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RADIATION AND SHIELDING MEASUREMENTS

DURING INITIAL OPERATION OF DRESDEN

NUCLEAR POWER STATION

By

W. D. Craig

February 15, 1961

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

SAN JOSE, CALIFORNIA

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TITLE PAGE

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TITLE

RADIATION AND SHIELDING MEASUREMENTS DURING INITIAL OPERATION OF DRESDEN NUCLEAR POWER STATION

ABSTRACT

This report summarizes the radiation and shielding measurements, and experience gained, during the initial operation of the 180 MW(e) Dresden Nuclear Power Station.

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CONCLUSIONS

As the first large boiling water reactor power plant, Dresden afforded the first comprehensive check of many radiation and shielding questions. Radiation scatter above and below the reactor vessel forced the addition of some concrete shield blocks. With this exception, the shielding as designed was adequate, and conservative, for operation and maintenance of the plant. No shielding was required around the turbine. Measurements of neutron flux and gamma heating in the reactor vessel showed both to be well within design criteria. Residual radiation at reactor shutdown is not a major factor in the maintenance of the plant at this time. Good agreement with theoretical predictions was noted for Nitrogen-16 radiation, controlling access to steam equipment and primary water circuit. Significant savings in shielding of the reactor and plant equipment are possible in the design of future plants, based upon data gathered at Dresden.

By cutting out this rectangle and folding on the center line, the above information can be fitted into a standard card file.

For list of contents—drawings, photos, etc. and for distribution see next page (FN-610-2).

INFORMATION PREPARED FOR ATOMIC POWER EQUIPMENT DEPARTMENT

TESTS MADE BY Wm. D. Craig
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Atomic Power Equipment Department

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T A B L E   O F   C O N T E N T S

I. INTRODUCTION

This report summarizes the radiation and shielding experience gained during the startup and initial operation of the Dresden Nuclear Power Station.

In this report are the first possible comparisons of measured dose rates versus predictions for a large boiling water reactor power plant. As in many other areas of Dresden's technology, the program of radiation measurements during startup and initial operation was necessary for three principal reasons:

1. A proof test of Dresden shielding adequacy, and/or determination of required shield revisions.

2. A gauge of the accuracy to which BWR shielding calculations may be made, as a guide in the design of succeeding plants.

3. Extension of basic shielding theory by measurements of those parameters upon which little information exists.

In order to satisfy these three requirements, a schedule of dose measurements was made for the several plant areas in which radiation would exist. This schedule called for general surveys, and also for particular measurements of points of interest. A copy of this schedule appears in the Appendix (Section VI) of this report.

Points of specific interest in the measurements included:

A. Reactor Enclosure Radiation

   1. Measurement of direct radiation through reactor shields.
   2. Determination of the radiation scattering problems above the top head, and below the bottom of the reactor.

3. Measurements of radiation from primary water system components.

B. Turbine Building Radiation

1. Determination of required turbine shielding, and/or maintenance access problems.

2. Measurement of Nitrogen-16 concentration in off-gas system and in condenser hot-well.

C. Radiation at Pressure Vessel Wall

1. Determination of gamma heating at vessel wall by calorimetry.

2. Measurement of neutron flux and spectrum at vessel wall.

D. Shutdown Radiation Measurements

II. SUMMARY AND CONCLUSIONS

A. Shielding and radiation protection at Dresden Station is adequate for safe operation, maintenance, and inspection of the plant.

B. All shields designed to protect against direct radiation in the Dresden plant are conservative, and such shields are being reduced in future plants. Shielding technology for large boiling water reactor power plants is now well established.

Concrete thicknesses actually needed around representative components of the Dresden system, in order to reduce the dose rate to less than 0.5 mrem/hr, are:

Turbine - None (with controlled-access area approximately 30-150 feet around turbine pedestal)

Condenser - 2-1/2 - 3 feet

Air Ejector and Off-Gas System - 4 feet

Feedwater Heaters - 2-4 feet, varying from first to last stage

Feedwater Pumps - None

Primary Steam Pipes - 3-1/2 feet

Secondary System - None

Feedwater Piping - None

Condensate Demineralizers - 2-2-1/2 feet

Primary Recirculating Water System and Secondary Steam Generators - 4 ft.

Primary Steam Drum - 4-1/2 feet

Cleanup Demineralizer - 3-3-1/2 feet

C. The influence of scattered radiation, and radiation streaming, is shown to be a factor in the design of large reactor power plant design.

D. The buildup of corrosion product activity with operating time in the primary system may be a major factor in influencing maintenance procedure of recirculation loop components. More operating time must be logged before meaningful predictions may be made.

E. Nitrogen-16 radiation, controlling plant shielding and access (during operation) to steam and recirculating water equipment, conformed reasonably to theoretical predictions. A significant example is the turbine operating floor, which was shown to require no shielding for routine inspection and maintenance.

F. Measurements were made of neutron flux and gamma heating just outside the reactor pressure vessel wall. The 40 year neutron irradiation exposure at the inside radius of the vessel wall, on core midplane elevation, will be about 1.6×10^{19} neutrons/cm^2 (nvt) from neutrons of greater than 0.1 Mev energy. This compares favorably with APED design criteria permitting no more than 5×10^{19} nvt ($>$ 0.1 Mev) in the design life of a pressure vessel.

Gamma heating in the vessel steel is a maximum of 23,500 BTU/hr-$ft.^3$ (0.03 watts/gm). This results in a calculated maximum tangential thermal stress of about 2,200 psi, tensile on the inside radius of the vessel. Since 10,000 psi is allowable under ASME Boiler and Pressure Vessel Code Case 1234, this is quite conservative.

III. RADIATION MEASUREMENTS

Gamma dose rates are entered in units of milliroentgens per hour, neutron dose rates in millirem per hour. Since the conversion from gamma milliroentgens to millirem is unity, the neutron and gamma dose rates are additive to units of millrem per hour. Neutron dose rates are followed by the designator "n". Gamma dose rates are undesignated.

Gamma measurements above 1 mr/hr were made with "Cutie Pie" portable ionization chamber ratemeters. Gamma measurements less than 1 mr/hr were made with portable Geiger counter-ratemeters and with precision dosimeters. Neutron measurements at 10, 22.5, 35, and 42 megawatts were made with portable BF3 counter-ratemeters and with Rudolph fast neutron ratemeters. Above 42 megawatts, neutron surveys were made with a double-moderator neutron dosimeter.*

Unless otherwise noted, measurements were made at a distance of 1 meter from the source surface.

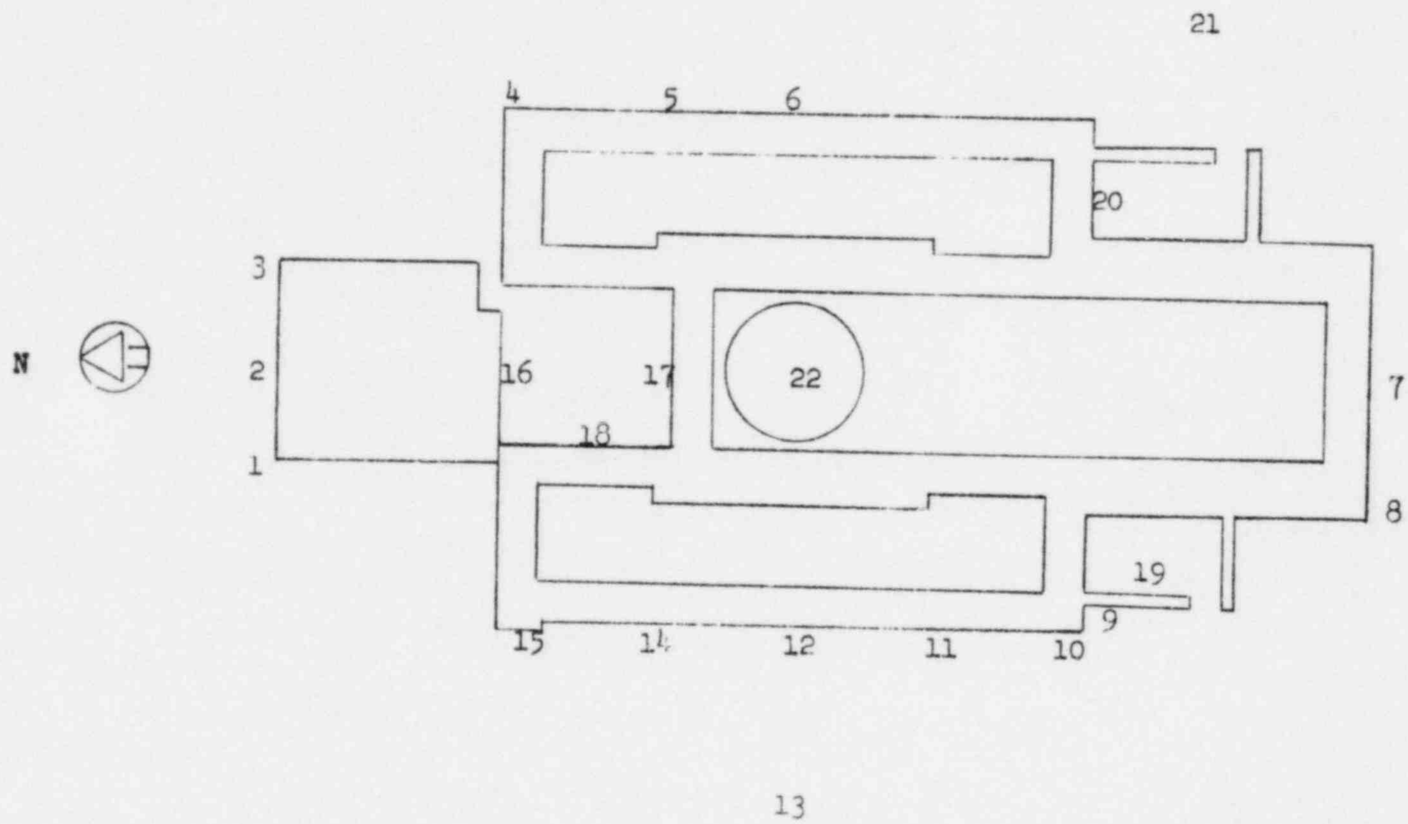
* For Description See - "Double Moderator Neutron Dosimeter," J. De Pangher, HW-57293, July 15, 1958.

A. Reactor and Enclosure Surveys During Operation
(See corresponding figures showing plan views of survey area and plant drawings in Appendix)

1. Special Radiation Survey Sheet No. 1
(Refer to Drawing 142F822 in Appendix)

Reactor Power (MWT)

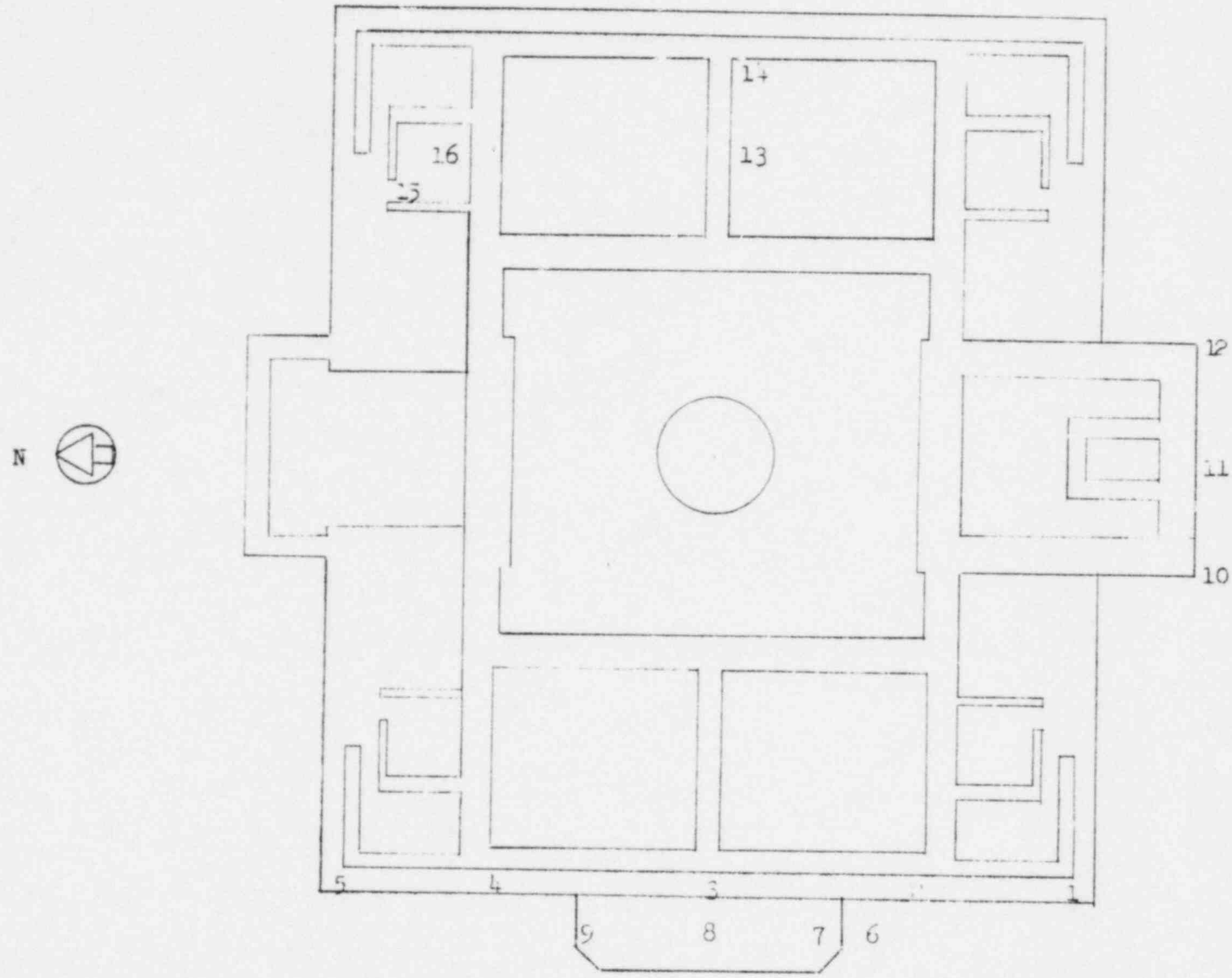
Point	10	22.5	42	309 Dual Cycle	315 Single Cycle	626 Dual Cycle
1. Primary steam drum, east-center		80 22 n				5000-8000 (Est.)
2. Primary steam drum, south-end		20 25 n				2000 1200 n (Est.)
3. Riser at southeast of steam drum		120 5 n				
4. Outside steam drum enclosure, El.634 south		0 0.02 n				0.5 0.1 n
5. Above steam drum enclosure, EL.649'6"		0 0.002 n				25 2 n



RADIATION MEASUREMENTS AT ELEVATIONS

565 ft. & 584 ft.

(For greater detail, see Drawing 142F823 in Appendix)



RADIATION MEASUREMENTS AT ELEVATION 548 ft.

(For greater detail, see Drawing 142F824 in Appendix)

2. Special Radiation Survey Sheet No. 2 (Continued)

	Reactor Power (MWT)						
	10	22.5	35	42	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
21. El. 565, 15' east of Rm. H		0.8 5.1 n					0 0.1 n
22. El. 584, above reactor top head	5 46 n					75 18 n	125 35 n

Surveys at 1, 22.5, and 35 MWT made before installation of reactor top heat shielding beams. After the 35 MWT survey, a 3 foot thickness of concrete shielding beams was installed above, and to the south of, the reactor top head to completely enclose reactor.

2. Special Radiation Survey Sheet No. 2

Reactor Power (MWT)

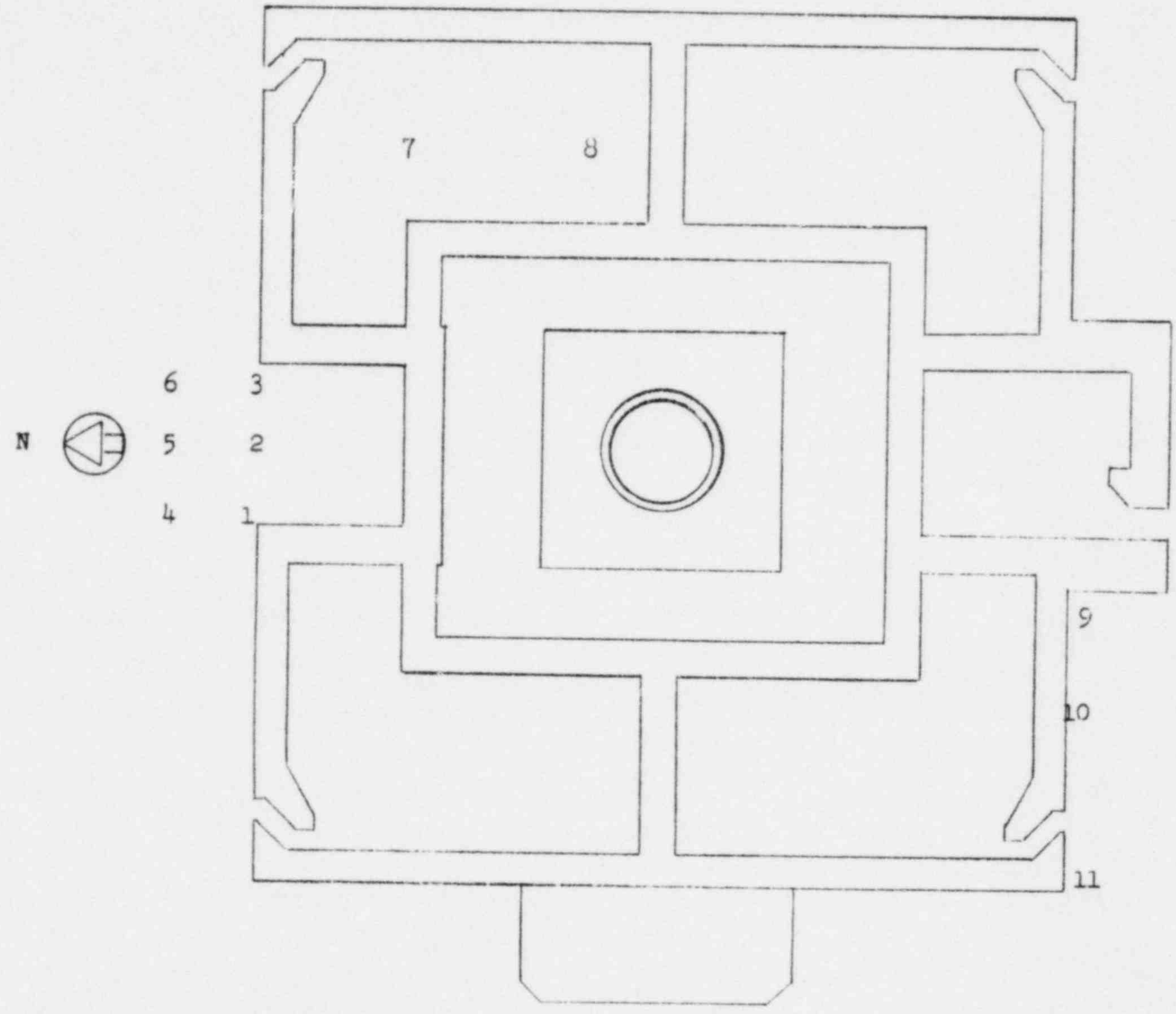
		10	22.5	35	42	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
2.	El. 565, north of reactor	0.3 3.5 n	0.6 6.5 n	1 9.5 n	0 0.1 n	0 0.25 n	0 0.58 n	0 0.8 n
4.	El. 565, east sec. shield wall					0 0.1 n	0 0.1 n	0 0.1 n
6.	El. 565, east sec. shield wall	0.2 1 n	0.5 1.8 n			0 0.1 n	0 0.1 n	0 0.1 n
7.	El. 565, south canal wall	0.2 1 n	0.5 3.1 n		0 0.1 n	0 0.1 n	0 0.1 n	0 0.1 n
9.	El. 565, elevator entrance	0.1 0.5 n		0.4 4 n		0 0.1 n	0 0.2 n	0 0.2 n
12.	El. 565, west sec. shield wall	0.2 1 n		0.5 4 n		0.2 0.1 n	0.2 0.1 n	0.3 0.1 n
13.	El. 565, over sec.stm.gen. D	0.2 1 n		0.4	0 0.1 n	0.2 0.1 n	0 0.2 n	0 0.1 n
15.	El. 565, west sec. shield wall	0.2 1 n				0.2 0.1 n	0.2 0.1 n	0.2 0.1 n
16.	El. 560'-6", bottom of stairs below wire guns	0.2 2 n				0.2 0.2 n	0.2 0.4 n	0.2 0.4 n
17.	El. 565, wire scanning platform	0.4 4 n				0.2 0.2 n	0.3 0.2 n	0.3 0.3 n
18.	El. 560'-6", center of leak detector room	0.4 2 n	0.5 5.6 n	4 n		0 0.1 n	0 0. 1 n	0 0.2 n
19.	El. 565, in instrument Rm. H				0 0.1 n			0 0.1 n

3. Special Radiation Survey Sheet No. 3

Reactor Power (MWT)

Point	10	22.5	35	42	309 Dual Cycle
1. El. 549, SW corner of walkway	0 0.2 n		0.4 1.4 n	0 0.1 n	0 0.1 n
3. El. 549, west walkway	0 0.2 n		0.4 0.5 n		0 0.1 n
5. El. 549, NW corner of walkway			0 0 n		0 0.1 n
8. El. 536'-6", Platform west			0.4 1.3 n		0 0.1 n
11. El. 548, Platform south	0 0.3 n		0.4 2 n		0. 0.1 n
13. El. 549, Inside SSG Room B on elevated platform	(These measurements reported on Special Radiation Survey Sheet No. 4)				
15. El. 548, Inside Nuclear Steam Inst. Room C			0 0 n		0 0.1 n
17. El. 549, North walkway	0.2 1 n		0.4 1.8 n		0 0. 15 n

See note at bottom of data, Survey Sheet No. 2.



RADIATION MEASUREMENTS AT ELEVATION 529 - 517 Ft.

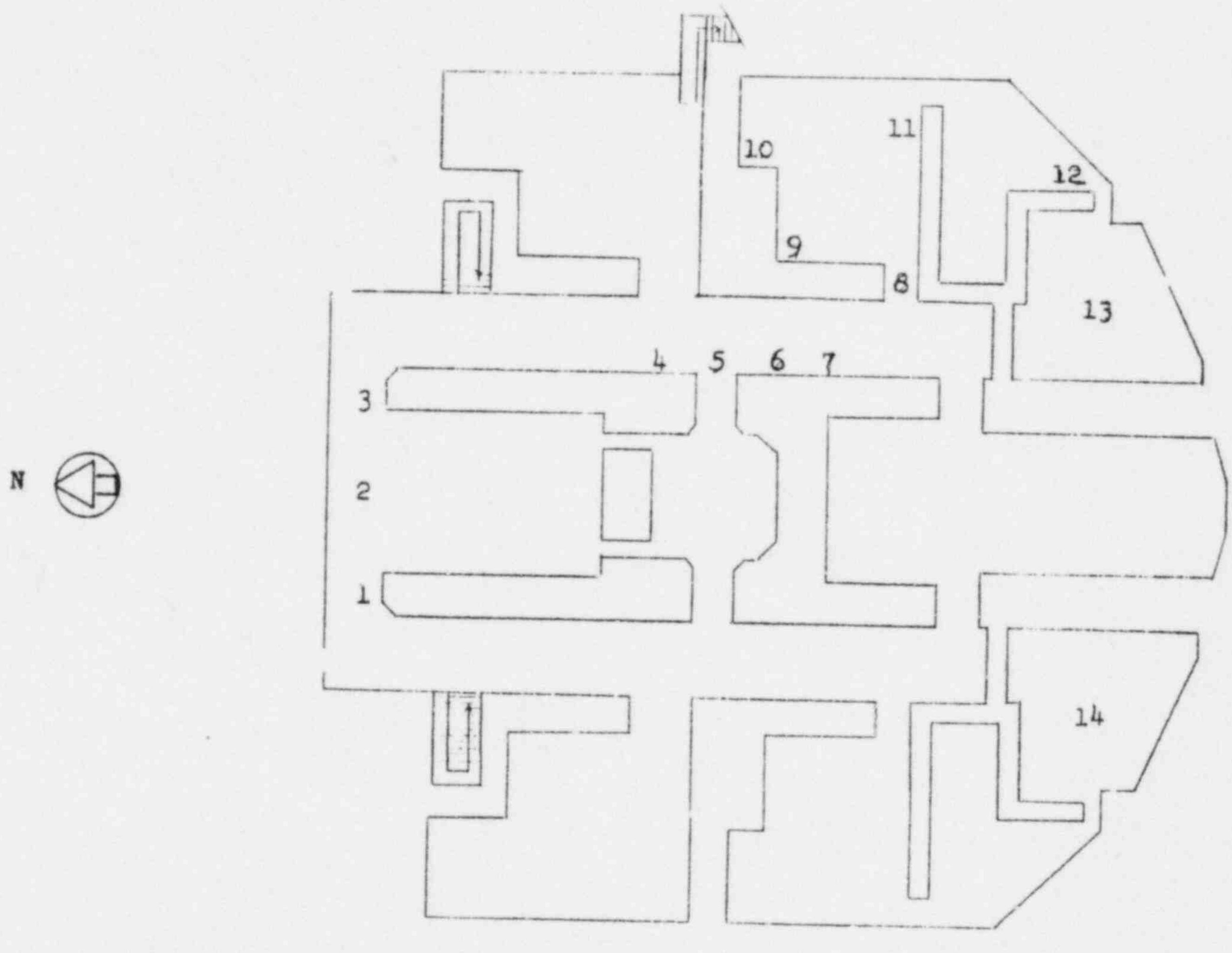
(For greater detail, see Drawings 142F825 and 142F826 in Appendix)

4. Special Radiation Survey Sheet No. 4

Reactor Power (MWT)

Point	10	22.5	35	42	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
2. El. 529'6", North, below crane hatch	0 0.25 n		0.4 1.3 n	0 0.1 n	0 0.22 n	0 0.2 n	0 0.3 n
7. El. 533, Downcomer SSG Room A	22.5				900 15 n	1000 15 n	1800 27 n
8. El. 533, SSG A	7				110	120 11 n	250 20 n
10. El. 529'-6" Outside south sec. shield wall	0				0 0.1 n		0 0.2 n
El. 535, Ext. Ion Ch. Guide Tube #1				0 30 n	0 200 n	0 1.6 n	0 2.5 n
El. 517, Between Primary and Secondary Shields					300 1600 n		300 240 n
El. 517, NE, in corridor outside door to pipeway vestibule					1 18 n		1 1 n

See note at bottom of data, Survey Sheet No. 2



RADIATION MEAS'JREMENTS AT ELEVATION 502 FT.

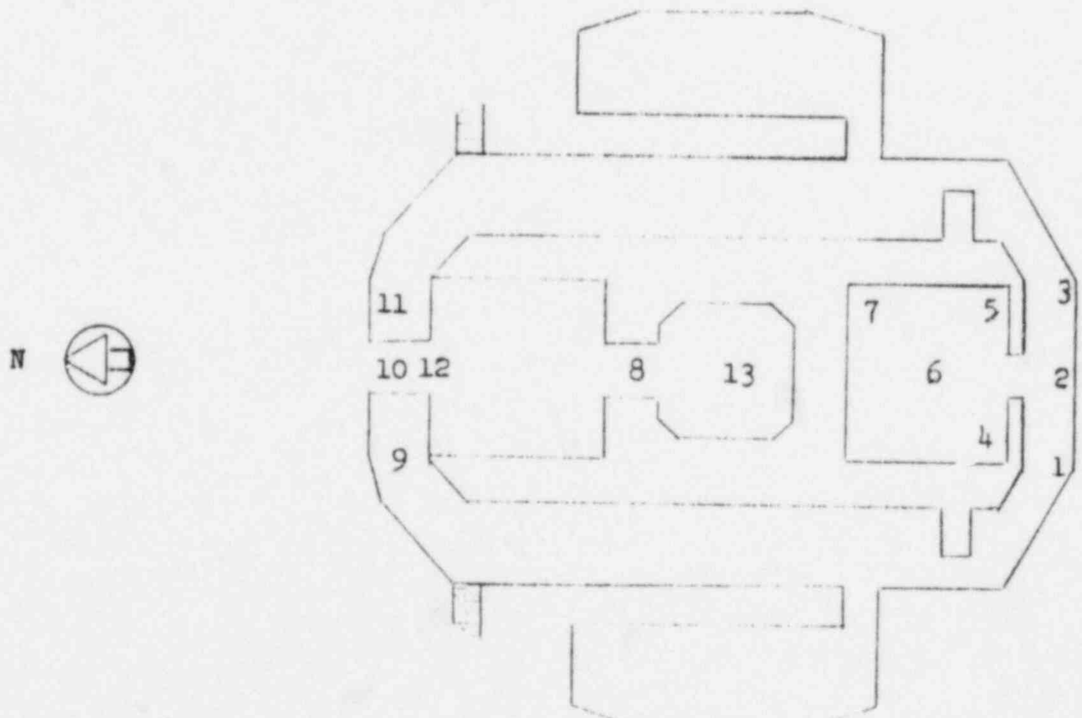
(For greater detail, see Drawing 142F827 in Appendix)

5. Special Radiation Survey Sheet No. 5

Point	Reactor Power (MWT) 10	22.5	35	42	309 Dual Cycle	315 Single Cycle	626 Dual Cycle
1. El. 502, corridor N of scram valve room	0		0 1 n		0.5 14 n	1.5 5 n	0.5 1 n
2. El. 502, corridor N of scram valve room	0		0 2 n	0 7.5 n	0.5 21 n	1.5 5 n	1 3 n
3. El. 502, corridor E of sub-pile room	0		2.5 n		8.5 9 n	0.3 0.2 n	0.4 0.2 n
5. El. 502, corridor E of sub-pile room	1.5 5 n		17 n	1.5 3.7 n	13.5 14 n	0.3 0.2 n	0.4 0.2 n
7. El. 502, corridor E of sub-pile room	0				0.5 0.1 n		0.2 0.2 n
8. El. 502, Instrument Room A, doorway	0				0 0.1 n		0.4 0.1 n
11. El. 502, Instrument Room A	0				0.1 n		
12. El. 502, Doorway to Cleanup Demin. A.	0.25				0 0.1 n		
13. El. 502, Between Cleanup Demin. A & Reg. HX A	0.25				70		
14. El. 502, Between Cleanup Demin. B & Reg. HX B	0			2 0 n	200 4 n		

Between the 35 and the 42 MWT surveys, a partial thickness of shield blocks was placed in the control drive installation ports in the primary shield at El. 502.

This was completed between the 309 and 315 MWT surveys. A laminated steel, cadmium and wood sliding door and a concrete labyrinth were placed in the passageway between the sub-pile room and the drive accumulator room, E. 488, between the 35 and 626 MWT surveys. Between the 315 and 626 MWT surveys, a wooden neutron shield cribbing was placed around the control drive hydraulic line penetrations of the primary shield at El. 502.



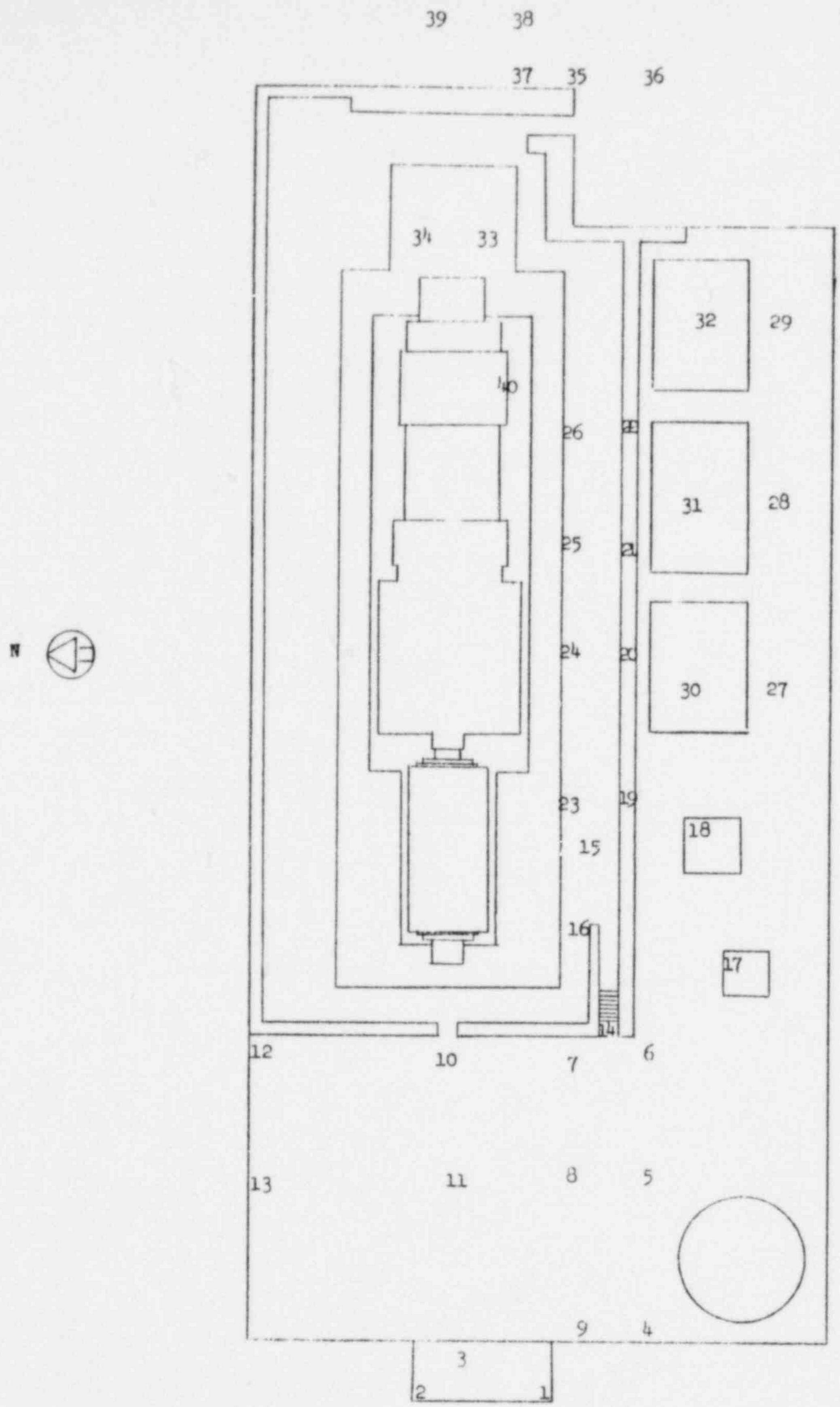
RADIATION MEASUREMENTS AT ELEVATION 488 FT.

(For greater detail, see Drawing 142F827 in Appendix)

6. Special Radiation Survey Sheet No. 6

Point	Reactor Power (MWT) 10	22.5	35	42	309 Dual Cycle	315 Single Cycle	626 Dual Cycle
1. El. 488, Corridor S of Reactor	0			0	0 0 n		0 0.1 n
2. El. 488, Corridor S of Reactor	0 0 n			0	0 0 n		0 0.1 n
6. El. 488, Control Drive Instrument Room	0 0 n			0	0 0 n		0 0 n
7. El. 488, Control Drive Instrument Room	0 0 n				0 0 n		0 0 n
8. El. 487, Doorway Between Accum. Room & Sub-Pile Room	1.5 10 n			7 12 n	100 40 n	90 30 n	4 6 n
9. El. 487, Corridor N of Accumula r Room	0 0 n				0.5 1.5 n	2	0.3 1 n
10. El. 487, Corridor N of Accumulator Room	0 0 n				2.0 2.4 n	2.5 2.2 n	0.5 2.5 n
12. El. 487, In Accumulator Room	0 0.7 n				2.5 6 n	3.5 4 n	0.5 3 n
13. El. 487, Below Reactor in Sub-Pile Room	10 40 n			70			1200 (Est) 5000 n (Est)

See note at bottom of data, Survey Sheet No. 5



RADIATION MEASUREMENTS AT ELEVATION 551 FT., TURBINE BUILDING

(For greater detail, see Drawing 142F700 in Appendix)

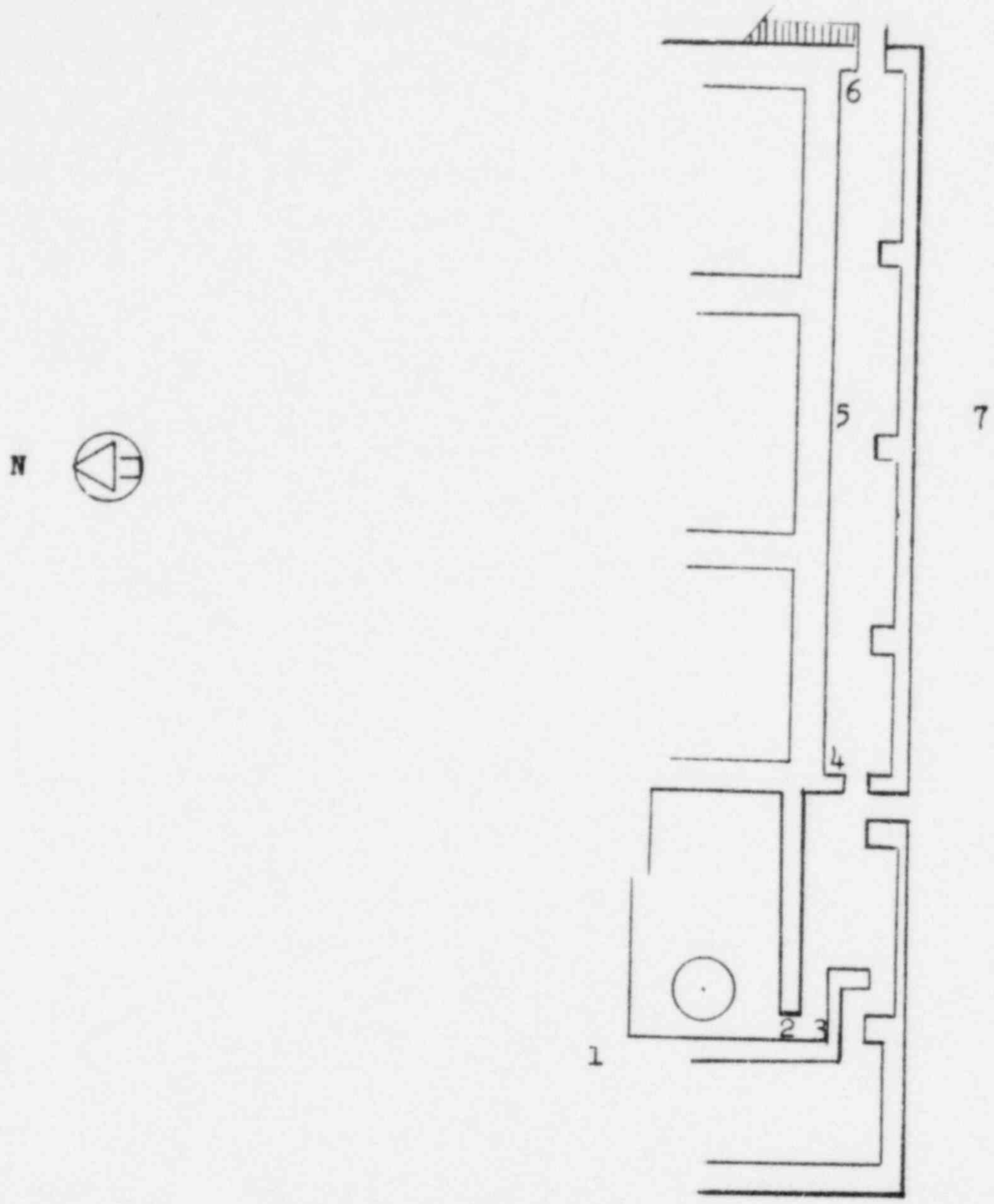
B. Turbine Building Surveys During Operation

1. Special Radiation Survey Sheet No. 8 (Gamma Only)

	Reactor Power (MWT)		
Point	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
1. El. 551, Stairs W. of Turbine	0	0	0.6
2. El. 542'-6" Stairs W. of Turbine	0	0	0.8
3. El. 534, Stairs W. of Turbine	0	0	0.8
4. El. 551, Turbine Floor West	0	0	1.2
5. " " " " "	0	0	2
6. " " " " "	0	0.5	2.8
7. " " " " "	0	0.5	2.6
8. " " " " "	0	0	3.3
9. " " " " "	0	0	1
10. " " " " "	0	0	0.3
11. " " " " "	0	0	1
12. " " " " "	0	0	2.2
13. " " " " "	0	0	1.4
14. El. 551, Stair to Cond. Pump	0	1	1
15. El. 531'-6" " " " "	0	1.6	4
16. El. 521'-6" " " " "	0	1.4	4.5
17. El. 551, Hatch	0	0.8	4
18. El. 551, Hatch above Con. Demin.	0.5	1.8	10
19. El. 551'-6", Curb S of Turbine	0.8	3	11
20. " " " " " " "	4	13	29.5
21. " " " " " " "	5	16	31
22. " " " " " " "	6.5	19	39

Special Radiation Survey Sheet No. 8 Continued

	Reactor Power (MWT)		
	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
23. El. 551, Floor Joint S of Turbine	1	4	10
24. " "	5	17	39
25. " "	7	19.5	39
26. " "	17.5	65	88
27. El. 551, Turbine Floor Over FWH Comp.	1.5	4	22
28. " "	4.5	11	36
29. " "	6	15	40
30. El. 551, Turbine Floor at Deck-Plate above FWH Comp.	3.5	7	26
31. " "	5.5	15	40
32. " "	11.5	13.5	45
33. El. 551, Turbine Front Std. Over Floor Grating	4	11	19
34. " "	4	11	19
35. El. 551'-3" Stairs E. of Turbine	0	0	0
40. Contact with H.P. Steam Line entering turbine steam chest	35	190	300

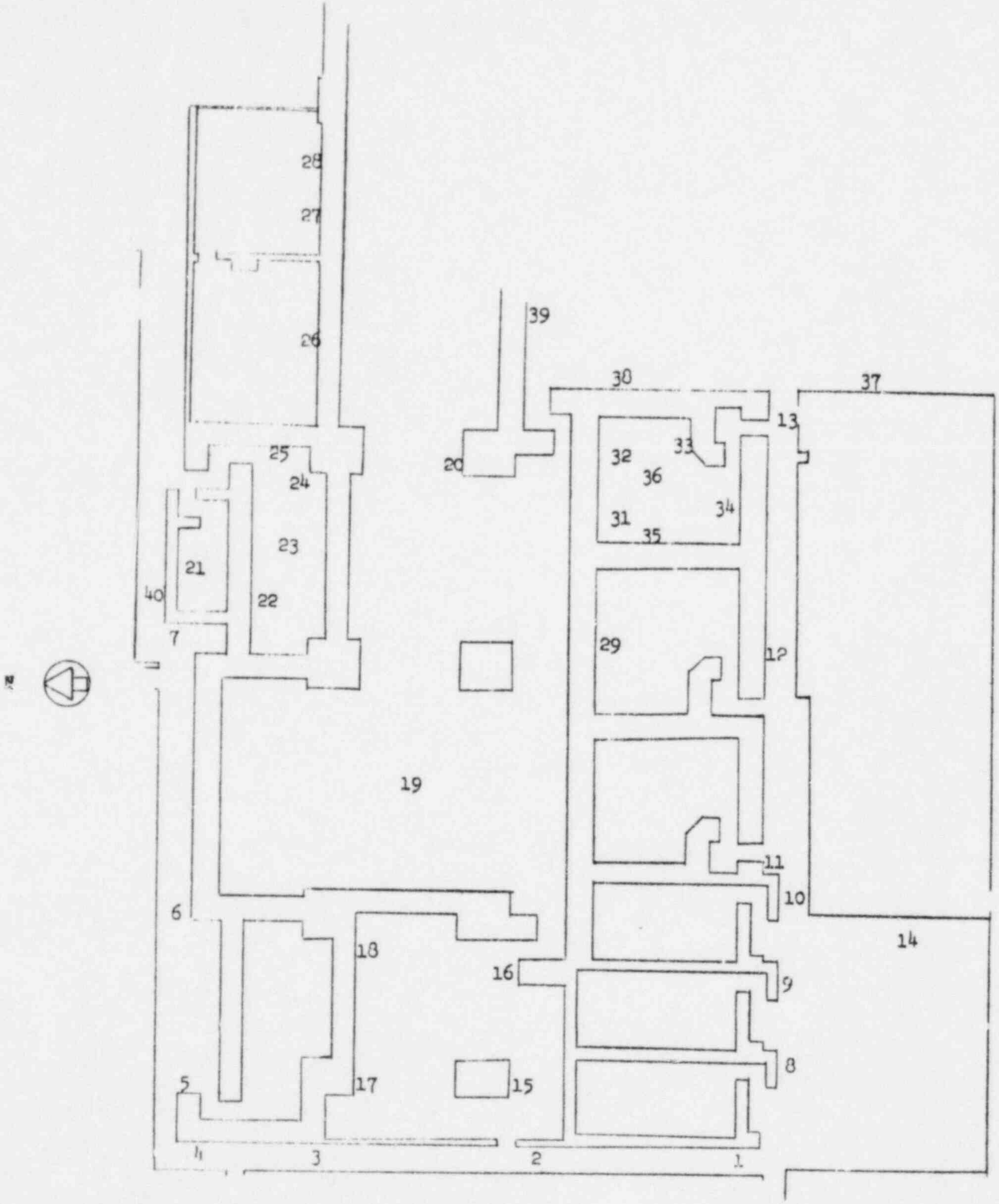


RADIATION MEASUREMENTS AT ELEVATION 531 FT., TURBINE BUILDING

(For greater detail, see Drawing 142F701 in Appendix)

2. Special Radiation Survey Sheet No. 9 (Gamma Only)

	Reactor Power (MWT)		
Point	309 Dual Cycle	315 Dual Cycle	626 Dual Cycle
1. El. 531'-6" Cond. Demin. C.	17	50	75
2. El. 531'-6" Window Penetration to Demin.	2	10	14
3. El. 531'-6" Window Penetration to Demin.	2	6	8
4. El. 531'-6" Valve Gallery	0	0	0
5. " "	0	0	0
6. " "	0	0	0
7. " "	0	0	0



RADIATION MEASUREMENTS AT ELEVATION 495 FT. - 517 FT. IN TURBINE BUILDING

(For greater detail, see Drawing 142F702 in Appendix)

3. Special Radiation Survey Sheet No. 10 (Gamma Only)

Point	Reactor Power (MWT) 309 Dual Cycle	315 Single Cycle	626 Dual Cycle
1. El. 517'-6" W. Corridor	0	0	0
2. " "	0	0	0
4. " "	0	0	0
5. El. 517'-6", N. Corridor	0	0	0
6. " "	0	0	0
7. " " Off-Gas Sampler	0.5	3	7
8. El. 517'-6", S. Corridor	0	0	0
9. " "	0	0	0
10. " "	0	0	0
11. " " by FWH Comp. #1	0	0	0
12. " "	0	0	0
13. " "	0	0	0
14. El. 517'-6" Pri. Feed Pump A	0	0	0
15. El. 514, Cond. Pump Rm.	0	0	0
16. " "	0	0	0
17. " "	0	0	0
18. " "	0	0	0
19. El. 495, Below Hot-well	2	11	12-50 range
20. El. 510'-6" Pri. Steam Line	235	390	800
21. El. 517'-6" Air Ejector Valve Sta.	0	0	0
22. El. 517'-6" N. Side Air Ejector A	245	0 Shut Down	800
23. El. 517'-6" E. Side " "	60	0 "	1000

Special Radiation Survey Sheet No. 10 Continued

	Reactor Power (MWT)		
Point	309 Dual Cycle	315 Single Cycle	626 Dual Cycle
24. El. 517'-6" Air Ejector Comp. A,N.	5	0 Shut Down	500
25. El. 517'-6" " " " ,E.	0.5	0 " "	50
26. El. 517'-6", Lube Oil Room	0	0	0
27. " " " "	0	0	0
28. " " " "	0	0	0
29. El. 512, FWH Comp. #2	12	31	120
31. El. 512, FWH, Comp. #3	28	130	30
32. " "	17	180	150
33. " "	22	70	120
34. " "	12	48	30
35. " "	80	280	300
36. " "	70	180	200
37. El. 517, Trackway	0	0	0
38. " "	0	0	0
39. " "	0	0	0
40. El. 517'-6", Off-Gas Monitor	0	0	0
41. El. 517'-6", Air Ejector Rm.B	1	380	1
42A. El. 512, FWH Comp. #2	15	25	80
42B. " " " "	11	14	80
11A. El. 512, FWH Comp. #1	--	0.5	30

C. Shutdown Radiation Surveys

The data listed below are representative of dose rates prevailing in areas within shielded compartments which will require periodic access for maintenance.

Dose Point, and Time After Shutdown	Dose Rate (mr/hr)
1. Reactor vessel top heat (contact)	
a. 2.5 hr. after shutdown from 315 MWT, core exposure 5,000 MWDT* (5/1/60)	17
b. 24 hr. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/2/60)	1.5
c. 36 hr. after shutdown from 425 MWT, core exposure 37,600 MWDT (11/16/60)	5
(Core history before this shutdown: For 4 hrs. previous to shutdown, 425 MWT; previous 12 hrs. at 626 MWT; previous 12 hrs. at 425 MWT; etc.)	
2. Below control drives, reactor vessel bottom head (15-30 ft from vessel)	
a. 0.3 hr. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	4
b. 1.5 hr. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	3
c. 28 hr. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/2/60)	0.5
c. 52 hr. after shutdown from 425 MWT, core exposure 37,600 MWDT (11/16/60) (Operating history as described in (1c) above.	5

* Megawatt-day (thermal)

3. Reactor bottom head, contact with control drive flanges (11/16/60).	10
(Core history as described in (1c) above)	
4. Cleanup demineralizer room, between demineralizer tank and regenerative heat exchanger	
a. 1.5 hrs. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	130
b. 3.7 hrs. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	120
c. 27.7 hrs. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	35
d. 18 days after shutdown from 315 MWT, core exposure 5,000 MWDT (5/19/60) (Per F. Brutschy & R. Osborne, VAL)	15
e. 14 days after shutdown from 626 MWT, core exposure 12,500 MWDT (9/9/60) (Per R. S. Gilbert, VAL)	60
5. Secondary steam generator, near bottom head	
a. 3 hrs. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/1/60)	15 side 18 bottom
b. 27 hrs. after shutdown from 315 MWT, core exposure 5,000 MWDT (5/2/60)	1.2 side 2 bottom
c. 18 days after shutdown from 315 MWT, core exposure 5,000 MWDT (5/19/60) (Per Brutschy & Osborne, VAL)	0.8 side 7 bottom
d. 14 days after shutdown from 626 MWT, core exposure 12,500 MWDT (8/8/60) (Per R. S. Gilbert, VAL)	2 side 70 avg. bottom
e. 36 hrs. after shutdown from 425 MWT, core exposure 37,600 MWDT (11/16/60) (Core history as described in (1c) above).	230 bottom

D. Environs Surveys

After addition of the top head shielding blocks, measurements at environs stations and survey points around sphere and turbine building during operation show dose rates to be quite low. At full power, the source of greatest influence on environs dose rates is the turbine, including crossover and moisture separator. Measurements taken with precision dosimeters located at various points on the roof of the administration building and access control building give dose rates of 0.3 - 0.5 mr/hr at 626 MWT, full steam flow. These are the highest dose rates at any uncontrolled environs area, and are within the 0.5 mrem/hr limitation of uncontrolled access areas at Dresden. Dose rates in office areas below the roof are even lower.

IV. DISCUSSION OF MEASUREMENTS

A. Reactor Surveys

Radiation surveys were begun at a reactor power of 10 MWt (April 1, 1960). This and subsequent surveys during the next two days at reactor powers up to 40 MWt showed that neutrons streaming above and below the reactor vessel would create untendable dose rates at increased reactor powers. These neutrons were escaping from the reactor vessel radially at the core elevation and were then streaming up and down the annulus between vessel and primary shield.

Above the vessel top head, the radiation scattered from the vertical canal walls, and the walls and top of the craneway, finally escaping from the shield through the large crane openings at the north and the south of the shield (El. 588'). Once outside the shield, these neutrons were able to shower all elevations of the sphere above El. 529', by scattering from the sphere shell itself. In addition, neutron dose rates were readily measurable in the areas around the sphere. Dose rates at a reactor power of about 22.5 MWt were:

El. 565' North (Point 2, Survey Sheet 2)	6 mrem/hr neutrons
(Highest floor designed to be accessible during reactor operation)	0.6 mr/hr gammas
El. 584' Above Reactor Top Head (before installation of shield blocks)	90 mrem/hr neutrons 10 mr/hr gammas

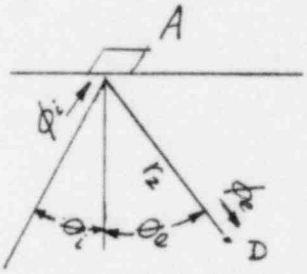
Below the bottom head of the reactor, scattered neutron radiation into elevations 488' and 502' gave dose rates in the order of 50 mrem/hr at 25 MWt in the control drive scram valve and accumulator rooms.

These conditions were corrected by: 1) The addition of a 3 foot thick concrete shield over the top head of the reactor, 2) Plugging holes in the concrete biological shield below the core, through which the control drive hydraulic piping passes into the drive valve room, 3) Adding a one foot thick concrete labyrinth and a steel sliding door between the sub-pile area (below the bottom head) and the control drive accumulator room, 4) Stacking concrete blocks to full primary shield thickness in the pipe passages through the primary biological shield, through which the risers and return recirculation piping passes, and 5) Stacking concrete blocks (2 feet thick) in large holes in primary shield just below the bottom head, through which the control drives were handled during their installation.

A tabulation of neutron and gamma dose rates at the various power levels appears in Section III of this report.

Classically, radiation scattering from a wall follows this empirical formula:

$$\phi_e = \frac{\phi_i A \alpha \cos \theta_i}{r_2^2}$$



where,

ϕ_e = scattered flux measured at detector D

ϕ_i = incident flux on scattering area A

A = scattering area

α = albedo (reflection or scattering coefficient) of wall area A for scatter through included angle $\theta_i + \theta_e$

r_2 = distance from A to receiver point

For strict application, $A << r_2^2$. This condition may be met by breaking the scattering surface into parts. For the case of neutron flux escaping out the north crane opening in the shield, after scatter off the roof and (east and west) side shields, this condition may be met. The total of the effective scattering areas is estimated to be 7,500 ft^2; the average distance from scattering areas to the 565 elevation dose point was 50 feet. At 22 MWt, dose rate above reactor top head in the canal was 80 mrem/hr. At the same power, the dose rate at the 565 elevation north point was 6.5 mrem/hr. The average value of ϕ_i is 30°. Then the albedo may be roughly calculated as,

$$\alpha = \frac{\phi_e \, r_2^2}{\phi_i \, A \cos\theta_i}$$

$$= \frac{6.5 \times 2.5 \times 10^3}{8 \times 10 \times 7.5 \times 10^3 \times 0.866}$$

$$= 3.1 \times 10^{-2}$$

for scattering off the concrete walls through an average included angle of 60°.

As seen in the tables of dose rate measurements, the dose rate above the reactor top head shield blocks (3 foot thickness of concrete) at full power (626 MWt) is 35 mrem/hr from neutrons, and 125 mr/hr from gammas. At the dose point on elevation 565, north, the scattered dose rate at full power is 0.8 mrem/hr. Average energy of the scattered neutrons is 0.6 Mev.

During full power operation, dose rates in all areas within the sphere are less than 1 mrem/hr (neutrons and gammas) except for the points listed below:

1. At the external ion chamber drive area (El. 535, north), the ion chamber guide tubes are a source of neutron streaming. Dose rates at some tubes during half power operation were as high as 75 mrem/hr. This situation was remedied by insertion of cylindrical tube plugs with a length of 12 inches of polyethylene and one inch of steel and lead. Dose rates from the hottest tubes are now reduced to 1-2 mrem/hr.

2. Spare instrument tubes leading down the annulus between reactor vessel and primary shield penetrate the shield at an elevation above the El. 548 floor. Some neutron streaming ($\sim$15 mrem/hr) is detected at the tubes. Since the streaming beam is well above the floor, over the head of any passerby, no plugging is necessary.

3. Neutron dose rates in the order of 5 mrem/hr exist in the drive valve room, locally around the hydraulic lines penetrating the primary biological shield.

4. In the pipeway vestibule area, El. 517, between primary and secondary shields, neutron leakage out the recirculation line penetrations in the primary shield gives rise to neutron dose rates of 60-100 mrem/hr. Since this area contains the recirculation piping and the primary steam lines with their associated high gamma fluxes from Nitrogen-16, no accessibility restriction accrues. However, neutron scattering in this area results in local dose rates of $\sim$ 5 mrem/hr in the El. 517 walkway near the doors into the vestibule area.

B. Turbine Building Surveys

Although reactor core and pressure vessel shielding was comparatively well known, the radiation levels and shielding requirements in the primary cooling circuit were yet to be proven.

The accessibility problems presented by radioactive isotopes in the steam and condensate systems of the plant were found to be of a smaller order of magnitude than contemplated during design. Carryover of Nitrogen-16 in the steam is the only sizeable contributor to dose rates in the steam and condensate systems, with negligible corrosion product carryover. Of 400 curies of Nitrogen-16 produced per second in the reactor core at full power, only 5 curies per second reach the turbine. The turbine is unshielded, and dose rates of 10-60 mr/hr [illegible] experienced by an operator during an inspection of bearings, front standard controls, etc. Dose rate at the generator collector rings is less than 0.5 mr/hr.

Radiation levels around the condenser vary from 3 mr/hr at the water boxes to 20 mr/hr beneath the hotwell, with a local hot spot (200 mr/hr) at the off-gas line take off. As a non-condensible, the majority of the Nitrogen-16 goes with the off-gas to the shielded air ejector room, in which radiation levels are as high as 1 r/hr.

The delay time (approximately 2 minutes) provided by the hotwell allows the residual Nitrogen-16 activity to die off before the consensate reaches feedwater pumps and heaters. The shielded feedwater heater compartments have radiation levels between 1 and 250 mr/hr within the compartments, depending upon the extraction stage of the regeneration steam to individual heaters and upon the condensate water level in the heaters and drain tanks.

Dose rates from primary recirculating lines in the secondary steam generator compartments average 2 r/hr at full power, with steam generator dose rates lightly less. Neutron dose rates of 20 mrem/hr exist in the secondary steam generator compartments. The majority of this is due to Nitrogen-17 decay, with some possible contribution due to scattered neutrons from the reactor vessel.

Correlation with Calculations

Fair correlation was found between the currently used method of calculation and measurements of Nitrogen-16 dose rates. The calculation method may be summarized by the following equations:

$$N = \sum_{E=10.5\,\text{Mev}}^{E=\infty} n_E \frac{\Sigma_{ACT(E)}}{\Sigma_{R(E)}}$$

N = Nitrogen-16 nuclei produced per fission neutron incore

n_E = fission neutrons produced in energy group $E + \Delta E$ per fission neutron in core, based upon the Watt-spectrum.

$\Sigma_{ACT(E)}$ = energy-dependent activation cross-section for the formation of Nitrogen-16 in the O^{16} (n,p) N^{16} reaction.*

*(Based upon BNL-325, Harvey and Hughes, Neutron Cross Sections, 7/1/55, p.85).

$\Sigma_{R(E)}$ = energy-dependent cross-section of core for removal of neutron from its initial energy group $E + \Delta E$ to a lower energy group. This includes hydrogen moderation, uranium capture, inelastic scattering by oxygen, structure, and fuel, and elastic scattering by oxygen.

The calculation of the contributions of the various energy groups is thus based upon a modified age calculation of the fission neutrons born above the reactor threshold of about 10.5 Mev. The result of this summation of the contributions is:

$N = 3.5 \times 10^{-6}$ Nitrogen-16 nuclei produced per fission neutron

Total production of Nitrogen-16 when the core is operating at the full thermal rating of 626 megawatts is thus,

$$N_p = 3.5 \times 10^{-6} \frac{\text{N-16 nuclei}}{\text{fiss.neut.}} \times 2.5 \frac{\text{fiss.neut.}}{\text{fission}} \times 3.1 \times 10^{10} \frac{\text{fiss.}}{\text{watt - sec}} \times 6.26 \times 10^{8} \text{ watts}$$

$N_p = 1.7 \times 10^{14}$ Nitrogen-16 nuclei produced in core per second.

This may be converted to curies by

$$C_p = 1.7 \times 10^{14} \frac{\text{N-16 nuclei}}{\text{sec.}} \times 0.0943 \frac{\text{disint.}}{\text{sec-nucleus}} \times \frac{1}{3.7 \times 10^{10}} \frac{\text{curie-sec}}{\text{disint.}}$$

C_p = 433 curies per second of Nitrogen-16 released from core.

As may be seen from above, this method of calculating Nitrogen-16 production does not depend upon a calculation of the neutron fluxes of the various energies in the core. Rather, it is assumed that all neutrons born above 10.5 Mev are degraded below the threshold before leaving the core.

The splitting of this activity between water and steam may be predicted by two methods. One is based upon observations at EBWR, and the other is based upon a chemical analysis of the reversible reactions involving ammonical and nitrate radical forms. Both agree that, at the average Dresden core exit steam quality of 5.5%, the percentage of Nitrogen-16 escaping the core in a steam-compatible form is about 5.5%. On this basis, 23.5 curies per second start from the core in the steam.

With a calculated 17-second delay time from core to turbine throttle, the entering rate of Nitrogen-16 flow through the turbine is 4.7 curies per second. On this basis, the calculated dose rate one meter from one 18-inch diameter steam pipe just upstream of the turbine is 410 mr/hr. The measured dose rate at this point was 600 mr/hr.

Recent measurements of turbine dose rates (11/60) have shown that a general reduction of about 35% has occurred after air in-leakage to the primary feedwater system has been reduced. This may be explained, not in terms of reduced production of N-16, but as a reduction of carryover of N-16 in the steam from the core and drum. With substantial air in-leakage, the primary water is nearly saturated with atmospheric nitrogen (N-14). This inhibits N-16 solution in water upon its formation and increases its carryover with steam. Minimizing air saturation of the primary water has resulted in reducing this carryover, and dose rates are closer to predictions.

It is interesting to note, as substantiating evidence, that Nitrogen-13 release from the stack dropped by a factor of four upon correction of the air in-leakage. Nitrogen-13 is produced by the N^{14} (n, 2n) N^{13} reaction, with the target Nitrogen-14 nuclei supplied by air leakage into the system. The fourfold reduction in Nitrogen-13 is much greater than the Nitrogen-16 reduction, since the target source of Nitrogen-13 was reduced. No less Nitrogen-16 is now being produced; the equilibrium has just been forced more in favor of the water phase.

C. Radiation Measurements at Reactor Vessel

Gamma heating and neutron flux were measured outside the reactor vessel wall near the core midplane, during full power operation measured values show each to be lower than the predicted values. This is principally due to the 452-channel core loading which existed during the measurements; the predictions were made on the basis of a full 488 channel core.

1. Neutron Fluxes in Vessel Wall

The neutron flux and spectrum measurements were made with the use of a series of threshold-reaction activation samples. The sample assembly (Capsule Carrier Assembly) is shown on Drawing 932C449 in the Appendix. The assembly was mounted in place of the "Dark Channel" external ion chamber in ion chamber guide tube #12, and positioned as close to the reactor vessel as possible. In position, the carrier was separated from the reactor vessel (at core elevation) by 1/2 inch of steel (inner shield caisson), the vessel insulation, and approximately 6 inches of air. At that position, two assemblies were irradiated for one hour. The results of specific activation determinations by B. F. Rider at Vallecitos Atomic Laboratory were:

0.5 Mev neutron flux (by Cl35 (n, α) P32)	2.2×10^8 n/cm^2 -sec.
2.9 Mev neutron flux (by S32 (n, p) P32)	3.6×10^7 n/cm^2 -sec.

5.0 Mev neutron flux (by Ni^{58} (n, p) Co^{58})	2.2×10^7 n/cm^2-sec
8.1 Mev neutron flux (by Al 27 (n,α) Na^{24})	4.2×10^6 n/cm^2-sec
0.1 Mev neutron flux (extrapolated from above)	8×10^8 n/cm^2-sec

Further extrapolation back to the outside radius of the pressure vessel gives

$$\phi_{n>0.1\,Mev} = 8 \times 10^8 e^{(.17 \times 1.27)} \times \left(\frac{215}{200}\right)$$

$$= 1.1 \times 10^9 \quad \text{n/cm}^2\text{-sec}$$

It was calculated that the neutron flux on the outside radius of the pressure vessel would be between 3×10^9/ and 7×10^9 n/cm^2-sec ($>$ 0.1 Mev). This was based on full core loading of 488 fuel bundles, or channels. The lower value determined experimentally may be partly reconciled with the theoretical value by examining the decrease in core radius in the 452 channel core. The absence of channels in the outer periphery increased the water reflector thickness in the direction of the samples by 3.5 inches. This increase in water path would theoretically decrease the fast ($>$0.1 Mev) flux by a factor of

$$e^{.12 \times 3.5 \times 2.54} = 2.9$$

This factor, applied to the range of theoretical prediction above, would decrease theoretical prediction of 3×10^9 - 7×10^9 n/cm^2-sec to a range of 1×10^9 - $2.4 \times$ [illegible] n/cm^2-sec. The effect of water annulus thickness upon vessel neutron exposure is evident.

The neutron flux at the inside radius of the vessel can be extrapolated from the measured value at the shield caisson. As noted above, extrapolation to the outside radius of the vessel gave a neutron flux of 1.1×10^9 n/cm^2-sec ($>$ 0.1 Mev). Similarly, further extrapolation to the inside radius of the vessel gives a fast ($>$ 0.1 Mev) neutron flux of

$$1.1 \times 10^9 \times e^{(.17 \times 15)} \times \frac{200}{185} = 1.5 \times 10^{10}$$ n/cm^2-sec with the 452 channel

core. This would give a 40-year neutron exposure at the vessel inside radius of 1.6×10^{19} n/cm^2 with a plant load factor of 0.85. If the core were loaded to its periphery, the 40-year exposure at the inside radius of the vessel would be 4.6×10^{19} n/cm^2. Present APED design criteria permits a maximum neutron exposure of 5×10^{19} n/cm^2 ($>$ 0.1 Mev) in vessel steel, based upon consideration of embrittlement and rise of transition temperature in the steel.

Gamma Heating in Vessel Wall

With the use of a special calorimeter, gamma heating was measured at the same point (inside channel #12 guide tube) as the neutron fluxes described above. The calorimeter used was composed of a slug of stainless steel of known mass, suspended and insulated within an evacuated thin-walled case. Both slug and case were fitted with thermocouples. Knowing the heat capacity of the slug, the heat transfer rate between slug and case, and the rate of rise of slug temperature in a gamma energy flux, the gamma heating could be evaluated. Thus, where $\left(\frac{dT}{dt}\right)_{SLUG}$ is the time rate of temperature rise in the slug.

Then

$$Q = \frac{C_p m \left(\frac{dT}{dt}\right)_{SLUG} + H\,(T_s - T_c)}{V}$$

Q = gamma heating in slug BTU/hr-ft^3.

C_p specific heat of slug, BTU/16-°F.

m = mass of slug, lb

H = total heat transfer coefficient between slug and case.

$H = H\ (T_s, T_c)$

T_s = slug temperature, °F

T_c = case temperature, °F

V = volume of slug, ft^3.

$H\ (T_s, T_c)$ was determined experimentally before the gamma heating measurement. At full reactor power, this data of slug and case temperature vs. time was recorded:

July 16, 1960 Time (Hrs)	Ts (mv)	Ts (°F)	Tc (mv)	Tc (°F)	Ts - Tc
1222	1.86	95.5	1.86	95.5	0
1325	1.99	99	1.90	97	2
1343	2.03	100.	1.92	97.5	3
1430	2.07	101.5	1.94	98	3.5

At case temperatures of 95 - 98°F,

$$H(T_s, T_c) = 3.6 \times 10^{-2}\ BTU/_{hr\text{-}°F}$$

$$C_p m = 8.5 \times 10^{-3}\ BTU/_{°F}$$

$$V = 1.42 \times 10^{-4}\ ft^3$$

Evaluating, Q varies between 775 and 825 BTU/hr-ft^3. The average gamma heating, calculated by averaging the data, is

$$Q = 800 \text{ BTU/hr-ft}^3$$

in steel, at a point just outside the 1/2-inch thick steel inner shield caisson.

To extrapolate the gamma heating to the outside and to the inside radii of the reactor pressure vessel, use may be made of the theoretically predicted slope of the gamma heating in the pressure vessel steel. The theoretically predicted gamma heating in the vessel steel* was:

$$Q = 2.6 \times 10^4 e^{-6.1 t} \quad \text{BTU/hr-ft}^3$$

t = thickness of steel, in feet, from the inside radius of pressure vessel.

Thus, at the inside radius t = 0 and gamma heating was predicted to be

$$Q = 26{,}000 \text{ BTU/hr-ft}^3$$

at the outside radius of the pressure vessel, following the prediction. The gamma heating was calculated to be

$$Q = 2.6 \times 10^4 e^{-6.1 (0.50)}$$
$$= 1235 \text{ BTU/hr-ft}^3.$$

Using the same exponential slope (6.1 ft^{-1}) to extrapolate the average measured value beyond the caisson, to the pressure vessel outside radius,

* "Gamma Heating in Dresden Pressure Vessel", by R. A. Mickle & W. D. Craig, March 9, 1960

$$Q = 800\ e^{6.1(\text{caisson thickness})} \times \frac{215}{200}$$

caisson thickness = 0.5 inch

= 0.0416 ft.

Then the extrapolated heating value at the vessel outside radius becomes

$$Q = 800\ e^{6.1 \times 0.0416} \times \frac{215}{200}$$

$$= 1110 \text{ BTU/hr-ft}^3.$$

This compares to the predicted value previously noted of 1235 BTU/hr-ft^3. The gamma heating at the inside radius, corresponding to 1110 BTU/hr-ft^3 at the outside radius, is 23,400 BTU/hr-ft^3 for the 452 channel core. The corresponding maximum tangential thermal stress is about 2000 psi (tensile) at the inside radius of the vessel, assuming perfect vessel insulation. For less than perfect insulation, maximum thermal stress should be lower.

Again, the theoretical value was based upon a 488-channel core loading and the measurements were made with a 452-channel core. Using the increase in water reflector thickness calculated and used in the neutron flux analysis above, 3.6 inches, the gamma heating at any point outside the water reflector may be theoretically compared for the full 488-channel core and the 452-channel core. The effective one-exponent fit of water attenuation of the gamma heating spectrum in Dresden reflector is

$$\mu_{eff} = 0.03 \text{ cm}^{-1}$$

For a water path increase of 3.5 inches, the ensuing attenuation is

$$e^{0.03 \times 3.5 \times 2.54} = 0.77.$$

Then, for a full 488-channel core, the gamma heating in the vessel steel will be $1/0.77$, or 1.3 times the gamma heating in the 452-channel core.

On the basis of the measured value of heating outside the caisson with the 452-channel core, the value of gamma heating at the inside radius of the pressure vessel with a 488-channel loading will be 30,400 BTU/hr-ft^3. This corresponds approximately to a maximum tangential thermal stress of 2600 psi (tensile) at the inside radius of the vessel.

The measured values of gamma heating cannot be taken as particularly precise data. The heat generation rate and the temperature differentials are lower than those for which the instrument was designed. The gamma heating measured represents an upper limit.

It should be noted that a series of thermocouples were located on the outside diameter of the pressure vessel during installation. It was originally planned to check gamma heating and thermal stress by noting the temperature differential between core inlet temperature (approximately corresponding to water temperature on the inside radius of the vessel) and the temperature indicated by a thermocouple on the outside of the vessel at core mid-plane height. This method did not give the desired results, due to:

1) The less-than-theoretical efficiency of the vessel insulation, which caused a lowering of the temperature differential across the vessel wall.

2) The small temperature differential ($\sim$10°F), which would exist across vessel wall at the measured rate of gamma heating, even with perfect vessel insulation.

3) The inability of the thermocouples to measure within an accuracy of 1-2% (or 5-10°F at 550°F).

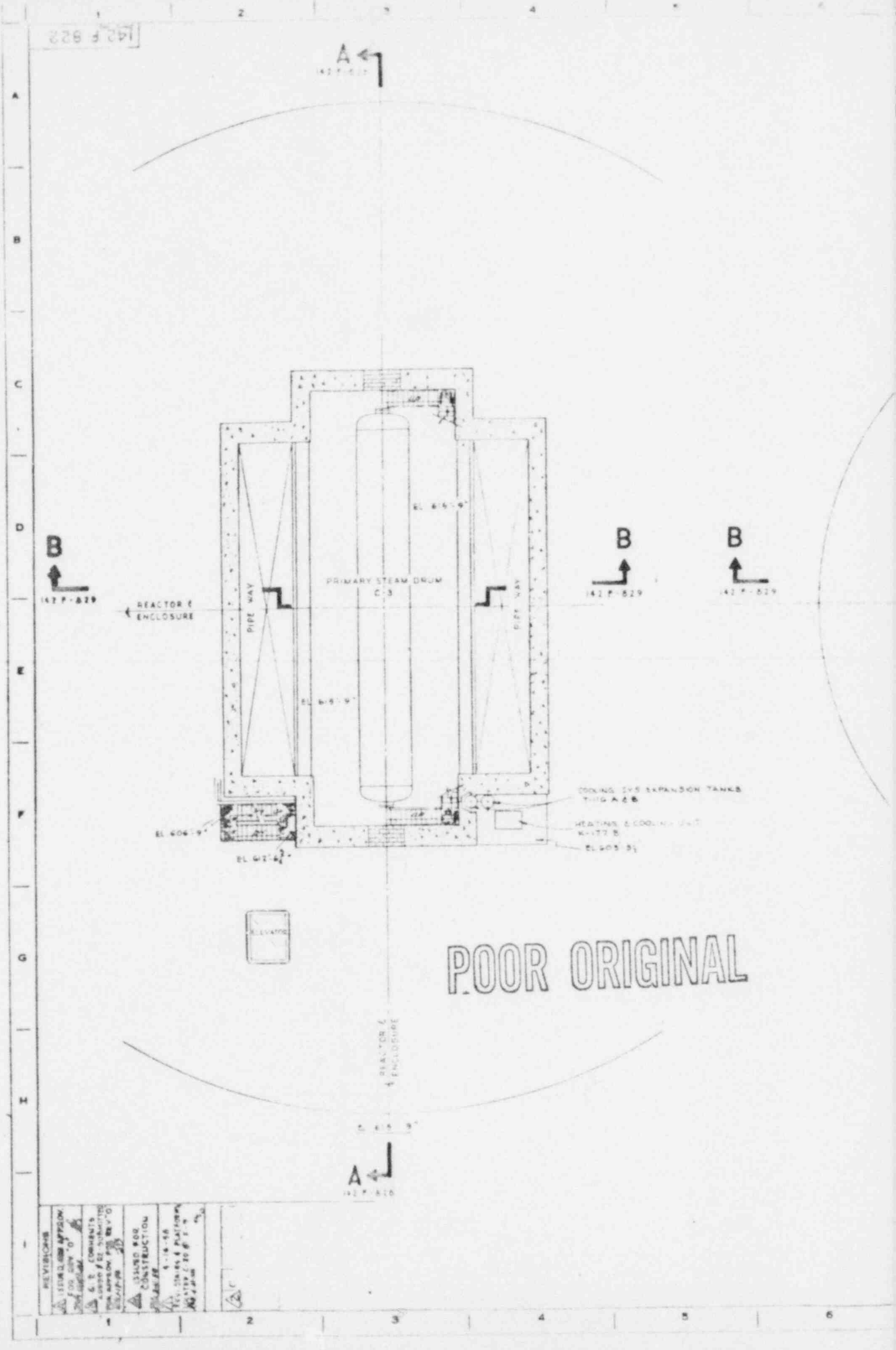
D. Shutdown Radiation Measurements

The program of shutdown radiation measurements has just begun. Several years of operation with attendant shutdowns and surveys must be accumulated before a real pattern of activity buildup will be evident. At this time, after about two months of effective full power operation, only minor quantitites of cobalt-60 (5.3 year half-life) are present. Iron-59 (45 day half-life), cobalt-58 (72 day half-life), and zirconium-95 (65 day half life) have yet to exhibit the full magnitude which they will reach. The majority of plant shutdown radiation, outside the core, at the present time is caused by manganese-56 (2.5 hour half-life) and copper-64 (12.8 hour half-life). Tables of shutdown dose rates around steam drum, steam generators, and piping are given in Section III-B of this report.

Direct radiation from the core is negligible after shutdown, in such areas as reactor top head. Dose rate above reactor top head during such operations as stud-heating for head removal, is controlled by scattered radiation from the core and structure around the core streaming up the annulus between vessel and primary shield. The dose rate from this source is about 50 mr/hr, 24 hours after shutdown. This is attenuated by filling the refueling canal with about two feet of water Another source is top head activation. At the present time, top head dose rates, at contact, are 5 mr/hr 36 hours after shutdown from full power operation. Within 12 hours after shutdown, the dose rate below the control drives in the sub-pile room is less than 5 mr/hr.

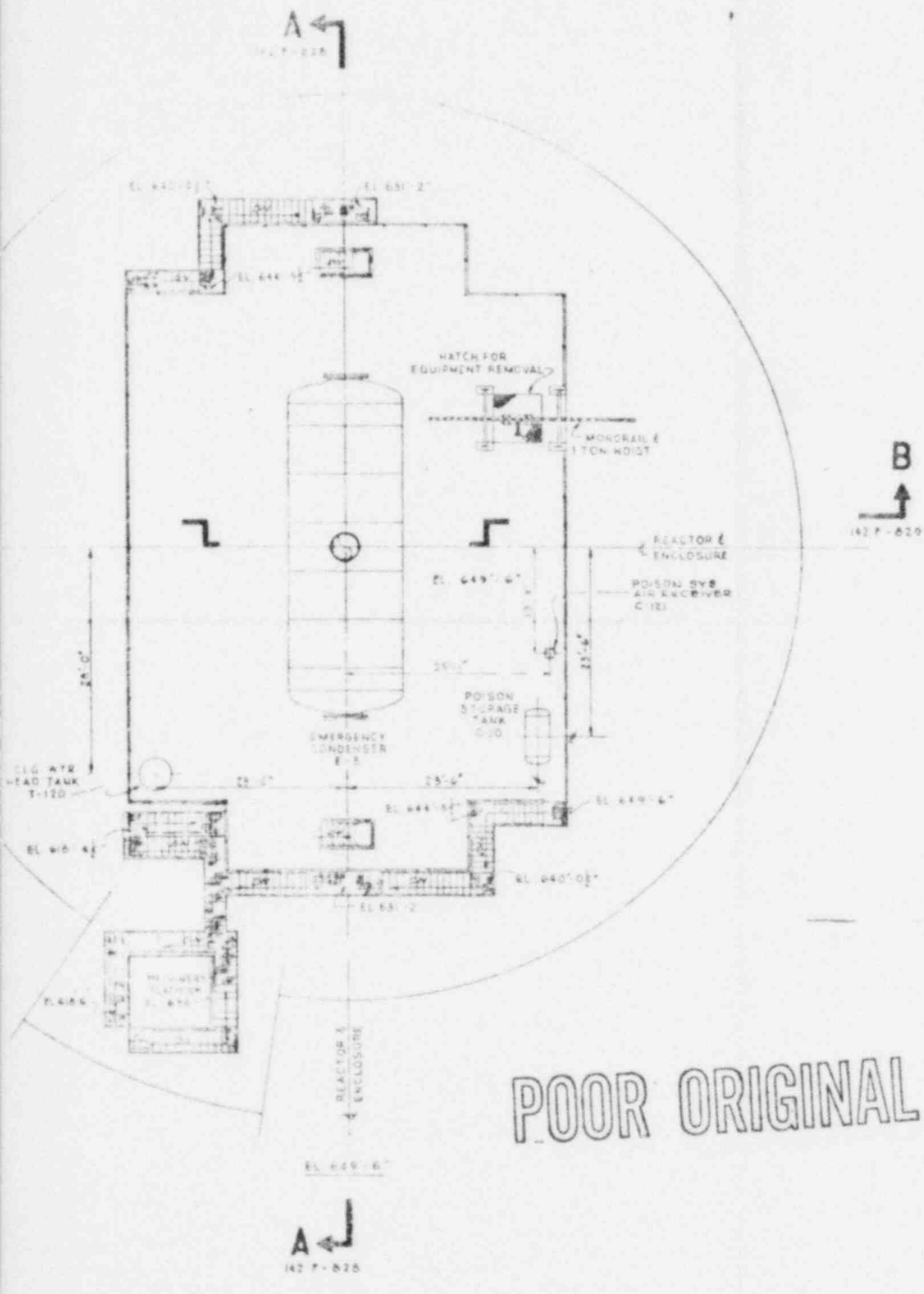
At shutdown, dose rates in secondary steam generator compartments are of the order of 100-200 mr/hr near recirculation piping and secondary steam generators. This decays to about 25 mr/hr in 24 hours except near local hot spots. Substantial buildup with reactor operating time may be expected, particularly around valves, flanges, pumps, etc.

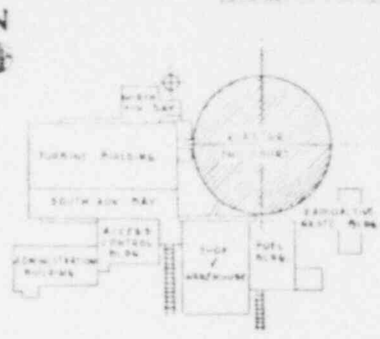
The two cleanup demineralizers and their associated regenerative heat exchangers represent the most concentrated radiation sources (outside reactor vessel) after reactor shutdown. Dose rates of 100-200 mr/hr exist in the cleanup demineralizer rooms one week after shutdown. In the first 48 hours after shutdown, dose rates around cleanup demineralizers and heat exchangers are several hundred mr/hr. To date, however, the demineralizer recirculating piping and pump exhibit small contamination; and the radiation levels in the room containing the demineralizer recirculating pump are 10 mr/hr or less, one week after shutdown. In the turbine building, the condensate demineralizers contain the only significant source at shutdown. All other equipment is capable of direct access within a few seconds of reactor shutdown.



142 F 828
A
B
142 F 829
REACTOR ℄ ENCLOSURE
PIPE WAY
PRIMARY STEAM DRUM
C-3
COOLING SYS EXPANSION TANKS
HEATING & COOLING UNIT
ELEVATOR
POOR ORIGINAL
REVISIONS
ISSUED FOR CONSTRUCTION

142 F 822





KEY PLAN

142F822

POOR ORIGINAL

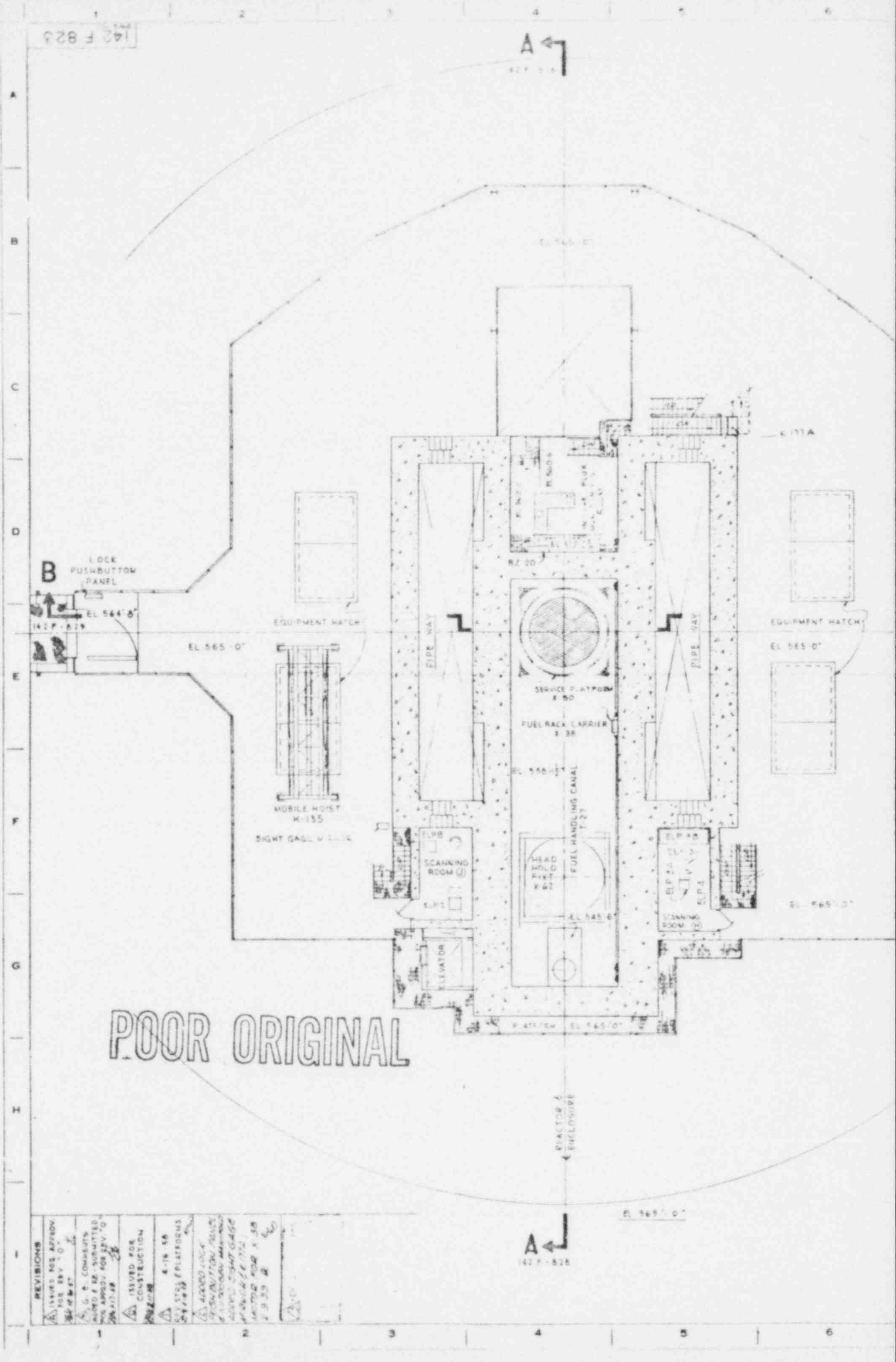
APPROVED
W. E. WEBSTER
Rev 0

0 5 10 15 20
SCALE IN FEET

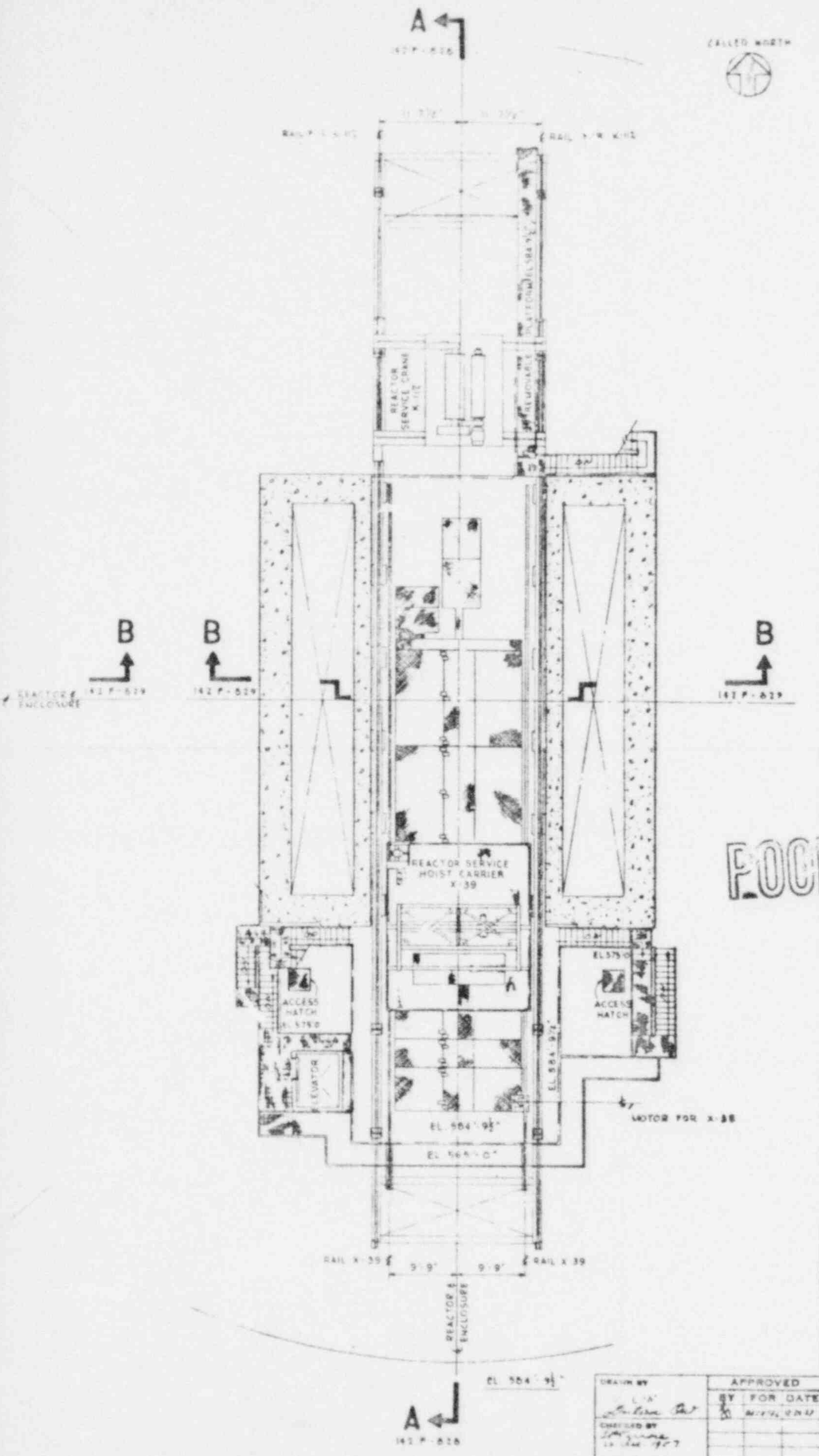
APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	JOB NO. 2348
1		AREA

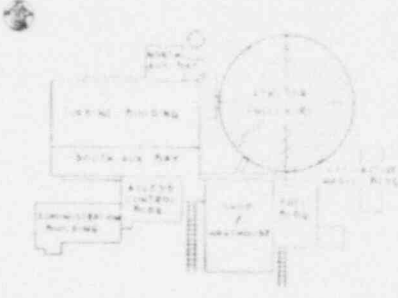
DRAWN BY	APPROVED		
	BY	FOR	DATE
CHECKED BY			
ENGINEER			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE REACTOR ENCLOSURE - EQUIP'T. LOC.
PLAN @ EL 615'-9" & 649'-6"
MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
SCALE 1/8" = 1'-0" DWG NO. 142 F 822 2 PTS.



POOR ORIGINAL





POOR ORIGINAL

142 F 823

APPROVED

SCALE IN FEET

APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	JOB NO 2348
1		AREA

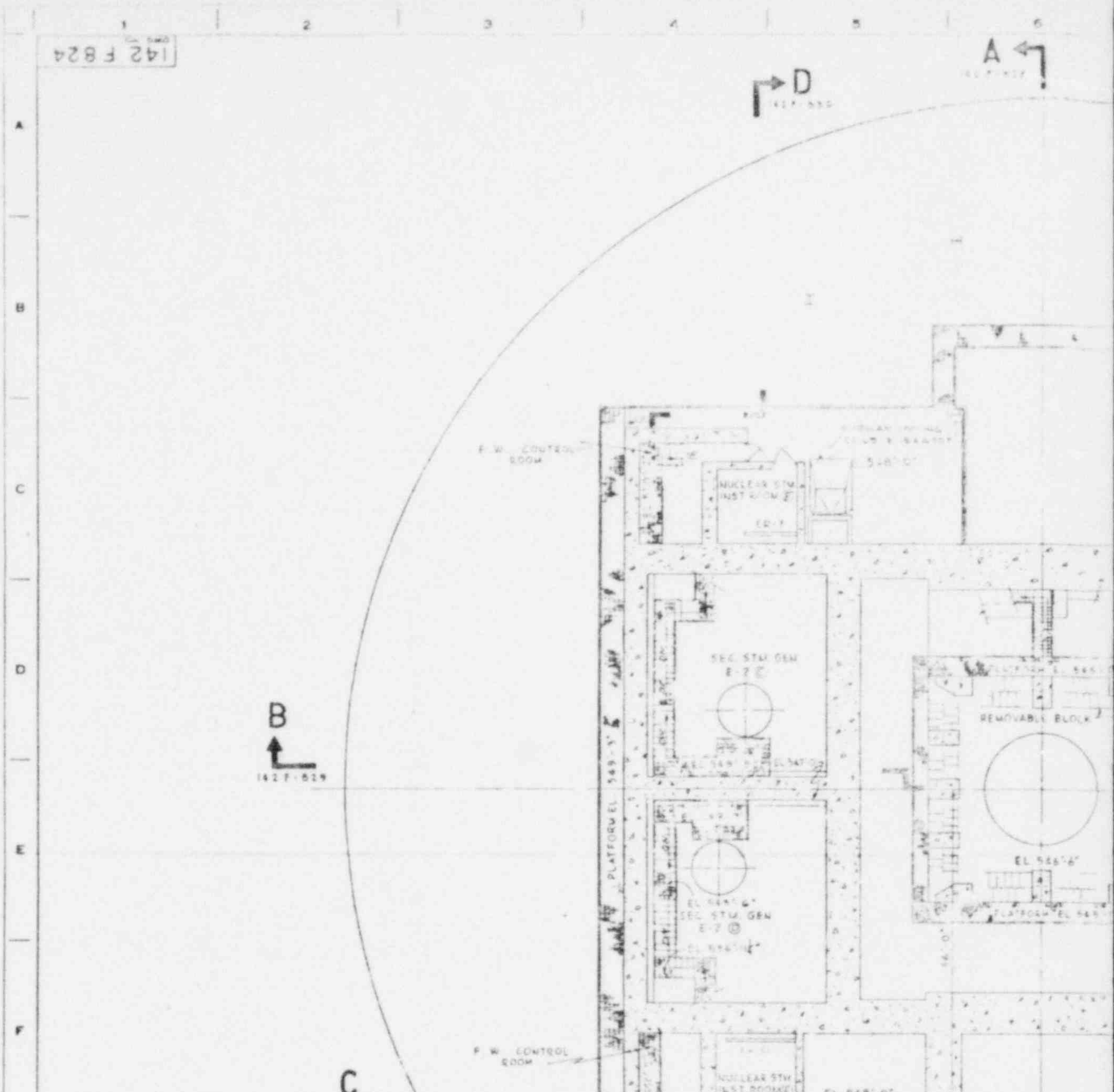
DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.

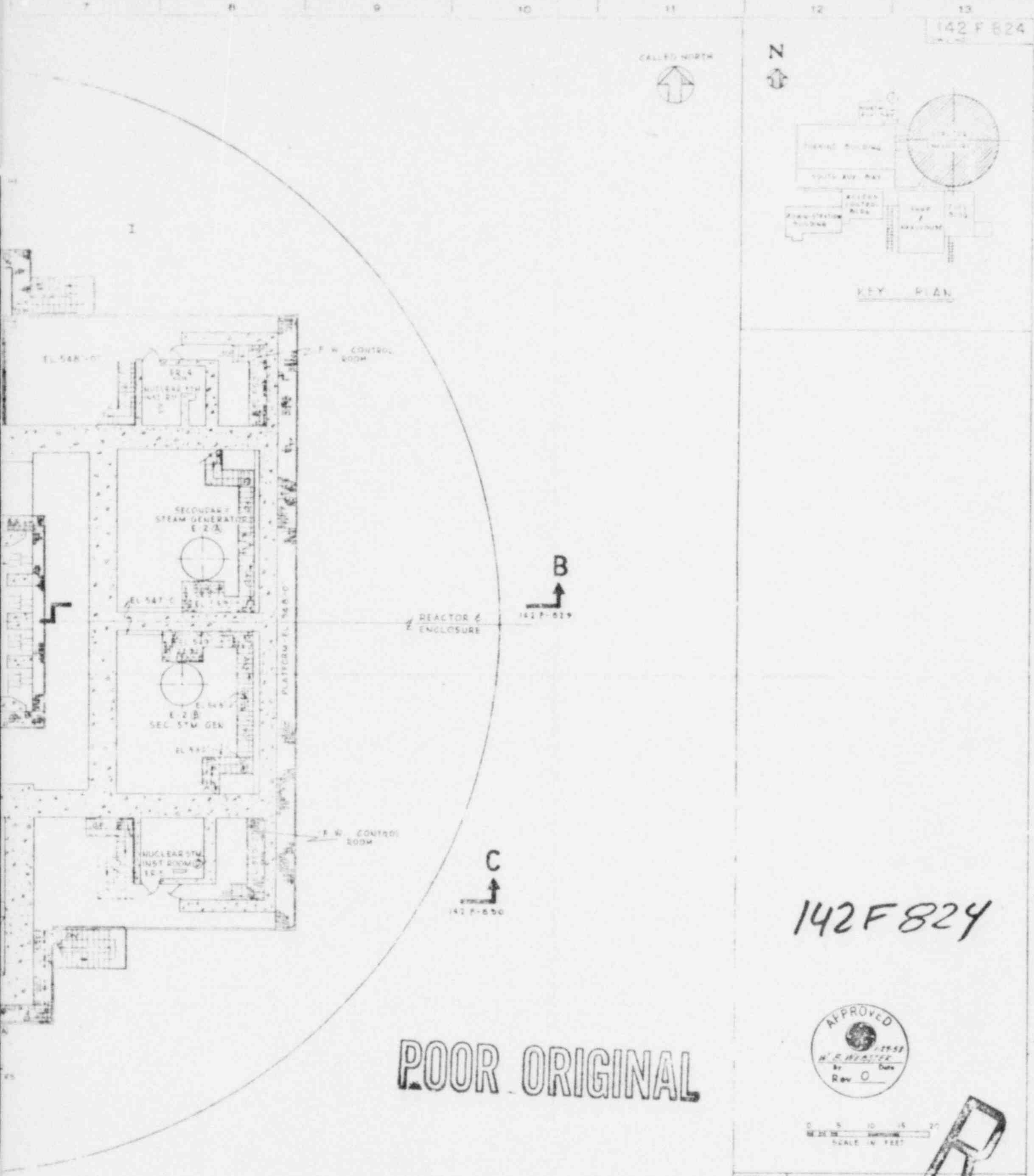
TITLE REACTOR ENCLOSURE - EQUIP'T LOC. PLANS @ EL. 565'-0" & 584'-9 1/2

MADE FOR DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY

SCALE 1/8" = 1'-0" DWG. NO. 142 F 823 REV. 3

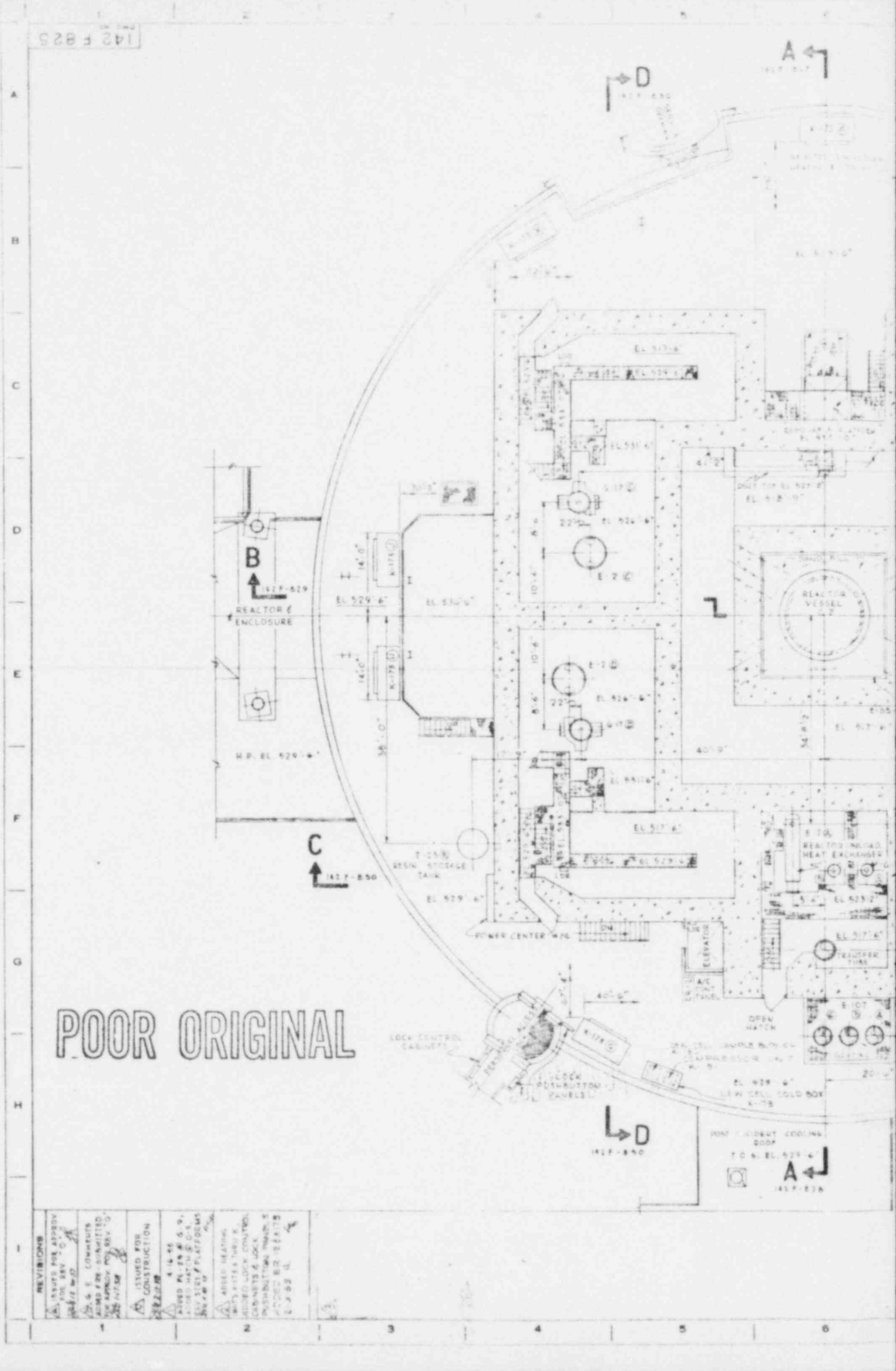


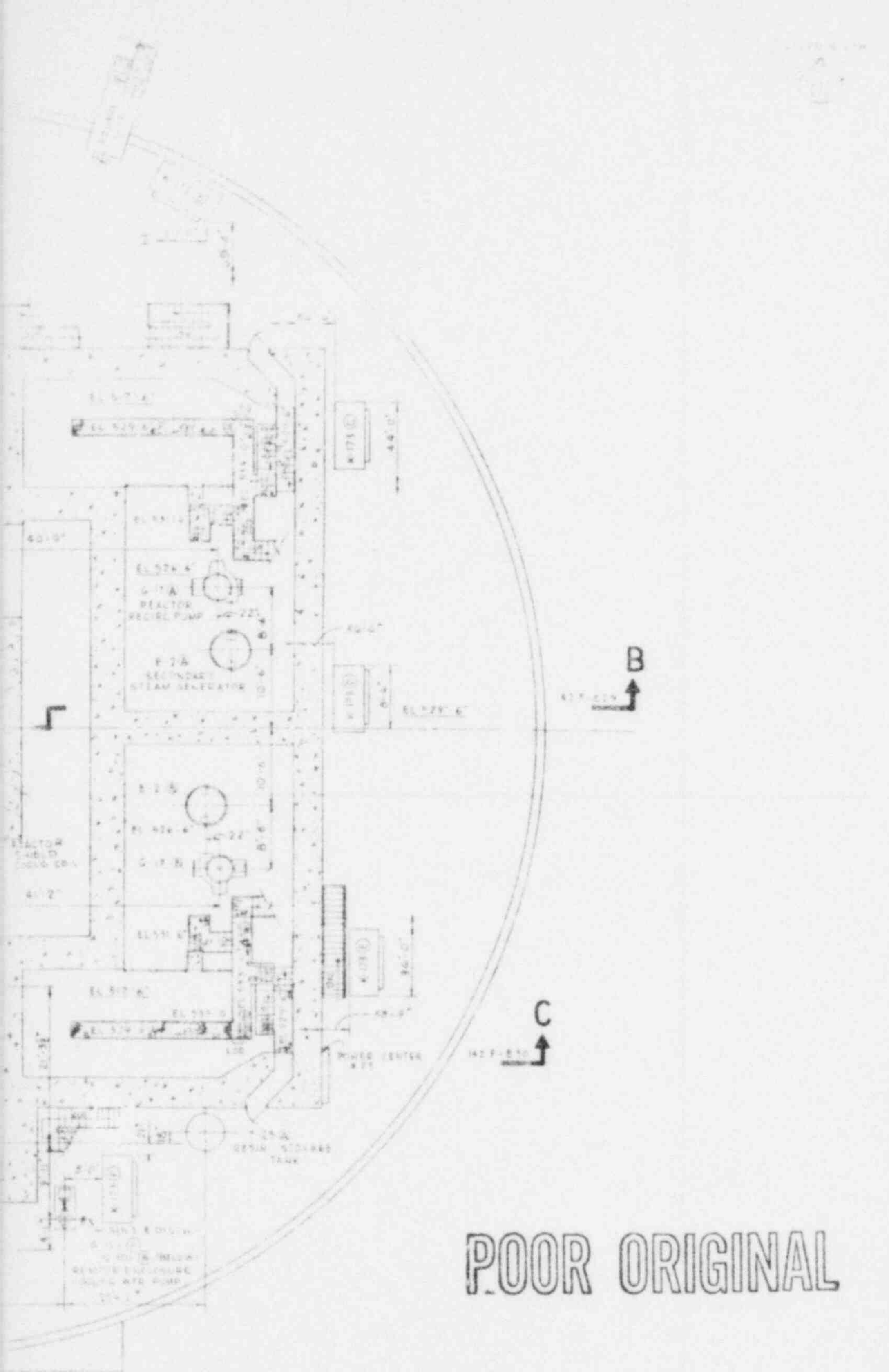
POOR ORIGINAL



PREPARED BY	BECHTEL CORPORATION, SAN FRANCISCO
JOB NO.	2348
	GENERAL ELECTRIC — ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE	REACTOR ENCLOSURE - EQUIP'T LOC. PLAN @ EL. 548'-0"
MADE FOR	DRESDEN NUCLEAR POWER STATION, COMMONWEALTH EDISON COMPANY
SCALE	1/8" = 1'-0"
DWG. NO.	142 F 824
REV.	2

DRAWN BY / CHECKED BY / ENGINEER — APPROVED: BY / FOR / DATE


142 F 825
REACTOR & ENCLOSURE
REACTOR VESSEL
REACTOR UNLOAD HEAT EXCHANGER
RESIN STORAGE TANK
POWER CENTER
ELEVATOR
OPEN HATCH
LOCK CONTROL CABINETS
LOCK PUSHBUTTON PANELS
TRANSFER TUBE
POST ACCIDENT COOLING ROOF
EL. 529'-6"
EL. 517'-6"
REVISIONS
ISSUED FOR CONSTRUCTION
POOR ORIGINAL



142 F 825

POOR ORIGINAL

APPROVED
Rev. 0

SCALE IN FEET

PREPARED BY
BECHTEL CORPORATION
SAN FRANCISCO

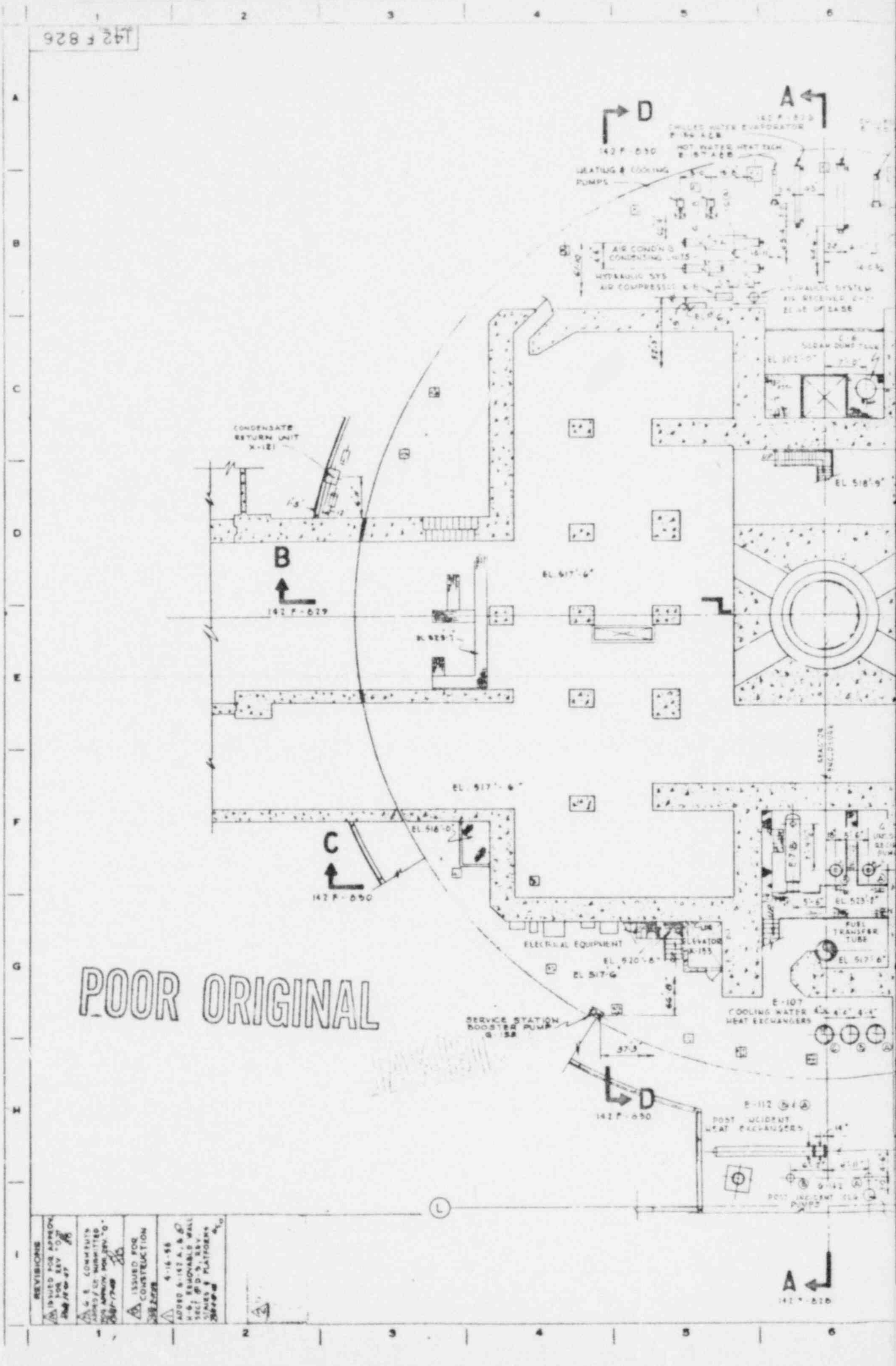
APPROVED
BY | FOR | DATE

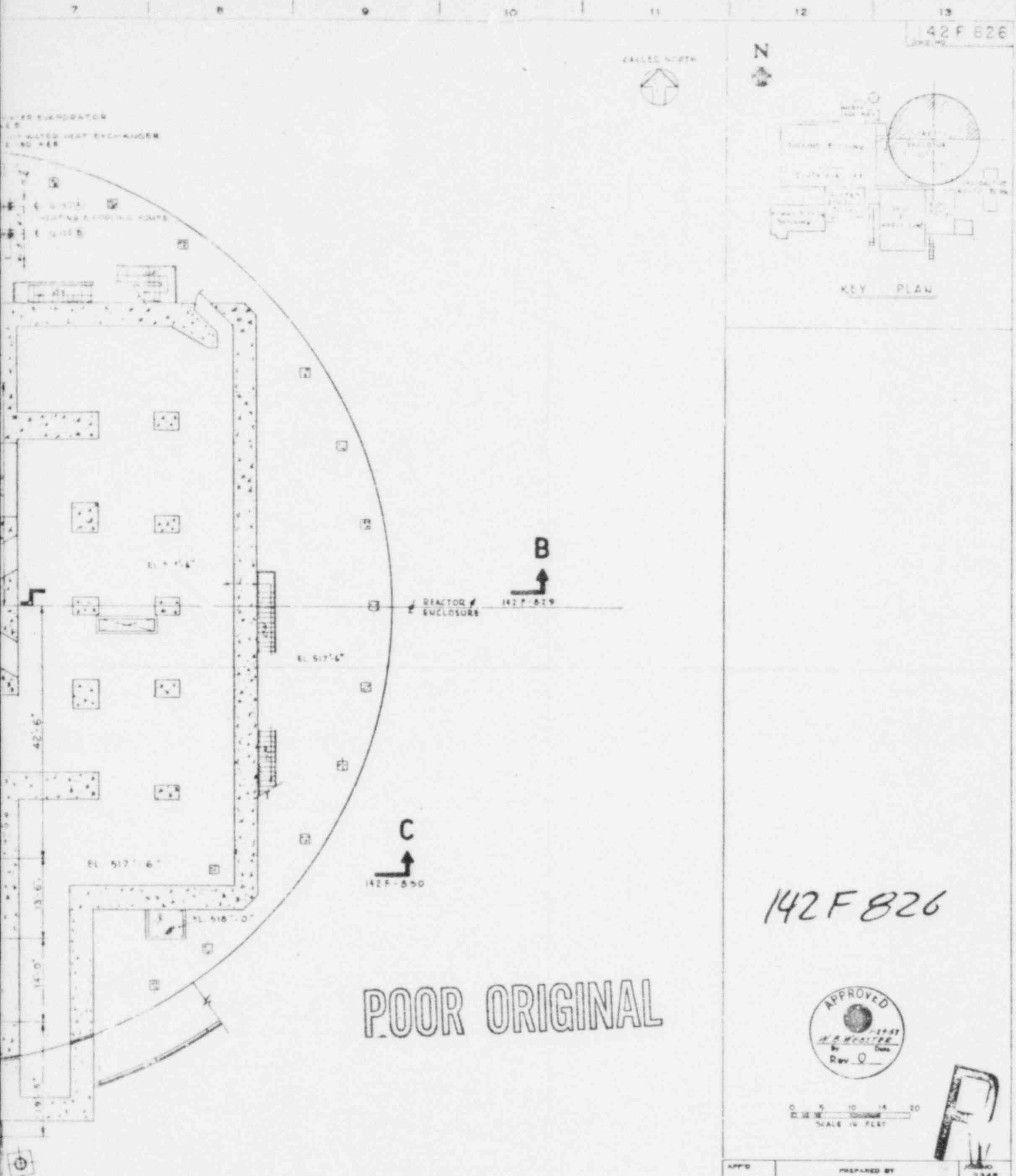
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.

REACTOR ENCLOSURE - EQUIP'T. LOC.
PLAN @ EL - 529'- 6"

DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY

DWG. NO. 142 F 825 | 3

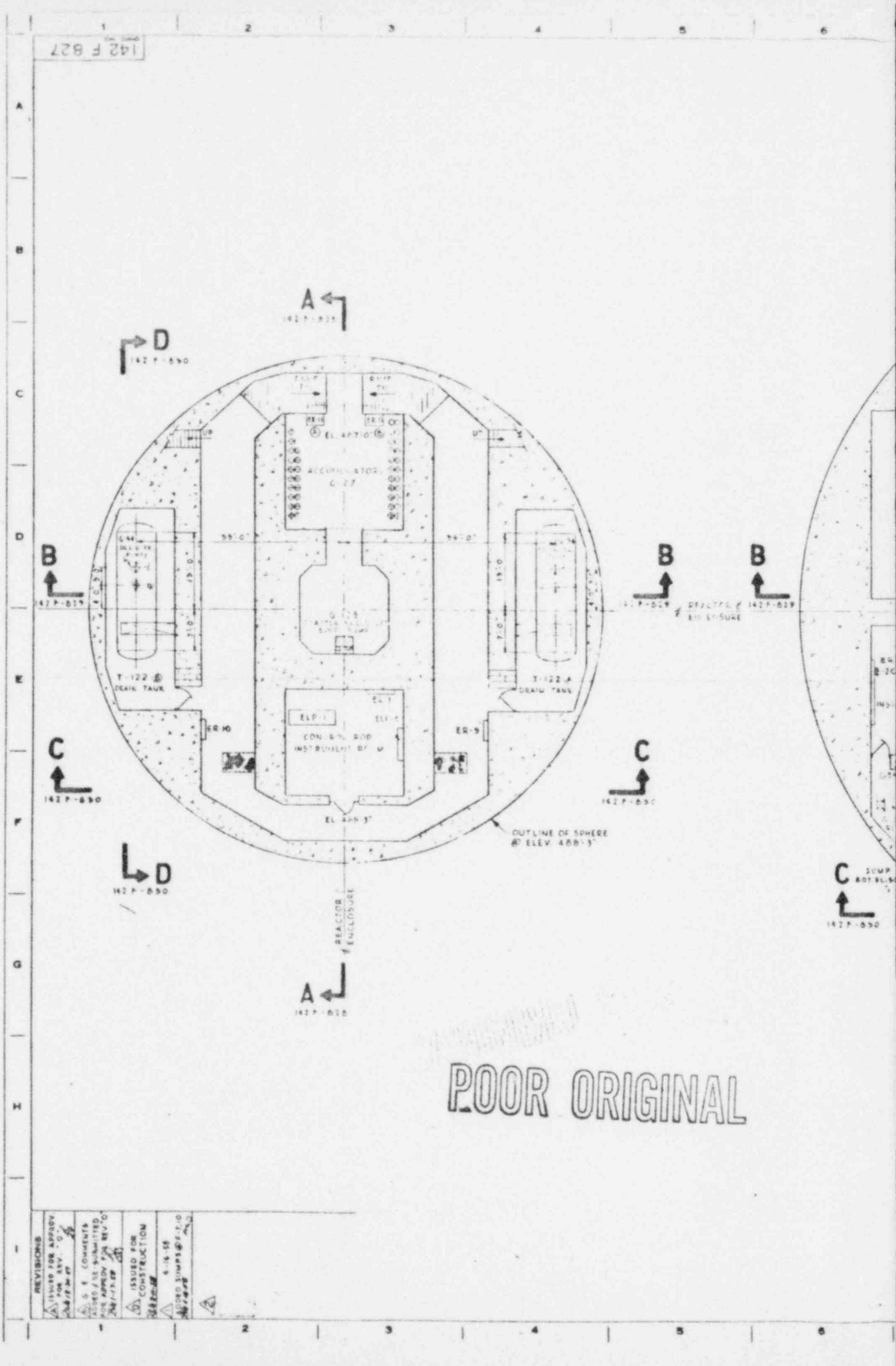

142 F 826
CONDENSATE RETURN UNIT K-121
HEATING & COOLING PUMPS
AIR CONDENSING CONDENSING UNITS
HYDRAULIC SYS AIR COMPRESSOR
HYDRAULIC SYSTEM
SCRAM DUMP TANK
EL. 517'-6"
EL. 518'-9"
EL. 518'-0"
EL. 520'-8"
EL. 517'-6
EL. 517'-6"
ELECTRICAL EQUIPMENT
ELEVATOR
SERVICE STATION BOOSTER PUMP
FUEL TRANSFER TUBE
E-107
COOLING WATER HEAT EXCHANGERS
POST ACCIDENT HEAT EXCHANGERS
142 F-829
142 F-830
142 F-828
REVISIONS
ISSUED FOR CONSTRUCTION
POOR ORIGINAL



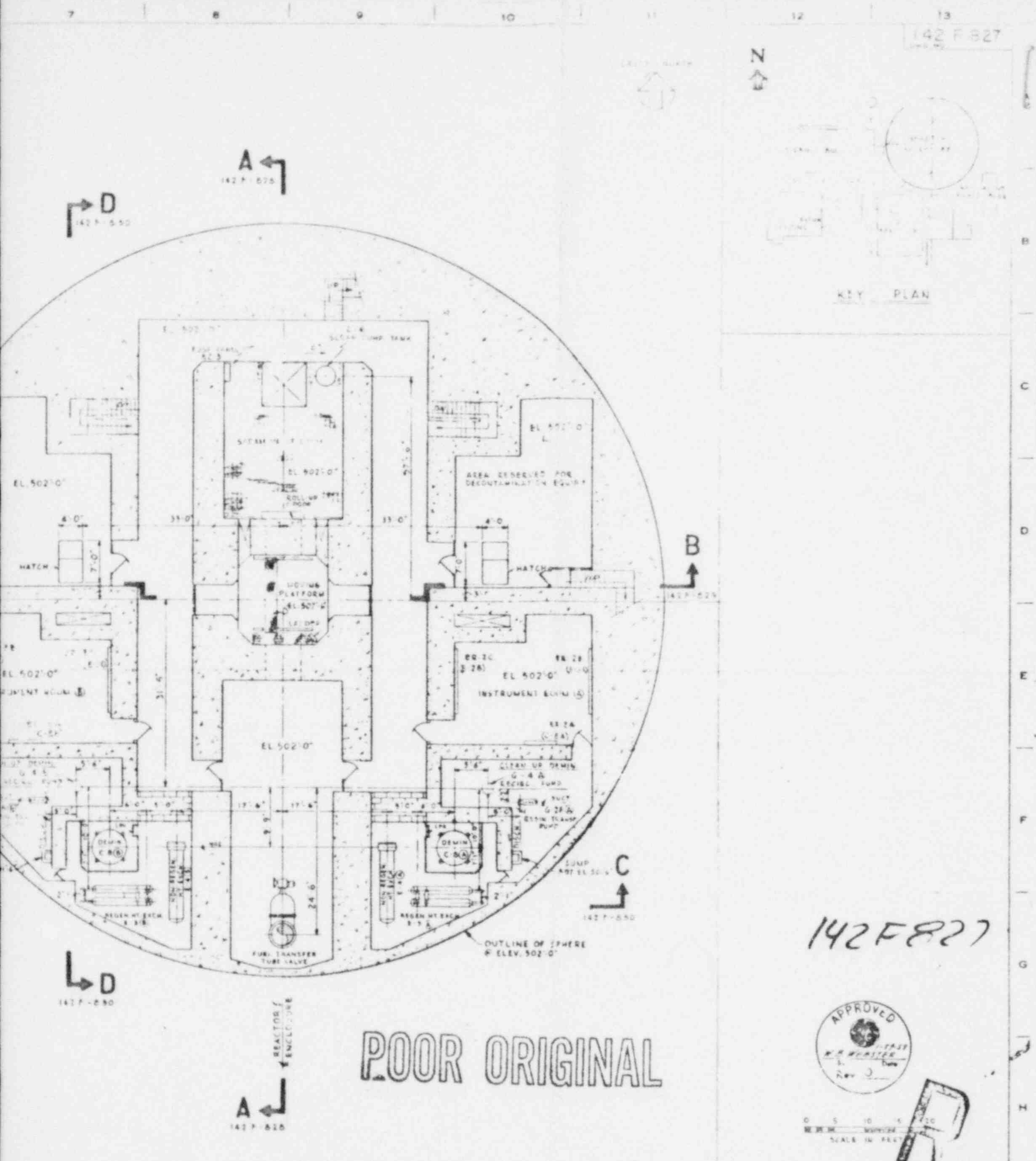
APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	2345
1		AREA

DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			
ENGINEER			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE REACTOR ENCLOSURE - EQUIP'T LOC. PLAN @ EL. 517'-6"
MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
SCALE 1/8" = 1'-0" DWG NO 142 F 826 REV 2

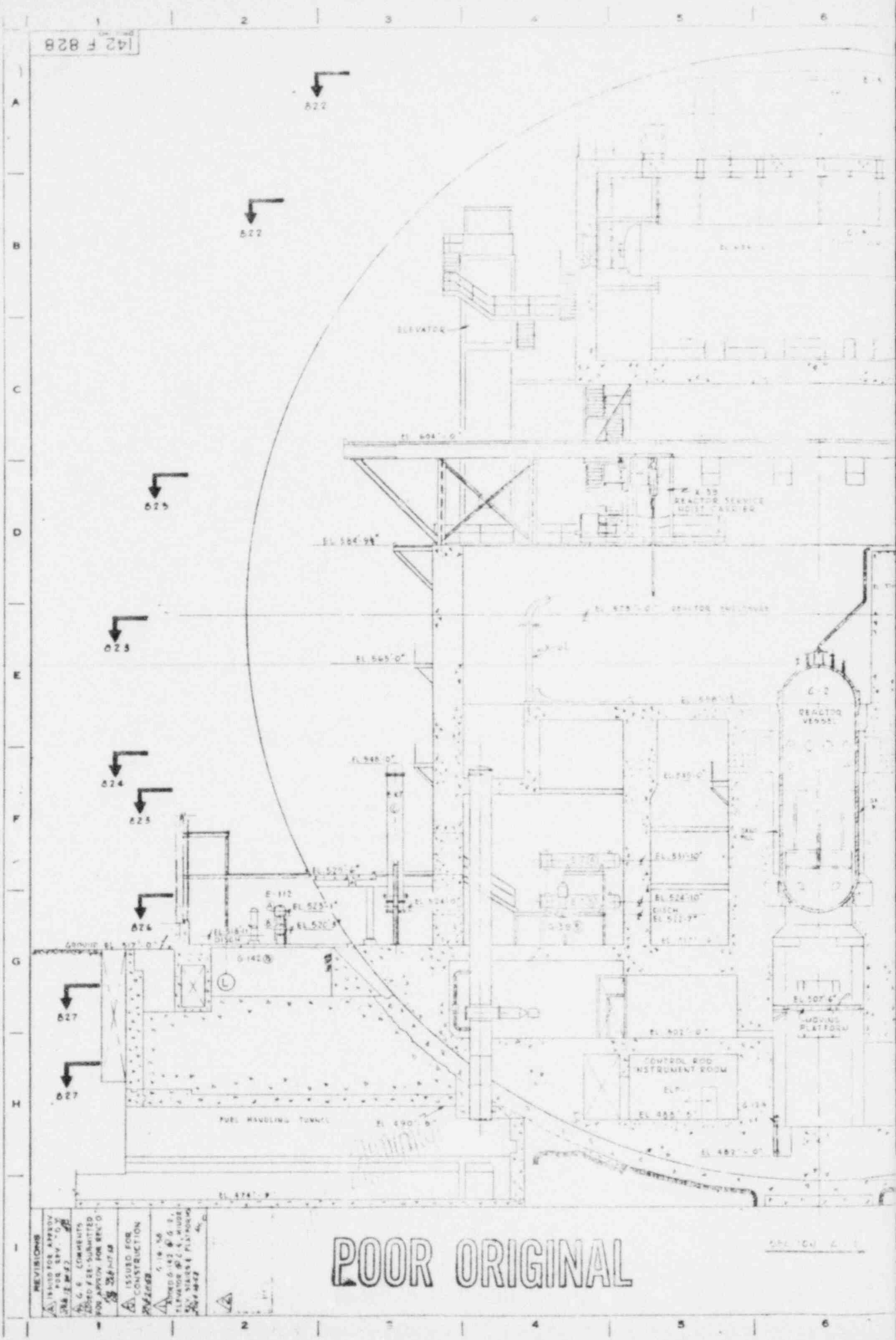


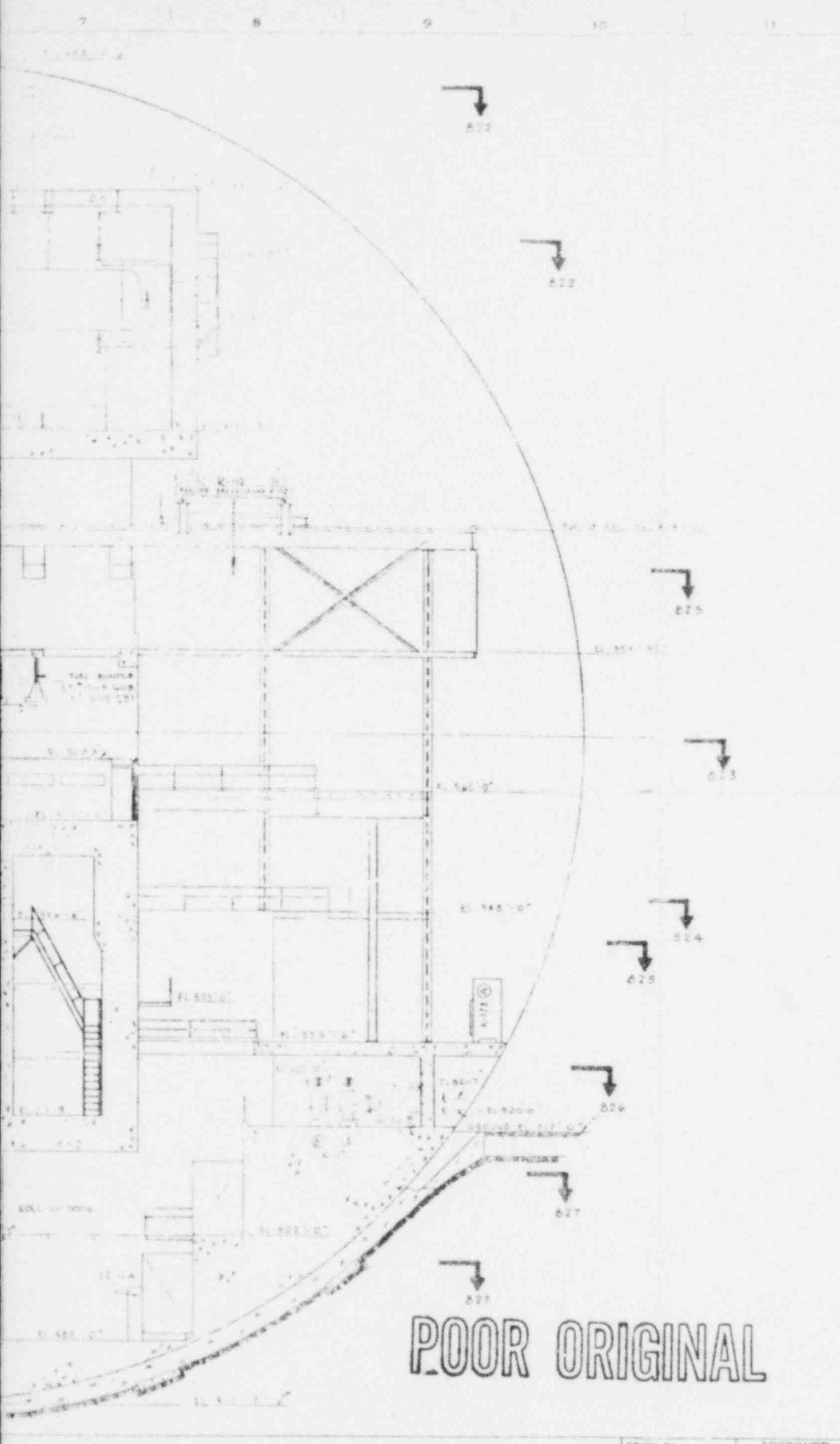
142 F 827
A
142 P-829
D
142 P-830
ACCUMULATORS
G-27
EL. 467'10"
B
142 P-829
T-122 B
DRAIN TANK
T-122 A
DRAIN TANK
REACTOR ℄ ENCLOSURE
C
142 P-830
ELP-1
CONTROL ROD
INSTRUMENT ROOM
ER-10
ER-9
EL. 488'-3"
OUTLINE OF SPHERE @ ELEV. 488'-3"
REACTOR ℄ ENCLOSURE
A
142 P-828
REVISIONS
ISSUED FOR APPROV FOR REV. "0"
G.E. COMMENTS ADDED / RE-SUBMITTED FOR APPROV FOR REV. "0"
ISSUED FOR CONSTRUCTION
6-16-58
ADDED SUMPS
POOR ORIGINAL

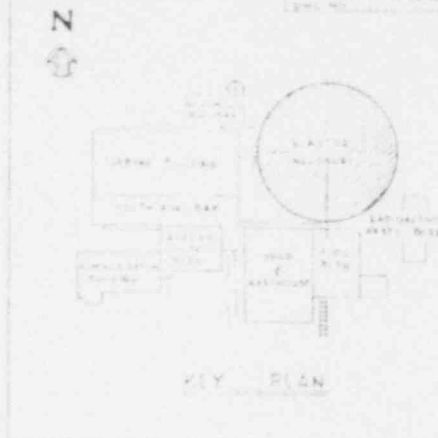


PREPARED BY	BECHTEL CORPORATION, SAN FRANCISCO
JOB NO.	2345
	GENERAL ELECTRIC — ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE	REACTOR ENCLOSURE - EQUIP'T. LOC. PLANS @ EL. 488'-3" & 502'-0"
MADE FOR	DRESDEN NUCLEAR POWER STATION — COMMONWEALTH EDISON COMPANY
DWG. NO.	142 F 827
REV.	2

DRAWN BY — CHECKED BY — ENGINEER — APPROVED BY FOR DATE


142 F 828
822
822
823
823
824
825
826
827
827
ELEVATOR
REACTOR SERVICE HOIST CARRIER
REACTOR VESSEL
MOVING PLATFORM
CONTROL ROD INSTRUMENT ROOM
FUEL HANDLING TUNNEL
REVISIONS
ISSUED FOR CONSTRUCTION
POOR ORIGINAL





142 F 828

142F828

APPROVED
H.B. Webster
By Date
Rev. 0

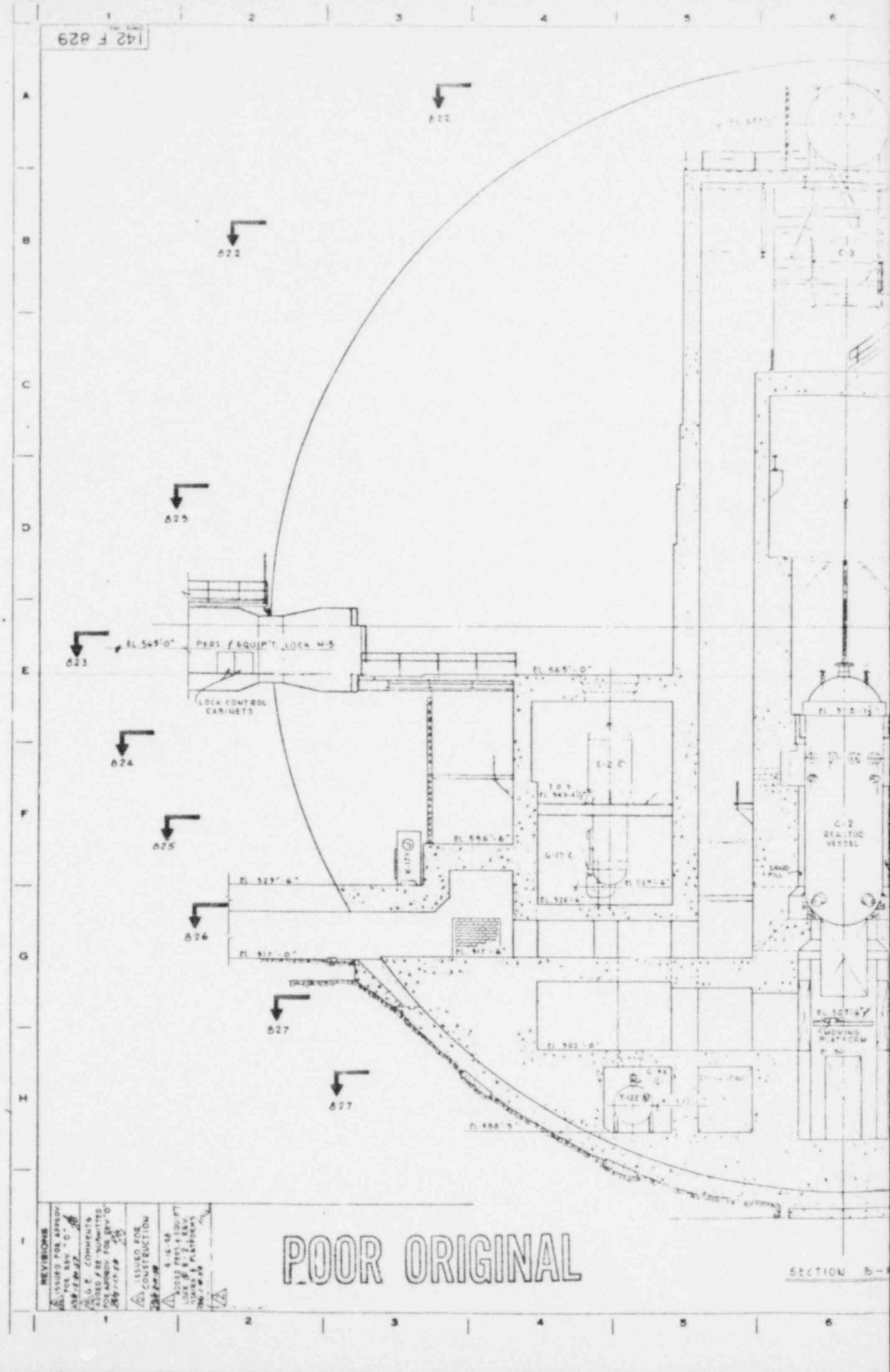
SCALE IN FEET

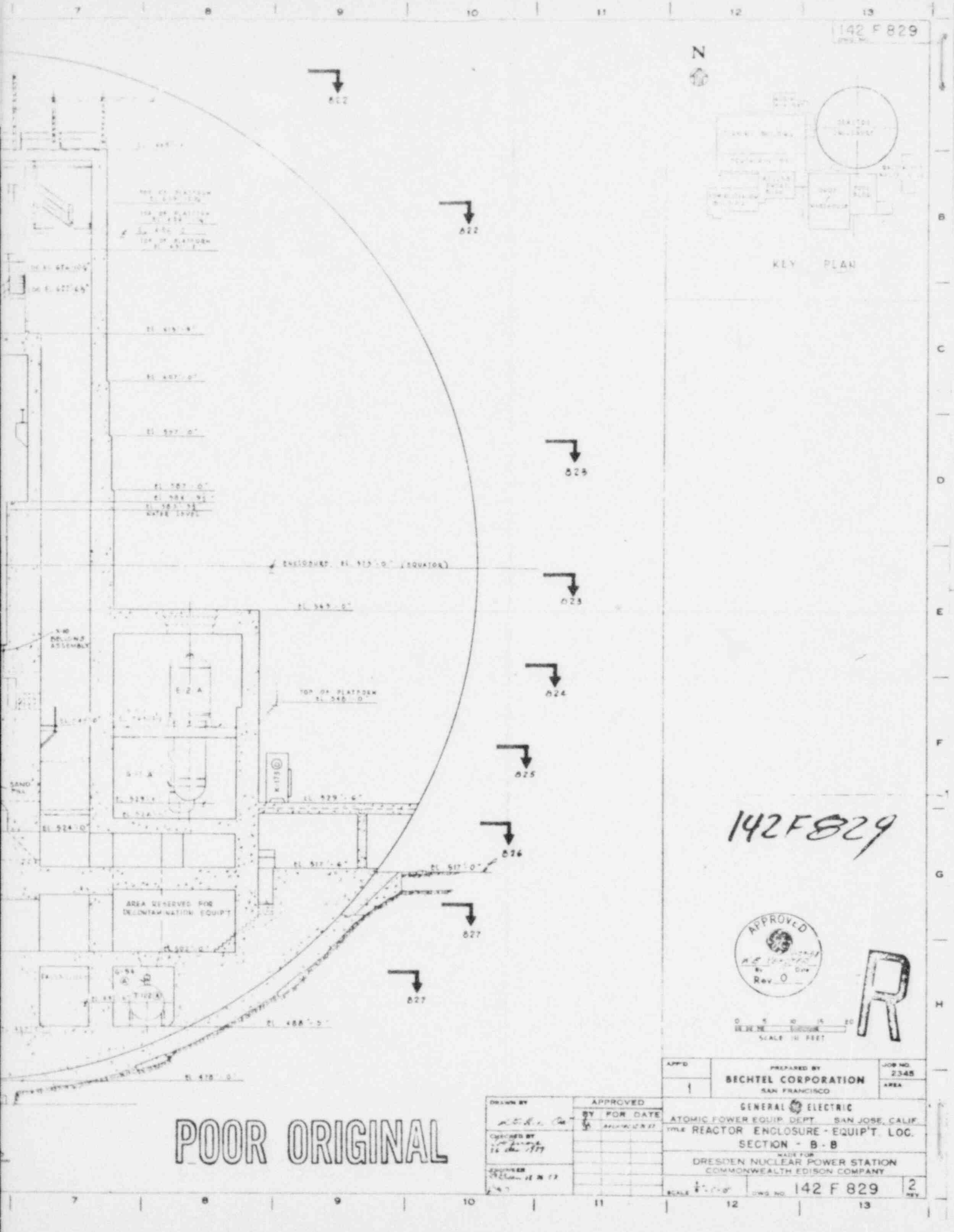
POOR ORIGINAL

APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	JOB NO 2345
1		AREA

DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			
ENGINEER			

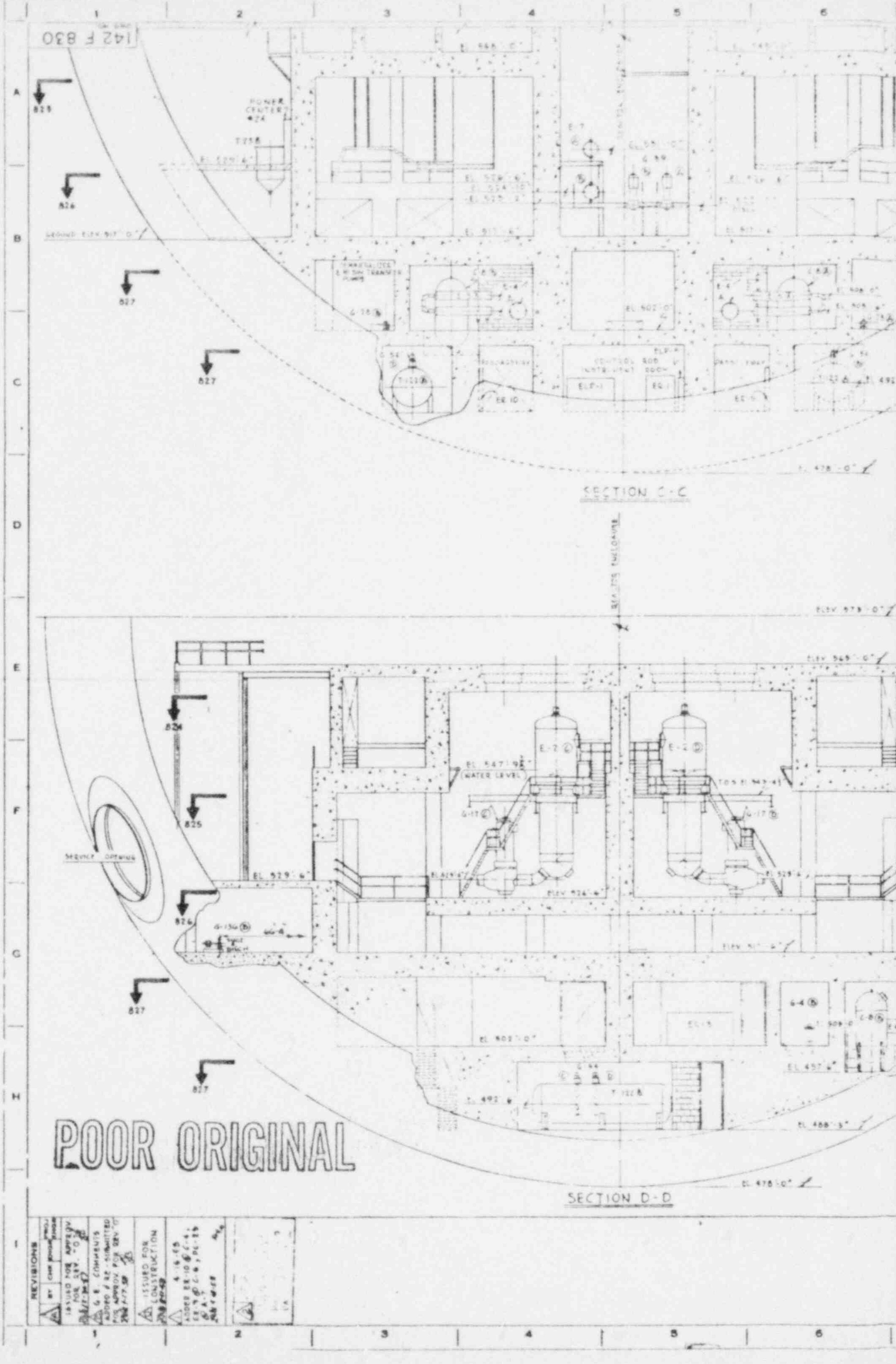
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE REACTOR ENCLOSURE - EQUIP'T. LOC.
SECTION - A-A
MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
SCALE | DWG. NO. 142 F 828 | 2 REV.

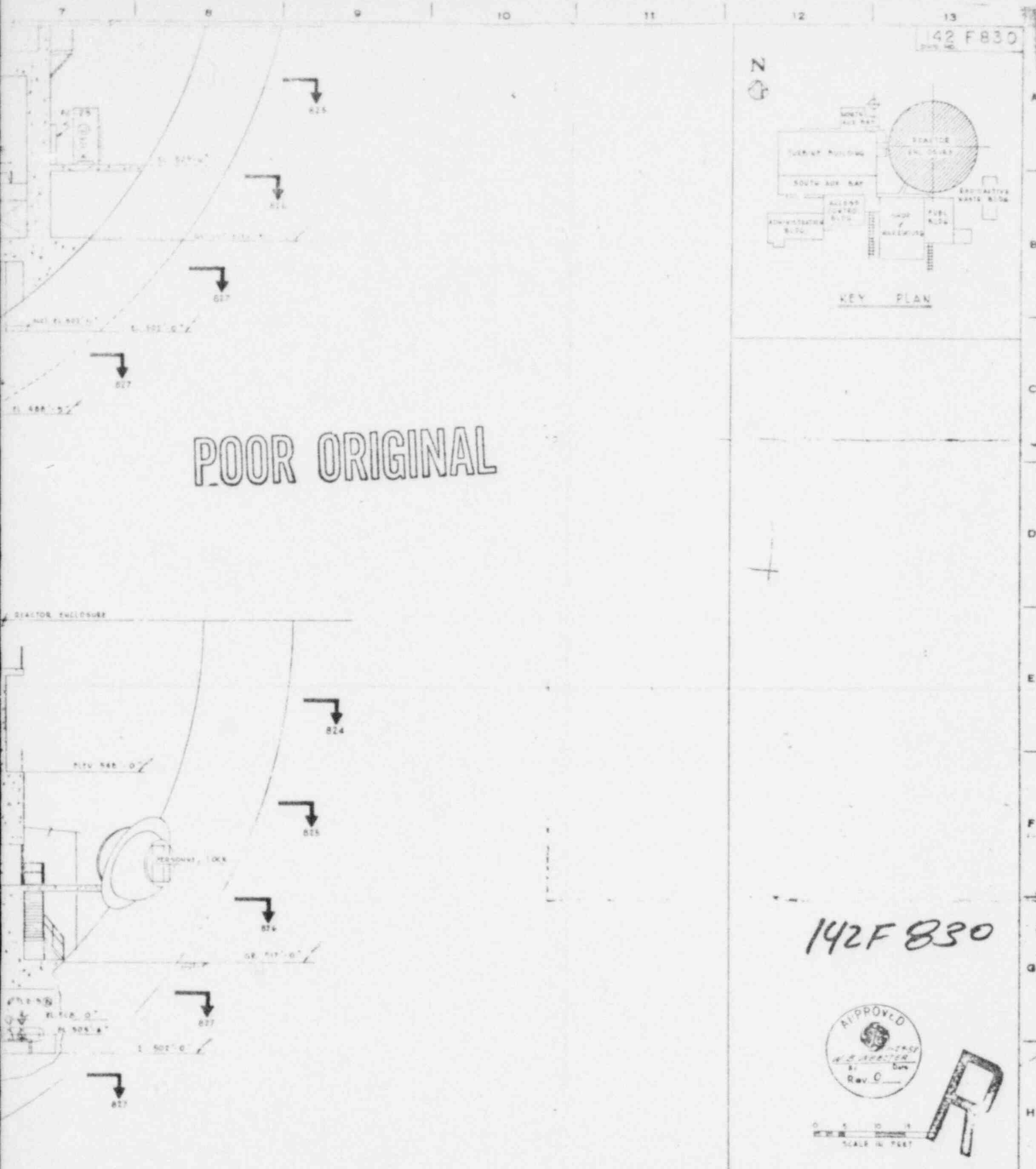

142 F 829
822
822
823
823
824
825
826
827
827
EL 549'-0"
PERS. & EQUIP'T. LOCK H-5
LOCK CONTROL CABINETS
EL 565'-0"
E-2 C
EL 556'-6"
EL 529'-6"
EL 517'-0"
EL 517'-6"
C-2 REACTOR VESSEL
SAND FILL
MOVING PLATFORM
EL 502'-0"
REVISIONS
ISSUED FOR CONSTRUCTION
POOR ORIGINAL
SECTION B-



POOR ORIGINAL

APP'D	1
PREPARED BY	BECHTEL CORPORATION, SAN FRANCISCO
JOB NO.	2345
	GENERAL ELECTRIC — ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE	REACTOR ENCLOSURE - EQUIP'T. LOC. SECTION - B - B
MADE FOR	DRESDEN NUCLEAR POWER STATION, COMMONWEALTH EDISON COMPANY
DWG NO.	142 F 829
REV	2


142 F 830
POWER CENTER
SECTION C-C
SERVICE OPENING
EL 529'-4"
WATER LEVEL
SECTION D-D
REVISIONS
ISSUED FOR CONSTRUCTION
1
2
3
4
5
6
A
B
C
D
E
F
G
H
I



APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO		JOB NO. 2348
CLASS 1			AREA

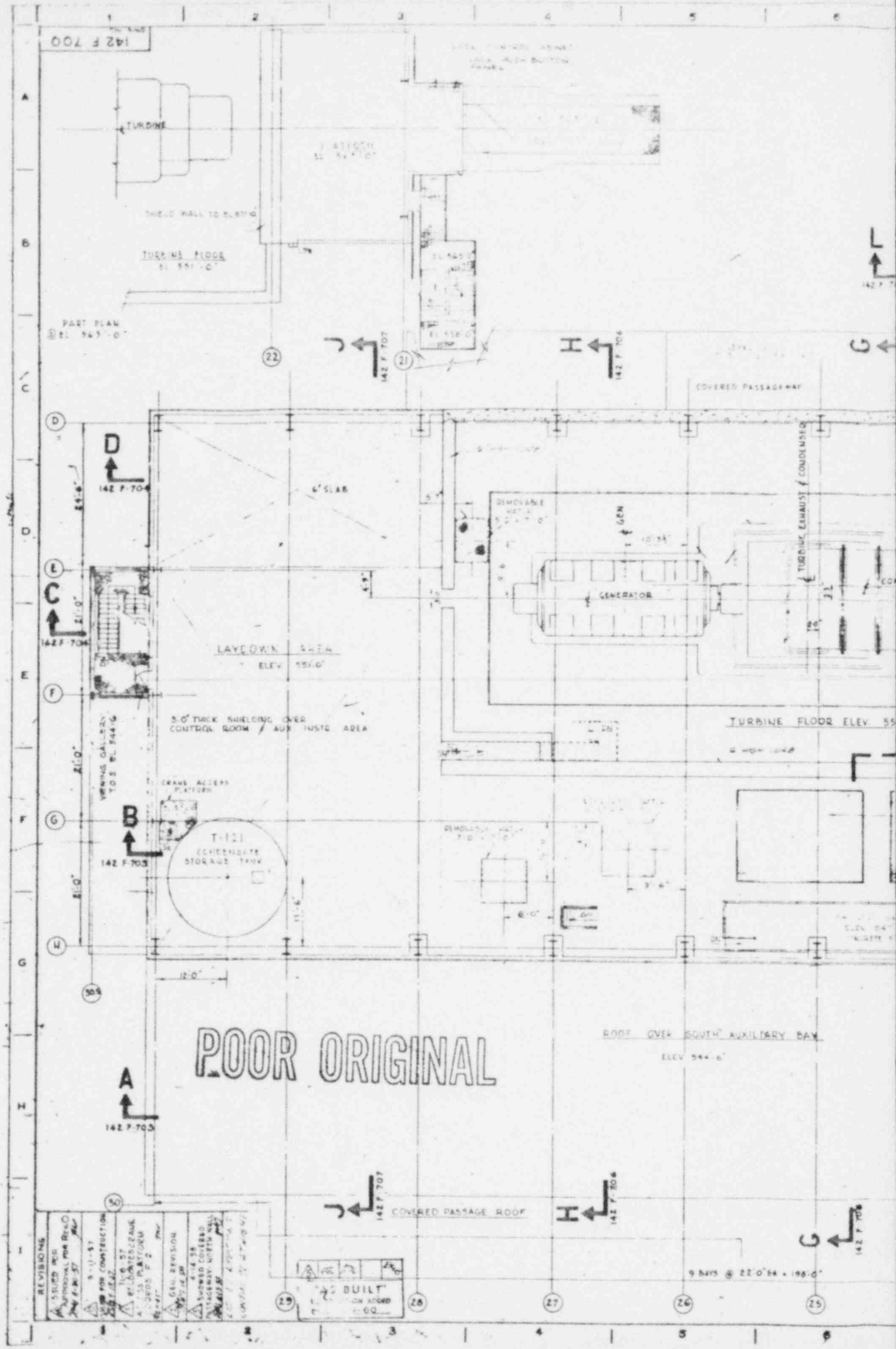
DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			
ENGINEER ... 12-26-67			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.

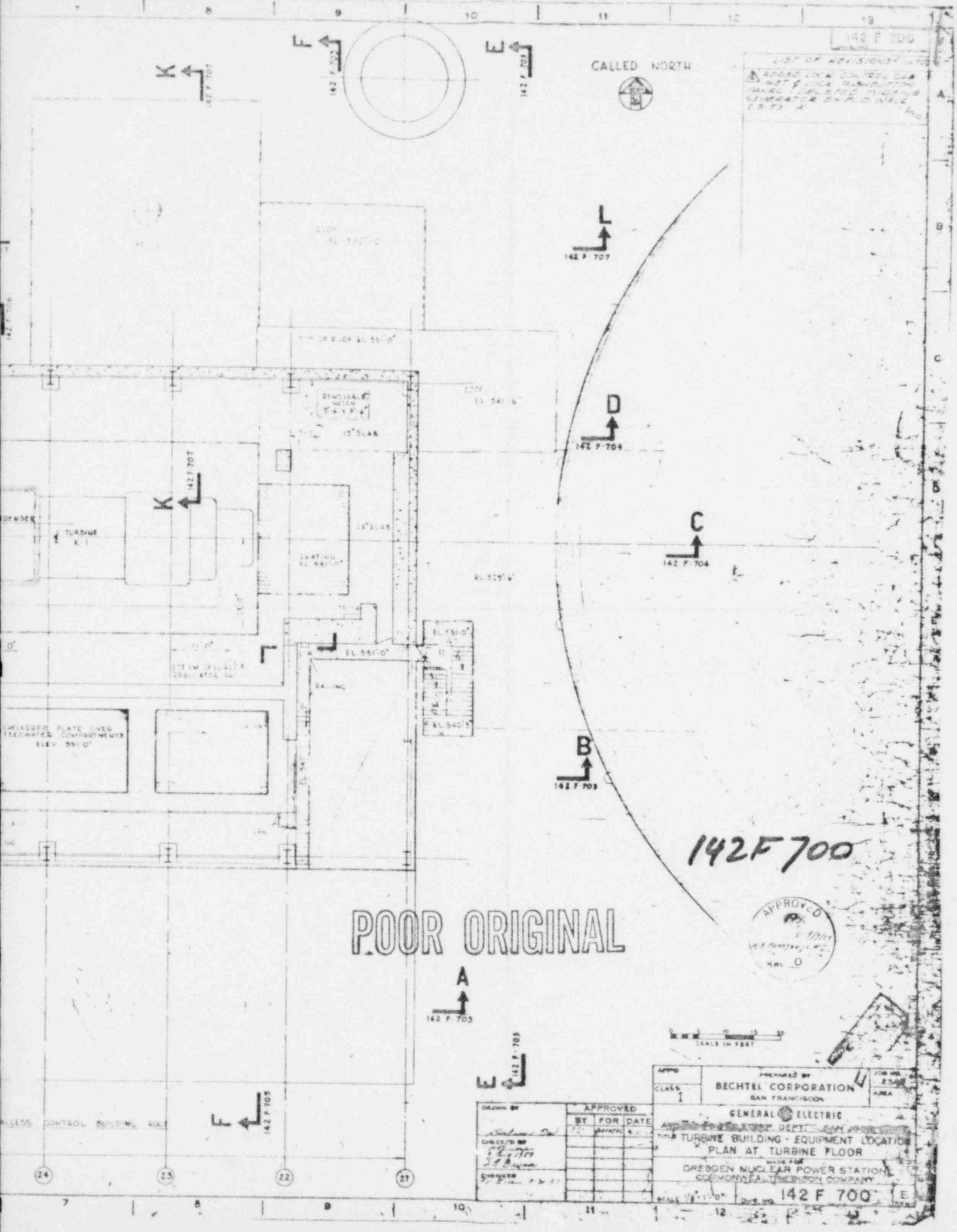
TITLE: REACTOR ENCLOSURE - EQUIP'T. LOC.
SECTIONS - C-C , D-D

MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY

SCALE 1/8" = 1'-0" DWG. NO. 142 F 830 SHEET 2

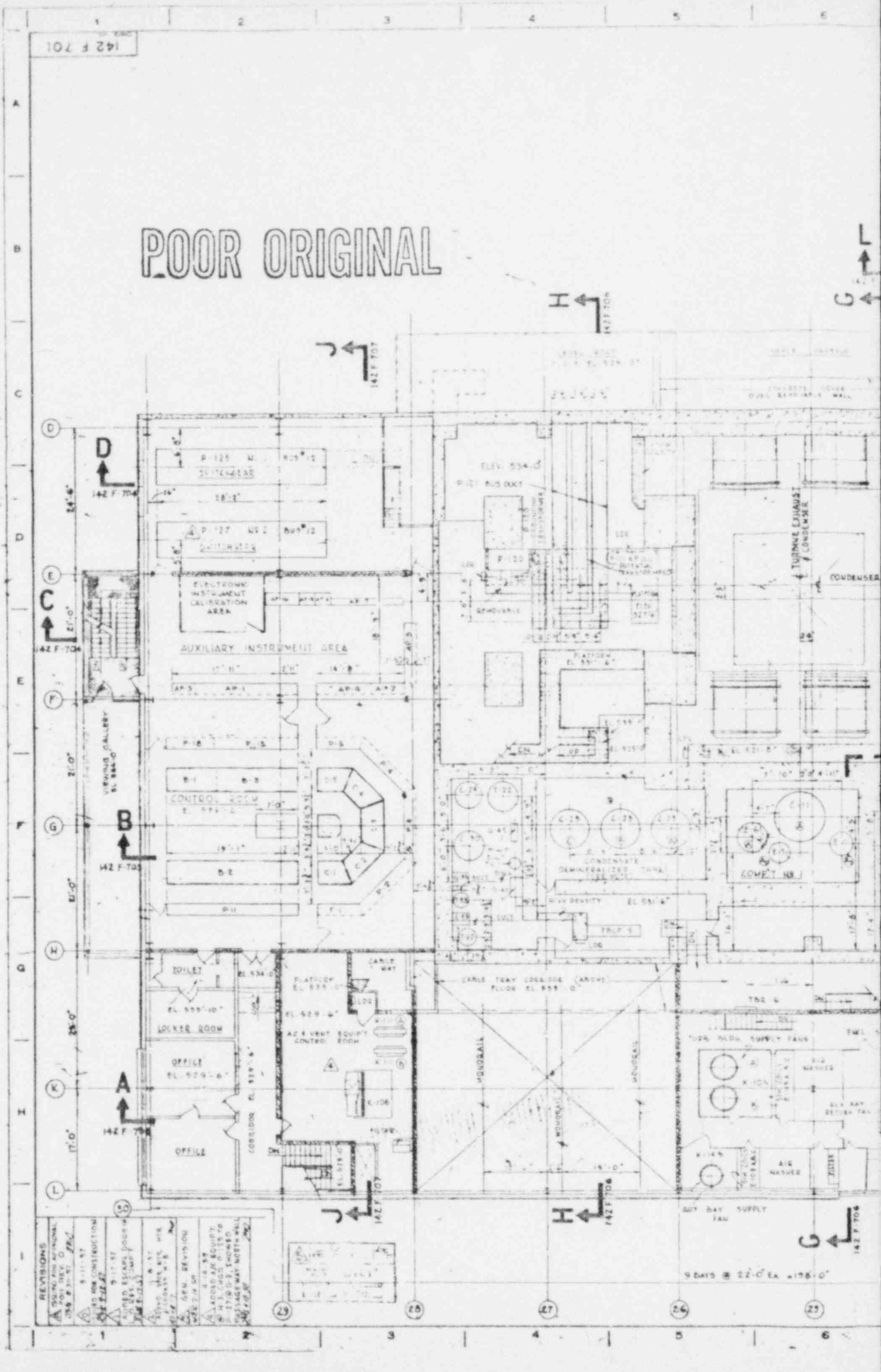


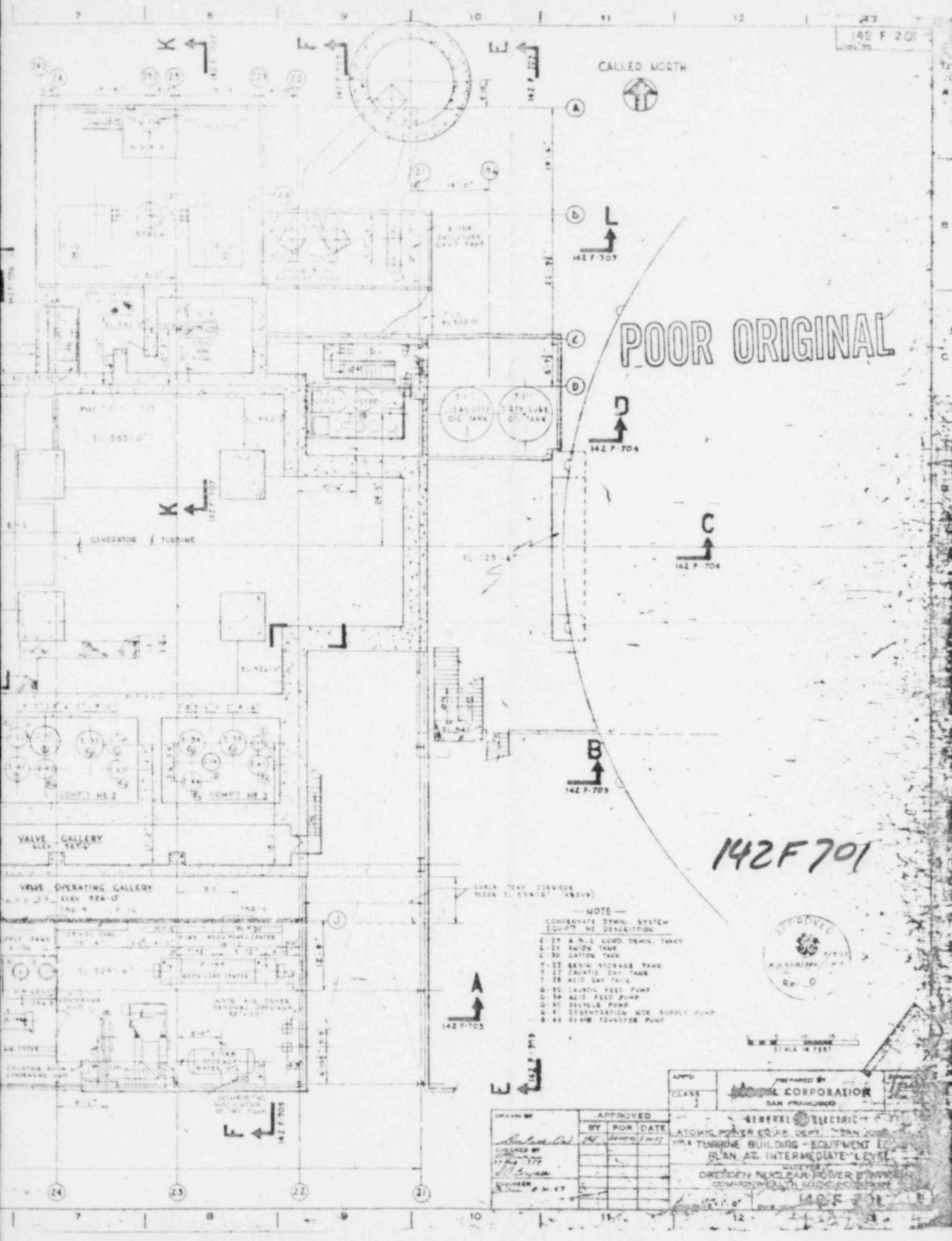
142 F 700
TURBINE
TURBINE FLOOR
EL 551'-0"
SHIELD WALL TO BLSTNG
PART PLAN
EL 563'-0"
COVERED PASSAGEWAY
D
142 F 704
C
142 F 704
B
142 F 703
A
142 F 703
J
142 F 707
H
142 F 706
G
L
4" SLAB
REMOVABLE HATCH
GEN
GENERATOR
TURBINE EXHAUST / CONDENSER
LAYDOWN AREA
ELEV 551'-0"
5'-0" THICK SHIELDING OVER CONTROL ROOM / AUX WASTE AREA
VIEWING GALLERY T.O.S. EL 544'-6"
CRANE ACCESS PLATFORM
T-121
CONDENSATE STORAGE TANK
TURBINE FLOOR ELEV 55
12'-0"
POOR ORIGINAL
ROOF OVER SOUTH AUXILIARY BAY
ELEV 544'-6"
COVERED PASSAGE ROOF
9 BAYS @ 22'-0" EA = 198'-0"
"AS BUILT"
REVISIONS
ISSUED FOR APPROVAL
ISSUED FOR CONSTRUCTION
GEN. REVISION
22
21
30
30A
29
28
27
26
25



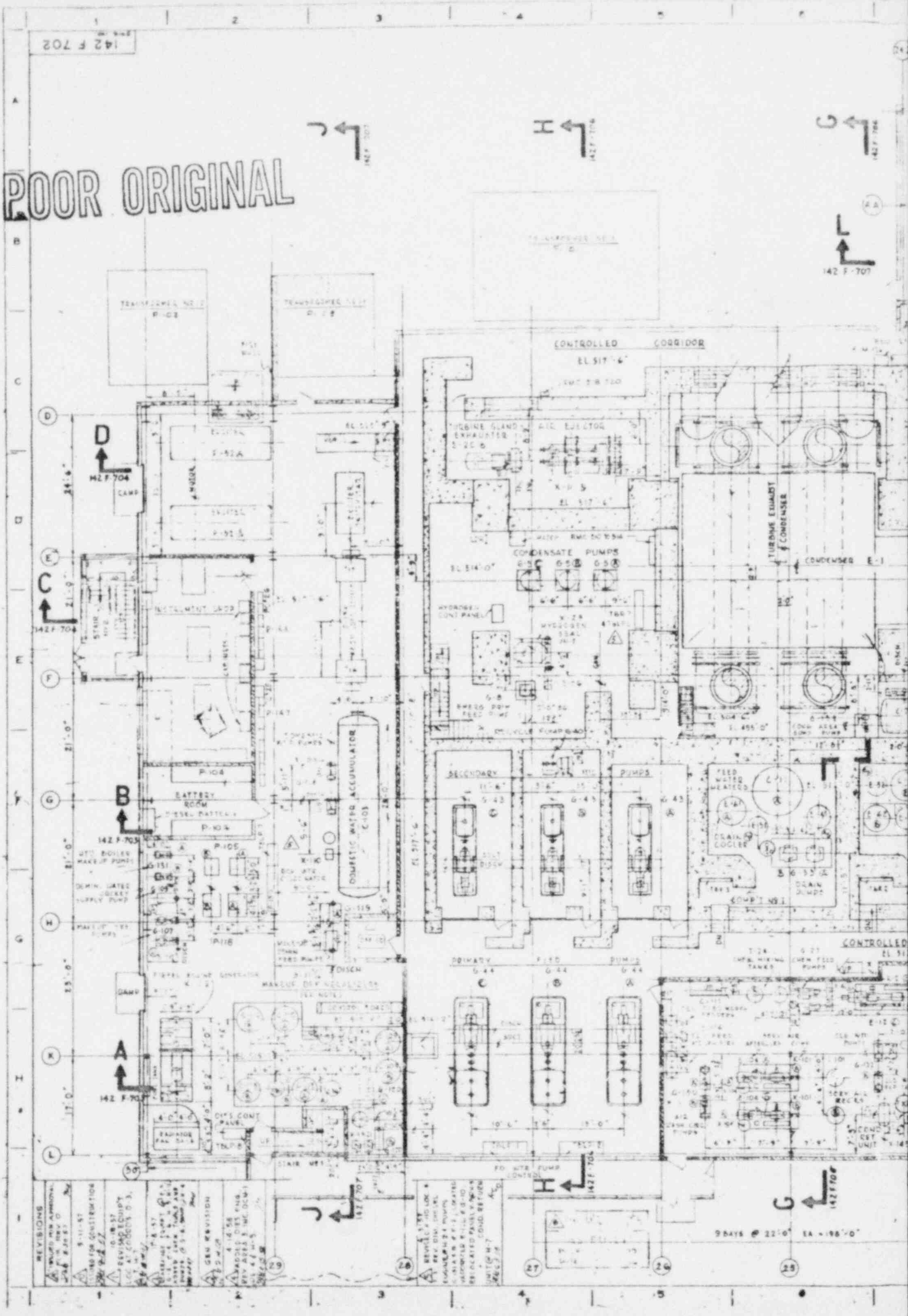
142 F 700
LIST OF REVISIONS
CALLED NORTH
TOP OF ROOF EL 551'-0"
REMOVABLE HATCH
12" SLAB
TURBINE
GRATING
EL 551'-0"
EL 540'-3"
RAILING
STEAM SEAL REGULATOR
CHECKERED PLATE OVER FEEDWATER COMPARTMENTS ELEV 551'-0"
142 F 707
142 F 704
142 F 703
142 F 706
142F 700
POOR ORIGINAL
APPROVED
REV 0
SCALE IN FEET
BECHTEL CORPORATION
SAN FRANCISCO
GENERAL ELECTRIC
TURBINE BUILDING - EQUIPMENT LOCATION
PLAN AT TURBINE FLOOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
142 F 700

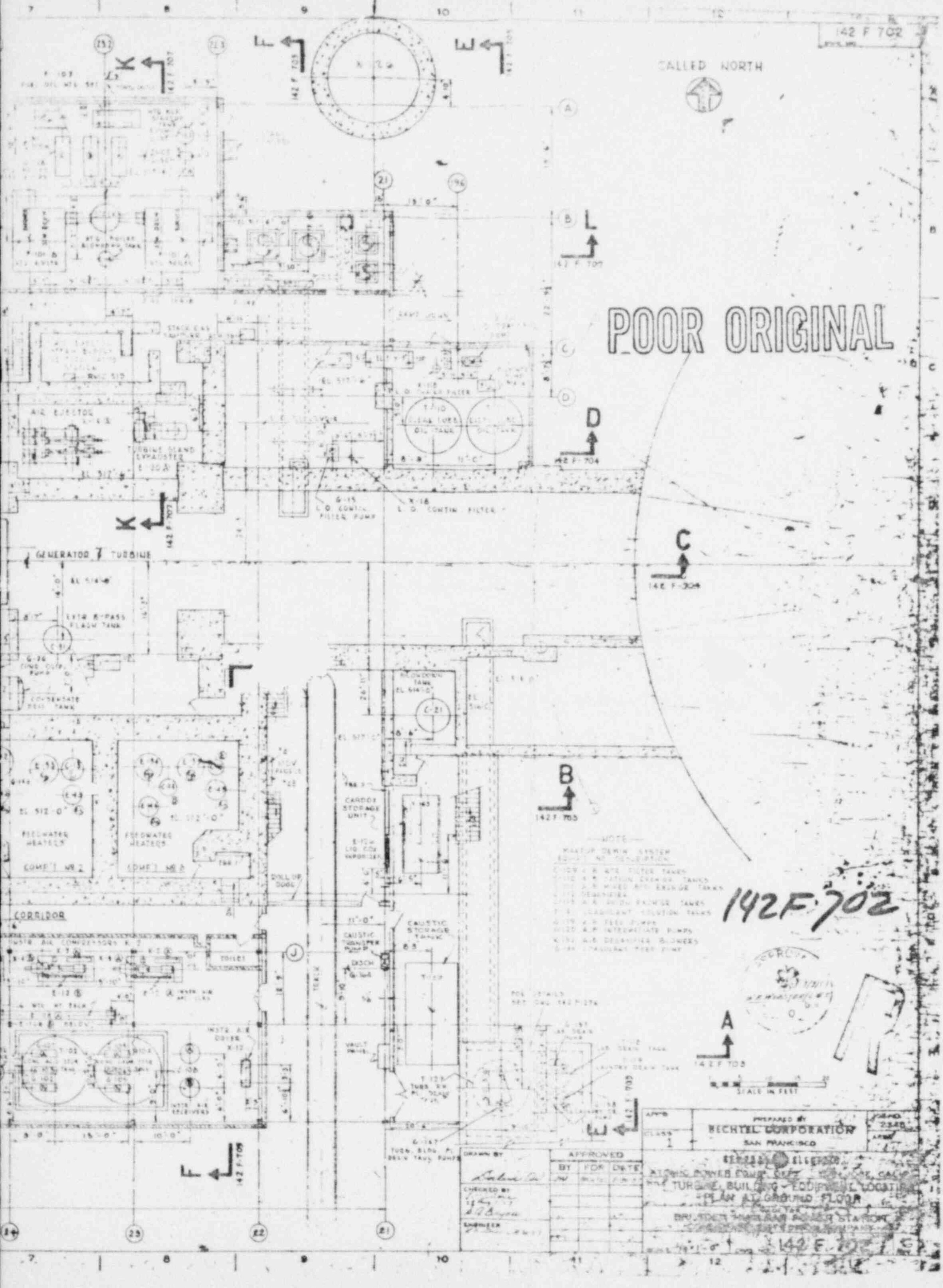
POOR ORIGINAL



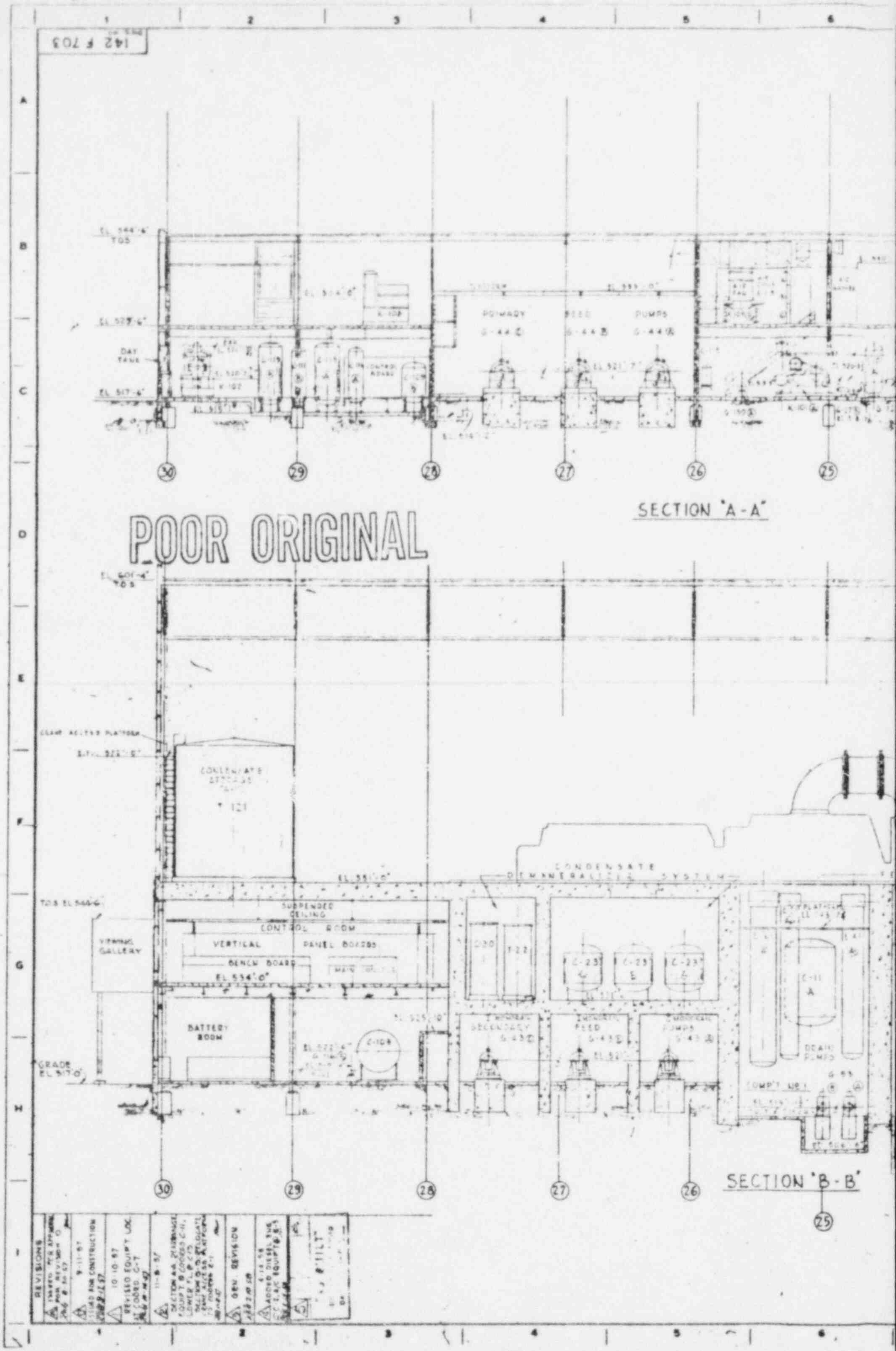


142 F 701
CALLED NORTH
POOR ORIGINAL
142F701
GENERATOR ℄ TURBINE
VALVE GALLERY
VALVE OPERATING GALLERY
COMPT NO 2
COMPT NO 3
CLEAN LUBE OIL TANK
DIRTY LUBE OIL TANK
EL 529'-6"
EL 555'-0"
— NOTE —
CONDENSATE DEMIN. SYSTEM
EQUIPT NO DESCRIPTION
C-29 A, B, C COND. DEMIN. TANKS
C-31 ANION TANK
C-30 CATION TANK
T-27 RESIN STORAGE TANK
T-27 CAUSTIC DAY TANK
T-28 ACID DAY TANK
G-50 CAUSTIC FEED PUMP
G-54 ACID FEED PUMP
G-60 RECYCLE PUMP
G-41 REGENERATION WTR SUPPLY PUMP
G-49 RESIN TRANSFER PUMP
SCALE IN FEET
APPROVED
Rev 0
PREPARED BY
CORPORATION
SAN FRANCISCO
CLASS
DRAWN BY
CHECKED BY
ENGINEER
APPROVED
BY
FOR
DATE
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE
TITLE TURBINE BUILDING - EQUIPMENT
PLAN AT INTERMEDIATE LEVEL
MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
142 F 701


142 F 702
POOR ORIGINAL
CONTROLLED CORRIDOR
EL 517'-6"
TURBINE GLAND EXHAUSTER
AIR EJECTOR
CONDENSATE PUMPS
TURBINE EXHAUST CONDENSER
CONDENSER E-1
INSTRUMENT SHOP
BATTERY ROOM
DOMESTIC WATER ACCUMULATOR C-103
SECONDARY
PUMPS
FEED WATER HEATERS
DRAIN COOLER
DRAIN PUMPS
PRIMARY FEED PUMPS
MAKEUP DEMINERALIZERS
CONTROLLED
STAIR
9 BAYS @ 22'-0" EA = 198'-0"
REVISIONS

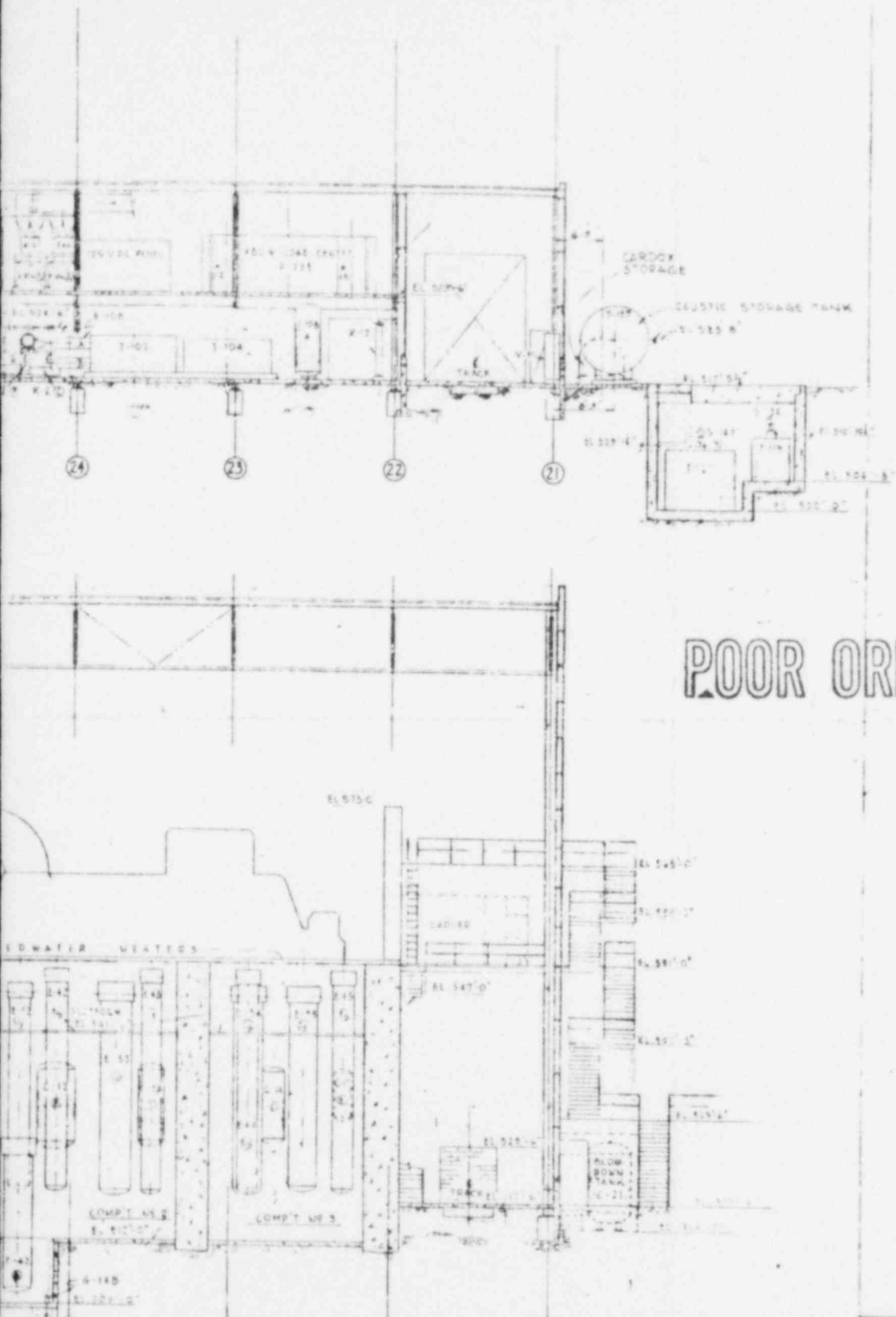


142 F 702
CALLED NORTH
POOR ORIGINAL
AIR EJECTOR
TURBINE GLAND EXHAUSTER
GENERATOR & TURBINE
L.O. CONTIN. FILTER PUMP
L.O. CONTIN. FILTER
CONDENSATE DRAIN TANK
FEEDWATER HEATERS
COMP'T NO 2
COMP'T NO 3
BLOWDOWN TANK
CARDOX STORAGE UNIT
CAUSTIC STORAGE TANK
CORRIDOR
TOILET
INSTR AIR DRYER
ROLL UP DOOR
NOTE
MAKEUP DEMIN. SYSTEM
142F-702
SCALE IN FEET
PREPARED BY
BECHTEL CORPORATION
SAN FRANCISCO
APPROVED
BY
FOR
DATE
DRAWN BY
CHECKED BY
ENGINEER
TURBINE BUILDING - EQUIPMENT LOCATION
PLAN AT GROUND FLOOR
142 F 702



142 F 703
SECTION "A-A"
POOR ORIGINAL
PRIMARY FEED PUMPS
DAY TANK
SECTION "B-B"
CONDENSATE STORAGE TANK
CONDENSATE DEMINERALIZER SYSTEM
SUSPENDED CEILING
CONTROL ROOM
VERTICAL PANEL BOARDS
BENCH BOARD
EL 534'-0"
VIEWING GALLERY
BATTERY ROOM
GRADE EL 517'-0"
C-104
FEED
PUMPS
REVISIONS
GEN. REVISION

142 F 703



POOR ORIGINAL

142 F 703

APPROVED Rev. 0

SCALE IN FEET

DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			

PREPARED BY BECHTEL CORPORATION SAN FRANCISCO

JOB NO. 2345

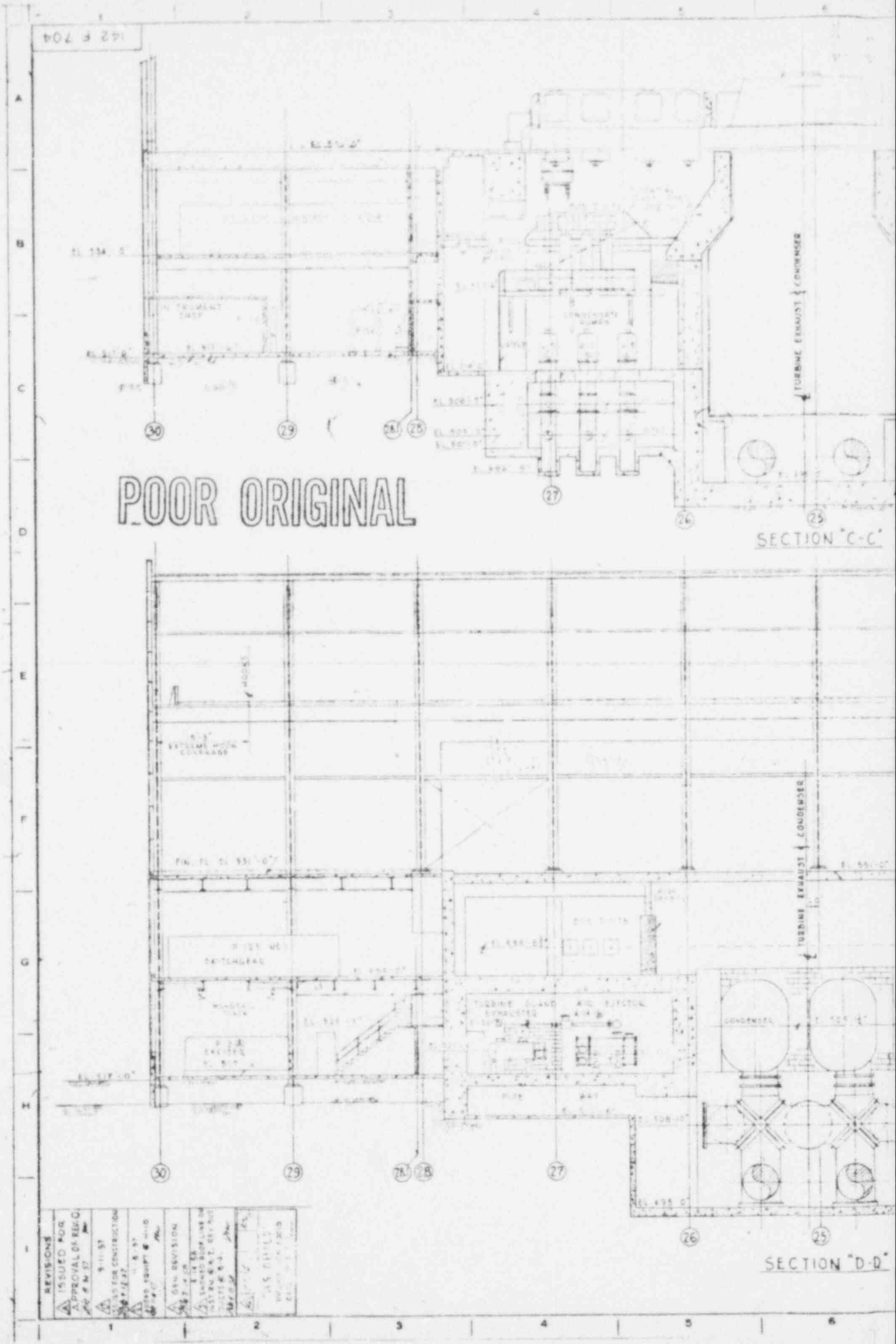
SCALE 1

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.

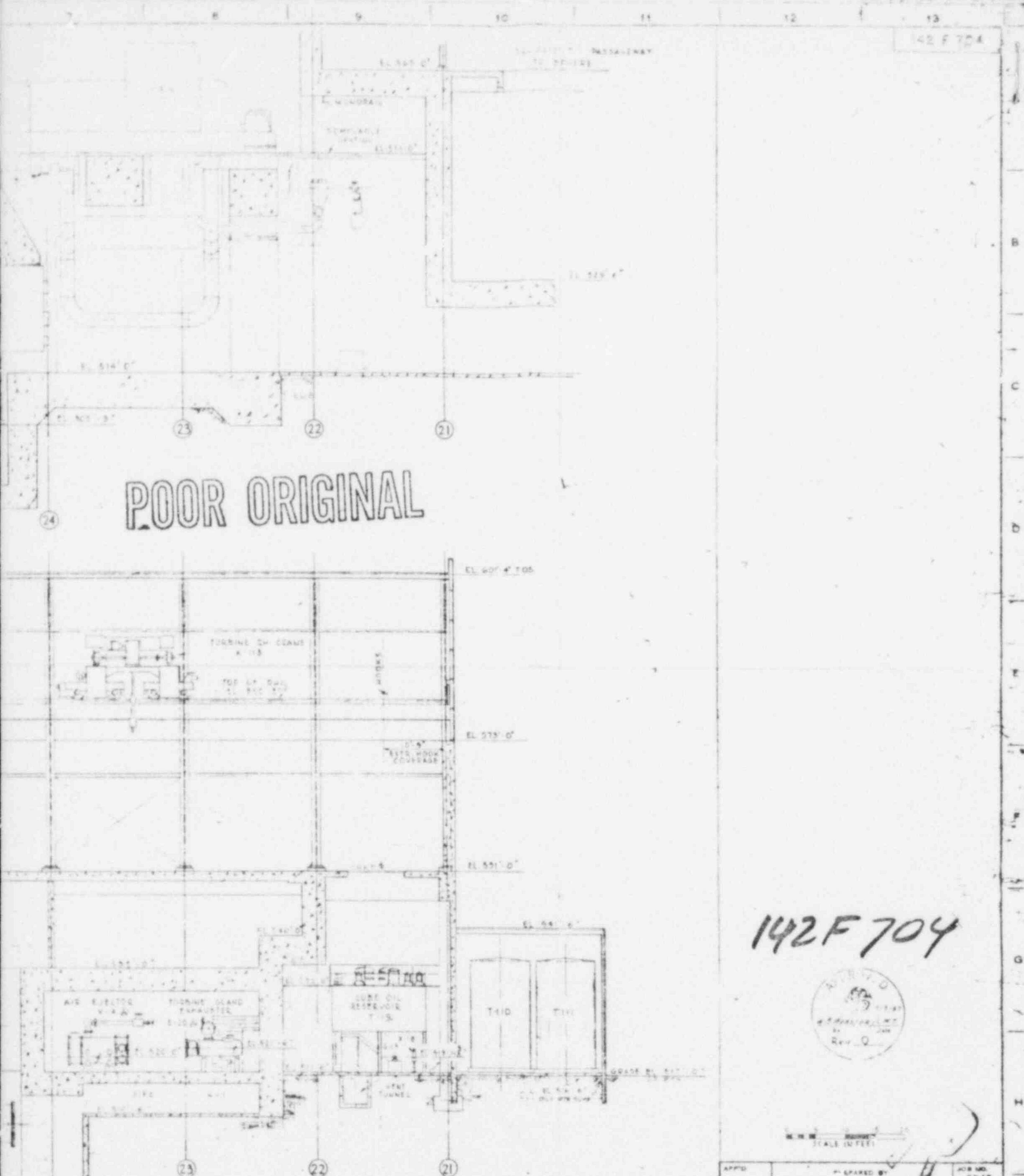
TITLE TURBINE BUILDING - EQUIPMENT LOCATION LONGITUDINAL SECTIONS "A-A" & "B-B"

MADE FOR DRESDEN NUCLEAR POWER STATION COMMONWEALTH EDISON COMPANY

SCALE 1/8"=1'-0" DWG. NO. 142 F 703 5



142 F 704
POOR ORIGINAL
TURBINE EXHAUST ℄ CONDENSER
SECTION "C-C"
EXTREME HOOK COVERAGE
SWITCHGEAR
TURBINE GLAND
EXHAUSTER
CONDENSER
PIPE WAY
SECTION "D-D"
REVISIONS
ISSUED FOR APPROVAL OF REV'G
ISSUED FOR CONSTRUCTION
GEN. REVISION



APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	JOB NO. 2348
CLASS 1		AREA

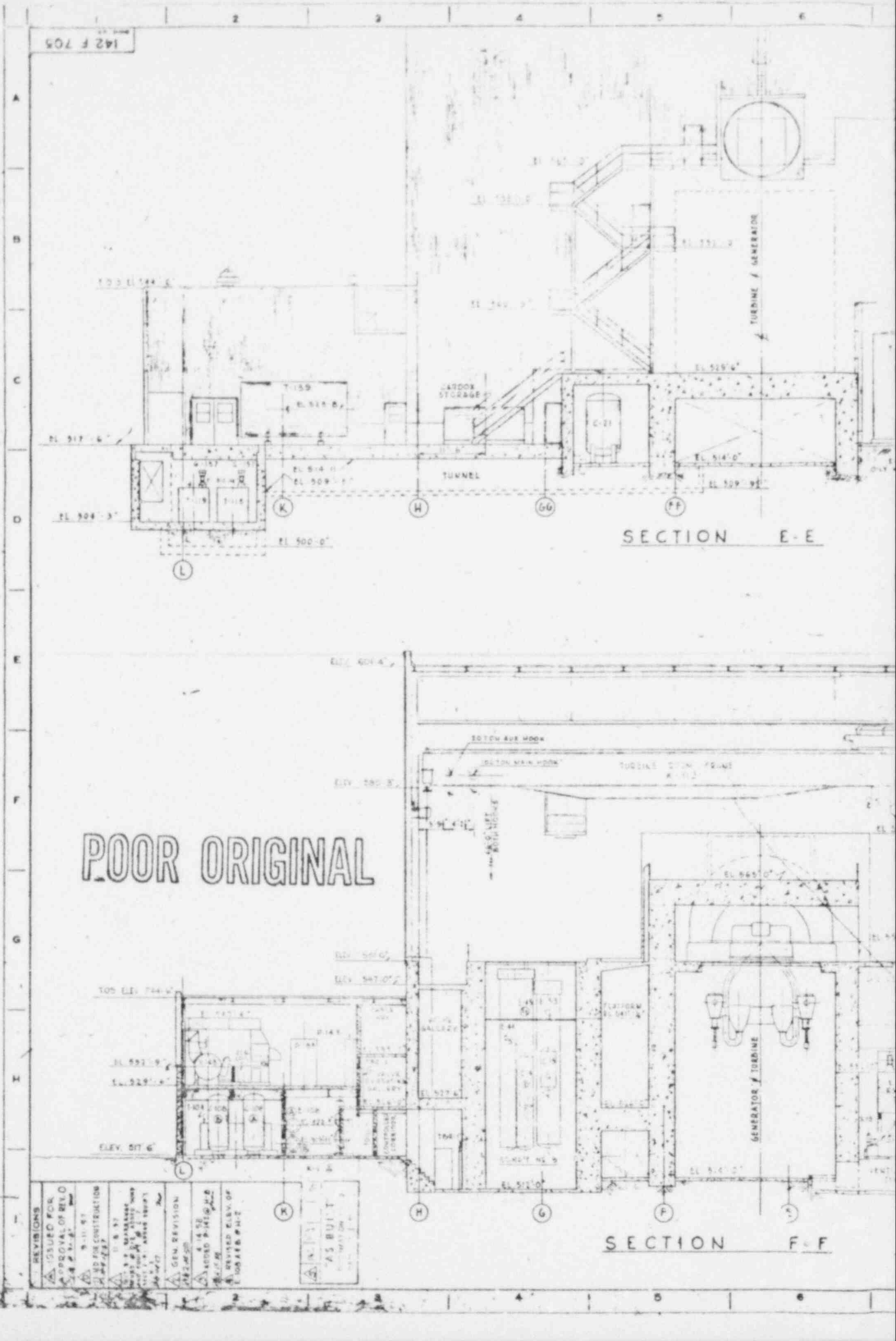
DRAWN BY	APPROVED BY	FOR	DATE
CHECKED BY			
ENGINEER			

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.

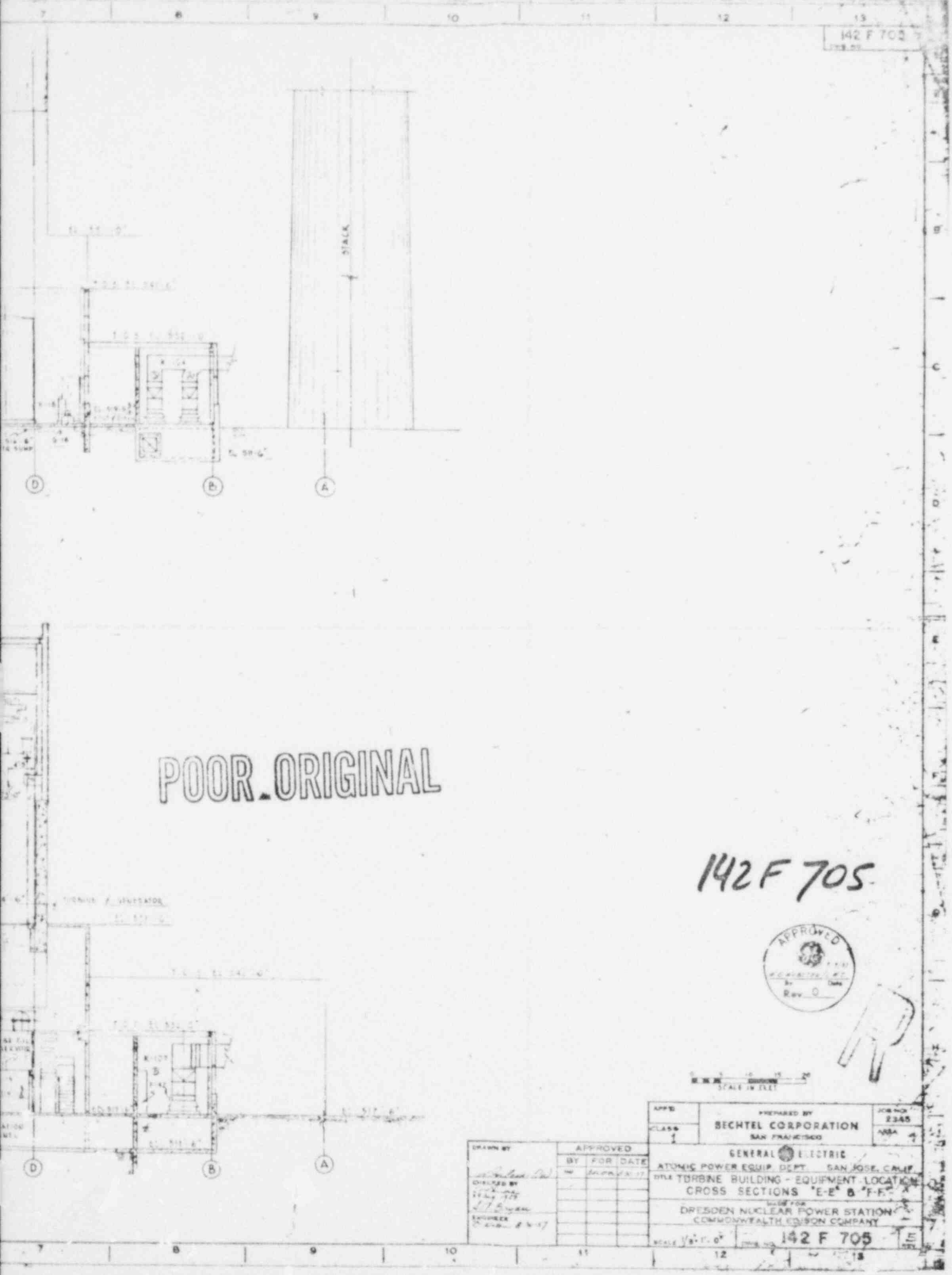
TITLE TURBINE BUILDING - EQUIPMENT LOCATION LONGITUDINAL SECTIONS: "C-C" & "D-D"

MADE FOR DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY

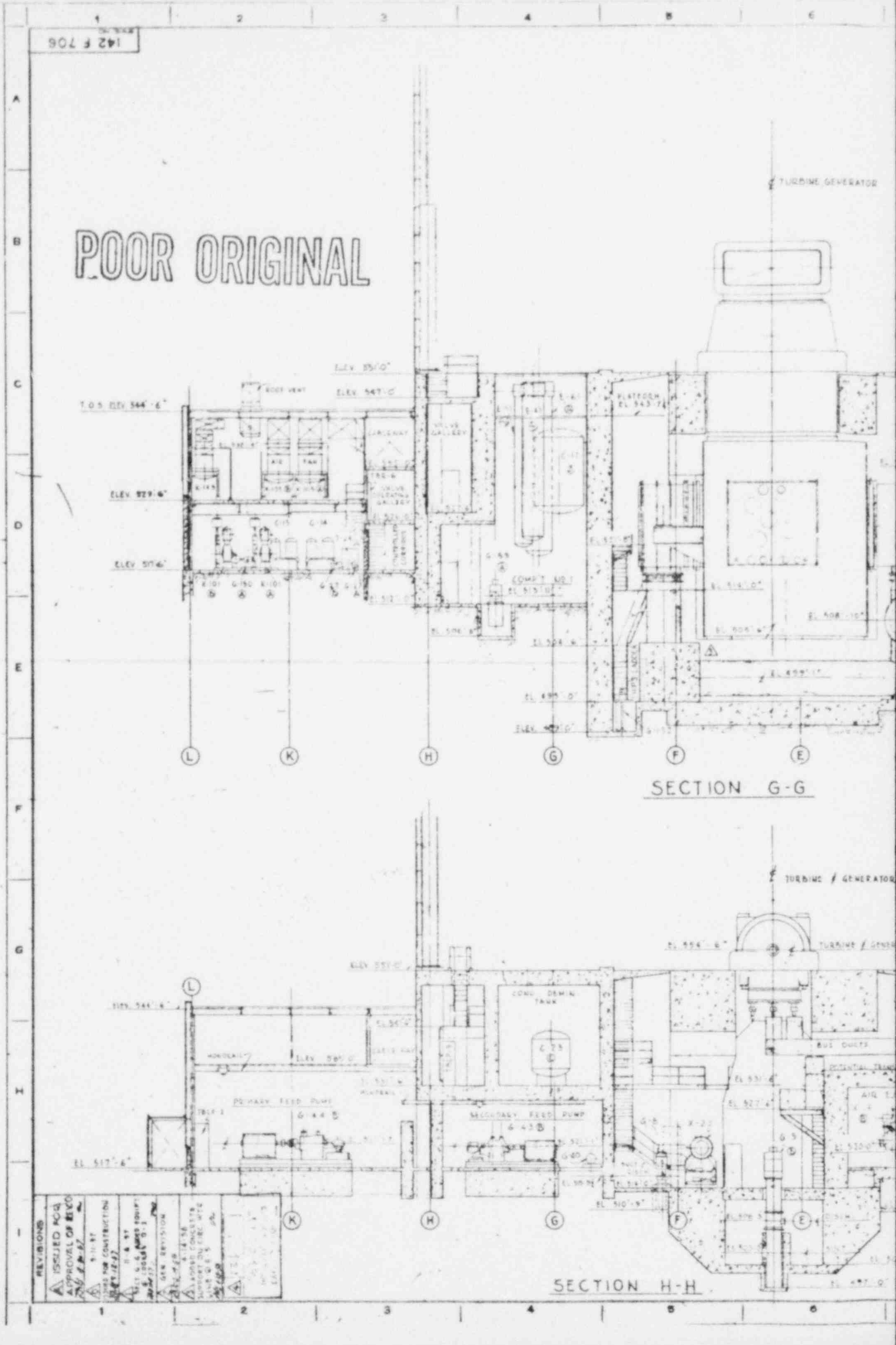
SCALE 1/8" = 1'-0" DWG. NO. 142 F 704 REV. 4



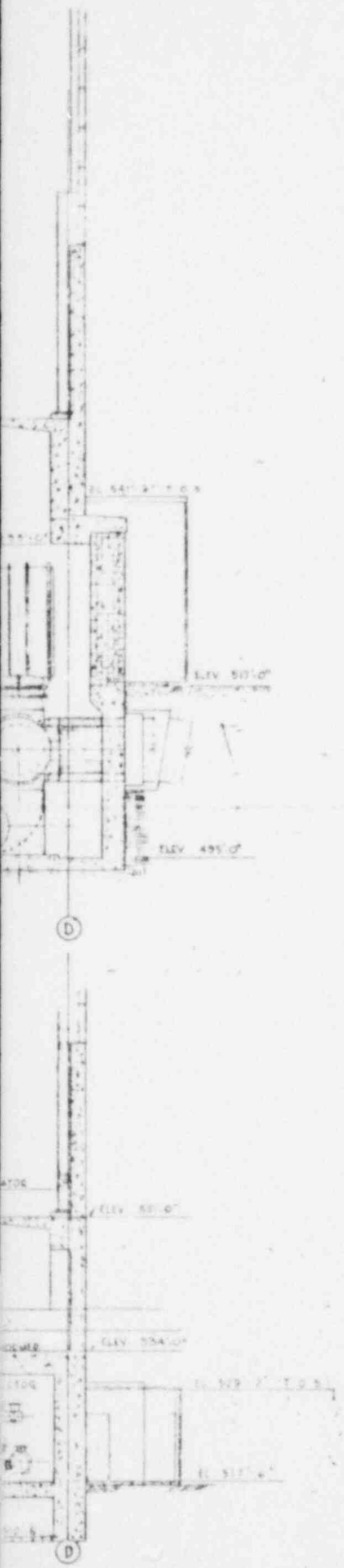
142 F 705
SECTION E-E
SECTION F-F
TUNNEL
CARDOX STORAGE
T-159
TURBINE ℄ GENERATOR
GENRRATOR ℄ TURBINE
EL 917'-6"
EL 904'-3"
EL 900'-0"
EL 529'-6"
EL 514'-0"
EL 509'-9½"
ELEV. 601'-4"
ELEV. 580'-8"
ELEV. 547'-0"
TOS ELEV. 744'-6"
ELEV. 517'-6"
EL 512'-0"
EL 514'-0"
EL 565'-0"
PLATFORM
SHAFT NO. 9
GALLERY
TURBINE ROOM CRANE
50 TON AUX HOOK
150 TON MAIN HOOK
REVISIONS
ISSUED FOR APPROVAL OF REV. 0
GEN. REVISION
"AS BUILT"
POOR ORIGINAL


142 F 705
STACK
D
B
A
POOR ORIGINAL
142 F 705
APPROVED
Rev. 0
SCALE IN FEET
PREPARED BY
BECHTEL CORPORATION
SAN FRANCISCO
JOB NO. 2345
CLASS 1
DRAWN BY
CHECKED BY
ENGINEER
APPROVED
BY
FOR
DATE
GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE TURBINE BUILDING - EQUIPMENT LOCATION
CROSS SECTIONS "E-E" & "F-F"
MADE FOR
DRESDEN NUCLEAR POWER STATION
COMMONWEALTH EDISON COMPANY
SCALE 1/8" = 1'-0"
DWG. NO. 142 F 705
E

POOR ORIGINAL



142 F 706



POOR ORIGINAL

142 F 706

APPROVED
Rev. 0

DRAWN BY	APPROVED			APP'D	PREPARED BY BECHTEL CORPORATION SAN FRANCISCO	JOB NO. 2343
CHECKED BY L. T. Bryan	BY	FOR	DATE	CLASS 1		AREA
ENGINEER						

GENERAL ELECTRIC
ATOMIC POWER EQUIP. DEPT. SAN JOSE, CALIF.
TITLE TURBINE BUILDING - EQUIPMENT LOCATION CROSS SECTIONS "G-G" & "H-H"
MADE FOR DRESDEN NUCLEAR POWER STATION COMMONWEALTH EDISON COMPANY
SCALE 1/8" = 1'-0" DWG. NO. 142 F 706 REV. 4

932C449

GENERAL ELECTRIC 932C449

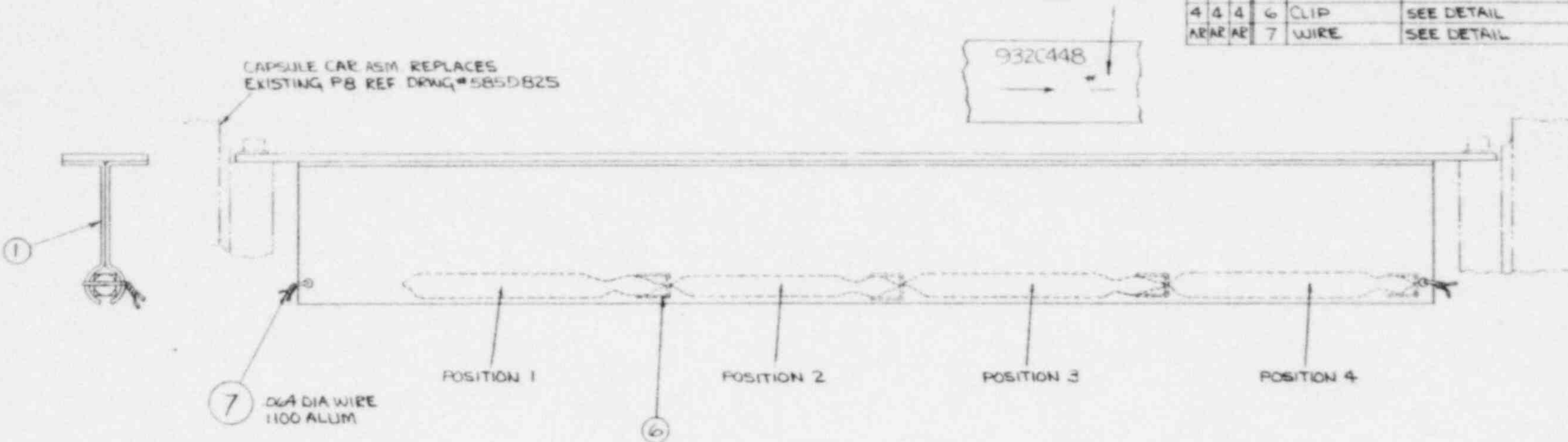
CAPSULE CARRIER ASM

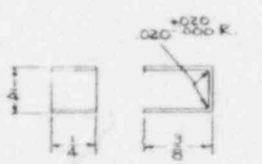
FIRST MADE FOR CEP

UNLESS OTHERWISE SPECIFIED USE THE FOLLOWING

APPLIED PRACTICES	SURFACES	FRACTIONS	DECIMALS	ANGLES
	✓	± 1/64	±	±

G3	G2	G1	PT NO		
1	1	1	1	CAP. CARRIER	932C448 G1
1	1	1	2	SULPHUR CAP	APED VAL CHEMISTRY
1	1	1	3	COBALT CAP	
1	1	1	4	NICKEL CAP	
1	1	1	5	MAG CHLOR CAP	APED VAL CHEMISTRY
4	4	4	6	CLIP	SEE DETAIL
AR	AR	AR	7	WIRE	SEE DETAIL

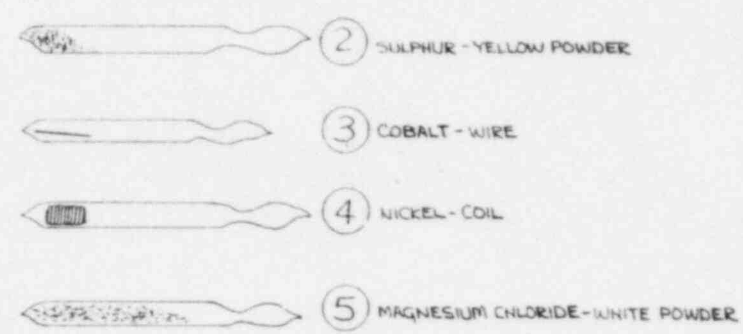




6 .020 THK
1100 ALUM
ANNEALED

POOR ORIGINAL

	POS 1	POS 2	POS 3	POS 4
G1	P2	P3	P5	P4
G2	P5	P3	P4	P2
G3	P4	P3	P2	P5



NOTES:

1. THE SULPHUR, COBALT, NICKEL & MAG CHLORIDE FILLED QUARTZ CAPSULES ARE SUPPLIED IN SETS OF FOUR CAPSULES, ONE FOR EACH MAT'L IN SERIES. FOR EXAMPLE SET#1 CONSISTS OF FOUR CAPSULES ONE SULPHUR, ONE COBALT, ONE NICKEL, & ONE MAG CHLORIDE
2. EACH SET OF FOUR CAPSULES SHALL BE INSTALLED AS INDICATED ONE SET PER EACH P1
3. SCRATCH CORRESPONDING SET NO. ON P1 IN SPACE PROVIDED AFTER ASM. OF CAPSULES.

REVISIONS | PRINTS TO

REV NO. 0

932C449

MADE BY C. Williams APRIL 27 1960
[illegible] MAR. 16 '61

APED SAN JOSE 932C449