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1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

VOLUME I

U.S. ATOMIC ENERGY COMMISSION
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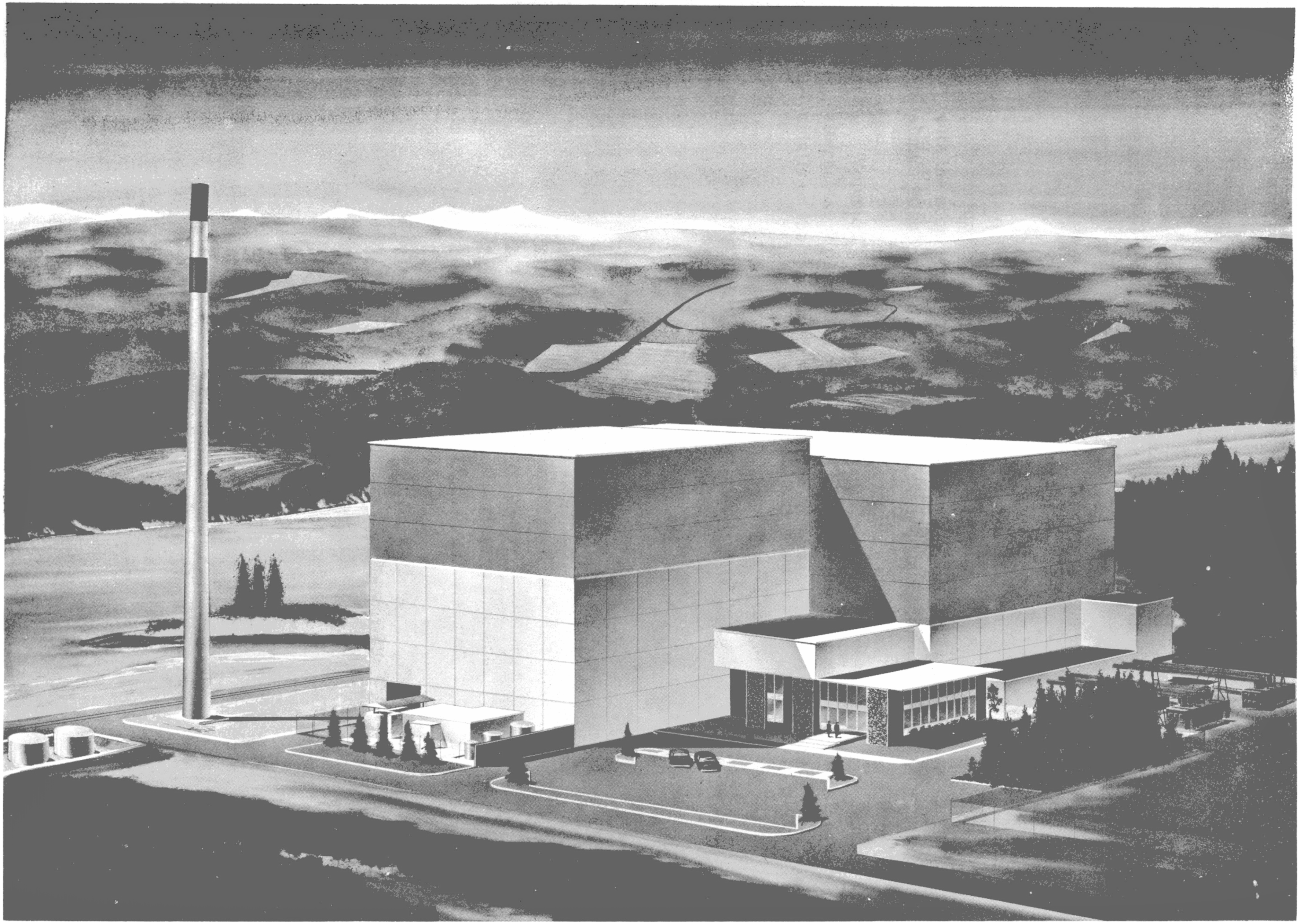
PROJECT AGREEMENT 36

ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA



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VOLUME 1

INTRODUCTION

This volume is a summary report of the 1000 MWe Boiling Water Reactor Plant Feasibility Study performed by the General Electric Company. Included in the volume are chapters dealing with the scoping and ground rules, the design criteria, future development potential, and conclusions. Also included are summaries of the plant economics and the equipment descriptions.

Volume 2 contains a detailed technical description of the Reference Plant and the various alternates. Volume 3 is composed of a complete set of drawings, as well as appendices presenting details on the reactor core calculations, the vessel fabrication plans, and the plant costs.

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CHAPTER I

GENERAL

This is the report of a study performed by the Atomic Power Equipment Department of the General Electric Company for the United States Atomic Energy Commission under Contract AT(04-3)-189, P. A. No. 36, effective June 1, 1963. The study was commenced in July, 1963 and finished in February 1964.

1. AUTHORIZATION AND OBJECTIVE

Under the contract, the General Electric Company was to study the technical and economic feasibility of a 1000 MWe Nuclear Power Plant, using one Boiling Water Reactor, and to compare this concept with a 1000 MWe plant utilizing two reactors of equal rating.

Of particular significance was the criterion, expressed in the project agreement, that the design and construction of the plant should be scheduled to permit startup in 1968. With such a schedule limitation, it was necessary to establish design parameters and to select a plant which could be warranted for the specified rating without requiring any effort on research and development.

The largest single-unit BWR plant currently on order is of 515 MW rating, and the largest twin-unit plant is 380 MW; therefore, it was considered that the 1000 MW plant would require a significant up-scaling of major system components and an extrapolation of technology in some areas. Detailed studies were made of such critical items as the reactor core, turbine, generator, pressure vessel and recirculation pumps in order to establish feasibility and meet the criteria of the project agreement.

2. SCOPE

The scope of this study included a comparative evaluation of numerous components and of several configurations for the various systems involved. A variety of site plans and building arrangements were studied, including five physical layouts of the twin turbine-generators.

Major equipment feasibility investigations were carried out by preparing detailed specifications and making special efforts to establish direct contact with the appropriate suppliers.

Standard components were specified for the reactor core, and numerous computer runs were made in order to establish that all safety margins and stability criteria were met, as well as to assure that the core was adequate to produce the required power at specified exposure levels.

Aside from the comparison of identical plants utilizing single or twin reactors, these plants were also compared with identical plants incorporating steam reheat in the turbine cycle. Thus four alternate plants were completely studied:

Arrangement A. Single Reactor Non-Reheat Cycle (Reference)
Arrangement B. Single Reactor Reheat Cycle
Arrangement C. Twin Reactor Non-Reheat Cycle
Arrangement D. Twin Reactor Reheat Cycle

To permit a realistic comparison of the single and twin-reactor concepts and the reheat and non-reheat cycles, all other plant equipment and parameters were made as nearly identical as possible in all four plant arrangements.

All plants are rated 1000 MWe (net), and use single-cycle boiling water reactors with internal steam separation and pressure-suppression containment. All cores have identical lattice and moderator-to-fuel volume ratio, control elements, fuel rod diameter and length, total water flow, and structural materials. In all plants the reactor steam is saturated at a pressure of 1015 psia, and the condenser vacuum is 1.5 inches Hg abs, at average river temperature.

In addition to the above, certain factors are specified as identical for the two non-reheat plants (A and C), and for the two reheat plants (B and D). The feedwater temperature, core exit steam quality, and the design of the turbine-generator sets and the condensate and feedwater systems are identical in each pair of plants. This permits a direct comparison between plants having one and two reactors operating on the same steam cycle.

The vessels for the single-reactor plants are identical, although the reheat cycle requires a smaller core than the non-reheat. The same applies to the two cases of twin-reactor plants.

For all four alternate plants calculations were made to establish design of the initial and the equilibrium cores. Fuel cycle costs were estimated for the four equilibrium cores, and the costs of the initial core for the Reference Plant were estimated on three different bases. Capital investment, operation and maintenance costs, and energy costs were estimated using the appropriate AEC ground rules.

3. GROUND RULES

The contract enumerated the following ground rules to be used in preparing the final report on this project:

1. The bases for site conditions, design conditions and cost estimating will be based on the Atomic Energy Commission's Nuclear Power Plant Cost Evaluation Handbook, March 15, 1962, as revised where applicable and as revised by agreement with the AEC for purposes of this study.

2. Technical design basis and reactor and power plant concepts are to be applicable to completing a plant design and its construction for 1968 startup.

3. Cost estimating comparisons will provide only a breakdown into major plant systems. The "reference design" (Nuclear Steam Supply System and Balance of Plant) will be supplied as a lump sum price. Cost estimates of the alternate designs of the nuclear steam supply systems will also be presented as a lump sum. Differences will be detailed on a dollars per kilowatt basis as applied to major subsystems and components.

4. ACKNOWLEDGMENTS

We wish to acknowledge the work of Ebasco Services, Inc. of New York, New York who served as the Architect-Engineer for this study. The Architect-Engineer's scope included feasibility studies of alternate arrangements, the preparation of procurement specifications and drawings of the "Balance of Plant", flow diagrams and project schedule. The Architect-Engineer also prepared cost estimates of the equipment, installation and civil works required within this scope.

We wish also to acknowledge the work of Babcock and Wilcox Company and Combustion Engineering, Inc. for their studies of the processes, costs and schedules appropriate to fabrication of the pressure vessel.

GPO 812-316-2

CHAPTER II

DESIGN CRITERIA AND EQUIPMENT FEASIBILITY

In the early stages of this study, particular attention was devoted to the selection of plant design conditions and system parameters. Because of the large plant capacity and, more significantly, the need to determine a design which did not require Research and Development programs – equipment capacity and design limits were major considerations. In addition to the reactor core, the major equipment items which received particular review of technical feasibility and economics of application were the reactor pressure vessel, reactor coolant pumps, turbine, generator, and reactor containment.

1. SELECTION OF PLANT DESIGN CONDITIONS

The net electrical output rating of the plant was established by the study contract at 1000 MW net. Design criteria were based on the turbine operating with admission of 965 psia saturated steam, 1.5 inches Hg abs exhaust pressure, and a plant water make-up rate of 0.5 percent. The auxiliary power requirements were estimated as 44,000 kw – therefore, a gross generator output of 1044 MW was fixed as a turbine-generator rating.

Of major significance to the nuclear steam supply system equipment selection and the plant energy cost is the coolant flow rate. This parameter directly establishes the reactor vessel and containment size, and the selection of the number and capacity of the recirculation pumps. A wide range of feedwater temperatures and related steam requirements were evaluated with respect to the resulting cycle efficiency and the effect on equipment cost and feasibility. Following are some of the more important variables studied in determining the design conditions of the plant:

Major Equipment Cost Items:

- Reactor Vessel Diameter
- Number of Pump Loops
- Capacity of Pump Loops
- Containment Size

Plant Performance and Operating Costs:

- Cycle Efficiency
- Coolant Pumping Power

System Parameters Which Determine Coolant Flow:

Feedwater Temperature
Turbine Steam Rate
Core Exit Steam Quality

The selection of the core coolant flow utilized a study of the plant variables as illustrated by Figure II-1.

The reactor recirculation flow rate requirement changes as a function of the selection of feed temperature and core exit steam quality. Higher flow rates result from a choice of higher feedwater temperatures. Although all points on the curves satisfy design criteria and limits, the zone of optimum economic selection occurs between points A and B.

This optimum design zone establishes a range of preferred values for feedwater temperature, core exit steam quality, core coolant flow, and other thermal-hydraulic parameters. The design point shown by the circle was selected at a relatively low feedwater temperature. At the lower feedwater temperature, the resulting lower equipment cost justified the sacrifice in heat rate, based on the predicted fuel cost and economic assumptions of the plant design. Pump feasibility and cost studies showed that six pump loops, each with 50,000 gpm capacity, gave the best plant economics, and that the design point fell well within the preferred zone limits.

The core coolant flow rate of 114,000,000 #/hr permitted the use of only six recirculation loops, utilizing pumps of a capacity within the capability of major pump suppliers. Six loops provide a considerable advantage over a layout with more loops, from the standpoint of containment design and of reactor vessel nozzle placement. The following plant design selections were made for the single reactor non-reheat cycle plant (arrangement A) based on the procedure illustrated by Figure II-1.

Core Coolant Flow Rate	114,000,000 lb/hr
Number of Recirculation Pumps	6
Pump Capacity	50,000 GPM
Feedwater Temperature	290.6 F
Net Efficiency	31.31%
Reactor Vessel Diameter	21 ft. 10 in.

2. REACTOR CORE

The feasibility of the core lattice is assured because the choice of fuel rod size, lattice configuration, and control element design and spacing are backed by operational experience in the Dresden, KAHL, and Humboldt Bay plants.

Design and operating experience with the 200 MWe Dresden station, which has a large low-leakage core (approximate leakage 3 percent), renders physics extrapolations to the 1000 MWe core (approximate leakage 1.5 percent) less severe than would appear from the factor of 5 extrapolation in power. Previous experience with the problems associated with power distribution in large decoupled cores gave considerable confidence in the ability to design a feasible 1000 MW core. Compliance with required stability limits and criteria gave further assurance that the 1000 MWe would meet its design objectives.

Advanced design concepts, such as increased control-element pitch, larger and higher reactivity-worth fuel bundles, soluble poison for cold shutdown, and programmed control-element sequencing have been avoided even though the incorporation of such concepts might make the system appear to have some economic advantages. Throughout the core design, an attempt has been made to impose design criteria and limits which will insure a feasible, safe, operable, and reliable core.

The following criteria were applied in the core design.

2.1 Design Exposure

The design exposures are 16,500 MWD/MTU and 22,000 MWD/MTU for the initial and equilibrium cores, respectively. Initial uranium enrichments were determined to meet these exposures, consistent with the fuel management plan adopted.

2.2 "End-of-Life" Definition

The end-of-life of the core is attained when there is 0.01 Δk remained in the inserted control elements, which serves as a margin for maneuvering the power distribution and for xenon transients during load following.

2.3 Stuck Control Element Criteria

The reactor is subcritical both hot and cold, with any one control element completely withdrawn, during any time of core life.

2.4 Limited Fuel Bundle Worth

Core safety is enhanced by a limitation of the maximum bundle size. The use of a scatter-pattern refueling scheme diminishes the fuel bundle reactivity worth.

2.5 Minimum Burnout Ratio

The minimum burnout ratio at any point in the core in all designs is greater than or equal to 1.5. This means that the actual heat flux at any point is less than 67 percent of the critical heat flux under conditions occurring at that point. The ratio is applied to the worst postulated design transient and the 120 percent overpower condition. The critical heat flux and its dependence on variables such as steam quality and mass flow rate is determined by experimental data obtained at the General Electric Heat Transfer Facility. Selection of the design limit lines as a line drawn beneath all data points, rather than a least squares fit to all points, results in additional design conservatism.

2.6 Maximum Fuel and Cladding Temperatures

The fuel rod heat flux is limited to a value which assures that the maximum fuel temperature is below the melting point of UO_2. The fuel rod plenum is sized such that the internal pressure from fission gases does not result in cladding hoop stresses which exceed 50 percent of the cladding ultimate strength at any time during core life. The maximum cladding temperature is selected to be less than the temperature which would cause accelerated ("break-away") corrosion of zircaloy.

2.7 Reactivity Coefficients

Core safety, stability, transient response, and system stability depend upon the interaction of the moderator temperature and void coefficients, the Doppler fuel temperature coefficient, and the power coefficient. These coefficients are determined by proper choices of fuel rod diameter, water-to-fuel ratio, and hydrogen-to-uranium ratio, such that:

—the temperature coefficient is negative when the reactor is hot;

—the power coefficient is negative for the minimum critical configuration in the cold condition;

—the power coefficient is sufficiently negative to prevent spatial Xenon oscillations for the given core size;

—the power coefficient is sufficiently small to provide coupled nuclear-hydraulic stability and minimize pressure variation pertubations; and

—the calculated moderator void coefficient and power coefficient assure operating system stability, spatial Xenon stability, good load-following response, and a safe response to a transient such as a turbine tripout.

2.8 Power Peaking Allowances

The design includes power peaking allowances (calculated by means of 1, 2, and 3 dimensional reactor analysis codes) which provide sufficient margin to allow for the worst operating states which the core will experience throughout life. These power peaking allowances are:

Radial power density distribution factor	1.30
Axial power density distribution factor	1.57
Inter-control element (azimuthal) power peaking factor	1.15
Local "hot-spot" peaking factor	1.30
Overpower operation allowance	1.20

Actual operating experience and continued comparison with operating data confirm, to the greatest extent possible, the validity of the above approach.

2.9 Selection of Conservative Physics Constants

Basic physics constants, such as the eta value for Pu-239 and the fission-product cross sections, in which there are at present wide experimental uncertainties, are taken slightly on the conservative side to insure a safe margin in calculated reactivity life and fuel cycle costs, and to assure that the cores will meet their stated design objectives.

2.10 Minimum Fuel Cycle Costs

Within the above design criteria, the selection of such parameters as water-to-fuel ratio, cladding-to-uranium ratio, and hydrogen-to-uranium ratio is determined by that combination of parameters yielding minimum fuel cycle costs coupled with optimum performance and stability.

2.11 Stability Design Criteria

The stability design criteria are:

1. The steady-state power will not oscillate in time.
2. The spatial reactor power will not oscillate enough to cause violation of fuel thermal limits.

3. Well damped power behavior will occur following a pertubation caused by control element movement, flow change, load change, etc.

4. Sufficiently low noise level will result from hydrodynamic-nuclear effects to prevent interference with in-core instrument indications.

3. REACTOR VESSEL

The reactor pressure vessel selection, design specifications and its fabrication feasibility is covered in detail in the Appendix. Of the four plant alternates studied, the single reactor non-reheat cycle required the largest reactor vessel. The fabrication of the vessel was found to be entirely feasible even though the vessel would be 21 feet 10 inches inside diameter with an 8-3/4-inch base metal wall, and would weigh approximately 800 tons. There is no portion of this vessel design with which there has not been prior experience in technique of fabrication. Vessel fabricators have formed, welded, and satisfactorily heat treated plate sections into cylindrical courses in greater thicknesses than 8-3/4-inches. The vessel flange section would be fabricated as three or four segments and welded into a continuous ring. The vessel heads would be fabricated by standard orange peel and formed dome construction. Some vessel fabricating plant equipment is adequate at the present time, and some is being modernized this year, so that several fabricators will be completely equipped to produce this vessel in the fall of 1964.

4. REACTOR WATER RECIRCULATION PUMPS

The six coolant recirculation pumps are each rated at 50,000 gpm and 100 feet total head. Each pump requires 1100 operating horsepower so that at rated load, the total pump power required is about 1/2 percent of the gross electrical output of the plant. The pumps are of vertical centrifugal type utilizing mechanical shaft seals. The seal was designed to provide maximum reliability and minimum maintenance for service in large central station power plants. Pumps with similar "low pressure per seal" arrangement are now in operation. Three manufacturers responded to the pump specification and quotation request. The selected pumps proved to be economically attractive compared to lesser-capacity units.

5. TURBINE-GENERATOR SET

The turbine-generators were carefully selected as representative of today's technology. They incorporate many components on which successful operating experience has been acquired. The rating of 522 MW on a single shaft is deemed a reasonable extrapolation from units presently in service and is comparable to units now on order and in the process of design and manufacture.

The economics of nuclear power combine a relatively high investment component of generation cost with a low fuel cost component. High plant availability and reliability are required for economic success. Steam turbine-generators for nuclear applications, therefore, should be designed to equal the availability and reliability standards for fossil fueled units. Significant extrapolation from successful designs, with regard to size, rating, stress, etc., does tend to increase the difficulty in achieving desired reliability levels. Efficient progress is made by limiting growth rates to prudent steps, with each increase made only after evaluation of the experience arising from the previous extension. The largest 1800 rpm single-shaft turbine-generator in service generates 350 MW. It is twice as large as any other in service in the industry. Units rated 400 MW are scheduled to be shipped in 1964. The largest tandem unit on order is rated 620 MW, for 1968 service. For these reasons, it is not prudent to consider 1000 MW single-shaft units at the present time. In the future, as successful experience is gained at the 600 MW level, it may be reasonable to consider reaching out to 800 or 900 MW, and later to 1000 or 1200 MW.

6. GENERATOR AND EXCITER

Historically, the growth of large turbine-driven generators has occurred in many steps of various magnitudes to meet particular needs of the utility industry. In general, these steps have been anticipated sufficiently to permit following orderly procedures of design, development, prototype production, and service evaluation. This type of approach is prudent to insure a high level of availability and reliability. At this time such requirements can be met with two generators, each rated 640,000 kva. A single unit rated 1,280,000 kva involves an abnormally large step in capacity and, therefore, is not consistent with trouble-free progress.

The two largest 1800 rpm generators in service have a capability of 391,111 kva each. A third unit capable of 527,778 kva recently completed factory running tests. One generator rated at 755,000 kva is on order for a nuclear installation.

Of particular importance in sizes beyond those listed, is the cyclic force due to current flowing in the stator coils, which tends to cause vibration of the insulated conductors in the slots. Methods of suppressing this vibration must be evaluated for various currents and such testing inherently involves long periods of time.

Extension of experience in the pouring of large ingots and in forging them into shafts will insure the availability of a high-quality forging for the one-piece body portion of the rotor.

Valuable information is obtained by running tests on each new and larger generator to confirm electrical, thermal, and mechanical calculations. Included in the latter item is the evaluation of stiffness of bearing supports and effect on critical-speed vibrations.

The first 1800 rpm, gap pickup conductor-cooled rotor for a generator rated 232,000 kva has been shipped, but is not yet in service. Further design, manufacturing, and service experience is desirable.

At the 640,000 kva level, excitation is only slightly above that available from gear shaft-driven exciters already shipped and is below that to be supplied on the existing order of a 755,000 kva unit. At 1,280,000 kva, the excitation is approximately 4700 kilowatts. As a single unit, this would constitute an abnormally large increase (more than 100 percent) over present experience with exciters. Terminal voltage may increase up to a possible level of 28 kv. Present experience includes voltages up to 24 kv. Application of the higher voltage will dictate sufficient evaluation of all insulated areas to insure ability to satisfactorily withstand the short-time effects of test voltages and long-time effects of normal voltage, such as slot discharge and surface corona.

7. LAST-STAGE BUCKETS OF TURBINE

Moisture erosion in low-pressure turbines is an exponential function of bucket tip speed, and is also related to moisture content. General Electric has had successful non-reheat operating experience with 38-inch last stage buckets where a low-pressure corrugated plate moisture separator was provided. Non-reheat units are on order, with 43-inch last stage buckets, and similar moisture separators will be provided. It is deemed unwise to consider offering the next non-reheat step, the 52-inch bucket, until service experience is acquired with the 43-inch bucket.

The reheat cycle leads to turbine designs in which the expansion from the reheater to the condenser takes place in an eight- or nine-stage six-flow span. Experience has been obtained on similar designs with 38-inch buckets under conventional steam conditions, but not with the longer last-stage buckets. Hence, the 38-inch last-stage bucket was chosen for this design study.

8. REACTOR CONTAINMENT

The primary containment selected for the reactor vessel and recirculation system is of the pressure-suppression type. This eliminates the necessity for a large dry capsule or sphere, and would assure suppression of the liberated energy, and containment of the released activity resulting from an accidental core coolant loss or water circuit rupture. Post incident pressures would be small and thus tend to eliminate the driving force for fission products attempting to escape through any leak in the drywell barrier.

This concept of containment was applied to the Humboldt Bay plant which is now in successful operation.

The drywell is not unusual from a construction standpoint, being similar in many respects to the reinforced concrete silo-like structures which are common in this country. Locks, pipe penetrations, etc., are of basically conventional design.

The reactor building itself, surrounding the pressure-suppression system, acts as a secondary containment. Fission products escaping through leakage in the drywell would be subject to decay in the building atmosphere and deposition on building surfaces.

An emergency ventilator maintains a slight vacuum in the building subsequent to an accident and discharges air via high efficiency particulate and charcoal filters to the plant stack. Any reactor building leakage would, therefore, be infiltration rather than exfiltration at ground level. The charcoal filter is included specifically to remove radioactive iodine.

This innovation reduces environs exposure to insignificant levels even if the drywell leak rate should substantially exceed the design and test leak rate.

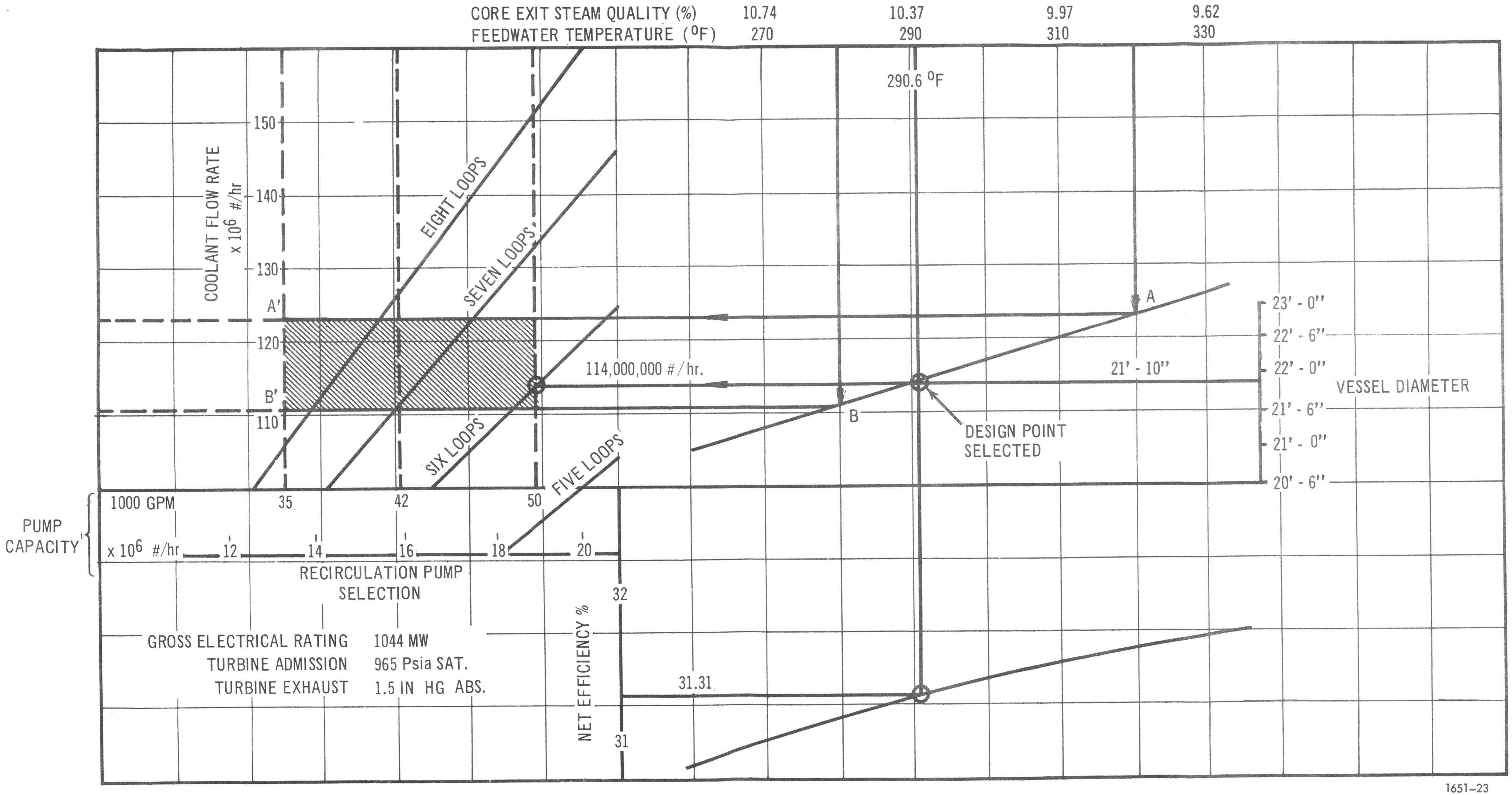

Figure II-1. Design Conditions and Equipment Selection

CHAPTER III

CONCLUSIONS

The present study definitely establishes the feasibility of a 1000 MWe Boiling Water Reactor Nuclear Power Plant, utilizing either one or two reactors and either the reheat or non-reheat steam cycle. All required equipment items are of the types in the current product lines of manufacturers. The 1000 MWe pressure vessel would be fabricated from steel of a thickness no greater than that utilized in vessels of several nuclear plants which are now in operation. No research or development would be required by any of these four plants. The following chapter does, however, describe an Advanced Boiling Water Reactor Plant concept, and the research and development program which it would require.

The core designs were based on minor extrapolations from the cores of operating reactors. A good water-to-fuel ratio was not sacrificed for the sake of compactness; and numerous calculations and computer runs showed the cores to be inherently stable and safe, with good load-following characteristics. The design work revealed no particular problems with any of the cores.

The largest turbine-generator in service (of the type required by nuclear plants) is rated at 350 MWe, and the largest one on order is 620 MWe. For the sake of realism, the 1000 MWe plants in this study were all designed with twin 500 MWe turbine-generators, even though this incurs a cost penalty compared to what would be presumed to be the cost of a single 1000 MWe turbine-generator when such a size becomes available.

Cost and economic summaries are presented in the following Tables. Greater detail may be found in the Appendix.

Table III-1 shows a summary of the construction costs for the four plants. It will be noted that the twin-reactor plants cost slightly more than the single-reactor equivalents. The additional cost of reactor-plant equipment in the twin-reactor plant is partially compensated by the smaller interest during construction which results from the shorter schedule. Most of the indirect construction costs are determined, in accordance with the AEC ground rules, as specified percentages. These percentages, for the single Reactor Non-Reheat Plant (Arrangement A) are as follows: general and administrative: 6.2%; miscellaneous construction costs: 1.0%; engineering, design, inspection, and Architect Engineer services: 11.1%; nuclear engineering: 3.6%; Contingencies: 10.0%; and interest during construction: 14.2%.

Total direct construction costs for Plant A are estimated as $97 per kilowatt. Adding indirect construction costs on the above basis, plus startup costs, land, and land rights, brings the total plant cost to $151.6/kw. In an actual commercial application of the 1000 MWe BWR power plant, the plant price would be determined using appropriate indirect costs applicable to the specific situation.

Fuel cycle costs are summarized in Table III-2. For the initial core of the Reference Plant only, estimates of the fuel cycle cost were made using three different sets of assumptions:

1) Prior AEC. Costs and procedures as listed in the Guide to Nuclear Power Cost Evaluation, U. S. Atomic Energy Commission, with revisions as of April 30, 1963.
2) 1969 AEC. Expected costs based on current AEC announcements.
3) 1969 Private. Same as "1969 AEC" except that private supply and ownership of the uranium is assumed, based on that option being available.

Assumption (1) was included primarily for reference purposes. Assumptions (2) and (3) lead to lower fuel cycle costs, reflecting the cost decreases, in the next five years, resulting from a growing fuel market as well as from improved technology and procedures.

Private financing of uranium was assumed to be at a rate of 10 percent per annum, compared to government financing at 4.75 percent. This factor partially offsets the lower price of uranium on the free market, which was assumed to be \$4.00 per pound of U_3O_8, as compared to government-owned uranium with a price of \$6.00/lb.

Thus the fuel component of energy cost estimated on the "1969 Private" basis is 0.030 mills per kilowatt-hour lower than that on the "1969 AEC" basis.

Fuel cycle cost estimates of the equilibrium cores for all four plant alternates are also presented in Table III-2. It is assumed that the cores will be fabricated and used in 1979, and that at that time private purchase and ownership of the fuel will be mandatory. Significant decreases are projected in the costs of the various chemical processes, the enrichment process, and in fuel fabrication, offsetting the increase in uranium price. Thus the fuel component of the energy costs estimated for 1979 is about 0.4 mill per kilowatt-hour lower than for the initial core in 1969. Part of this difference is due to the increased burnup of the equilibrium core (22,000 MWD/MTU) compared to the initial core (16,500 MWD/MTU).

Total annual costs and energy costs at equilibrium are summarized in Table III-3. It will be noted that the twin-reactor plants show a slight disadvantage compared to their single-reactor counterparts (C compared to A, and D compared to B). As required by AEC ground rules, power generation costs for all plants were estimated on the basis of an 80 percent load factor. It is probable, however, that the twin-reactor plants could operate at a higher factor than the equivalent single-reactor plants, because refueling time is 29 percent less for the twin reactors, and also a twin plant should be able to operate at half capacity during some types of forced outages which would require complete shutdown of the single-reactor plant. A load factor of 83 percent would make energy from Plant C cost 4.91 mills/KWh, and from Plant D cost 5.03 -- slightly better than the equivalent single-reactor plants operating at 80 percent load factor.

The plants have an energy cost between 4.93 and 5.16 mills per kilowatt-hour, which would make them competitive in many sections of the United States as well as abroad.

The number of major components in a large power plant has an effect on the total cost of power generation. An investigation of the capital investment portion of the cost of generation found that an arrangement similar to Plants C and D with twin reactors and twin turbine-generators has the lowest long-term investment cost. The arrangement similar to Plants A and B with a single reactor and twin turbine-generators requires a higher long-term investment, and an arrangement using a single reactor and a single turbine-generator requires the highest investment of all. The investment study covered a 19-year expansion; and present worth was calculated, assuming the same price for all plant arrangements. Details of the study are presented in Appendix V.

It was assumed that a given system was expanded between 1968 and 1971 through addition of units having one or another of the aforementioned arrangements. The system was then expanded from 1972 to 1986 using only twin-reactor twin T-G units. Thus, the immediate effect (on present worth) of the various units installed in the first four years was evident.

Assuming a forced outage rate of 0.03, the advantage of the twin-reactor twin T-G expansion over the single-reactor twin T-G expansion was $54 million, and the advantage over the single-reactor single T-G expansion was $84 million. The twin units would have shown even greater advantages if the twin elements had been cross tied, and also if one reactor and T-G of the twin unit were assumed to start operation a year after the other -- system demand permitting.

This study shows that system expansion with twin units instead of single units, has additional long-term investment advantages which are not shown by a simple comparison of plant prices.

TABLE III-1

SUMMARY OF CONSTRUCTION COST ESTIMATE

(Thousands of Dollars)

AEC Acct. No.	Description		Non-Reheat 1 Reactor A	Reheat 1 Reactor B	Non-Reheat 2 Reactors C	Reheat 2 Reactors D
21	Structures and Improvements		10 157	10 615	12 051	12 509
22	Reactor Plant Equipment		53 631	52 693	55 517	54 699
23	Turbine-Generator Plant Equipment		29 364	36 023	29 356	36 015
24	Accessory Electric Equipment		3 321	3 420	3 337	3 438
25	Miscellaneous Power Plant Equipment		585	625	550	590
	Total Direct Construction Cost		97 058	103 376	100 811	107 251
	Indirect Construction Cost					
	General and Administrative	(1)	6 018	6 306	6 149	6 542
	Subtotal		103 076	109 682	106 960	113 793
	Miscellaneous Construction Costs	(1%)	1 031	1 097	1 070	1 138
	Subtotal		104 107	110 779	108 030	114 931
	Engrg., Design and Inspect. A&E Serv.	(2)	11 556	12 186	11 883	12 527
	Subtotal		115 663	122 965	119 913	127 458
	Nuclear Engrg.	(3)	4 164	4 304	4 317	4 461
	Startup Costs		507	507	630	630
	Subtotal		120 334	127 776	124 860	132 549
20	Land and Land Rights		360	360	360	360
	Subtotal		120 694	128 136	125 220	132 909
	Contingency	(10%)	12 069	12 814	12 522	13 291
	Subtotal		132 763	140 950	137 742	146 200
	Interest during Construction	(4)	18 852	20 015	15 840	16 813
	Total Capital Cost		$151 615	$160 965	$153 582	$163 013
	Applied Percentages	(1)	6.2	6.1	6.1	6.1
		(2)	11.1	11.0	11.0	10.9
		(3)	3.6	3.5	3.6	3.5
		(4)	14.2	14.2	11.5	11.5

TABLE III-2

SUMMARY OF FUEL CYCLE COST ESTIMATE

	Initial Core			Equilibrium Core			
Plant Arrangement Cost Assumptions	A Prior AEC	A 1969 AEC	A 1969 Private	A 1979 Private	B 1979 Private	C 1979 Private	D 1979 Private
Fuel Costs in $/kg U charged							
Fabrication	115.00	115.00	115.00	100.00	100.00	100.00	100.00
Uranium depletion	117.20	101.67	86.94	119.82	120.64	121.92	122.13
Use charge on uranium	18.41	15.38	26.39	45.89	44.91	46.42	46.91
Reprocessing*	36.29	34.12	34.04	25.15	25.15	25.16	25.16
Spent fuel shipping	16.58	4.40	4.40	3.30	3.30	3.30	3.30
Plutonium credit	(60.23)	(43.60)	(43.60)	(48.70)	(48.70)	(48.90)	(48.90)
Total	243.25	226.97	223.17	245.46	245.30	247.90	248.60
Fuel Costs in Mills/kwh							
Fabrication	0.928	0.928	0.928	0.605	0.579	0.606	0.579
Uranium depletion	0.945	0.820	0.701	0.725	0.698	0.739	0.708
Use charge on uranium	0.148	0.124	0.213	0.278	0.260	0.281	0.272
Reprocessing*	0.293	0.275	0.275	0.152	0.145	0.152	0.146
Spent fuel shipping	0.134	0.035	0.035	0.020	0.019	0.020	0.019
Plutonium credit	(0.486)	(0.352)	(0.352)	(0.295)	(0.282)	(0.296)	(0.283)
Total	1.962	1.830	1.800	1.485	1.419	1.502	1.441
Working capital for core fabrication	0.267	0.175	0.175	0.152	0.142	0.151	0.144
Total including working capital on fabrication	2.229	2.005	1.975	1.637	1.561	1.653	1.585
Unit electrical energy yield, MWH/kgU	123.99	123.99	123.99	165.32	172.87	165.00	172.55

*Includes chemical separation, conversion and losses.

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TABLE III-3

SUMMARY OF POWER GENERATION COST ESTIMATE

Plant:	(ref.) A	B	C	D
Number of Reactors	1	1	2	2
Steam Cycle	Non-reheat	Reheat	Non-reheat	Reheat
Annual costs ($\$10^3$):				
Capital cost - 14%	21, 176	22, 485	21, 451	22, 771
Land and land rights - 10%	36	36	36	36
Nuclear liability insurance	356	352	356	352
Fuel, including U financing - equilibrium	10,405	9,942	10,530	10,098
Working capital for core fabrication - 10%	1,069	994	1,058	1,013
Operation and maintenance	1, 448	1, 448	1, 801	1, 801
Working capital for operation and maintenance - 10%	51	50	58	57
TOTAL	34,541	35,307	35,290	36,128
Energy costs (mills/kwh): (1)				
Capital cost	3. 022	3. 208	3. 061	3. 249
Land and land rights	0. 005	0. 005	0. 005	0. 005
Nuclear liability insurance	0. 051	0. 050	0. 051	0. 050
Fuel, including U financing equilibrium	1. 485	1. 419	1. 502	1. 441
Working capital for core fabrication	0. 152	0. 142	0. 151	0. 144
Operation and maintenance	0. 207	0. 207	0. 257	0. 257
Working capital for operation and maintenance	0. 007	0. 007	0. 008	0. 008
TOTAL	4. 93	5. 04	5. 04	5. 16

(1) Assuming 80% load factor

CHAPTER IV

FUTURE DEVELOPMENTS

Selected design features and underlying technical bases for the 1000 MWe boiling water reactor, described in other sections of this report, are appropriate for immediate offering and start of construction in 1964. The design of the reference BWR plant and of all alternate arrangements is based on current technology and requires no substantial new development work to achieve the 1000 MWe rating within the time schedule. Thus, the reference plant design has the advantage of being immediately available for construction in sizes up to and including 1000 MWe. However, this design does not fully reflect the extent of additional development potential for design and performance improvement available for the boiling water reactor.

Several anticipated design advances are closely inter-related and are considered in combined form as a single design concept, the 1000 MWe Advanced Boiling Water Reactor (ABWR). This is described in Section 1 below.

Aside from this, new advances in steel technology indicate that in the near future, satisfactory pressure vessels can be built with wall thicknesses which are considerably less than those currently required for identical service. These steels are discussed in Section 2 below.

1. THE ADVANCED BOILING WATER REACTOR

The 1000 MWe ABWR is conceived as a higher power density, single-cycle forced circulation BWR plant providing substantially reduced capital and net power costs compared with the reference 1000 MWe BWR plant, Arrangement A. These economies will be made possible by a combination of advanced features which result in substantially reduced coolant flow, reduced reactor vessel size, fewer control rod drives, fewer and smaller external recirculation loops, a generally smaller and more compact shielding, containment and building arrangement, and improved nuclear and mechanical performance of the fuel. It is expected that these improvements can be applied to all power ratings of boiling water reactors, over the range from 100 MWe through 1000 MWe.

The ABWR depends on successful accomplishment of an associated program of development effort in a number of areas of reactor design. The estimated cost of this development program, upon which the ABWR design would be based, is of the order of $5,000,000.

The ABWR concept, as described briefly in the next section, is intended to be appropriate for start of construction in 1966 and demonstration of full-power rating of 1000 MWe in 1970, with the underlying development program assumed to be started in 1964. Since varying degrees of development effort and technical risk are involved in realization of each of the separate design features described, it is anticipated that modified versions of some of them may be included in new BWR designs offered on earlier schedules than assumed for the ABWR concept.

1.1 Advanced Design Features

Principal advanced design features assumed for the 1000 MWe ABWR, which will provide the expected improvements in technical and economic performance are as follows.

A. Higher optimum power density and fuel specific power (25 to 30 percent increase), to be brought about, with no significant change from present fuel rod sizes, by a combination of improved thermal design criteria, improved lattice designs giving smaller local power peaking, and improved gross power distribution resulting from more reliable calculation methods and improved in-core flux measurement.

B. Improved control system, featuring an optimum combination of:

   – Burnable poisons for partial compensation of fuel cycle burnup.

   – Mechanical control rods in reduced number (40 to 50 percent reduction in number) for shimming and fast scram.

   – Separation of control rods into "shim" type and "fast scram & shim" type with corresponding simplification of drive design and cost reduction, as permitted.

   – Coolant flow rate variation for automatic load following (assumed in reference 1000 MWe BWR design).

C. Substantially reduced flow rate per unit power (40 percent reduction), to be obtained by improvement of the critical heat flux and stability design criteria, by development of fuel designs incorporating special design features which raise the steam quality at which the critical heat flux occurs, and by use of lattice designs which provide reduced boiling flow fraction and reduced steam void reactivity coefficients.

D. Reduction of the external recirculation system to two loops and use of internal jet pumps for plant sizes up to 1000 MWe, compared with six loops required for the reference 1000 MWe plant designed according to present day criteria. This is made possible by the reduced coolant flows provided, together with the use of internal jet pumps.

E. Reduction of the reactor vessel to a size no larger than necessary to house the core and internal jet pumps (two to three feet reduction in vessel diameter). This is made possible by the reduced coolant flow and modification of the internal steam separator equipment to exploit the higher core exit steam quality.

F. Improved building arrangement and containment design to fully exploit the cost reductions in this area made available by the smaller reactor vessel and reduced external coolant circulation system employed.

G. Improved fuel design which, in addition to being capable of 22,000 MWD/MTU average discharge exposure at the higher heat fluxes employed, provides:

- —Special heat transfer and fluid flow devices which will permit a reduction in coolant flow per unit thermal power, or an increase in exit steam quality.
- —Suitable and economic burnable poison carrier devices, to partially compensate for the reduced number of mechanical control rods employed.
- —Near-optimum placement of non-boiling moderator in the lattice cell, to reduce local flux peaking and to reduce steam void reactivity defect and void coefficients of reactivity at the higher steam qualities employed.
- —Increased facility of channel recovery and component disassembly, to reduce down-time requirements associated with refueling.

The 1000 MWe ABWR plant incorporating these anticipated advanced design features in near-optimum combination would employ a single reactor operating at 1000 psi pressure in a direct, single, non-reheat cycle using approximately 291 F feedwater return temperature. The plant would be rated at approximately 1000 MWe net electrical power with a reactor thermal rating of 3,200 MWt, providing a net cycle heat rate of approximately 10,920 Btu/kw-hr.

Table IV-1 summarizes typical design and performance parameters estimated for the 1000 MWe ABWR, in comparison with the reference 1000 MWe BWR. These design parameters are premised on majority success in achievement of the goals of the underlying development program on which the ABWR would be based. The level of technical risk involved and magnitude of associated development work required is believed to be appropriate for plants of advanced

TABLE IV-1

COMPARISON OF BWR DESIGN PARAMETERS
(Plant Ratings of 1000 MWe)

		Advanced BWR Plant (1970 Operation)	Reference BWR Plant (1968 Operation)
Type of cycle		Single, non-reheat	Single, non-reheat
Net electrical power	MWe	1,000	1,000
Reactor thermal power	MWt	3,200	3,194
Reactor pressure	psi	1,000	1,000
Feedwater temperature	F	290.6	290.6
Average core exit steam quality		0.18	0.10
Coolant flow	10^6 lbs/hr	65	114
Maximum heat flux (peak trans.)	Btu/hr-ft^2	420,000–450,000	399,000
Core power density	Kw/liter	42–50	35.4
Fuel specific power	Kw/Kg-U	18–21	15.3
Ave. moderator steam vol. fraction		0.27–0.30	0.27
Number of control rods		150–180	376
Control rod pitch	inches	12–13	10
Active fuel length	inches	144–150	144
Fuel rod diameter	inches	0.48–0.55	0.55
Channel coolant/moderator vol. fract.		0.55–0.60	0.7
Reactor vessel inside diameter	feet	19'-20'	21'–10"
Reactor vessel wall thickness	inches	3-7/8	8-3/4
Number of external flow loops (size)		2 (24-in.)	6 (32-in.)
Type of pumps		Internal jet pumps + 2 Centrifugal	6 Centrifugal
Nuclear Control:			
–Burnable poison		–Initial & Equil. cycles } in optimum combination	–Temporary poison for initial load
–Control rods		–50% "Scram & Shim" rods 50% "Shim" rods } in optimum combination	–100% "Scram" rods
–Automatic load following		–Flow control	–Flow control
Equilibrium fuel cycle exposure	MWD/T-U	20,000	20,000
–Batch size for annual reloading		25–30%	20%

design scheduled for start of construction in 1966 with full-power operation in 1970, provided associated development work is initiated in 1964.

1.2 Development Costs

The development tasks and the estimated costs to accomplish the above presented goals are:

Task	I	—Critical Heat Flux Development	$ 663,000
Task	II	—Power Distribution	$ 400,000
Task	III	—Power Stability	$ 652,000
Task	IV	—Burnable Poison	$ 914,400
Task	V	—Flow Control	$ 205,000
Task	VI	—Steam Separation	$ 524,000
Task	VII	—Jet Pump	$ 367,300
Task	VIII	—Fuel Development	$ 600,000
Task	IX	—Advance Design	$ 450,000
		TOTAL	$4,775,700

2. PROSPECTIVE MATERIALS FOR PRESSURE VESSEL

There are four new materials which appear to be promising prospects for pressure vessels over 4 inches thick. Tables IV-2 and IV-3 list the chemical composition of these materials, the anticipated tensile properties, and the allowable stresses in comparison with two current steels. Based on these anticipated properties, the wall thickness for a 21 foot 10 inch ID, 1000 psig vessel is shown assuming that the ASME Section III Code prevails.

The Ni-Cr-Mo steel is probably the most promising at the present time. The U.S. Steel Corporation developed this steel and it is being evaluated by several laboratories and fabricators. Fabrication and welding procedures are being developed and evaluated by Combustion Engineering. This steel with small additions of vanadium has been employed extensively for turbine-generator rotor forgings.

The 2-1/4 Cr-1Mo material in the quenched-and-tempered condition has been used extensively by the Navy and the missile industry for gas receivers. Emanuel and Griffin reported "Properties and Fabrication of High-Strength 2-1/4 percent Chromium-1 percent Molybdenum Materials" in the Welding Research Supplements of the Welding Journal, Volume XXVI, No. 9, September 1962, Pages 393S–399S. An ASME Code Case governing the use of this material is currently being developed by the ASME Boiler and Pressure Vessel Code Committee.

The SSS-100 material was developed by Armco Steel Company and it has been approved for use in thicknesses up to two inches by ASME Code Case 1298. In addition to the benefits of higher strength and the associated benefits of thinner sections (less weight, easier inspection, etc.), these steels also have lower NDT (nil ductility transition) temperatures than do A302B and A336 Modified. Limited results of irradiation tests indicate that the irradiation causes an increase in NDT temperature which is essentially of the same magnitude in these new quenched-and-tempered steels as in A302B. If these results are borne out by further tests, the fact that their initial NDT is significantly lower than for A302B will indicate that these new materials will have a greater margin of safety against brittle fracture in service.

2.1 General Electric Evaluation Programs

The Atomic Power Equipment Department of General Electric is irradiating tensile and impact specimens of quenched-and-tempered 2-1/4 CR-1Mo plates, forgings, and castings and SSS-100 plate materials in the Dresden Reactor. All of the specimens are from material over six inches thick. Specimens of the Ni-Cr-Mo steel will be placed in the Dresden Reactor at the next scheduled shutdown.

2.2 Activity of the Pressure Vessel Research Committee of the Welding Research Council

The PVRC has initiated a program to evaluate materials for pressure vessels over four inches thick. This program is intended to be an extension of current programs of the PVRC Materials and Fabrication Divisions which are studying materials for pressure vessels up to four inches thick. A quenched-and-tempered 2-1/4 Cr-1Mo vessel will be cycled by the Southwest Research Institute as an extension of the current PVRC program.

2.3 Activity of the American Society for Testing Materials

The ASTM has task groups actively engaged in developing specifications for pressure vessel materials over four inches thick, which include the plate and forging materials listed above.

2.4 Activity of the American Society of Mechanical Engineers

The ASME Boiler and Pressure Vessel Committee has been instrumental in initiating and encouraging the activity of both the PVRC and the ASTM and will accept the materials and assign allowable stress intensity values as soon as adequate information is developed. Not only the nuclear industry, but also the petroleum and chemical industries are very interested in these materials for thick-walled pressure vessels.

TABLE IV-2

CHEMICAL COMPOSITION OF PRESSURE VESSEL STEELS

Material	C max	Mn	P max	S max	Si	Cr	Ni	Mo	Other
ASTM A302B	.25	1.15-1.50	.035	.040	.15-.30	--	--	.45-.60	
ASTM A336 modified	.27	.50-.80	.040	.050	.15-.35	.25-.45	.50-.90	.55-.70	
2-1/4 Cr-1Mo	.20	.30-.60	.035	.035	.50 max	2.00-2.50	--	.90-1.10	
Ni-Cr-Mo	.25	.20-.40	.02	.02	.30 max	1.5-2.0	2.75-3.9	.40-.60	
Ni-Cr-Mo-V	.25	.20-.40	.02	.02	.30 max	1.5-2.0	2.75-3.9	.40-.60	V = .05-.10
SSS-100	.12-.20	.4-.7	.035	.040	.20-.35	1.4-2.0	--	.40-.60	Ti or V = .04-.10 Cu = .20-.40 B = .0015-.006

TABLE IV-3

MECHANICAL PROPERTIES AND ALLOWABLE STRESSES FOR PRESSURE VESSEL STEELS

Material	Yield Strength (psi)	Ultimate Tensile (psi)	Elong. (%)	Allowable Stresses at 600F (psi) (ASME Section III)	Wall Thickness of 21'-10" I.D. 1000 psi Vessel (inches)
ASTM A302B	50,000	80,000	20	26,700	6-1/2
ASTM A336 modified	50,000	80,000	20	26,700	6-1/2
2-1/4Cr-1Mo	100,000*	130,000*	18*	40,000**	4-1/4
Ni-Cr-Mo	100,000*	130,000*	18*	40,000**	4-1/4
Ni-Cr-Mo-V	100,000*	130,000*	18*	40,000**	4-1/4
SSS-100	100,000*	130,000*	18*	40,000**	4-1/4

* Anticipated properties

** Assumed values based on anticipated properties

CHAPTER V
PLANT DESCRIPTION SUMMARY

TABLE V-1
PLANT DATA

PLANT ARRANGEMENT**		(Ref) A	B	C	D
Number of Reactors		1	1	2	2
Number of Turbine-Generators		2	2	2	2
Turbine Cycle		Non-Reheat	Reheat	Non-Reheat	Reheat
Net Electrical	(MW)	1000	1000	1000	1000
Gross Electrical	(MW)	1044	1044	1044	1044
Thermal Rating	(MW)	3194	3054	1600*	1530*
Net Cycle Efficiency	(%)	31.31	32.74	31.25	32.68
Net Plant Heat Rate	(Btu/kwh)	10,900	10,420	10,920	10,440
REACTOR					
Operating Pressure	(psia)	1015	1015	1015	1015
Operating Temperature		Sat.	Sat.	Sat.	Sat.
Total Core Flow	(10^6 lb/hr)	114	114	57*	57*
Core Exit Quality	(%)	10.37	10.00	10.37	10.00
Core Entrance Subcooling	(Btu/lb)	28.2	26.5	28.2	26.5
Feedwater Temperature	(°F)	290.6	296.7	290.6	296.7
REACTOR VESSEL					
Inside Diameter	(ft-in.)	21-10	21-10	16-0	16-0
Overall Length (incl. head)	(ft-in.)	64-10	64-10	57-5	57-5
Vessel Weight	(Kips)	1600	1600	793	793
Head Weight	(Kips)	265	265	125	125
Wall Thickness (Total)	(in.)	9	9	6-3/4	6-3/4

* Each reactor

** See description of arrangements in Chapter I.

TABLE V-1 (Continued)

RECIRCULATION LOOPS

Number		6	6	3*	3*
Pump Capacity	(gpm)	50,000	50,000	50,000	50,000
Pump Head	(ft)	100	100	100	100
Motor Rating	(HP)	1250	1250	1250	1250
Power Input to Motor	(KW)	870	870	870	870
Suction Size	(in.)	32	32	32	32
Suction Velocity	(fps)	24.6	24.6	24.6	24.6
Discharge Size	(in.)	30	30	30	30
Discharge Velocity	(fps)	28.0	28.0	28.0	28.0

AUXILIARY POWER

Percent of Gross Electrical	(%)	4.23	4.23	4.23	4.23

TURBINE-GENERATOR

Designation		TC4F-43	TC6F-38	TC4F-43	TC6F-38
Number of Casings		4^x	4^x	4^x	4^x
Speed	(rpm)	1800	1800	1800	1800
Exhaust Pressure	(in. Hg abs)	1.5	1.5	1.5	1.5
Make Up	(%)	0.5	0.5	0.5	0.5
No. of Heater Stages		3^x	3^x	3^x	3^x
Turbine Heat Rate	(Btu/kwh)	10,270	9822	10,270	9822

* = each reactor

x = each turbine generator

1. REACTOR CORE

The nuclear and thermo-hydraulic criteria have been selected to insure safe, stable operation within the limits of maximum cladding and fuel temperatures, maximum fuel bundle and control element worths, and minimum and maximum allowable negative reactivity coefficients. In a large, loosely coupled core of 500 to 1000 MWe, spatial xenon stability, good load following response with the moderator as a control variable, and inherent power distribution self-flattening are important design objectives which were achieved by establishment of the moderator-to-fuel volume ratio of 2.4, fuel rod diameter of 0.555 inch and core exit steam quality of 0.1037. Core safety is achieved by limiting the maximum control element pitch to 10 inches, and maximum fuel bundle size to 36 rods.

Design life is consistent with the energy extraction of 16,500 MWD/metric ton of uranium averaged over the initial core discharge batches, and an equilibrium core discharge exposure of 22,000 MWD/MTU. The "end of a fuel cycle interval" is defined as occurring when there is a total of one percent Δk present in the control elements inserted in the core, with the reactor at full power with equilibrium xenon.

Core data for all plants are tabulated in Table V-2.

TABLE V-2

CORE DATA

1000 MW Boiling Water Reactor Plant Feasibility Study

Plant Arrangement		Reference A	B
Number of Fuel Rods per Bundle		36	36
Number of Fuel Bundles		1528	1420
Fuel Rod O.D.	(in.)	0.555	0.555
Cladding Thickness	(in.)	0.035	0.035
Effective Active Fuel Length	(in.)	144	144
Equivalent Core Diameter	(in.)	220.5	212.6
Circumscribed Core Diameter	(in.)	228.0	221.4
Average Core Power Density	(KW/L)	35.45	36.5
Moderator to Fuel Ratio (Total Core H_2O/UO_2)*		2.4	2.4

* Volume in cold condition

TABLE V-2 (Continued)

Plant Arrangement		A	B
Heat Transfer Area	(ft^2)	95,910	89,130
Fuel Assembly Weight (incl. channel)	(lbs)	473	473
Total Weight of UO_2	(lbs)	520,100	483,400
Fuel Bundle Replacement per Cycle	(%)	20	20
Cladding Material		Zircaloy-2	Zircaloy-2
Channel Material		Zircaloy-4	Zircaloy-4
Number of Control Elements		376	349
Control Element Poison Material		B_4C	B_4C
Number of Temporary Control Curtains		708	656
Number of In-Core Monitors		89 x 4	83 x 4

Plant Arrangement		C	D
Number of Fuel Rods per Bundle		36	36
Number of Fuel Bundles		756	724
Fuel Rod O. D.	(in.)	0.555	0.555
Cladding Thickness	(in.)	0.035	0.035
Effective Active Fuel Length	(in.)	144	144
Equivalent Core Diameter	(in.)	155.1	151.8
Circumscribed Core Diameter	(in.)	165.5	158.1
Average Core Power Density	(KW/L)	35.9	35.8
Moderator to Fuel Ratio (Total Core H_2O/UO_2)*		2.4	2.4
Heat Transfer Area	(ft^2)	47,450	45,450

*Volume in cold condition

TABLE V-2 (Continued)

Plant Arrangement		C	D
Fuel Assembly Weight (incl. Channel)	(lbs)	473	473
Total Weight of UO_2	(lbs)	257.300	246,400
Fuel Bundle Replacement per Cycle	(%)	20	20
Cladding Material		Zircaloy-2	Zircaloy-2
Channel Material		Zircaloy-4	Zircaloy-4
Number of Control Elements		185	177
Control Element Poison Material		B_4C	B_4C
Number of Temporary Control Curtains		340	324
Number of In-Core Monitors		43 x 4	41 x 4

		Arrangement A (Ref)	
		Initial Core	Equilibrium Core
Average Discharge Exposure	(MWD/MTU)	16,500	22.000
Initial Enrichment (Average)	(w/o)	2.01	2.38
U Content of Discharge	(w/o)*	97.62	96.94
Enrichment of Discharged U	(w/o)	0.81	0.82
Total Pu Content of Discharge	(w/o)*	0.634	0.733
Fissile Pu Content of Discharge	(w/o)*	0.436	0.487

*Weight percent of initial uranium

TABLE V-2 (Continued)

		Arrangement B	
		Initial Core	Equilibrium Core
Average Discharge Exposure	(MWD/MTU)	16,500	22,000
Initial Enrichment (Average)	(w/o)	2.02	2.39
U Content of Discharge	(w/o)*	97.62	96.94
Enrichment of Discharged U	(w/o)	0.82	0.82
Total Pu Content of Discharge	(w/o)*	0.634	0.733
Fissile Pu Content of Discharge	(w/o)*	0.436	0.487

*Weight percent of initial uranium

		Arrangement C	
		Initial Core	Equilibrium Core
Average Discharge Exposure	(MWD/MTU)	16,500	22,000
Initial Enrichment (Average)	(w/o)	2.05	2.42
U Content of Discharge	(w/o)*	97.62	96.94
Enrichment of Discharged U	(w/o)	0.84	0.84
Total Pu Content of Discharge	(w/o)*	0.634	0.733
Fissile Pu Content of Discharge	(w/o)*	0.439	0.489

*Weight percent of initial uranium

TABLE V-2 (Continued)

		Arrangement D	
		Initial Core	Equilibrium Core
Average Discharge Exposure	(MWD/MTU)	16,500	22,000
Initial Enrichment (Average)	(w/o)	2.06	2.43
U Content of Discharge	(w/o)*	97.62	96.94
Enrichment of Discharged U	(w/o)	0.85	0.85
Total Pu Content of Discharge	(w/o)*	0.634	0.733
Fissile Pu Content of Discharge	(w/o)*	0.440	0.489

*Weight percent of initial uranium

1.1 Lattice

A lattice utilizing 36-rod fuel bundles was selected, as shown in figures I-1 and I-2. Cruciform control elements occupy the intersecting water gaps between alternate fuel assemblies. Of the intersections unoccupied by control elements, one in four contain in-core flux monitor assemblies. Control of initial excess reactivity of the cold clean core is supplemented by temporary plates or "curtains", as shown in figure I-1. These curtains are removed before the first refueling of the core, and are never again required.

Several selected fuel rods at the corners of each bundle have lower uranium enrichment than the others. This improves the power distribution in the fuel bundles and thus allows a significant reduction in the amount of heat transfer surface required to maintain design power.

The ten-inch control-element pitch is a practical value to minimize individual worth of the elements and to allow adequate clearance between control drive mechanisms below the vessel.

1.2 Moderator-to-Fuel Volume Ratio

To preclude detrimental startup transients, the water-to-fuel volume ratio is specified as large as is allowed by cold lattice coefficients. Stability to xenon and load-following requirements make a large negative power coefficient desirable, whereas a small coefficient would be dictated by considerations of overall system stabi.ity, coupled nuclear-thermohydraulic stability, and the minimizing of perturbations due to pressure changes. Because of the large decrease in water density with an increase in enthalpy, which is characteristic of the boiling process, there is considerable design margin for good load following and spatial xenon stability. Therefore, the power coefficient is made as small as possible within the limits established by the end-of-life cold void coefficient. The water-to-fuel ratio of 2.4 (cold) reduces the hot operating coefficients to a minimum and results in a large system stability margin.

1.3 Fuel Assemblies

Fuel material is UO_2 contained in Zircaloy tubing. Free volume within the rod is sufficient to prevent excessive pressure buildup due to fission gas formation over the design life of the fuel. The rods in each fuel bundle are positioned at intermediate points with spacers and by tie plates at the ends.

The diameter of 0.555 inch of fuel rod is based on economic considerations, consistent with meeting the thermal design criteria on fuel temperature and on margin against fuel rod thermal burnout.

Each fuel bundle is encompassed by a Zircaloy channel which serves several functions. It provides guidance and a bearing surface to the control elements, allows effective water flow control (in combination with orificing of the fuel assembly support grid), and provides mechanical support and protection during fuel handling operations.

1.4 Control Materials

The control elements contain boron carbide granules compacted in Type-304 stainless steel tubes. The tubes are structurally contained and formed into cruciform shape by a stainless steel shroud. End fittings are provided for coupling to the control rod drive and to facilitate handling of the elements.

The temporary poison is provided as metal plates, or "curtains" made of stainless steel alloy material which contains boron in low concentration. The curtain fits between fuel channels, is supported by the upper core grid guide, and is provided with a handle for grappling.

1.5 Design "Hot Spot" Factors

Thermal design of the reactor, including selection of the core size and effective heat transfer area, steam quality, total recirculation flow, inlet subcooling, and internal flow distribution, is based on the concept and application of "hot spot" factors.

The calculated "hot spot" factors express in simple numerical form the design allowance for the combined effects (on the fuel rod heat flux and temperature) of the gross and local steady-state power density distributions, adjustments of the control elements and the reactor power level, power and neutron flux instrumentation errors, and operational transients.

1.6 Core Cooling

Because of economic considerations, the design steam volume fraction is selected to be as high as is believed to be presently feasible, limited by (1) boiling burnout caused by excessive steam formation rates at the fuel rod surfaces, and (2) nuclear and hydraulic stability. Selection of a value for the steam volume fraction provides the basis for the design of the core cooling system.

Correct distribution of core water flow among the fuel channels is established by accurately calibrated replaceable orifices at the fuel channel inlets. The orifices control the flow distribution and, hence, the coolant conditions within prescribed limits throughout the full range of core operation.

1.7 Fuel Management

Fuel management planning is based on reloading 20 percent of the core every 12 months, beginning approximately 27 months after startup. Margin in the capability of the moveable control elements, however, will permit varying the intervals between refuelings through change of the refueling batch size.

The reactor is designed to be refueled in a "scatter-pattern": (1) Depleted fuel is discharged from a one-in-four scatter pattern in the central 80 percent of the core; (2) fuel from the peripheral region in the outer 20 percent of the core is moved into the vacant locations in the scatter region; and (3) fresh fuel is inserted in the vacant periphery. Once the fuel has been moved into the scatter pattern, it is not moved again until discharge.

Based on an 80-percent plant load factor, the initial core will be operated approximately 22 months before removal of the temporary control curtains. Approximately 27 months after startup, 20 percent of the core will be discharged at an average exposure of 11.300 MWD per metric ton of uranium and replaced with fresh fuel. Discharge exposure, as a function of batch number, varies until equilibrium exposure conditions are established. The average exposure of the first five discharge batches (first core) will be 16,500 MWD per metric ton of uranium. Sufficient control capacity is available in the moveable control elements to permit exposures in excess of 22,000 MWD/MTU for equilibrium cores.

1.8 Reactivity Coefficients

The reactivity coefficients of importance in boiling water reactor designs are:

1. The moderator void coefficient
2. The moderator temperature coefficient
3. The Doppler fuel temperature coefficient
4. The over-all power coefficient

The volumes of the core material, primarily water and fuel, are chosen so that:

—The calculated reactivity coefficients in the cold condition would allow a maximum worth control element to be fully withdrawn at maximum operator control speed from any just-critical core arrangement, without damage occurring to the fuel.

—The calculated moderator void coefficient and power coefficient assure operating system stability, spatial xenon stability, good load following response, and a safe response to transients such as a turbine tripout.

1.9 Control Requirements

The negative reactivity requirements of the control system are as follows: the initial cold clean core has a positive reactivity of approximately 0.283 Δk. Temporary poison curtains provide 0.13 Δk, and the initial worth of the moveable control elements in the cold clean core is 0.23 Δk. At equilibrium conditions, the control system needs to control only 0.16 Δk. The shutdown margins for the initial and equilibrium cores are 0.077 Δk and 0.040 Δk, respectively. The maximum reactivity in any one control element is approximately 0.02 $\Delta k/k$, under equilibrium conditions.

Following are the safety criteria which were considered in control system design.

Stuck Element Margin

The reactor is subcritical by a margin of at least 0.02 Δk, both hot and cold, with any one control element completely withdrawn, during any time in the life of the core. Use of liquid poisons is not required to maintain subcriticality.

Control Element Maximum Worth

The control elements are of such type and arranged in such a manner that the sudden withdrawal of the maximum-worth element in the worst operating configuration would not cause vessel damage.

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Scram Rate

The design provides sufficient negative reactivity and adequate insertion speed to assure that the reactor will shut down safely (i.e., without fuel damage) under a number of operating conditions or accidents resulting from operator error or equipment malfunction, such as a momentary loss of reactor feedwater or a full load rejection.

1.10 Core Power Distribution

Figure V-3 presents the axial variation in average power and other core properties. The curves pertain to the initial core, in which the temporary poison curtains aid in the flattening of power distribution. Control element position, as shown in the "fractional control inserted" curve, was chosen to make the minimum burnout ratio (in the hottest channel) as large as possible. The peak-to-average power is 1.2. Equilibrium Xenon, spatial distribution of the fuel temperature, and Doppler reactivity have significant effects on the power distribution and were included in the calculated power shape.

2. NUCLEAR STEAM SUPPLY - OTHER EQUIPMENT AND SYSTEMS

2.1 Reactor Vessel, Internals and Mounting

The reactor vessel is fabricated of 8-3/4-inch carbon steel plate, a material which has seen many years of service in operating reactor pressure vessels. The 21-foot 10-inch diameter and 54-foot 10-inch length make the vessel body larger than those currently in use. However, several manufacturers will be completely equipped to fabricate and test this vessel by the end of 1964. This matter is thoroughly discussed in the Appendix.

The weight of the fuel and control rod drives is evenly distributed over the bottom of the vessel, which rests on a skirt on the reactor foundation. The top head is sealed with two O-rings and held on with studs.

Steam separators and driers are mounted inside, above the core, and removed for refueling. Steam lines leave the sides of the vessel and control rod drives enter the bottom, so that no major disconnections are necessary before removing the head.

Instruments make appropriate measurements and control of temperature, pressure, water level, etc.

2.2 Reactor Water Recirculation

After steam is stripped from the boiling mixture at the top of the core, the water is recirculated to the bottom of the core in six pumping loops which contain flow-control valves.

Control of these valves permits the reactor to accompany turbine load demands, since variation in flow rate changes moderator density. Flow control is used to accompany load changes over approximately the upper one-third of any operating range.

2.3 Control Rod Drives

The control rod drives incorporate hydraulic pistons driven by water from a separate feed system, backed up by various emergency sources. The drives are mounted below the reactor, and may be replaced without removing the reactor head.

2.4 Auxiliary Water Systems

A part of the reactor water is bled off continuously, filtered, demineralized, and returned to the reactor. This removes corrosion products and other impurities, many of which are radioactive–thus inhibiting a build-up of active deposits in the plant equipment or a deposition of film on the fuel which might diminish heat transfer.

Auxiliary pumping systems also provide cooling for the reactor during shutdown, and cooling for the spent fuel storage pool.

2.5 Emergency Systems

If the reactor should become isolated from the main condensers, it would be shut down immediately and residual heat shunted to an emergency condenser.

Should core cooling fail, the reactor would shut down and a nozzle system would automatically come into service to spray water over the core.

If several control elements should fail to insert properly, negative reactivity could be supplemented by addition of liquid poison to the reactor coolant.

2.6 Fuel and Core Component Handling

The area above and to the sides of the reactor is arranged for refueling work and other servicing of the reactor internals. When the reactor head is removed, the water in the reactor may be raised into the upper portion of the drywell and placed in communication with storage pools for fuel and reactor internal equipment. With this communication established, radioactive components may be moved freely between the reactor and the pools, using water as the only shielding medium.

3. CONTROL AND INSTRUMENTATION

The control and instrumentation system enables the operator to start up, operate, and shut down the plant, and provides protection for personnel and equipment.

The Plant Control Room in the administration building is the main operating center. The operating controls are mounted on consoles, while the indicating and recording instruments are mounted on vertical panels behind the consoles.

3.1 Reactor Control and Load Following

As load demand varies, the power output of the plant may be controlled by varying the flow of reactor recirculating water, or by moving the control elements in the core.

Flow-control load following makes use of the thermal characteristics of the reactor cooling water, which serves as the moderator. Steam bubbles generated in the core decrease the average moderator density, and thus cause a decrease in reactivity. In order to increase reactor power production, therefore, it is only necessary to increase water flow to sweep some of the steam void from the core and thus increase the average moderator density. As the reactor power increases, more steam bubbles are formed, and the reactor stabilizes at a new higher power level. Power reduction is accomplished by the reverse process. This type of load following does not require movement of control elements. Flow is controlled by motor-driven valves in the main recirculation loops below the reactor. Load level can be followed thus over approximately the upper 30 percent of the power range.

The reactor has hydraulically-driven neutron-absorbing control elements, each of which may be moved individually from the Plant Control Room. The elements are capable of controlling the reactor over the entire range up to rated power. No liquid poisons are required.

3.2 Neutron Monitoring

Four separate systems are used to monitor neutron level from startup to full power operation. Each system has several channels tied into the recording-annunciating panels of the control room, and certain functions are arranged to cause reactor scram.

Aside from the more conventional ion chambers located in the reactor shield, a large number of tiny fission chambers are distributed throughout the reactor core. These are of great assistance to the operator in his control element maneuvering to promote optimum fuel use and to flatten the power distribution.

3.3 Reactor Protection System

The reactor protection system provides for rapid shutdown ("scram") of the reactor in the event of transients or malfunctions which could lead to a potentially unsafe condition.

Dual trip-channels are provided, equipped with two or more sensors in each channel, so that (1) even though a sensor may malfunction and fail to trip, an incipient unsafe condition will still result in reactor scram, and (2) a trip due to erroneous operation of a sensor circuit will not cause a spurious scram, because scram is initiated only when both channels are tripped.

4. TURBINE AND STEAM EQUIPMENT

The plant utilizes two identical turbine-generator sets. Arrangements A and C operate on a non-reheat cycle, whereas reheat is included in Arrangements B and D. This choice of alternates has permitted economic and operational comparison of the two cycles at this particular rating and under the specified steam conditions. The reasons for choice of two shafts, instead of a single shaft, are presented in Chapter II of this volume.

In the non-reheat cycle, saturated steam is admitted through stop and control valves to the single-flow high-pressure element and exhausts at about 180 psia. It is carried to the double-flow intermediate-pressure element, from which it exhausts at about atmospheric pressure to four cross-arounds containing moisture separators. From these separators, the steam passes to two double-flow low-pressure elements, from which it exhausts at 1.5 inches Hg abs to the condensers.

In the reheat cycle, the saturated steam is admitted to the double-flow high-pressure element, from which it exhausts into four moisture separators. The steam, reduced to 1 percent moisture, passes to four two-stage steam reheaters. The first stage utilizes extraction steam from the high-pressure turbine, and the second stage, steam directly from the reactor. The reheated steam is admitted to three double-flow low-pressure elements, from which it exhausts to the main condensers at 1.5 inches Hg abs. The reheat piping is provided with stop, intercept and relief valves.

Special provisions for moist steam contained in the turbines include grooved moisture-extracting buckets, drainage pockets in each diaphragm, and drains from exhaust casing steam guides. The moisture separators in the cross-arounds are of the corrugated plate type. Water extracted from the separators and the turbine is piped to the feedwater heater system.

4.1 Turbine Control

The two turbines are equipped with pressure regulators which position the turbine control valves and the controlled bypass valves in such a way as to maintain reactor pressure essentially constant. Plant output is determined by reactor steaming rate, which is controlled by variation in reactor recirculation flow, or by core control element movement.

A main bypass system, sized to handle 40 percent of rated steam, admits steam directly to the main condensers in the event of a load rejection. If the rejection is of 40 percent or less, the reactor will drop to the new load level. If the rejection is greater than 40 percent, the reactor will scram, but the rise in pressure is designed to be below the setting of the safety valves, so that they would not normally blow.

5. CONDENSER AND FEEDWATER SYSTEMS

The systems for handling condensate and for supplying reactor feedwater are of basically conventional design. Principal deviation is due to the fact that the steam will carry a certain quantity of radioactive gases resulting from the radiolytic decomposition of the water as it passes through the reactor. The presence of these active components requires that special handling be given to the noncondensible gases discharged from the air ejectors, vacuum pump, and gland seal condenser. These discharged gases are sent to the off-gas system, described in section 8, Radioactive Wastes.

5.1 Condensers and Condensate

The condensers are single-pass type, utilizing straight Admiralty metal tubes welded to silicon bronze tube sheets. Each condenser is equipped with divided water boxes to permit shutting down half of the condenser if a leak should occur.

All condensers are supplied with cooling water from four tunnels, two serving each turbine. The two tunnels to one turbine are connected so that each tunnel feeds a half water box on each of the condensers. Thus, if one tunnel should be shut down, all of the low-pressure elements of that turbine would discharge into condensers which had been equally reduced in capacity. The condensers discharge through four tunnels to a seal well connected to the canal.

The intake structure is part of the turbine building, so that the circulating water pumps can be serviced by the turbine-building crane. Two pumps discharge into each tunnel.

Air ejectors and vacuum pumps are used in a conventional manner, except that they discharge to systems especially designed to monitor, control and release the radioactive gases.

Before being stored or entering the feedwater heaters, condensate passes through demineralizers to remove corrosion products and other impurities. Regeneration of the demineralizer resins is controlled from a local panel, some of the operations being manual, and some automatic.

5.2 Reactor Feedwater

Demineralized condensate passes through a series of three feedwater heaters, with pumps located between the second and third. Heating is by means of water and steam from extraction lines and seals and the moisture separators. All heaters are of the horizontal, U-tube, removable shell type.

6. ELECTRICAL SYSTEMS

The electrical systems are essentially identical for all plant arrangements. Each of the two 1800 rpm main generators is rated 640 mva at 22 kv, and provided with a gear-driven exciter. Generator rotors are cooled with hydrogen at 45 psig, and the stators are water cooled.

Switching, control and protective relaying equipment for the main electrical system are centralized on a board located in the plant control room.

Auxiliary power sources include:

1. Startup A-C at 4160 volts, taken from the 345 KV switchyard bus.
2. Normal Running A-C at 4160 volts, taken from each generator.
3. Emergency A-C at 4160 volts, provided by a 2500 kva diesel-driven generator.
4. Utility D-C at 250 volts, supplied by a 120-cell battery which is continuously float charged.

In addition to the 4160-volt systems, power is distributed at 480 and 120/208 volts.

7. SERVICE SYSTEMS

Service water pumps located at the intake structure supply water to various cooling systems in the plant.

Some of this water passes through heat exchangers in the reactor building. Water in the secondary circuit of these exchangers is in a closed loop, and is pumped through the cooling coils of the reactor shield, the non-regenerative heat exchanger, the drywell cooler and other coolers located in the reactor building.

Strained river water is pumped directly through the cooling coils of turbine building equipment, such as the hydrogen coolers, feedwater pumps, air compressors, turbine oil cooler, etc.

A storage tank is provided for condensate; and plant make-up requirements are supplied by processing fresh water through a demineralization system.

Well water is chlorinated for the sanitary system and heated for showers and basins.

Pumps mounted on the intake structure supply water to a fire hydrant loop in the yard, which in turn feeds hose outlets in the radwaste building and a fire loop in the turbine building. Chemical extinguishers are available in the reactor building.

Dried and filtered instrument air is supplied by two automatic-start compressors. Service air is available from a separate system.

Filtered air for ventilation of the turbine building is passed through heating coils or a spray washer to control temperature. The reactor building is maintained at a slight negative pressure by automatic control of the intake and outlet blowers. Isolation valves would operate in the event of an incident.

The radwaste building has controlled ventilation to permit selective purging as well as full-time control of any radioactive gases released in the processes.

All radioactive or potentially radioactive gases from the plant are discharged through the stack at a predetermined maximum rate.

8. RADIOACTIVE WASTES

Continuous- or batch-monitoring of radioactivity is employed throughout the plant so that radioactive material is never released from the plant in an uncontrolled manner. Release to the environs is expected to be well within permissible limits. Waste processing facilities are sized to handle peak loads so that waste accumulation should never limit plant operation.

Radioactive gases, which are primarily activation products in the water, are collected by the off-gas system, allowed to decay to a safe level, and then discharged to the atmosphere through the stack. The height of the stack is such that atmospheric dilution prevents any significant increase in radioactivity in the vicinity at ground level.

Liquid wastes result from sampling, refueling, leakage, and washdown of contaminated areas. These wastes pass to the radwaste building where they are processed. Purified waste water may be pumped to the condensate storage tank for reuse or may be diluted and discharged to the cooling water canal. Liquid wastes may also be concentrated, transformed to solids, and then stored or shipped off-site.

Radioactive solid wastes include demineralizer resins, filter sludge, contaminated tools, and reactor core parts such as monitors and control elements. Normally, these are put in packages or casks and shipped off-site. Some tools and equipment may be decontaminated and returned to service.

9. CIVIL AND STRUCTURAL

The layout of the plant is compact, with the reactor building adjoining and sharing common walls with the turbine and administration buildings.

The plant is arranged and provided with facilities which make it convenient to protect working personnel against possible radiation and contamination hazards. By means of shielded corridors and stairways, personnel can pass to or from any of the controlled areas without exposure to other areas. Most shielding is of structural concrete–heavy concrete is not used.

Shielding is provided for the control room and for other areas, such that operating personnel receive a radiation dose which is below the permissible limit. In addition, the control room is shielded to limit dose rate to 0.5 rem/8 hours following a hypothetical* accident.

In the twin-reactor arrangements, all normal maintenance and refueling work may proceed on one reactor while the other is in full operation.

Shielding throughout the plant is adequate to permit continued operation with the quantity of fission products in the water which would result from a hypothetical** maximum amount of defective fuel.

The reactor pressure vessels in all plant arrangements are each contained in a drywell with piping to independent pressure-suppression chambers. A bridge crane above the reactor handles the drywell head, vessel head, and all major core components.

The turbine building is enclosed, with the turbine-generators placed side by side and isolated from the building structure to reduce noise. The generator-end of the building houses the switchgear, motor-control centers, diesel-driven generator, and station battery.

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* During refueling or involving loss of coolant.

**Equivalent to one curie per second of a mixture of noble gases.

The circulating water intake structure is part of the turbine building, which results in short tunnels to the condensers, and easy access to the pumps for maintenance. Tunnels from the condensers lead to a seal well which empties into the discharge canal. Valved bypass conduits divert warm discharge water to the intake, for ice-control during severe cold periods.

The radwaste building is primarily below grade, heavily-shielded, and remotely operated.

An underground unit treats non-radioactive sewage before discharging it to the river.

The buildings for the office, machine shop and warehouse are of standard construction.

10. PLANT STARTUP, OPERATION AND MAINTENANCE

A single Plant Control Room is provided for operation of the reactor, the turbine-generators, and the main switching. The Reference Plant should use an operating staff of approximately 78 men, which number may be increased or decreased, depending upon the practices of the operating Utility. Turbine inspection and plant maintenance work are performed during the annual refueling by the regular plant staff.

10.1 Plant Startup

Pre-operational tests include checkout of the initial loading instrumentation and special attention to all safety devices. Fuel is then loaded to obtain minimum critical configuration with and without temporary control curtains. Moderator coefficients are measured and control elements calibrated.

The core is loaded to rated size, more coefficient tests are run, and then the head installed and the reactor pressurized. Power is increased in small increments, with many measurements and observations at each step.

10.2 Normal Station Operation

Startups are made on the basis of predicted critical configurations. During operation at power, the operator maneuvers the control elements to give the best flux flattening and optimum fuel burn-up. Over the upper part of the power range, load is followed by the automatic recirculation flow control system.

The reactor is shut down by inserting the control elements, bypassing steam as necessary, and adjusting the pressure regulator downward until the reactor is sufficiently cool to permit the shutdown cooling system to replace the main recirculation loops.

A load rejection of up to 40 percent of rated plant power is handled routinely by automatic steam bypass operation and adjustment of reactor power. A larger load rejection (up to 100 percent) scrams the reactor but should not blow the safety valves.

10.3 Refueling and Related Operations

Refueling would normally be on an annual basis, and would require replacement of 20 percent of the core, and shuffling of 20 percent of the fuel within the core. The time from reactor shutdown to full operation is estimated at 19 days for Plant Arrangements A and B, and 13.6 days for each reactor of Arrangements C and D, assuming 85-percent efficiency of personnel and equipment.

10.4 Maintenance

Much of the plant equipment has sufficient over-capacity so that units such as pumps, demineralizers and instruments can be withdrawn from service for maintenance during reactor operation.

Other equipment, which has insufficient over-capacity, or which is mounted in radiation zones, is serviced during refueling outages. This applies to the turbines, recirculation pumps, core components, and control rod drives.

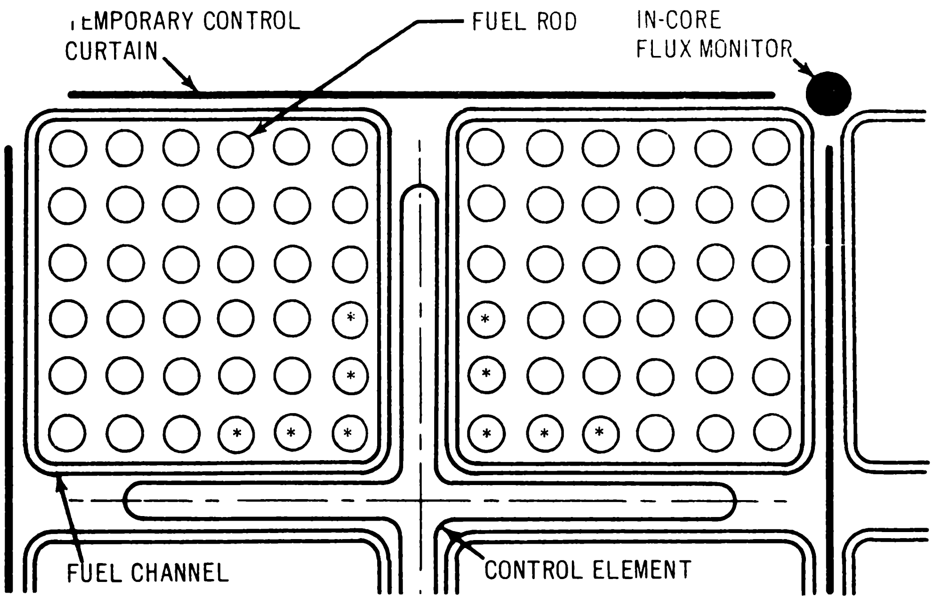
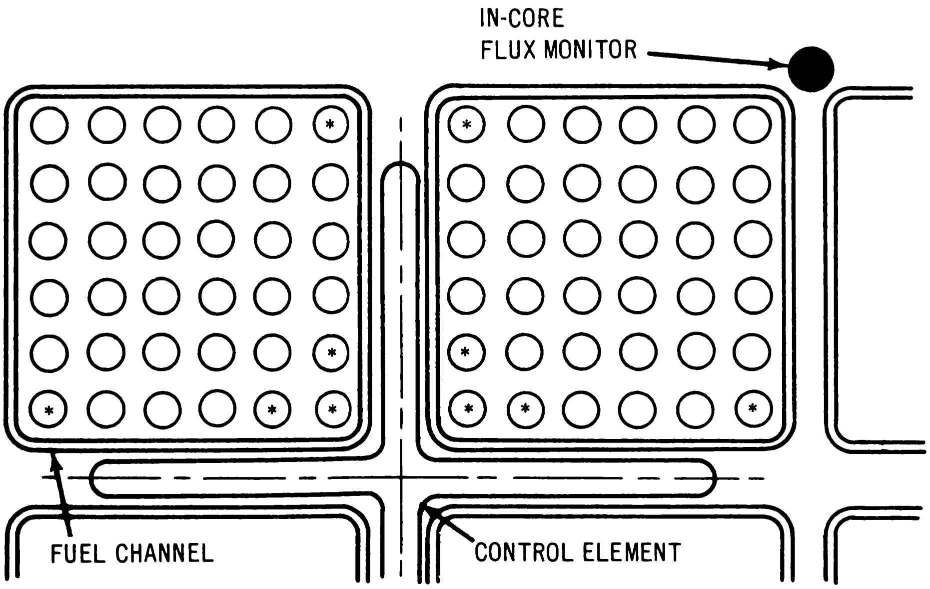


Figure V-1. Initial Core Lattice



NOTE:

(1) SPECIAL CORRECTED CORNER RODS AS DENOTED BY (*) HAVE DIFFERENT ENRICHMENTS THAN STANDARD RODS IN BUNDLE.

(2) FUEL BUNDLE CONTAINS 36 FUEL RODS

1651-13

Figure V-2. Equilibrium Core Lattice

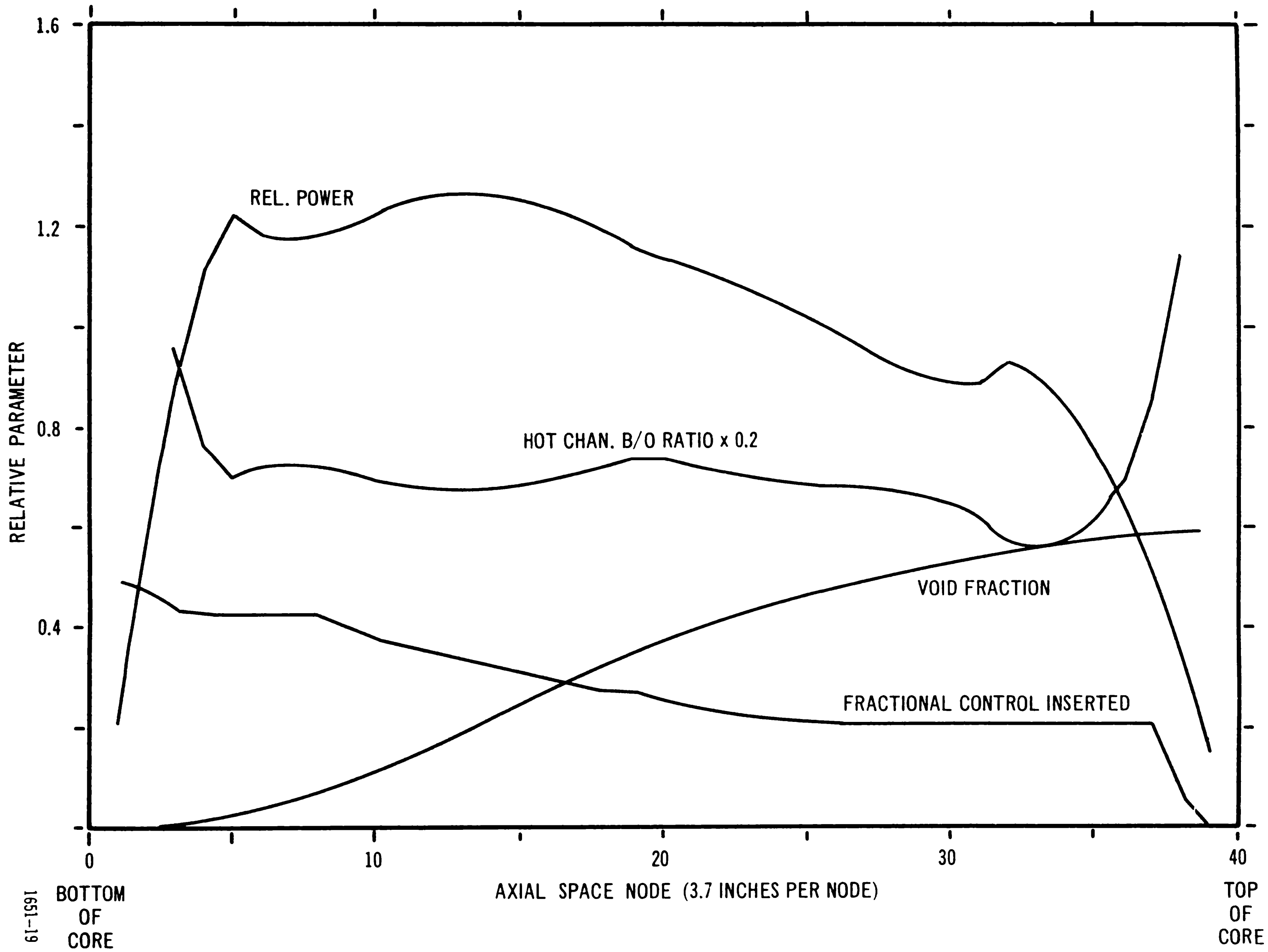


Figure V-3. Spatial Core Properties - Clean Core, 1000 MWe BWR, Reference Arrangement A

CHAPTER VI

SPECIAL SERVICES AND SCHEDULE

1. PROJECT MANAGEMENT

Successful completion, on schedule, of projects such as this large boiling water reactor nuclear power plant requires close coordination of all phases of engineering, procurement and construction, and substantial planning by all principals involved. Experience has demonstrated the value of a project organization to accomplish these objectives. The plant price includes assigning to the project an experienced Project Management organization with sufficient authority to assure that all responsibilities are met.

2. TRAINING

Comprehensive training facilities and qualified instructors are included in the price of this plant. Instruction is provided to fit requirements, and special arrangements for supplemental training may be made.

3. STARTUP SERVICES

The plant price includes the services of experienced supervisors for startup and development of related operating procedures. This assistance by qualified, licensed personnel will be furnished from the time prior to fuel loading until the plant is in full operation.

4. SCHEDULE

The schedules of design, construction, and startup are shown in the tables that follow. It will be noted that the schedule is one year longer for Plant Arrangement A than for Plant Arrangement C. This is primarily due to the 42-month delivery schedule on the large vessel, compared to 30 months average for the two vessels in the twin-reactor plant. This factor also causes the interest during construction to be larger for Plant A than for C. AEC ground rules determine the interest for A as 14.2%, compared to 11.5% for C; a difference of 3 million dollars. This is probably unrealistic, as the expenditure pattern for Plant A could be considerably different from that of Plant C. Two months of engineering and design would precede issuance of the vessel inquiry. This could be followed by several months in which expenditures are virtually nil. Later on a normal engineering schedule would be resumed, and construction would start 9 to 10 months later than for Plant C.

It is also probable that the vessel vendor could reduce the time schedule estimate on the large vessel, if given an opportunity to make a more detailed fabrication study.

LEGEND

- I INQUIRY ISSUED
- O ORDER PLACED
- ▼ DELIVERED AT SITE

PREPARED BY CONSTRUCTION DEPARTMENT

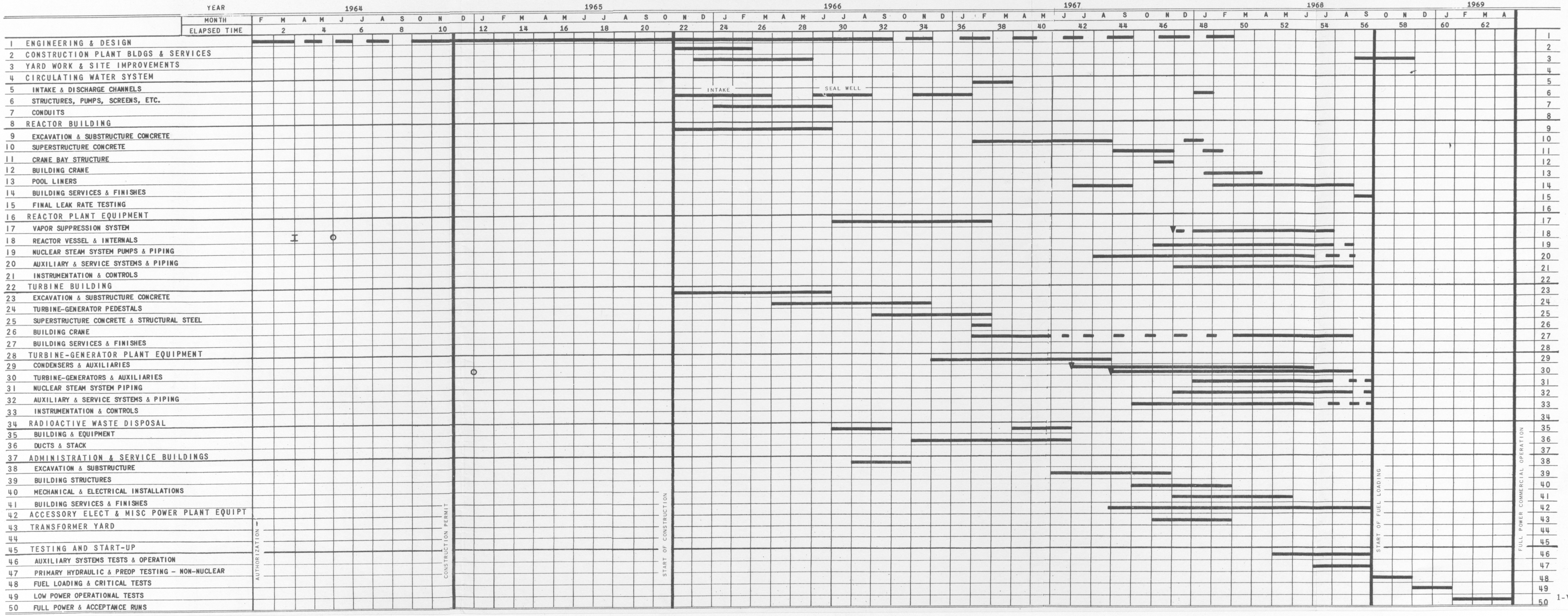
PROPOSED CONSTRUCTION SCHEDULE

1000 MWE BOILING WATER REACTOR PLANT FEASIBILITY STUDY

ARRANGEMENT A - ONE REACTOR NON-REHEAT CYCLE

FOR ARRANGEMENT B - ONE REACTOR REHEAT CYCLE-SLIGHT ADJUSTMENTS TO INDIVIDUAL CONSTRUCTION ITEMS WILL NOT CHANGE OVER-ALL LENGTH OF SCHEDULE.

PLACE NEW YORK, N.Y. DATE DECEMBER 1963



LEGEND

I INQUIRY ISSUED
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▼ DELIVERED AT SITE

PREPARED BY CONSTRUCTION DEPARTMENT

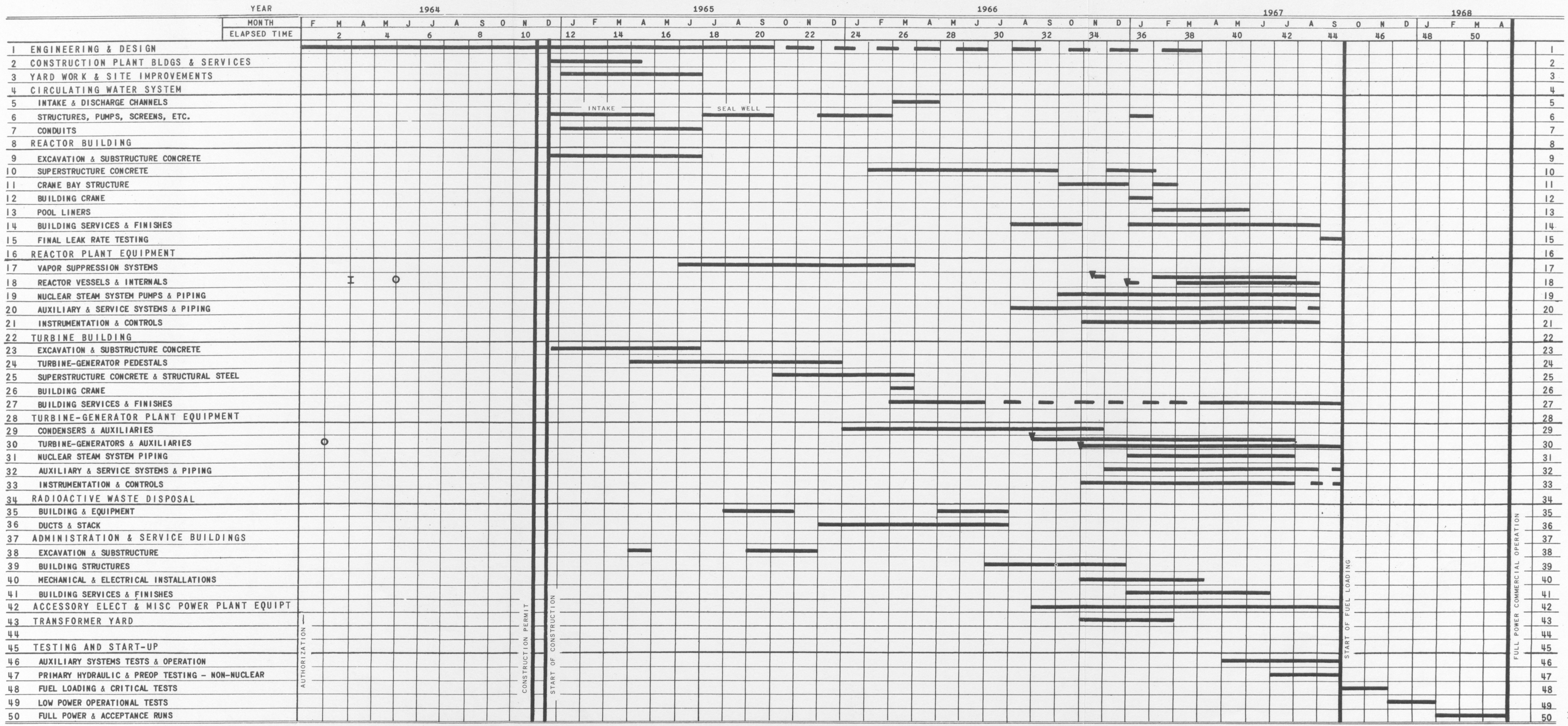
PROPOSED CONSTRUCTION SCHEDULE

1000 MWE BOILING WATER REACTOR PLANT FEASIBILITY STUDY

ARRANGEMENT C - TWO REACTORS NON-REHEAT CYCLE

FOR ARRANGEMENT D — TWO REACTORS REHEAT CYCLE- SLIGHT ADJUSTMENTS TO INDIVIDUAL CONSTRUCTION ITEMS WILL NOT CHANGE OVER-ALL LENGTH OF SCHEDULE.

PLACE NEW YORK, N.Y. DATE DECEMBER 1963



CHAPTER VII

BASIC DRAWING LIST

The following is a listing of reference General Electric Company and Ebasco Services Inc. drawings designated as the basic drawings required for use with this report. Other reference drawings are filed at the end of the report following the appendices.

The figure numbers shown in the first column of the drawing list correspond to circled numbers located in the upper right corner of each drawing.

Figure No.	Drawing No.	Title
1	3250-M-1	Plot Plan, Arrangement A
2	237-E-285	Plant Flow Diagram, Arrangement A
3	3250-M-8	Heat Balance, Arrangement A
4	3250-M-5	General Arrangement, Ground Floor Plan, Arrangement A
5	3250-M-6	General Arrangement, Sections A-A, Arrangement A
6	-	Reactor Assembly
7	3250-E-1	Electrical One Line Diagram

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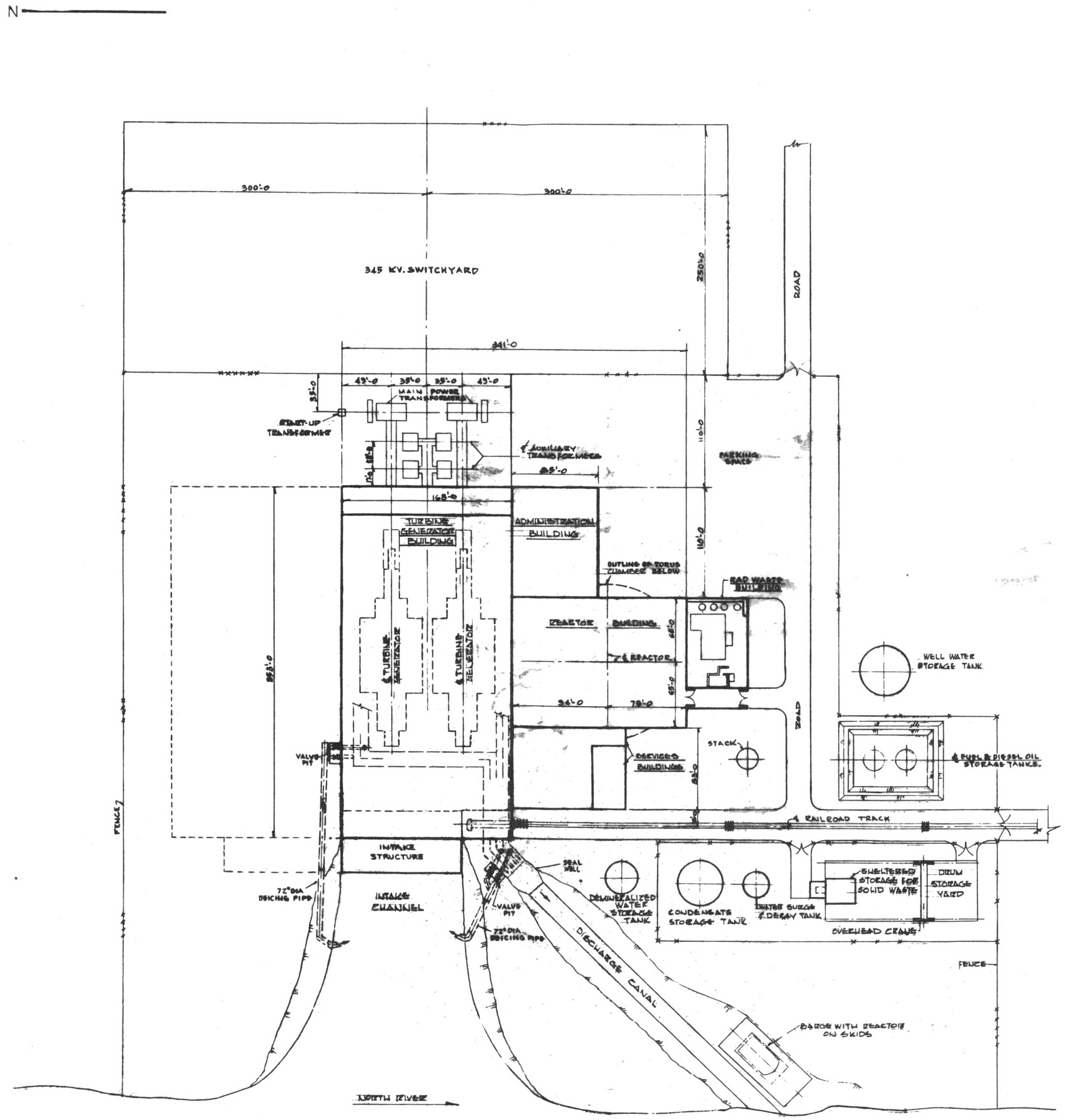
1



PLOT PLAN – ARRANGEMENT A

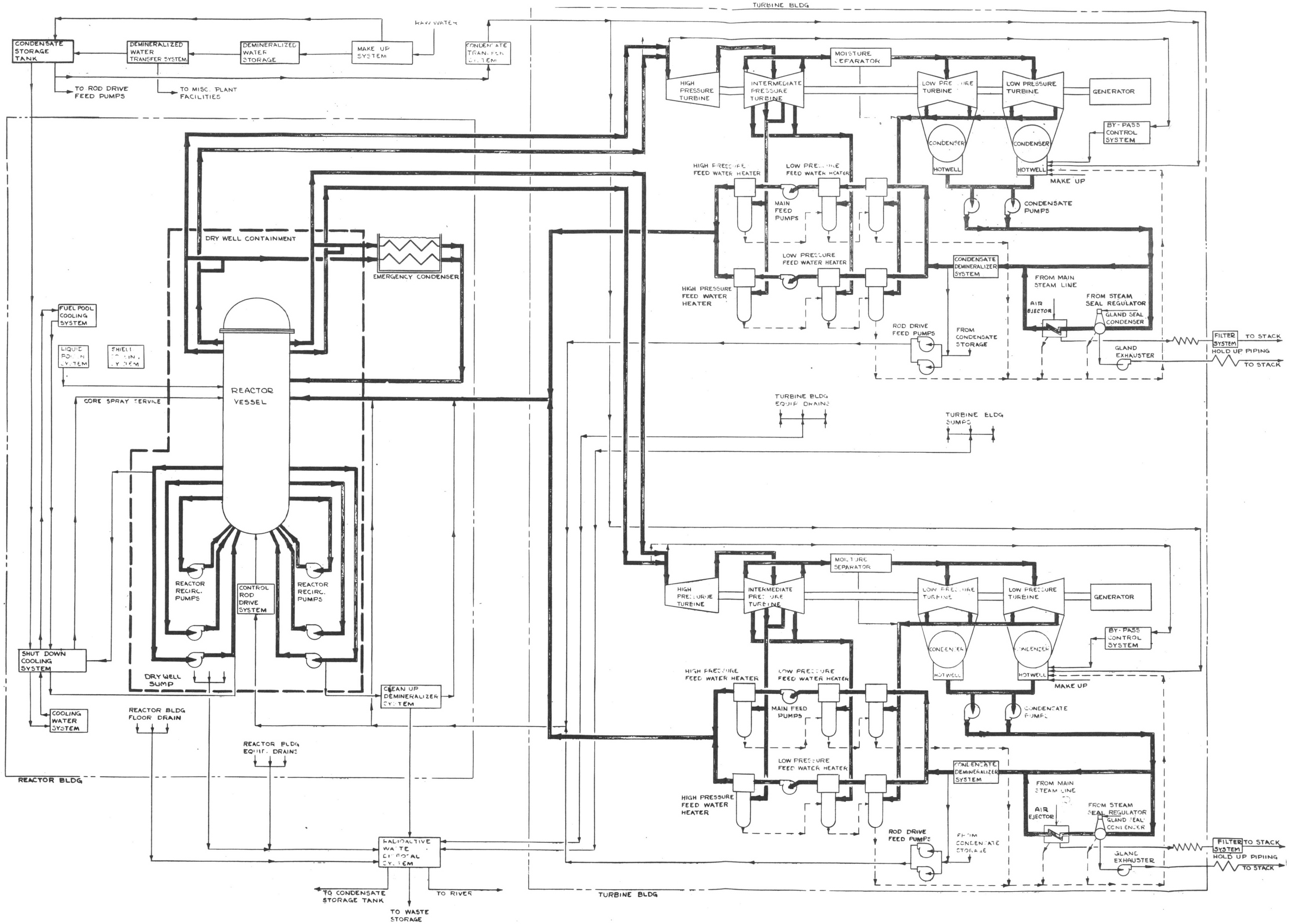
SIMILAR TO DRAWING NO. 3250–M–1

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

EBASCO SERVICES INCORPORATED
ENGINEERS AND CONTRACTORS

2



PLANT FLOW DIAGRAM – ARRANGEMENT A

SIMILAR TO DRAWING NO. 237E285

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC

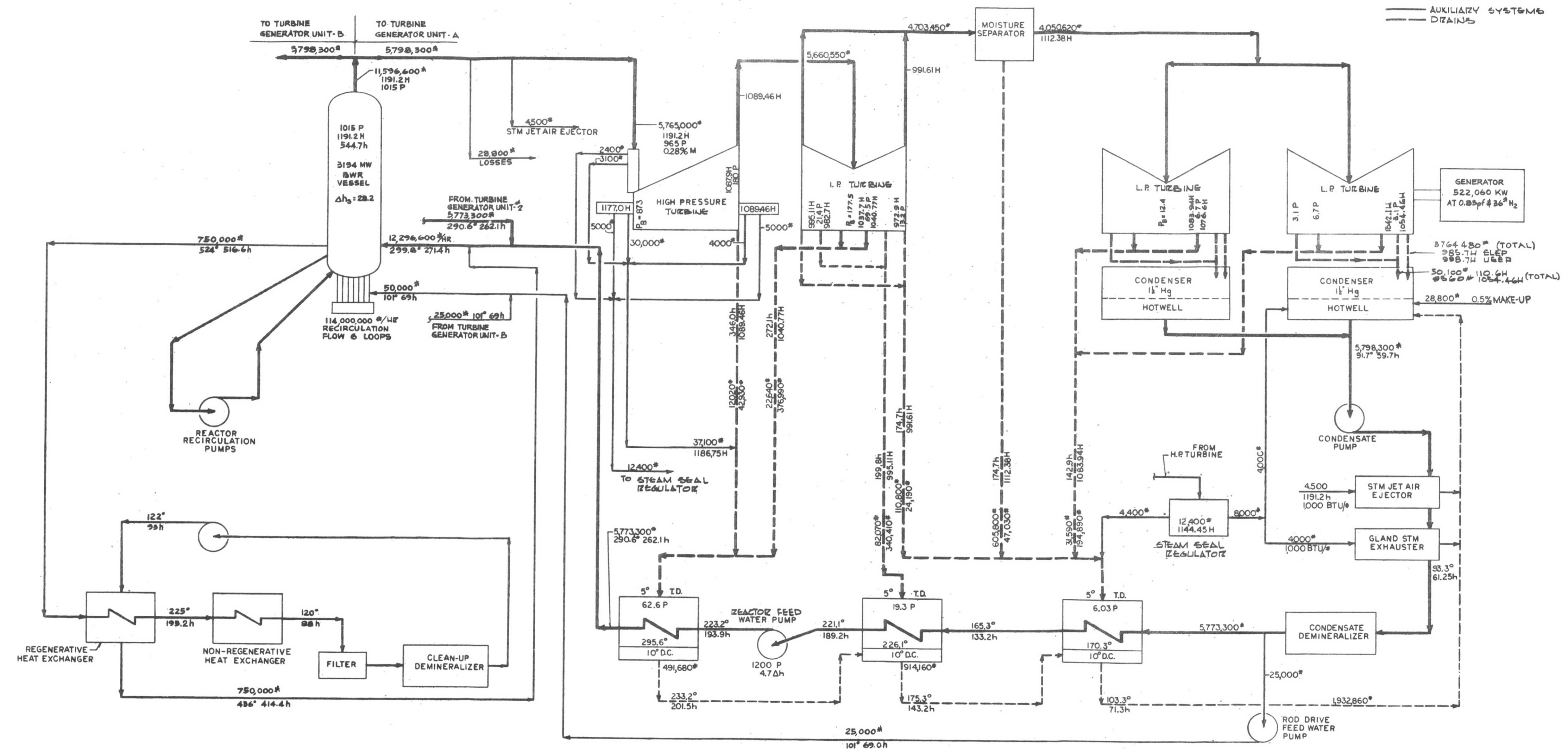
ATOMIC POWER EQUIPMENT DEPARTMENT

GPO 812-316-9

3

LEGEND

- # FLOW-LBS/HR
- P PRESSURE-PSIA
- ° TEMPERATURE
- H, h ENTHALPY-BTU/LB
- P_B BOWL PRESSURE
- ELEP EXPANSION LINE END POINT
- UEEP USED ENERGY END POINT
- STEAM & FEED
- BLEED
- AUXILIARY SYSTEMS
- DRAINS



ESTIMATED GROSS PLANT HEAT RATE $= \dfrac{3194 \text{ MW} \times 3.413 \times 10^6 \text{ BTU/MW-HR}}{1{,}044{,}120 \text{ KW}} = 10{,}440$ BTU/KW-HR

ESTIMATED NET PLANT HEAT RATE $= \dfrac{3194 \text{ MW} \times 3.413 \times 10^6 \text{ BTU/MW-HR}}{1{,}000{,}000 \text{ KW}} = 10{,}900$ BTU/KW-HR

NOTE:
IN ORDER TO GENERATE 1000 MW FOR ARRANGEMENT "A" TWO IDENTICAL TURBINE GENERATOR UNITS AS SHOWN ABOVE WILL BE REQUIRED.

HEAT BALANCE – ARRANGEMENT A

SIMILAR TO DRAWING NO. 3250–M–8

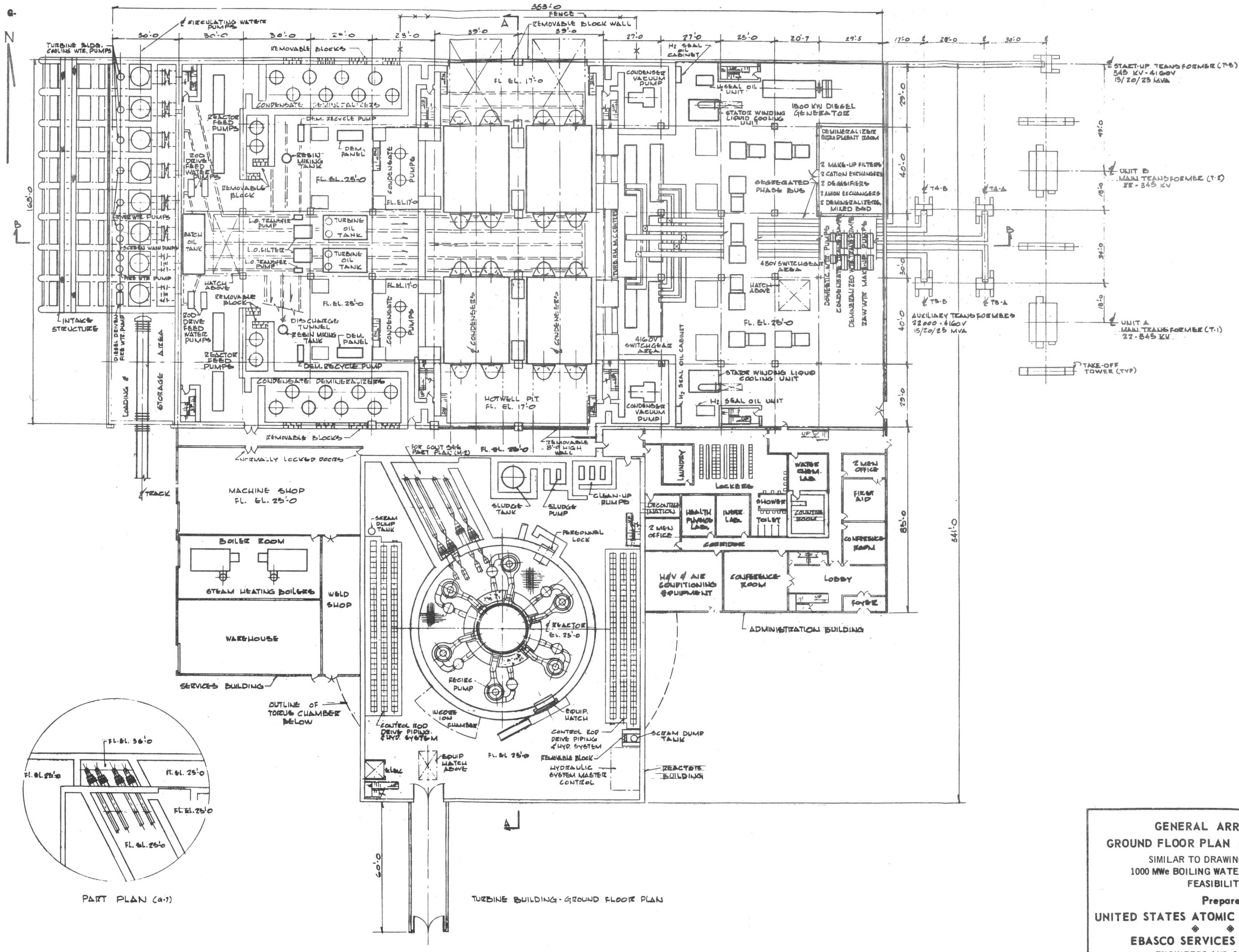
1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

EBASCO SERVICES INCORPORATED

ENGINEERS AND CONTRACTORS



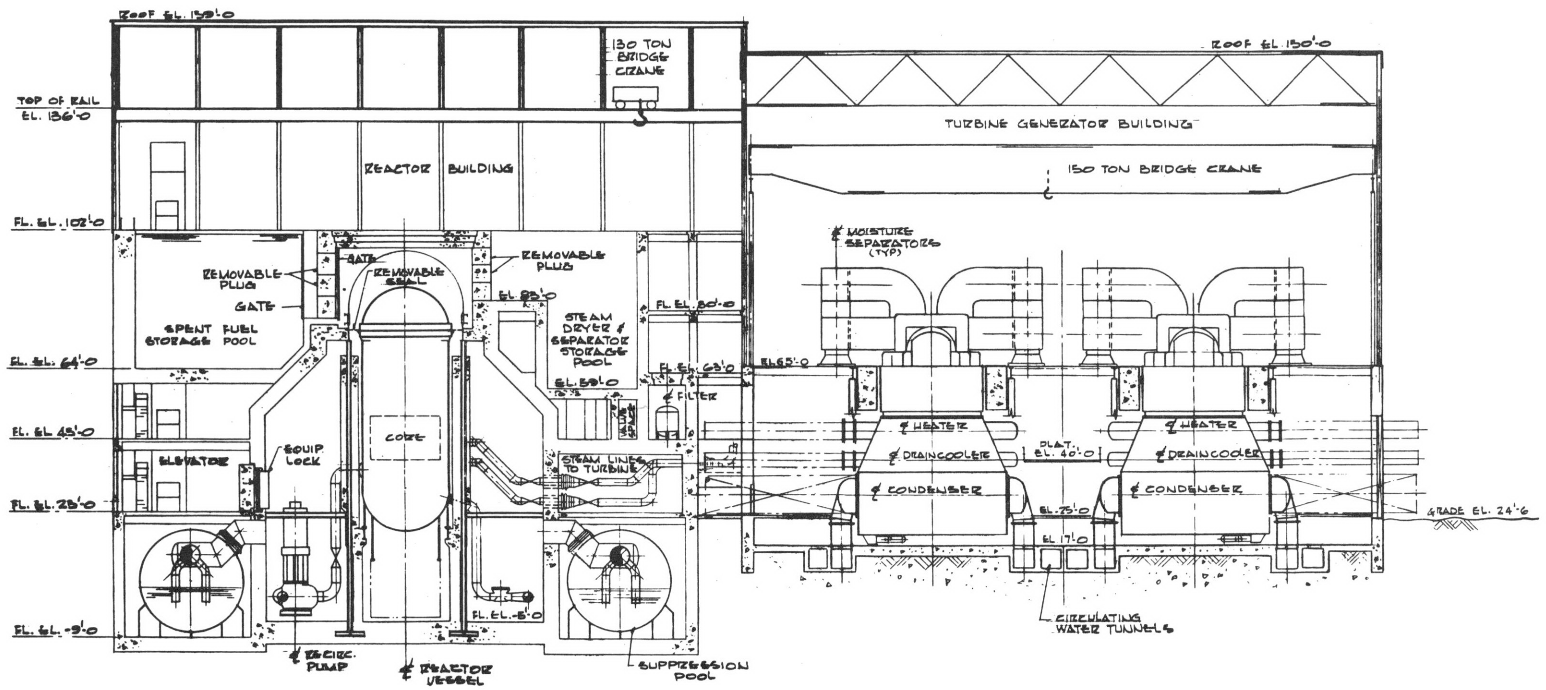
GENERAL ARRANGEMENT
GROUND FLOOR PLAN – ARRANGEMENT A

SIMILAR TO DRAWING NO. 3250–M–5
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

EBASCO SERVICES INCORPORATED
ENGINEERS AND CONTRACTORS

GPO 812–316–10



SECTION A-A

GENERAL ARRANGEMENT
SECTIONS A-A – ARRANGEMENT A
SIMILAR TO DRAWING NO. 3250-M-6
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
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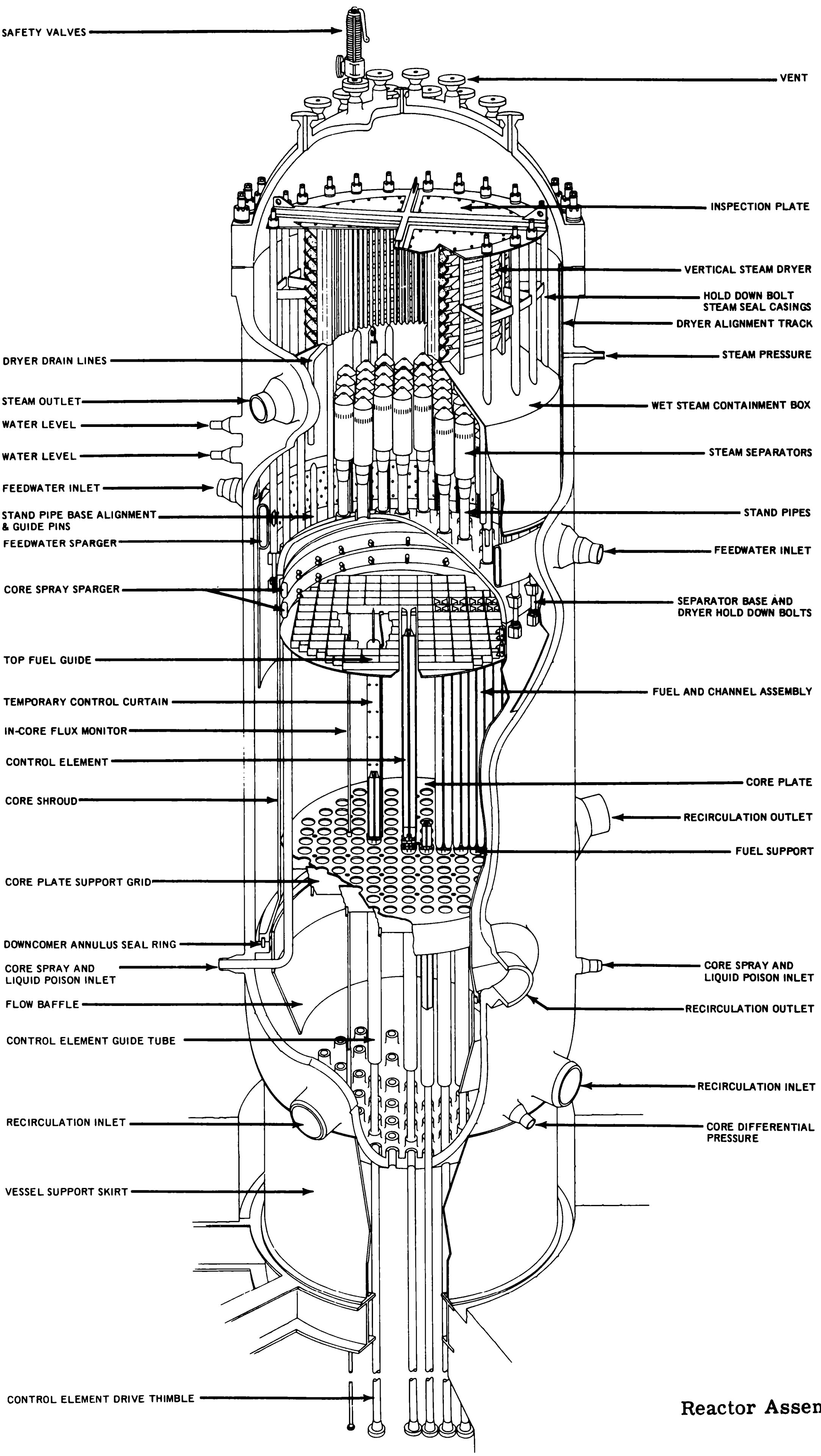
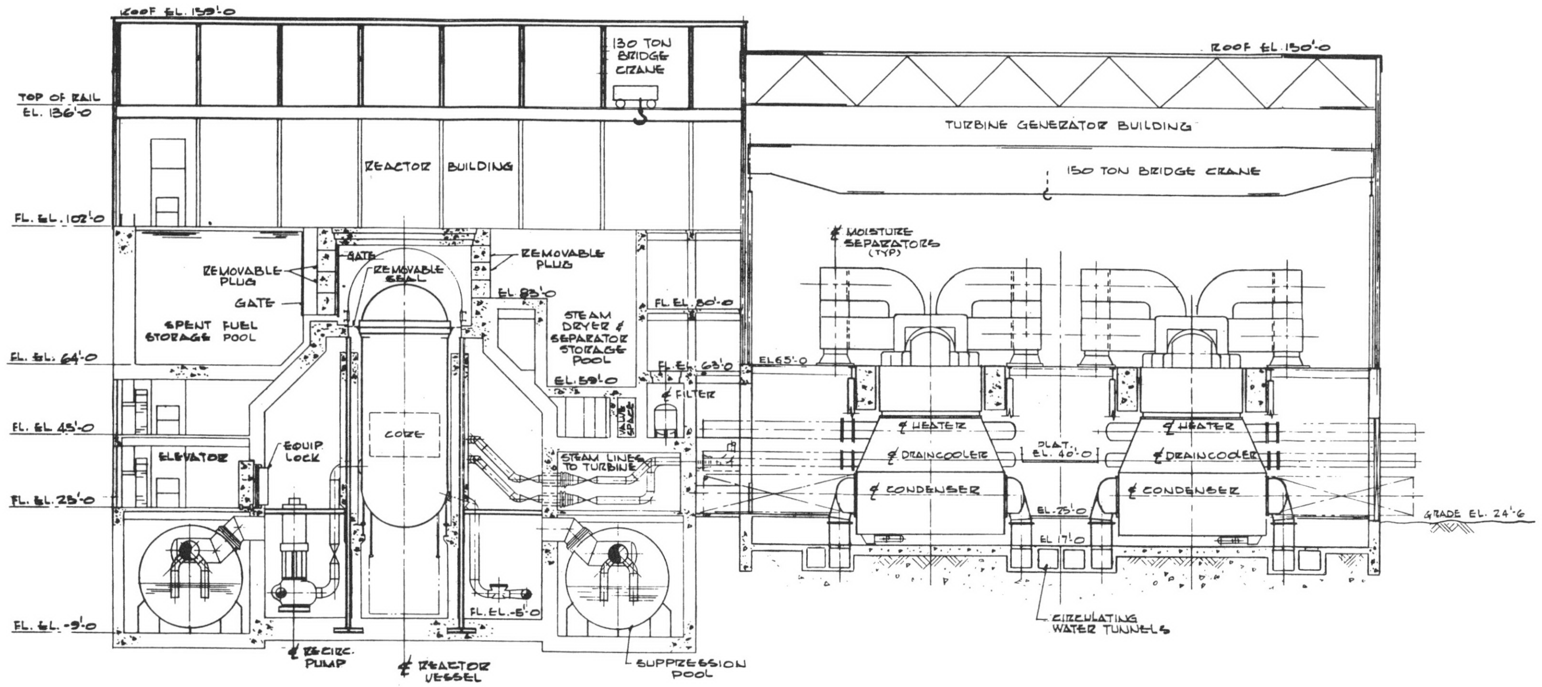


Reactor Assembly



SECTION A-A

GENERAL ARRANGEMENT
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