. Because the equipment and personnel hatches can be exposed to low ambient temperature, they are constructed of ASTM A201, Gr. B., firebox steel normalized by heating to 1700°F and cooling in still air and Charpy tested to a minimum of 15 ft-lb at -50°F. One weld specimen is prepared for each 50 feet of weld performed by each welder and Charpy tested fo a minimum of 15 ft-lb at -50°F.

No specific corrosion allowance is included for the liner, which is a minimum of 1/4" thick. The inside surface of the liner is covered with a three-coat system of sprayed vinyl plastic paint to a total thickness of 6 to 8 mils. This

type of paint is particularly effective for resisting atmospheric corrosion. The outer surface of the steel liner is in direct contact with concrete which, due to its alkaline nature (pH of at least 12) provides adequate corrosion protection for the steel in contact with it.

The mat is founded on firm rock, resting directly on the rock or on fill concrete which rests directly on firm rock.

The mat transmits all loads from the reinforced concrete pressure container and all internal structures to the firm rock.

TABLE 4.2.4-1

REACTOR CONTAINMENT MATERIALS OF CONSTRUCTION

Item	Material Specification
A. Liner	
Shell, Bottom, and Dome Plates	ASTM A442, Gr. 60
Piping Penetration Sleeves	ASTM A333, Gr. 0
Piping Penetration Reinforcing Rings	ASTM A442, Gr. 60
Piping Penetration Sleeve Reinforcing Bar Anchoring Rings and Plates	ASTM A442, Gr. 60
Rolled Shapes	ASTM A131, Gr. C
Reinforcing Bar Bridging Rings	ASTM A204, Gr. C, Firebox normalized
Reinforcing Bar Anchoring Ring and Plates	ASTM A300, Cl. 1, Firebox A201 Gr. B
Equipment Hatch Insert	ASTM A300, Cl. 1, Firebox A201 Gr. B
Equipment Hatch Flanges	ASTM A300, Cl. 1, Firebox A201 Gr. A
Equipment Hatch Head	ASTM A300, Cl. 1, Firebox A201 Gr. B
Personnel Hatch	ASTM A300, Cl. 1, Firebox A201 Gr. B
B. Welding Electrodes	
Material Joined	**Specification**
Carbon Steel to Carbon Steel	ASTM E7018
Stainless Steel to Stainless Steel	ASTM E308
Carbon Steel to Stainless Steel	ASTM E310
C. Concrete Shell and Interior Structure	
Reinforcing Steel	ASTM A432, Designation 14s and 18s
Cement	ASTM C150, Type II low alkali
Structural Steel	A-36

4.2.5 Penetrations

General

All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts and access hatches contain positive pressure zones which prevent outleakage under all conditions, including a hypothetical accident involving major rupture of the reactor coolant system. Typical electrical and pipe penetrations are shown by Figure 4.2.5-1.

In general, a penetration consists of a sleeve imbedded in the reinforced concrete wall and welded to the containment liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the joint. The pipe, electrical conductor cartridge, duct or access hatch passes through the imbedded sleeve and the ends of the annulus are closed off. The annuli are pressurized by an outside air supply to at least 47 psig, the maximum pressure expected during a hypothetical accident. The plant air supply has as a backup a supply of nitrogen gas capable of 24 hours of service.

Electrical Penetrations

Electrical conductors penetrating the containment range in size from No. 20 AWG stranded thermocouple leads to 7/8 inch solid copper rods. As shown by Figure 4.2.5-1, electrical penetrations consist of a sealed cartridge inserted into the sleeve imbedded in the reinforced concrete wall; the sleeves are 8-inch steel pipe.

The basic penetration cartridge consists of a 7-inch O.D. steel tube with welded end plates, through which pass the sealed conductors. The cartridges are of two types, the first being for straight-through conductors, and the second type being for cables which are jointed within the cartridge. The cable joints within the cartridge are of the sealed bayonet type with compression drawup. After the cartridges have been assembled, they are bench tested with Freon gas for leak tightness and then inserted into the sleeves. The cartridges are held tightly in the penetration sleeves by bolted flanges sealed with double O-rings at the inside

of the containment wall. The cartridges are seal-welded to expansion bellows on the outside of containment to completely close the annulus between the cartridges and sleeves.

The spaces between the O-rings, the annuli and the insides of the cartridges are pressurized by the outside air supply. Each of these spaces can be disconnected from the air supply for individual (and independent) halogen leak test.

Piping Penetrations

Approximately one hundred penetrations are provided for fluid carrying and for air purge ventilating piping. Several capped penetrations are included as spares. Details of the penetrations are shown on Figure 4.2.5-1.

Most pipes penetrating the containment connect to equipment inside and outside of the containment, and are for either high-temperature or moderate-to-low temperature service. Other pipes, such as for purge air, connect the containment volume to the outside atmosphere.

In all cases, a piping penetration consists of an imbedded sleeve with the ends welded to the penetrating pipe; the ends are blind flanges and an expansion bellows in the sleeve on the outside of the containment permits relative motion of the pipe with respect to the sleeve. The volume between pipe and sleeve is pressurized by the outside air supply, and each volume can be disconnected from the air supply for individual halogen leak test.

Penetrations for high-temperature pipes (steam and feedwater, for example) have cooling coils in the spaces between the pipes and sleeves to protect the sleeve and the surrounding containment wall from excessive temperatures.

The modes of isolating these pipes during a high-pressure containment incident are covered in Section 4.2.2.

Access Port and Air Lock

An equipment hatch of approximately 16 foot I.D., built of welded steel and having a double gasketed flanged and bolted dished door, is located at grade (see Figure 4.2.5-2). Equipment up to and including the size of the reactor closure

O-ring seals can be moved into and out of containment through this hatch. The dished door is handled by an auxiliary hoist extended from the frame of the containment polar crane. The hatch barrel is imbedded in the containment wall. All weld seams at the containment liner joint, at the flanges and in the dished door have test channels for periodic leak test. The space between the double-gaskets of the door flanges is continuously pressurized by the outside air supply.

A 7-foot I.D. personnel hatch penetrates the dished door of the equipment hatch. The personnel hatch is a double door, hydraulically-latched welded steel assembly with an 18-inch emergency manhole in the inner door. An equalizing valve connects the personnel hatch with the interior of the containment vessel for the purpose of equalizing pressure in the personnel hatch with that in the containment. Emergency access to the personnel hatch from inside the containment is available in the event of a hydraulic failure of the inner door; a special tool, located near the door on the containment side, is used to unlatch the inner door. Once inside the personnel hatch, the inner door may then be closed. Should the outer door of the personnel hatch fail to open due to a hydraulic malfunction, a telephone in the hatch may be used to summon help. The outer door may also be opened from outside by means of another special tool, stored outside the containment, permitting egress from the personnel hatch. The inner door of the personnel hatch is provided with an 18-inch manhole, bolted from inside the hatch, to permit access to the containment in the event of a failure of the inner personnel hatch door. Hatch closures are of the double-gasketed type and spaces between the double-gaskets are continuously pressurized by the outside air supply. Pressure is relieved from the double-gasket spaces prior to opening the joints and during the time that personnel are within containment.

Fuel Transfer Penetration

A 20-inch O.D. fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration, as indicated by Figure 4.2.5-2, consists of a 20-inch stainless steel pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer

tube and connects the reactor refueling canal with the spent fuel pit. The tube is fitted with a standard gate valve in the refueling canal and a sluice gate valve in the spent fuel pit. These valves prevent leakage through the transfer tube during accident conditions. The outer pipe is welded to the containment liner and provision is made, by use of a special seal ring, for freon gas leak testing all welds essential to the integrity of the penetration. Bellows expansion joints are provided on the outer pipe to compensate for any differential movement between the two pipes.

The annulus between the inner and outer pipes on the containment end of the transfer tube is continuously pressurized by the outside air supply.

Construction Openings

One 27-inch I.D. opening with a bolted and double-gasketed cover is located in the dome at the top of the vessel. This opening is for construction ventilation and will be permanently closed at the conclusion of the construction work. After the cover is bolted in place, removable external concrete shielding is installed over the opening. The space between the gaskets is pressurized by the outside air supply.

Large equipment will be passed into the reactor containment through a temporary opening in the side wall. This opening will be permanently closed after this equipment is in place and before the containment is completed and finally tested.

4.2.6 Inspection and Testing

Preoperational Inspection and Tests

Preoperational inspections and tests are performed in several stages which lead finally to the leak rate tests.

Fabrication and Construction Testing

The containment mat and lower portion of the side walls are erected and lined early in the project. This segment of the liner is tracergas tested using test channels as shown in Figure 4.2.2-1. The test channels are structural steel channels, or other appropriate shapes, which are installed over every welded joint below ground grade. The test channels are welded so as to form a space into which freon is

admitted and the channel-to-liner weld tested for seam leakage. The test channels do not form a single continuous channel but are segmented for convenience in testing. Test gas is introduced through threaded connections at appropriate points to insure a homogeneous test gas throughout the channel. After testing, the gas is expelled from the channels and the threaded connections plugged. All channels are available for retesting as required throughout plant life except those in the bottom mat which are covered by 24 inches of concrete.

Periodic compression tests are made on the concrete as placed at the construction site to assure compliance with specifications.

To insure the integrity of the Cadweld connections as actually installed on the containment reinforcing, quality control utilizes a random sampling procedure. The Cadwelds randomly selected in the field are removed and tested to destruction. A plot of the test results is maintained to demonstrate the integrity of the joints. From this information, the average yield strength, and deviations from it, will be computed. Calculation of the statistical tolerance limit is expected to demonstrate a confidence level that at least 99% of the Cadwelds will have a minimum ultimate strength of 75,000 psi if all were tested to destruction.

In addition, a continuing statistical sampling procedure will be used throughout the construction to insure that the process is not deviating from the originally established distribution data.

Shop fabricated liner components are dimensionally checked and shop welding is spot radiographed at least to the requirements of paragraph UW-52 of the ASME Code for Unfired Pressure Vessels. All other welds are dry checked.

The containment liner is next completed above grade as a free standing vessel. All welds in the side walls and dome are double butt welds.

During erection, continued dimensional checks are made and all welding above grade is spot radiographed at least to the requirements of paragraph UW-52 of the ASME Code for Unfired Pressure Vessels. All other welds are dye checked. Inspections are carried out by both the erector and the engineers.

All welding and the qualification of welding procedures and welding operators is in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. Acceptance standards and procedures for radiographic inspections are in accordance with Section I, paragraph p-102-H of the ASME Boiler Code. Acceptance standards and procedures for liquid penetrant inspection are in accordance with ASTM standard E-165-60T.

After completion, the entire liner is tested at 8 psi gage with air and 10 wt % freon. This test results in a tensile stress in the liner of approximately 17,000 psi. All welds and gasket joints are inspected and checked for leakage using a halogen leak detector on its most sensitive range; leaks greater than 0.05 cubic inches per day can be found by this technique. Any leaks found are chipped out and repaired and the test repeated as necessary.

After the gross freon test, polyethylene test channels are taped over each of the welded liner seams which are not already covered with steel test channels. The polyethylene channels are indicated by Figure 4.2.2-1. Concrete is then placed for the side walls and dome. The polyethylene channels provide a space between the liner and concrete which can be used for future leak testing. To use these channels, the liner will be drilled and tapped from the inside to permit the introduction of freon into the channel area. The seams can then be checked using a halogen leak detector. This testing is not intended to replace periodic or continuing leak rate testing, but will permit a check of individual welds should the need arise.

With the liner completed and the concrete poured and cured, the container is tested with air at 115 per cent of design or 54 psig. This pressure is held for one hour and insures the structural integrity of the container.

Initial Leak Rate Test

The design leak rate is 0.1 per cent of the contained volume in 24 hours at 47 psig. It has been demonstrated that, with good quality control during erection, this is a reasonable requirement. Large containers approximately the size of this vessel normally show leak rates of the order of 0.1% per day or less, however, this leakage is believed to occur at penetrations and closures. Leakage through penetrations and closures is blocked by the Containment Penetration Pressurization System.

The initial leak rate test is performed, after completion of construction and installation of penetrations, at five pressures up to and including the design value of 47 psig. This will establish the variation of leak rate with pressure. The test is performed using the reference volume method. Every effort is made to demonstrate the leaktightness of the reference volume system. The entire reference volume system is pressurized to a minimum of 100 psi gage air containing 20 wt % freon. All reference volume joints are bagged with plastic and the system held at this pressure for 48 hours. The reference volume system, especially the joints, is checked with a halogen leak detector to demonstrate integrity.

In addition to the usual calculation of leak rate as a function of pressure differential, air is returned to the reactor containment at the conclusion of each test step through a precision gas meter until the differential pressure is returned to its original condition. This provides a check on the calculated leak rate. Reactor containment ambient temperature and humidity are also measured during the course of the test to provide further backup information.

The initial leak rate test consists of establishing a leak rate at approximately 47, 35, 23, 12 and 5 psi gage. Because the containment is a thick walled concrete structure, short term temperature or meteorological variations should not have any appreciable effect on the containment ambient temperature and pressure. It should, therefore, be possible to establish meaningful leak rates in a shorter term test than might be required on a bare steel vessel. The containment will be held at each test pressure for a minimum of 12 hours to determine the leak rate at

that pressure. If the leak rate determination is inconclusive after 12 hours, the test will be continued. Following a successful test, pressure will be lowered to the next test point slowly over a period of approximately 12 hours. The results of these tests will be used to establish flow models by which the measurements made at low pressure can be extrapolated to the design conditions.

During these leak-rate tests the penetration positive pressure zones which will be normally pressurized by the penetration pressurization system will remain unpressurized.

Postoperational Tests

The containment is equipped with a continuous leakage monitoring system. This system consists of the same arrangement previously described under the initial leak rate test and is used in the following manner.

1. The container is charged with air to a pressure of approximately 1 psig.
2. Over a period of time, leakage from the containment causes a drop in pressure.
3. When pressure reaches a prescribed limit the containment is recharged to 1 psig.
4. The air charge is metered to provide a direct measure of leakage over the period since the last charge.
5. Periodic checks are made of containment pressure, temperature and humidity, to detect development of abnormal leak rates.

The possibility of in-leakage to the containment through the pressurized penetrations is ruled out by periodic leakage tests of the penetration pressurization system. This system is tested by isolating it from the pressure source and observing rate of pressure decay. In addition, during normal operation, an integrating air supply flow recorder indicates abnormal system air consumption. Detection of system leak rates of a magnitude significant to the results of the continuous containment leakage monitoring will result in location and repair of the pressurization system

As previously stated, provisions are made for leak tests of all penetrations. Such tests will be performed whenever there is any indication of excess leakage. The containment temperature, pressure and humidity are continuously monitored to provide for accurate determination of the containment atmospheric properties during operation. These measurements, as well as the permanently installed reference volume system, will be used to determine changes in containment leak rate (see Section 4.2.5). Special devices are provided as part of the air distribution system for continuously monitoring the penetrations.

Each pressurized volume of each penetration can be disconnected from the outside air supply for individual and separate halogen leak test.

Provisions have been made to pressurize the personnel hatch with tracer gas in order to test the tightness of the hatch and door gaskets.

4.2.7 Containment Penetration Pressurization System

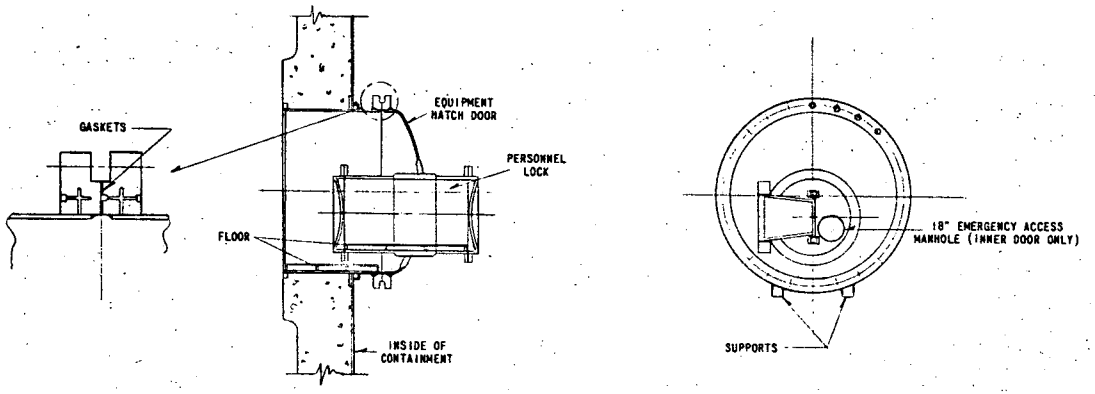
The function of the containment penetration pressurization system is to prevent leakage of containment air through penetrations under all conditions by supplying air above containment design pressure to the positive pressure zones incorporated in the penetration design. The design features of the penetrations are described in Section 4.2.5.

The containment penetration pressurization system is shown on the Process Flow Diagram, Figure 4.2.7-1. The system utilizes a regulated supply of clean and dry compressed air from the plant compressed air system to maintain pressure in all penetrations whenever plant operating conditions require the containment to be closed.

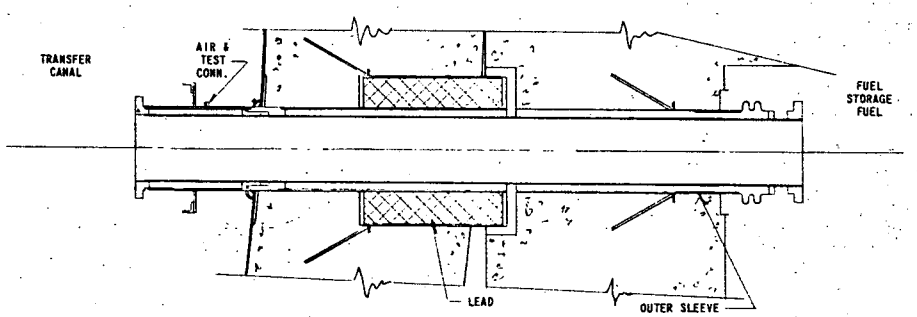
Nitrogen cylinders provide a standby source of gas pressure. The regulators on the nitrogen supply are set to deliver at a slightly lower pressure than the normal regulated air supply pressure. Thus, in the event of failure of the normal air supply pressure, the penetration pressure requirements will automatically be maintained by the nitrogen supply.

Each penetration air supply line can be isolated for infrequent halogen or helium leak testing. A capped tubing connection is provided in each supply line to allow injection of the leak-test gas.

Leakage from the system (and potential leakage from penetrations) is checked by continuous measurement of the integrated makeup air flow.



EQUIPMENT AND PERSONNEL HATCH



FUEL TRANSFER PENETRATION

FIGURE 4.2.5-2

DFTM.	J. Robinson	7-27-65	WESTINGHOUSE ELECTRIC CORPORATION ATOMIC POWER DIV., PITTSBURGH, PA., U.S.A.
CHKD.			TITLE: HATCHES AND FUEL TRANSFER PENETRATIONS
DES. ENG.	M. DeRosa	7/27/65	
MFG. ENG.			
MTLS. ENG.			
APP.			SCALE
APP.			DIMENSIONS IN INCHES
DFTG. SUPV.	R. Slaughter	7/27/65	DO NOT SCALE

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SUB 1

NEXT ASSY.

REF. DWG.

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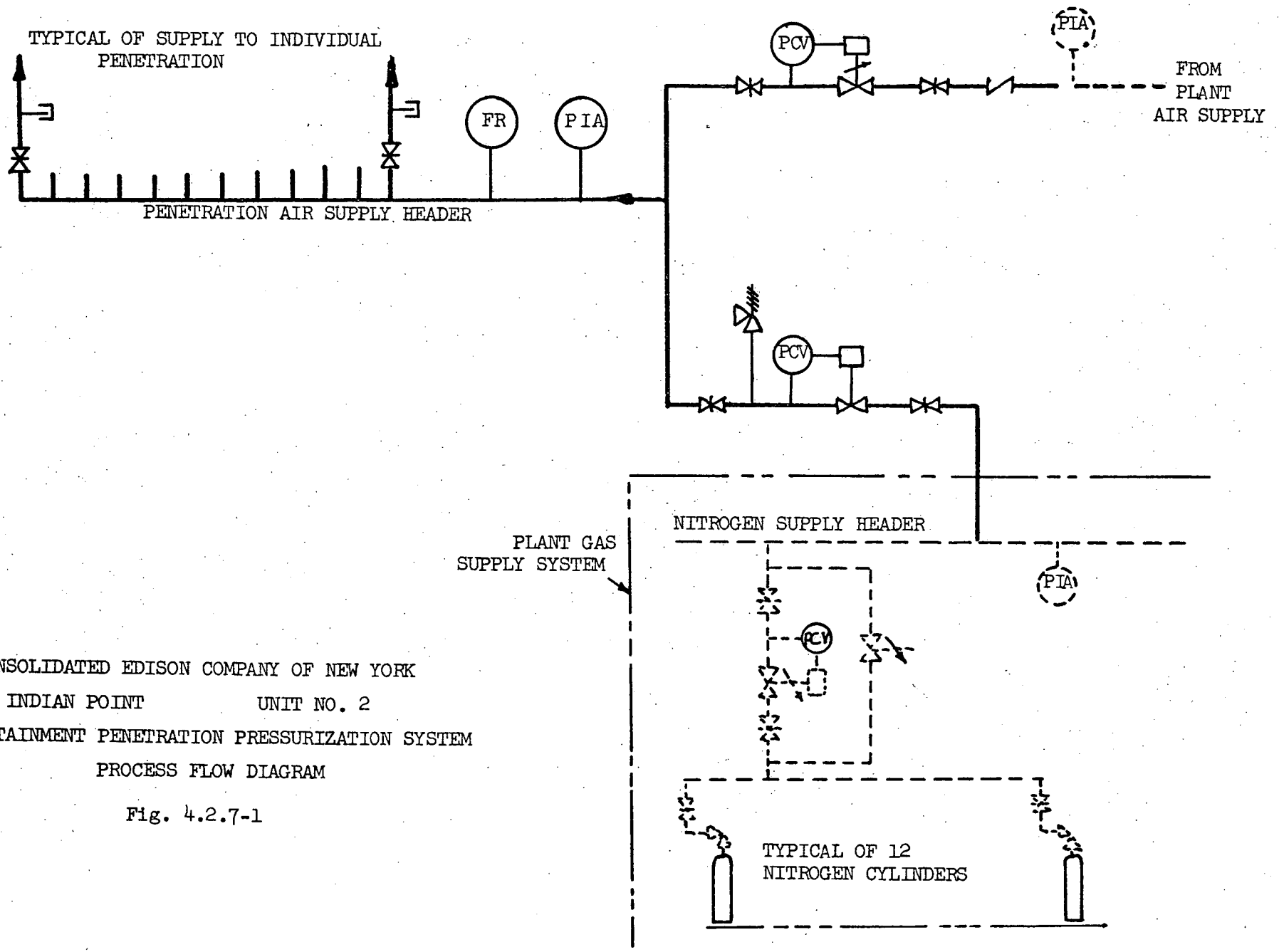


CONSOLIDATED EDISON COMPANY OF NEW YORK

INDIAN POINT UNIT NO. 2

CONTAINMENT PENETRATION PRESSURIZATION SYSTEM

PROCESS FLOW DIAGRAM

Fig. 4.2.7-1

4.3 Air Recirculation Filtration

4.3.1 Performance Objectives

The air recirculation filtration is provided by the Ventilation System as a back-up for the pressurized penetrations and testable welds in the containment. The filtration function is to reduce the concentration of fission products in the containment atmosphere and thereby limit the potential release of fission products to the environment following a loss of reactor coolant.

Air recirculation filtering capacity is sufficient to limit the two hour adult thyroid dose to less than 300 rem at the exclusion distance of 520 meters for the following conditions:

1. Containment leak rate of 0.1% per day. This assumes no credit for the effectiveness of pressurized penetrations and testable welds in the containment.
2. Site meteorology corrected for building wake effects.
3. No filtration of organic iodine - This assumes no credit for the ability to filter organic forms of iodine at 100% relative humidity with the highly activated coconut charcoal recently demonstrated at ORNL.
4. Fission product releases for complete core meltdown per TID 14844. This assumes no credit for safety injection in limiting core meltdown.
5. Partial effectiveness of the filtration equipment. This assumes one of the installed units is unavailable at the time of the loss of coolant.

4.3.2 Description

The air recirculation filtration is accomplished using four direct driven centrifugal fans, in the Ventilation System which are also utilized in air recirculation cooling. Each has a capacity of 65,000 cfm under accident conditions. Four filtering units, one for each fan, are provided. Each filtering unit includes a

moisture separator, an absolute filter and a highly activated charcoal filter in series.

During normal operation, the filter units are not in use. Motor operated louvers in each unit close off the filter inlet and allow recirculation air flow to by-pass the filter. In this way, the high degree of charcoal activation is maintained. Under accident conditions, the louvers are automatically operated to block the by-pass flow around the filters and to direct recirculation air flow through the filter units. The discharge of air is through the air recirculation duct work which is designed to give even distribution throughout the containment.

The charcoal filters are provided with removable test elements for use in verifying continued effectiveness of these filters. Provisions for periodic in-place testing will also be included.

4.4 Air Recirculation Cooling

4.4.1 Performance Objectives

The air recirculation cooling capacity of the Ventilation System is sized to remove the normal heat loss from equipment and piping in the reactor containment during plant operation and to remove sufficient heat from the reactor containment following the initial loss of coolant pressure transient to keep the containment pressure from exceeding safe limits. The fans and cooling units continue to remove heat after the loss of coolant and reduce the pressure close to atmospheric within the first 24 hours.

The fans, motors, electrical connections, and all other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a loss of coolant.

4.4.2 Description

The air recirculation cooling function is accomplished using four direct connected motor driven centrifugal fans, distribution duct work, and four cooling coil assemblies. The ventilation system also includes steam heating coils for containment heating when required.

The recirculation fans are designed to run continuously. Each fan draws air from the annular space between the crane support wall and the containment wall. In normal operation, the filter units are by-passed and the air passes through the cooling coils, through the fans, and through the distribution ducts to the various parts of the containment. Following a loss of coolant, the air passes through the filter units after being cooled and is then distributed.

The air recirculation units are located in the annulus between the crane support wall and the containment wall in separate quadrants. The shielded location makes inspection of the equipment possible during power operation if required.

Three of the four fans and coolers have sufficient capacity to hold the containment pressure within design limits after a loss of coolant.

5.0 SAFETY ANALYSIS

5.1 Containment Integrety

5.1.1 Method of Containment Transient Analysis

ENERGY RELEASE TO THE CONTAINMENT

If a reactor coolant system rupture is assumed to occur, the following energy additions to the containment are considered: (1) initial coolant blowdown, (2) continued boiling of residual water remaining in the reactor vessel after the initial coolant release, (3) burning in the containment atmosphere of hydrogen gas which is produced as a result of any zirconium-water reaction in the core, and (4) transport of sensible heat stored in the core (as a result of fission product decay and zirconium-water reaction) by flashing of safety injection coolant prior to the time that safety injection coolant reaches the subcooled condition in the core.

Coolant Blowdown Energy

For the assumed rupture, the loss of coolant adds a large amount of energy to the containment. The rate of energy addition to the containment is limited by the location and size of the assumed break and the pressure and temperature in the reactor coolant system. Calculations have shown that the hypothetical accident (complete severance of a coolant pipe connected to the reactor vessel and free discharge from both ends) results in a more rapid energy input to the containment and produces a higher peak pressure in the containment than any of the smaller breaks which can be assumed. The remaining discussion applies to the hypothetical accident.

For this assumed break, the pressure in the reactor coolant system falls from operating pressure to the peak pressure in the containment in 12 seconds. In the same period of time the liquid volume in the system drops to a low value. The rate of coolant energy release to the containment is computed by use of the

LOCO[1] digital computer code. The computation determines mass and enthalpy flux through the break as a function of time.

In the calculation the nature of the discharge process is characterized by two phases in which appropriate equations and thermodynamic properties are used to satisfy mass, energy and momentum requirements as a function of pressure and time. Mass and energy balances written for the reactor coolant contents, in conjunction with equations describing the discharge flow rate, provide the basis for establishing energy and mass blowdown into the containment.

Phase I: Discharge of subcooled water through the break results in a rapid system depressurization to saturation pressure. During the period of subcooled blowdown, the reactor trip is initiated by low pressurizer pressure. Prior to the trip, full reactor power is added to the coolant but this heat is removed by forced convection heat transfer in the steam generator tubes. After the reactor trip is completed, with an allowance for trip delay, core residual heat (delayed neutron residual fission plus fission and capture product decay) and sensible heat are transferred to the coolant. Sensible heat includes the heat stored in both the core and core structures, and a portion of the thick metal in the reactor vessel.

Phase II: Further expansion of the coolant water through the rupture results in a change to flashing two-phase flow after a very short time, and discharge of coolant into the containment is restricted by choking flow. Energy is added to the coolant from the core and metal parts, which are above saturation temperature, and is removed by loss of coolant from the vessel and by heat transfer to the steam generator secondary water. When the reactor coolant water level falls

1. WCAP-1678, "LOCO - Digital Computer Code for Loss-of-Coolant Accident Analysis."

below the steam generator tube inlet level, all heat removed from the coolant is by flow of coolant through the break into the containment.

Discharge of the homogeneous steam-water mixture continues to add mass and energy to the containment until almost no water remains in the system. Because of the high blowdown rate for a complete severance of a large reactor coolant pipe, the rapid steam evolution results in transport of large quantities of entrained liquid. A distinct interface may not exist as the core becomes uncovered. The computer calculation shows, and indeed it is likely, that very little water will remain in the bottom of the reactor vessel because of the rapid blowdown.

Energy from Hydrogen-Oxygen Reaction

During the period between completion of the initial blowdown and addition of safety injection water, the transfer of heat into the containment vessel by convection from the reactor coolant system insulation and by the small residual steam flow is relatively ineffective. Consequently, most of the residual heat generation serves to raise the temperature of the core, internals structure, and reactor vessel. As the zircaloy cladding temperature in any portion of the core rises above the threshold at which a significant rate of reaction occurs, additional energy is produced and stored inside the reactor vessel. Unless the steam evolution rate exceeds the maximum possible consumption rate established by the parabolic rate law, all steam evolved is considered to contribute to the core stored heat by reacting with zirconium and the energy addition to the containment during this portion of the transient is a result of the burning of the hydrogen generated by the zirconium-water reaction.

The released hydrogen gas, which is well above the spontaneous ignition temperature, migrates to the containment vessel as it is produced and burns in the air-steam mixture. The hydrogen-oxygen reaction is the only significant source of

energy reaching the containment from the zirconium-water reaction between the time that blowdown is complete and safety injection starts.

If any zirconium-water reaction occurs before cooling by safety injection becomes effective, only a fraction of the total zirconium available for reaction will be consumed, because the safety injection system is actuated in a very short time and the core temperature rise does not progress to a great extent. The energy released from this reaction does not enter the containment concurrently with the blowdown energy, since cladding temperature does not increase during blowdown. An influx of safety injection water is necessary to liberate the stored heat and safety injection cooling does not begin until after blowdown is complete.

Energy from Flashing Safety Injection Coolant

During the coolant blowdown phase, the pressurizer low pressure signals cause the reactor trip and also actuate safety injection in conjunction with low pressurizer water level signals. The delay in opening injection flow paths and starting injection pumps results in a short period of core heating.

Residual heat generation and the zirconium-water reaction continue to increase the stored energy in the core and reactor vessel until safety injection water reaches the vessel. After safety injection water reaches the reactor vessel, the thermal energy stored in the core, reactor vessel and internal structures is released into the containment as the injection water is evaporated. Evaporation of the injection water continues to add energy to the containment until subcooled heat transfer is achieved in the core and vessel at essentially containment saturation conditions.

Long-term cooling of the core is accomplished by terminating injection from the large refueling water storage tank and recirculating spilled coolant to

the core. Once the core is covered again by the injected water and the reactor coolant system is cooled and depressurized, spilled coolant is returned from the containment vessel sump through a heat exchanger and into the reactor vessel.

CORE THERMAL TRANSIENT

Evaluation of the core thermal transient following a loss-of-coolant accident involves predicting the pellet-clad temperature as a function of time, using heat source and heat sink equations with proper accounting for changes in physical properties of the two as temperature changes.

A transient digital computer program, LOCTA-R, is used to evaluate fuel pellet and cladding temperatures following a loss of coolant accident.

The starting point for calculation of the clad and fuel pellet temperature transients is the coolant blowdown transient following a loss-of-coolant accident. The temperatures are calculated from the solution of the heat source and conduction equations for the fuel rods. Zirconium-water reaction energy evolved in the cladding is added to fuel element initial sensible heat and residual heat storage, to obtain the time and space variation of temperatures.

The calculation is based upon a heat balance, which considers heat transfer between the fuel pellet and cladding by specifying a gap film coefficient, and between the cladding and the coolant by specifying a coolant film coefficient and sink temperature. The calculation is performed at a point in the fuel, and considers residual heat generation in the fuel and the generation of reaction heat in the cladding. Three fuel pellet nodes and one clad node are employed in the calculation over a unit length; axial conduction is assumed to be zero. Physical properties (e.g., thermal conductivity, specific heat of the fuel, clad, and oxide film) are varied as function of temperature. By summing the calculations for various points in the core (which are expressed as fractions of the hot spot heat flux), the overall core heat source is established. One hundred per cent of

the zirconium mass is considered available for reaction.

The zirconium-water reaction is accounted for by use of the parabolic rate equation and a specified heat of reaction. When clad temperature reaches approximately 1800°F (1000°C) as a result of residual heat generation the zirconium-water reaction is initiated.

Results from this computation include the average fuel pellet and clad temperature at any point, the integrated residual and reaction heat, heat release to the sink, and the per cent of zirconium that reacted, as a function of time.

5.1.2 Evaluation of Containment Pressure and Temperature Transients

Calculation of containment pressure and temperature transients is accomplished by use of the digital computer code COCO (Containment Code). The method of analysis used in this code is described briefly in the following paragraphs.

The analytical model used is restricted to the containment volume and structure. Transient phenomena within the reactor coolant system affect containment conditions by means of convective mass and energy transport through the pipe break. For analytical rigor and convenience, the containment air-steam-water mixture is further separated into two systems.

The first system consists of the air-steam phase, while the second is the water phase. Sufficient relationships to solve the problem are provided by the equations of conservation of mass and energy as applied to each system, together with appropriate equations of state and boundary conditions. As thermodynamic conditions may vary during the transient, the equations have been derived for all possible cases of superheated or saturated steam and subcooled or saturated water. The following are the major assumptions made in the analysis:

1. Discharge mass and energy flow rates through the reactor coolant

system break are established from the coolant blowdown and core thermal transient analysis (described in the preceding paragraphs).

2. At the break point, the discharge flow separates into steam and water phases. The water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the containment.

3. Spatial dependence is ignored. The steam-air mixture has uniform properties, as does the water phase. More specifically, thermal equilibrium between the air and steam is assumed.
This does not imply thermal equilibrium between the steam-air mixture and the water phase.

4. Air is taken as an ideal gas, while compressed water and steam tables are employed for water and steam thermodynamic properties. During the transient, there is energy transfer from the air-steam and water systems to the containment structure and equipment within the shell. Provision is made in the analysis for heat transfer through and heat storage in both interior and external walls. The film heat transfer coefficient between the containment contents and a wall surface may be specified either as a function of time or temperature difference.

Provision is also made in the analysis for several engineered safeguards including ventilation fan coolers, internal spray, and recirculation of sump water. To account for other miscellaneous forms of heat release or removal, functions giving heat input or removal versus time may also be supplied.

The containment pressure transient curve for the maximum hypothetical accident (29 inch I.D. double ended break) with the full safety injection system operating and 3 of 4 ventilation fans and coolers operating is shown on Figure 5.1-1.

The full safety injection operation includes three high head safety injection pumps plus two low pressure (residual heat removal) pumps, and two charging pumps.

Twelve seconds after the assumed loss-of-coolant accident, the containment pressure peaks due to the reactor coolant system blowdown and begins to drop due to the heat absorbed by the fan-coolers and the steel and concrete structure of the containment and internal structure. Twelve seconds after the accident the safety injection system is initiated and borated water from the refueling water storage tank is injected into the core via the reactor coolant loops. At this time the safety injection flow releases the core stored heat, residual heat, and heat of the thick and thin metal parts of the reactor coolant system to the containment. Any zirconium water reaction energy and the accompanying burning of hydrogen generated by this reaction starts at about twenty seconds.

This rapid release of heat becomes greater than the rate of heat absorption by the steel, concrete, safety injection water and ventilation fan-coolers and the containment pressure begins to increase. At approximately 280 seconds the core is covered with safety injection water and the zirconium water reaction is stopped. Continued safety injection removes heat from the core by adding to the sensible heat of the injected fluid. The containment pressure continues to decrease rapidly. Operation of the recirculation heat exchanger maintains the pressure at a low value throughout the remainder of the post-accident period. This phase of the transient is not shown on Figure 5.1-1.

Figure 5.1-1 also shows the containment pressure transient that would occur for the first 2000 seconds following the maximum hypothetical accident if no engineered safeguards operate. In this case no safety injection flow or ventilation cooling was assumed to occur.

During the first fifteen seconds of this assumed loss-of-coolant accident the containment pressure transient is essentially the same as described

with safety injection and ventilation cooling. The zirconium water reaction starts approximately twenty seconds after the accident and is assumed to proceed according to the generally accepted parabolic rate law. It is conservatively assumed that sufficient water is present to carry on this reaction. The hydrogen released by this reaction is assumed to burn as it enters the containment. This results in the gradual rise in the containment pressure. After about 1300 seconds the heat released to the containment is exceeded by the heat transfer to the containment structure and the pressure begins to decrease.

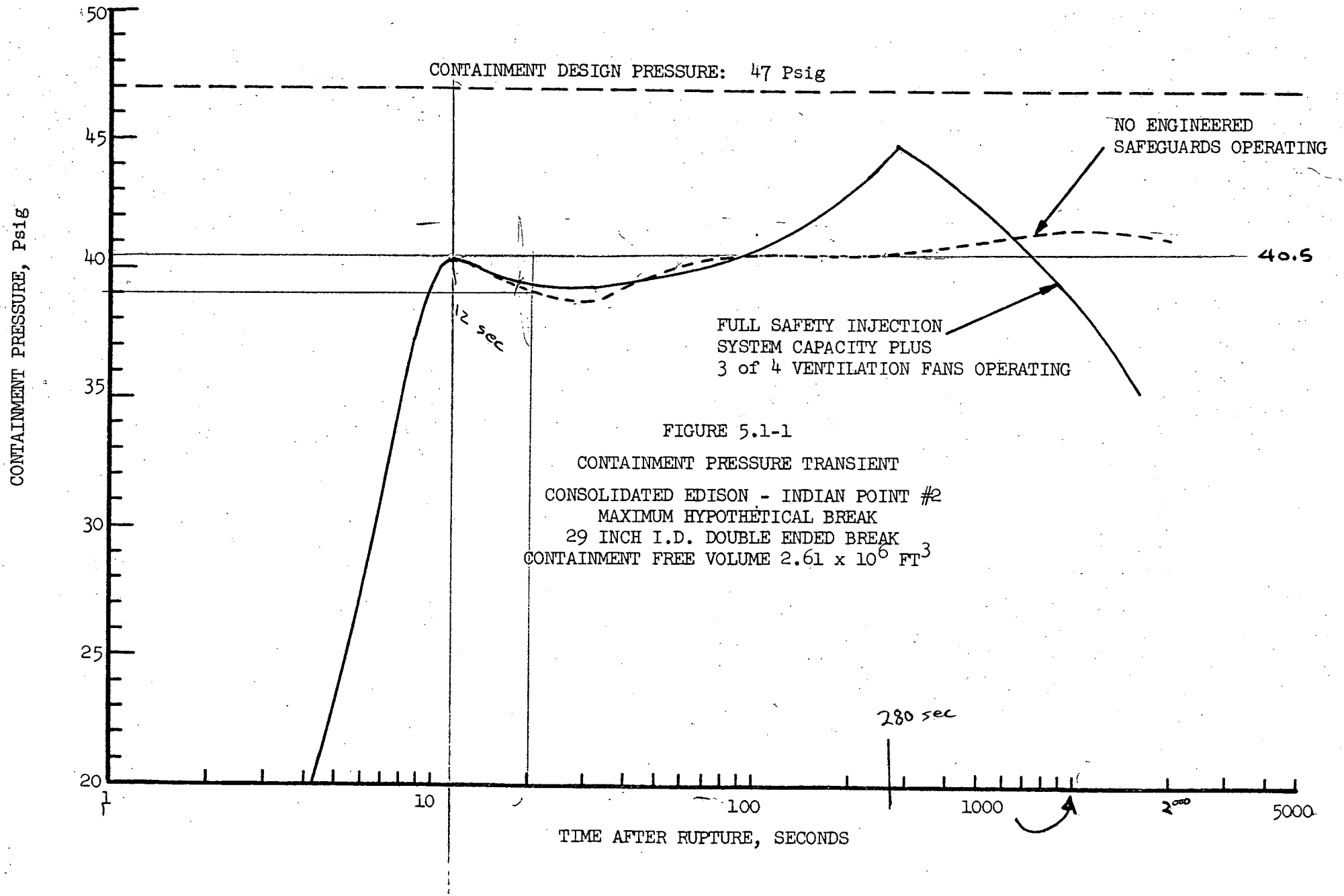


FIGURE 5.1-1

CONTAINMENT PRESSURE TRANSIENT

CONSOLIDATED EDISON - INDIAN POINT #2
MAXIMUM HYPOTHETICAL BREAK
29 INCH I.D. DOUBLE ENDED BREAK
CONTAINMENT FREE VOLUME 2.61×10^6 FT^3

5.2 OFF-SITE CONSEQUENCES OF A MAJOR LOSS OF COOLANT

5.2.1 Full Effectiveness of Engineered Safeguards Systems

In order to demonstrate that the engineered safeguards systems will provide adequate protection for the public, a preliminary evaluation has been performed of the off-site consequences of the largest source of public exposure for a pressurized water reactor, a major loss of coolant.

Although the design, fabrication, inspection and testing of the various components and systems of this plant make the sudden severance of a reactor coolant pipe an incredibility, such an occurrence is postulated to evaluate the upper limit of potential public exposure.

Following the sudden severance of the reactor coolant pipe, the reactor coolant rapidly blows down to the containment causing a pressure increase in the containment and leaving the core uncovered. Proper functioning of the engineered safeguards will initiate the safety injection system to recover the core with borated water and limit the extent of core meltdown and metal-water reaction. With the pressurized penetration system as described, the major source of potential leakage from the containment will be effectively blocked. The further degree of leak tightness afforded by the periodic verification of weld integrity gives a high degree of confidence that theleakage fraction of this containment system during the post-accident conditions will be very much less than 0.1 per cent per day which is a nominal design value for containment systems of this type without pressurized penetrations and testable welds. Considering these factors, the expected post-accident releases and off-site consequences under these conditions are expected to be very small when compared to the exposure limits of 10CFR100.

Further analyses have been performed, however, to evaluate the off-site consequences of this loss of coolant neglecting the improvement in containment

leak tightness which is expected to be achieved with pressurized penetrations and testable welds and with no credit for the operation of the safety injection system to prevent core meltdown. In addition, the effect of the presence of some forms of organic iodines which may not be subject to highly efficient filtration has been evaluated.

5.2.2 Partial Effectiveness of Engineered Safeguards with No Unfilterable Organic Iodines

The first condition of partial effectiveness of the engineered safeguards is evaluated assuming that any organic iodines which might be found in the reactor or in the containment following the accident are removed by the air recirculation filtration system with the same efficiency as the elemental forms of iodine. This assumption is based on recent work at the Oak Ridge National Laboratories (Ref. ORNL-3864) indicating that a highly active grade of coconut charcoal exhibits greater than 99 per cent efficiency for organic iodines at 100 per cent relative humidity and does not exhibit breakthrough observed with other charcoals.

The following model describes the assumptions and methods used to calculate the off-site exposures. Following the loss-of-coolant, a 100% core meltdown was assumed based on the model described in TID 14844 with the following saturation inventory release fractions from the core:

Noble gases	-	100%
Halogens	-	50%
Particulates	-	1%

Following their release from the reactor vessel, one half of the halogens (i.e. 25% of the core halogen inventory as assumed in TID 14844) would plate out very rapidly and become unavailable for leakage. The remaining 25 per cent of the halogens are assumed to remain airborne indefinitely, subject only to

depletion by natural decay, leakage and filtration.

For the post-accident containment conditions, either one of the containment cooling systems (spray or air recirculation coolers) in conjunction with the safety injection system will prevent the containment design pressure from being exceeded but is not assumed effective in rapid pressure reduction. Also, no credit has been taken for the expected improvement in leak tightness of this containment system, so that a leak rate of 0.1 per cent per day is assumed to exist for the first 24 hours after the accident. The cooling systems are expected to bring the containment pressure to near atmospheric in about one day, but a conservative extrapolation of leak rate versus pressure (leak rate being proportional to the square root of the pressure ratio) and a continuing internal pressure of 3 psig have been assumed, so that the leak rate for the next 30 days is taken as 0.045 per cent per day.

As previously described, the air recirculation filtration system is equipped with four fan-filter units, each having a capacity of 65,000 cfm. Three of those units are assumed to be operable during the transient so that a recirculation rate of 4.5 containment volumes per hour is available, based on a containment free volume of 2.61 million cubic feet. An overall filtration efficiency of 90 per cent per pass was assumed to determine a reduction constant, λ_f, of 4.05 hr^{-1} which was applied to all the iodine in the containment available for leakage. The reduction was determined by applying λ_f as an exponential removal coefficient.

The initial halogen source available for leakage was determined using the following parameters: A reactor power level of 2758 MWt,

Isotope	Saturation Inventory
I-131	2.47×10^4 C/MWt
I-132	3.74×10^4 C/MWt
I-133	5.63×10^4 C/MWt

Isotope	Saturation Inventory
I-134	6.47×10^4 C/MWt
I-135	5.03×10^4 C/MWt

A dose-equivalent inventory of I-131 was calculated using the above inventories and the following TID 14844 dose-per-curies factors for the individual isotopes of iodine:

Isotope	Dose per Curie Inhaled
I-131	1.48×10^6 rad
I-132	5.35×10^4 rad
I-133	4.00×10^5 rad
I-134	2.50×10^4 rad
I-135	1.24×10^5 rad

The dose equivalent inventory of iodine 131 as a function of time was obtained by multiplying the initial iodine 131 inventory by the following factors, that take into consideration the radioactive decay of all the iodine isotopes and the contribution of the tellurium isotopes.

Time	Equivalence Factor
0	1.880
Aver. first 2 hrs	1.837
Aver. next 22 hrs	1.472
Aver. next 30 days	.3423

Using the above figures, an initial dose equivalent inventory of iodine 131 of 3.2×10^7 curies is available for leakage.

The meteorological dispersion of the leakage from the containment has been calculated using the Sutton model of dispersion along with the measured dispersion parameters of Indian Point. The Sutton model has been modified to account for additional dispersion of the leakage due to turbulence in the wake of

the containment building. Conservative dispersion characteristics applicable to three time periods were selected and the doses calculated for each period.

The Sutton equation for the dispersion of a point source at ground level gives the ground level plume concentration as a function of distance:

$$\chi = \frac{2Q}{\pi \bar{u} C_y C_z x^{2-n}} e^{-\left(\frac{y^2}{C_y^2 x^{2-n}}\right)}$$

where C_y C_z and n are the dispersion parameters, $\bar{u}$ is the wind speed y is the lateral distance from the plume center line, x is the downwind distance and Q is the point source release term.

Using a method suggested by Gifford and Fuquay, this formula can be modified to account for building dilution as follows:

$$\chi/Q = \frac{2 \quad e^{-\left(\frac{y^2}{C_y^2 x^{2-n}}\right)}}{\bar{u}\left(\pi C_y C_z x^{2-n} + A\right)}$$

where A is the cross sectional area of the containment building. The first period of the dose calculation utilized this modified dispersion formula, a building area of 2000 square meters, and the inversion parameters assumed in TID 14844 which are conservative for the Indian Point site.

Category	C_y	C_z	n	$\bar{u}$
Inversion-I	0.4 $m^{n/2}$	0.07 $m^{n/2}$	0.5	1 m/sec

This first period comprises the first two hours after the accident. The direction of the 1 meter per second wind is assumed to be constant throughout the period.

The second period is the next 22 hours following the accident, during which the same inversion condition exists, but the average wind speed from the same direction is assumed to be 2 meters per second.

The third period is from 24 hours after the accident to 31 days after the accident. During this period, the meteorological conditions are assumed to be randomly distributed among the categories listed below:

Category	Fraction	$1/\bar{u}$	C_z	C_y	n
Lapse - L_1	0.139	0.575	0.48	0.6	0.2
Lapse - L_2	0.061	0.191	0.43	0.53	0.3
Neutral - N	0.378	0.358	0.39	0.47	0.4
Inversion - I	0.424	0.493	0.07	0.40	0.5

The parameters $\bar{u}$, C_y, C_z, n for L_1, L_2 and N are those measured at the site, and those for I are again the TID 14844 assumptions. These site parameters represent a very conservative 30 day period as they are inversions for 42% of the time. Because the winds are not expected to be from the same direction throughout the 30 day period, the dispersion formula was modified to account for long term variability of the mean wind direction. The most adverse distribution was assumed to result in a maximum of 35 per cent of the winds blowing in one 20° sector. The dispersion formula used is:

$$(\chi/Q)_{30} = \sum_i \frac{2fF_i}{\beta\, \bar{u}_i (\sqrt{\pi} C_{z_i} x^{2-n_i/2} + A)}$$

This expression is obtained by integrating the Sutton equation from $-\infty$ to $+\infty$ in the y direction and then averaging the concentration over the desired sector, β, for the appropriate fraction of the time, f. The other parameters have been defined with F_i being the fraction of the time any particular weather category

exists. As stated, β = 0.354 radian (20°) and f = 0.35.

Dispersion Factors

Distances meters	(χ/Q) 2 hours	(χ/Q) 22 hours	(χ/Q) 30 days
400	7.40×10^{-4}	3.70×10^{-4}	1.60×10^{-4}
700	5.52×10^{-4}	2.76×10^{-4}	8.70×10^{-5}
1,000	3.96×10^{-4}	1.98×10^{-4}	5.47×10^{-5}
2,000	2.03×10^{-4}	1.01×10^{-4}	2.44×10^{-5}
4,000	8.21×10^{-5}	4.10×10^{-5}	5.83×10^{-6}
7,000	3.53×10^{-5}	1.76×10^{-5}	2.21×10^{-6}
10,000	2.23×10^{-5}	1.11×10^{-5}	1.19×10^{-6}
20,000	8.40×10^{-6}	4.20×10^{-6}	3.53×10^{-7}

Adult breathing rates have been assumed in the dose calculations and are 1.25 m^3/hours for the first 2 hours, 0.795 m^3/hours for the next 22 hours and 20 m^3/day for the next 30 days. Using the models and assumptions outlined above the off-site doses were calculated using the following equations:

$$D_2 = 2 \times B_2 \times L_2 \times \bar{A}_2 \times (\chi/Q)_2 \times \frac{1}{DRF_2} \times DCF_{131}$$

D_2 = Integrated two hour dose to the adult thyroid, rads

B_2 = Breathing rate for 2 hours — m^3/hour

L_2 = Leak rate for 2 hours — sec^{-1}

$\bar{A}_2$ = Average dose equivalent curies I-131 available for leakage over the first 2 hours, neglecting filtration — curies

$(\chi/Q)_2$ = Two hour dispersion factor — sec/m^3

DRF_2 = Reduction factor averaged over two hours due to recirculation filtration: $DRF_2 = \frac{2\lambda f}{1-e^{-2\lambda f}}$

DCF_{131} = Dose conversion factor for I-131 — rad/curie

For the particular case analyzed

$B_2 = 1.25$ $\quad$ $L_2 = 1.16 \times 10^{-8}\ sec^{-1}$

$\bar{A}_2 = 3.13 \times 10^7$ curies $\quad$ $DRF_2 = 8.1$

$DCF_{131} = 1.48 \times 10^6$ rad/curies

$D_2 = 1.66 \times 10^5\ (\chi/Q)_2$ rads

For the 22 hour dose calculations, the equation is essentially the same:

$$D_{22} = 22 \times B_{22} \times L_{22} \times \bar{A}_{22} \times (\chi/Q)_{22} \times \left(\frac{e^{-2\lambda f}}{DRF_{22}}\right) \times DCF_{131}$$

D_{22} =	Integrated dose to an adult thyroid over the next 22 hours	rads
B_{22} =	Breathing rate for the next 22 hours	m^3/hr
L_{22} =	Leak rate for next 22 hours	sec^{-1}
$\bar{A}_{22}$ =	Average dose equivalent curies of I-131 available for leakage during the next 22 hours, neglecting filtration	curie
$(\chi/Q)_{22}$ =	Next 22 hour dispersion factor	sec/m^3
DRF_{22} =	Reduction factor averaged over the 22 hours due to recirculation filtration	

$$DRF_{22} = \frac{22\lambda_f}{1-e^{-22\lambda f}}$$

For the 22 hours case analyzed here

$B_{22} = 0.795\ m^3/hr$ $\quad$ $L_{22} = 1.16 \times 10^{-8}\ sec^{-1}$

$\bar{A}_{22} = 2.51 \times 10^7$ curies $\quad$ $DRF_{22} = 89$

$DCF_{131} = 1.48 \times 10^6$ rads/curies

$D_{22} = 25.4\ (\chi/Q)_{22}$

The 30 day dose is calculated in much the same manner, although for this particular case, its contribution to the total dose is negligible. The

integrated dose versus distance for this case is shown on Figure 5.2-1.

The direct radiation dose from this accident have also been evaluated. The infinite time integrated whole body dose at the surface of the containment is less than 100 mr, so that there is no direct radiation hazard to the operating personnel of Unit No. 1 or Unit No. 2 and to the public.

5.2.2 Sensitivity to the Behavior of Organic Forms of Iodine

A further analysis has been undertaken to determine the sensitivity of the safeguards evaluation to the assumption in Section 5.2.1 that all forms of iodine, including the organic form, are susceptible to the combination of particulate and charcoal filtration. Should further research and development fail to support the practicality of using special grades of charcoal to trap and retain organic iodides, it must be assumed that some fraction of the volatile iodine remains an available source for leakage to the environment. While the magnitude and biological significance of this residual source are subject to some uncertainty at this time, the following inferences can be drawn from published reports and personal communications with investigators in this field.

a) The most credible explanation for the synthesis of organic iodine requires that inorganic iodine (e.g. Molecular I_2) must first diffuse to a metallic surface and be deposited on that surface in order to undergo the synthesis reaction.

b) The yield of organic iodine which takes place in the interior of the reactor vessel and coolant piping is expected to be small (perhaps 1-2 cent).

c) The human uptake factors for organic iodine are not known, but would be expected to be different from those applied to the more soluble forms of iodine.

d) Once absorbed on charcoal, molecular iodine resists conversion to the organic form.

A preliminary conclusion can be drawn from the fact that the removal coefficient for filtration of molecular iodine ($4.05\ hr^{-1}$) is an order of magnitude higher than the removal coefficient for deposition on surfaces (assuming a 0.05 cm/sec deposition velocity in an enclosure the size of the reactor containment): that is, that the fate of molecular iodine released to the containment is much more likely to be permanent retention on charcoal than conversion to the organic form. This conclusion is further supported by the fact that a large fraction of the iodine is removed from the synthesis chain by interaction with liquid water condensing in the air, in the recirculation air coolers and on other cool surfaces.

The present evaluation is made assuming that the net effective yield of organic iodine is five percent of that which is initially available for leakage from the containment, and that no reduction of this source occurs with passage through the particulate or charcoal filters.

The dose equations are modified as follows:

$$D_2 = 2 \times B_2 \times L_2 \times \bar{A}_2 \times (\chi/Q)_2 \times \left[\alpha + \frac{1-\alpha}{DRF_2}\right] \times DCF_{131}$$

where the terms are as previously defined and α is the fraction of unfilterables.

For the case of $\alpha = 0.05$

$$D_2 = 2.24 \times 10^5\ (\chi/Q)_2 \quad \text{rads}$$

For the 22 hour dose calculation

$$D_{22} = 22 \times B_{22} \times L_{22} \times \bar{A}_{22} \times (\chi/Q)_{22} \left[\alpha + \frac{(1-\alpha)e^{-2\lambda_f}}{DRF_{22}}\right] \times DCF_{131}$$

Again the terms are as previously defined so that

$$D_{22} = 3.77 \times 10^5\ (\chi/Q)_{22}$$

The thirty day dose is given by

$$D_{30} = 30 \times B_{30} \times L_{30} \times \bar{A}_{30} \times (\chi/Q)_{30} \left[\alpha + \frac{(1-\alpha)e^{-24\lambda_f}}{DRF_{30}}\right] \times DCF_{131}$$

These are similar to the previously defined terms except that they are for the 30 day period.

For the case of interest:

$B_{30} = 20\ m^3/day$ $\qquad$ $L_{30} = 5.22 \times 10^{-9}\ sec^{-1}$

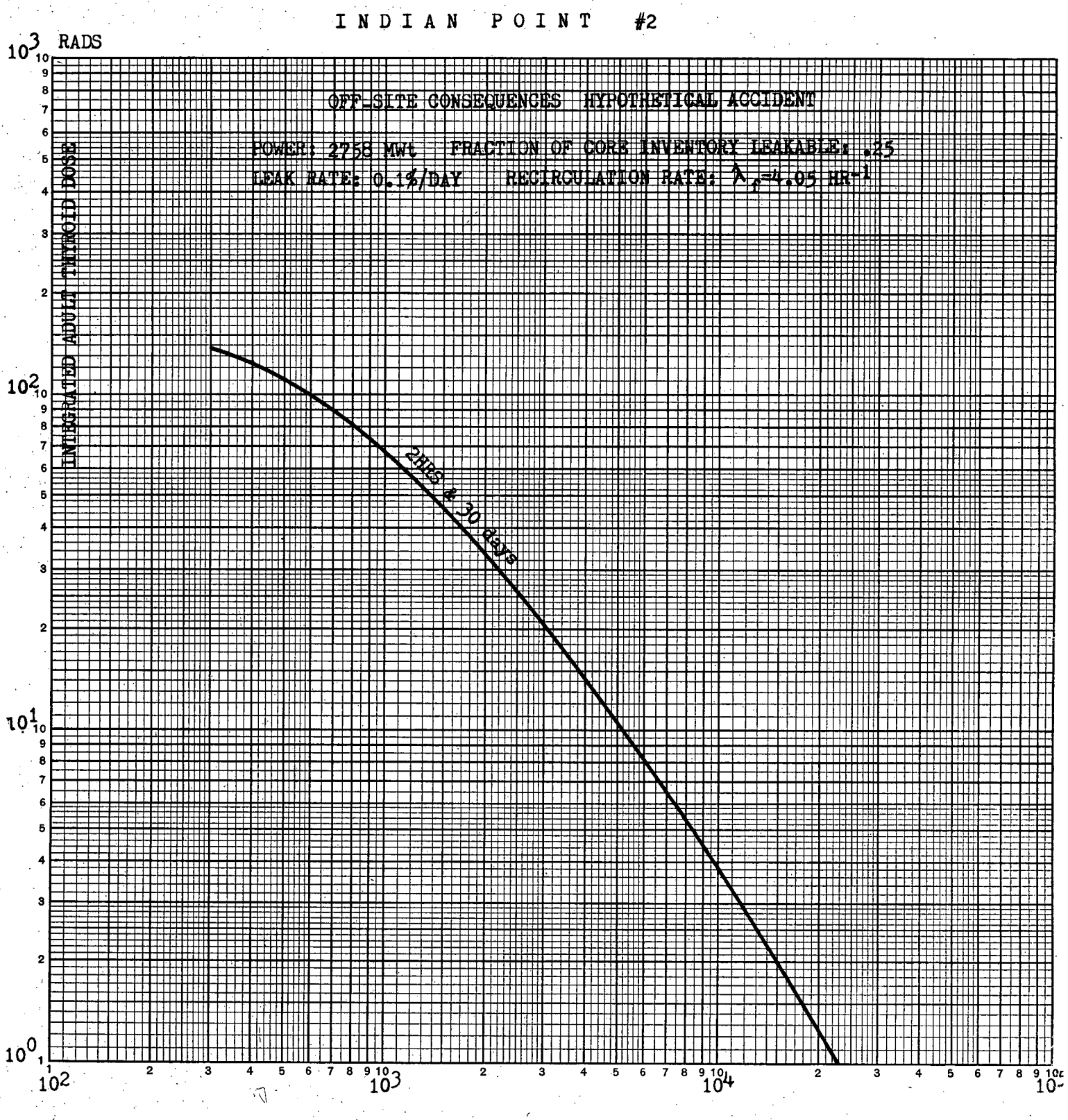
$\bar{A}_{30} = 5.83 \times 10^6$ curies $\qquad$ $\alpha = 0.05$

$DRF_{30} = 2.92 \times 10^3$ $\qquad$ $DCF_{131} = 1.48 \times 10^6$ rads/curies

$D_{30} = 1.35 \times 10^6\ (\chi/Q)_{30}$ rads

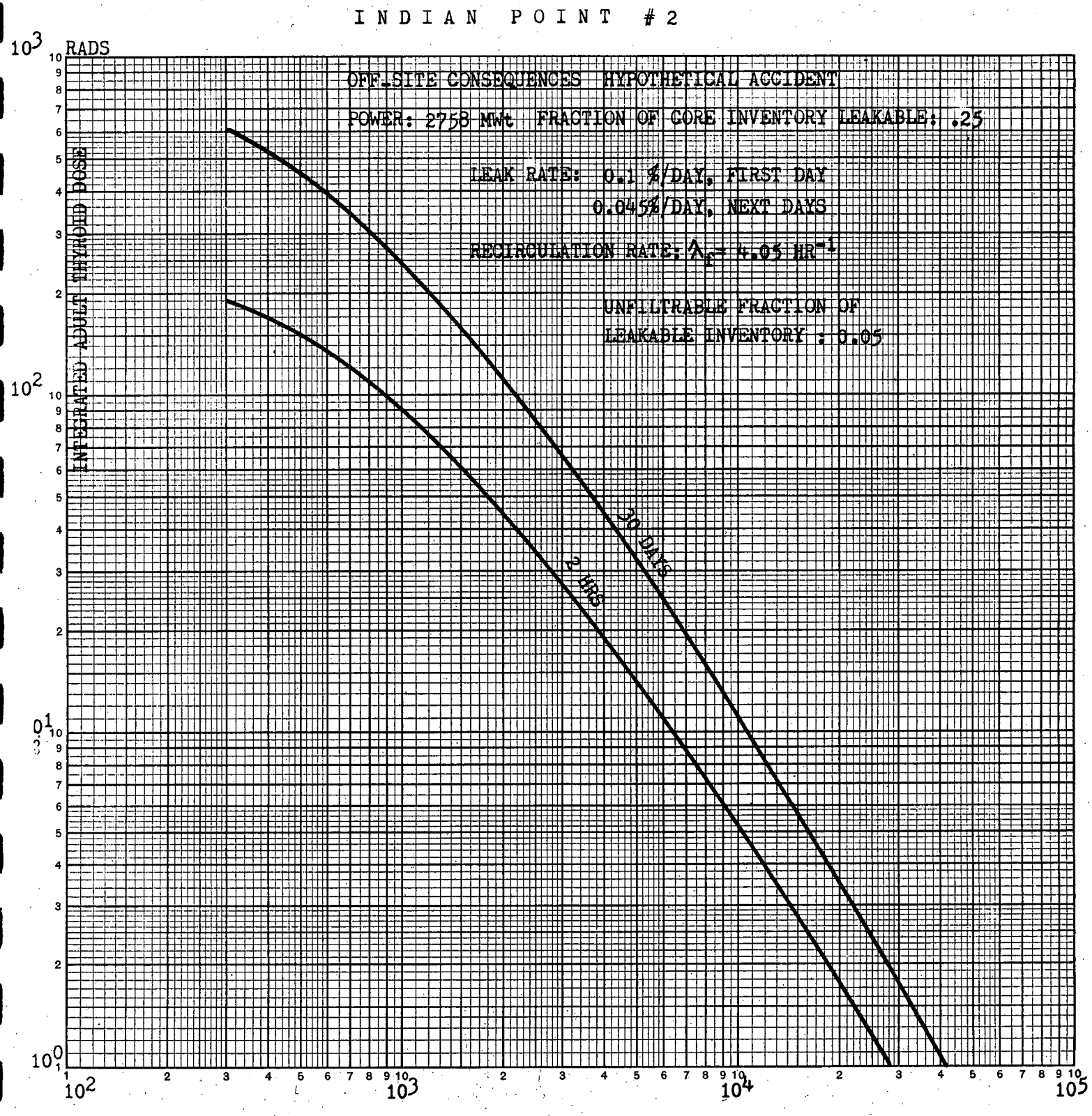
The integrated doses for 2 hours and for 30 days are shown on Figure 5.2-2. The 30 day dose assumes that the same person received the 2 hour, 22 hour and 30 day doses as calculated.

The direct radiation whole body doses are the same as those calculated in Section 5.2.1.



Case I

Figure 5.2-1



Case II

Figure 5.2-2

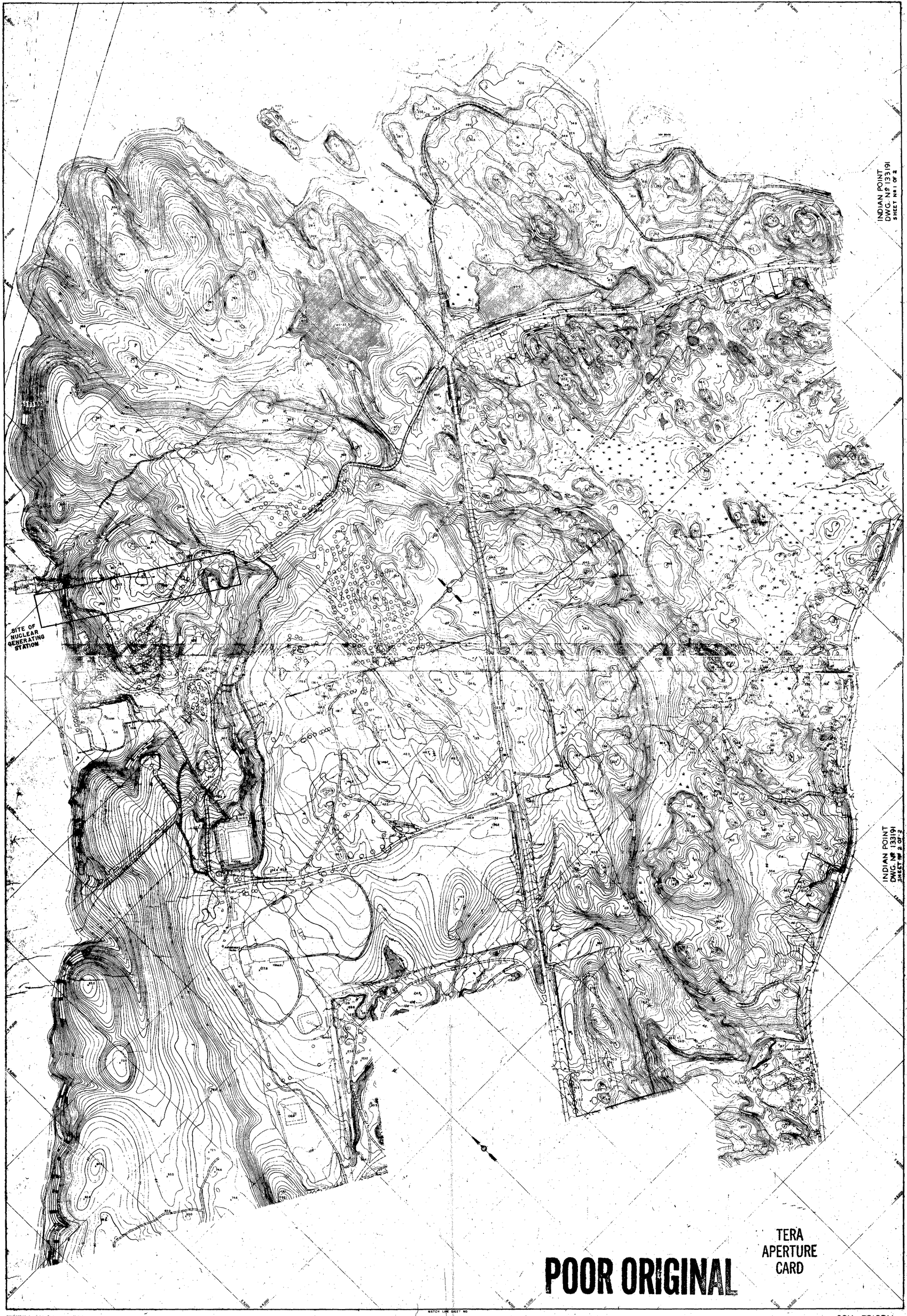
A (16)

8110210077-01-14

FIGURE 1

Con Edison
Dwg. 133191

FIGURE 1.



TERA
APERTURE
CARD

POOR ORIGINAL

TOPOGRAPHY COMPILED BY PHOTOGRAMMETRIC METHODS
FROM AERIAL PHOTOGRAMMETRIC ENGINEERS, INC.
AERIAL PHOTO FLIGHT DATE: 1 NEW YORK - DENVER, COLORADO
AERIAL PHOTO FLIGHT DATE: 2 NEW YORK - HOUSTON, TEXAS

MATCH LINE SHEET NO

SCALE 1:1200

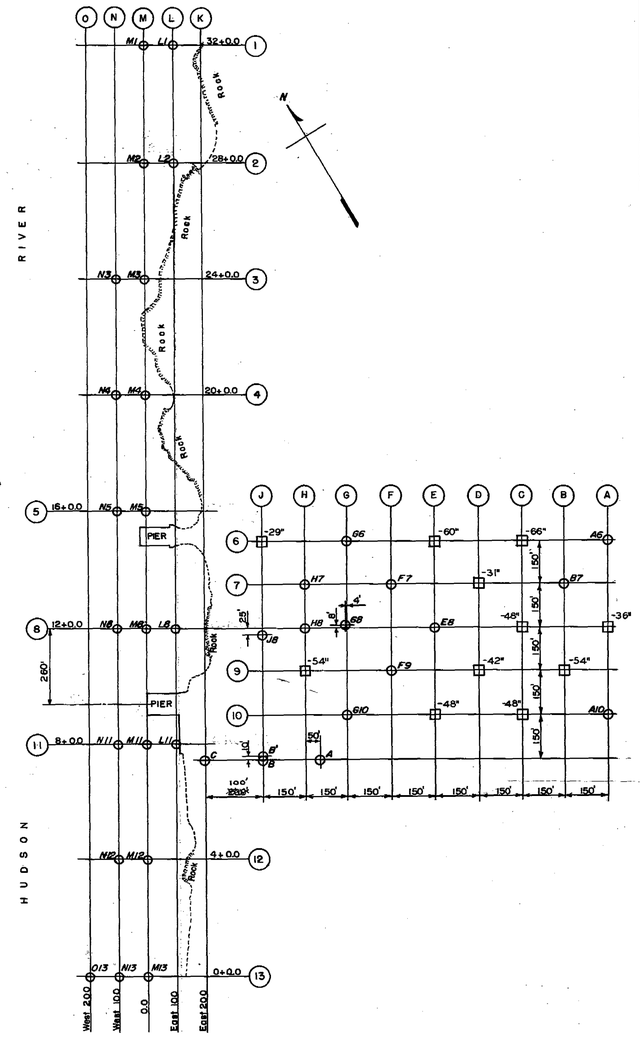
CONTOUR INTERVAL

PREPARED MAY 1969
BY: WILBUR IRISH & ELLS
906 SOUTH ST., HANSCOM, MASS.

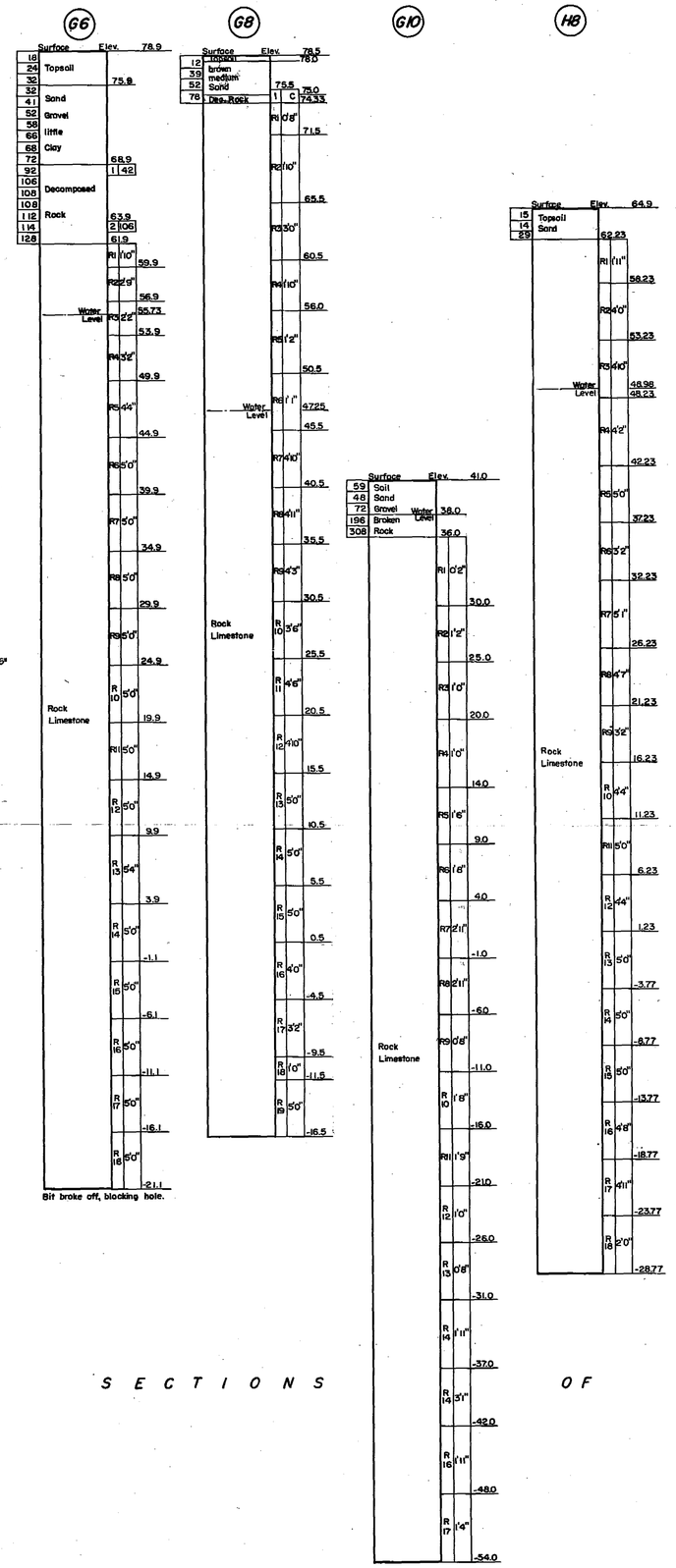
CON-EDISON
TOPOGRAPHIC MAP
OF
BUCHANAN AREA
INDIAN POINT
DWG. Nº 133191
SHEET NO. 2 OF 2

FIGURE 3

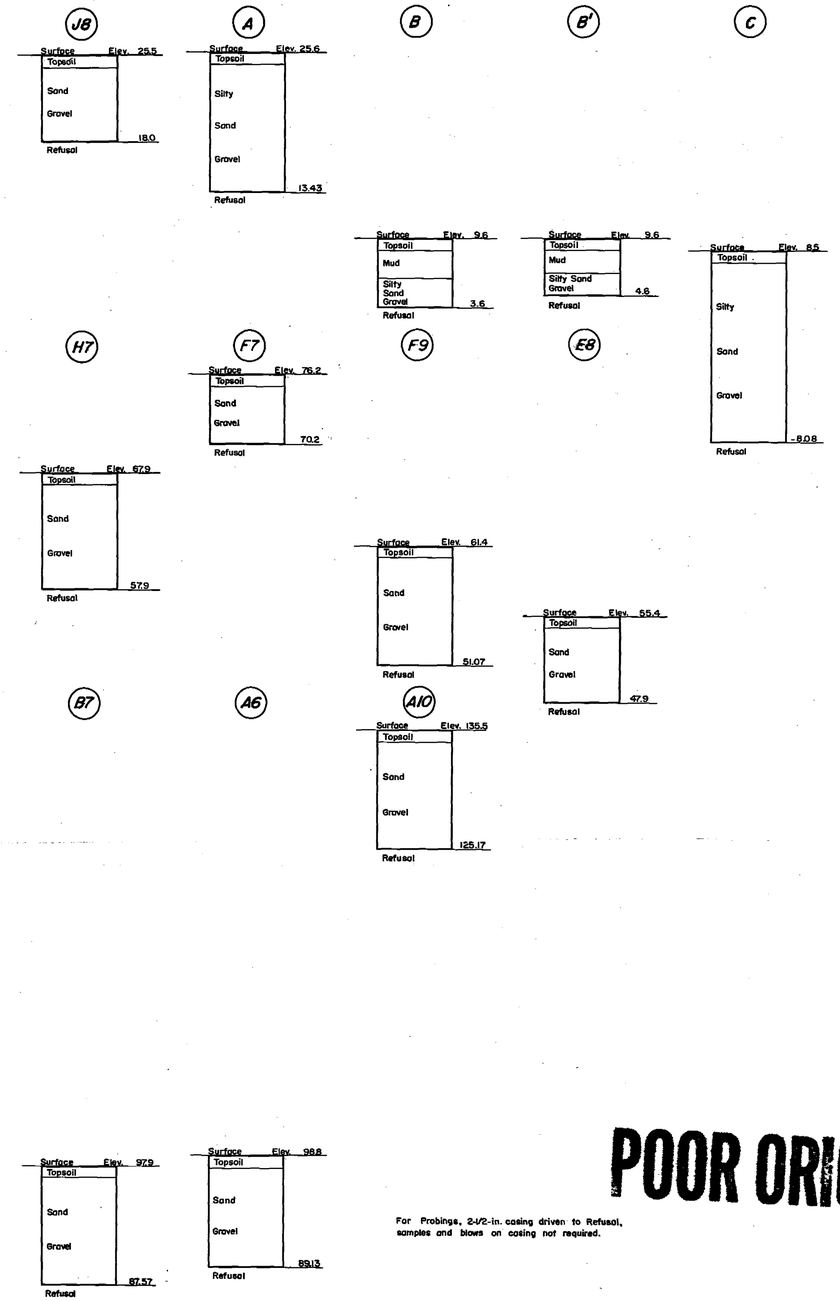
Con Edison
Dwg. 132848



PLAN



SECTIONS OF BORINGS



For Probings, 2-1/2-in. casing driven to Refusal, samples and blows on casing not required.

POOR ORIGINAL

NOTES

A/B C

A is elevation of bottom of sampling device at start of penetration.
B is number of sample or drilling run (preceded by "R").
C is number of blows of 300-lb. hammer falling 18 inches taken to drive 2-in. split sampler 1 foot. "C" here, sample taken from casing. If sample is run of rock core (size AX) feet and inches recovered between top and bottom elevations for run appear here.
At left of sections (except land probings) is number of blows (same hammer and fall) taken to drive 2-1/2-in. casing each foot, and odd distance shown to rock.
Elevations refer to Mean Sea Level, Sandy Hook.

Scale: Plan 1" = 200'; Section 1" = 5'.

INFORMATION OBTAINED BY CON. EDISON CO. SURVEYORS:

-48" Indicates location of rock below surface as obtained by probing.

For soundings see dwgs. 132860, 132861, & 132862.

TERA APERTURE CARD

SEE DWG. 132850

TEST BORINGS
MADE BY
WARREN GEORGE
136 LIBERTY STREET NEW YORK 6, N. Y.

Henry D. Hammond

REVISIONS

7-5-56		1

CORRECTED DIMENSION ON PLAN BETWEEN LINE J & K AGREE WITH CON EDISON KEY PLAN. 7-5-56 L.P.

132848-1

LETTERS "A" TO "K" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT

BORINGS

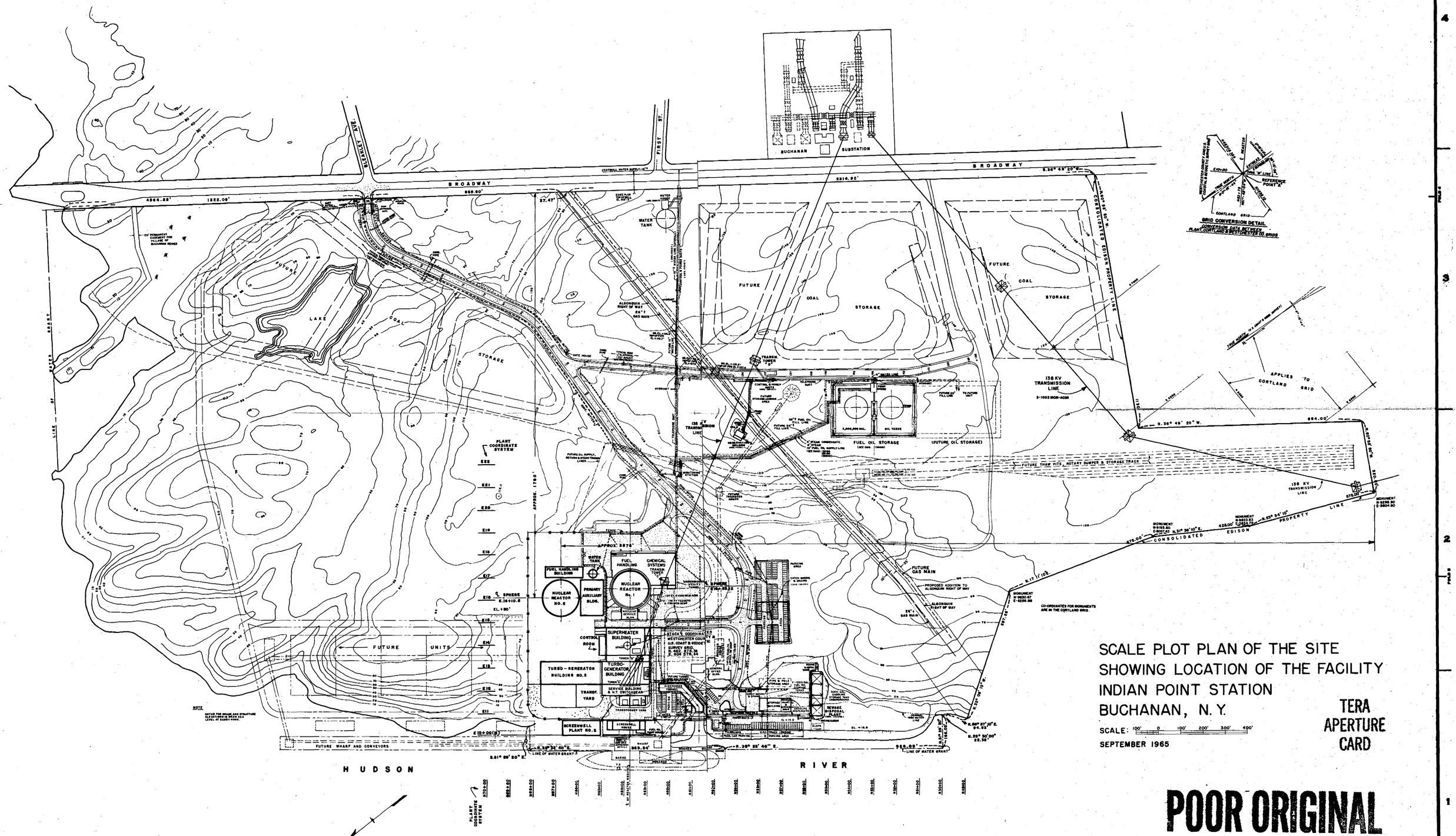
SHEET 1 OF 2

SCALE 1" = 200' / 1" = 5'	DRAWN BY		RECORDED 6-11-58
J.O. 54-300 M	CHECKED BY		ENG.
DIAG. / ELECT.	MECH.	SUPPL.	DIST.

APPROVALS

CHIEF DRAFTSMAN	5-13-58
INSIDE PLANT ENG.	
MECH. PLANT ENG.	
STRUCTURAL ENGINEER	6-1-58
DISTRIBUTION ENGINEER	
OUTSIDE PLANT ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
CHIEF PROJ. ENG.	6/10/58
OPERATION	

SCALE PLOT PLAN OF SITE



SCALE PLOT PLAN OF THE SITE
SHOWING LOCATION OF THE FACILITY
INDIAN POINT STATION
BUCHANAN, N. Y.

SCALE: 100' 0 100' 200' 300' 400'

SEPTEMBER 1965

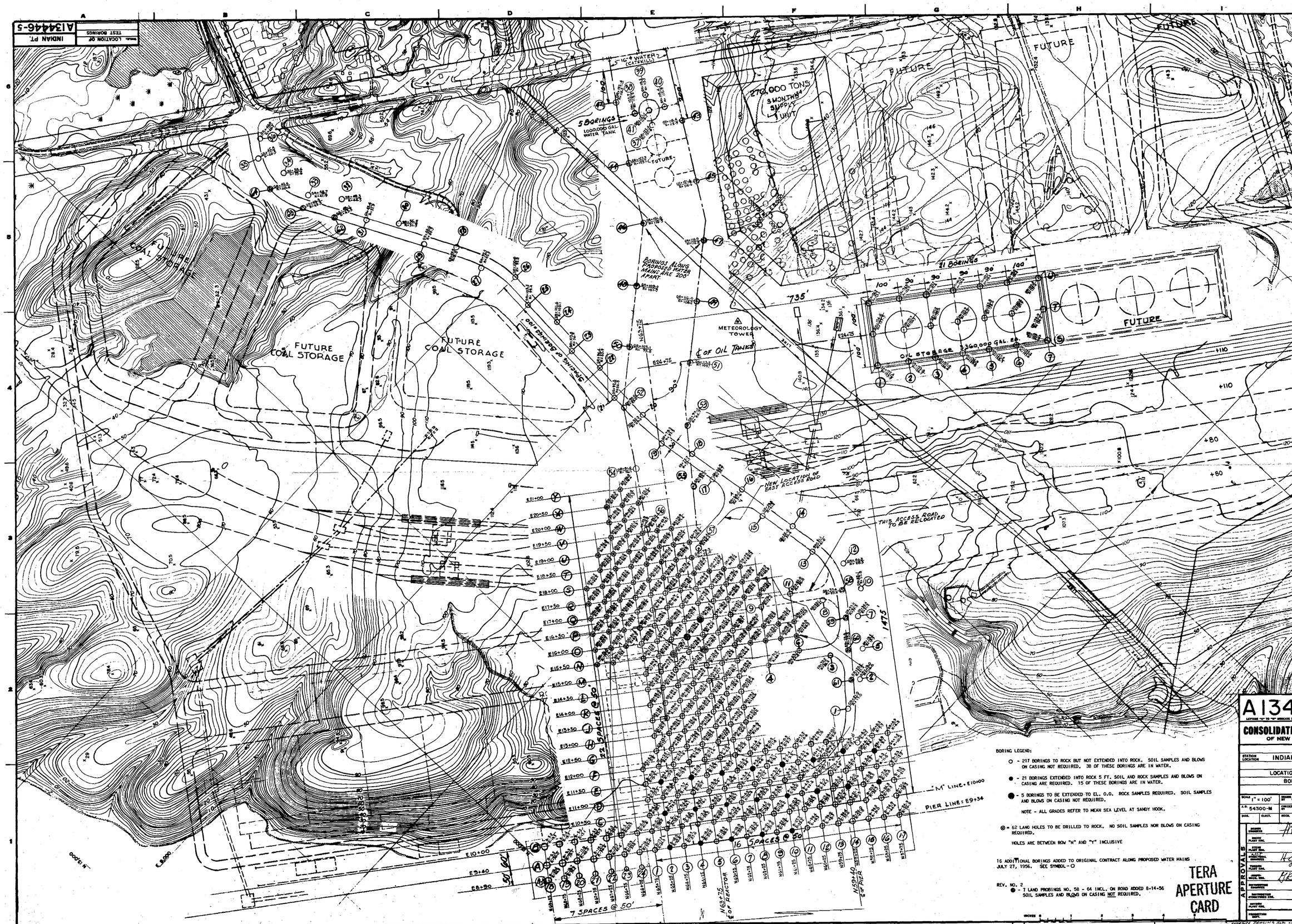
TERA APERTURE CARD

POOR ORIGINAL

LOCATION OF TEST BORINGS

UNIT #1

Dwg. A134446-5



BORING LEGEND:

○ - 217 BORINGS TO ROCK BUT NOT EXTENDED INTO ROCK. SOIL SAMPLES AND BLOWS ON CASING NOT REQUIRED. 38 OF THESE BORINGS ARE IN WATER.

● - 21 BORINGS EXTENDED INTO ROCK 5 FT. SOIL AND ROCK SAMPLES AND BLOWS ON CASING ARE REQUIRED. 15 OF THESE BORINGS ARE IN WATER.

● - 5 BORINGS TO BE EXTENDED TO EL. 0.0. ROCK SAMPLES REQUIRED. SOIL SAMPLES AND BLOWS ON CASING NOT REQUIRED.

NOTE - ALL GRADES REFER TO MEAN SEA LEVEL AT SANDY HOOK.

⊗ = 62 LAND HOLES TO BE DRILLED TO ROCK. NO SOIL SAMPLES NOR BLOWS ON CASING REQUIRED.

HOLES ARE BETWEEN ROW "N" AND "Y" INCLUSIVE

16 ADDITIONAL BORINGS ADDED TO ORIGINAL CONTRACT ALONG PROPOSED WATER MAINS JULY 27, 1956. SEE SYMBOL - ○

REV. NO. 2
● - 7 LAND PROBINGS NO. 58 - 64 INCL. ON ROAD ADDED 8-14-56 SOIL SAMPLES AND BLOWS ON CASING NOT REQUIRED.

TERA APERTURE CARD

POOR ORIGINAL

A134446-5

LETTERS "A" TO "M" ARRIGATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION	INDIAN POINT
	LOCATION OF TEST BORINGS
SCALE	1" = 100'
DRAWN BY	C. McINTYRE
DESIGNED BY	12-27-56 C.
A.O.	54300-M
DRAWN	
ELECT.	
MECH.	
STRUCT.	
SURV.	

APPROVALS

DESIGN ENGINEER	A Pirhl 6-7-56
PLANT ENG.	
ESTIMATE PLANT ENG.	
ELECTRICAL ENGINEER	H J Meinke 6-11-56
TURBINE PLANT ENG.	
MECH. ENG.	H.R. McLean 6/11/56
INSTRUMENTATION ENGINEER	
INSTRUMENTATION ESTIMATING ENG.	
GENERAL PLANT ENG.	
CONSTRUCTION DEPT.	

EVERYTHING CONTAINED HEREON AND THEREON, TITLE TO INDICATION SHALL REMAIN WITH CEASE

INDIAN PT.
LOCATION OF TEST BORINGS
A134446-5

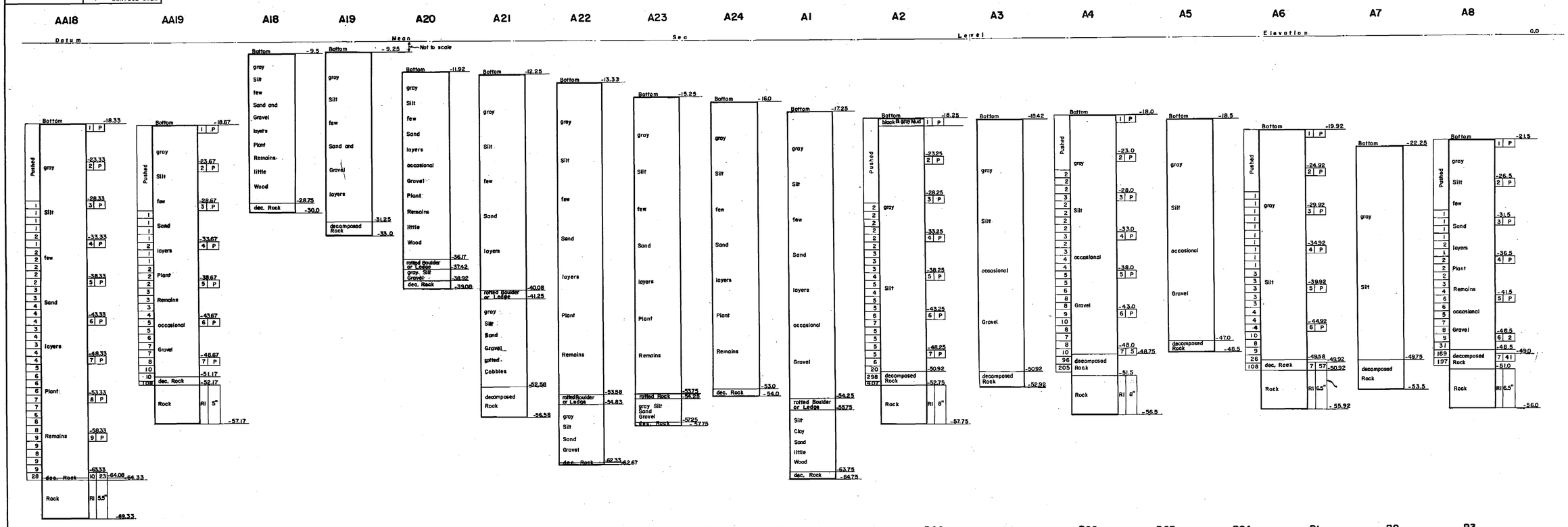
REVISIONS
5 RELOCATED EAST ACCESS ROAD C.McI 3-13-57

TEST BORINGS

UNIT #1

Dwg. 136308

TEST BORINGS	INDIAN POINT
SHEET 1 OF 6	136308-0



NOTES

All borings carried into hard rock.

All soundings (made by driving 2-1/2-in. casing) carried to refusal.

A is elevation of bottom of sampling device at start of penetration.

B is number of sample or drilling run (preceded by "R").

C is number of blows of 300-lb. hammer falling 18 inches taken to drive 2-in. split sampler 1 foot. For rock core, inches (or feet and inches) recovered appear here. "P", pushed down.

At left of boring sections is number of blows (same hammer and fall - also used to drive casing for soundings) taken to drive 2-1/2-in. casing each foot and add distance shown at bottom.

No samples or casing - blows record required on soundings.

Ground water not reached by land soundings.

Grid holes arranged from North to South. Others in numerical order, except 5 land borings in bed rock are grouped.

All bits used for coring rock size AX, producing 1-1/8-in. core.

REFERENCE DRAWINGS

134446	LOCATION OF TEST BORINGS
136309	SHEET NO. 2 OF 6
136310	SHEET NO. 3 OF 6
136311	SHEET NO. 4 OF 6
136312	SHEET NO. 5 OF 6
136313	SHEET NO. 6 OF 6

S E C T I O N S O F B O R I N G S

Scale: 1" = 5'

POOR ORIGINAL

LICENSED PROFESSIONAL ENGINEER HENRY D. HAWKSWORTH UNIVERSITY OF THE STATE OF NEW YORK SEL 18220

Henry D. Hawksworth

TEST BORINGS MADE BY WARREN GEORGE 136 LIBERTY STREET NEW YORK 6, N.Y.

REVISIONS

		0

RELEASED FOR CONSTRUCTION CMC 12-24-56

TERA APERTURE CARD

136308-0

LETTERS "A" TO "R" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT TEST BORINGS SHEET 1 OF 6

SCALE	DRAWN BY	RECORDED
1" = 5'-0"		9-7-56 R
J.O. 54300-M	CHECKED BY	ENG
DIAG / ELECT / MECH / STRUCT / DIST		

APPROVALS

CHIEF DRAFTSMAN	J.P. Kelly 9/5/56
INSIDE PLANT ENG.	
MECH. PLANT ENG.	
STRUCTURAL ENGINEER	J.R. Galloway 9/6/56
DISTRIBUTION ENGINEER	
OUTSIDE PLANT ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
CHIEF PROJ. ENG.	J. Gallon 9/6/56
OPERATION	

STANDARD CO. SYRACUSE, N.Y. U.S.A.

TEST BORINGS

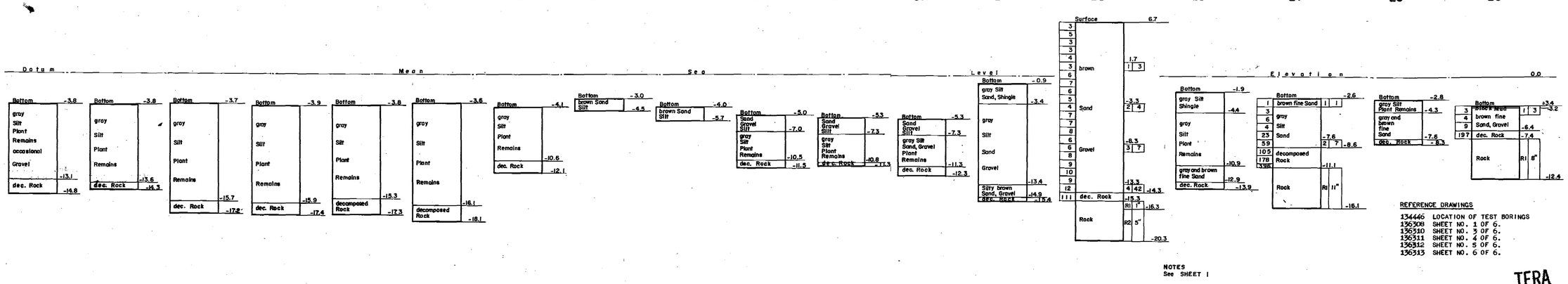
UNIT #1

Dwg. 136309

136309-0

TEST BORINGS
SHEET 2 OF 6
INDIAN POINT

REVISIONS
RELEASED FOR CONSTRUCTION CMK 12-24-58



S E C T I O N S

Scale: 1" = 5'

NOTES
See SHEET 1

REFERENCE DRAWINGS

134446 LOCATION OF TEST BORINGS
136308 SHEET NO. 1 OF 6.
136310 SHEET NO. 3 OF 6.
136311 SHEET NO. 4 OF 6.
136312 SHEET NO. 5 OF 6.
136313 SHEET NO. 6 OF 6.

TERA APERTURE CARD

TEST BORINGS
MADE BY
WARREN GEORGE
136 LIBERTY STREET NEW YORK 6, N.Y.

Cost 1

136309-0
LETTERS "A" TO "N" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT
TEST BORINGS
SHEET 2 OF 6

SCALE 1" = 5'-0" | DRAWN BY | RECORDED 9-7-58
J.O. 54300-M | CHECKED BY | ENG.
DIAG. | ELECT. | MECH. | STRUCT. | DIST.

APPROVALS
CHIEF DRAFTSMAN J. P. Kiebel 9/8/58
INSIDE PLANT ENG.
MECH. PLANT ENG.
STRUCTURAL ENGINEER J. R. Holloway 9/8/58
DISTRIBUTION ENGINEER
OUTSIDE PLANT ENGINEER
DISTRIBUTION STRUCTURES ENG.
CHIEF PROJ. ENG. J. W. Allen 9/8/58
OPERATION

Sidney Co. Washington, D.C. U.S.A.

TEST BORINGS

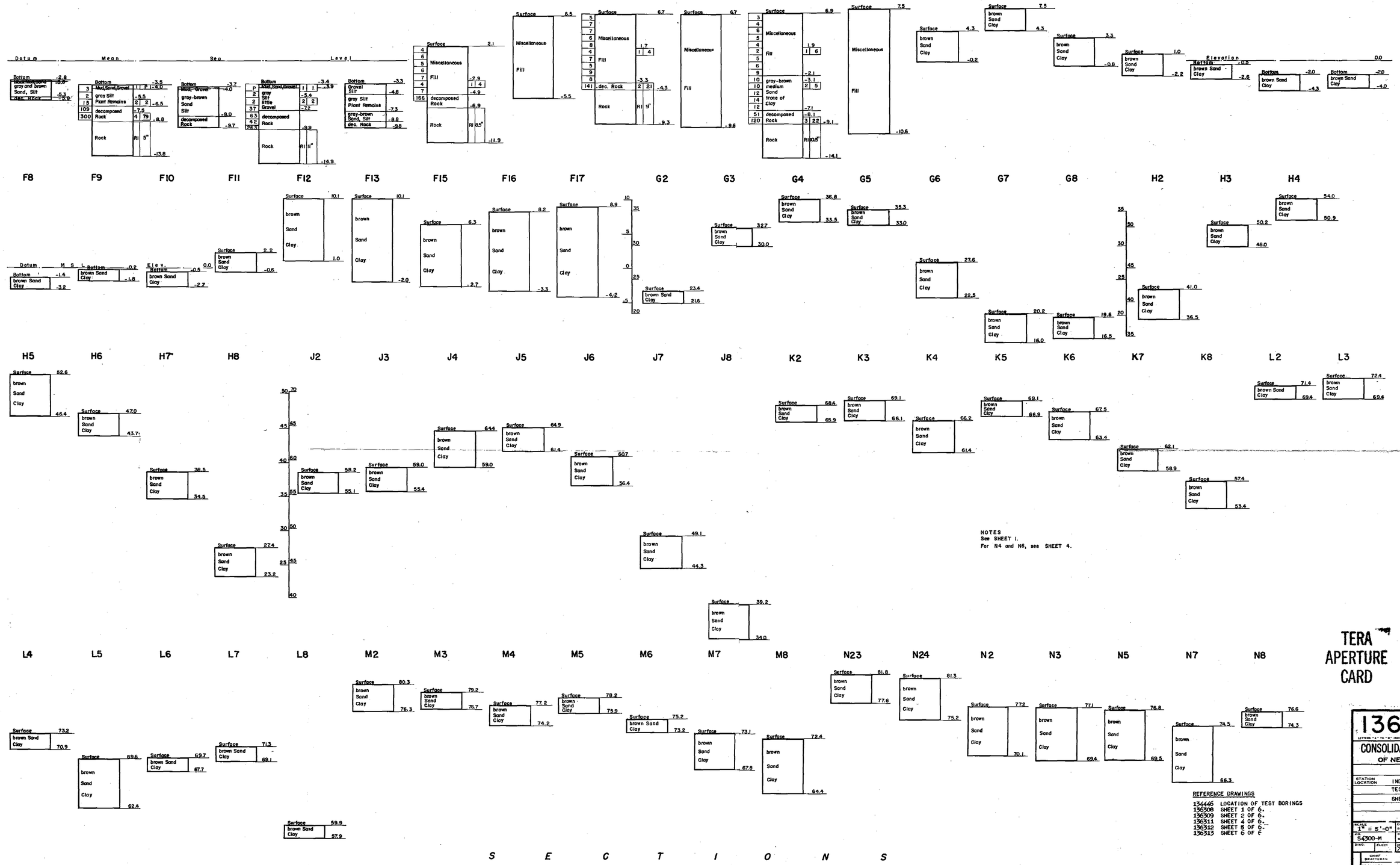
UNIT #1

Dwg. 136310

136310-0

INDIAN POINT

TEST BORINGS - SHEET 3 OF 6



NOTES
See SHEET 1.
For N4 and N6, see SHEET 4.

REFERENCE DRAWINGS

134446	LOCATION OF TEST BORINGS
136308	SHEET 1 OF 6.
136309	SHEET 2 OF 6.
136311	SHEET 4 OF 6.
136312	SHEET 5 OF 6.
136313	SHEET 6 OF 6

TERA APERTURE CARD

REVISIONS

		0

RELEASED FOR CONSTRUCTION CWS 12-28-56

136310-0

LETTERS "A" TO "H" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT TEST BORINGS SHEET 3 OF 6

SCALE 1" = 5'-0"	DRAWN BY	RECORDED T.J.O'K
JOB 54300-M	CHECKED BY	ENG
DIAG. / ELECT.	MECH. / STRUCT.	DIST

APPROVALS

CHIEF DRAFTSMAN	H.P. Kelly 9/6/56
INSTR. PLANT ENG.	
MECH. PLANT ENG.	
STRUCTURAL ENGINEER	A.R. Gilloury 9/6/56
DISTRIBUTION ENGINEER	
OUTSIDE PLANT ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
CHIEF PROJ. ENG.	J. Kuhnelson 9/6/56
OPERATION	

TEST BORINGS MADE BY WARREN GEORGE

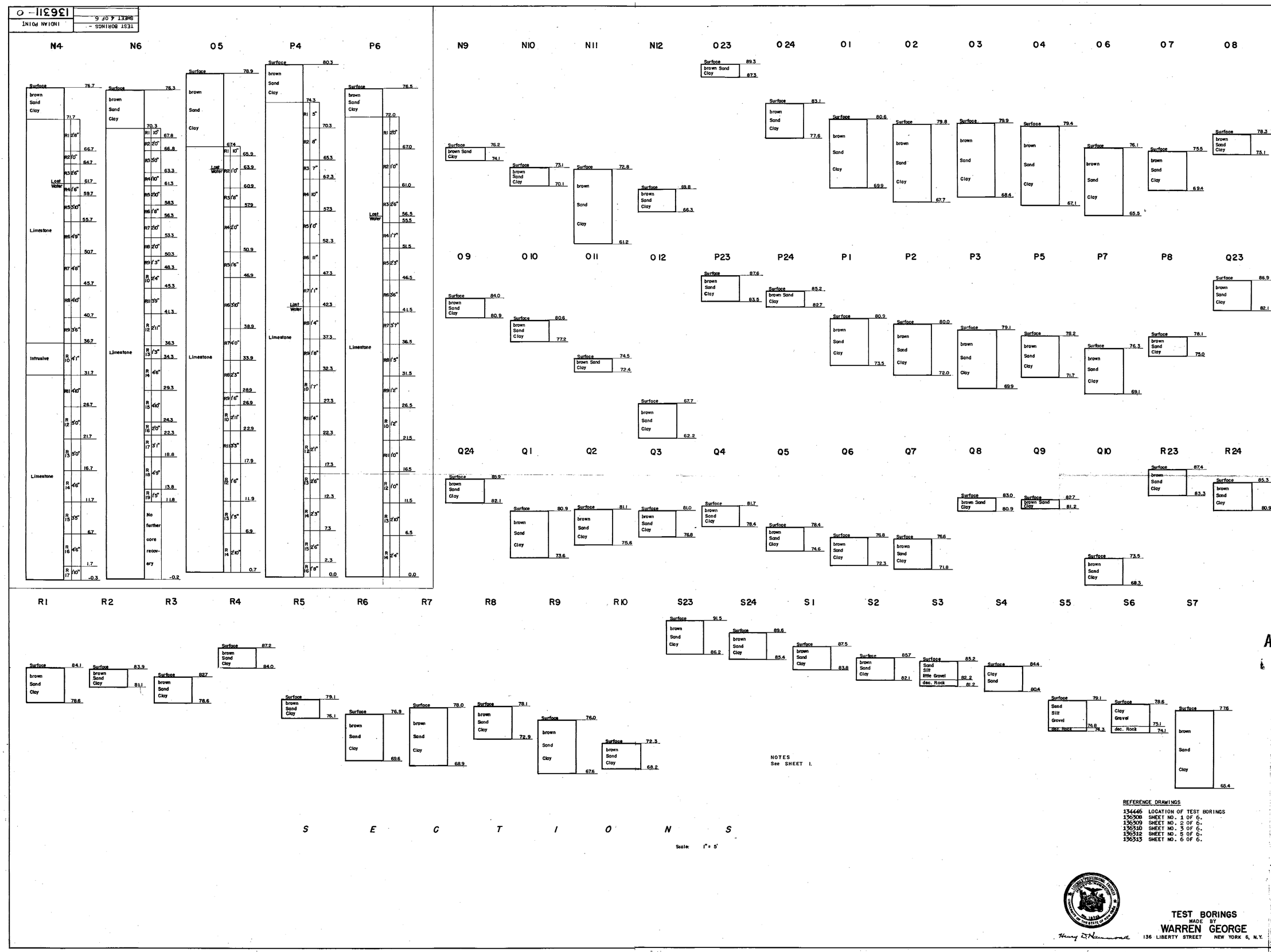
136 LIBERTY STREET NEW YORK 6, N.Y.

TEST BORINGS

UNIT #1

Dwg. 136311

TEST BORINGS - INDIAN POINT
SHEET 4 OF 6
136311-0



NOTES
See SHEET 1.

REFERENCE DRAWINGS

134446	LOCATION OF TEST BORINGS
136308	SHEET NO. 1 OF 6.
136309	SHEET NO. 2 OF 6.
136310	SHEET NO. 3 OF 6.
136312	SHEET NO. 5 OF 6.
136313	SHEET NO. 6 OF 6.

Henry D. Hammond

TEST BORINGS
MADE BY
WARREN GEORGE
136 LIBERTY STREET NEW YORK 6, N.Y.

TERA APERTURE CARD

REVISIONS
0 RELEASED FOR CONSTRUCTION CMC 12-24-56

136311-0
LETTERS "A" TO "N" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT
TEST BORINGS
SHEET 4 OF 6

SCALE 1" = 5'-0" | DRAWN BY | RECOMMENDED
J.O. 54300-M | CHECKED BY | ENG.
DIAG. | ELECT. | STRUCT. | DIST.

APPROVALS
CHIEF DRAFTSMAN S. P. Kuhls 9/6/56
INSIDE PLANT ENG.
MECH. PLANT ENG.
STRUCTURAL ENGINEER J. R. Kilcommons 9/6/56
DISTRIBUTION ENGINEER
OUTSIDE PLANT ENGINEER
DISTRIBUTION STRUCTURES ENG.
CHIEF PROJ. ENG. J. W. Ballou 9/6/56
OPERATION

TEST BORINGS

UNIT #1

Dwg. 136312

TEST BORINGS	136312-0
SHEET 5 OF 6	
INDIAN POINT	

NOTES
See SHEET 1.
Probings 1-64 arranged in approximate physical order along road from 1 (SW end) to 36 (NE end) including probings for 16-in. water mains which were made at and NW of road crossing; remainder arranged from Water Tank back along line of 16-in. mains to road crossing.

S E C T I O N S

Scale: 1" = 5'

REFERENCE DRAWINGS
136446 LOCATION OF TEST BORINGS
136308 SHEET NO. 1 OF 6.
136309 SHEET NO. 2 OF 6.
136310 SHEET NO. 3 OF 6.
136311 SHEET NO. 4 OF 6.
136313 SHEET NO. 6 OF 6.

TERA APERTURE CARD

TEST BORINGS
MADE BY
WARREN GEORGE
136 LIBERTY STREET NEW YORK 6, N.Y.

Henry D. Hammond

REVISIONS
0 RELEASED FOR CONSTRUCTION CMG 12-24-56

136312-0
LETTERS "A" TO "N" INDICATE DRAWING IS NOT FOR CONSTRUCTION
CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT
TEST BORINGS
SHEET 5 OF 6

SCALE 1" = 5'-0" | DRAWN BY | RECORDED 9-7-56 R.
JO. 54300-M | CHECKED BY | LHG
DIAG. | ELECT. | MECH. | STRUCT. | DIST.

APPROVALS
CHIEF DRAFTSMAN J. P. Kuehl 9/5/56
INSIDE PLANT ENG.
MECH. PLANT ENG.
STRUCTURAL ENGINEER J. R. Kulkarney 9/6/56
DISTRIBUTION ENGINEER
OUTSIDE PLANT ENGINEER
DISTRIBUTION STRUCTURES ENG.
CHIEF PROJ. ENG. J. Kulkarney 9/6/56
OPERATION

TEST BORINGS

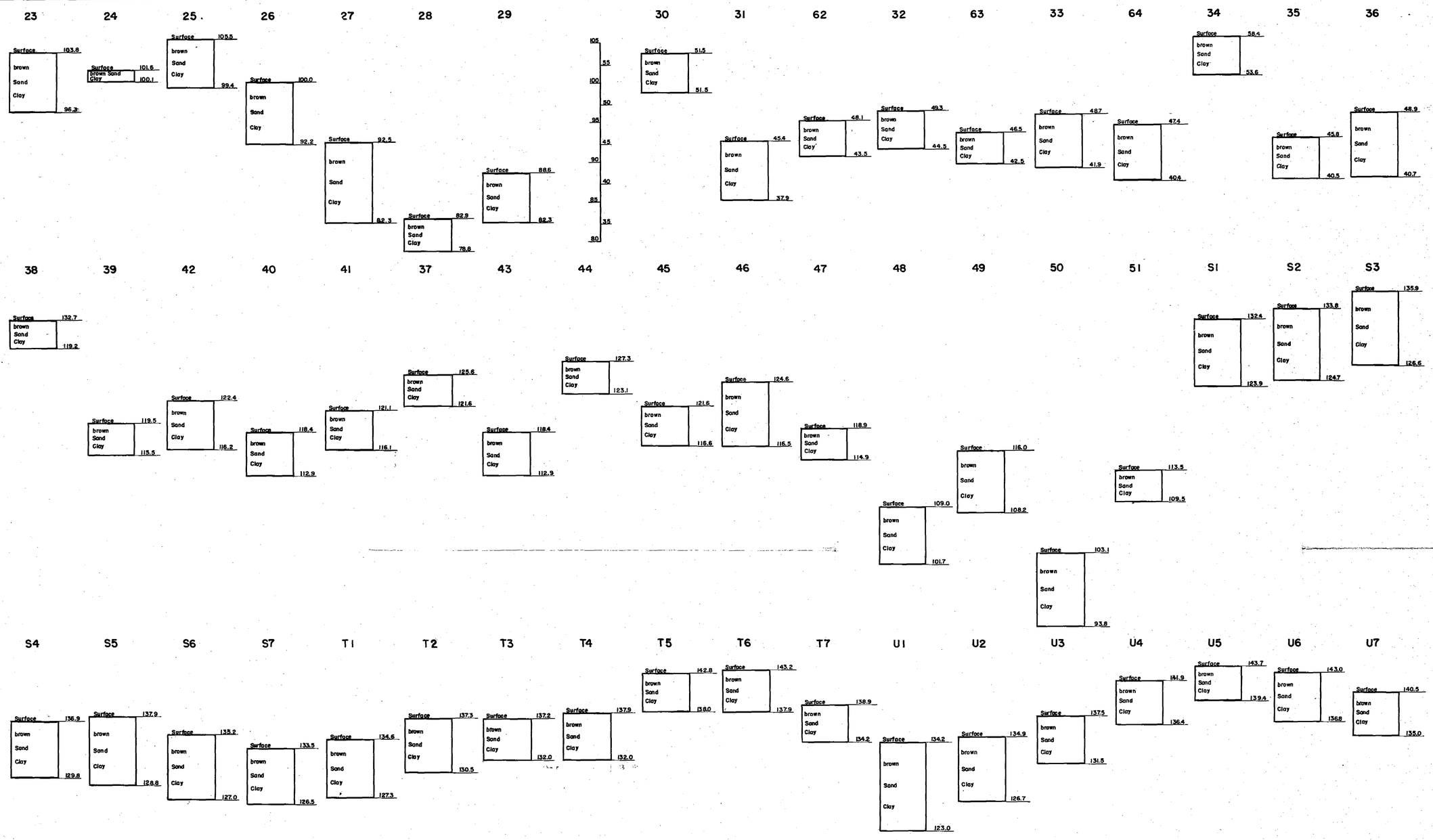
UNIT #1

Dwg. 136313

136313-0

INDIAN POINT

TEST BORINGS - SHEET 6 OF 6



S E C T I O N S

Scale: 1" = 5'

NOTES
See SHEET 1.

REFERENCE DRAWINGS

134446	LOCATION OF TEST BORINGS
136308	SHEET NO. 1 OF 6.
136309	SHEET NO. 2 OF 6.
136310	SHEET NO. 3 OF 6.
136311	SHEET NO. 4 OF 6.
136312	SHEET NO. 5 OF 6.

TERA APERTURE CARD

REVISIONS

11-29-56	0

RELEASED FOR CONSTRUCTION
CMS 12-24-56

136313-0

LETTERS "A" TO "M" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO. OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT
TEST BORINGS
SHEET 6 OF 6

SCALE 1" = 5'-0"	DRAWN BY		RECORDED 9-7-56 R
J.O. 54300-M	CHECKED BY		ENG
DIAG.	ELECT.	MECH. STRUCT	DIST.

APPROVALS

CHIEF DRAFTSMAN	F. P. Kuhl 9/5/56
INSIDE PLANT ENG.	
MECH. PLANT ENG.	
STRUCTURAL ENGINEER	J. P. Gallagher 9/6/56
DISTRIBUTION ENGINEER	
OUTSIDE PLANT ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
CHIEF PROJ. ENG.	J. W. Hallon 9/6/56
OPERATION	

LICENSED PROFESSIONAL ENGINEER HENRY D. HAMMOND STATE OF NEW YORK

Henry D. Hammond

TEST BORINGS
MADE BY
WARREN GEORGE
136 LIBERTY STREET NEW YORK 6, N.Y.

LOCATION OF TEST BORINGS

UNIT #2

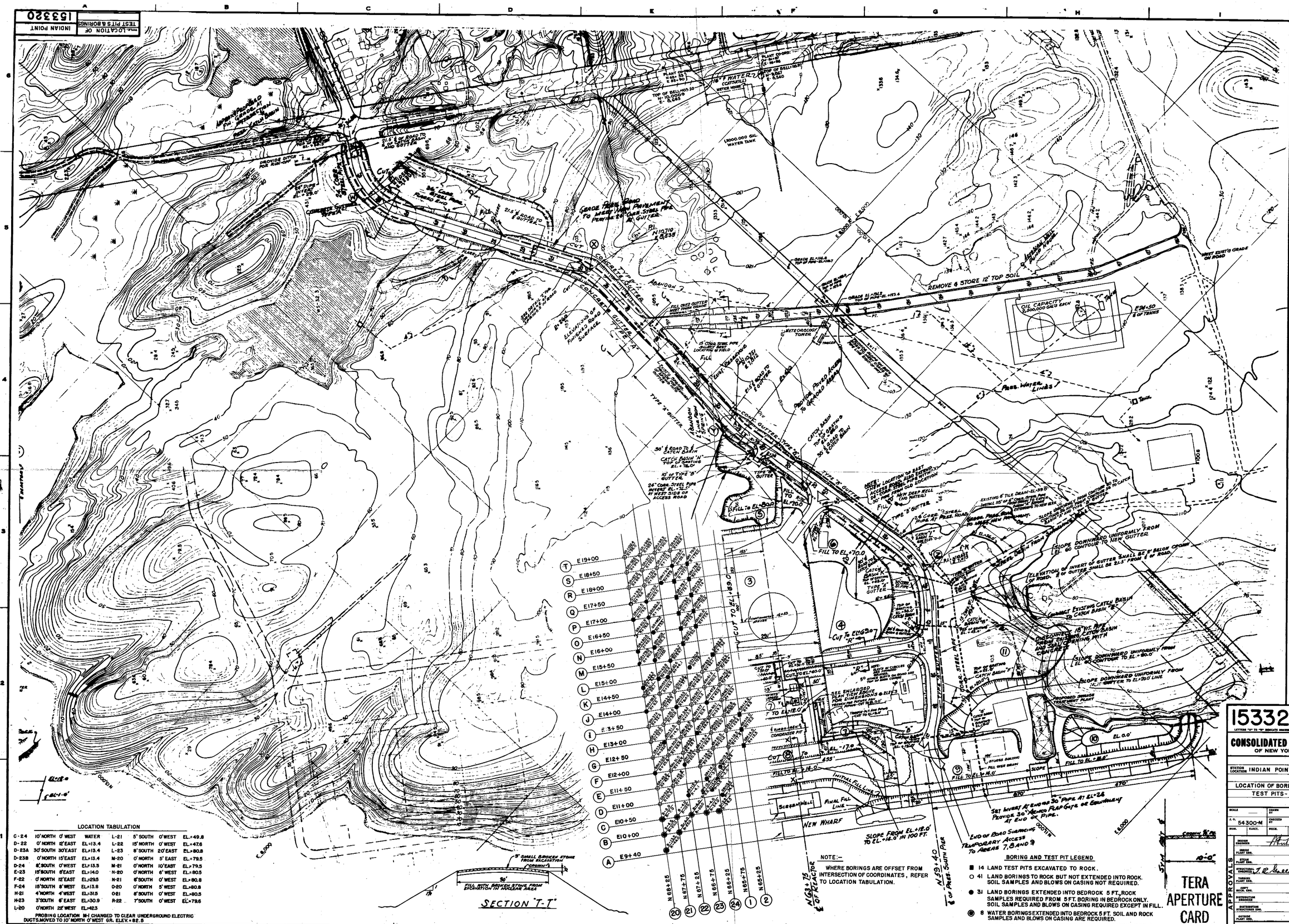
Dwg. 153320

153320

TEST PITS & BORINGS

LOCATION OF

INDIAN POINT



LOCATION TABULATION

C-24	10' NORTH	0' WEST	WATER	L-21	5' SOUTH	0' WEST	EL.=49.8
D-22	0' NORTH	12' EAST	EL.=13.4	L-22	15' NORTH	0' WEST	EL.=47.6
D-23A	30' SOUTH	30' EAST	EL.=13.4	L-23	8' SOUTH	20' EAST	EL.=80.8
D-23B	0' NORTH	15' EAST	EL.=13.4	M-20	0' NORTH	5' EAST	EL.=79.5
D-24	8' SOUTH	0' WEST	EL.=13.5	M-21	0' NORTH	10' EAST	EL.=79.5
E-23	15' SOUTH	6' EAST	EL.=14.0	N-20	0' NORTH	6' WEST	EL.=80.5
F-22	0' NORTH	12' EAST	EL.=29.5	N-21	6' SOUTH	0' WEST	EL.=80.8
F-24	15' SOUTH	8' WEST	EL.=13.8	O-20	0' NORTH	5' WEST	EL.=80.8
H-21	4' NORTH	4' WEST	EL.=31.5	O-21	8' SOUTH	0' WEST	EL.=80.5
H-23	3' SOUTH	6' EAST	EL.=30.9	R-22	7' SOUTH	0' WEST	EL.=79.6
L-20	0' NORTH	22' WEST	EL.=42.3				

PROBING LOCATION M-I CHANGED TO CLEAR UNDERGROUND ELECTRIC DUCTS. MOVED TO 10' NORTH 0' WEST GR. ELEV.=82.5

NOTE:- WHERE BORINGS ARE OFFSET FROM INTERSECTION OF COORDINATES, REFER TO LOCATION TABULATION.

BORING AND TEST PIT LEGEND

- ■ 14 LAND TEST PITS EXCAVATED TO ROCK.
- ○ 41 LAND BORINGS TO ROCK BUT NOT EXTENDED INTO ROCK. SOIL SAMPLES AND BLOWS ON CASING NOT REQUIRED.
- ● 31 LAND BORINGS EXTENDED INTO BEDROCK 5 FT., ROCK SAMPLES REQUIRED FROM 5 FT. BORING IN BEDROCK ONLY. SOIL SAMPLES AND BLOWS ON CASING REQUIRED EXCEPT IN FILL.
- ◉ 8 WATER BORINGS EXTENDED INTO BEDROCK 5 FT. SOIL AND ROCK SAMPLES AND BLOWS ON CASING ARE REQUIRED.

THE DATUM FOR ALL ELEVATIONS IS M.S.L. AT SANDY HOOK.

REVISIONS

RELEASED FOR FIELD WORK SPEC. SE 125-60	12-8-60	0
ADDED PROBING DATA — CES	6-6-61	1
ADDED PROBING DATA — CES	11-12-61	2
ADDED PROBING DATA — FES	11-21-61	3
ADDED PROBING ROBING-FTEST PIT DATA	5-31-64	4

153320-4

LETTERS "A" TO "M" INDICATE DRAWINGS IN USE FOR CONSTRUCTION

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT

LOCATION OF BORINGS AND TEST PITS - 1960

SCALE		DRAWN BY	RECORDED BY F.H.C.R.
J.O. 54300-M		CHECKED BY	ENG.
ENGR.	ELECT.	MECH.	STRUCT.	DIST.

APPROVALS

DESIGN ENGINEER	4-11-61
DIRECT PLANT ENG.	
STEAM PLANT ENG.	
STRUCTURAL ENGINEER	T. R. Kelleway 4/11/61
INDIRECT PLANT ENG.	
CONT. MECH. ENG.	
DISTRIBUTION ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
OUTSIDE PLANT ENG.	
PRODUCTION DEPT.	

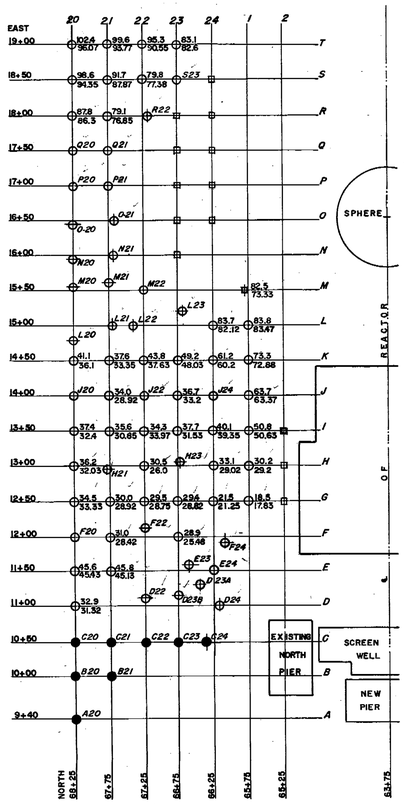
TERA APERTURE CARD

POOR ORIGINAL

TEST BORINGS

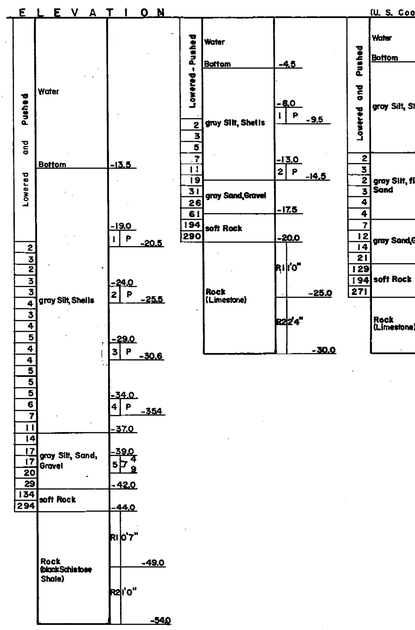
UNIT #2

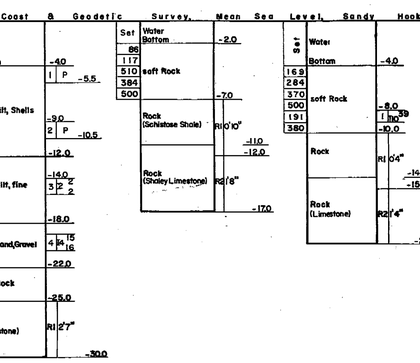
Dwg. 154276

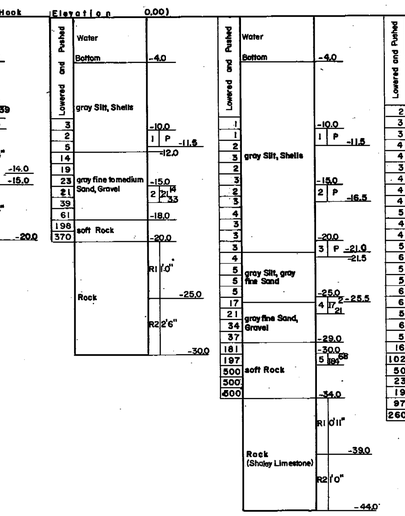


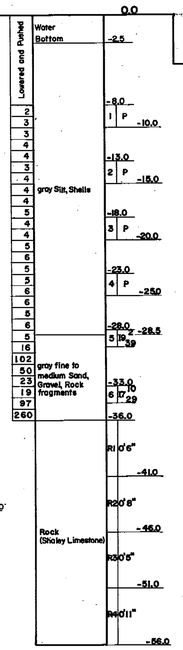
LEGEND
32.9 (El. Surface) Probing
31.32 (El. Refusal)
21 Boring, Land
21 Boring, Water
Test Pit

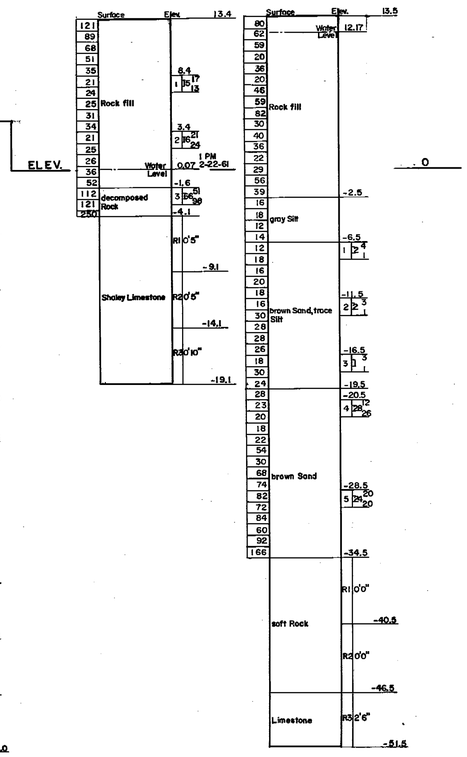
PLAN



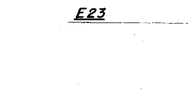


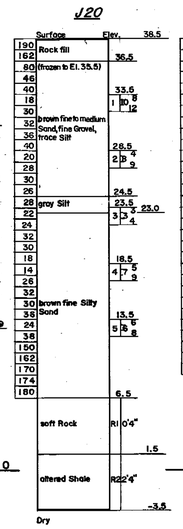


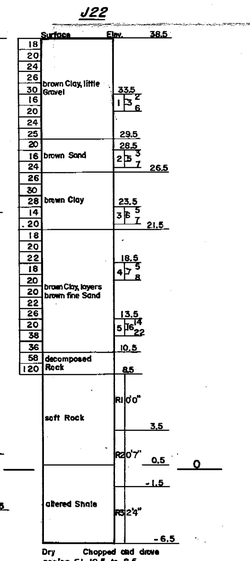


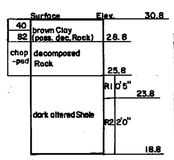


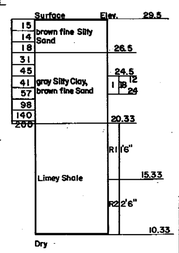
NOTES
See SHEET 2

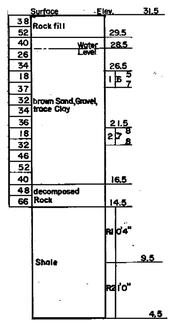


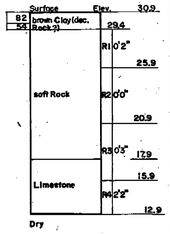


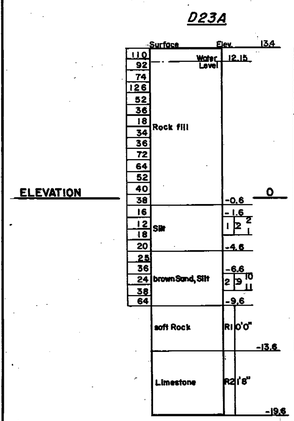


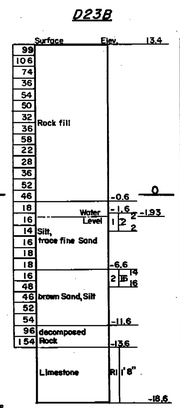


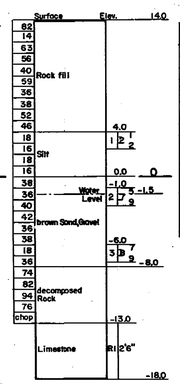


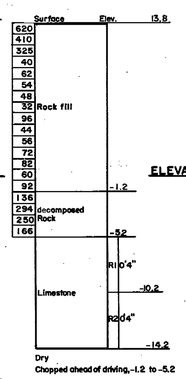


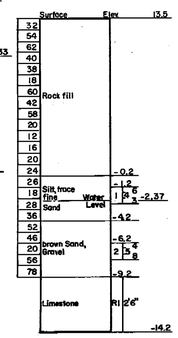












ELEVATION

TERA APERTURE CARD

CONTRACTOR
WARREN GEORGE ASSOCIATES
95 RIVER STREET HOBOKEN, N.J.

SHEET 1 OF 2 SHEETS

SHEET #1

154276-0

LETTERS "A" TO "R" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION: INDIAN POINT

BORING, PROBING & TEST PIT DATA
UNIT # 2

SCALE	DRAWN BY H. HAMMOND	RECORDED
J.O. 60200-M	CHK'KED BY	ENG.
DIAG. / ELECT. / MECH. / STRUCT.		DRFT.

APPROVALS

DESIGN ENGINEER	4-11-61
INDIAN PLANT ENG.	
STEAM PLANT ENG.	
STRUCTURAL ENGINEER	J. R. Galloway 4/11/61
TURBINE PLANT ENG.	
ASST MECH. ENG.	
DISTRIBUTION ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
OUTSIDE PLANT ENG.	
PRODUCTION DEPT.	

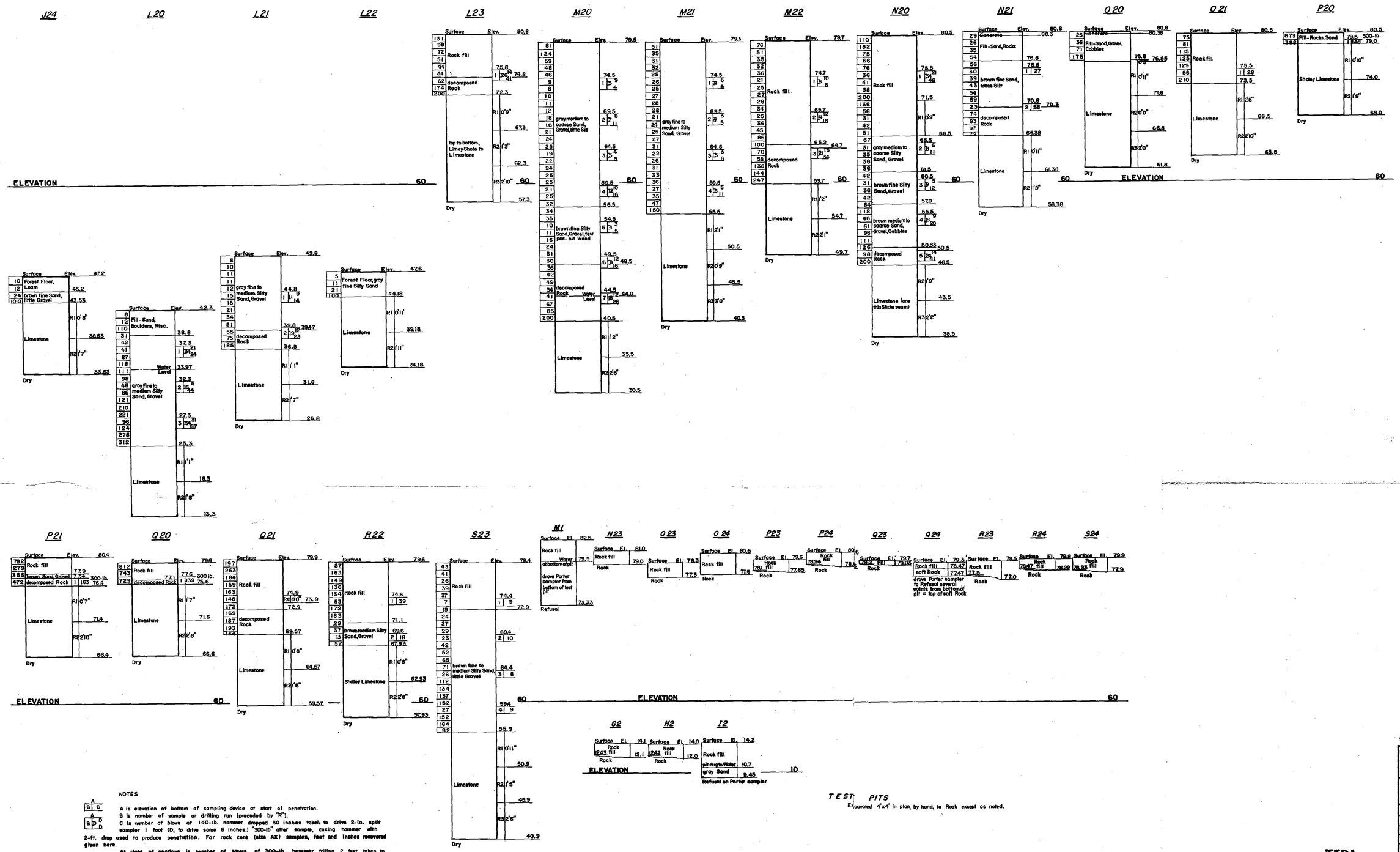
18,705

TEST BORINGS

UNIT #2

Dwg. 154277

154277



NOTES

A is elevation of bottom of sampling device at start of penetration.
B is number of sample or drilling run (preceded by 'R').
C is number of blows of 140-lb. hammer dropped 30 inches taken to drive 2-in. split sampler 1 foot (D, to drive same 6 inches.) "300-lb" after sample, casing hammer with 2-ft. drop used to produce penetration. For rock core (size AX) samples, feet and inches recovered given here.

At right of sections is number of blows of 300-lb. hammer falling 2 feet taken to drive 2-1/2-in. casing each foot, and odd distance shown to refusal.

For Probings, and where noted on Test Pits, 1-in. Porter Sampler driven to refusal with 35-lb. hammer falling 30 inches.

TEST PITS
Excavated 4'x4' in plan, by hand, to Rock except as noted.

TERA APERTURE CARD

CONTRACTOR
WARREN GEORGE ASSOCIATES
95 RIVER STREET HOBOKEN, N.J.

18,705

SHEET 2 OF 2 SHEETS

SHEET # 2

154277-0

LETTERS "A" TO "M" INDICATE DRAWING IS NOT FOR CONSTRUCTION

CONSOLIDATED EDISON CO.
OF NEW YORK, INC.

STATION LOCATION INDIAN POINT

BORING, PROBING & TEST PIT DATA
UNIT # 2

SCALE	DRAWN BY H. HAMMOND	RECORDED
J.O. 60200-M	CHECKED BY	ENG.
DIAG. / ELECT. / MECH. / STRUCT.		DATE

APPROVALS

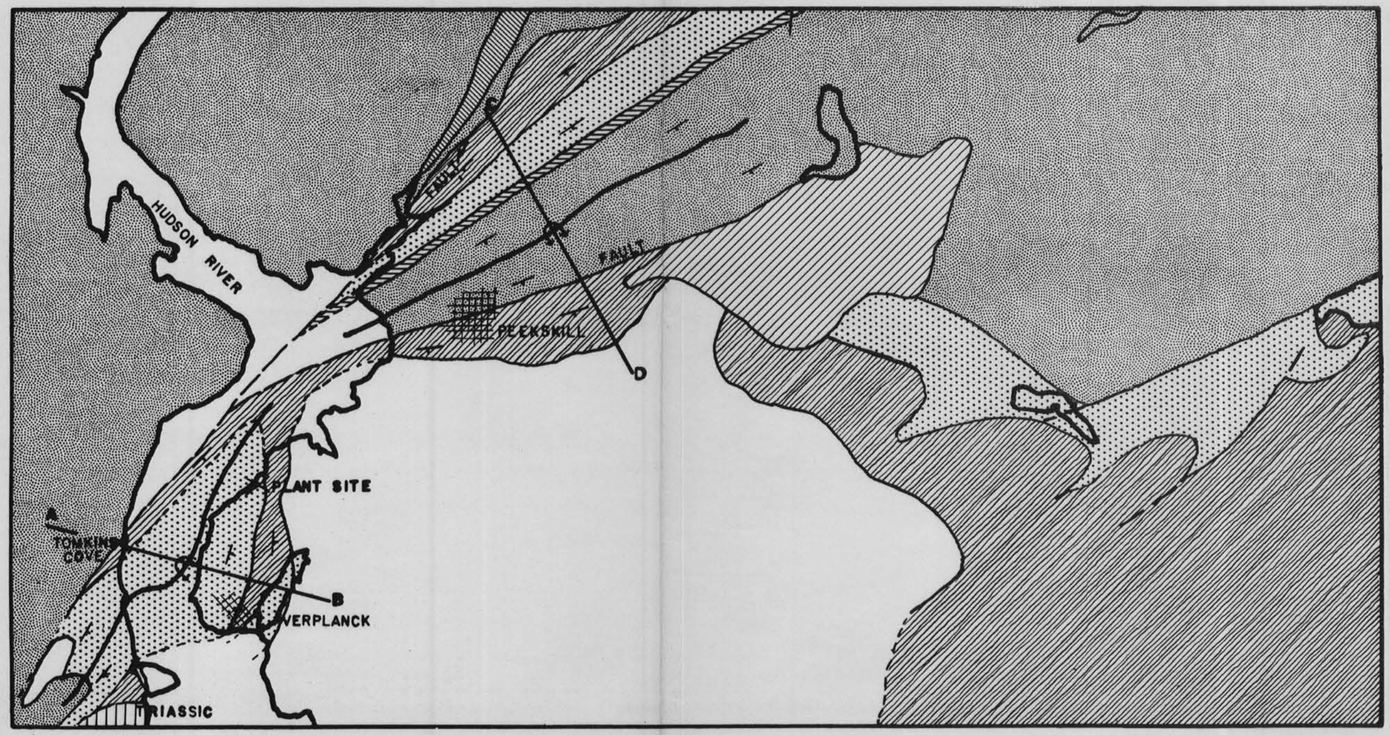
DESIGN ENGINEER	4-11-61
INDIAN PLANT ENG.	
STEAM PLANT ENG.	
STRUCTURAL ENGINEER	J. R. Kennedy 4/11/61
TURBINE PLANT ENG.	
ASS'T MECH. ENG.	
DISTRIBUTION ENGINEER	
DISTRIBUTION STRUCTURES ENG.	
OUTSIDE PLANT ENG.	
PRODUCTION DEPT.	

A (17)

FIGURE 2
GENERALIZED GEOLOGIC MAP OF THE LOWER THIRD OF THE WEST POINT QUADRANGLE, NEW YORK.

FORMATION BOUNDARIES, FAULT LINES, AND IGNEOUS CONTACTS AFTER C. P. BERKEY. STRUCTURAL AXES AND CORRELATION OF FORMATIONS BY SIDNEY PAIGE.

FORMATIONS SOUTH OF PEEKSKILL, AND EAST OF THE HUDSON RIVER, REGARDED BY BERKEY AS OF "DOUBTFUL" AGE AND MAPPED AS INWOOD LIMESTONE AND MANHATTAN SCHIST, ARE HERE MAPPED AS CAMBRO-ORDOVICIAN IN AGE.



DIORITE & NORITE
PEEKSKILL GRANITE
CORTLANDT COMPLEX

HUDSON RIVER SHALE AND PHYLLITE

WAPPINGER LIMESTONE

POUGHQUAG QUARTZITE

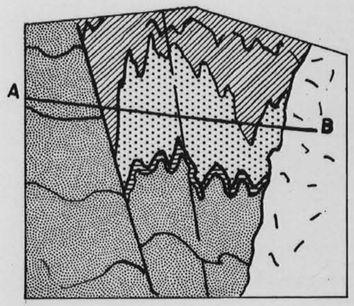
LIMESTONE (PALEOZOIC)?

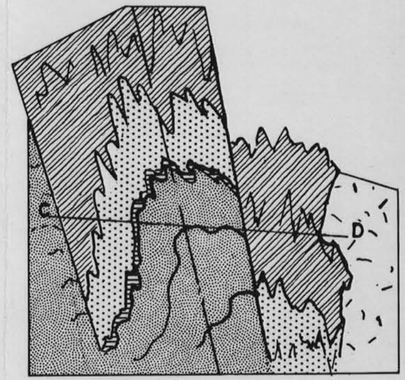
GRANITOID GNEISSES (PRE-€)

OVERTURNED AXIS

FAULTS

0 1 2 3 4
SCALE IN MILES





POOR ORIGINAL

TERA APERTURE CARD

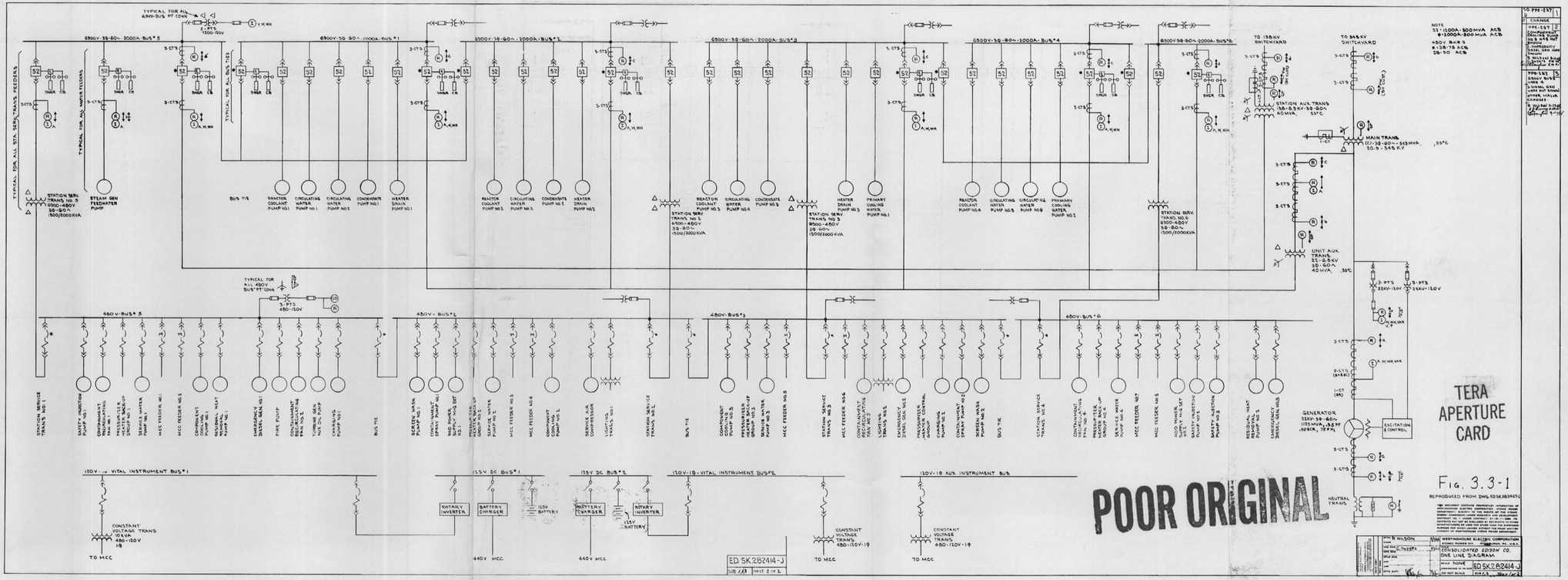


Fig. 3.3-1

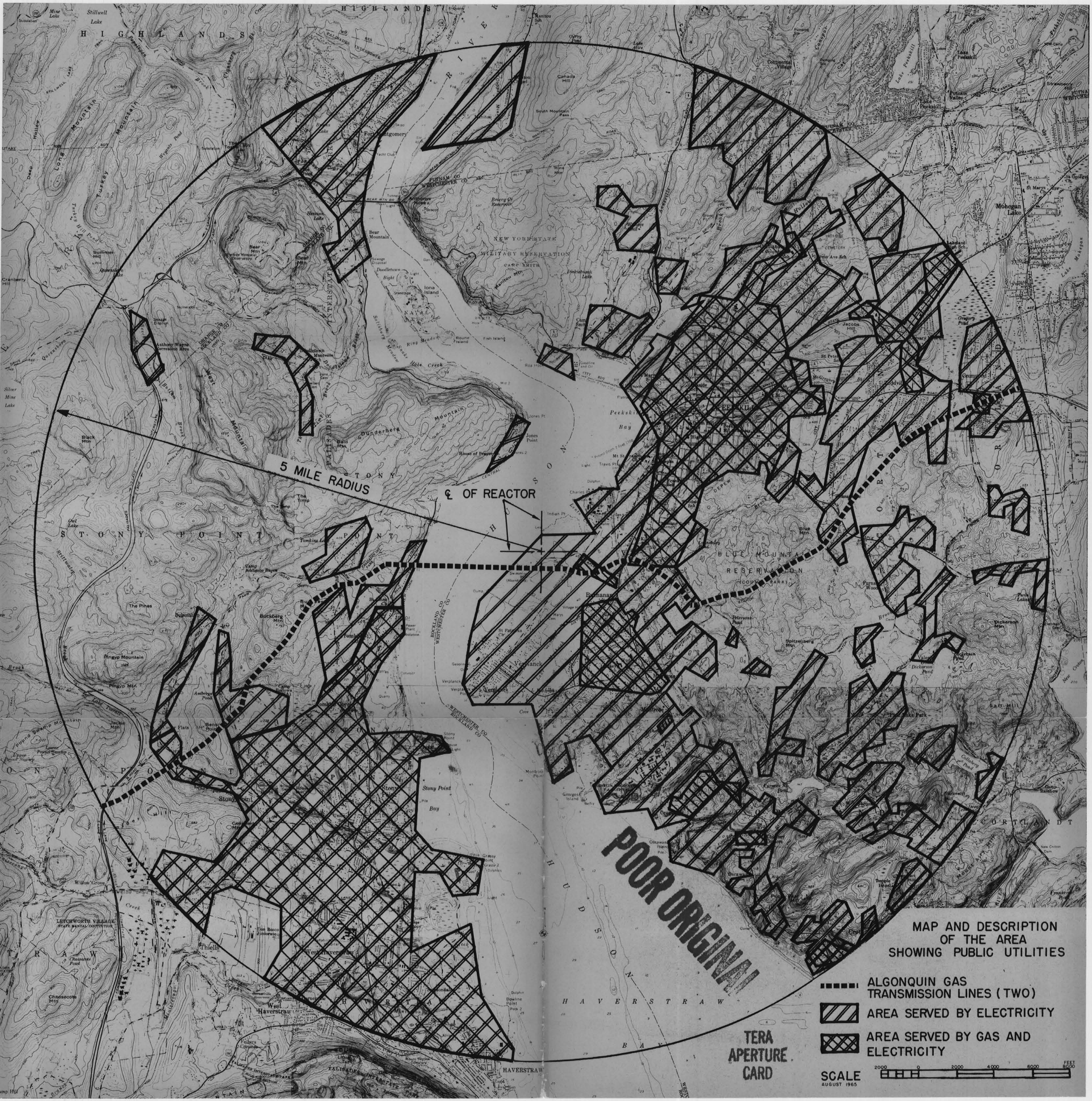
REPRODUCED FROM DWG. ED.5K.282414-J

MAP AND DESCRIPTION

OF THE AREA SHOWING

PUBLIC UTILITIES

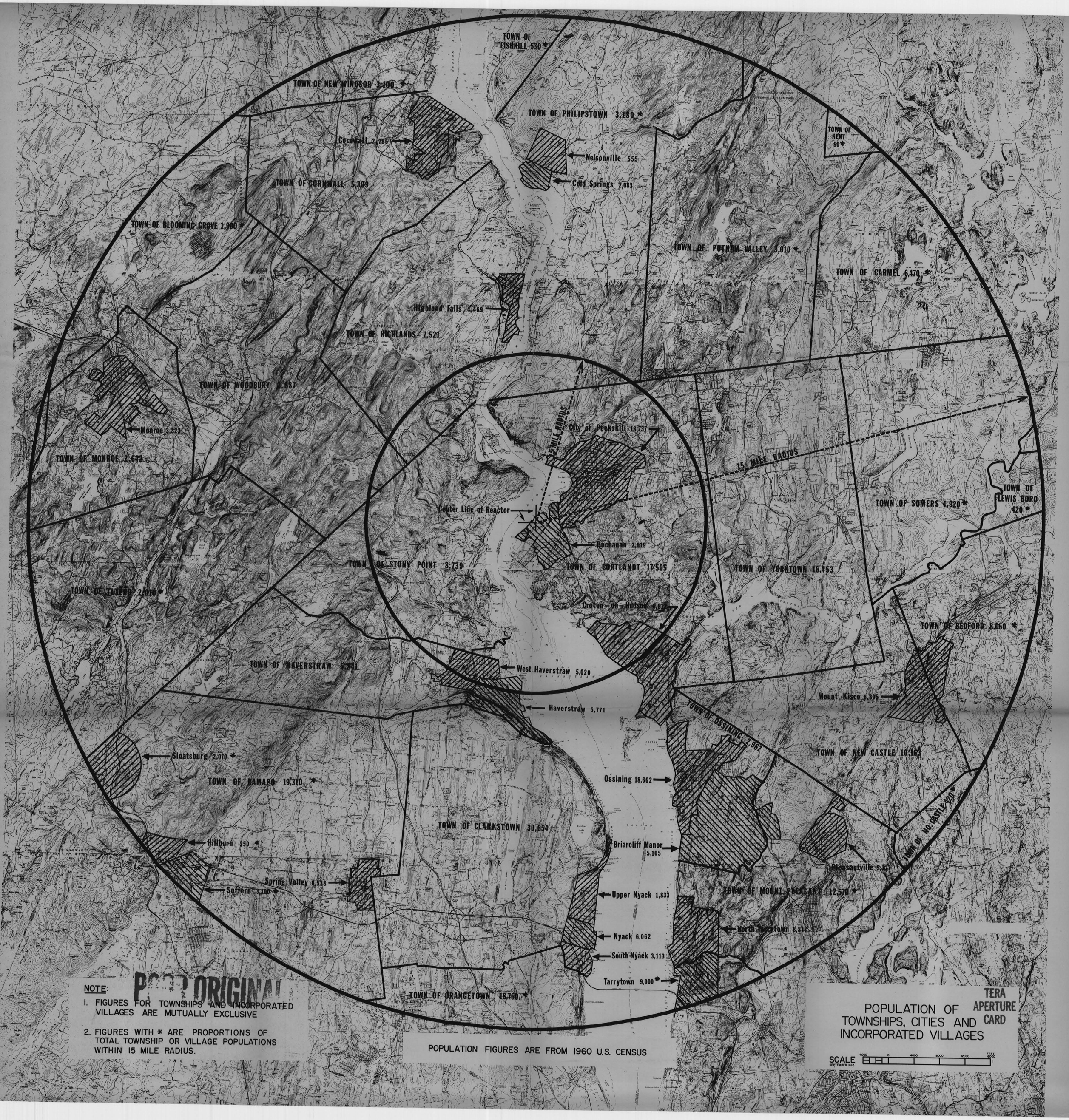
77-01-07


5 MILE RADIUS
℄ OF REACTOR
POOR ORIGINAL
TERA APERTURE CARD
MAP AND DESCRIPTION OF THE AREA SHOWING PUBLIC UTILITIES
ALGONQUIN GAS TRANSMISSION LINES (TWO)
AREA SERVED BY ELECTRICITY
AREA SERVED BY GAS AND ELECTRICITY
SCALE
AUGUST 1965
2000 0 2000 4000 6000 8000 FEET
HIGHLANDS
STONY POINT
NEW YORK STATE MILITARY RESERVATION
BLUE MOUNTAIN RESERVATION
PEEKSKILL
CORTLANDT
HAVERSTRAW
HUDSON RIVER
Peekskill Bay
Haverstraw Bay
Stony Point Bay
Bear Mountain
Iona Island
Anthony Wayne Recreation Area
Dunderberg Mountain
Verplanck
Buchanan
Stony Point
West Haverstraw
Haverstraw
Thiells
Letchworth Village State Mental Institution
Mohegan Lake
Lake Peekskill
Lake Meahagon
Silver Mine Lake
Black Mtn
Long Mountain
Turkey Mountain
Summer Hill
Cranberry Hill
Bald Mtn
Jones Point
Indian Pt
Blue Mtn
Dickerson Mtn
Spitzenberg Mtn
Pingyp Mountain
Buckberg Mtn
Cheesecote Mtn
Tomkins Cove
Canada Hill
South Mountain Pass
Manitou Mtn
Dieterichs Lake
Round Island
Fish Island
Ring Meadow
Stockade Hole Creek
Doodletown Bight
Brewer Cr Reservoir
Hessian Lake
Bear Mtn Bridge
Anthony Rose
Fort Montgomery
Jacobs Hill
Gregory Pond
Crompond
Continental Village
Strawberry Hill
Putnam Valley
Lake Peekskill
Oscawana Island
Georges Island
Bowline Point
Dolphin
Grassy Point
Willow Grove
Tion Bosco Juniorate
Cheesecote Pond
Palisades Interstate Park
Rockland Co
Westchester Co
Putnam Co
Orange Co

POPULATION OF

TOWNSHIPS, CITIES AND

INCORPORATED VILLAGES



NOTE:

1. FIGURES FOR TOWNSHIPS AND INCORPORATED VILLAGES ARE MUTUALLY EXCLUSIVE
2. FIGURES WITH * ARE PROPORTIONS OF TOTAL TOWNSHIP OR VILLAGE POPULATIONS WITHIN 15 MILE RADIUS.

POOR ORIGINAL

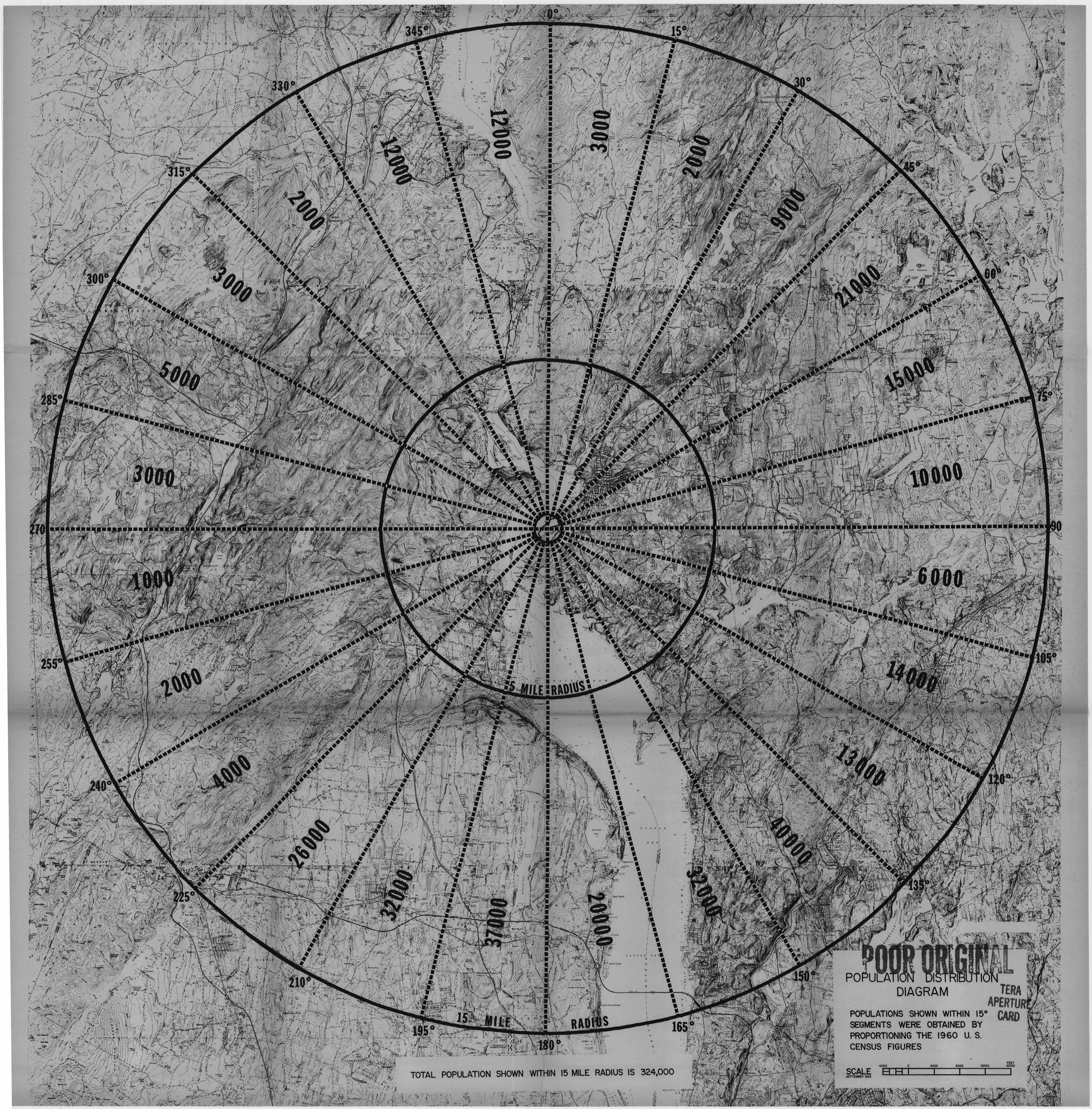
POPULATION FIGURES ARE FROM 1960 U.S. CENSUS

POPULATION OF TOWNSHIPS, CITIES AND INCORPORATED VILLAGES

TERA APERTURE CARD

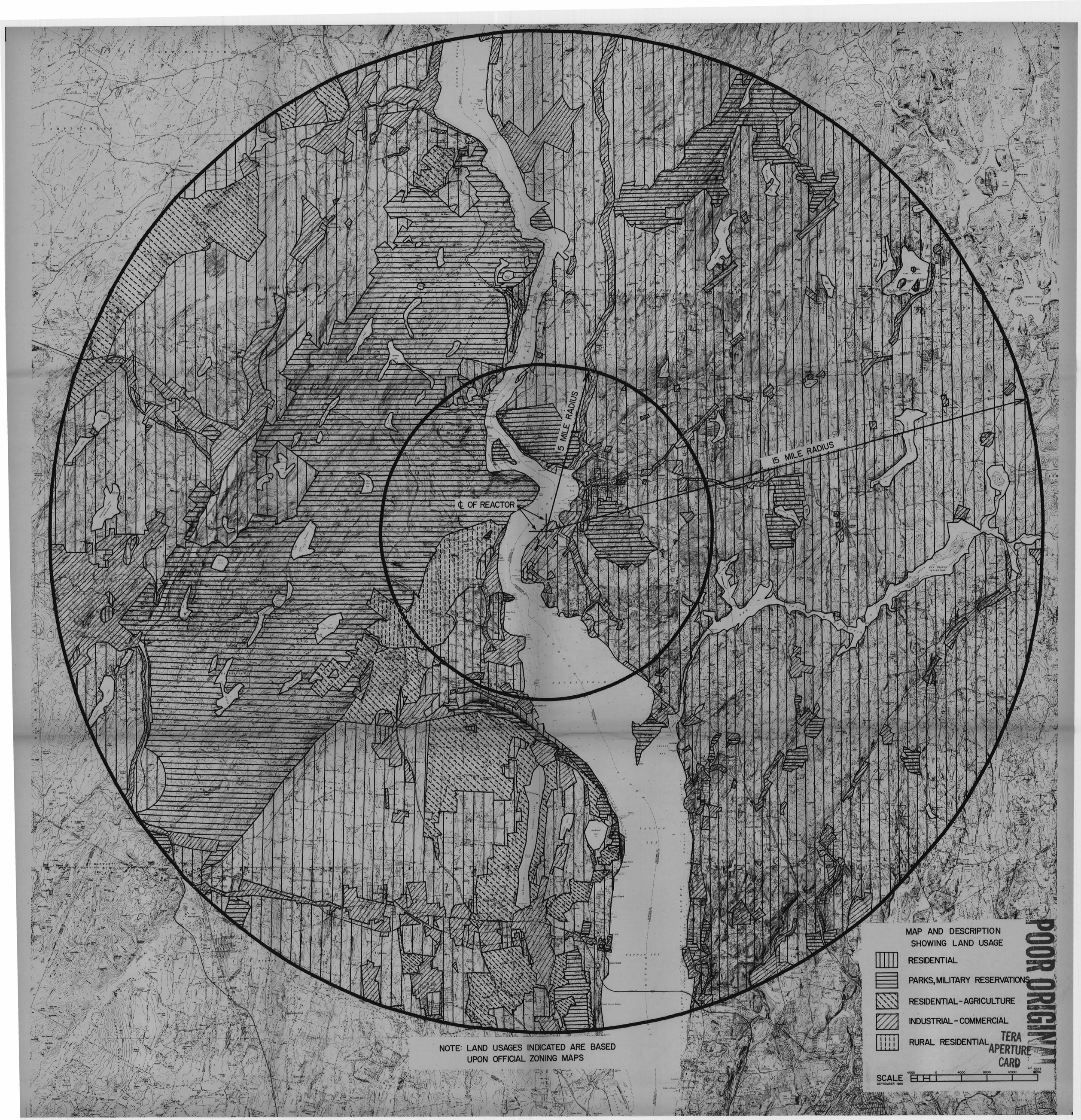
SCALE SEPTEMBER 1965 — 4000 0 4000 8000 12000 FEET 16000

POPULATION DISTRIBUTION DIAGRAM

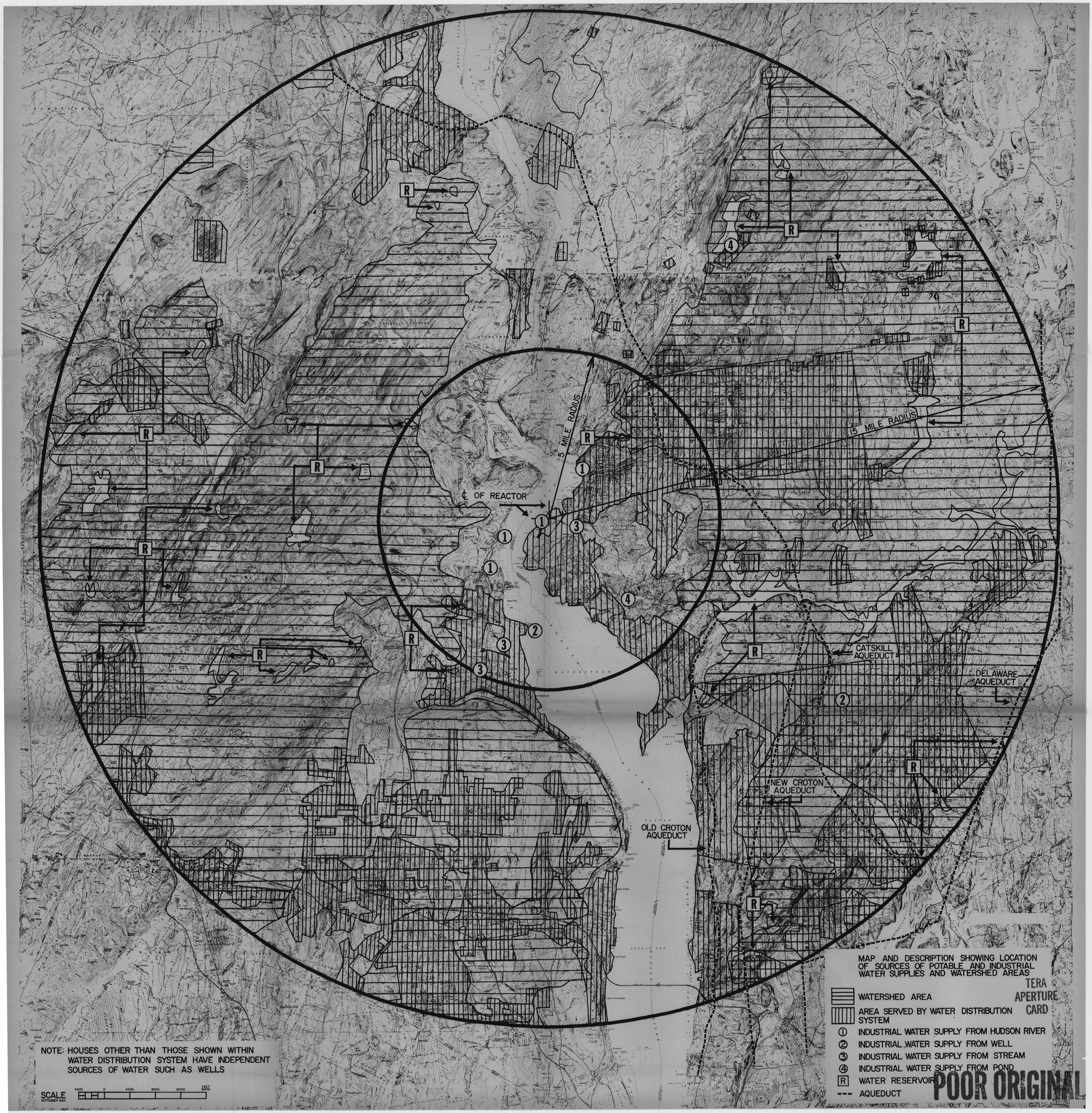

0°
15°
30°
45°
60°
75°
90
105°
120°
135°
150°
165°
180°
195°
210°
225°
240°
255°
270
285°
300°
315°
330°
345°
12000
12000
3000
2000
9000
21000
15000
10000
6000
14000
13000
40000
32000
20000
37000
32000
26000
4000
2000
3000
5000
3000
1000
2000
15 MILE RADIUS
POOR ORIGINAL
POPULATION DISTRIBUTION DIAGRAM
TERA APERTURE CARD
POPULATIONS SHOWN WITHIN 15° SEGMENTS WERE OBTAINED BY PROPORTIONING THE 1960 U. S. CENSUS FIGURES
SCALE
SEPTEMBER 1965
4000 0 4000 8000 12000 FEET 16000
TOTAL POPULATION SHOWN WITHIN 15 MILE RADIUS IS 324,000

MAP AND DESCRIPTION

SHOWING LAND USAGE


15 MILE RADIUS
15 MILE RADIUS
℄ OF REACTOR
MAP AND DESCRIPTION
SHOWING LAND USAGE
RESIDENTIAL
PARKS, MILITARY RESERVATIONS
RESIDENTIAL - AGRICULTURE
INDUSTRIAL - COMMERCIAL
RURAL RESIDENTIAL
SCALE
SEPTEMBER 1965
4000 0 4000 8000 12000 16000 FEET
NOTE: LAND USAGES INDICATED ARE BASED
UPON OFFICIAL ZONING MAPS
POOR ORIGINAL
TERA
APERTURE
CARD

MAP AND DESCRIPTION SHOWING LOCATION OF SOURCES OF POTABLE AND INDUSTRIAL WATER SUPPLIES AND WATERSHED AREAS



TERA APERTURE CARD
POOR ORIGINAL

MAP AND DESCRIPTION

OF THE AREA

SHOWING SEWAGE SYSTEMS

MAP AND DESCRIPTION
OF THE AREA
SHOWING SEWAGE SYSTEMS
AREA SERVED BY SEWAGE SYSTEM
SEWAGE TREATMENT PLANT
INDUSTRIAL WASTE TREATMENT PLANT
TERA
APERTURE
CARD
SCALE
AUGUST 1965
2000 0 2000 4000 6000 FEET 8000
℄ OF REACTOR
5 MILE RADIUS
HIGHLANDS
STONY POINT
CORTLANDT
HUDSON RIVER
HAVERSTRAW BAY
Peekskill Bay
Stony Point Bay
BLUE MOUNTAIN RESERVATION (COUNTY PARK)
Bear Mountain
Fort Montgomery
Iona Island
Indian Pt
Buchanan
Verplanck
Montrose
Tomkins Cove
Stony Point
West Haverstraw
HAVERSTRAW
Peekskill
Mohegan Lake
Crompond
Toddville
Van Cortlandtville
Lake Peekskill
Oscawana