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# 1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

VOLUME III

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
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ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA

VOLUME 3

# INTRODUCTION

Volume 3 contains the Appendices and a complete set of drawings related to the 1000 MWe Boiling Water Reactor Plant Feasibility Study performed by the General Electric Company for the United States Atomic Energy Commission under Contract AT(94-3)-189 P. A. No. 36, of June 1, 1963.

Volume 1 presents a summary of the results of the study, and Volume 2 contains complete technical descriptions of the Reference Plant and the alternates. The four plant arrangements which were studied are as follows:

Arrangement A - Single Reactor Non-Reheat Cycle - Reference Plant

Arrangement B - Single Reactor Reheat Cycle

Arrangement C - Twin Reactor Non-Reheat Cycle

Arrangement D - Twin Reactor Reheat Cycle.

All reactors are of the single-cycle type with internal steam separation, and send saturated steam directly to twin turbine-generators.

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# APPENDIX I

## CORE CALCULATIONS

The following sections describe the physics calculations made to determine core power distributions, core stability, exposure behavior and fuel management patterns for the 1000 MW plant (core data are summarized in Table XI-1 of Volume 2.). From a physics standpoint, the cores for both the 500 MWe and 1000 MWe reheat plants are essentially the same as for the non-reheat plants. Hence the physics calculations have been limited to the non-reheat plants, but apply equally well to the reheat plants.

Preliminary estimates were made to determine important core design parameters so that studies could be initiated regarding mechanical design, system thermal hydraulics, the vessel, and the containment. The detailed physics and thermal hydraulic studies indicated that the preliminary estimates were conservative. However, schedule and cost considerations precluded revision of the design to include the results of the detailed studies and therefore, the reference design includes some margin relative to the results reported herein.

### 1. SUMMARY OF IMPORTANT RESULTS

One, two and three dimensional methods have been applied to the analysis of the 500 and 1000 MWe cores. Increased amounts of computing time, above what is normally required for smaller cores, were necessary to insure proper convergence of the power distribution and eigenvalues in the three-dimensional studies.

Calculations of the power distribution in the 1000 MWe core at zero exposure exhibit considerable sensitivity to control element positioning, a characteristic of large low-leakage loosely coupled cores. This sensitivity is due in large part to the $\Delta k$ reactivity difference between a controlled and uncontrolled portion of the core relative to the small neutron leakage component. As exposure progresses, this $\Delta k$ difference is reduced, and the core becomes less sensitive to control element movements.

The effects of xenon and Doppler reactivity defect, both of which exhibit self-flattening effects on the power distribution, were studied in three dimensions. Power peaking without xenon is increased approximately 4-5 percent above the peaking observed when xenon has had time to reach equilibrium. Void-dependent Doppler has been shown to have considerable effect in controlling the magnitude and position of the power peak.

Investigations of large core stability have shown that the large negative power coefficient, characteristic of a boiling water reactor, contributes greatly to core stability. An approximate

threshold for sustained xenon oscillation at power coefficients of -0.014$\Delta$k/k per unit change in power has been determined.[6] The calculated power coefficient for the 1000 MWe core was found to be approximately -0.09$\Delta$k/k per unit change in power, based upon a 10 percent decrease in total core power.

The use of control elements for load following was found to reduce the stability threshold for spatial xenon oscillations; however, flow control was found to be adequate for load changes of 30 percent of full power.

Three dimensional exposure studies indicate that acceptable peak-to-average power values, somewhat less than design values, can be maintained throughout core life by proper control element sequencing. Due to the increased sensitivity of the power distributions to control element insertion at the beginning of life, greater reliance on in-core instrumentation and/or pre-calculated control patterns must be made for the 1000 MWe core.

A 5-batch scatter pattern refueling, with fresh fuel loaded in the periphery was selected as the fuel management scheme. Calculations of the exposure behavior through the first reload confirm the ability to control the power distribution throughout life.

## 2. CORE POWER DISTRIBUTION

### 2.1 Methods

The FLARE code developed by Fischer[1] for three dimensional, coupled nuclear thermal hydraulic analyses of boiling water reactors, was used extensively in calculations for the 1000 MWe plant. The model includes feedback from power to voids, xenon, and fuel temperature (Doppler). Fischer, et al.,[2,3,4] have shown that FLARE results agree to within 10 to 15 percent of measured power distributions taken from the startup physics experiments for the Big Rock Point, RWE-KAHL, Humboldt Bay, and Dresden plants. A limited amount of exposure data (operating control patterns) from the RWE-KAHL plant[5] have indicated good agreement with predicted operating eigenvalues, and hence calculated exposure distributions. No comparisons of FLARE-calculated exposure distributions with those actually obtained from isotopic analyses from an operating plant have been made, however.

Explicit representation of control elements, coupled with a full three-dimensional representation of the reactor with nuclear and thermal-hydraulic effects included makes FLARE a powerful tool for investigation of boiling water reactors.

In addition to FLARE, other methods of analysis developed for study of axial power distributions and investigation of transient effects, results from studies of other plants currently being built or being planned, and a vast background of cumulative experience in the design of BWR's have been utilized to establish a feasible, operable 1000 MW boiling water reactor plant.

A number of one-dimensional coupled nuclear-thermo-hydraulic optimization studies were completed. The neutron diffusion equations were solved by use of the power control method[11] which provides very rapid convergence by forcing the eigenvalue (multiplication) to 1.0 and iterating on the control absorption distribution to achieve a desired power shape. This model includes coupled void, Doppler, xenon, thermo-hydraulic, and long term fuel irradiation effects and has provision for optimization of axial control element sequencing so as to achieve the best power or burnout ratio distributions throughout the life of the plant. A number of one-dimensional axial studies were completed to optimize the axial distribution of burnable poisons in the initial core.

It is important to realize that the physics extrapolation from a 200 MW BWR plant is much smaller than the factor of 5 extrapolation of electrical megawatts output, plant size, turbine-generator rating, etc. As shown in Table I-1, from the physics standpoint, cores of Dresden size with three percent leakage are nearly "infinite", and hence the correct physics extrapolation factor is more like a factor of two. Thus, physics problems in the 1000 MWe core are expected to be the same kind of problems which are met in normal day-to-day operation at the Dresden Station, i.e., those problems which are characteristic of a large, low leakage, loosely coupled core.

TABLE I-1
COMPARISON OF BWR PLANTS

| Station | (MWe) Power | $(1 + M^2B_g^2)$ Approximate Leakage |
|---|---|---|
| Dresden | 200 | 1.03 |
| Oyster Creek | 500 | 1.02 |
| 1000 MWe Plant | 1000 | 1.013 |

## 2.2 Power Distribution

For the 1000 MWe core, representation of each fuel bundle explicitly in FLARE was not possible because of current limitations on total allowable space nodes. Hence each node in the x and y directions represented four fuel bundles clustered about their associated control element. Z direction nodes were chosen such that the spacing between nodes was approximately 12 inches in the z direction as compared to 10 inches in the x and y directions. For the 500 MWe plant, with a smaller number of fuel bundles, representation of each bundle was possible. One quarter core symmetry was used in all calculations.

In obtaining the power distributions shown in figure I-1 for the 1000 MWe core, and in figure I-2, for the 500 MWe core typical of a two-reactor plant, several trial settings of the control elements were necessary in order to obtain an eigenvalue of 1.000 ±0.005 and acceptable values of the over-all peak-to-average power ratio.

The preliminary design of the 1000 MWe core, which was needed for initiation of the vessel and plant studies, was based on an over-all peak-to-average power of 2.26, composed of a 1.3 radial peaking factor, a 1.57 axial peaking factor, and an azimuthal factor of 1.15. The over-all peak-to-average thus excludes local inter-bundle peaking factors.

For both the 1000 MWe core and for the 500 MWe core, over-all peak-to-average power ratios in the range of 1.8 to 2.0 were shown to be feasible, due principally to the self-flattening characteristic of the low-leakage core with a large negative power coefficient. In a core containing many times the minimum critical mass and in which a large amount of decoupling exists, the fact that better-than-design values of the peak-to-average power are achievable is negated somewhat by the greater sensitivity of the power distribution to small (with respect to uncontrolled-controlled $\Delta k$) local reactivity effects. Although a peak-to-average power of less than 2.0 is possible, conservative BWR design practice dictates retention of the 2.26 figure as a design value. The increased sensitivity to control element motion and to xenon and Doppler reactivity contributions is discussed in Section 2.3 below.

In figures I-1 and I-2, numerical values are shown representing control element position, bundle power, relative power and bundle exposure, arranged in clockwise order about the symbol representing the control element. Both bundle power and bundle exposure represent averages taken axially along the 4 bundles represented by the single x-y space node. The relative powers listed were local powers taken across an x-y section of the reactor, at the axial elevation at which the over-all peak power occurred.

The values presented in figures I-1 and I-2 are representative of the power distribution which one could obtain, but they are in no sense optimized to yield the lowest possible peak-to-average power. As is evident, there are many possible control element patterns, each with its own characteristic power distribution, some worse, some better than that shown. With the large number of control elements shown and the great flexibility of a large core, shaping and positioning of the power to a desired place or magnitude within the core is practically unlimited. Throughout the study, control element position was regarded as a continuous variable, whereas in actual operation small but discrete steps are used in positioning a control element. This does not limit the validity of the quoted distributions.

Figure I-3 presents a power contour plot in the x-z directions for the initial 1000 MWe core at 0 MWD/STU. It can be seen that the power is very strongly peaked toward the bottom of the core, with "power valleys" coincident with inserted control rods, and "power peaks" between rods. Because of the high control element density toward the center of the core, a large zone of reduced power is observed there.

Axial effects, typical of a partially controlled bundle and an uncontrolled bundle are exhibited at radial nodes 8 and 9 respectively. At radial node 9, the power is forced to the bottom of the core due to high void fraction near the top of the core. The power is depressed at radial nodes 8 and 10 up to the end of the control elements at axial node 10.4. Because of the resulting low void content (low power production in the channel) off the tip of the rod at axial node 11 and because of the rapid rise in reactivity there, the power rises, and then finally, because of leakage effects, decreases. The fully inserted control elements at radial nodes 2 and 4 cause the power to be depressed in the entire channel, relative to channels containing partially inserted control elements, except at the core center where control element density is high.

The information presented in figure I-2 for the 500 MWe core indicates that substantially the same peaking factors and power distributions are obtainable in a 500 MWe core as in the 1000 MWe core. Because of the greater emphasis placed on determining the feasibility of a single core producing 1000 MWe, further core studies of the 500 MWe plant were not performed. It is evident from consideration of initial power distributions that the 1000 MWe plant suffers no major disadvantages and could yield lower design power peaking than allowed in the preliminary estimate.

## 2.3 Sensitivity to Control Element Movement

In a large core containing many critical masses, a great amount of decoupling exists when the dimensions of the core are much greater than the mean free path of fast and resonance neutrons. Thus, areas of the core exist which behave independently of each other, with the core eigenvalue determined by the most reactive area. Changes in control element position can therefore shift the position of a critical region, and in some cases, also cause the power distribution to change position abruptly. In the trial calculations to obtain a minimum peak-to-average power ratio, this type of power shifting was observed on several occasions. Figures I-4 and I-5 indicate the rather pronounced shift in the power peak from the bottom of the core (node 2: figure I-4) to the top of the core (at node 11: figure I-5), caused by the withdrawal of elements 3, 8 and 5, 8 (and their symmetry elements 8, 3 and 8, 5) 9.6 inches and 7.2 inches respectively. (From nodes 10.2 to 9.4 and from 10.2 to 9.6.) A change in radial position as well as axial position is also observed, as indicated by the rectangle enclosing the peak powers in the two figures.

A more pronounced radial shift in power, in which the axial location was unchanged from its initial position at the top of the core (node 11) is demonstrated in figures I-6 and I-7. The withdrawal of elements in positions 2, 9 and 9, 2 from an insertion of 10.5 to an insertion of 9.4 (13.2-inch increment) causes the power to shift toward the periphery. The effect of a slight asymmetry in control element position on the power distribution is also shown in figure I-7. (Compare element positions 1, 4 and 4, 1.) The peak power in each octant of the core differs appreciably in magnitude, but occurs at the same relative location.

## 2.4 Xenon and Doppler Effect on Power Distributions

In large cores, the amount of reactivity required to achieve a given degree of power flattening is less than in a small core. Thus, local xenon and fuel temperature coefficients, unimportant in small cores, became important means of power flattening in the 1000 MWe core. It is evident that control element positioning and refueling decisions require careful analysis to determine the effects of xenon and Doppler coefficient in large cores.

The increased sensitivity of large cores, to xenon and Doppler effects, is shown in figures I-8, I-10, I-11 and I-12 which illustrate the calculated power distribution for the following conditions, respectively:

1) Including void-dependent Doppler and xenon

2) Including void-dependent Dopper, excluding xenon

3) Excluding xenon, excluding void-dependent Doppler

4) Including xenon, excluding void-dependent Doppler

Figure I-9 indicates the control pattern used in the preceding conditions. By a comparison of figures I-10, I-11, and I-12 with figure I-8, the relative flattening caused by xenon and Doppler can be determined. The case excluding Doppler is of academic interest only, since full power operation without some Doppler reactivity effect is non-physical. Figure I-13 shows an axial power distribution comparison for the above cases. In each instance, the bundle containing the highest peak-to-average power was selected for the axial plot. In all cases the control element pattern was left unchanged from that of figure I-9, hence the eigenvalue is indicative of the core average effect of xenon or Doppler. The effect of void dependence of the Doppler coefficient upon the power distribution is the most pronounced effect in figure I-13, causing the peak power to shift from the bottom of the core to the top if the void dependence is removed. Calculated values of the $\Delta k/k$ due to Doppler at zero voids and 45 percent voids were 0.0098 and 0.015 respectively. The effects of xenon and Doppler are thus seen to be effective in reducing the power peaking.

## 2.5 Optimization of Axial Control Element Positioning

A number of one-dimensional axial studies were utilized to determine the effects of control element withdrawal on the power distribution and to determine the extent to which it could be optimized. Detailed three dimensional calculations indicated that:

- The axial power distribution is most difficult to control. The horizontal power tends to self-flatten because of the inherent large negative power coefficient of a BWR.

- A one-dimensional analysis results in a conservative average axial power distribution. The power distribution is flattened and the axial non-uniformity is decreased by outward, horizontal, exposure-dependent progression of the power which cannot be observed in a one-dimensional analysis.

On the basis of the preceding considerations it was concluded that if the average axial power distribution can be shown to be controllable in a one-dimensional analysis, similar conditions should be achievable in three-dimensional analysis or in operation.

The one-dimensional, axial studies indicated that the power distribution could progressively depreciate as the control elements are withdrawn if the operator has no knowledge of future operating states when he is scheduling control. Optimization of the control element patterns at each step to achieve the best possible power distribution causes continued depreciation. The reactor should be operated so that the limiting conditions are constant in time. This can only be achieved by utilizing non-uniform fuel irradiation to compensate for the withdrawal of control elements.

Non-optimum control element withdrawal is illustrated in figures I-14 through I-20 which present the spatial variation of the average core relative axial power, fractional control element insertion, and void content. Also shown is the burnout ratio in the hot channel (divided by 5). Each mesh interval represents a 3.7 inch spacing. The initial operating state represented by figure I-14 has been optimized to maximize the minimum burnout ratio by axial distribution of the control elements. This operating condition and power distribution are held relatively constant to an average core exposure of 6000 MWD/STU by movement of control elements, as shown in figure I-15. Because of greater fuel irradiation in the bottom of the core the power gradually re-distributes towards the top of the core as shown in figures I-16 and I-17 for 9000 and 12,000 MWD/STU, respectively. At 13,000 MWD/STU when the control elements are fully withdrawn the power peaks to the bottom of the core, as illustrated by figure I-18. The limiting condition, a minimum burnout ratio of 1.7, is reached at both 12,000 and 13,000 MWD/STU. Because of the void dependence of the allowable heat flux, a greater peak can be tolerated in the bottom of the core.

The space-dependence of the average relative isotopic composition and the infinite multiplication is shown in figure I-19 for an average core exposure of 13,000 MWD/STU. The plutonium concentration is relatively flat because the greater conversion in the high-void regions compensates for the reduced exposure in these zones. The peak-to-average axial exposure is less than 1.2.

The reactor was kept critical in the interval between 12000 and 13000 MWD/STU by cutting out feedwater heaters to lower the inlet temperatures. The exposure-dependence of a number of important properties of the core is shown in figure I-20.

The time-dependence of the limiting conditions can be improved by initially allowing a greater peak-to-average power, which produces non-uniform fuel irradiation to reduce subsequent increases in peak-to-average as the control elements are withdrawn. This is illustrated by figures I-21 to I-24. The power is initially peaked more strongly to the bottom of the core, figure I-21; progresses upward in the core to a lesser extent than before, figure I-22; and reaches a final relatively flat shape with all control elements withdrawn at an average exposure of 13000 MWD/STU, figure I-23. After approximately 6000 MWD/STU average exposure the control elements should be scheduled in an all-out/all-in manner. This operating pattern greatly eases operational scheduling of the control elements.

The time-dependence of several important quantities is shown in figure I-24. The limiting burnout ratio which is reached at 12,000 MWD/STU, is significantly greater than the limiting condition experienced in the alternate mode of operation. The average axial peak-to-average power decreases with exposure. As in the previous case, it is assumed that the subcooling can be increased by cutting out feedwater heaters near the end of the cycle (this condition results in a poorer heat rate). This advantageous mode of operation was not considered in determining the fuel costs. The final initial core operating state shown in figure I-23 illustrates the importance of non-uniform fuel irradiation which will tend to flatten the partial-batch scatter-refueled equilibrium reactor.

The detailed spatial burnup studies indicated greater exposure than the preliminary studies; thus providing potential for a reduction in initial fuel enrichment and a decrease in fuel costs. The studies also indicated that the one percent reactivity maneuvering allowance in the design is not needed if the reactor is operated optimally.

## 2.6 Spatially Distributed Burnable Poison

Considerably improved power distributions can be obtained by use of spatially distributed burnable poison. End-of-life power shapes similar to that illustrated by figure I-23 can be achieved on the initial core by spatial distribution of boron in temporary poison sheets.

The operating states of a core partially controlled with spatially distributed boron are shown in figures I-25 to I-29. This operating state is slightly non-optimum in that too great a burnable poison concentration was utilized in the center of the temporary poison curtains. The initial peak-to-average of 1.2, figure I-25, is maintained throughout the operating interval as shown in figures I-26 to I-29. The minimum burnout ratio of 2.2 occurs initially and improves thereafter. As in the previous cases, banking of the control elements after approximately 6000 MWD/STU provides the best operating conditions. The plutonium concentration peaks in the top of the core as shown in figure I-28.

Additional optimization should result in further improvement in the thermal capabilities. In the initial preliminary estimates of the power capability of the 1000 and 500 MWe plants no credit was taken for spatially distributed burnable poisons or for non-uniform fuel irradiation. These two considerations indicate improved power capability relative to the preliminary estimates and the reference designs.

## 3. CORE STABILITY

The term stability includes many aspects other than prevention of undamped oscillations. The reactor must be operable and follow changes in load well. Basic criteria for stability are:

- The total power must be constant without a tendency to oscillate during steady-state operation.

- The power distribution within the core must be regulated without spatial fluctuations serious enough to cause hydrodynamic problems or violation of fuel thermal limits. In addition, spatial xenon effects must be heavily damped.

- External disturbances, such as control element position changes, or load changes must be regulated without serious overshoot or power oscillation.

- The hydrodynamic-nuclear noise must be maintained at a low enough level so as not to interfere with in-core instrument indications or power output.

The major factors in design of large BWR's for stable operation are as follows:

### 3.1 Spatial Coupled Nuclear-Thermal-Hydraulic Stability

All of the effects contributing to spatial stability are not completely understood but the evidence from operation of Dresden indicates no problem areas. For example, in one test [8] Dresden was operated at void content three times rated (30 percent) on both forced and natural circulation with no apparent tendency for coupled spatial oscillations or system instability. The long fuel time constant, and large negative Doppler and void feedback are important contributors to spatial stability.

Entrance and exit effects can be important to spatial stability. Mixing in the inlet plenum must be adequate and in plants with internal steam separation the size of the exit plenum must be large enough to decouple the core from spatial effects in the steam separators.

Spatial instability problems can be promoted by incorrect loading of the reload fuel. In a large decoupled core it is undesirable to establish large independent zones in the core which have appreciably different properties. For example; out-in, few-zone refueling can result in an

infinite multiplication ($k_\infty$) distribution in which $k_\infty$ is less than 1.0 in the central zone near the end of a refueling interval. In a large decoupled core the different zones will respond differently to external flow and subcooling perturbations. A "scatter" refueling pattern which distributes the fuel of different exposure levels uniformly over the core is preferable from the standpoint of maintaining spatial stability. Studies of this means of refueling are discussed in a following section.

### 3.2 Spatial Xenon Stability

The large negative power coefficient of reactivity, which is characteristic of a BWR, provides strong damping of xenon. In a recent paper Chernick[10] discusses the importance of the power coefficient in maintaining stability to xenon. In a BWR the ratio of the power reactivity defect to the xenon defect is approximately 1.0, which, on the basis of linear theory, guarantees stability to xenon.

At Dresden spatial xenon shifts have been observed following a dual-cycle load change. The shift lasts for approximately 15 hours and then disappears. Therefore, xenon is heavily damped in the Dresden core.

In spatial dynamics analyses which have been completed for large BWR's it has not been possible to excite spatial xenon effects. This is discussed below.

### 3.3 System Stability

Over-all system stability, as determined by parameters such as transit times and pressure drops in the recirculation loops, the over-all void reactivity and Doppler feedback gains, the relative pressure drops in the core and steam separators and the size of the plenum between the core and steam separators, is considered to be the major stability limit on design. Bounding calculations of system stability for large BWR's indicate that stability margin can be optimized by variation of the inlet orificing of the core, modification of the core lattice design to change the feedback gain, variation of the size of the plenum above the core, modification of pump characteristics, and variation of the design of the internal steam separators.

The desired trend in the core design is to reduce the negative void feedback gain to the minimum acceptable from the standpoint of spatial stability to xenon and from considerations of providing good load-following capability. The negative feedback gain of the core is optimized by adjustment of parameters such as water-to-fuel volume ratio, core void content, inlet subcooling, recirculation flow, and fuel rod diameter. Water-to-fuel volume ratio is the most important independent parameter which is varied to achieve acceptable reactivity coefficients. A gain margin of two and a phase margin of 45 degrees relative to the unstable threshold are considered suitable margins to provide good operability.



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The major problem in design for a stable system is development of a detailed understanding of all the contributing factors so that the system can be pushed to maximum performance. There appear to be no fundamental factors which prevent adjustment of the variables so as to achieve stable performance. Operating BWR's such as Dresden and Big Rock Point have appreciable stability margin.

Results from studies of the above factors influencing core stability are discussed in the sections which follow for the core of Reference Plant Arrangement A.

## 3.4 Power Coefficient and Spatial Xenon Stability

In a BWR, the bulk of the coolant is at saturation and as a result the BWR has a large negative power coefficient of reactivity. The water density change per unit enthalpy change is approximately a factor of ten greater in a boiling system than in a non-boiling system. The power coefficient in typical BWR's ranges from approximately -1 to $-7 \times 10^{-4}$ $\Delta k/k$ per Kw/liter, depending on the mode and magnitude of perturbation. The calculated value for the 1000 MWe core was $-2.88 \times 10^{-4}$ $\Delta k/k$ per Kw/liter for a 10 percent decrease in power. A large negative power coefficient is desirable from the standpoint of providing good load-following response and damping of spatial xenon oscillations. For stability it is desirable that the feedback from power to reactivity be negative and small.

Studies [6] of the effect of spatial xenon oscillation in a large BWR have shown that a large negative power coefficient, greater than -0.014 $\Delta k/k$ per $\Delta P/P$ insures stability against xenon oscillations. In addition, it has been shown [7] that flow control, by perturbing the entire reactor, is the most effective means to follow load variations. The spatial stability threshold is reduced by use of control elements for load following.[7]

## 3.5 Flow Control

Reactor control by means of control element insertion is the usual means of short and long-term reactor control. However, control elements perturb only the local regions of the core, as demonstrated in Section 2.3 and the control of a large core, in which small differences in reactivity can lead to appreciable shifts in the power distribution, cause a very difficult optimization problem. Control of the reactor by means of changes in the coolant flow rate is then of interest for large cores because a coolant flow change perturbs the entire reactor, leading to a reasonably uniform reactivity insertion or removal, in the horizontal planes. An important feature of core response following a flow change is that the power and void distributions tend to return to approximately the same initial state after the transient effects have subsided.

Figure I-30 indicates the steady state power and flow distributions which result following a power decrease of 10 percent. In obtaining data presented in figure I-30 the power was reduced and an initial eigenvalue was calculated. The core flow was then adjusted such that the eigenvalue

returned to 1.0 ± 0.005. Comparison may be made between figures I-30 and I-31, for reduced flow and for rated flow, respectively, and an eigenvalue of 1.004. The power and void distribution (figure I-31) does not shift appreciably for the reduced power represented by figure I-30. The small differences, usually less that 5 percent can be attributed to the void dependence of the Doppler coefficient. If no Doppler effect were present, heat balance considerations require that void and power distribution return to their initial state.

### 3.6 Transient Effects Using Flow Control

The response of the 500 MWe core to a 50 percent negative 10 second flow ramp is shown in figure I-32. The minimum burnout ratio rises throughout the transient and the response is smooth. The heat power closely follows the flow decrease. The steady state axial peak-to-average fission power increases and the power distribution shifts slightly toward the bottom of the core.

The power range which can be controlled is shown in figure I-33 as a function of percent flow change for the 500 MWe core in the clean condition. The power-flow relationship is linear for this particular design and operating state. The model is idealized in that it is assumed that there is no change in inlet temperature when the flow is changed. Other variables such as control of the water level in the vessel and steam separation appear to be more limiting than the core with respect to determining the maximum power range which can be controlled with flow.

## 4. THREE-DIMENSIONAL EXPOSURE STUDIES AND FUEL MANAGEMENT

To determine the exposure behavior of the 1000 MWe core, two sets of three dimensional exposure calculations were made. In the first, no attempt was made to hold the peak-to-average power ratio to within current design limits, by movement of control elements. A non-restrictive limit on the allowable variation of the core eigenvalue in an exposure step before adjustment to criticality once again by movement of control elements was used, and varied by at least 0.02 $\Delta k$. Adjustment to criticality at each succeeding exposure step was made such that the initial eigenvalue was 1.0 ± 0.01. However, control element interchanges, important to maintain a uniform exposure distribution and minimize power peaking off the tips, were performed at intervals of 1000 MWD/STU, at every exposure step. In the second exposure set, the peak-to-average power ratio was constrained to a value less than 2.0, the allowable change in eigenvalue at a burnup step was held less than or equal to 0.01 $\Delta k$, resulting in 500 MWD/STU exposure steps, and the initial eigenvalue at the beginning of each burnup step was adjusted such that as burnup progressed, the eigenvalue stayed centered about 1.0, with a deviation of 0.005 above and below this value. Control element interchanges were performed as in Exposure Set 1, at every 1000 MWD/STU, or every other exposure step.

Because the first set of exposure calculations was considerably less restrictive, exposure to the end of first core life was possible. It should be noted here that the calendar time, and to some extent the computing time, for a three dimensional exposure study, without automated control element movements, approaches infinity as the restrictions governing the study are increased in severity. At each burnup step, adjustment of the control elements to yield a specific value of peak-to-average power and an eigenvalue within acceptable limits often required 5 to 10 times the amount of computing time as the burnup step itself.

The second exposure set was an attempt to make the exposure calculation represent as nearly as possible actual operating conditions during exposure, and hence more restrictive assumptions were used. While all possible means were used to expedite the calculations, it was found that with the restrictions outlined above, the calculation of the burning of 1000 MWD/STU could only be made in a week of calendar time. Exposure to a total burnup of only 5000 MWD/STU was obtained because of time limitations. However, the results from the study indicated that, with good in-core monitoring, the actual reactor could be operated safely without exceeding the current design margins.

A comparison of the two sets of exposure studies is given in Tables I-2 through I-5, and plotted in figures I-34, I-35, and I-36. It is again emphasized that Set 1, although carried out until the end-of-life for core one, does not represent obtainable peak-to-average power - but it does indicate the magnitude of possible peaking factors which would be obtained if careful control element programming were not used. Figure I-35 indicates the improvement in the peak-to-average channel power with exposure, decreasing approximately linearly with exposure. The peak-to-average channel power at 10,000 MWD/STU is significantly lower (1.13) than the design value of 1.3; however, no benefit from the lower value was included in the design. Thus, considerably improved power capability relative to the preliminary estimates used in the reference design is indicated.

The initial enrichment used in both these studies was 2.27 atom percent U-235; the final exposure for a single BATCH of the first core was 10,000 MWD/STU. With multiple (5) batch fuel management, there are two possible approaches left open to the designer. A reduction of enrichment, or an increase in exposure for a given enrichment is possible with multiple-batch fuel management. Considerations of the desired refueling interval (1 year for the 1000 MWe plant) usually indicate that it is preferable, in the transition to the equilibrium core, to reduce the initial enrichment of the initial core, followed by an increase in enrichment of the reload batches until equilibrium is obtained. The enrichment for the initial 1000 MWe core was determined to be 2.05 wt percent for an exposure of 15,000 MWD/STU and 2.4 wt percent to attain an equilibrium exposure of 20,000 MWD/STU. Initial enrichments, U-235 depletion, and plutonium discharged for the 1000 and 500 MWe core are shown in Tables V-2 of Volume 1 and XI-1 of Volume 2 (these Tables are identical).

TABLE I-2

Peak-to-Average Power, Location, and Initial and Final Eigenvalues for Exposure Set 1.
No Restrictions on Maximum Power, Exposure Step Length 1000 MWD/STU,
Maximum Change in Eigenvalue Approximately 2 Percent,
Control Element Interchange Every 1000 MWD/STU

| Core Average Exposure (MWD/STU) | Peak/Avg. Power | Location X, Y, Z, node | Peak/Avg. Channel Power | Location (X, Y) | Initial Eigenvalue | Final Eigenvalue |
|---|---|---|---|---|---|---|
| 0 | 2.78 | 1, 1, 2 | 1.44 | 9, 1 | 0.9949 | |
| 1,000 | 2.39 | 1, 1, 2 | 1.41 | 9, 1 | | 0.9800 |
| Control Element Interchange | | | | | | |
| 1,000 | 2.35 | 9, 1, 11 | 1.37 | 10, 1 | 1.012 | |
| 2,000 | 2.08 | 9, 1, 11 | 1.36 | 10, 1 | | 0.9919 |
| Control Element Interchange | | | | | | |
| 2,000 | 2.04 | 2, 8, 11 | 1.28 | 9, 1 | 0.9916 | |
| 3,000 | 1.82 | 1, 8, 11 | 1.26 | 9, 1 | | 0.9721 |
| Control Element Interchange | | | | | | |
| 3,000 | 2.55 | 6, 1, 10 | 1.24 | 8, 1 | 1.015 | |
| 4,000 | 2.17 | 7, 1, 10 | 1.23 | 8, 1 | | 0.9887 |
| Control Element Interchange | | | | | | |
| 4,000 | 2.21 | 6, 5, 10 | 1.26 | 1, 1 | 0.9885 | |
| 5,000 | 1.91 | 6, 5, 10 | 1.25 | 1, 1 | | 0.9649 |
| Control Element Interchange | | | | | | |
| 5,000 | 2.59 | 1, 1, 10 | 1.21 | 6, 1 | 0.9987 | |
| 6,000 | 2.17 | 1, 1, 10 | 1.19 | 6, 1 | | 0.9730 |
| Control Element Interchange | | | | | | |
| 6,000 | 2.51 | 2, 1, 9 | 1.27 | 1, 1 | 1.005 | |
| 7,000 | 2.13 | 2, 1, 9 | 1.24 | 1, 1 | | 0.9740 |
| Control Element Interchange | | | | | | |
| 7,000 | 2.44 | 1, 1, 8 | 1.19 | 4, 1 | 0.9977 | |
| 8,000 | 2.09 | 1, 1, 9 | 1.16 | 6, 1 | | 0.9715 |
| Control Element Interchange | | | | | | |
| 8,000 | 2.32 | 6, 5, 6 | 1.20 | 1, 1 | 0.9905 | |
| 9,000 | 2.04 | 6, 5, 6 | 1.18 | 1, 1 | | 0.9659 |
| Control Element Interchange | | | | | | |
| 9,000 | 2.37 | 5, 5, 5 | 1.15 | 4, 1 | 0.9978 | |
| 10,000 | 2.17 | 7, 6, 4 | 1.13 | 4, 1 | | 0.9732 |

TABLE I-3

Peak-to-Average Power, Location, and Initial and Final Eigenvalues for Exposure Set 2. Peak-to-Average Power Kept Less Than 2.0; Exposure Step Length 500 MWD/STU, Maximum Change in Eigenvalue Approximately 1 Percent, Control Element Interchange Every 1000 MWD/STU

| Core Average Exposure (MWD/STU) | Peak/Avg. Power | Location X, Y, Z, node | Peak/Avg. Channel Power | Location (X, Y) | Initial Eigenvalue | Final Eigenvalue |
|---|---|---|---|---|---|---|
| 0 | 1.89 | 7, 1, 2 | 1.33 | 9, 1 | 1.0036 | |
| 500 | 1.83 | 7, 1, 3 | 1.33 | 9, 1 | | 0.9958 |
| 500 | 1.95 | 9, 2, 11 | 1.34 | 9, 1 | 1.0021 | |
| 1,000 | 1.92 | 9, 2, 11 | 1.33 | 9, 1 | | 0.9941 |
| Control Element Interchange | | | | | | |
| 1,000 | 1.74 | 10, 1, 3 | 1.29 | 10, 1 | 1.0053 | |
| 1,500 | 1.73 | 10, 1, 3 | 1.28 | 10, 1 | | 0.9969 |
| 1,500 | 1.68 | 8, 8, 5 | 1.18 | 10, 1 | 1.0023 | |
| 2,000 | 1.65 | 8, 8, 5 | 1.18 | 10, 1 | | 0.9949 |
| Control Element Interchange | | | | | | |
| 2,000 | 1.95 | 5, 4, 11 | 1.20 | 9, 1 | 1.0031 | |
| 2,500 | 1.86 | 8, 7, 10 | 1.20 | 9, 1 | | 0.9930 |
| 2,500 | 1.96 | 10, 1, 10 | 1.22 | 9, 1 | 1.0052 | |
| 3,000 | 1.92 | 7, 8, 6 | 1.22 | 9, 1 | | 0.9953 |
| Control Element Interchange | | | | | | |
| 3,000 | 1.92 | 1, 9, 10 | 1.38 | 10, 1 | 1.0051 | |
| 3,500 | 1.85 | 1, 9, 10 | 1.37 | 10, 1 | | 0.9957 |
| 3,500 | 1.98 | 5, 5, 11 | 1.25 | 10, 1 | 1.0072 | |
| 4,000 | 1.89 | 5, 5, 11 | 1.25 | 10, 1 | | 0.9974 |
| Control Element Interchange | | | | | | |
| 4,000 | 1.84 | 1, 1, 11 | 1.30 | 10, 1 | 1.0024 | |
| 4,500 | 1.74 | 1, 1, 11 | 1.29 | 10, 1 | | 0.9922 |
| 4,500 | 1.93 | 3, 5, 10 | 1.25 | 10, 1 | 1.004 | |
| 5,000 | 1.84 | 3, 5, 10 | 1.25 | 10, 1 | | 0.9941 |

TABLE I-4

Ratios of Maximum and Minimum Exposure to Average Exposure, Set 1.

| Core Average Exposure (MWD/STU) | $E_{max}/E_{avg}$ | Location (X, Y, Z) | $E_{min}/E_{avg}$ | Location (X, Y, Z) |
|---|---|---|---|---|
| 0 | 0 | | 0.0 | - |
| 1,000 | 2.78 | 1, 1, 2 | 0.28 | 8, 9, 1 |
| 2,000 | 1.98 | 9, 1, 11 | 0.24 | 8, 9, 1 |
| 3,000 | 2.0 | 9, 2, 11 | 0.22 | 8, 9, 1 |
| 4,000 | 1.98 | 9, 1, 11 | 0.18 | 8, 9, 1 |
| 5,000 | 1.97 | 9, 1, 10 | 0.17 | 8, 9, 1 |
| 6,000 | 1.95 | 9, 2, 10 | 0.16 | 8, 9, 1 |
| 7,000 | 1.86 | 9, 1, 10 | 0.15 | 8, 9, 1 |
| 8,000 | 1.76 | 9, 2, 10 | 0.14 | 8, 9, 1 |
| 9,000 | 1.65 | 9, 2, 10 | 0.14 | 8, 9, 1 |
| 10,000 | 1.52 | 9, 2, 10 | 0.15 | 8, 9, 1 |

TABLE I-5

Ratios of Maximum and Minimum Exposure to Average Exposure, Set 2.

| Core Average Exposure (MWD/STU) | $E_{max}/E_{avg}$ | Location (X, Y, Z) | $E_{min}/E_{avg}$ | Location (X, Y, Z) |
|---|---|---|---|---|
| 0 | 0 | - | 0 | - |
| 500 | 1.88 | 7, 2, 2 | 0.15 | 8, 9, 1 |
| 1,000 | 1.86 | 9, 2, 11 | 0.15 | 8, 9, 1 |
| 1,500 | 1.75 | 9, 2, 11 | 0.21 | 8, 9, 1 |
| 2,000 | 1.65 | 9, 2, 11 | 0.25 | 8, 9, 1 |
| 2,500 | 1.70 | 9, 2, 11 | 0.27 | 8, 9, 1 |
| 3,000 | 1.73 | 9, 2, 11 | 0.28 | 8, 9, 1 |
| 3,500 | 1.72 | 9, 2, 11 | 0.29 | 8, 9, 1 |
| 4,000 | 1.72 | 8, 2, 11 | 0.30 | 8, 9, 1 |
| 4,500 | 1.72 | 8, 2, 11 | 0.33 | 8, 9, 1 |
| 5,000 | 1.70 | 8, 2, 11 | 0.35 | 8, 9, 1 |

A comparison of the ratios $E_{max}/E_{avg}$ and $E_{min}/E_{avg}$ (the ratio of maximum exposure to average exposure and the ratio of minimum exposure to average exposure, respectively) for two sets of exposure studies indicate that a substantial increase in final exposure is possible for Set 2, due to maintaining a considerably better peak-to-average power ratio throughout life. Exposure and power distribution are shown at exposure steps of 1000 MWD/STU for Exposure Set 2 in figures I-37 through I-43.

Since Exposure Set 1 was carried to the end-of-core-life, the first core reload refueling operation was performed and calculated. New fuel consisting of 1/5 of the core loading was loaded into the peripheral position designated as "new" in Table I-6. The exposure distribution, as well as the control element setting and power distribution after bringing the core critical are shown in figure I-44. Because at the first refueling the core contains fuel of high exposure in the center and zero exposure on the periphery, more relative insertion of elements near the periphery fuel is necessary to control the reactor and to prevent excessive peaking on the periphery. It is noteworthy that the peripheral elements themselves are not required to be inserted to control the power peaking. Computer and calendar time limitations prevented continuing the refueling until equilibrium was attained; however, on successive refueling to equilibrium, the exposure level of the most exposed fuel varies, and hence the well known transient behavior of the discharged fuel exposure is obtained.

TABLE I-6

First Core Refueling, with New Fuel Placed in Periphery and A One-in-Four Scatter Pattern Used in Central Region

A. Original Fuel Location (One Quadrant of Core)

| | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| 1, 1 | 1, 2 | 1, 3 | 1, 4 | 1, 5 | 1, 6 | 1, 7 | 1, 8 | 1, 9 | 1, 10 | 1, 11 |
| 2, 1 | 2, 2 | 2, 3 | 2, 4 | 2, 5 | 2, 6 | 2, 7 | 2, 8 | 2, 9 | 2, 10 | 2, 11 |
| 3, 1 | 3, 2 | 3, 3 | 3, 4 | 3, 5 | 3, 6 | 3, 7 | 3, 8 | 3, 9 | 3, 10 | 3, 11 |
| 4, 1 | 4, 2 | 4, 3 | 4, 4 | 4, 5 | 4, 6 | 4, 7 | 4, 8 | 4, 9 | 4, 10 | 4, 11 |
| 5, 1 | 5, 2 | 5, 3 | 5, 4 | 5, 5 | 5, 6 | 5, 7 | 5, 8 | 5, 9 | 5, 10 | |
| 6, 1 | 6, 2 | 6, 3 | 6, 4 | 6, 5 | 6, 6 | 6, 7 | 6, 8 | 6, 9 | 6, 10 | |
| 7, 1 | 7, 2 | 7, 3 | 7, 4 | 7, 5 | 7, 6 | 7, 7 | 7, 8 | 7, 9 | | |
| 8, 1 | 8, 2 | 8, 3 | 8, 4 | 8, 5 | 8, 6 | 8, 7 | 8, 8 | 8, 9 | | |
| 9, 1 | 9, 2 | 9, 3 | 9, 4 | 9, 5 | 9, 6 | 9, 7 | 9, 8 | | | |
| 10, 1 | 10, 2 | 10, 3 | 10, 4 | 10, 5 | 10, 6 | | | | | |
| 11, 1 | 11, 2 | 11, 3 | 11, 4 | | | | | | | |

B. Refuel Fuel Location (Boxes Show Fuel Assembly Locations Following First Batch Scatter Reload)

| | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| [5, 9] | 1, 2 | [1, 11] | 1, 4 | [2, 11] | 1, 6 | [3, 11] | 1, 8 | [4, 11] | 1, 10 | New |
| 2, 1 | 2, 2 | 2, 3 | 2, 4 | 2, 5 | 2, 6 | 2, 7 | 2, 8 | 2, 9 | 2, 10 | New |
| [11, 1] | 3, 2 | [9, 5] | 3, 4 | [7, 9] | 3, 6 | [8, 9] | 3, 8 | [5, 10] | 3, 10 | New |
| 4, 1 | 4, 2 | 4, 3 | 4, 4 | 4, 5 | 4, 6 | 4, 7 | 4, 8 | 4, 9 | 4, 10 | New |
| [11, 2] | 5, 2 | [9, 7] | 5, 4 | [6, 9] | 5, 6 | [6, 10] | 5, 8 | New | New | |
| 6, 1 | 6, 2 | 6, 3 | 6, 4 | 6, 5 | 6, 6 | 6, 7 | 6, 8 | New | New | |
| [11, 3] | 7, 2 | [9, 8] | 7, 4 | [10, 6] | 7, 6 | [9, 6] | 7, 8 | New | | |
| 8, 1 | 8, 2 | 8, 3 | 8, 4 | 8, 5 | 8, 6 | 8, 7 | 8, 8 | New | | |
| [11, 4] | 9, 2 | [10, 5] | 9, 4 | New | New | New | New | | | |
| 10, 1 | 10, 2 | 10, 3 | 10, 4 | New | New | | | | | |
| New | New | New | New | | | | | | | |

## REFERENCES


1. Fischer, D. L., and Harriman, J. M., "Simplified Analytical Model for Similation of Boiling Water Reactors" paper presented at San Francisco ANS meeting on Nuclear Performance of Power Reactor Cores, September, 1963.

2. Fischer, D. L., and Crowther, R. L., "Nuclear Characteristics of Large Advanced Boiling Water Reactors" paper presented at San Francisco ANS meeting on Nuclear Performance of Power Reactor Cores, September, 1963.

3. Kalman, M. P., "The Calculation of the Humboldt Bay Beginning-of-Life Power Distribution and a Comparison with Gamma Probe Measurements," General Electric, Unpublished, October 1, 1963.

4. West, D. L., "Analysis of the Dresden Half-Power Gamma Probe Data using QED (FLARE)" GE (APED) Unpublished, July 28, 1963.

5. Fischer, D. L., and Bellan, R. T., "FLARE Calculations of RWE Lifetime Performance," GE (APED) Unpublished, October 23, 1963.

6. Fischer, D. L., and Bellan, R. T., "On Xenon Transients in Large Power Reactors" GE (APED) Unpublished, September 23, 1963.

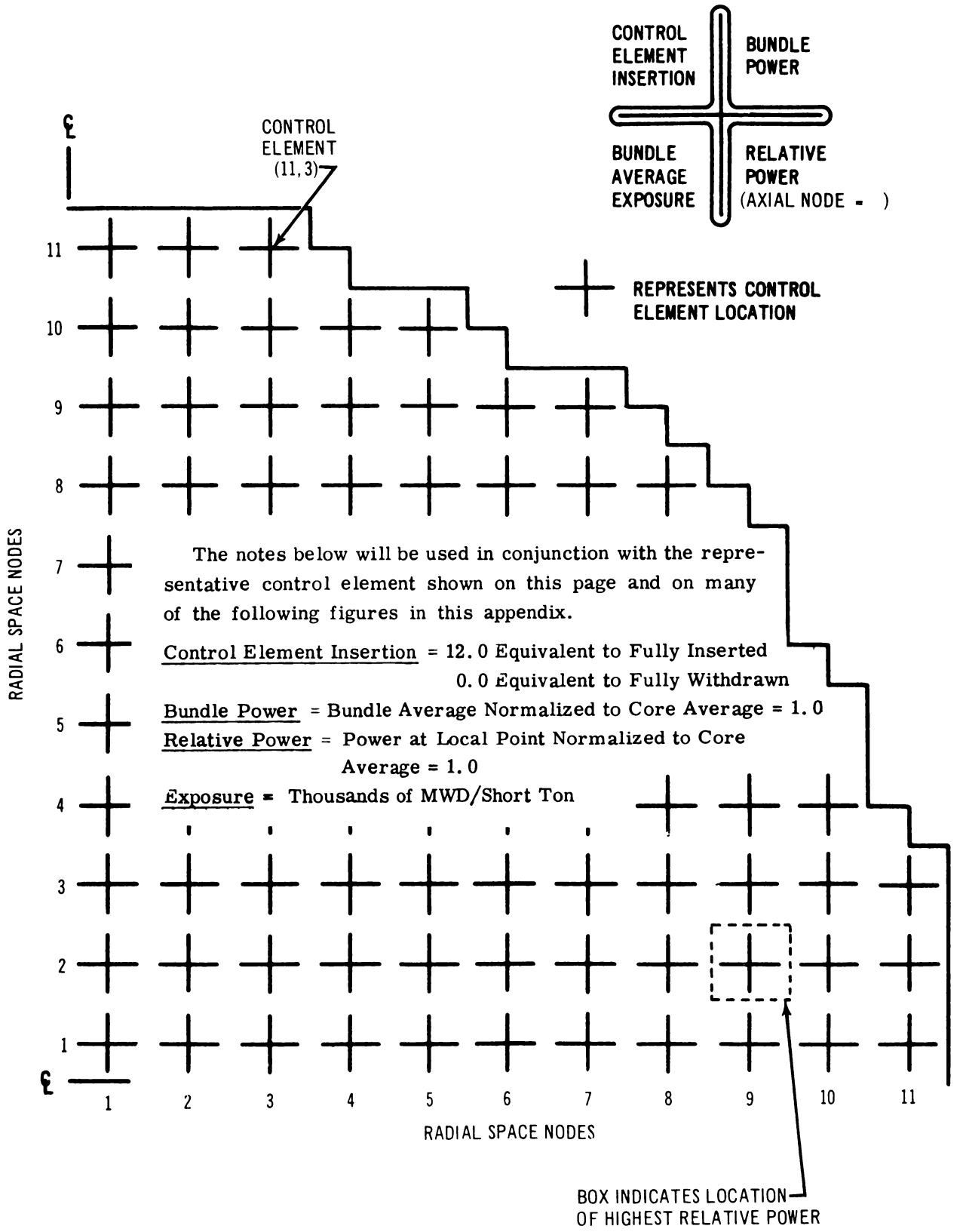
7. Fischer, D. L., Personal communication on controlled Xenon transients using flow control and control rods, December 10, 1963.

8. Bray, A. P.,Collett, W. I., Cook, W. H., Parkos, G. R., Pflasterer, G. R., and Owen, E. R., Dresden High Void Test, Unpublished, 1961.

9. Brammer, H. A., Parkos, G. R., "Refueling Methods for Large Boiling Water Reactors" paper presented at San Francisco ANS meeting at Nuclear Performance of Power Reactor Cores, September, 1963.

10. Chernick, J., "A Review of Non-Linear Reactor Dynamics Problems," BNL 774 (T-291) (1962).

11. Crowther, R. L., The Power Control Method for Solution of Diffusion Equations Describing Large Non-Linear Inhomogeneous Reactors. Transactions ANS, 6 No. 2 (November, 1963).


CONTROL ELEMENT INSERTION
BUNDLE POWER
BUNDLE AVERAGE EXPOSURE
RELATIVE POWER (AXIAL NODE - )
CONTROL ELEMENT (11,3)
REPRESENTS CONTROL ELEMENT LOCATION
The notes below will be used in conjunction with the representative control element shown on this page and on many of the following figures in this appendix.
Control Element Insertion = 12.0 Equivalent to Fully Inserted
0.0 Equivalent to Fully Withdrawn
Bundle Power = Bundle Average Normalized to Core Average = 1.0
Relative Power = Power at Local Point Normalized to Core Average = 1.0
Exposure = Thousands of MWD/Short Ton
RADIAL SPACE NODES
RADIAL SPACE NODES
BOX INDICATES LOCATION OF HIGHEST RELATIVE POWER

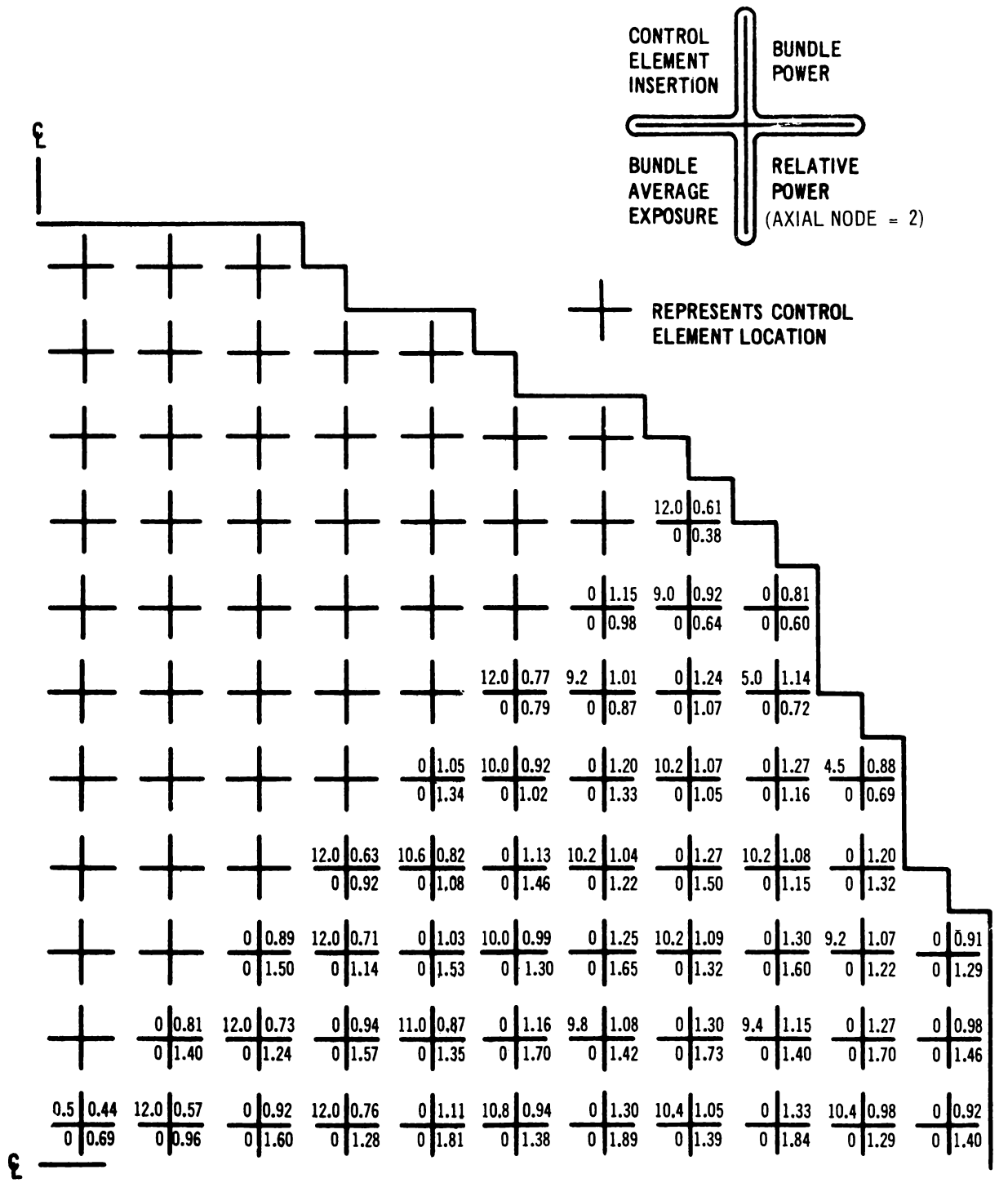



Figure I-1. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 0, Eigenvalue = 1.004

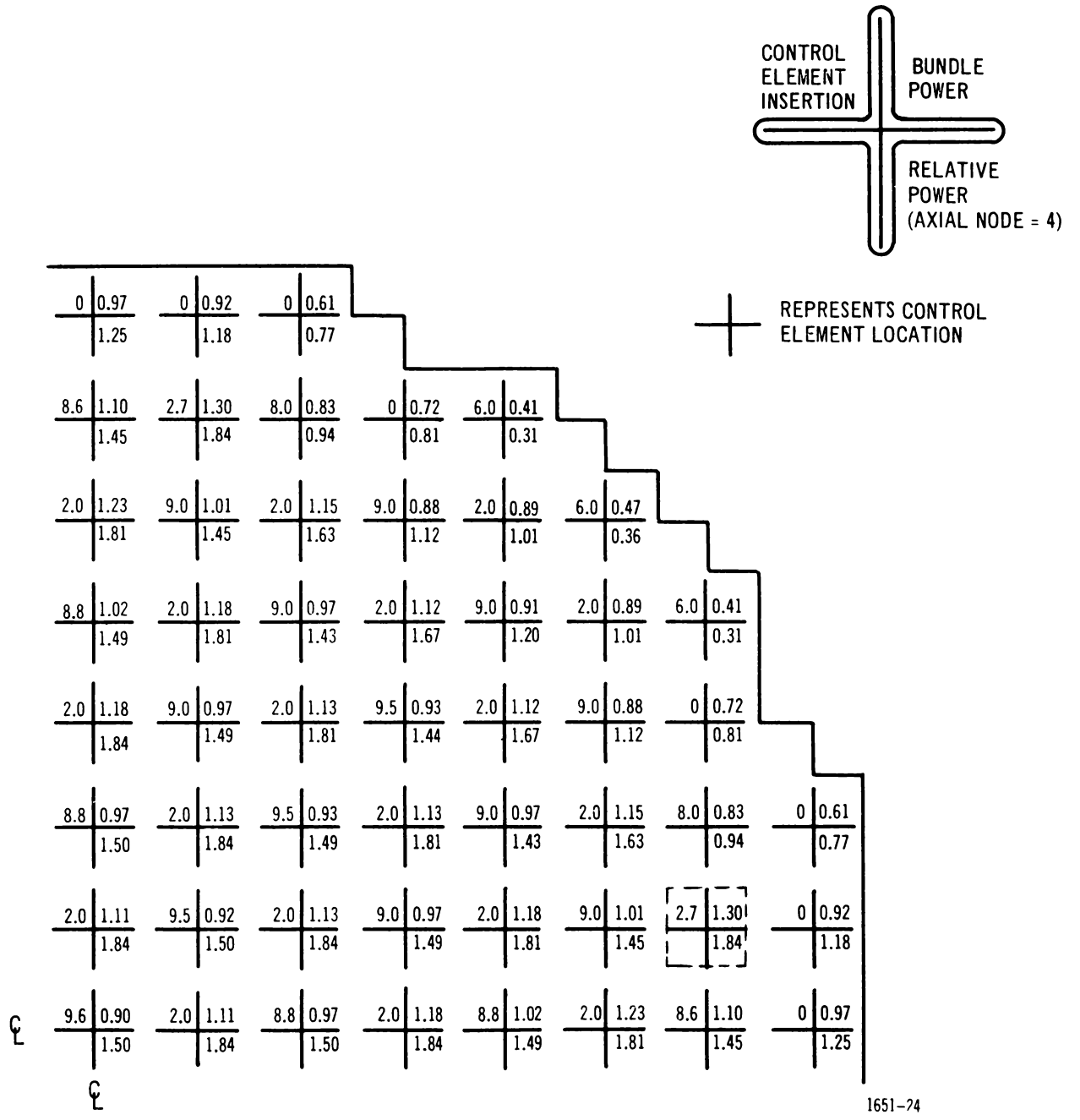



Figure I-2. 500 MWe Core, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, Core Average Exposure = 0, Eigenvalue = 1.004

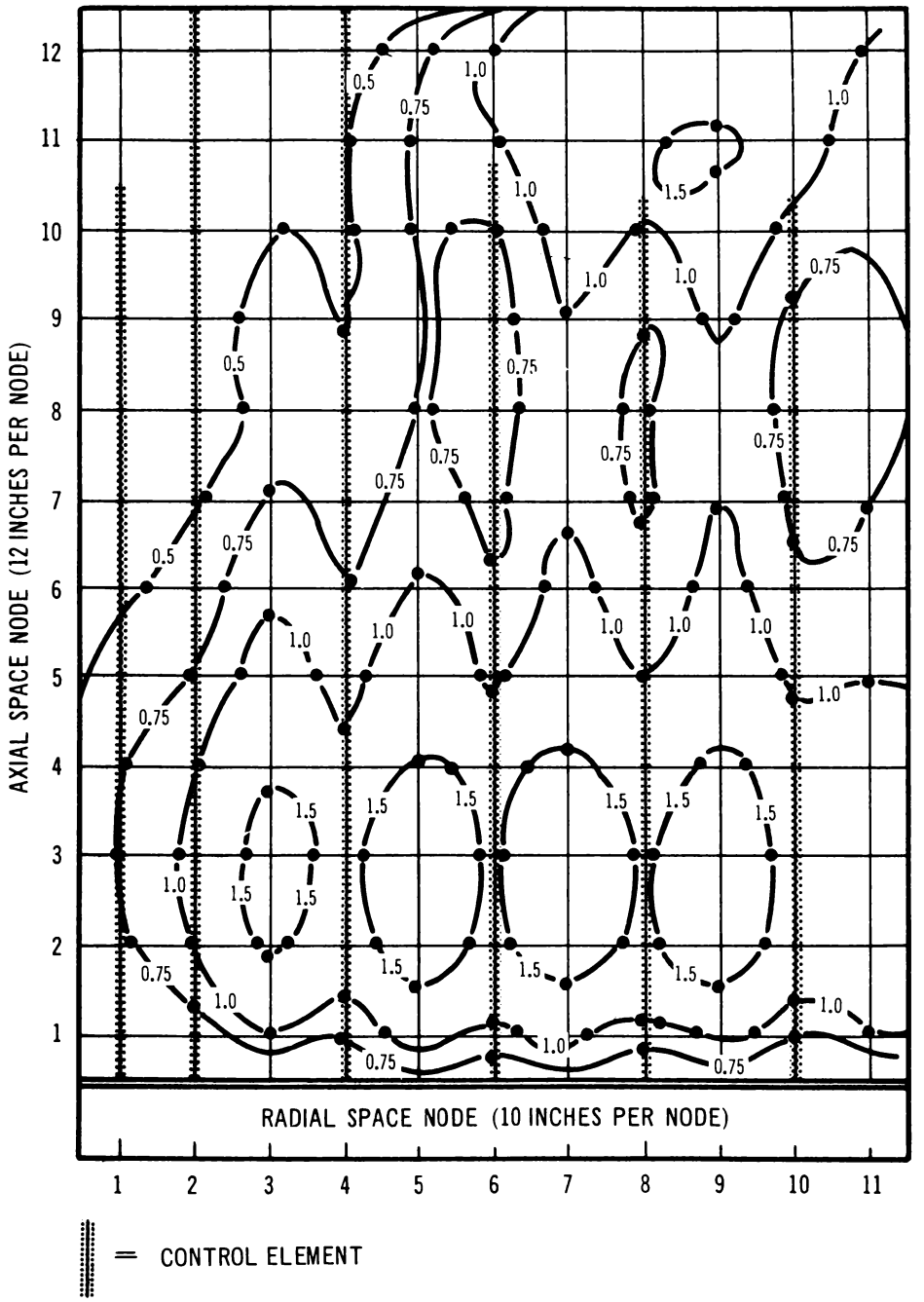



Figure I-3. Power Contours for the Initial Core at Beginning of Life

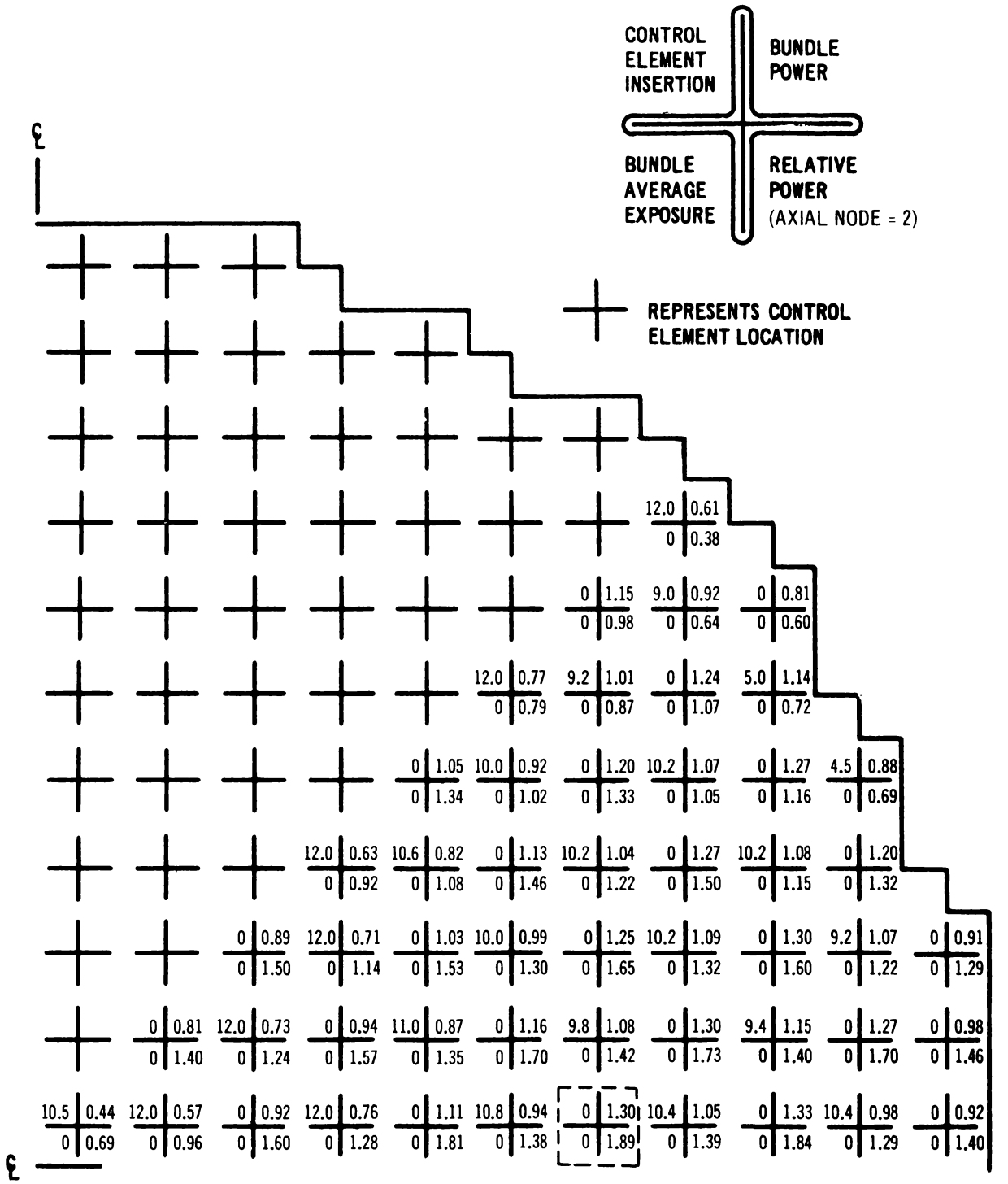



Figure I-4. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 0, Eigenvalue = 1.004

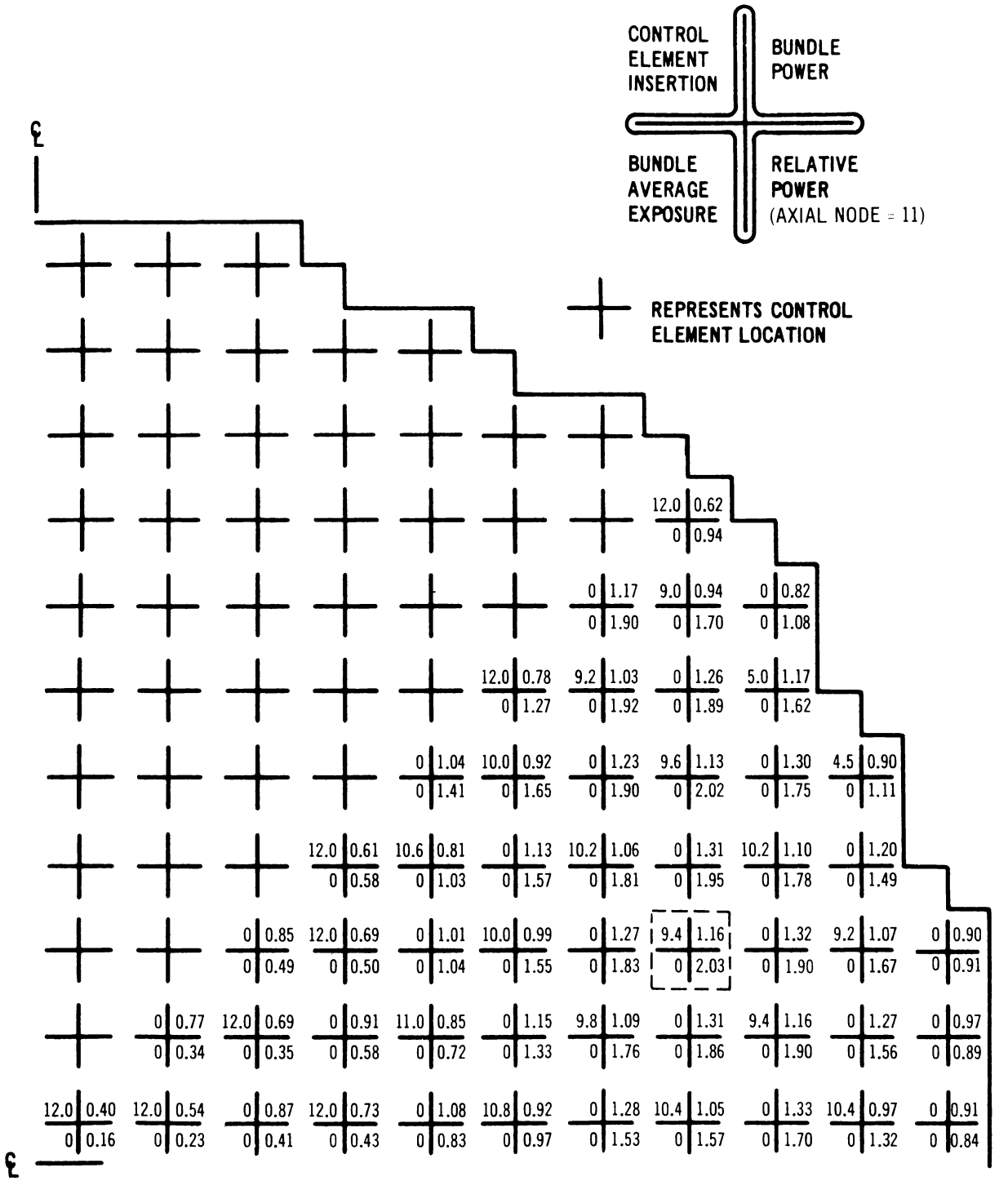



Figure I-5. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 0, Eigenvalue = 1.007

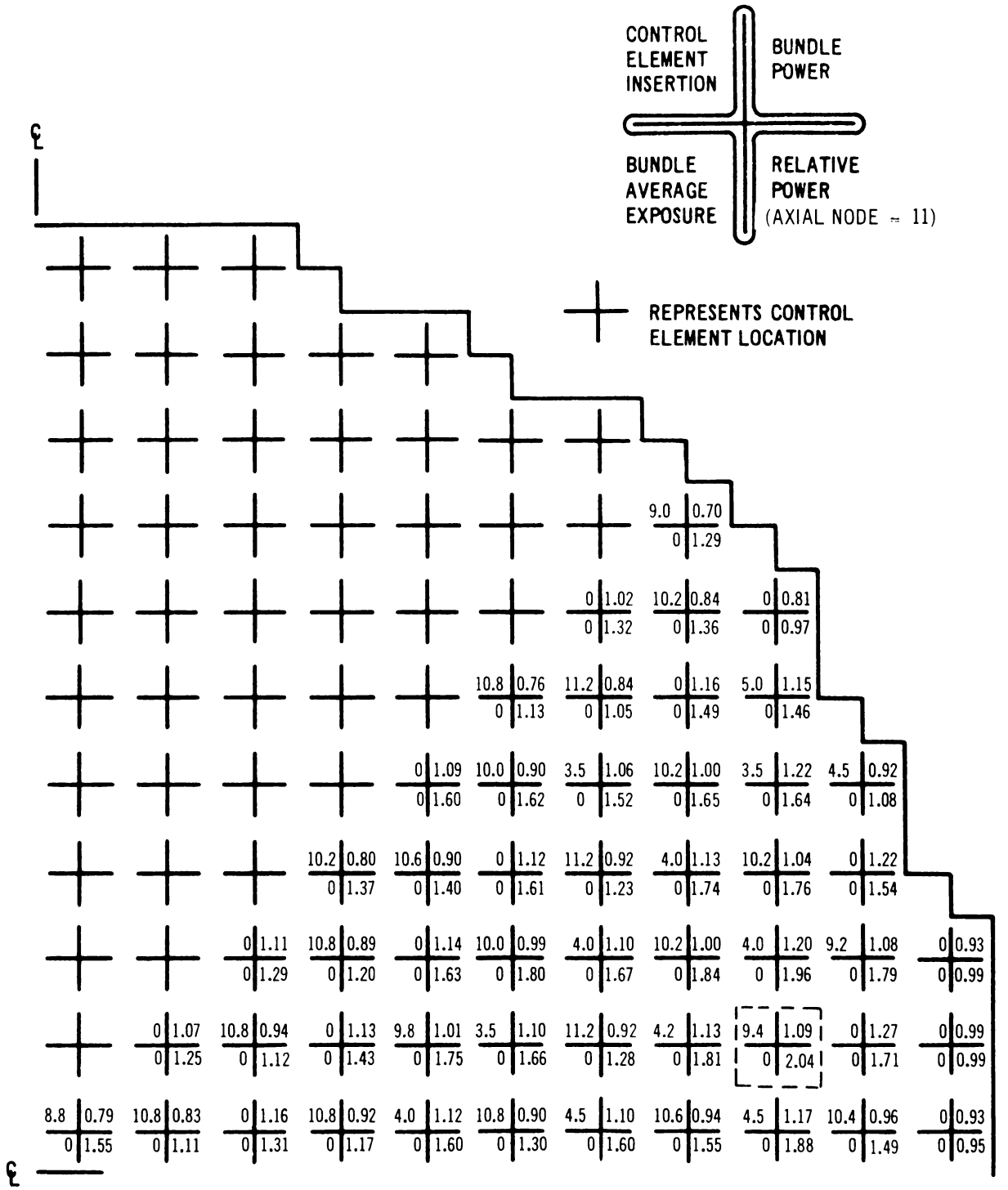



Figure I-6. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure. Core Average Exposure = 0, Eigenvalue = 0.998



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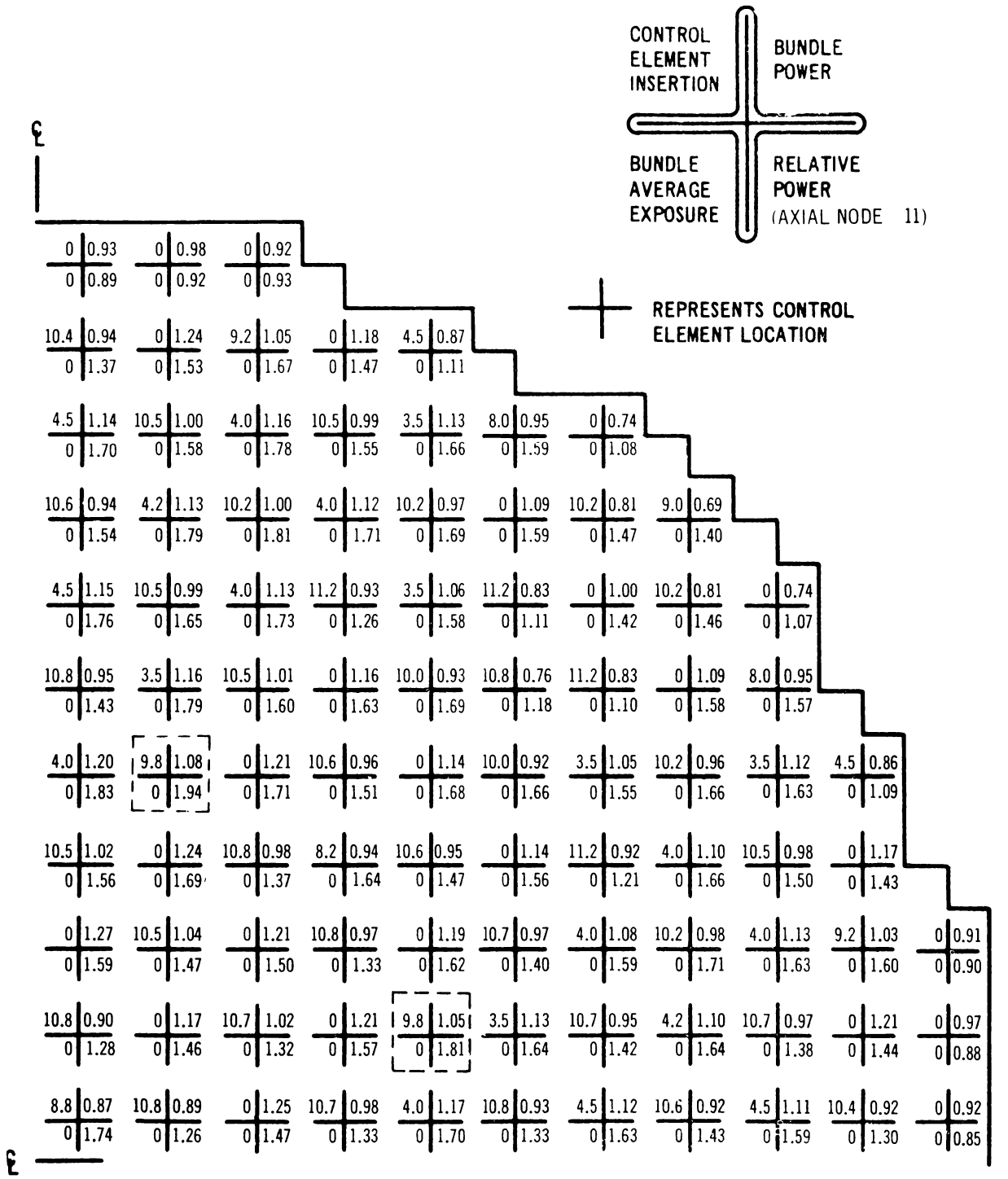


Figure I-7. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 0, Eigenvalue = 0.996

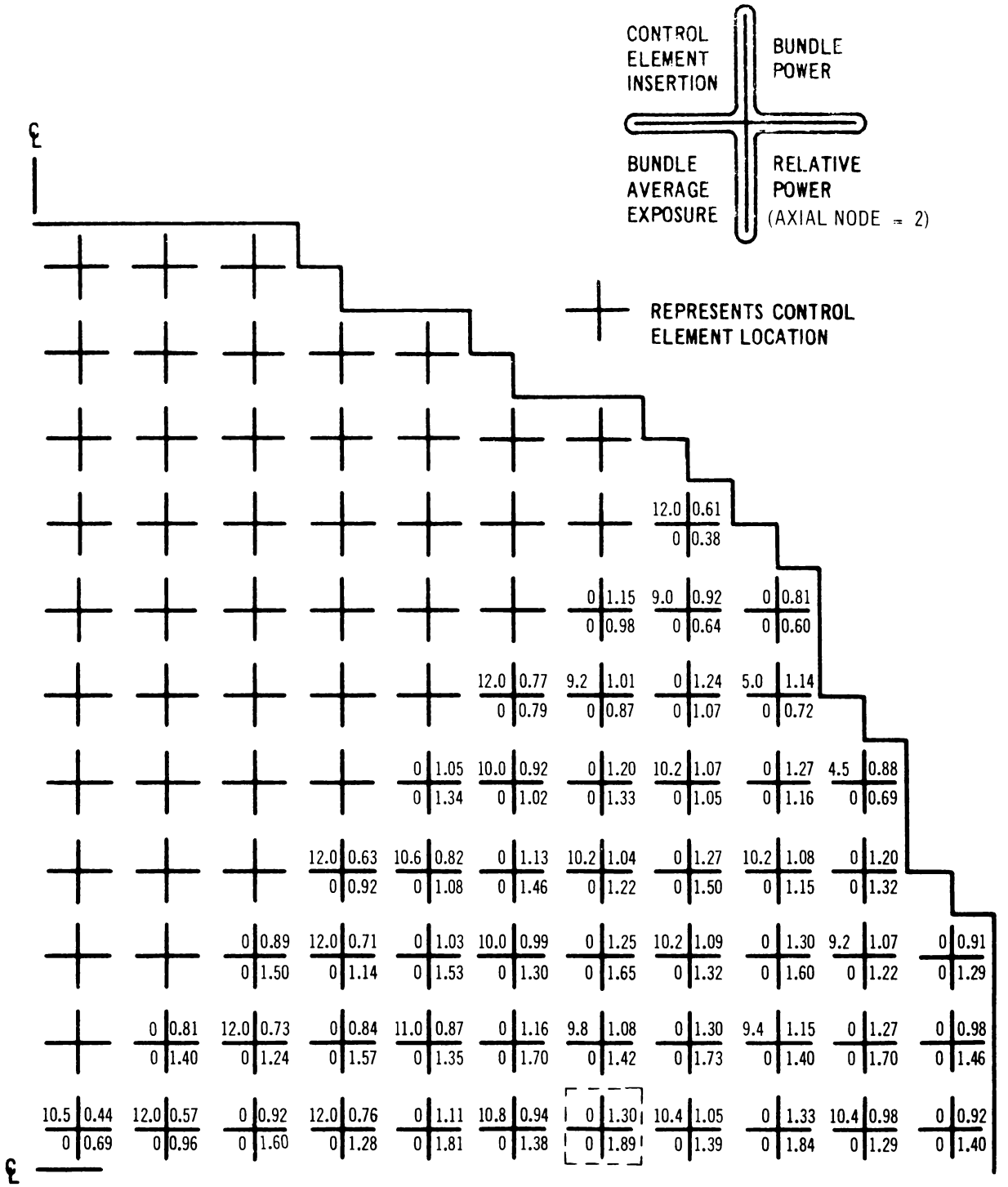


Figure I-8. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Void-Dependent Doppler and Xenon Effects Included, Eigenvalue = 1.004

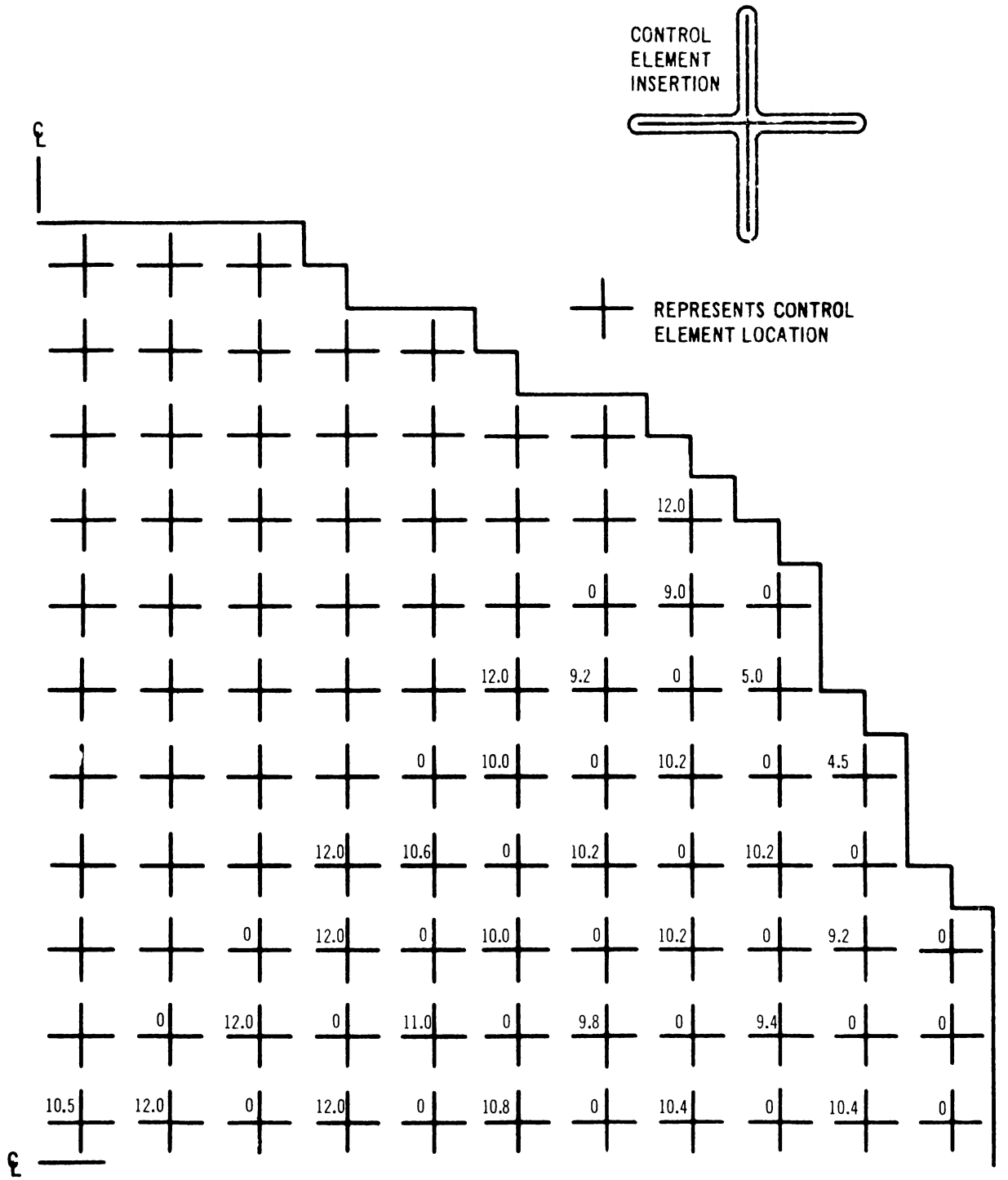


Figure I-9. Core of Arrangement A. Control Element Insertion. for All Xenon and Doppler Calculations

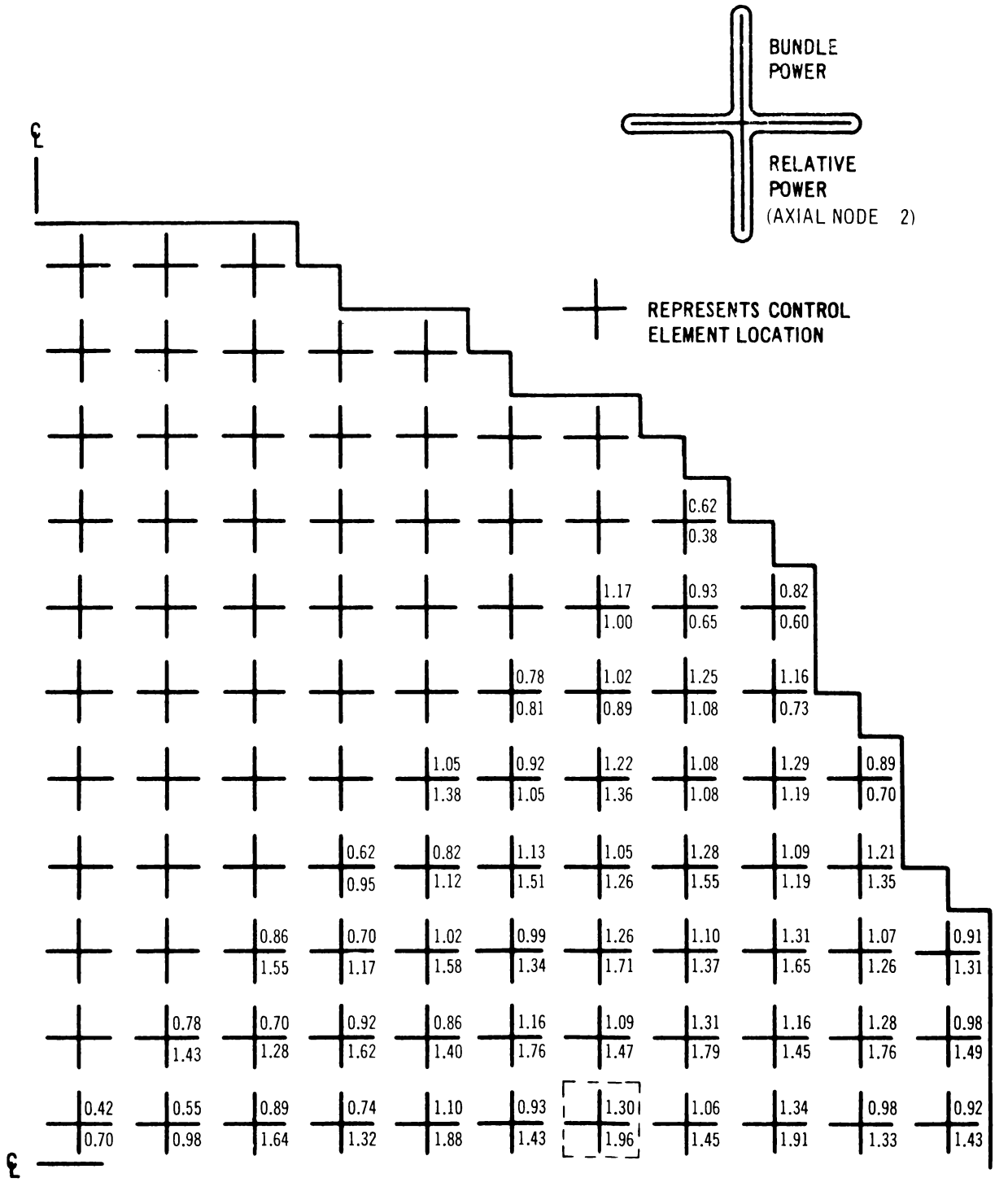


Figure I-10. Core of Arrangement A, Bundle Power, Relative Power at Plane Containing Peak Power, Core Average Exposure = 0, Eigenvalue = 1.029, Void-Dependent Doppler Effect Included, Xenon Effect Included

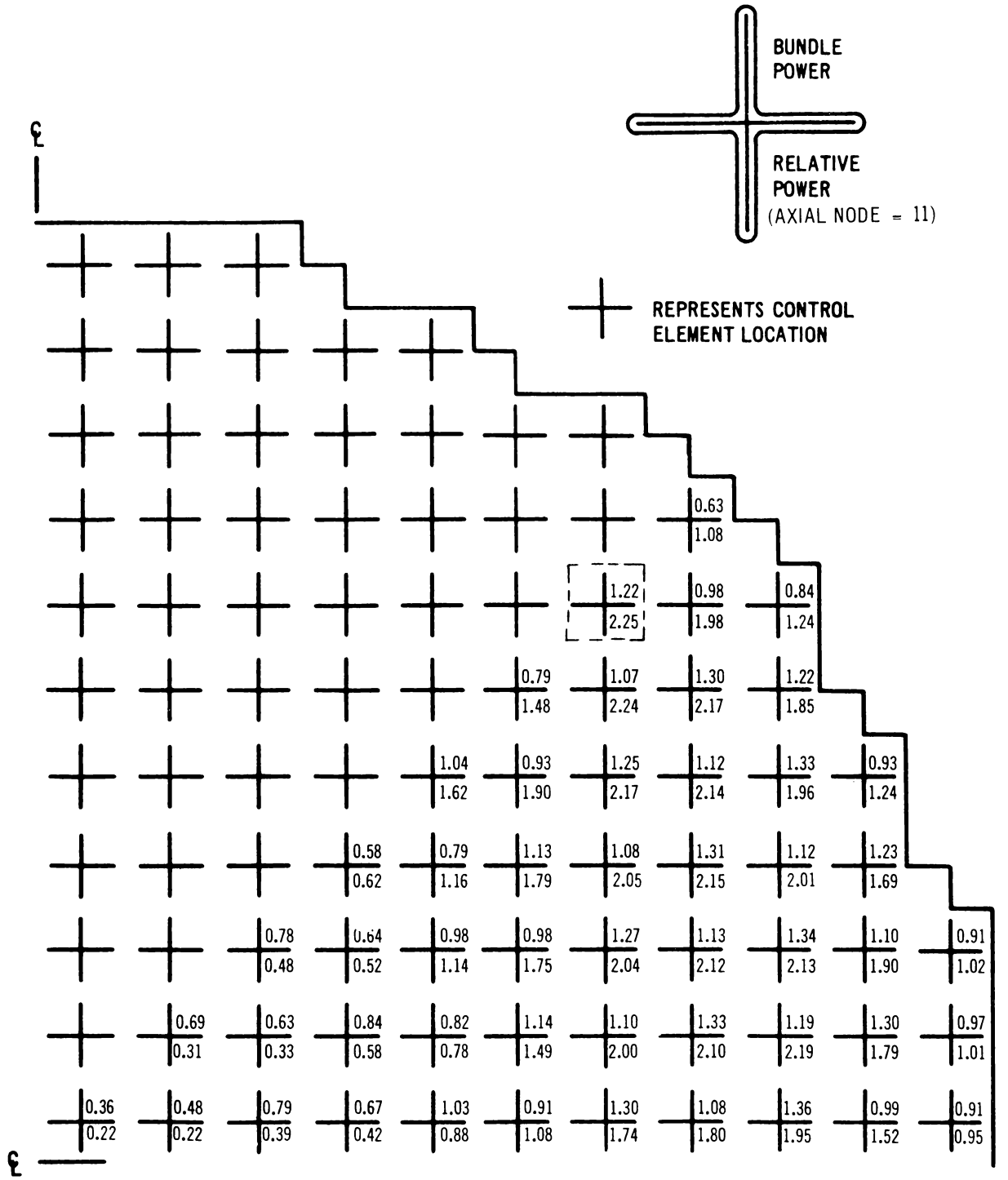


Figure I-11. Core of Arrangement A, Bundle Power and Relative Power at Plane Containing Peak Power. Core Average Exposure = 0, Eigenvalue = 1.052, Void-Dependent Doppler Effects Excluded, Xenon Effects Excluded

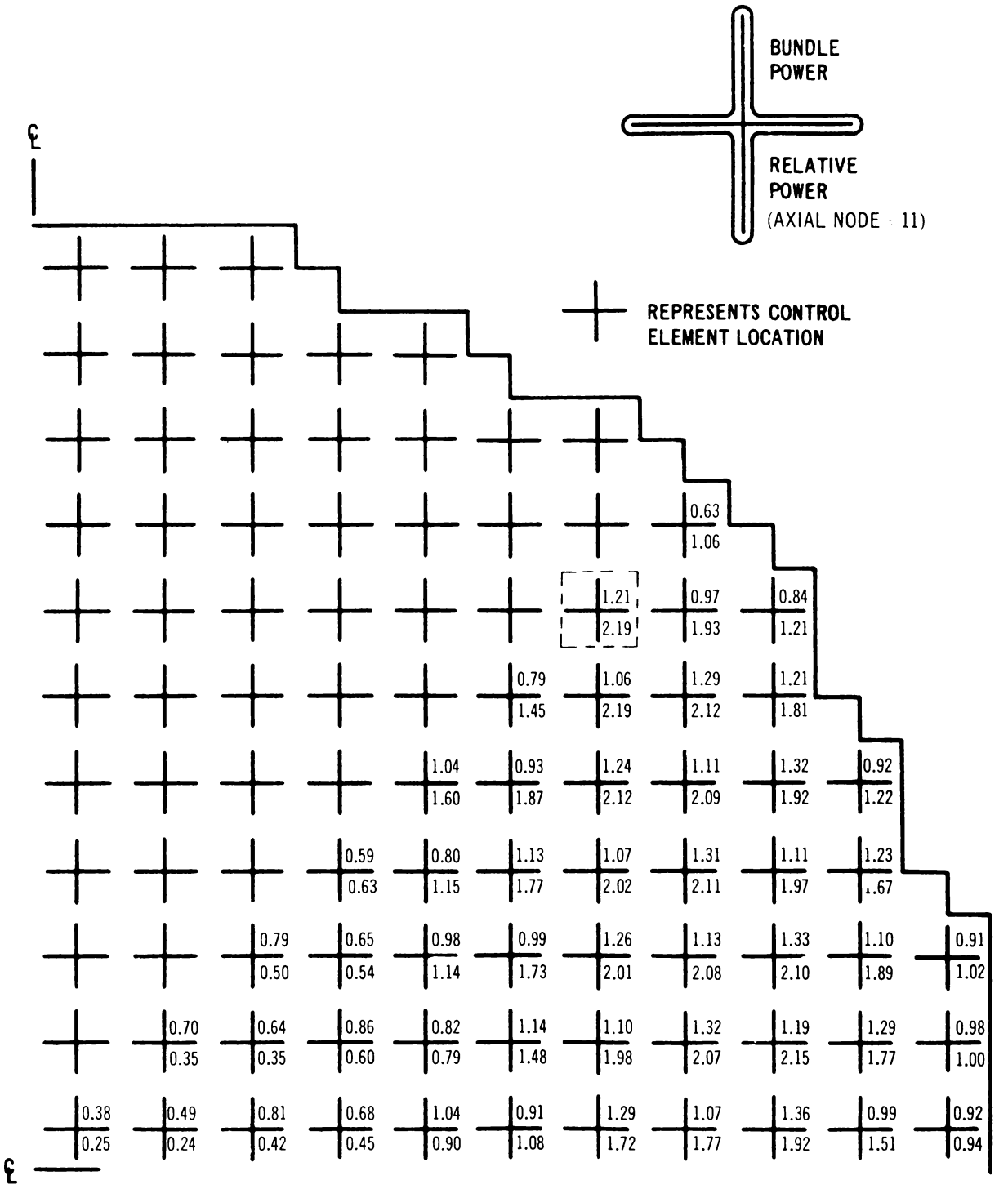


Figure I-12. Core of Arrangement A, Bundle Power, Relative Power at Plane Containing Peak Power, Core Average Exposure = 0, Eigenvalue 1.027, Void-Dependent Doppler Excluded, Xenon Effects Included

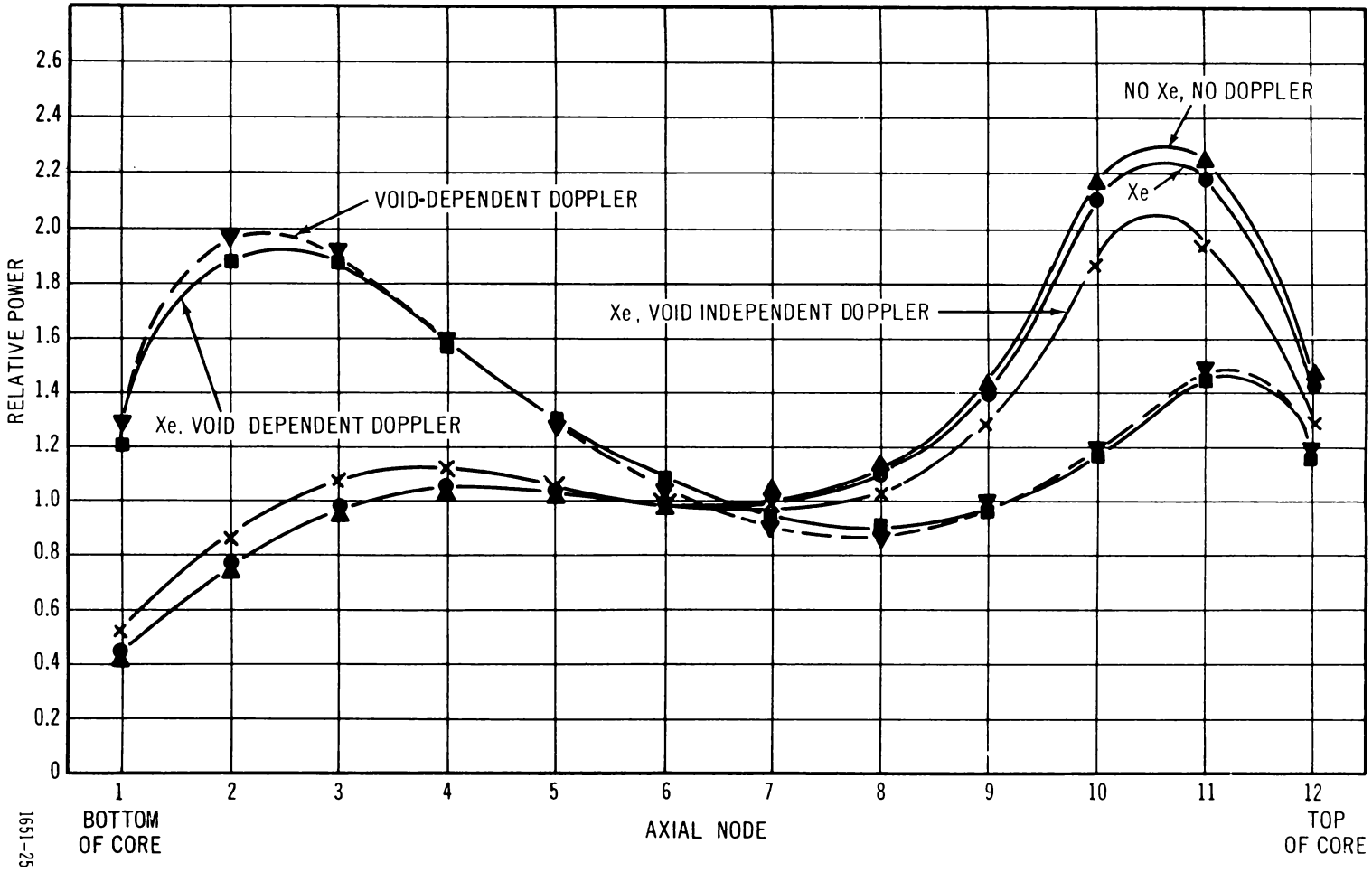


Figure I-13. The Effects of Xenon and Doppler on the Axial Power Distribution, Taken Through the Bundle Containing the Peak Power

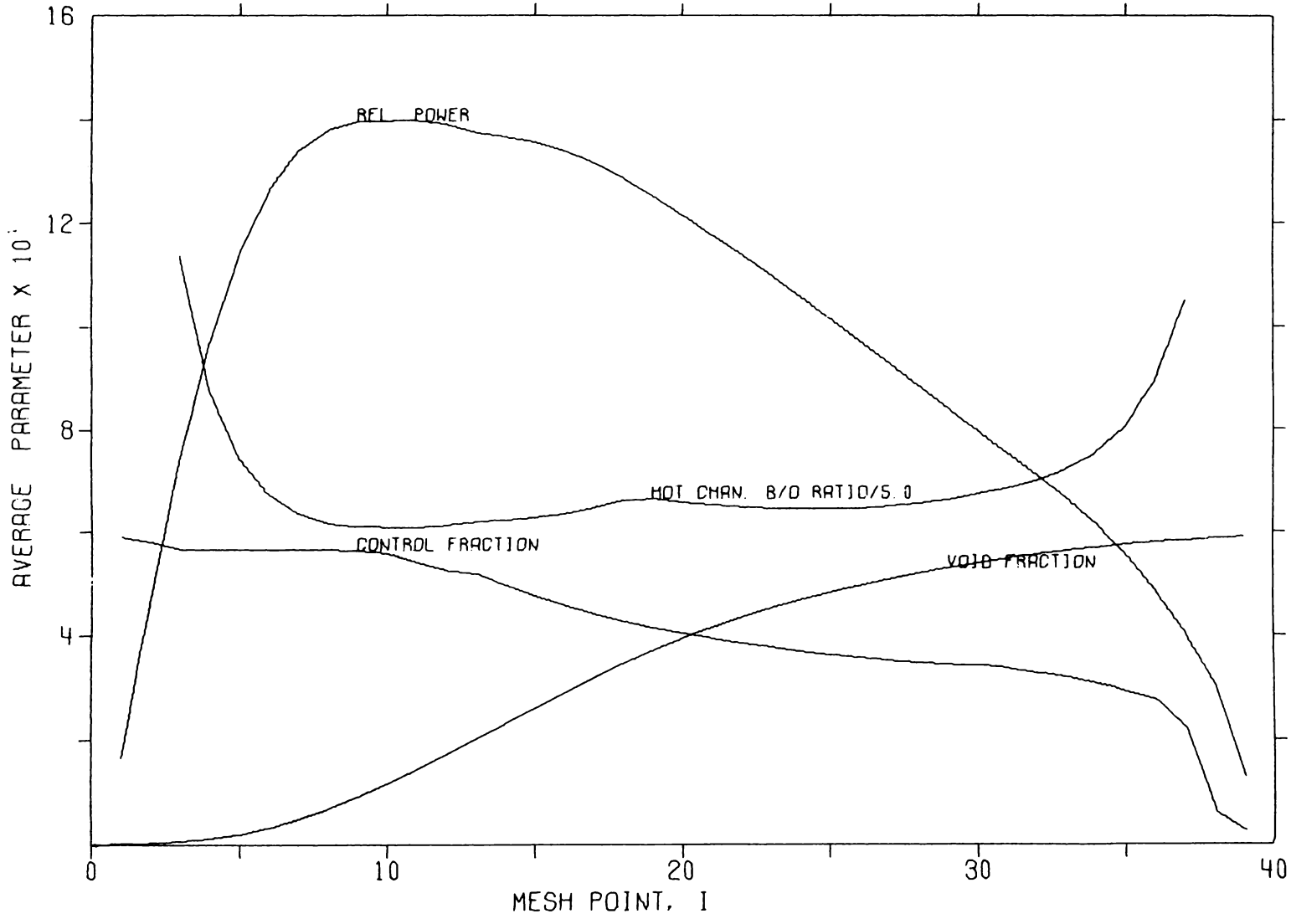


Figure I-14. Axial BWR Plot, Case Burni, Exposure = 0.0 MWD/STU

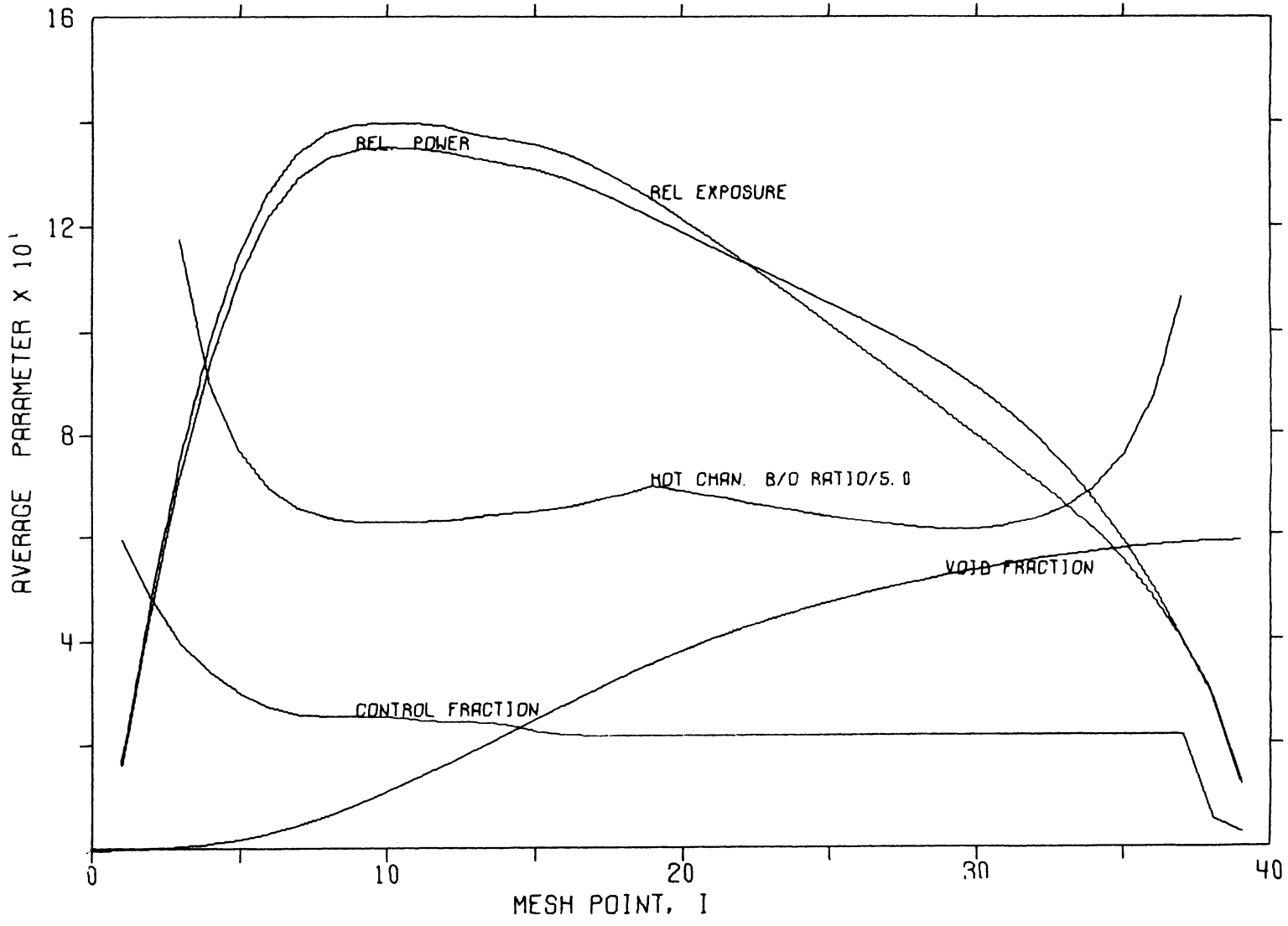


Figure I-15. Axial BWR Plot, Case Burni, Exposure = 6000 MWD/STU

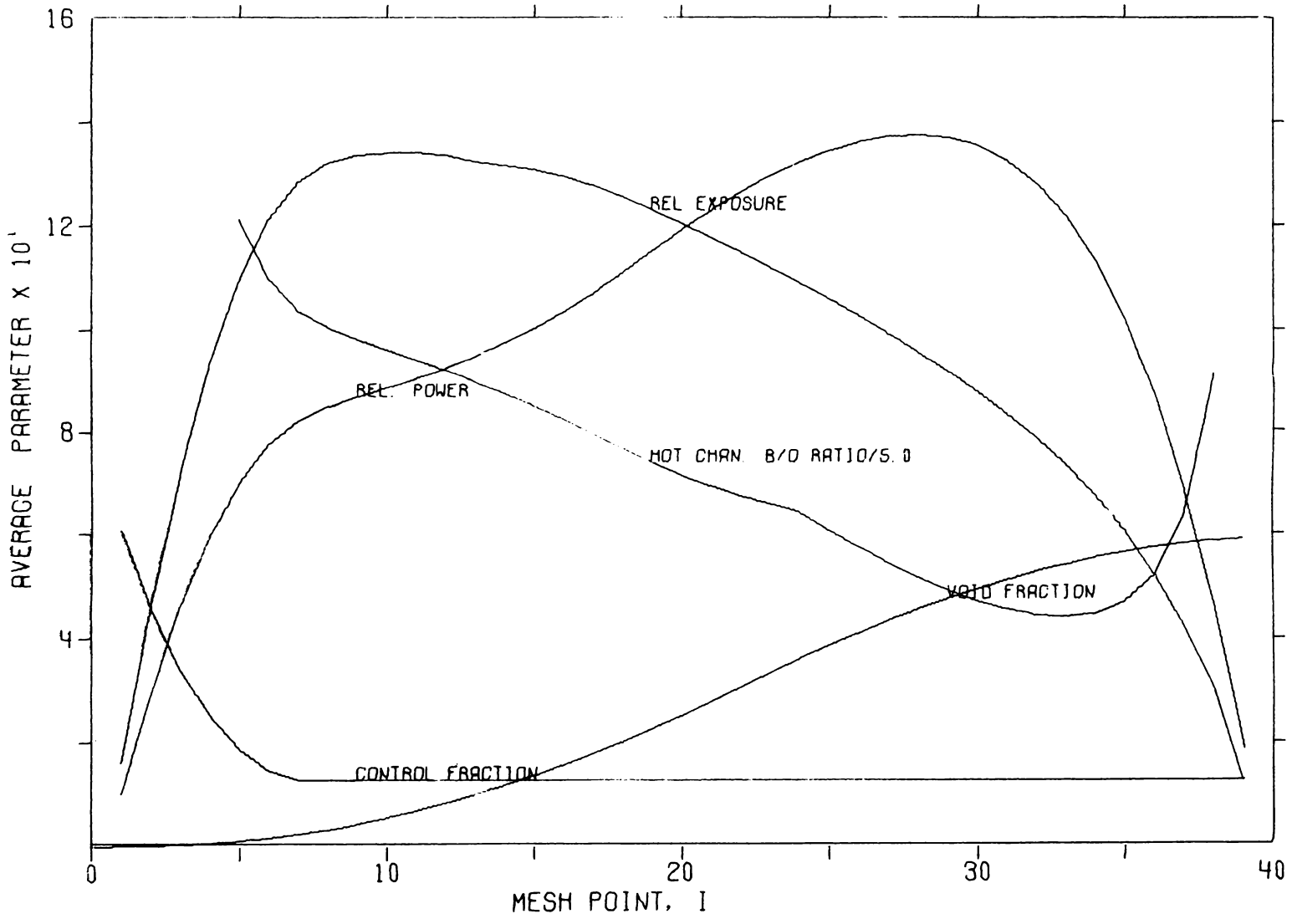


**Figure I-16.** Axial BWR Plot, Case Burni, Exposure = 9000 MWD/STU

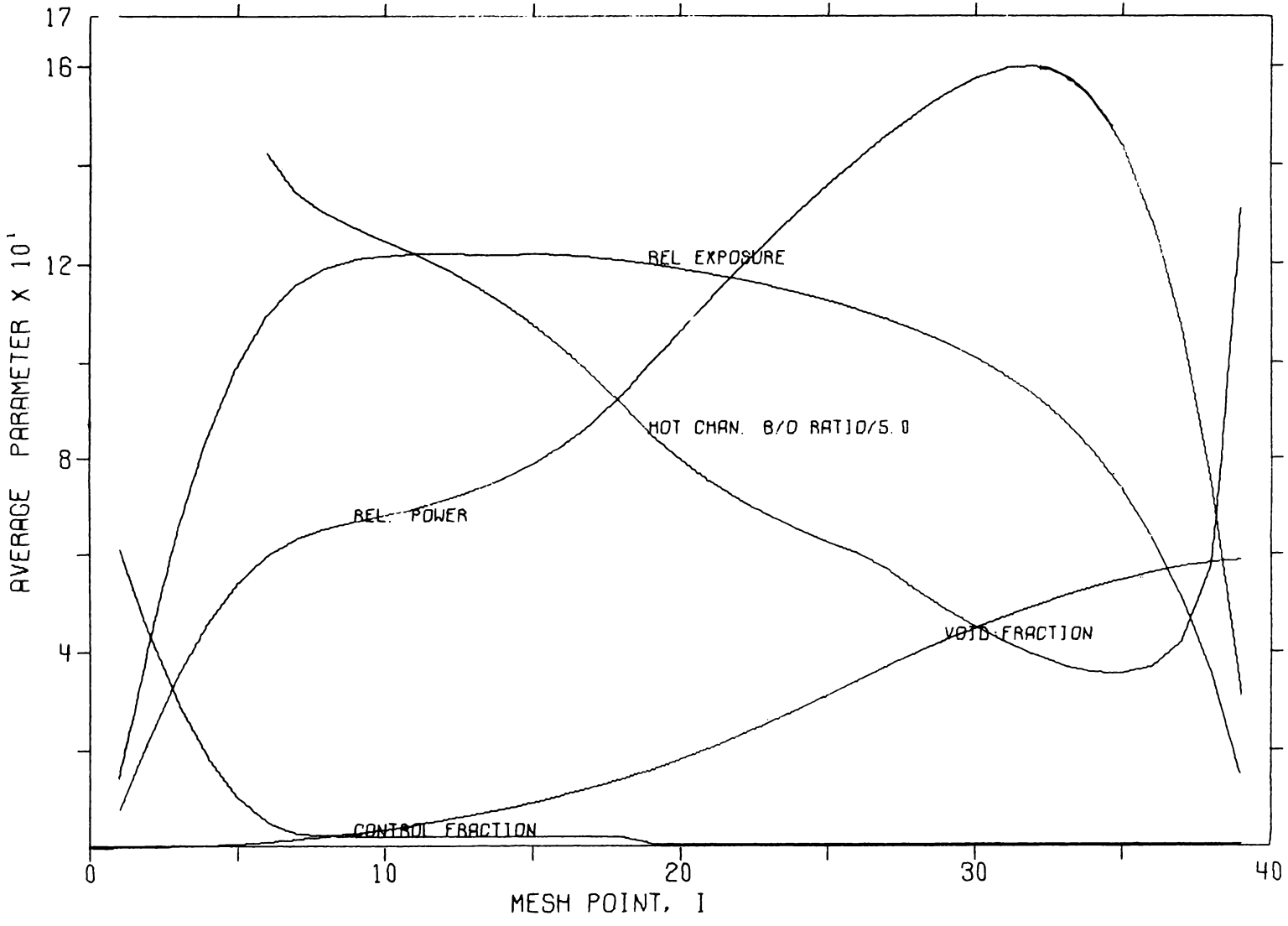


Figure I-17. Axial BWR Plot, Case Burni, Exposure = 12,000 MWD/STU

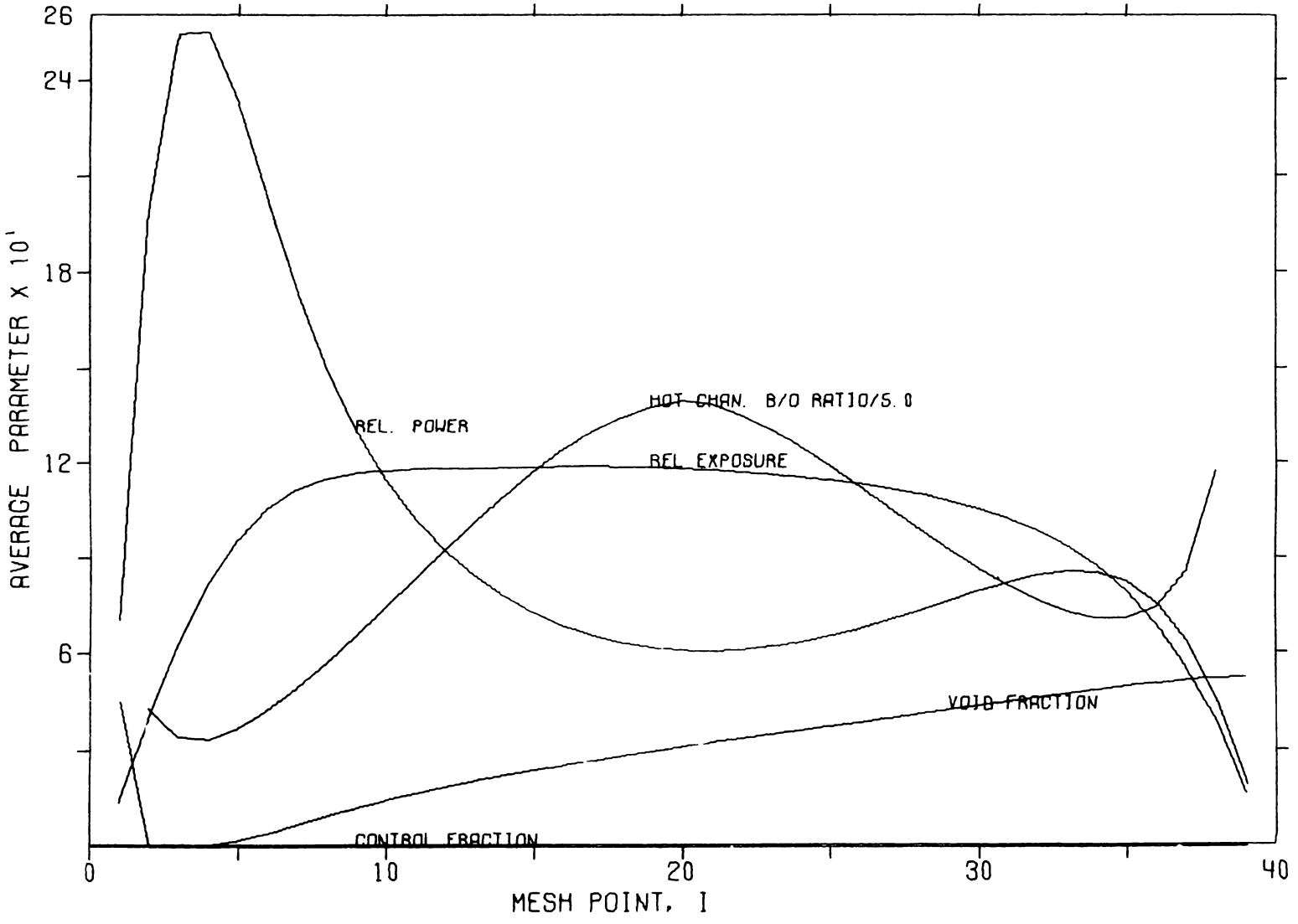


Figure I-18. Axial BWR Plot, Case Burni, Exposure = 13,000 MWD/STU

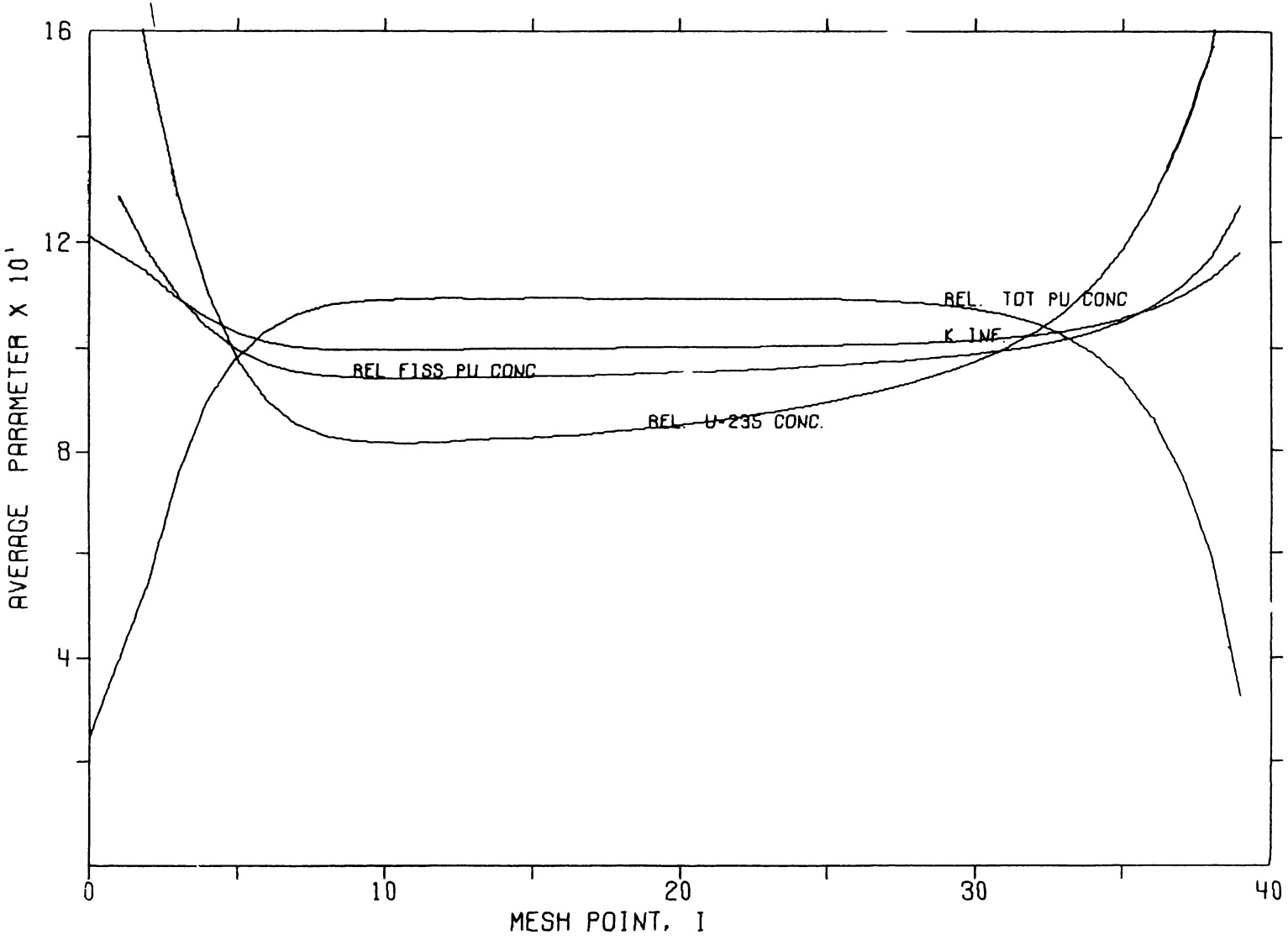


Figure I-19. Axial BWR Plot, Case Burni, Exposure = 13,000 MWD/STU

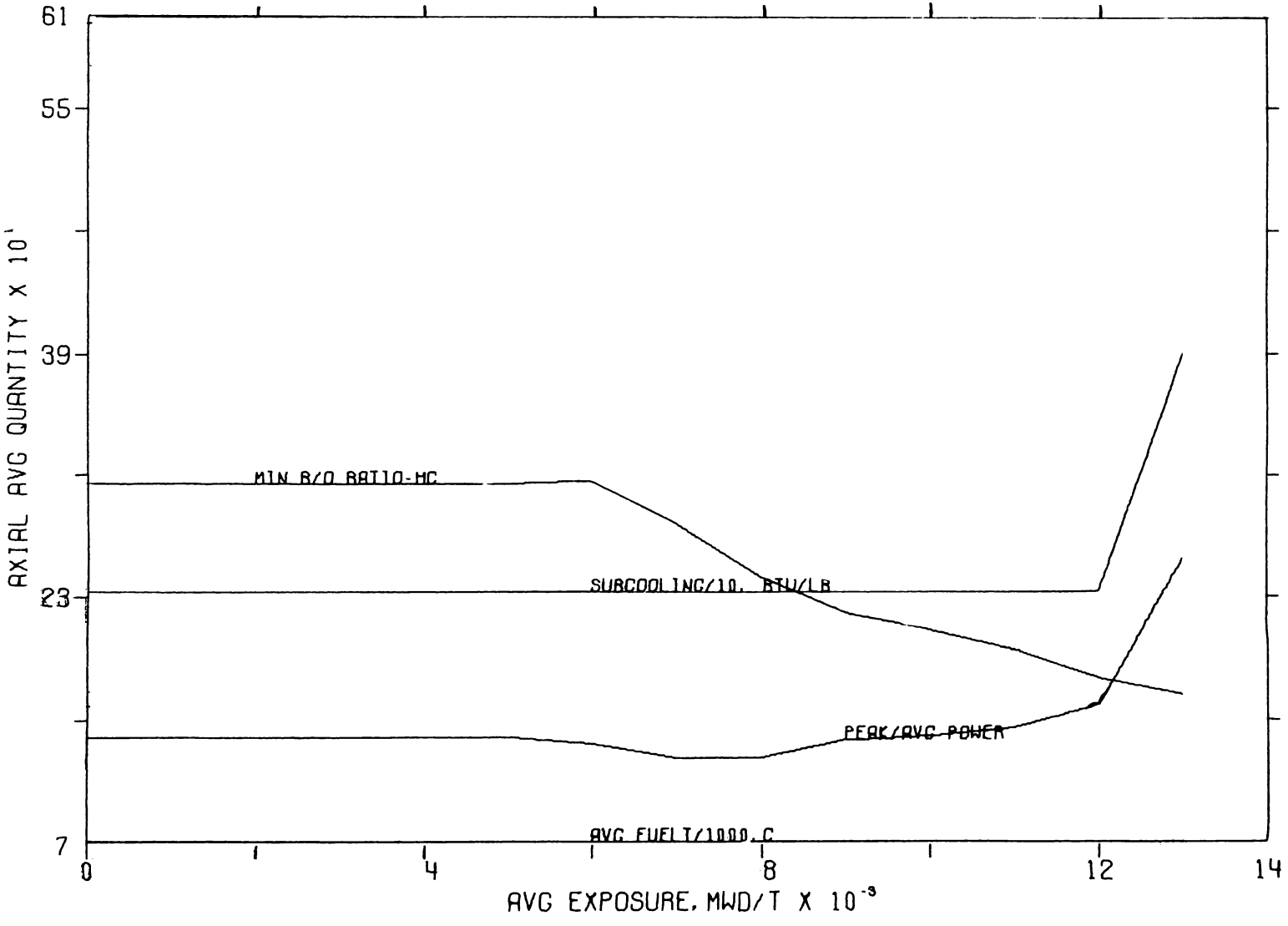


**Figure I-20.** Average BWR Properties, Case Burni

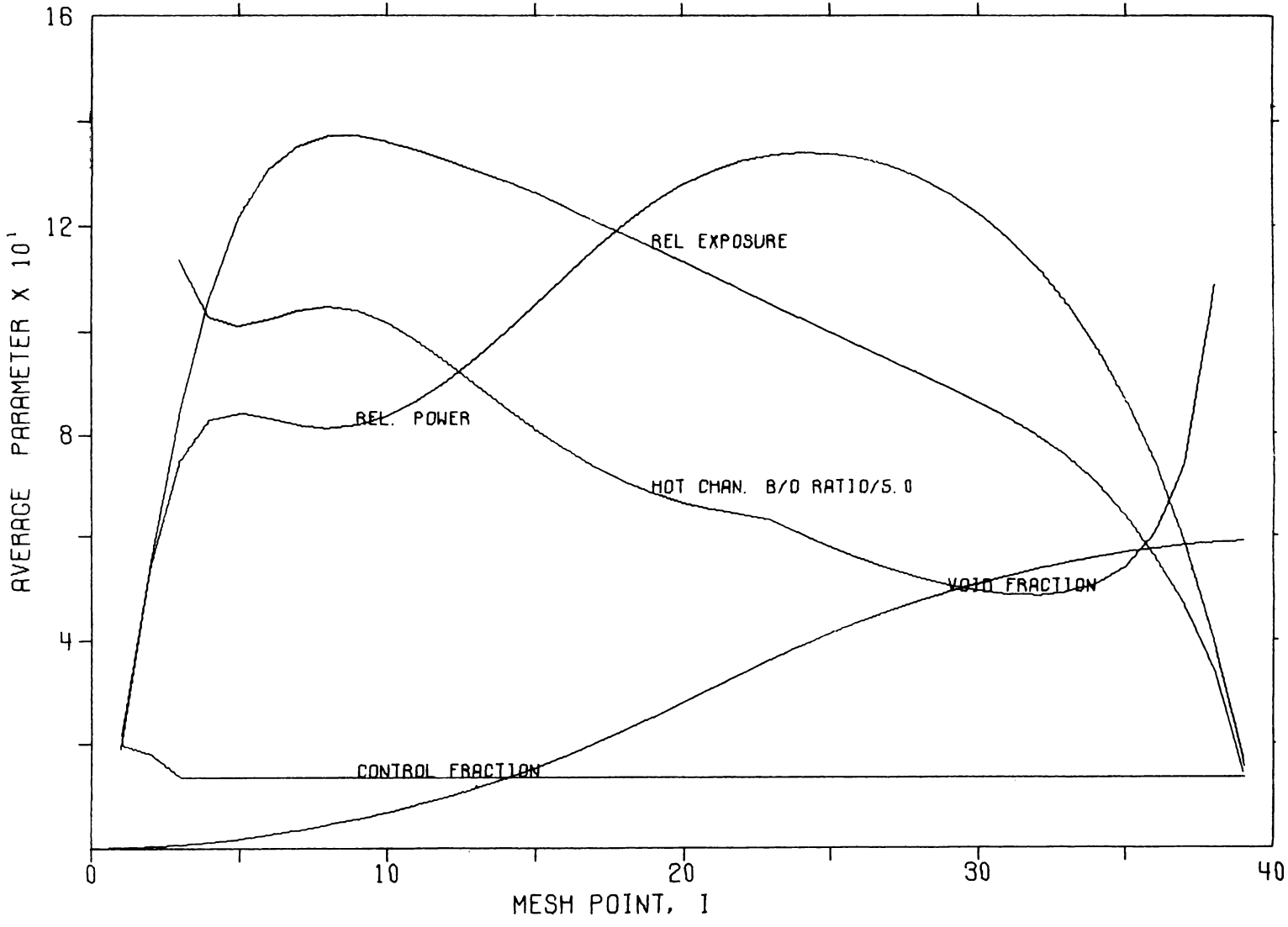


Figure I-21. Axial BWR Plot, Case BROPC, Exposure = 0.0 MWD/STU

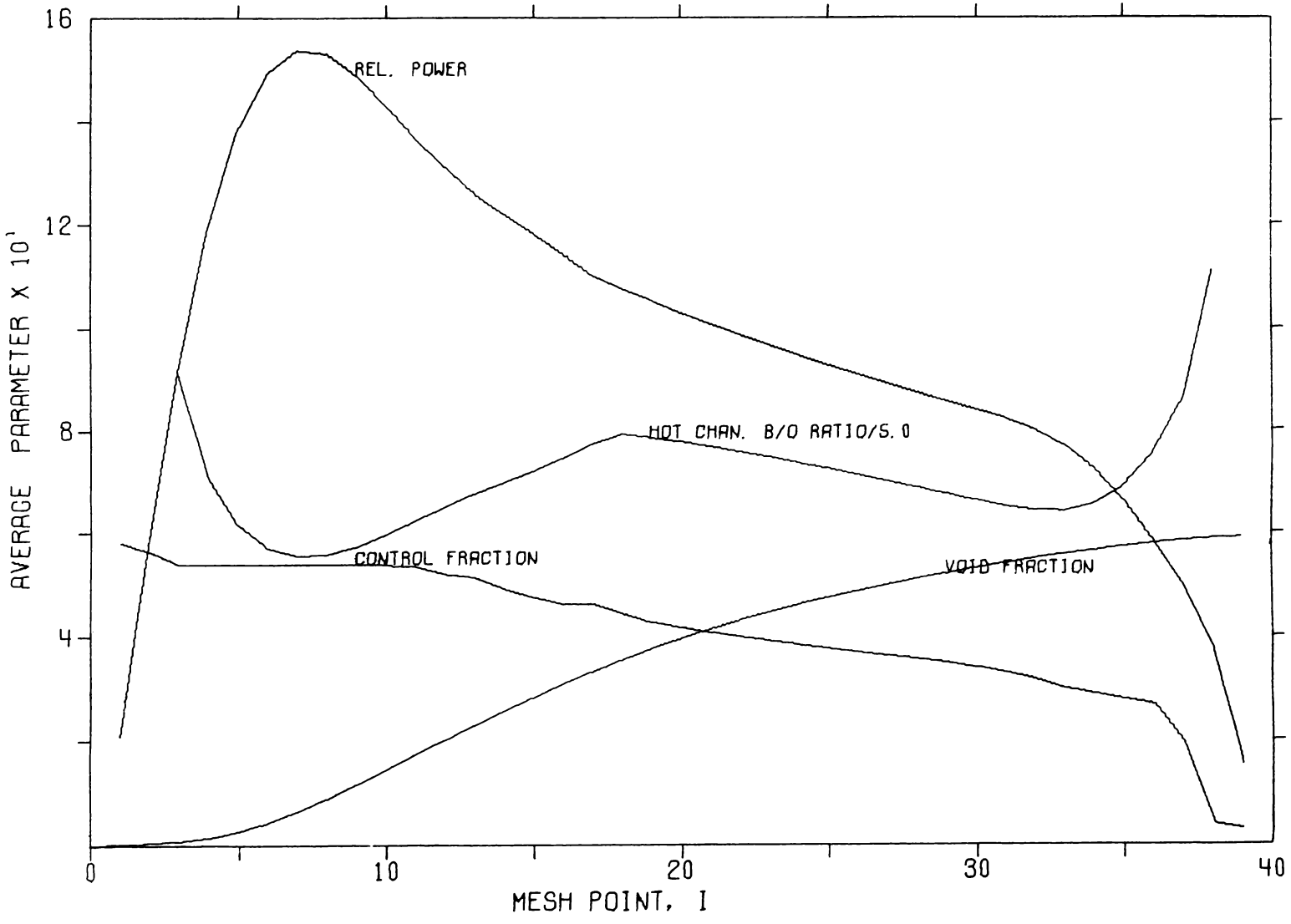


**Figure I-22.** Axial BWR Plot, Case BROPC, Exposure = 9000 MWD/STU



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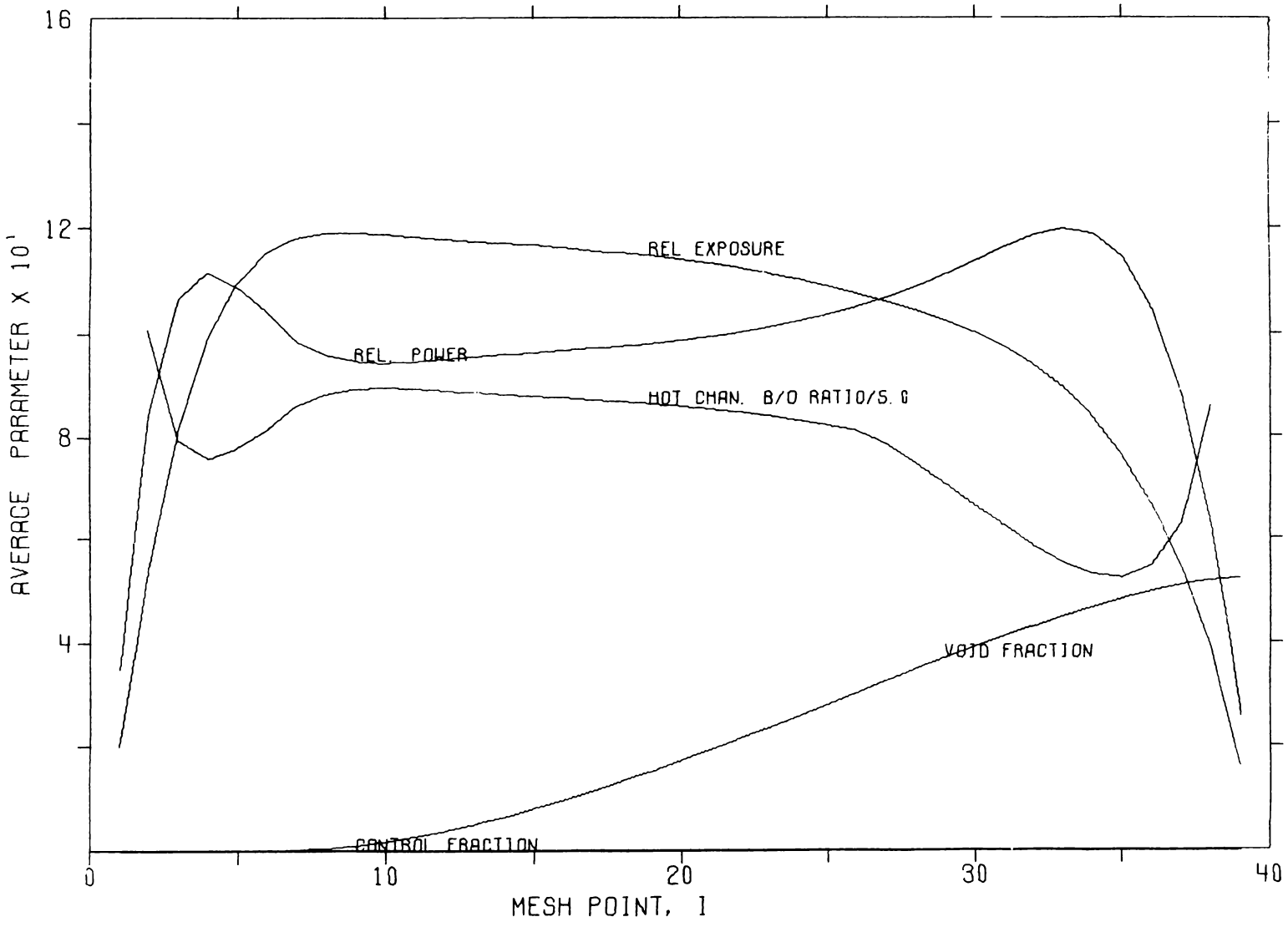



Figure I-23. Axial BWR Plot, Case BROPC, Exposure = 13,000 MWD/STU

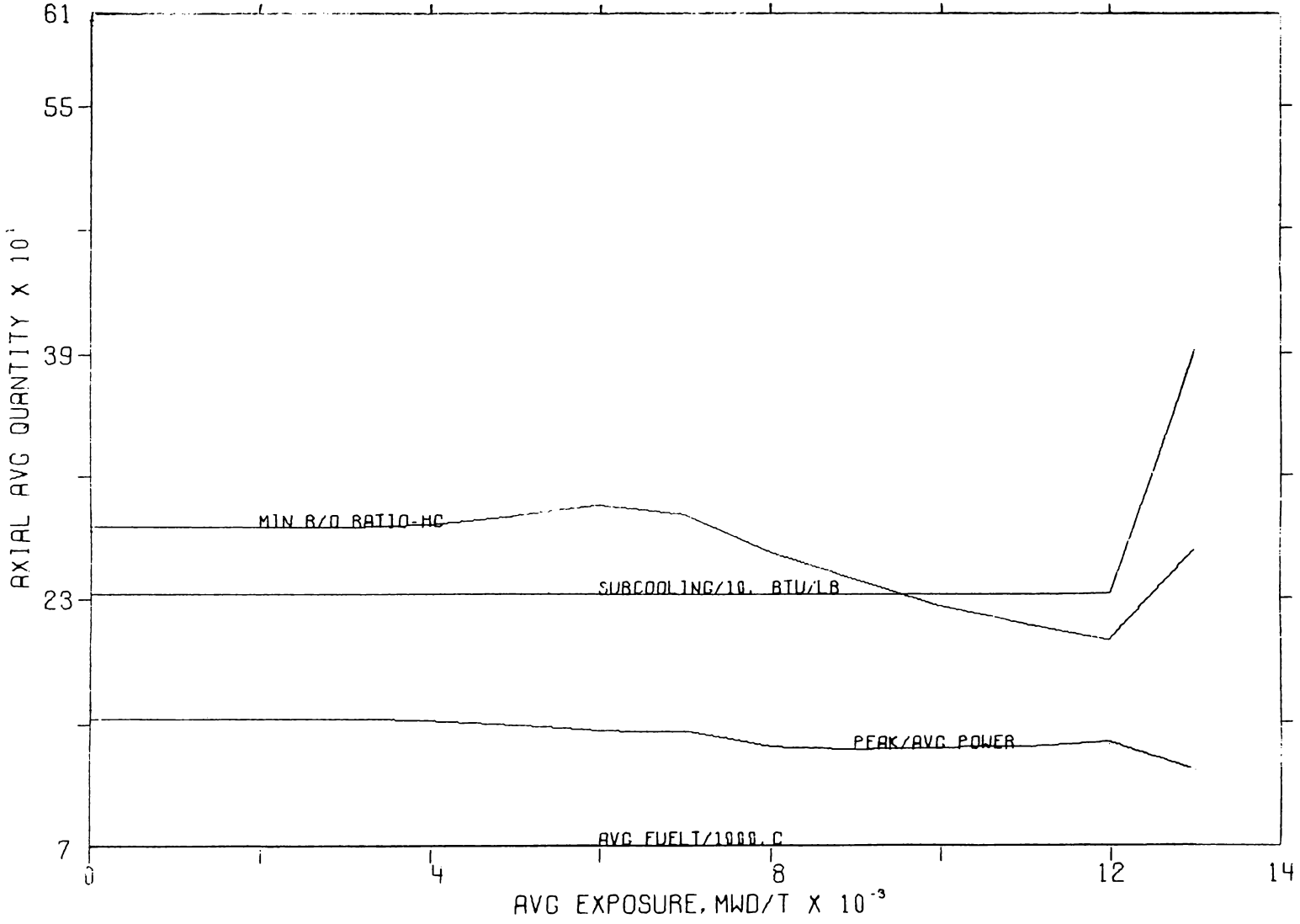



**Figure I-24.** Average BWR Properties, Case BROPC

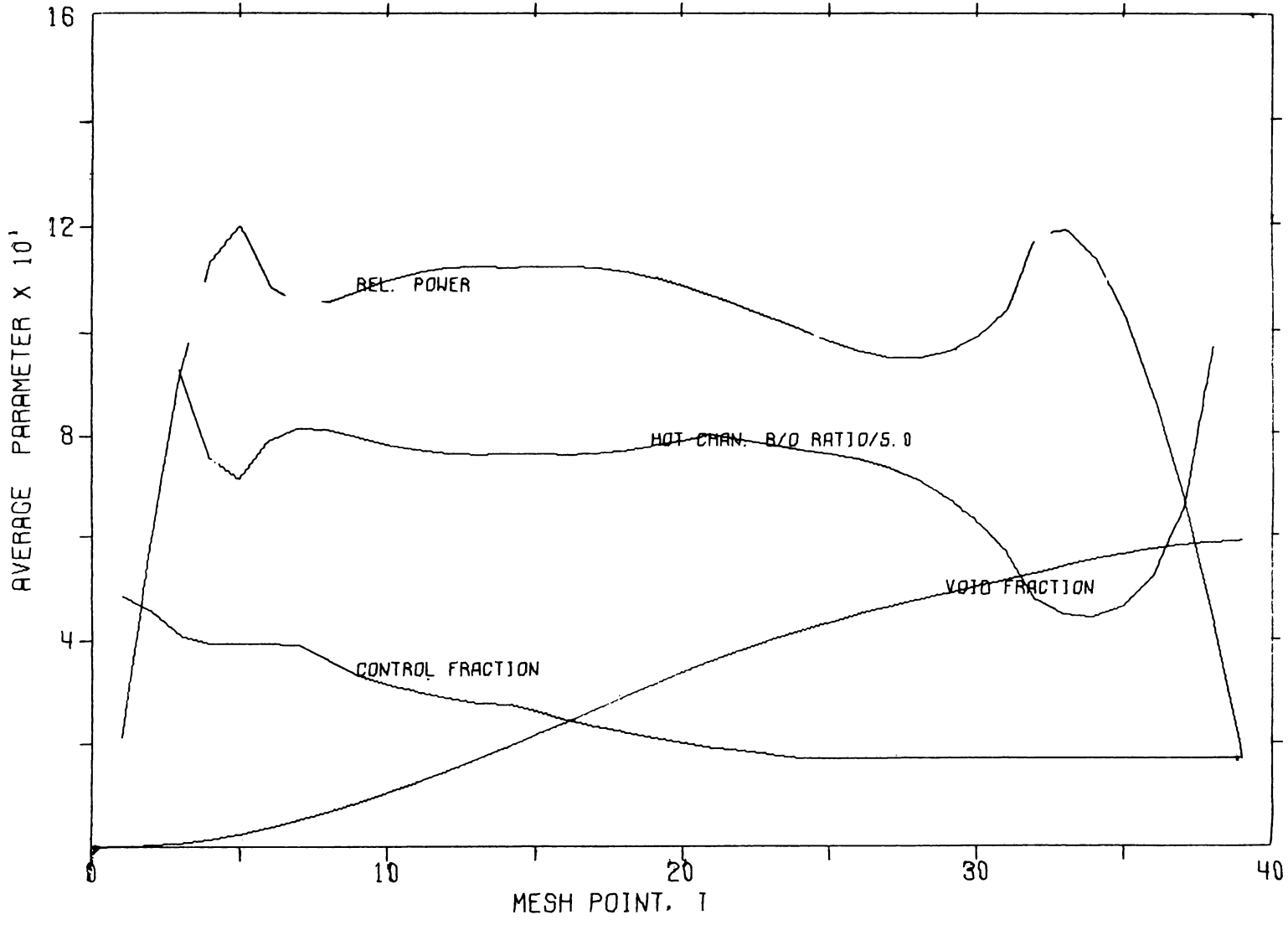



Figure I-25. Axial BWR Plot, Case SPRE9, Exposure = 0.0 MWD/STU

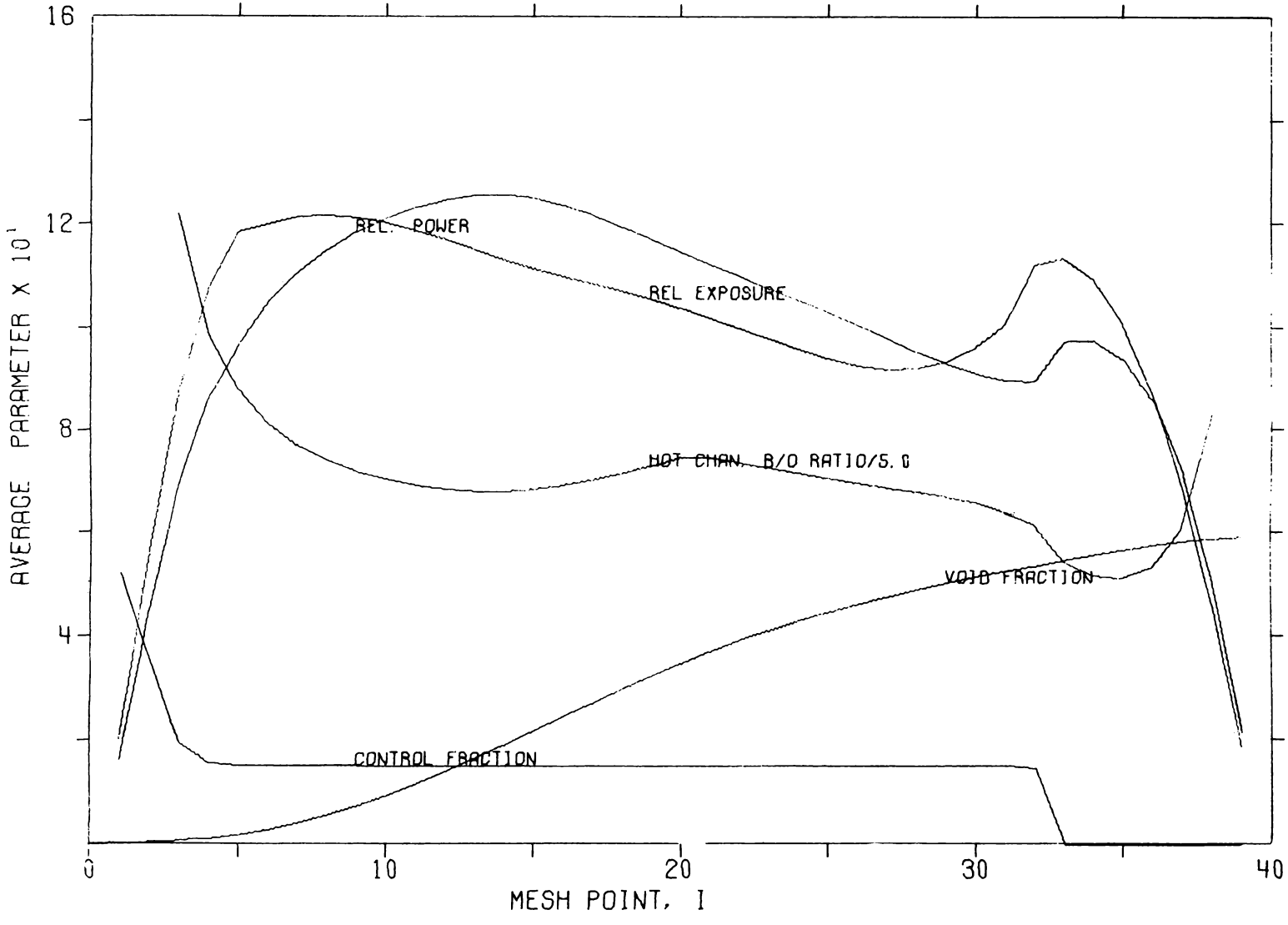



**Figure I-26.** Axial BWR Plot, Case SPRE9, Exposure = 9000 MWD/STU

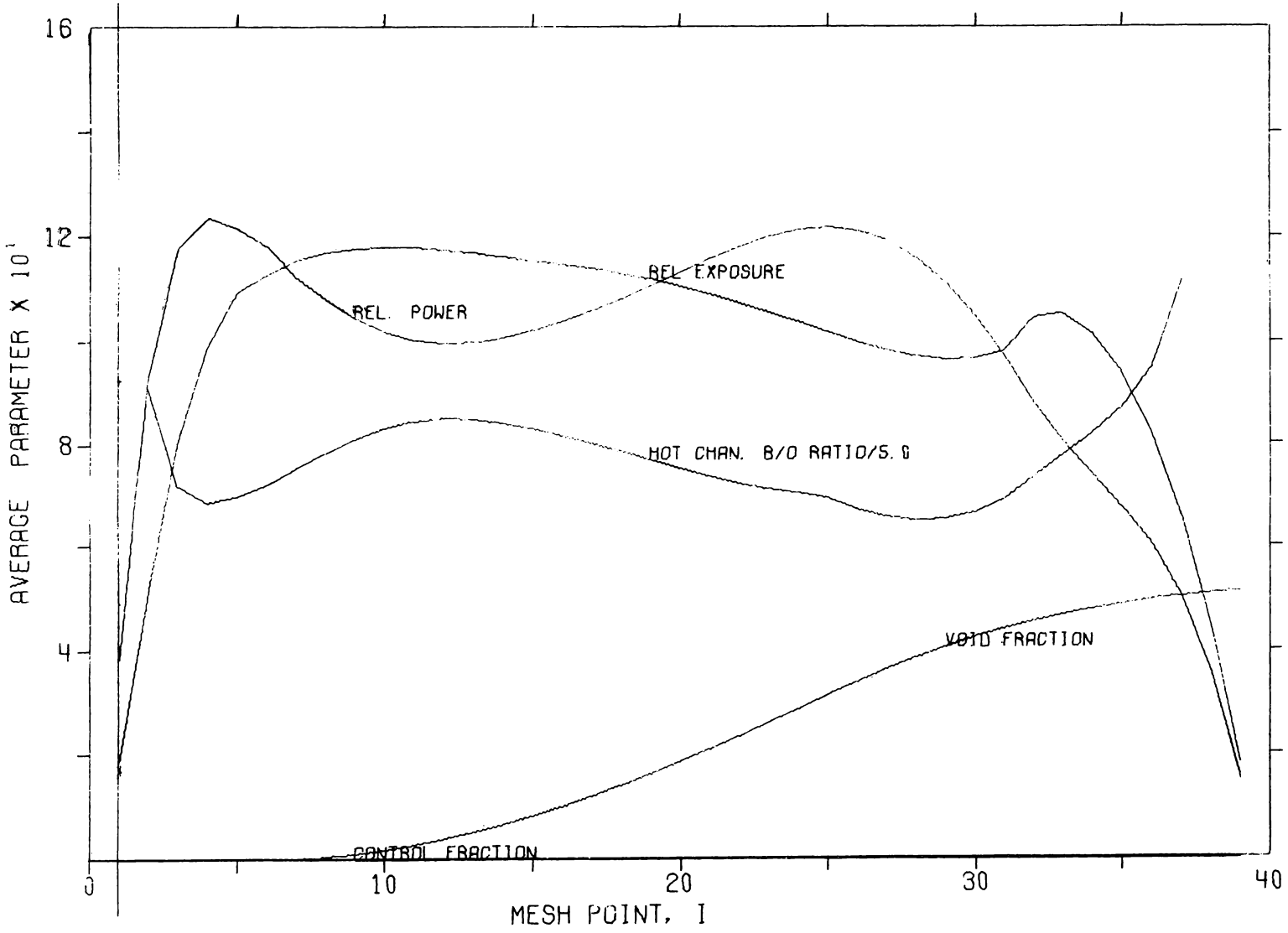



Figure I-27. Axial BWR Plot, Case SPRE9, Exposure = 13,000 MWD/STU

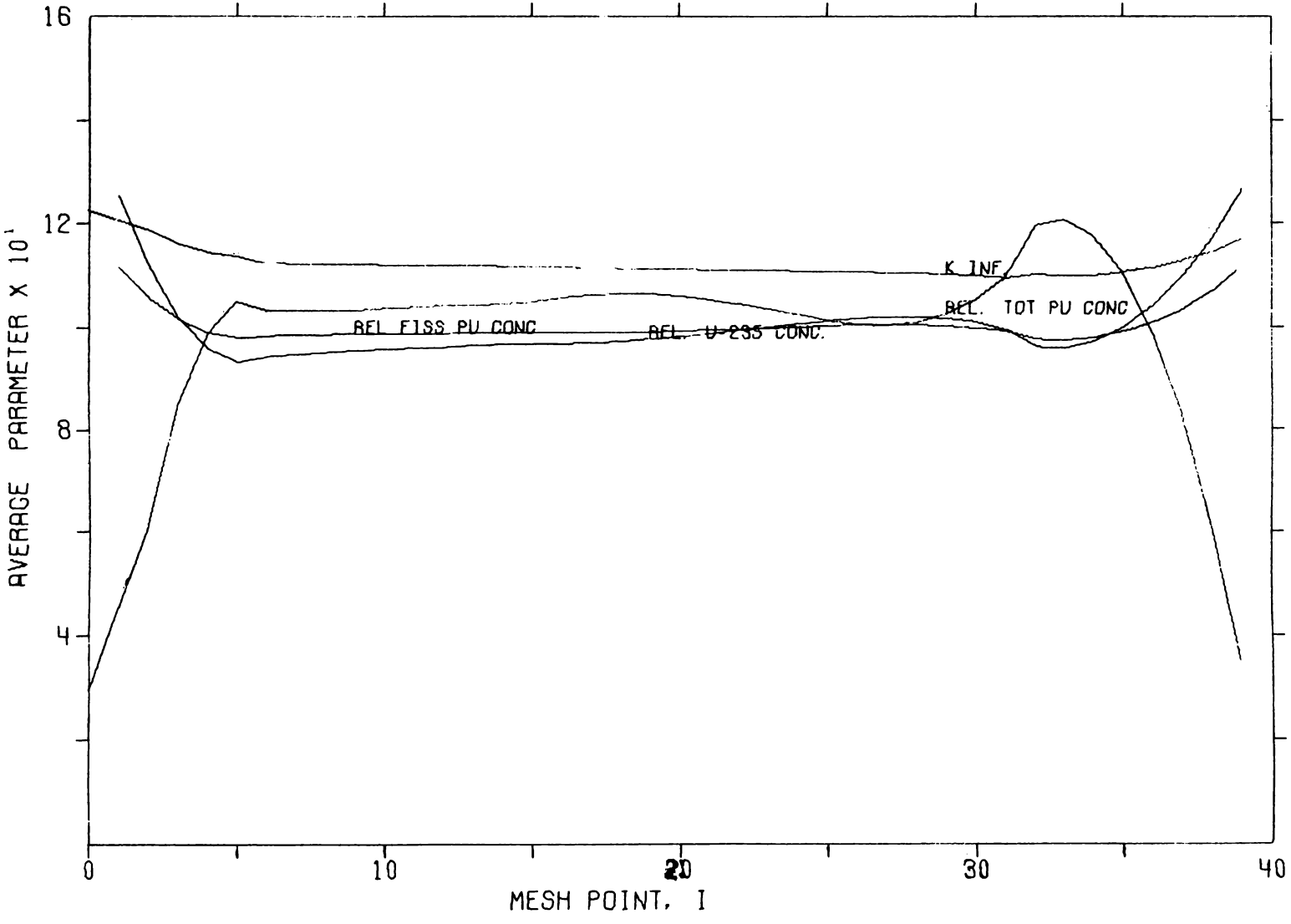



Figure I-28. Axial BWR Plot, Case SPRE9

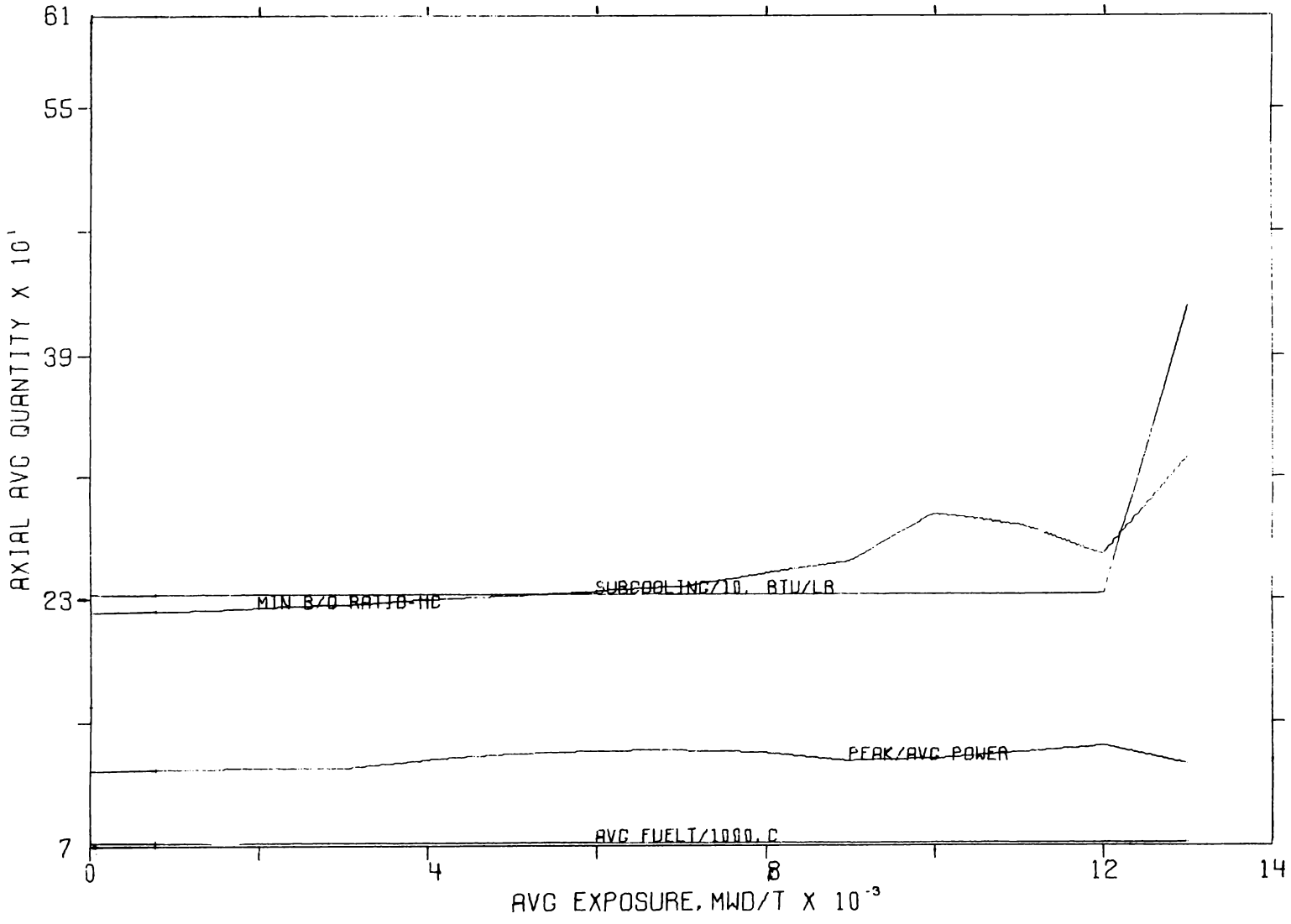



Figure I-29. Average BWR Properties, Case SPRE9

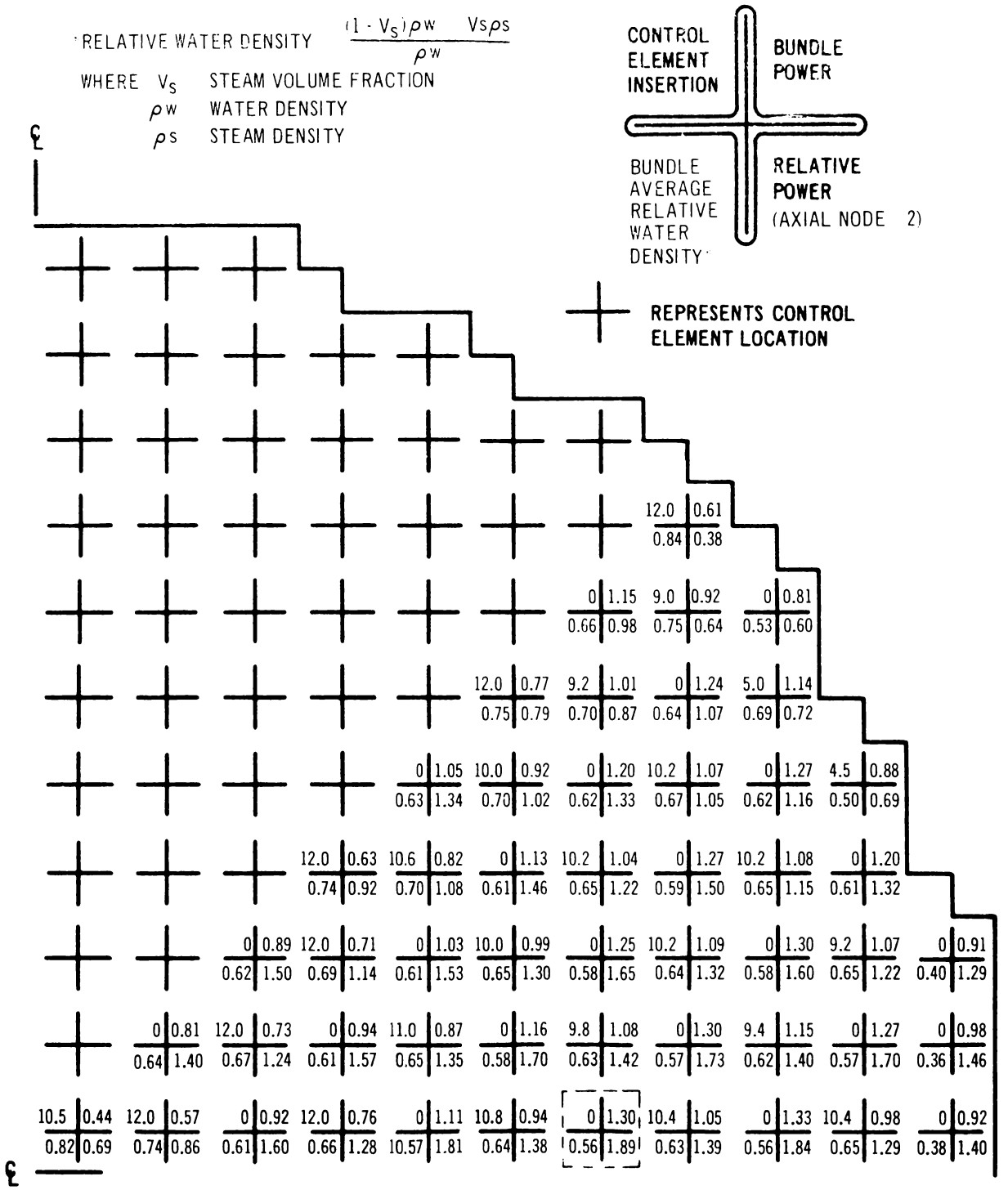



Figure I-30. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Relative Water Density, Rated Power, Rated Flow, Eigenvalue = 1.004

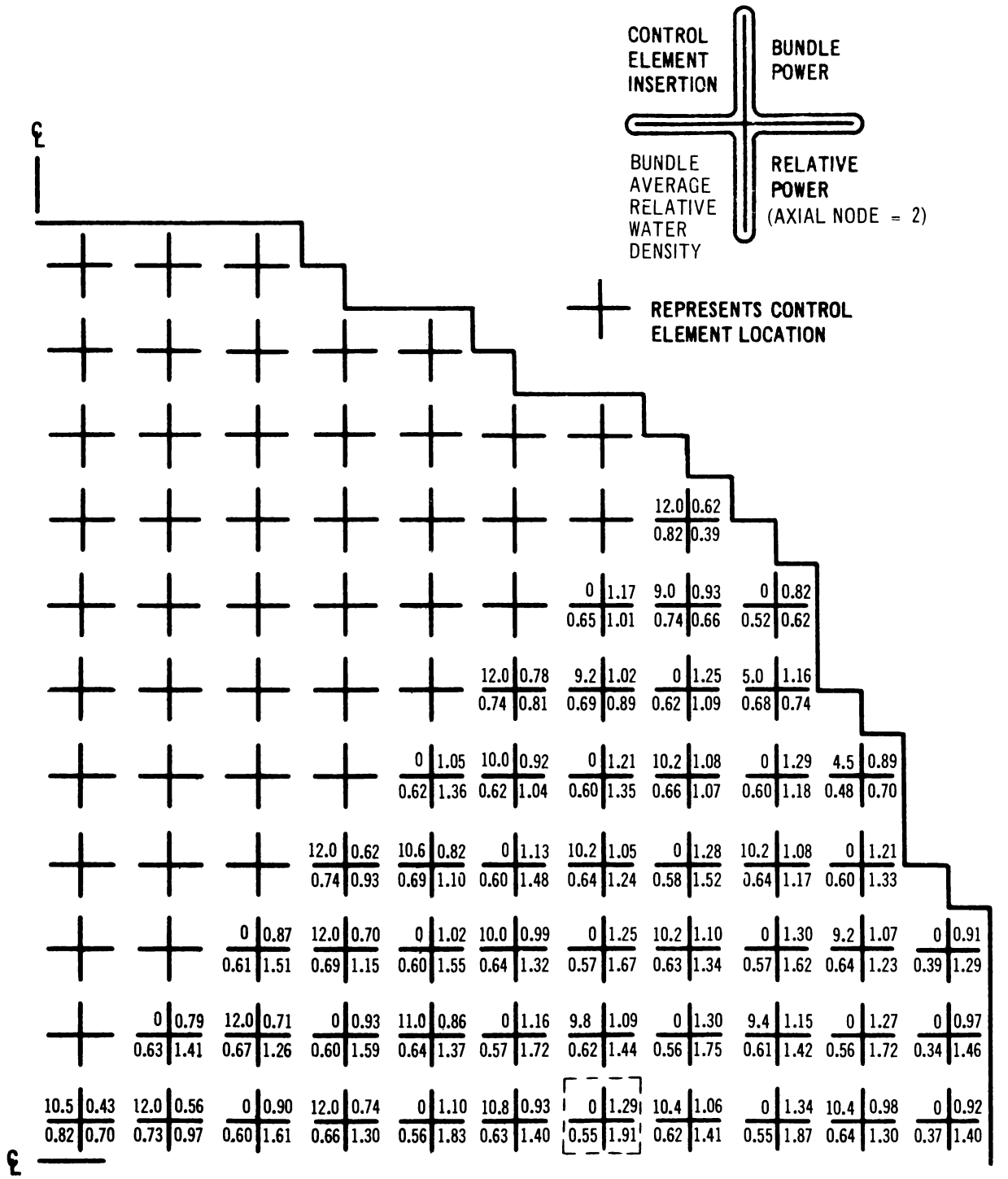
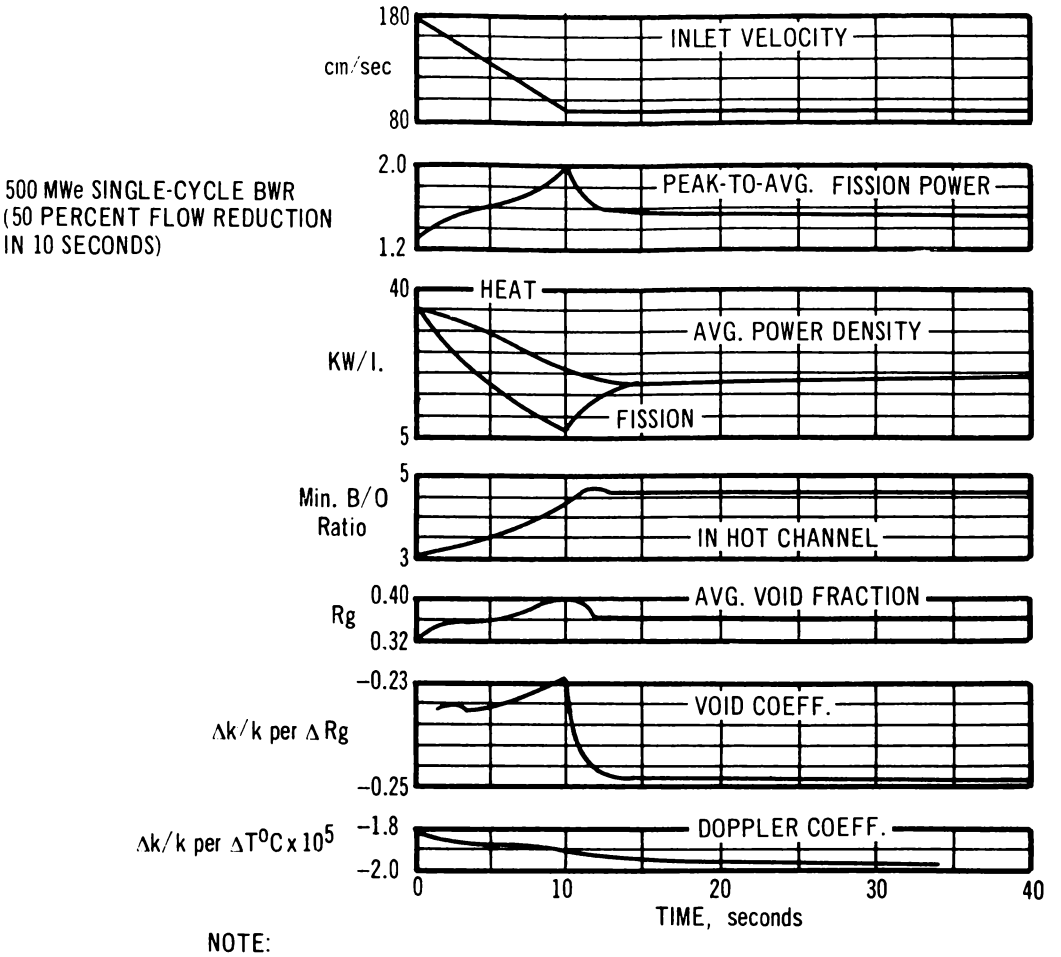



Figure I-31. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Relative Water Density, Power = 0.9 × Rated Power, Flow Adjusted to Yield Eigenvalue = 1.004




NOTE:

Rg = AVERAGE VOID VOLUME IN CORE AS PERCENT OF TOTAL MODERATOR SPACE VOLUME IN CORE.

Figure I-32. Flow Ramp

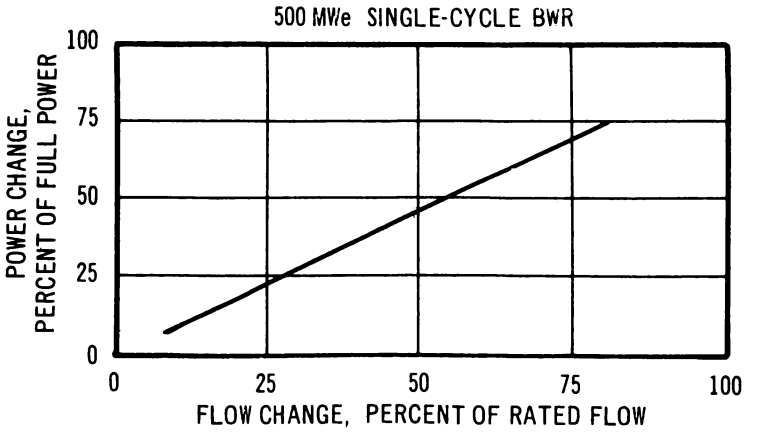



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Figure I-33. Power Flow Map

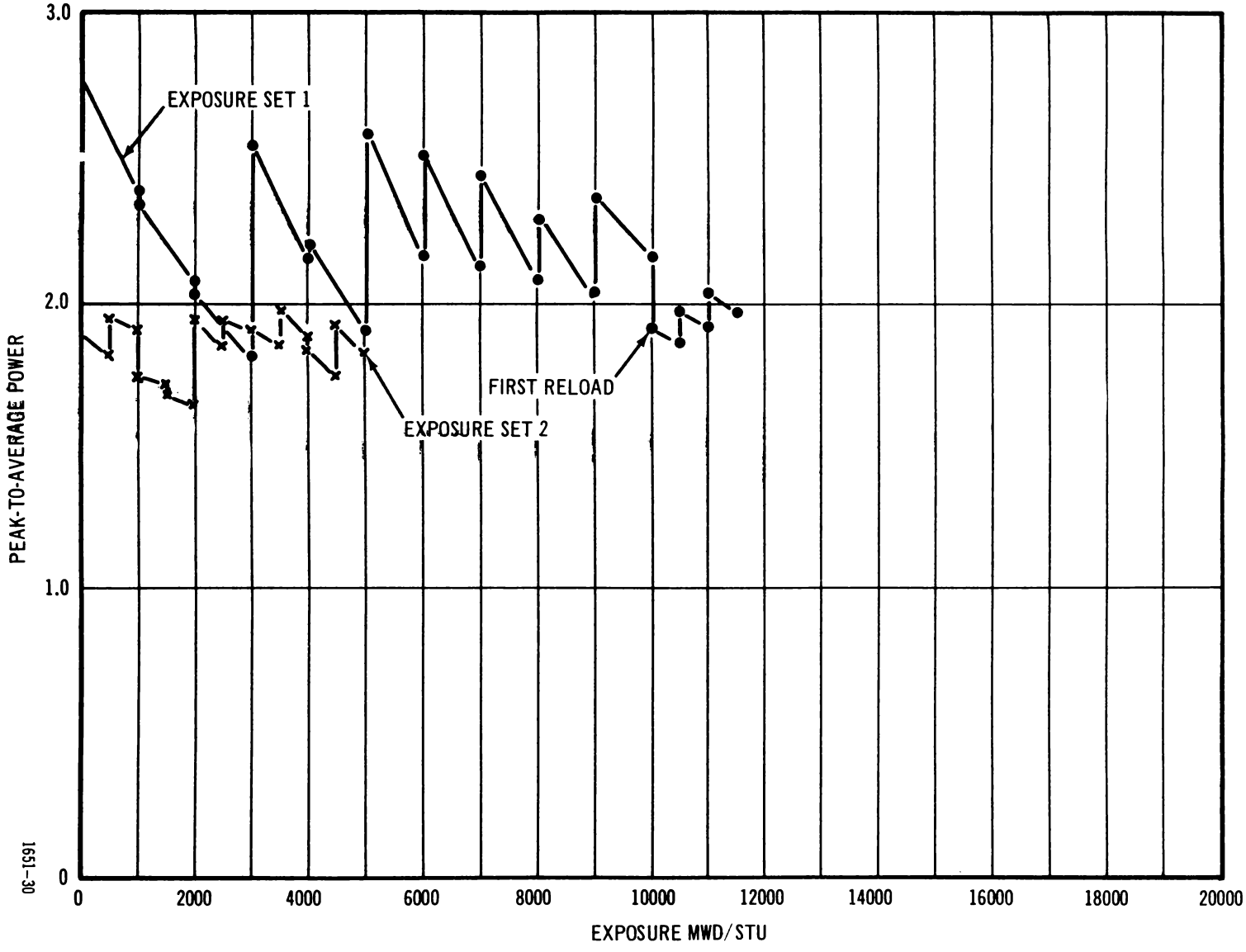



Figure I-34. Variation of Peak-to-Average Power with Exposure

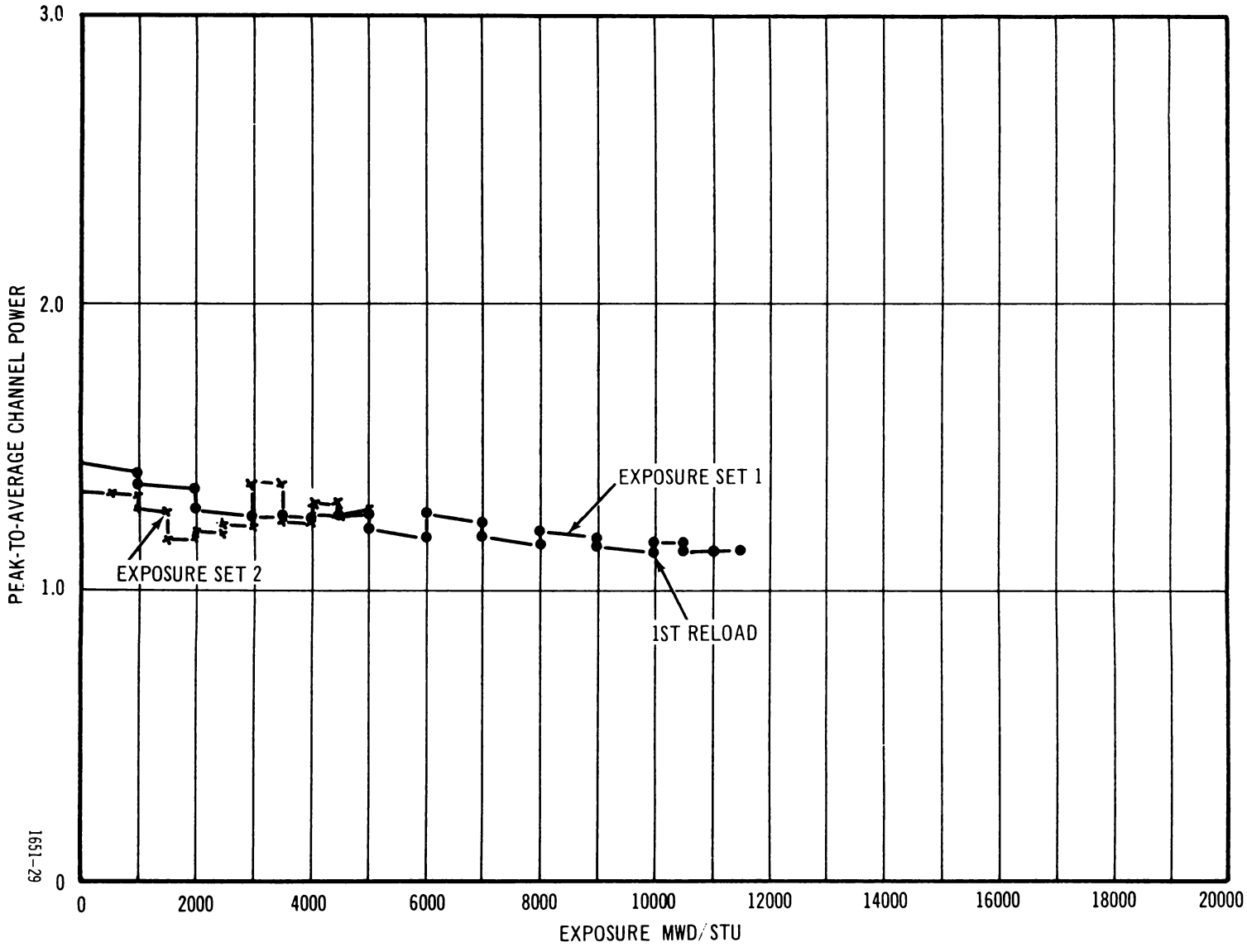



Figure I-35. Variation of Peak-to-Average Channel Power with Exposure

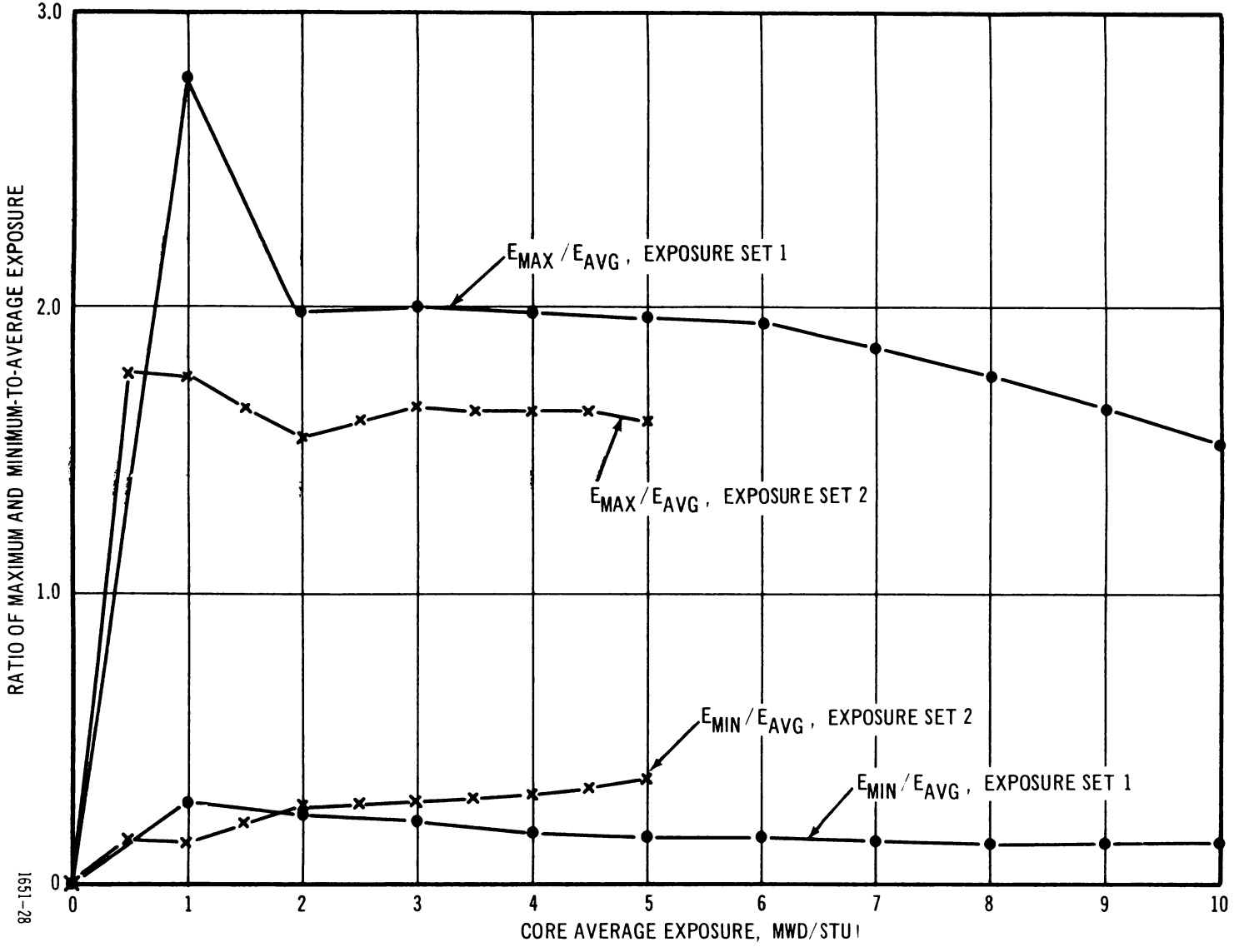



Figure I-36. Variation of $E_{max/E_{avg}}$ and $E_{min/E_{avg}}$ with Exposure

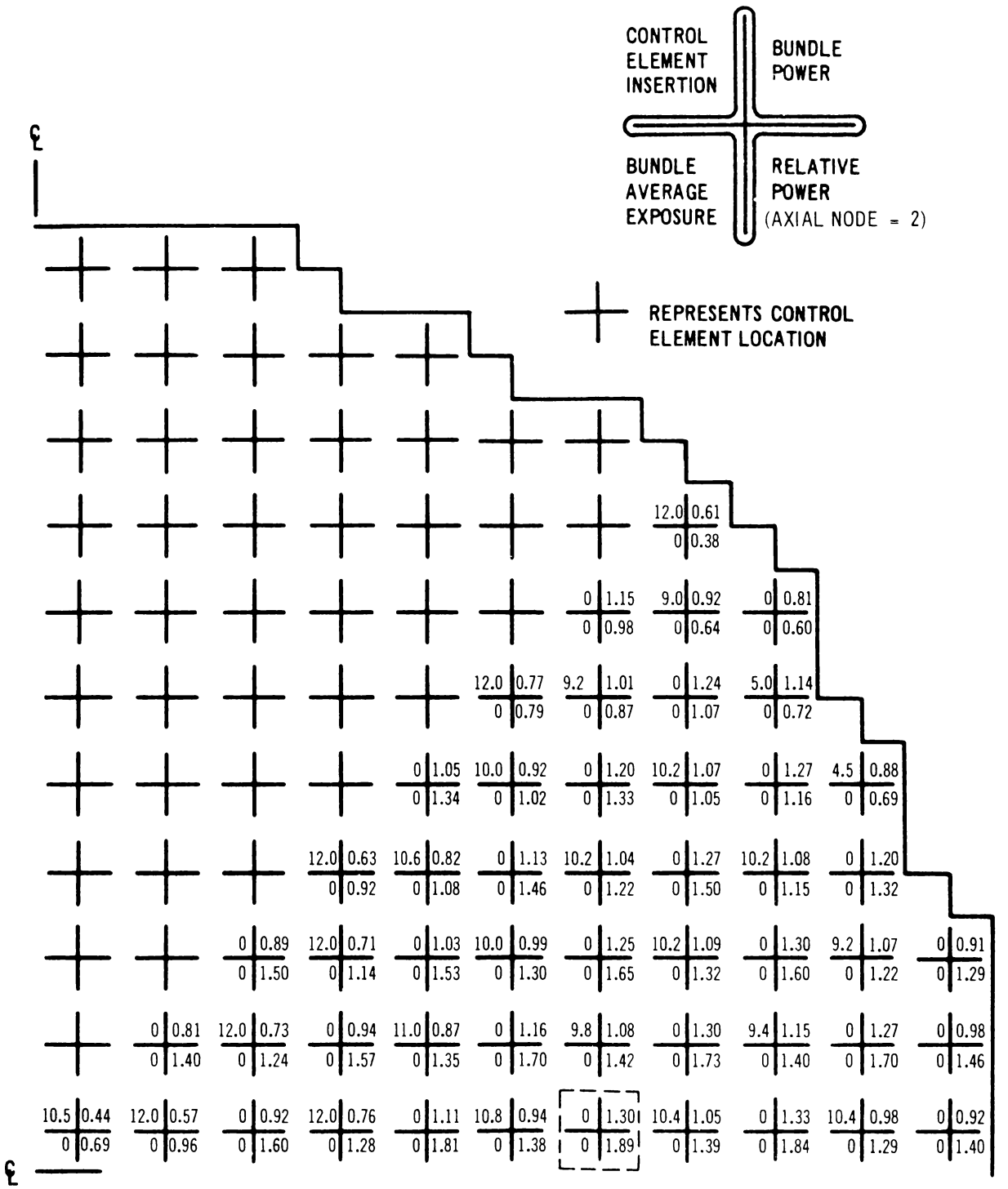



Figure I-37. Core of Arrangement A. Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure. Core Average Exposure = 0. Eigenvalue = 1.004

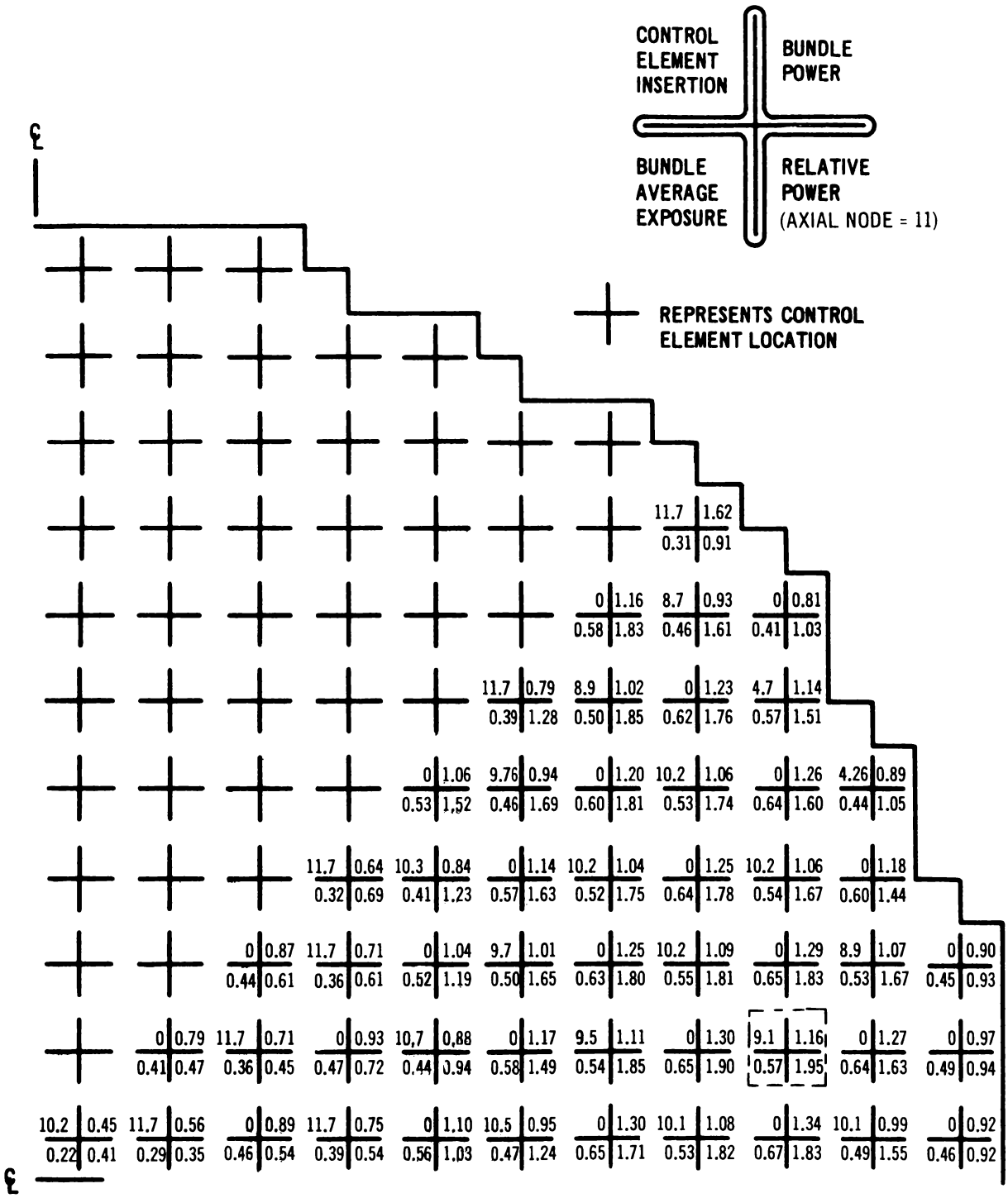



Figure I-38. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 500 MWD/STU, Eigenvalue = 1.002

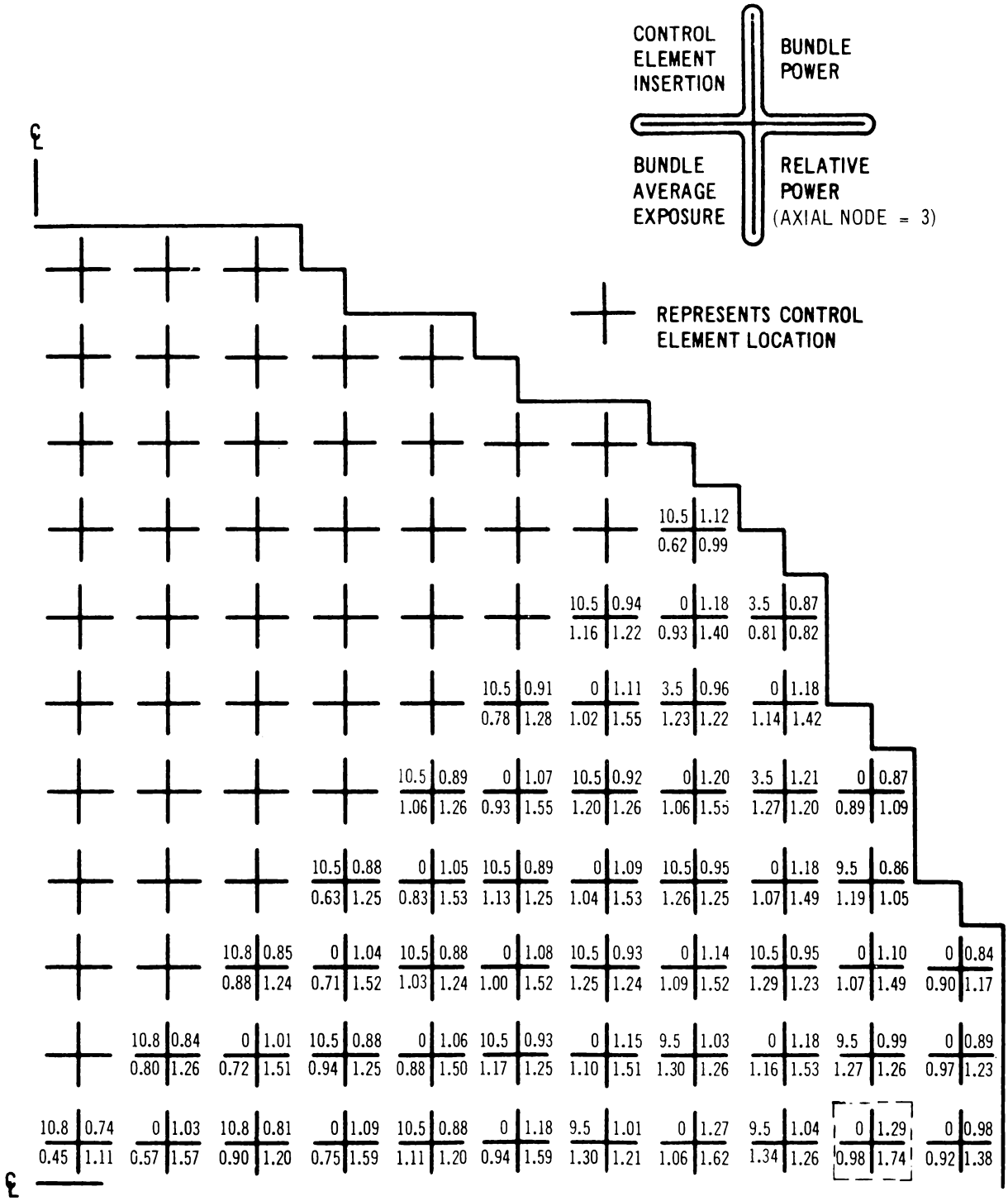



Figure I-39. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure 1000 MWD/STU, Eigenvalue = 1.005

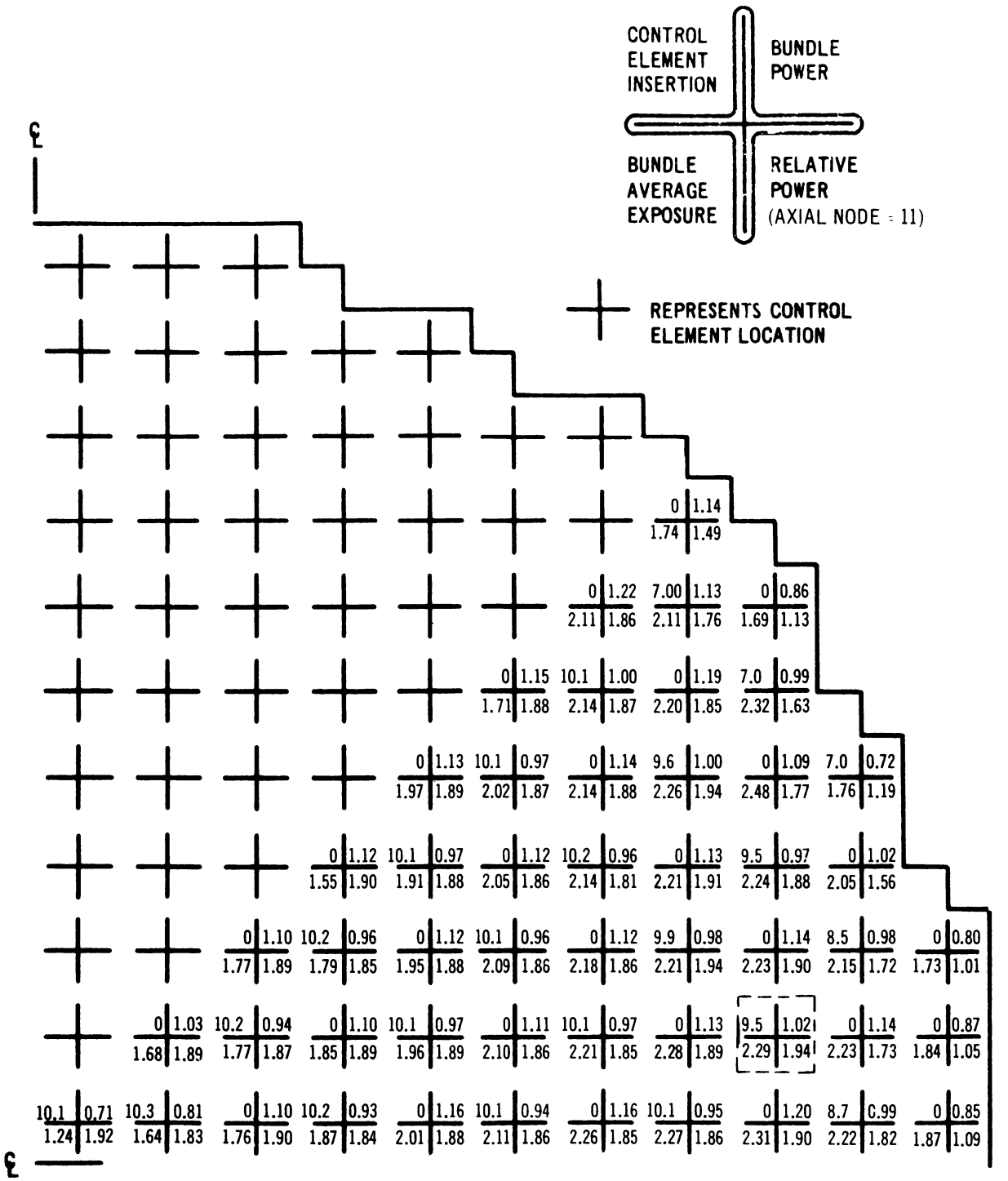



Figure I-40. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 2000 MWD/STU, Eigenvalue = 1.003

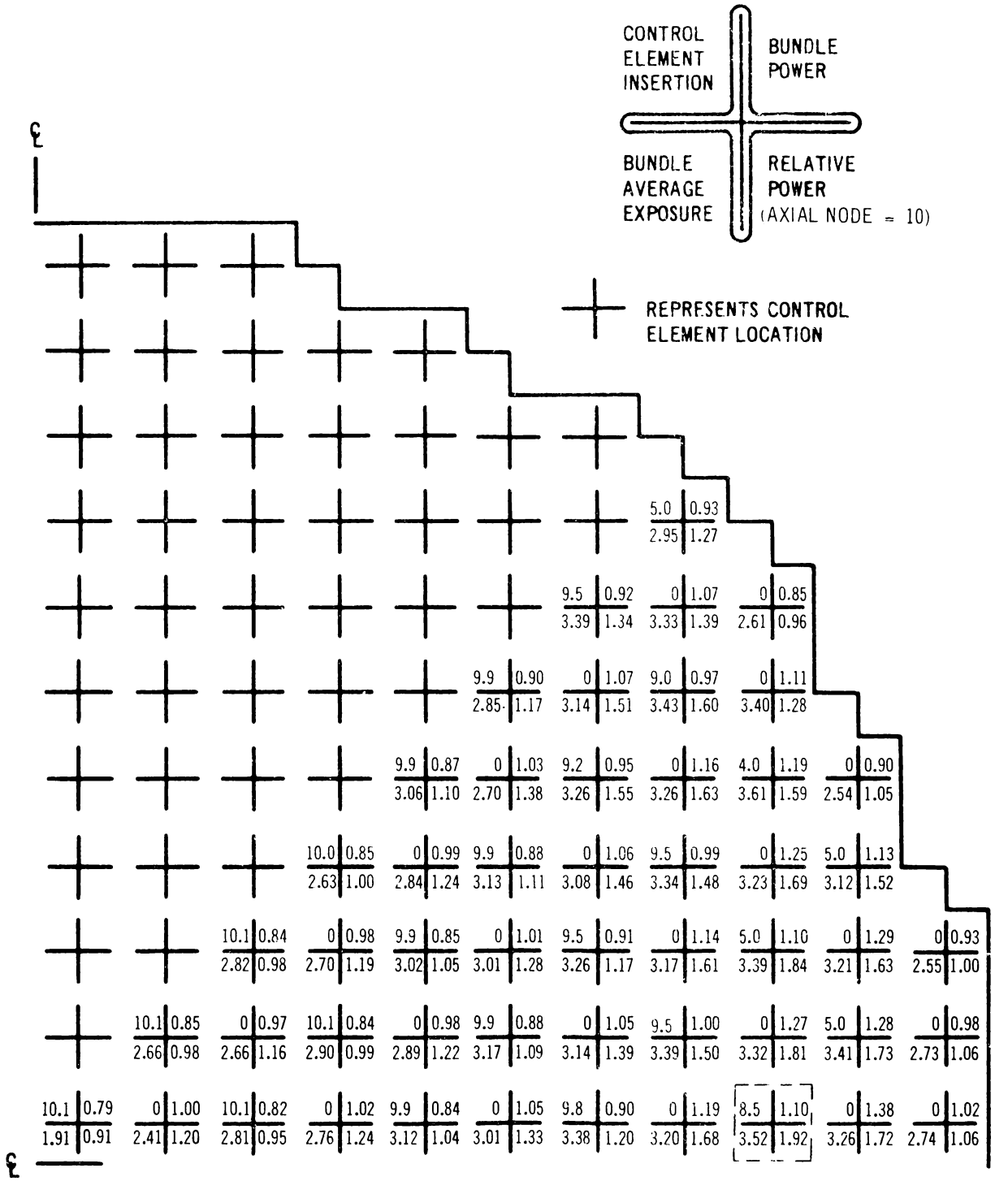



Figure I-41. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 3000 MWD/STU, Eigenvalue = 1.005

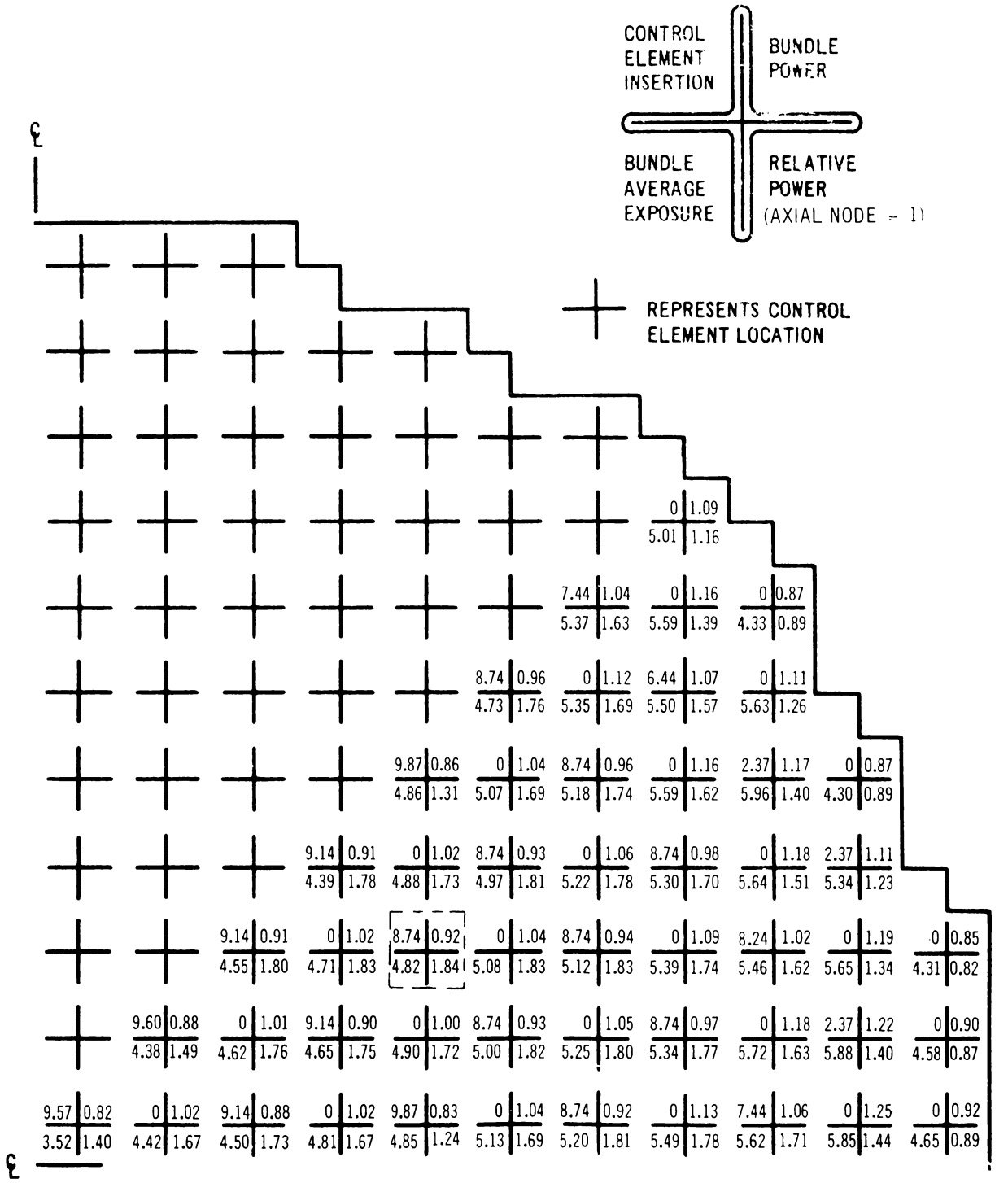



Figure I-43. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 5000 MWD/STU, Eigenvalue = 1.004

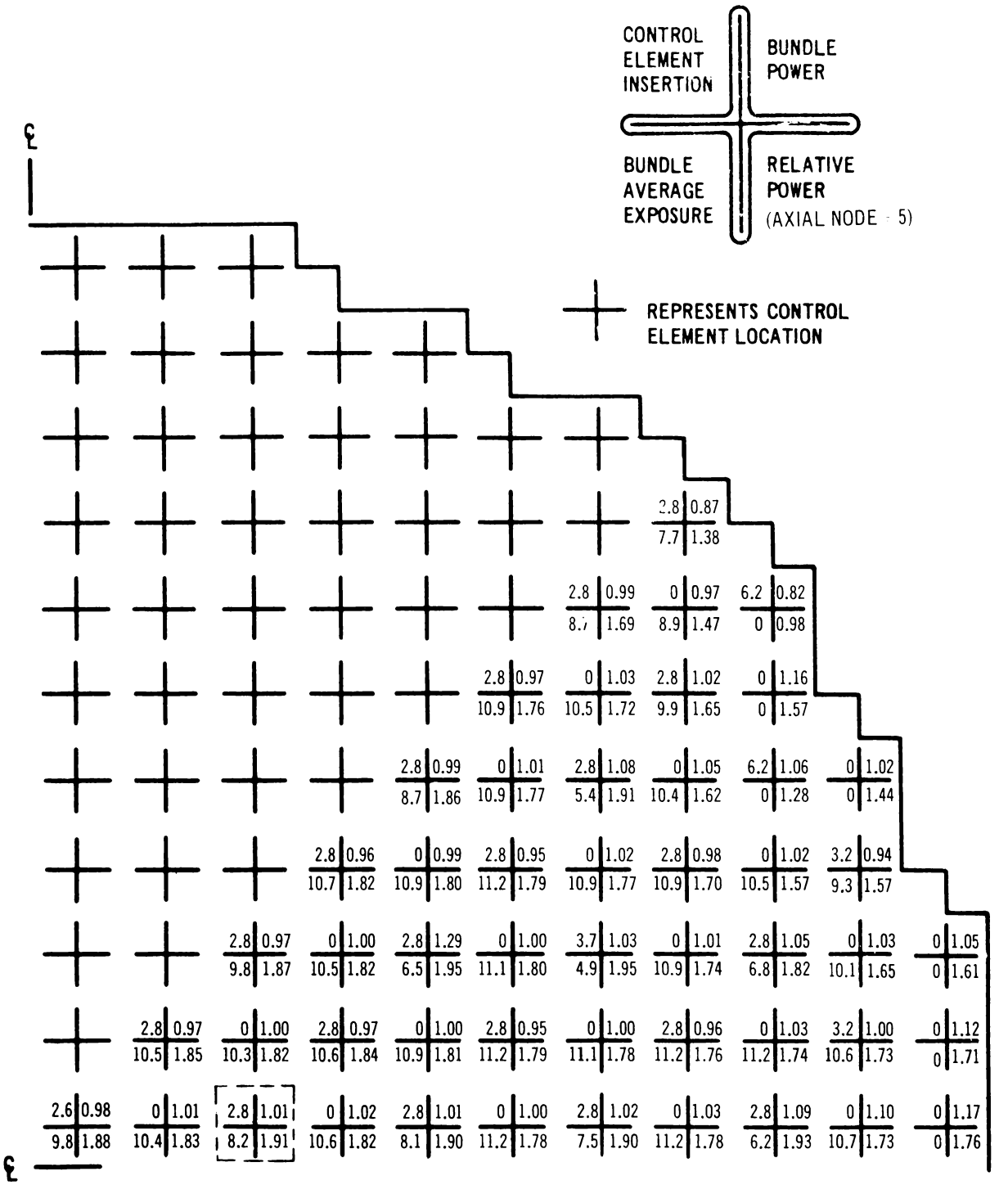



Figure I-44. Core of Arrangement A, Control Element Insertion, Bundle Power, Relative Power at Plane Containing Peak Power, and Exposure, Core Average Exposure = 10,000 MWD/STU, Eigenvalue = 1.004

# APPENDIX II

## FABRICATION OF PRESSURE VESSEL

The single-unit 1000 MW plant described in this report requires a vessel body of 21 feet-10 inches inside diameter and 54 feet-10 inches outside length, fabricated from carbon steel plate 8-3/4 inches thick, as shown in Figure 2-1.

The feasibility of fabricating such a vessel for this plant depends upon existing experience with the forming, welding, and heat treating of plate of this thickness, and upon the capacity of U. S. manufacturers to produce a vessel of this size to the required codes and specifications.

Procurement specifications, drawings, and delivery schedule were submitted to leading manufacturers of pressure vessels. The specifications included a questionnaire related to vessel feasibility, required development work, fabricators' capability, and vessel transportation problems. Conferences were held with key management and engineering personnel at the responding companies, and all aspects of the design, fabrication, testing and transportation were discussed.

### 1. FEASIBILITY

Experience in the fields of metallurgy and vessel fabrication is extensive and indicates that no portion of the 21-foot–10-inch I. D. vessel will require development of techniques or procedures that have not already been applied successfully in the construction of heavy-walled vessels. Fabricators have formed, welded and satisfactorily heat-treated plate sections into cylindrical vessel courses in greater thicknesses than the 8-3/4 inches required by ASME Code Section I for this vessel. Should new Code Section III prevail at the time of vessel fabrication, the wall would need to be only 6-1/2 inches thick, which would diminish the fabrication and handling time and costs.

The flange is too great a diameter to obtain as a single continuous forged ring. It will be obtained as three or four segments and welded into a continuous ring at the fabrication plant. The vessel heads will be fabricated by standard orange-peel and formed-dome construction.

Existing capacity of equipment in the plants of some of the fabricators, while in certain cases inadequate at this writing, is being expanded in 1964, as discussed under "Capability" below. Delivery of the vessel will be 3-1/2 years following placement of an order, and determines the specified plant construction schedule.

## 2. MATERIALS

Plate material for the 8-3/4 inches thick vessel wall is made in accordance with ASME Specification SA302B, and forging material is in accordance with SA336 modified in accordance with ASME Code Case 1236.

SA302B plate material up to 9 inches thick has been employed in the construction of vessels for nuclear reactors which are now in operation. This thickness is considered to be the upper limit to which this material may be manufactured and heat treated to obtain consistent and uniform properties throughout. SA336-Modified forgings have been used for reactor vessel flanges of section thicknesses equal to those required for this vessel.

SA302B and SA336-Modified alloys were designed to be used in thinner sections in the normalized condition, but develop the specified properties in sections from 4 inches to 9 inches thick by quenching and tempering. They have been shown to develop the desired microstructure and properties at 9-inch thickness by controlling the alloy content at the upper limits to ensure maximum hardenability, and by employing very rapid quenching techniques.

A comprehensive documentation of this general subject was published in the Welding Research Supplements of the Welding Journal, Volume XXVII, No. 10, October 1962, Pages 322S–447S. Mr. R. E. Lorentz, Jr, presented the 1962 Adams Lecture entitled, "Utilization of Quenching and Tempering for Improvement in Properties of Low-Alloy Steels in Heavy Thickness for Welded Construction".

## 3. WELDING

Welding materials and procedures have been developed and employed for construction of vessels 9 inches thick manufactured from SA302B and SA336-Modified. No difficulties in welding are anticipated.

## 4. DEVELOPMENT

Inasmuch as the vessel fabricators have had past experience in all phases of required technique, there is no new development required. Establishing weld procedures and proper handling sequence will be firmed up while waiting for delivery of plates and forgings from material suppliers. Fabricators expressed confidence that their existing heat treatment procedures would obtain the desired minimum mechanical properties within the interior of the heavy wall sections and that no specific development would be required in this regard.

## 5. CAPABILITY

Several fabricators are completely equipped to form and prepare the plates for this vessel. Equipment for fabricating, handling and heat-treating the entire vessel exists in some shops, whereas others are in the process of expanding. In 1964, there will be several companies with facilities for making and testing the complete vessel, including the use of linear accelerators for radiographing the welds.

## 6. SCHEDULING

Assuming February, 1964 as the date for placement of the vessel order, engineering on the vessel would be sufficiently complete to order materials, plates and forgings, within one month.

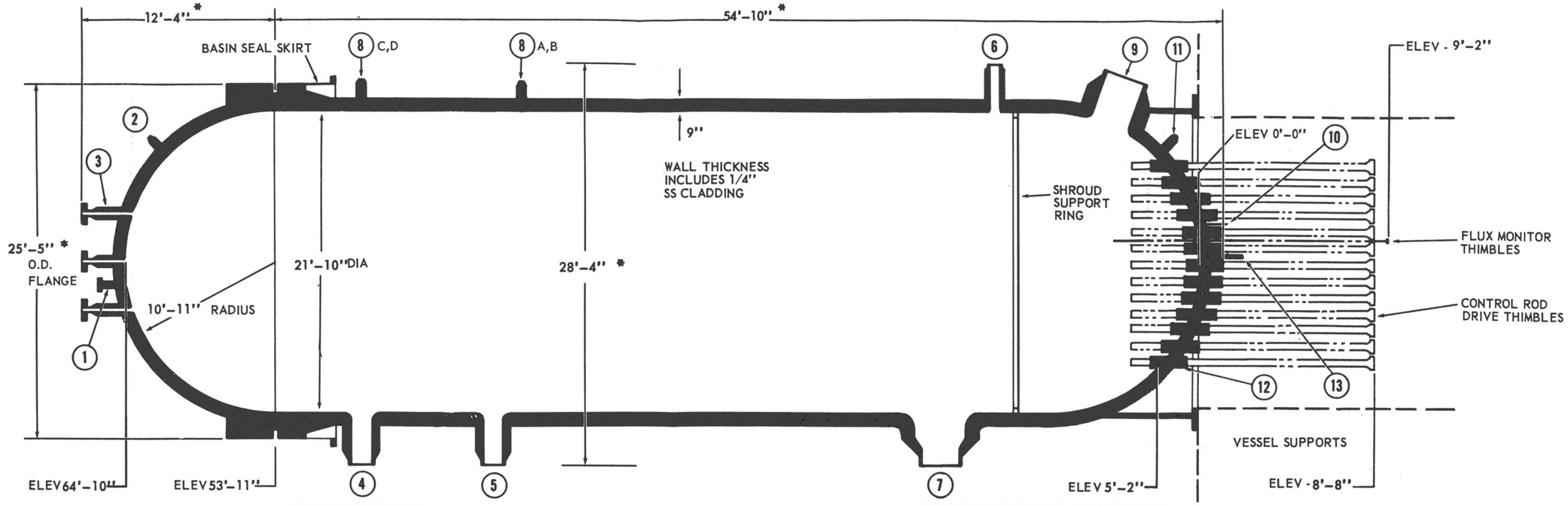
Plate sections would commence arriving at the plant approximately three months later. All head and cylinder parts can be formed and prepared in existing plant facilities. These parts would proceed immediately to fabrication, testing heat treatment, either in existing facilities or in new facilities which will be ready by the time they are needed.

The completed vessel will be delivered to the reactor site 3-1/2 years after placement of the order.

Transportation of the vessel by barge to the dock at the site will not be significantly more difficult than for smaller size vessels.

## 7. 500 MW VESSEL

Figure 2-2 presents the principal dimensions of the vessel for the 500 MW reactor. Because of the smaller size, thinner wall and lighter weight, the vessel will be easier to build than that for the 1000 MW reactor. Existing facilities can be used.




**NOZZLE SCHEDULE**

| No. | QUANT | FUNCTION | SIZE AND DESCRIPTION | LOCATION |
|---|---|---|---|---|
| | | TOPHEAD | | |
| 1 | 1 | VENT | 2" FLANGE | TOP HEAD |
| 2 | 0 | INSTRUMENTATION | – NOZZLE | TOP HEAD |
| 3 | 41 | PRESSURE RELIEF | 4" FLANGE | TOP HEAD |
| | | CYLINDER | | |
| 4 | 4 | OUTLET STEAM | 24"NOZZLE | ℄ EL: 48'–6" EQ. SPACE |
| 5 | 6 | FEEDWATER | 10"THERMAL SLEEVE | ℄ EL: 38'–3" EQ. SPACE |
| 6 | 3 | CORE SPRAY | 6"THERMAL SLEEVE | ℄ EL: 12'–0" EQ. SPACE |
| 7 | 6 | RECIRCULATION OUTLET | 40/32"REDUC. NOZZLE | ℄ EL: 16'–0" EQ. SPACE |
| 8 | | INSTRUMENTATION | | |
| | 6 | A. PROTECTION SYSTEM | 1" NOZZLE | ℄ EL: 38'–3" EQ. SPACE |
| | 4 | B. WATER LEVEL | 1" NOZZLE | ℄ EL: 38'–3" EQ. SPACE |
| | 2 | C. STEAM PRESSURE | 1" NOZZLE | ℄ EL: 50'–4" EQ. SPACE |
| | 1 | D. FILL LEVEL | 1" NOZZLE | ℄ EL: 49'–10" EQ. SPACE |
| | | BOTTOM HEAD | | |
| 9 | 6 | RECIRCULATION INLET | 32"NOZZLE | BOTTOM HEAD |
| 10 | 89 | FLUX MONITOR | 1½" NOZZLE | BOTTOM HEAD |
| 11 | 2 | CORE DIFF. PRESSURE | 1" NOZZLE | BOTTOM HEAD |
| 12 | 376 | CONTROL ROD DRIVE | 5" I.D. BUSHING NOZ. | BOTTOM HEAD |
| 13 | 1 | DRAIN | 2" NOZZLE | BOTTOM HEAD |
| 14 | 36 | THERMOCOUPLE TAPS | | DISTRIBUTED OVER VESSEL |

**REACTOR VESSEL WT. & LOAD DATA**

| | | |
|---|---|---|
| DESIGN PRESSURE | 1250 psig | |
| DESIGN TEMPERATURE | 575°F | |
| DESIGN LIFE | 40 YEARS | |
| EARTHQUAKE LOAD | 10% VERTICAL LOADING | |
| OPERATING PRESSURE | 1000 psig | |
| OPERATING TEMPERATURE | 546°F | |
| TOP HEAD & INSULATION WT | 271 | Kips |
| VESSEL & INSULATION WT | 1620 | Kips |
| INTERNALS–BOTTOM HEAD LOAD | 1051 | Kips |
| SHROUD SUPPORT RING LOAD | 296 | Kips |
| F.W. SPARGER SUPPORT LOAD | 9 | Kips |
| TOTAL INTERNALS DRY WT. | 1356 | Kips |
| WATER WT–REFUELING | 1873 | Kips |
| VESSEL SUPPORT LOAD–REFUELING | 5049 | Kips |
| WATER WT–OPERATIONS | 591 | Kips |
| VESSEL SUPPORT LOAD–OPERATIONS | 5186 | Kips |
| BOTTOM HD LOAD PER C.R.D. NOZZLE | 3 | Kips |
| NET THIMBLE WT | 147 | Kips |
| NET VESSEL BODY WT | 1600 | Kips |
| NET TOP HEAD WT | 265 | Kips |

(NOTE: 1 Kip = 1,000 Lbs)

NOTE: DRAWING NOT TO SCALE

* APPROX. OVERALL DIMENSIONS FOR SHIPPING

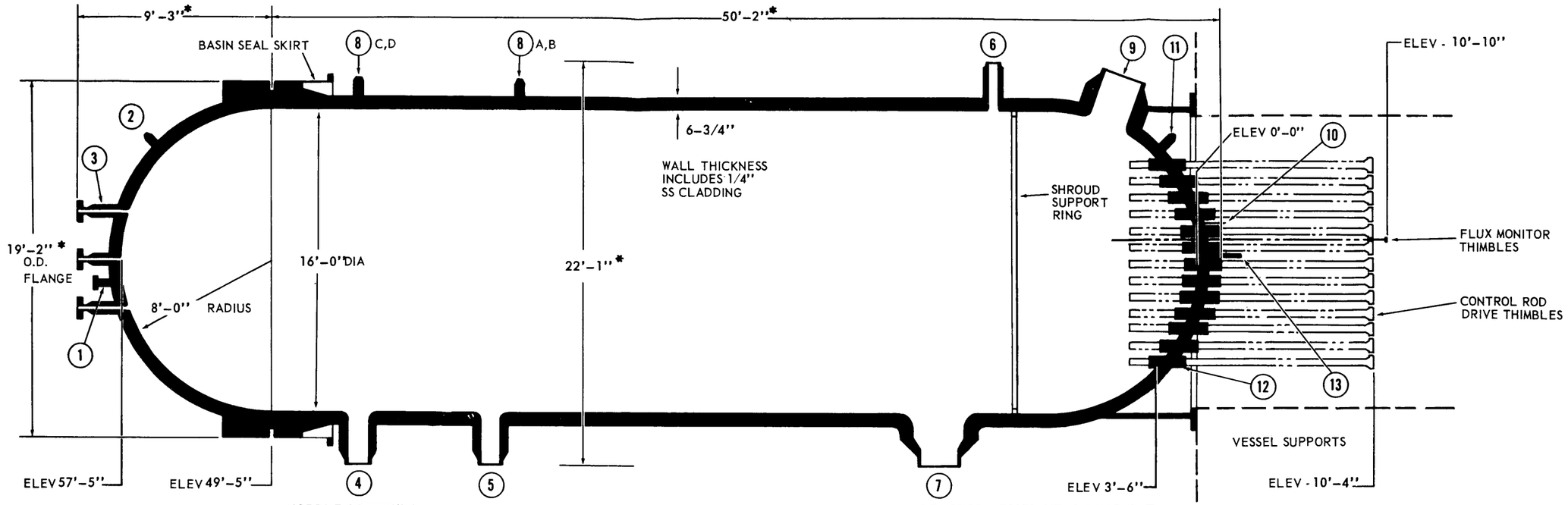
**PLANT REFERENCE**

SIZE: 1000 MWe

CYCLE: SINGLE NON-REHEAT

NOTES:
1. VESSEL TO BE IN ACCORDANCE WITH: ASME BOILER & PRESSURE VESSEL CODE SECT. VIII & SECT. 1, PAR. P–102, LATEST EDITION AND ADDENDA, CASE INTERPRETATIONS ON NUCLEAR PRESSURE VESSELS, AND GOVERNING REGULATIONS AT REACTOR SITE LOCATION.
2. MATERIAL: VESSEL, NOZZLE, & TOPHEAD, BASE MAT'L EQUIVALENT TO: ASME SA–336 & SA–302B. INTERNAL CLADDING MAT'L. EQUIVALENT TO: ASTM A–371–ER308.
3. TO BE SUPPLIED WITH VESSEL FOR FIELD INSTALLATION: CONTROL ROD DRIVE THIMBLES – LENGTH: 14'–6" FLUX MONITOR THIMBLES – LENGTH: 15'–2"

Figure 2-1. 1000 MWe Vessel Specification




NOTE: DRAWING NOT TO SCALE

* APPROX. OVERALL DIMENSIONS FOR SHIPPING

## NOZZLE SCHEDULE

| No. | QUANT | FUNCTION | SIZE AND DESCRIPTION | LOCATION |
|---|---|---|---|---|
| | | TOPHEAD | | |
| 1 | 1 | VENT | 2" FLANGE | TOP HEAD |
| 2 | 0 | INSTRUMENTATION | - NOZZLE | TOP HEAD |
| 3 | 21 | PRESSURE RELIEF | 4" FLANGE | TOP HEAD |
| | | CYLINDER | | |
| 4 | 4 | OUTLET STEAM | 18" NOZZLE | ℄ EL: 44'-10" EQ. SPACE |
| 5 | 4 | FEEDWATER | 8" THERMAL SLEEVE | ℄ EL: 35'-4" EQ. SPACE |
| 6 | 2 | CORE SPRAY | 6" THERMAL SLEEVE | ℄ EL: 9'-1" EQ. SPACE |
| 7 | 3 | RECIRCULATION OUTLET | 40/32" REDUC. NOZZLE | ℄ EL: 12'-10" EQ. SPACE |
| 8 | | INSTRUMENTATION | | |
| | 6 | A. PROTECTION SYSTEM | 1" NOZZLE | ℄ EL: 35'-4" EQ. SPACE |
| | 4 | B. WATER LEVEL | 1" NOZZLE | ℄ EL: 35'-4" EQ. SPACE |
| | 2 | C. STEAM PRESSURE | 1" NOZZLE | ℄ EL: 45'-6" EQ. SPACE |
| | 1 | D. FILL LEVEL | 1" NOZZLE | ℄ EL: 45'-0" EQ. SPACE |
| | | BOTTOM HEAD | | |
| 9 | 3 | RECIRCULATION INLET | 32" NOZZLE | BOTTOM HEAD |
| 10 | 43 | FLUX MONITOR | 1½" NOZZLE | BOTTOM HEAD |
| 11 | 2 | CORE DIFF. PRESSURE | 1" NOZZLE | BOTTOM HEAD |
| 12 | 185 | CONTROL ROD DRIVE | 5" I.D. BUSHING NOZ. | BOTTOM HEAD |
| 13 | 1 | DRAIN | 2" NOZZLE | BOTTOM HEAD |
| 14 | 36 | THERMOCOUPLE TAPS | | DISTRIBUTED OVER VESSEL |

## REACTOR VESSEL WT. & LOAD DATA

| | | |
|---|---|---|
| DESIGN PRESSURE | 1250 psig | |
| DESIGN TEMPERATURE | 575°F | |
| DESIGN LIFE | 40 YEARS | |
| EARTHQUAKE LOAD | 10% VERTICAL LOADING | |
| OPERATING PRESSURE | 1000 psig | |
| OPERATING TEMPERATURE | 546°F | |
| TOP HEAD & INSULATION WT | 127 Kips | |
| VESSEL & INSULATION WT | 807 Kips | |
| INTERNALS-BOTTOM HEAD LOAD | 520 Kips | |
| SHROUD SUPPORT RING LOAD | 125 Kips | |
| F.W. SPARGER SUPPORT LOAD | 5 Kips | |
| TOTAL INTERNALS DRY WT. | 650 Kips | |
| WATER WT-REFUELING | 1110 Kips | |
| VESSEL SUPPORT LOAD-REFUELING | 2660 Kips | |
| WATER WT-OPERATIONS | 361 Kips | |
| VESSEL SUPPORT LOAD-OPERATIONS | 2750 Kips | |
| BOTTOM HD LOAD PER C.R.D. NOZZLE | 3 Kips | |
| NET THIMBLE WT | 72 Kips | |
| NET VESSEL BODY WT | 793 Kips | (NOTE: 1 Kip = 1,000 Lbs) |
| NET TOP HEAD WT | 125 Kips | |

## PLANT REFERENCE

SIZE: 500 MWe

CYCLE: SINGLE NON-REHEAT

NOTES:
1. VESSEL TO BE IN ACCORDANCE WITH: ASME BOILER & PRESSURE VESSEL CODE SECT. VIII & SECT. 1, PAR. P-102, LATEST EDITION AND ADDENDA, CASE INTERPRETATIONS ON NUCLEAR PRESSURE VESSELS, AND GOVERNING REGULATIONS AT REACTOR SITE LOCATION.
2. MATERIAL: VESSEL, NOZZLE, & TOPHEAD. BASE MAT'L EQUIVALENT TO: ASME SA-336 & SA-302B. INTERNAL CLADDING MAT'L. EQUIVALENT TO: ASTM A-371-ER308.
3. TO BE SUPPLIED WITH VESSEL FOR FIELD INSTALLATION: CONTROL ROD DRIVE THIMBLES - LENGTH: 14'-6" FLUX MONITOR THIMBLES - LENGTH: 15'-2"

GPO 812-316-3

# APPENDIX III

## INSTALLATION OF PRESSURE VESSEL AND GENERATORS

The reactor vessel for Alternates A and B weighs approximately 800 tons, without the head. Because of its weight and physical dimensions delivery will be made to the site by barge. Delivery and unloading will need to be coordinated with the river water level, but this is not expected to present any problems.

The vessel will be on skids and in a horizontal position on the delivery barge. The barge will be positioned over a prepared bed at the river's edge, in or adjacent to the circulating water discharge channel, and will be lowered by sinking the barge onto this bed. With the barge thus stabilized, the vessel will be skidded off onto the river bank and from there to a position at the south end of the reactor building. At this point the upper end of the vessel will be positioned in a special jacking frame and carriage designed to bring the vessel to a vertical position, and then raised so that the reactor bottom will be above an opening in the south wall of the spent fuel storage pool at an approximate elevation of 76 feet.

The upper part of the jacking frame is so designed that it will move on rails placed on temporary supports in the fuel storage pool to a position directly above the final location of the reactor vessel within the building. The vessel will then be jacked down to position.

In order to move the vessel into the reactor building in the manner described above, it will be necessary to leave the south wall of the spent fuel storage pool open at approximately the 76 foot elevation and above until the reactor vessel is installed. The concrete in this section of wall will be completed immediately thereafter. A majority of the crane bay steel and roof trusses can be erected prior to installation of the vessel, without interference with movement of the jacking frame carriage.

Prior to its use in positioning the reactor vessel, the jacking frame will be utilized to unload and install the generators, which will be delivered by rail at the east end of the turbine building before equipment is installed in the transformer yard. The generators will be jacked to the level of the operating floor (elevation 65 feet), and moved west across that floor to their foundations. If delivery of the generator stators is made by barge, they will be skidded to the jacking frame in a manner similar to that proposed for the reactor vessel.

# APPENDIX IV

## COSTS

Plant construction and operating costs were determined on the basis of AEC ground rules and certain other assumptions pertinent to this study.

### 1. FUEL COSTS

The basic assumptions related to the economics of nuclear fuels as developed by the General Electric Company and reviewed by the U. S. Atomic Energy Commission are described below. Two periods of time were chosen; initially, 1969, representing the probable earliest year of fuel utilization for a 1000 MWe plant; and 1979, when the cores attain equilibrium.

#### 1.1 General Industry Conditions

It is assumed that the nuclear power industry experiences a growth between now and 1979 equivalent to that predicted in the U. S. AEC report to the President, and that the United States reactor complex during that period is primarily fueled with slightly enriched uranium. The technology will be a further evolution of currently developed water reactor cycles and not an abrupt transition to radically different cycles for a significant installed nuclear capacity. It is assumed that a possible reduction in the rate of weapons material production will not cause an increase in charges for use of government facilities; for example, the price for isotopic separation in government diffusion plants is assumed not to increase. It is further assumed that the option of private ownership of nuclear fuel will become available prior to 1969, and will have become mandatory by 1979.

The 1969 forecasts assume that the fuel is prepared in 1969 and charged into a reactor for initial operation that same year. The 1979 forecast is for that same reactor rather than a reactor created in 1979. The reactor is, however, expected to be able to utilize most of the advancements made in fuel cycle technology and economics during its first 10 years of life and thereafter.

The forecasts are made in terms of constant 1964 dollars, and do not include escalation except in the specific case of 1969 reprocessing, which has been escalated from a slightly lower current base.

Fuel exposure for the 1969 core is stipulated as 16.500 thermal megawatt-days per metric ton and 22,000 MWD/MT in 1979. The 22,000 MWD/MTU exposure is considered available for all fuel beyond the initial core.

## 1.2 Raw Material Costs

During 1969, under the AEC stretchout program for raw materials procurement, ore concentrate ($U_3O_8$ or yellow cake) is expected to be procured by the government at $6.00/lb. The regulations stipulate a maximum of $6.70/lb, but the average cost is taken as $6.00. It is assumed that a competitive free market for ore concentrates will exist in 1969 at a level of $4.00/lb. It is assumed that in 1969 there is an option to lease fuel from government and/or utilize private ownership at financing rates discussed below. In 1979, the government leasing option is assumed to be no longer available.

Conversion cost of ore concentrate ($U_3O_8$ to $UF_6$) is assumed in 1969 to be $1.00/lb of uranium whether government or private, which represents a slight improvement over the current level of approximately $1.15/lb. In 1979, this cost is expected to be reduced by a factor of two, due primarily to higher throughputs even without process development beyond fluoride volatility conversion.

## 1.3 Isotopic Separation

The current AEC charge of $30/kg of separative work is anticipated to remain constant through 1969 and to fall to $25/kg during the next 10 years by achievement of operating efficiencies and nearly complete amortization of plant and equipment. Cost increases from reduced operating levels or standby charges occasioned by reduced production of defense materials are assumed to be carried in defense budgets and not liquidated through the nuclear power program. No changes in technical methods are anticipated during the forecast period. Gas centrifuges or similar new processes are not anticipated, although diffusion plants in Britain, France and possibly the Soviet bloc will create competition in the 1970's.

## 1.4 Fabrication Price

The price of fabrication for the initial core is taken as $115/kg of uranium including use charge during fabrication, conversion of $UF_6$ to $UO_2$, losses and scrap reprocessing. This price is anticipated to drop to $100/kg of uranium in 1979 depending primarily on increase in volume of business. This price is based on a commitment today for delivery at that time. If an estimate were made of a quotation in 1979 for then-current delivery, it

would definitely be lower than a quotation today. Speculation about such a case could yield a price as low as $50/kg. Conditions which can be expected to reduce fabrication costs during this 10-year period are design of less-costly fuel assemblies, employment of advanced manufacturing techniques such as vibratory compaction rather than sintering of pellets, and metallurgical experience providing cladding material of lower cost alloys.

## 1.5 Spent Fuel Shipment

A 1969 charge of $2.00/lb of uranium for transportation of spent fuel from reactor site to reprocessing site assumes a maximum distance of 1000 miles in the continental United States and rail access to reactor and reprocessing facilities. It is anticipated that this figure will be reduced to $1.50/lb of uranium in 1979, because of increased operating economy due to greater volume of spent fuel shipment and more efficient utilization of shipping casks.

## 1.6 Reprocessing

The private plant reprocessing charge for a New York State site is assumed as $23,500/metric ton of uranium escalated to $24,000/metric ton in 1969 (all other numbers are in constant U.S. dollars). A turnaround charge differing from previous AEC turnaround predictions is assumed as $8,000/metric ton, for a total of $32,000/ton U fuel discharged, including turnaround. By 1979, the rate is expected to fall to $23,500/metric ton, and the requirement for turnaround will be eliminated by accepting the less costly penalty for interface mixing of batches run through the reprocessing plant in quick succession, with some degradation of the higher isotopic batches. It is anticipated that solvent extraction will continue to be the main process utilized throughout the period, and that advanced processes such as fluoride volatility reprocessing can not be counted on for reducing these costs.

## 1.7 Reconversion

The chemical conversion of uranyl nitrate (the chemical form of the reprocessing plant product) to uranium hexafluoride for re-enrichment is assumed to be $1.00/lb in 1969, falling by 25 percent in the subsequent ten years. Earlier charges for conversion of plutonium nitrate to plutonium metal have been dropped from fuel cost on the basis that the AEC will procure plutonium nitrate for recycle into thermal reactors and for use in fast reactors. These will use oxides converted from nitrate, thus practically eliminating the need for plutonium as a metal.

## 1.8 Plutonium Credit

The current probable procurement price is $10.00/gm for the fissile isotopes Pu-239 and Pu-241 in the form of nitrate, $Pu(NO_3)_4$. This AEC price is based on the estimated

fuel value of plutonium in thermal power reactors. Procurement for critical experiments, safety experiments, and prototypes is assumed to consume all power reactor-produced plutnoium in the U. S. through the early 1970's.

1.9 Financing Rate

1969 fuel lease from the government is assumed 4.75 percent, which is the current use-charge rate. In the alternate case of private ownership in 1969, a carrying charge of 10 percent is assumed, which is extended through 1979, representing a non-depreciating fixed charge for the uranium inventory included in the reactor operator's equity.

The Assumptions are tabulated in Table IV-1.

Table-IV-1

FUEL COST ASSUMPTIONS

| Plant Arrangement Cost Assumptions (2) | | Initial Load | | | Equilibrium Core | | | |
|---|---|---|---|---|---|---|---|---|
| | | A Prior AEC (1) | A 1969 AEC | A 1969 Private | A 1979 Private (1) | B 1979 Private | C 1979 Private (per reactor) | D 1979 Private (per reactor) |
| **A. Design Parameters** | | | | | | | | |
| 1. Fuel enrichment when charged to reactor | % | | | | | | | |
| a. Standard fuel rods - 31 per assembly | | 2.10 | | | 2.44 | 2.45 | 2.48 | 2.49 |
| b. Special fuel rods - 5 per assembly | | 1.45 | | | 2.01 | 2.02 | 2.05 | 2.06 |
| c. Average | | 2.01 | | | 2.38 | 2.39 | 2.42 | 2.43 |
| 2. Fuel enrichment when discharged from reactor, all fuel rods | % | 0.81 | | | 0.82 | 0.82 | 0.84 | 0.85 |
| 3. Average fuel exposure | MWD/STU | 15,000 | | | 20,000 | 20,000 | 20,000 | 20,000 |
| | MWD/MTU | 16,500 | | | 22,000 | 22,000 | 22,000 | 22,000 |
| 4. Plutonium concentration in discharged fuel | | | | | | | | |
| a. Total plutonium | g/kgU(3) | 6.49 | | | 7.56 | 7.56 | 7.56 | 7.56 |
| b. Fissile plutonium only (Pu-239, 241) | g/kgU(3) | 4.47 | | | 5.02 | 5.02 | 5.04 | 5.04 |
| 5. Rated gross power level (thermal) | MWt | 3,194 | | | 3,194 | 3,054 | 1,600 | 1,530 |
| 6. Rated net power level (electrical) | MWe | 1,000 | | | 1,000 | 1,000 | 500 | 500 |
| 7. Reactor fuel load per core | MTU | 208.0 | | | 208.0 | 193.3 | 102.9 | 98.57 |
| 8. Total U in discharge of this load | MTU | 203.0 | | | 201.6 | 187.4 | 99.75 | 95.55 |
| **B. Operating Parameters Set by Industry** | | | | | | | | |
| 1. Predicted plant operating factor | % | 80 | | | 80 | | | |
| 2. Average fabrication time-batch | days | 50 | | | 30 | | | |
| 3. Shipping time, fabricator to reactor (4) | days | 20 | | | 20 | | | |
| 4. Time interval between delivery of fuel batch to reactor site and charging to reactor (4) | days | 60 | | | 60 | | | |
| 5. Shipping time, reactor to chemical processing site | days | 20 | | | 20 | | | |
| 6. Process lag time, spent fuel processing | | | | | | | | |
| a. Separation | days | 30 | | | 30 | | | |
| b. Conversion | days | 5 | | | 5 | | | |
| 7. Spare fuel maintained on hand at all times, exclusive of discrete charging batches (5) | MTU | 2.18 | | | 2.18 | 2.04 | 1.09 | 1.09 |
| 8. Batch size charged to reactor per refueling | MTU | 41.60 | | | 41.60 | 38.66 | 20.58 | 19.71 |
| 9. Uranium discharged from reactor per refueling | MTU | 40.60 | | | 40.32 | 37.48 | 19.95 | 19.11 |
| 10. Number of discharged batches per chemical processing cycle | | 1 | | | 1 | | | |
| **C. Operating Parameters Set by Spent Fuel Processor** | | | | | | | | |
| 1. Minimum decay cooling period for irradiated fuel | days | 120 | 120 | 120 | 120 | | | |
| 2. Irrecoverable loss of U during chemical separation | % (6) | 1.0 | 1.0 | 1.0 | 1.0 | | | |
| 3. Irrecoverable loss of Pu during chemical separation | % (6) | 1.0 | 1.0 | 1.0 | 1.0 | | | |

Table-IV-1 (Continued)

| Plant Arrangement Cost Assumptions (2) | | Initial Load | | | Equilibrium Core | | | |
|---|---|---|---|---|---|---|---|---|
| | | A Prior AEC (1) | A 1969 AEC | A 1969 Private | A 1979 Private (1) | B 1979 Private | C 1979 Private | D 1979 Private |
| C. Continued | | | | | | | | |
| 4. Irrecoverable loss of U during conversion | % (7) | 0.3 | 0.3 | 0.3 | 0.3 | | | |
| 5. Irrecoverable loss of Pu during conversion | % (7) | 1.0 | NA | NA | NA | | | |
| 6. Chemical separation plant, processing rate | MTU/day | 1.0 | 1.0 | 1.0 | 1.0 | | | |
| 7. Chemical separation plant, turnaround time | days | 8 | (See item E. 10) | | (See item E. 10) | | | |
| 8. Uranium conversion plant, processing rate | MTU/day | 1.0 | 1.0 | 1.0 | 1.0 | | | |
| D. Economic Parameters Set by Industry | | | | | | | | |
| 1. New fuel processing cost - FOB fabrication plant, freight allowed to site. Includes all costs prior to delivery at site except use charge during shipment from fabricator to site. | $/kgU | 115.0 | 115.0 | 115.00 | 100.00 | | | |
| 2. Shipping charge, reactor to chemical processing site | $/kgU (8) | 16.0 | 4.40 | 4.40 | 3.30 | | | |
| 3. Fixed charge rate, working capital | %/year | 13 | 10 | 10 | 10 | | | |
| 4. Working capital factor, core fabrication (9) | % | 60 | 51 | 51 | 51 | | | |
| E. Economic Parameters Set by AEC or Industry | | | | | | | | |
| 1. Use charge rate, nuclear fuel material (10) | %/year | 4.75 | 4.75 | 10 | 10 | | | |
| 2. Bases for determining uranium values | | | | | | | | |
| a. Cost of ore | $/lb$U_3O_8$ | 8.00 | 6.00 | 4.00 | 6.00 | | | |
| b. Conversion of ore to $UF_6$ | $/lbU | 1.15 | 1.00 | 1.00 | .50 | | | |
| c. Cost of feed to enrichment plant - derived from items a and b | $/kgU | 23.50 | 17.80 | 12.60 | 16.70 | | | |
| d. Separative cost | $/kgU | 30.00 | 30.00 | 30.00 | 25.00 | | | |
| e. Feed enrichment | % | 0.7115 | 0.7110 | 0.7110 | 0.7110 | | | |
| f. Tails assay (11) | % | 0.2531 | 0.2834 | 0.2834 | 0.2633 | | | |
| 3. Uranium value prior to irradiation, as $UF_6$ | $/kgU | | | | | | | |
| a. Standard fuel rods - 31 per assembly | | 157.05 | 133.63 | 111.54 | 147.29 | 148.12 | 150.63 | 151.46 |
| b. Special fuel rods - 5 per assembly | | 90.35 | 75.23 | 61.04 | 111.91 | 112.72 | 115.15 | 115.97 |
| c. Weighted average value | | 147.79 | 125.52 | 104.53 | 142.38 | 143.20 | 145.70 | 146.53 |
| 4. Uranium value at discharge, as $UF_6$ - all rods | $/kgU | 31.34 | 24.43 | 18.02 | 23.27 | 23.27 | 24.53 | 25.17 |
| 5. Conversion charge, UNH to $UF_6$ | $/kgU (12) | 5.60 | 2.20 | 2.20 | 1.65 | | | |
| 6. Conversion charge, Pu nitrate to Pu metal | $/g (13) | 1.50 | NA | NA | NA | | | |
| 7. Plutonium credit | | | | | | | | |
| a. Value of all isotopes, as metal | $/g | 9.50 | NA | NA | NA | | | |
| b. Value of fissile isotopes (Pu-239, 241) as nitrate | $/g (14) | NA | 10.00 | 10.00 | 10.00 | | | |
| 8. Shipping charge, chemical processing site to fuel supplier | $/kgU | 1.00(15) | (Included in item D. 2) | | (Included in item D. 2) | | | |

Table-IV-1 (Continued)

| | Plant Arrangement Cost Assumptions (2) | | Initial Load | | | Equilibrium Core | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | A Prior AEC (1) | A 1969 AEC | A 1969 Private | A 1979 Private (1) | B 1979 Private | C 1979 Private | D 1979 Private |
| E. | Continued | | | | | | | | |
| 9. | Chemical separation plant, basic charge | | | | | | | | |
| | a. AEC plant | $/day | 17,000 | NA | NA | NA | | | |
| | b. Private plant | $/MTU | NA | 24,000 | 24,000 | 23,500 | | | |
| 10. | Chemical separation plant, turnaround charge | | | | | | | | |
| | a. AEC plant | | (See item C.7) | NA | NA | NA | | | |
| | b. Private plant | $/MTU | NA | 8,000 | 8,000 | NA | | | |

NOTES: NA = Not Applicable

(1) Data listed in first column of Initial Load or Equilibrium Core apply to remaining columns unless listed otherwise.

(2) Basically two sets of cost conditions are forecast: those expected to prevail in 1969 for initial load operation (for both AEC- and privately-supplied uranium) and those assumed to prevail in 1979 for equilibrium operation. Cost data previously used by AEC is also applied to the initial load for comparative purposes. The sets of data are as follows:

Prior AEC - costs and procedures as listed in the Guide to Nuclear Power Cost Evaluation, with revisions as of 4/30/63.

1969 AEC - expected costs based on current announcements

1969 Private - change of uranium supply from AEC to private (assuming optional private ownership) with other costs same as "1969 AEC" case.

1979 Private - forecast costs (assuming mandatory private ownership of uranium)

(3) Per kg U discharged

(4) Time for initial load assumed to be same as for equilibrium, for purposes of comparison. Time for initial load would actually be longer.

(5) Assumption is 1% of number of assemblies in core load, rounded up to a whole number.

(6) Percentage of weight of element fed to separation

(7) Percentage of weight of element fed to conversion

(8) For "Prior AEC" case - cost per kg U shipped
For all other cases - cost per kg U charged to reactor

(9) For "Prior AEC" case - per AEC Guide. For all other cases - based on average value of fabrication inventory in core (50%) and quantity of spare fuel on hand (1% of core load).

(10) The term "use charge" is used for the leasing or financing of uranium whether AEC or privately supplied.

(11) For the AEC cases, values are optimized from given cost data. For the private cases, values are not derived from the corresponding cost data; they are optimized from assumed AEC cost conditions, as follows:

| | 1969 | 1979 |
|---|---|---|
| Feed cost ($/kg) | 17.80 | 17.80 |
| Separative cost ($/kg) | 30.00 | 25.00 |

(12) Per kg U fed to conversion

(13) Per g Pu fed to conversion

(14) Per g of fissile isotopes

(15) Per kg U shipped

## 2. OPERATION AND MAINTENANCE COSTS

The size of the plant staff was determined as 78 for Arrangements A and B, and 95 for C and D. These numbers, however, could be increased or decreased, depending upon the policies of the operating utility. A sum for contracted maintenance is included in the services costs.

Table IV-2 presents the total operating and maintenance costs, plus the details of employee salaries. As specified by the AEC ground rules, fringe benefits and payroll taxes have been included as 20 percent of direct payroll. Table IV-3 shows details of the projected costs for consumable Materials and Services.

TABLE IV-2

ESTIMATED OPERATION AND MAINTENANCE COSTS

| | Salary or Wage Rate ($/yr) | Single Reactor No. of Personnel | Single Reactor Annual Expense ($) | Twin Reactors No. of Personnel | Twin Reactors Annual Expense ($) |
|---|---|---|---|---|---|
| **Plant Management** | | | | | |
| Station Superintendent | 15,000 | 1 | 15,000 | 1 | 15,000 |
| Assistant Superintendent | 12,000 | 1 | 12,000 | 1 | 12,000 |
| Clerk-Steno Storekeeper | 5,200 | 2 | 10,400 | 2 | 10,400 |
| Clerk-Storekeeper | 5,200 | 4 | 20,800 | 4 | 20,800 |
| Secretary-Typist | 4,800 | 1 | 4,800 | 1 | 4,800 |
| Subtotal | | 9 | 63,000 | 9 | 63,000 |
| **Technical Staff** | | | | | |
| Supervising Engineer | 10,800 | 1 | 10,800 | 1 | 10,800 |
| Instrument Engineer | 9,000 | 1 | 9,000 | 1 | 9,000 |
| Radiochemist | 10,800 | 1 | 10,800 | 1 | 10,800 |
| Nuclear Engineer | 10,800 | 1 | 10,800 | 1 | 10,800 |
| Results Engineer | 8,000 | 1 | 8,000 | 1 | 8,000 |
| Health Physics Supervisor | 8,400 | 1 | 8,400 | 1 | 8,400 |
| Lab. Technician | 5,900 | 4 | 23,600 | 4 | 23,600 |
| Subtotal | | 10 | 81,400 | 10 | 81,400 |
| **Operating Staff** | | | | | |
| Shift Superintendent (L) | 10,800 | 5 | 54,000 | 5 | 54,000 |
| Senior Control Operator (L) | 7,800 | 5 | 39,000 | 5 | 39,000 |
| Control Operator (L) | 7,600 | 5 | 38,000 | 10 | 76,000 |
| Turbine Operator | 7,300 | 5 | 36,500 | 5 | 36,500 |
| Equipment Attendent | 6,200 | 5 | 31,000 | 10 | 62,000 |
| Relief Control Operator (L) | 7,600 | 5 | 38,000 | 5 | 38,000 |
| Relief Equipment Operator | 7,300 | 5 | 36,500 | 5 | 36,500 |
| Janitor | 4,700 | 4 | 18,800 | 4 | 18,800 |
| Watchman | 4,700 | 1 | 4,700 | 1 | 4,700 |
| Subtotal | | 40 | 296,500 | 50 | 365,500 |

(L) = Licensed Reactor Operator

TABLE IV-2 (Continued)

| | Salary or Wage Rate ($/yr) | Single Reactor No. of Personnel | Single Reactor Annual Expense ($) | Twin Reactors No. of Personnel | Twin Reactors Annual Expense ($) |
|---|---|---|---|---|---|
| Maintenance Staff | | | | | |
| Maintenance Supervisor | 10,800 | 1 | 10,800 | 1 | 10,800 |
| Maintenance Foreman | 8,000 | 2 | 16,000 | 2 | 16,000 |
| Instrument Mechanic - A | 6,800 | 3 | 20,400 | 4 | 27,200 |
| Instrument Mechanic - B | 6,000 | 3 | 18,000 | 4 | 24,000 |
| Electrician - A | 6,800 | 2 | 13,600 | 3 | 20,400 |
| Mechanic - A | 7,000 | 4 | 28,000 | 4 | 28,000 |
| Mechanic - B | 6,000 | 2 | 12,000 | 4 | 24,000 |
| Helper | 5,000 | 2 | 10,000 | 4 | 20,000 |
| Subtotal | | 19 | 128,800 | 26 | 170,400 |
| Total Direct Payroll (rounded) | | 78 | 570,000 | 95 | 680,000 |
| Fringe Benefits at 20% | | | 114,000 | | 136,000 |
| Total Payroll | | | 684,000 | | 816,000 |
| Consumable Materials and Services | | | 764,000 | | 985,000 |
| Grand Total | | | 1,448,000 | | 1,801,000 |
| Mills/kwh at 80% capacity factor | | | 0.207 | | 0.257 |

TABLE IV-3

CONSUMABLE MATERIALS AND SERVICES

($/yr, in thousands)

| | Single Reactor | Twin Reactors |
|---|---|---|
| Operating supplies | 250 | 300 |
| Chemicals and resins | 90 | 108 |
| Waste disposal | 50 | 60 |
| Building heat | 80 | 88 |
| Communications | 24 | 24 |
| Maintenance supplies | 170 | 255 |
| Contract maintenance | 100 | 150 |
| Total | 764 | 985 |

## 3. COST OF CONSTRUCTION

Table IV-4 presents details of the direct construction costs of Reference Plant A. The price differential of the equivalent twin-reactor plant (Arrangement C) is compared to Plant A in Table IV-5, which shows details of Account No. 22.

TABLE IV-4

ESTIMATE OF DIRECT CONSTRUCTION COSTS

REFERENCE PLANT ARRANGEMENT A

| Account No. | Description | Cost (Labor and Equip.) (Thousands of $) | |
|---|---|---|---|
| 20 | Land and Land Rights | | $ 360 |
| 21 | Structures and Improvements | | |
| 211 | Ground Improvements | $ 915 | |
| 212 | Buildings | | |
| 01-212 | Auxiliary Buildings | $ 454 | |
| 03-212 | Turbine Bay and Auxiliary Bay | | |
| .1 | Excavation and Backfill | $ 118 | |
| .2 | Bearing Piles and Caissons | -0- | |
| .3 | Substructure Concrete | 206 | |
| .4 | Superstructure | 2 303 | |
| .5 | Stacks (Supported by Building) | -0- | |
| .6 | Building Services | 378 | |
| .7 | Painting | 66 | |
| .8 | Non-structural Liners | -0- | |
| | Subtotal | $ 3 071 | |
| 04-212 | Bay Control Service and Office Bldg. | $ 450 | |
| 06-212 | Screen Well Structure | | |
| .1 | Excavation and Backfill | $ 87 | |
| .2 | Bearing Piles and Caissons | -0- | |
| .3 | Substructure Concrete | 348 | |
| .4 | Superstructure | 18 | |
| .5 | Stacks (Supported by Building) | -0- | |
| .6 | Building Services | 6 | |
| .7 | Painting | 1 | |
| .8 | Non-structural Liners | -0- | |
| | Subtotal | $ 460 | |

TABLE IV-4 (Continued)

| Account No. | Description | Cost (Labor & Equip.) | |
|---|---|---|---|
| 09-212 | Spent Fuel Handling Structure | -0- | |
| 11-212 | New Fuel Storage & Decontamination | -0- | |
| 12-212 | Warehouse (Permanent) | $ 360 | |
| 15-212 | Well Pump House | $ 7 | |
| 18-212 | Gate House | $ 2 | |
| 26-212 | Chlorine Storage | $ -0- | |
| 29-212 | Yard Area Craneway | $ -0- | |
| 218 | Stacks | $ 85 | |
| 219 | Reactor Containment Structure | | |
| .1 | Excavation and Backfill | $ 235 | |
| .2 | Bearing Piles and Caissons | -0- | |
| .3 | Substructure Concrete | 355 | |
| .4 | Superstructure | 2 908 | |
| .5 | Stacks (Supported by Building) | -0- | |
| .6 | Building Services | 331 | |
| .7 | Painting | 94 | |
| .8 | Liners | 430 | |
| | Subtotal | $ 4 353 | |
| | Total Cost, Account 21, Structures and Improvements | | $10 157 |
| 22 | Reactor Plant Equipment | | |
| 221 | Reactor Equipment | $32 434 | |
| 222 | Heat Transfer System | 8 248 | |
| 223 | Nuclear Fuel Handling and Storage Equipment | 1 245 | |
| 225 | Radioactive Waste Treatment and Disposal | 896 | |
| 226 | Instrumentation and Control | 5 088 | |
| 227 | Feedwater Supply and Treatment | | |
| .1 | Raw Water Supply System | 52 | |
| .2 | Makeup Water Supply | 278 | |
| .3 | Steam Generator Feedwater Purification | w/222.4 | |
| .4 | Feedwater Heaters | 1 358 | |
| .5 | Feedwater Pumps and Drives | 275 | |
| .7 | Misc. Tanks | w/system | |
| .8 | Equipment Foundations | w/equip. | |
| | Total Cost, Item 227 | | $ 1 963 |

TABLE IV-4 (Continued)

| Account No. | Description | Cost (Labor & Equip.) | |
|---|---|---|---|
| 228 | Steam Condensate & Feedwater Piping | $ 3 049 | |
| 229 | Other Reactor Plant Equipment | 708 | |
| | Total Cost, Account 22, Reactor Plant Equipment | | $53 631 |
| 23 | Turbine-Generator Units | | |
| 231 | Turbine-Generators | | |
| .1 | Foundation | $ 538 | |
| .2 | Turbine-Generators | 24 600 | |
| | Installation | 681 | |
| | Total Cost, Item 231 | $25 819 | |
| 232 | Circulating Water System | $ 1 219 | |
| 233 | Condensers | $ 2 249 | |
| 234 | Central Lubricating System | w/ T-G | |
| 235 | Turbine Plant Boards, Instruments and Controls | w/ T-G | |
| 236 | Turbine Plant Piping | w/ Acct. 228 | |
| 237 | Auxiliary Equipment for Generators | $ 60 | |
| 238 | Other Turbine Plant Equipment | 17 | |
| | Total Cost, Acct. 23, T-G Units | | $29 364 |
| 24 | Accessory Electrical Equipment | | 3 321 |
| 25 | Misc. Power Plant Equipment | | 585 |
| | Total Direct Cost (Construction) | | $97 058 |

TABLE IV-5

COMPARATIVE COSTS OF NON-REHEAT PLANTS

Arrangement A – Single Reactor (Reference Plant)

Arrangement C – Twin Reactor

Price Differentials Arrangement A vs. Arrangement C

| Account No. | Description | Thousands of Dollars | $/KW |
|---|---|---|---|
| 221.1 | Reactor Vessel | -1128 | -1.128 |
| 221.2 | Reactor Controls | - 158 | -0.158 |
| 221.3 | Reactor Shield Cooling Piping | + 78 | +0.078 |
| 221.4 | Reactor Auxil. Heating and Cooling Sys. | +1359 | +1.359 |
| 221.7 | Reactor Plant Cranes and Hoists | (included with 223) | |
| 222.4 | Coolant Sampling and Purification Sys. | + 239 | +0.239 |
| 223 | Nuclear Fuel Handling and Storage Eqpt. | + 469 | +0.469 |
| 225 | Radioactive Waste Treatment and Disposal | 0 | 0 |
| 226 | Nuclear Instrumentation and Control | + 701 | +0.701 |
| 227 | Feedwater Supply and Treatment | 0 | 0 |
| 228 | Condensate and Feedwater Piping | + 326 | +0.326 |
| 229 | Other Reactor Plant Equipment | 0 | 0 |

(-) Minus sign indicates that price for Arrangement C is lower than that for Reference Plant Arrangement A.

(+) Plus sign indicates the opposite.

* The vessel price differential represents the vessel fabricators' cost estimates based on analysis of the specifications. The large vessel, although of similar weight to two smaller ones, does require newly developed handling equipment and techniques. Therefore, although 1000 MW vessel fabrication is entirely feasible at the present time, greater margins can be justified in estimates of the cost of labor and handling equipment. When a few large vessels have been built, the cost differential per unit weight may be expected to diminish.

# APPENDIX V

## EFFECTS OF FORCED OUTAGE RATE AND PLANT ARRANGEMENT ON CAPITAL REQUIREMENTS

The number of major components used, and the reliability of each, has an effect on the total power requirements of a system. Considering a system expansion utilizing initially either single-reactor units (like Arrangements A or B) or twin-reactor units (like Arrangements C or D), probability calculations were made to determine the required expansion pattern for a hypothetical system. Based on this, calculations were made of the cumulative present worth of investment for 19 years of expansion. Following are the assumptions which were utilized:

1) The original system (in 1967) consists of 37 units with a total capacity of 8345 MWe. Units of original system are assumed to have a forced outage rate of 0.02 throughout the expansion.

2) Original system load is 7562 MWe, and grows at a rate of 7 percent per year.

3) The system is expanded for 19 years, with the first 1000 MWe plant added in 1968.

4) Units added are between 9 and 12 percent of system size, and are rated at 1000, 1350, 1900, 2500 or 3300 MWe.

5) All units - new and old - are maintained four weeks per year, and new units maintained the first year of operation.

6) The acceptable loss-of-load probability is 0.15.

7) All units added in a given year have the same price, irrespective of unit arrangement.

8) The price of each 1000 MWe unit added in 1968 is $151,615,000. (This is the price of the Reference Plant as determined in the AEC Study.)

9) The price of a larger unit, relative to a smaller one is expressed by the ratio $P_1 / P_o = (MW_1 / MW_o)^{0.637}$

10) Prices decrease at the rate of 10 percent during the first 10 years and 5 percent during the second 10 years.

Six cases are presented, based on three types of units and two assumptions regarding forced outage rate (F.O.R.). The unit arrangements are designated as follows:

"C-D" Units -- twin reactor twin T-G, similar to Plant Arrangements C and D, except not cross-tied, i.e., behaves as two independent 500 MW (or larger) units, installed in the same year.

"A-B" Units -- single reactor twin T-G, similar to Plant Arrangements A and B.

"E" Units -- single reactor single T-G.

The six cases differed from each other only in plant arrangement and forced-outage rates, as follows:

Case 1: "C-D" units used for all additions from 1968 to 1986. F.O.R. of 0.02, assigned 0.01 to each reactor and 0.01 to each T-G.

Case 2: "A-B" units added in 1968-71, and C-D added in 1972-86. F.O.R. same as Case 1.

Case 3: "E" units added in 1968-71, and C-D added in 1972-86. F.O.R. same as Case 1.

Case 4: Unit arrangements added same as Case 1. F.O.R. of units added 1968-86 is 0.03, assigned 0.015 to each reactor and 0.015 to each T-G.

Case 5: Unit arrangements added same as Case 2. F.O.R. same as Case 4.

Case 6: Unit arrangements added same as Case 3. F.O.R. same as Case 4.

Table 5-1 presents the expansion patterns for the six cases. The cumulative present worth of the 19 year expansion is shown in Table 5-2. As discussed in the "assumptions", all plant arrangements were assumed to have the same price, in order to point up the differences caused by forced outage rate and by various plant arrangements.

TABLE V-1

EXPANSION PATTERNS

(Original System in 1967: 8345 MW installed)

| CASE | 1 | 2 | 3 | 4 | 5 | 6 |
|---|---|---|---|---|---|---|
| Unit Arrgt. (1968-71 Add) | C-D | A-B | E | C-D | A-B | E |
| F.O.R. of Units Added 1968-86 | 0.02 | 0.02 | 0.02 | 0.03 | 0.03 | 0.03 |
| YEAR | | | UNIT ADDED (MW) | | | |
| 1968 | 1000 | 1000 | 1000 | 1000 | 1000 | 1000 |
| 1969 | -- | 1000 | 1000 | -- | 1000 | 1000 |
| 1970 | 1000 | -- | 1000 | 1000 | -- | 1000 |
| 1971 | 1000 | 1000 | -- | 1000 | 1000 | -- |
| 1972 | -- | 1350 | 1350 | 1350 | 1350 | 1350 |
| 1973 | 1350 | -- | -- | -- | -- | 1350 |
| 1974 | 1350 | 1350 | 1350 | 1350 | 1350 | -- |
| 1975 | -- | -- | 1350 | -- | 1350 | 1350 |
| 1976 | 1350 | 1350 | -- | 1350 | -- | -- |
| 1977 | 1900 | 1900 | 1900 | 1900 | 1900 | 1900 |
| 1978 | -- | -- | -- | -- | 1900 | 1900 |
| 1979 | 1900 | 1900 | 1900 | 1900 | -- | -- |
| 1980 | 1900 | 1900 | 1900 | 1900 | 1900 | 1900 |
| 1981 | -- | -- | -- | -- | 2500 | 2500 |
| 1982 | 2500 | 2500 | 2500 | 2500 | -- | -- |
| 1983 | -- | 2500 | 2500 | 2500 | 2500 | 2500 |
| 1984 | 2500 | -- | -- | -- | -- | -- |
| 1985 | 2500 | 2500 | 2500 | 2500 | 2500 | 2500 |
| 1986 | 3300 | 3300 | 3300 | 3300 | 3300 | 3300 |

TABLE V-2

CUMULATIVE PRESENT WORTH OF 19-YEAR EXPANSION

| CASE | Cumulative Present Worth Costs of 19-year Expansion (Millions of $) | Difference of Cases With F.O.R. of 0.02 (Compared to Case 1) (Millions of $) | Difference of Cases With F.O.R. of 0.03 (Compared to Case 4) (Millions of $) |
|---|---|---|---|
| 1 | 1,190 | 0 | |
| 2 | 1,232 | +42 | |
| 3 | 1,258 | +68 | |
| 4 | 1,216 | | 0 |
| 5 | 1,270 | | +54 |
| 6 | 1,300 | | +84 |

GPO 812-316-1

# APPENDIX VI

The following is a listing of reference General Electric Company and Ebasco Services drawings for the plants and systems covered in Volumes I and II and in the first four appendices of this document.

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.

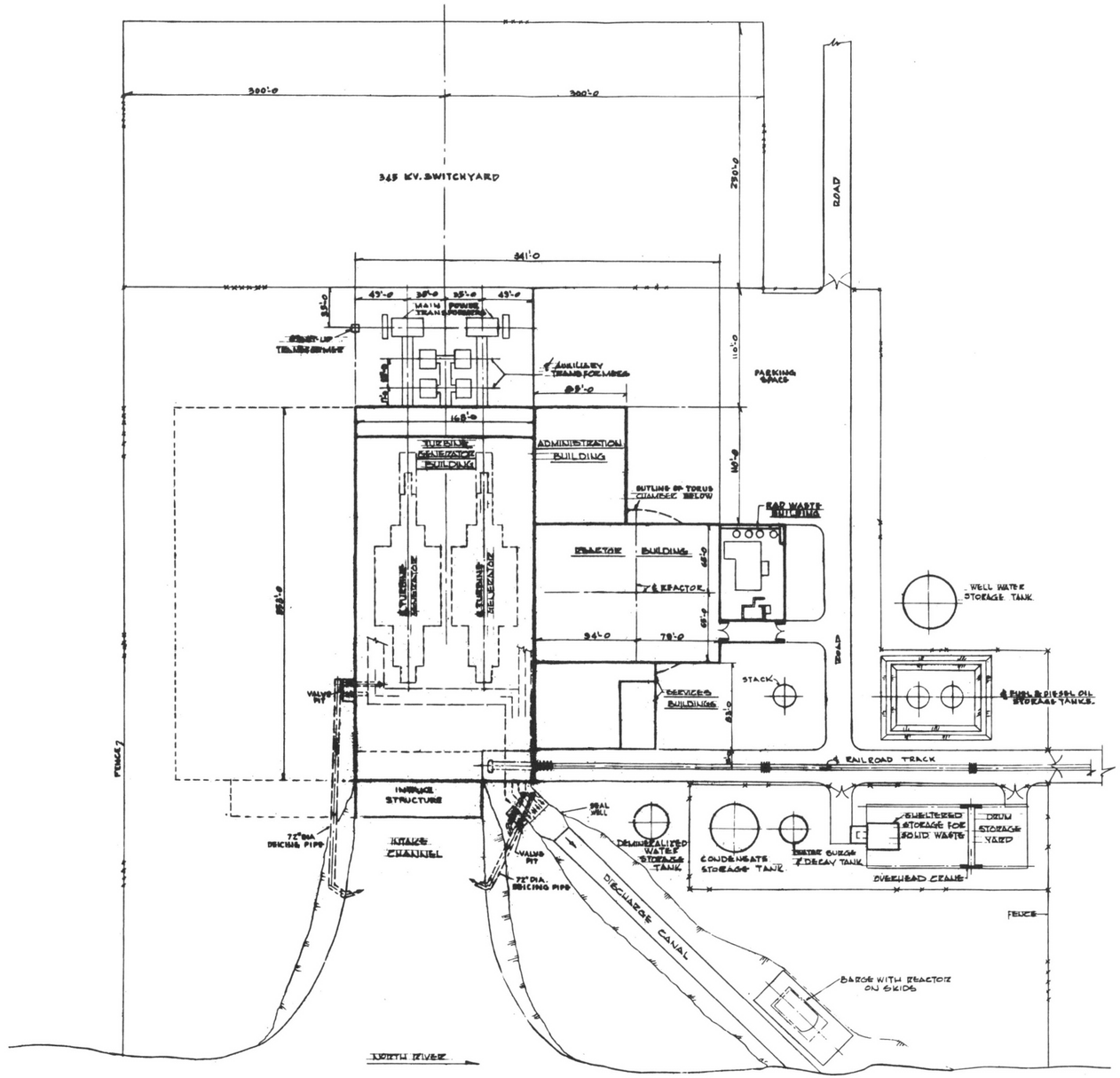
## Drawing List

| Figure No. | Drawing No. | Title |
|---|---|---|
| 1 | 3250-M-1 | Plot Plan, Arrangement A |
| 2 | 3250-M-21 | Plot Plan, Arrangement B |
| 3 | 3250-M-42 | Plot Plan, Arrangement C |
| 4 | 3250-M-3 | General Arrangement Operating Floor Plan, Arrangement A |
| 5 | 3250-M-4 | General Arrangement - Mezz Floor Plan, Arrangement A |
| 6 | 3250-M-5 | General Arrangement - Ground Floor Plan, Arrangement A |
| 7 | 3250-M-6 | General Arrangement - Sections A-A, Arrangement A |
| 8 | 3250-M-7 | General Arrangement - Sections B-B, Arrangement A |
| 9 | 3250-M-23 | General Arrangement Operating Floor Plan, Arrangement B |
| 10 | 3250-M-24 | General Arrangement - Mezz Floor Plan, Arrangement B |
| 11 | 3250-M-25 | General Arrangement - Ground Floor Plan, Arrangement B |
| 12 | 3250-M-26 | General Arrangement - Sections A-A, Arrangement B |
| 13 | 3250-M-27 | General Arrangement - Sections B-B, Arrangement B |
| 14 | 3250-M-42 | General Arrangement - Ground Floor Plan, Arrangement C |
| 15 | 148F210 (8 shts) | Reactor Building, Arrangement A, B |
| 16 | 237E294 (6 shts) | Reactor Building, Arrangement C, D |
| 17 | 3250-S-1 | Intake Structure |
| 18 | 148F230 | Radwaste Building Arrangement |
| 19 | 3250-M-8 | Heat Balance, Arrangement A |
| 20 | 3250-M-28 | Heat Balance, Arrangement B |

Drawing List (Continued)

| Figure No. | Drawing No. | Title |
|---|---|---|
| 21 | 237E285 | Plant Flow Diagram, Arrangement A |
| 22 | 237E288 | Plant Flow Diagram, Arrangement B |
| 23 | 237E287 | Plant Flow Diagram, Arrangement C |
| 24 | 237E289 | Plant Flow Diagram, Arrangement D |
| 25 | 237E283 | P and I Diagrams - NSSS, Arrangement A,B |
| 26 | 237E284 | P and I Diagrams - NSSS, Arrangement C,D |
| 27 | 3250-M-11 | Flow Diagram Make-up System and Water Storage |
| 28 | 3250-M-12 | Flow Diagram Service and Cooling Water System |
| 29 | 107C4488 | NSSS Cooling Water Diagram, Arrangement A |
| 30 | 107C4619 | NSSS Cooling Water Diagram, Arrangement C |
| 31 | 148F226 | P and ID Clean-up Demineralizer, Arrangement A |
| 32 | 148F227 | P and ID Clean-up Demineralizer, Arrangement C |
| 33 | 798D628 | Condensate Demineralizer, Arrangement A, C |
| 34 | 3250-M-13 | Flow Diagram Turbine Steam and Extraction System, Arrangement A |
| 35 | 3250-M-33 | Flow Diagram Turbine Steam and Extraction System, Arrangement B |
| 36 | 3250-M-14 | Flow Diagram Condensate and Feedwater System, Arrangement A |
| 37 | 3250-M-34 | Flow Diagram Condensate and Feedwater System, Arrangement B |
| 38 | 3250-M-16 | Flow Diagram Heater Drips, Drains and Vents, Arrangement A |
| 39 | 3250-M-36 | Flow Diagram Heater Drips, Drains and Vents, Arrangement B |
| 40 | 237E298 | Radwaste Disposal System |
| 41 | 107C4630 | Process Off-Gas System, Arrangement A, C |
| 42 | 3250-H and V-1 | Flow Diagram Heating, Vent. and Air Condition Systems |
| 43 | 237E136 | Control Room Panel Arrangement, Arrangement A, C |
| 44 | 762D755 | Reactor Protection System |
| 45 | 107C4627 | Power Supply-Control and Protection System |
| 46 | 762D750 | Neutron Monitor System |
| 47 | 798D862 | In-Core Flux Monitoring System |
| 48 | 798D863 | Process Radiation Monitor System |
| 49 | 107C628 | Area Radiation Monitor System |
| 50 | 3250-E-1 | Electrical One Line Diagram |
| 51 | 943A863 | Instrument Symbols |

N




**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

GPO 812–316–5

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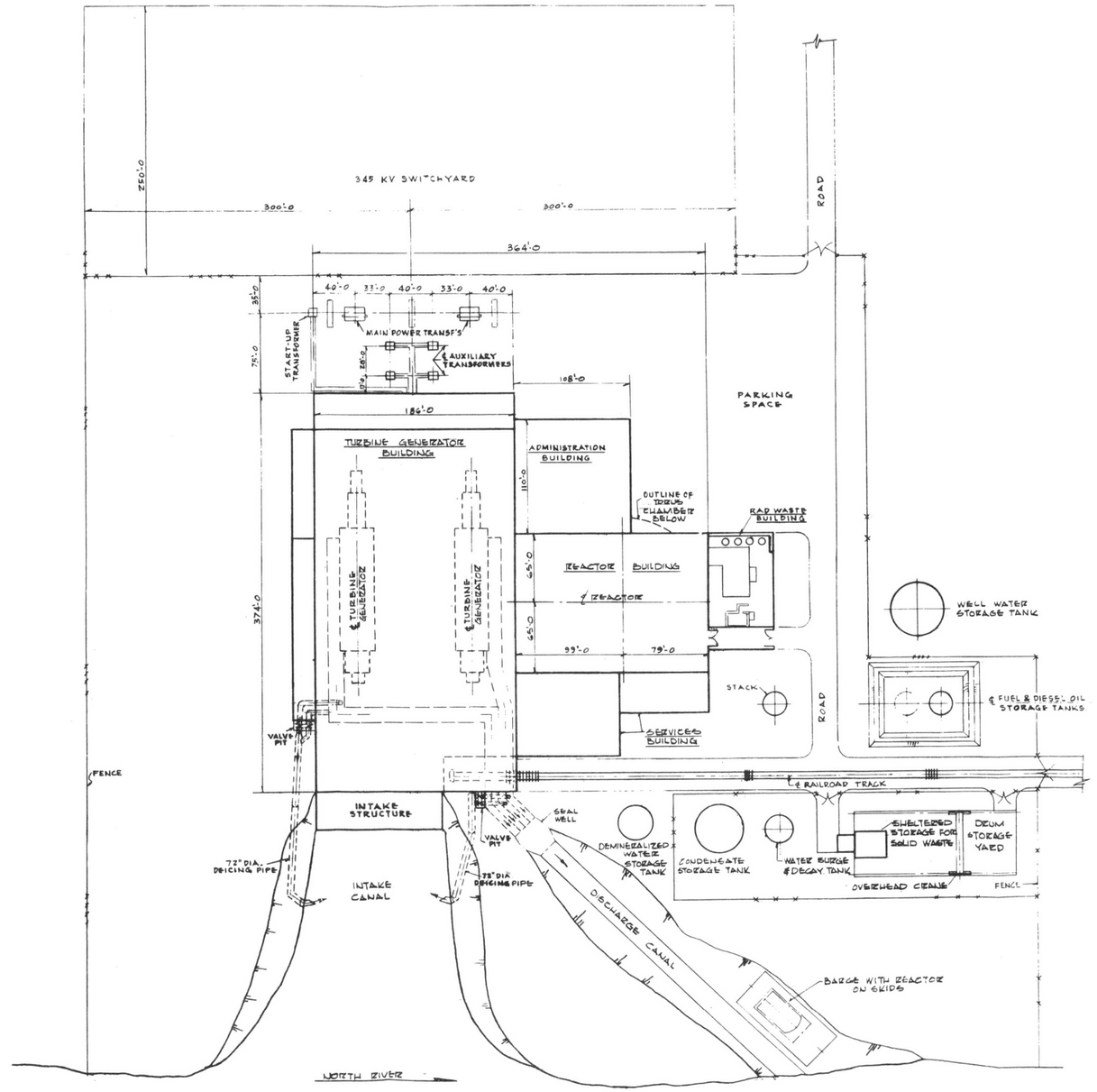



**PLOT PLAN – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 3250–M–21

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

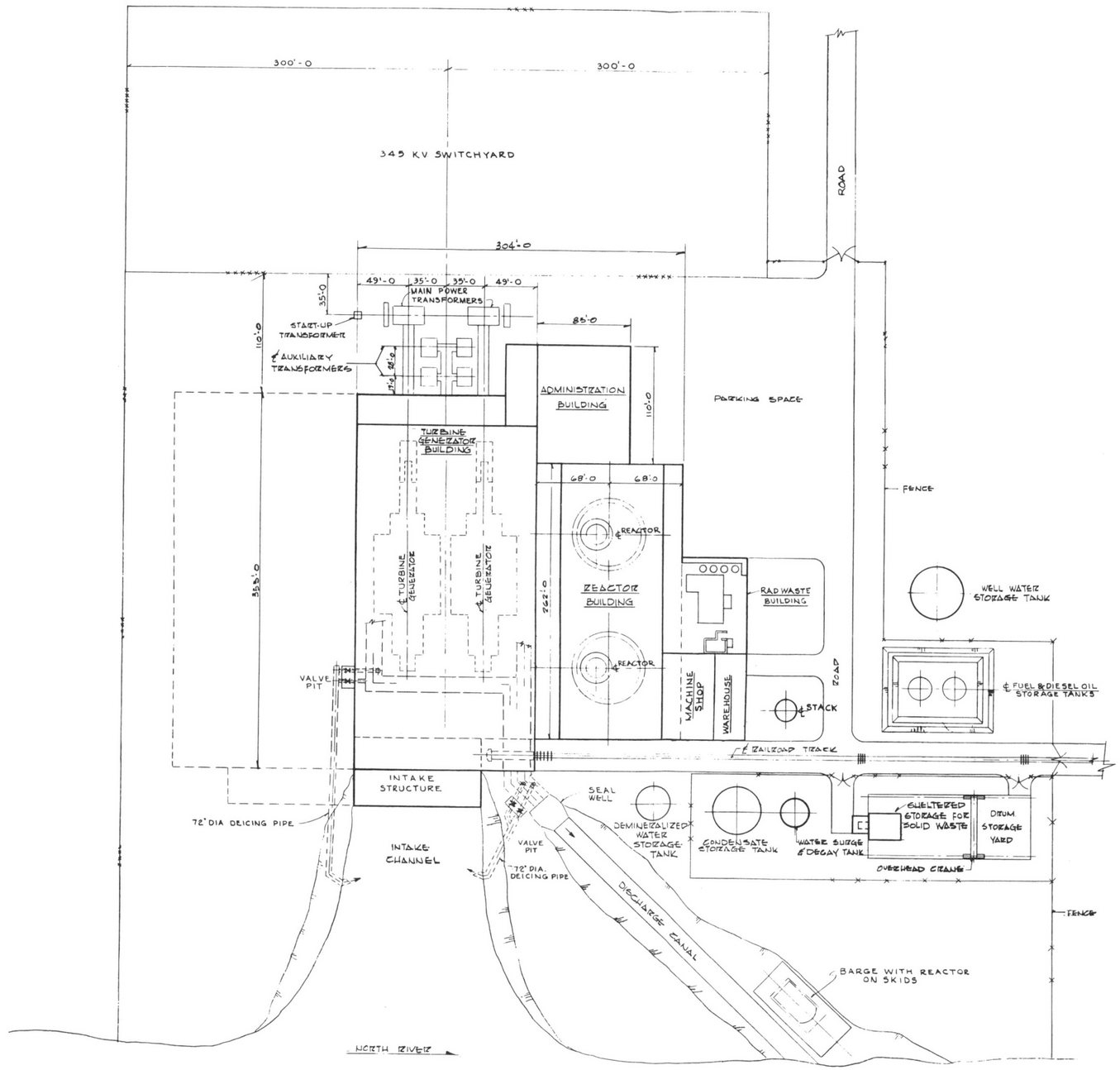
**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

◆ ◆ ◆

**EBASCO SERVICES INCORPORATED**

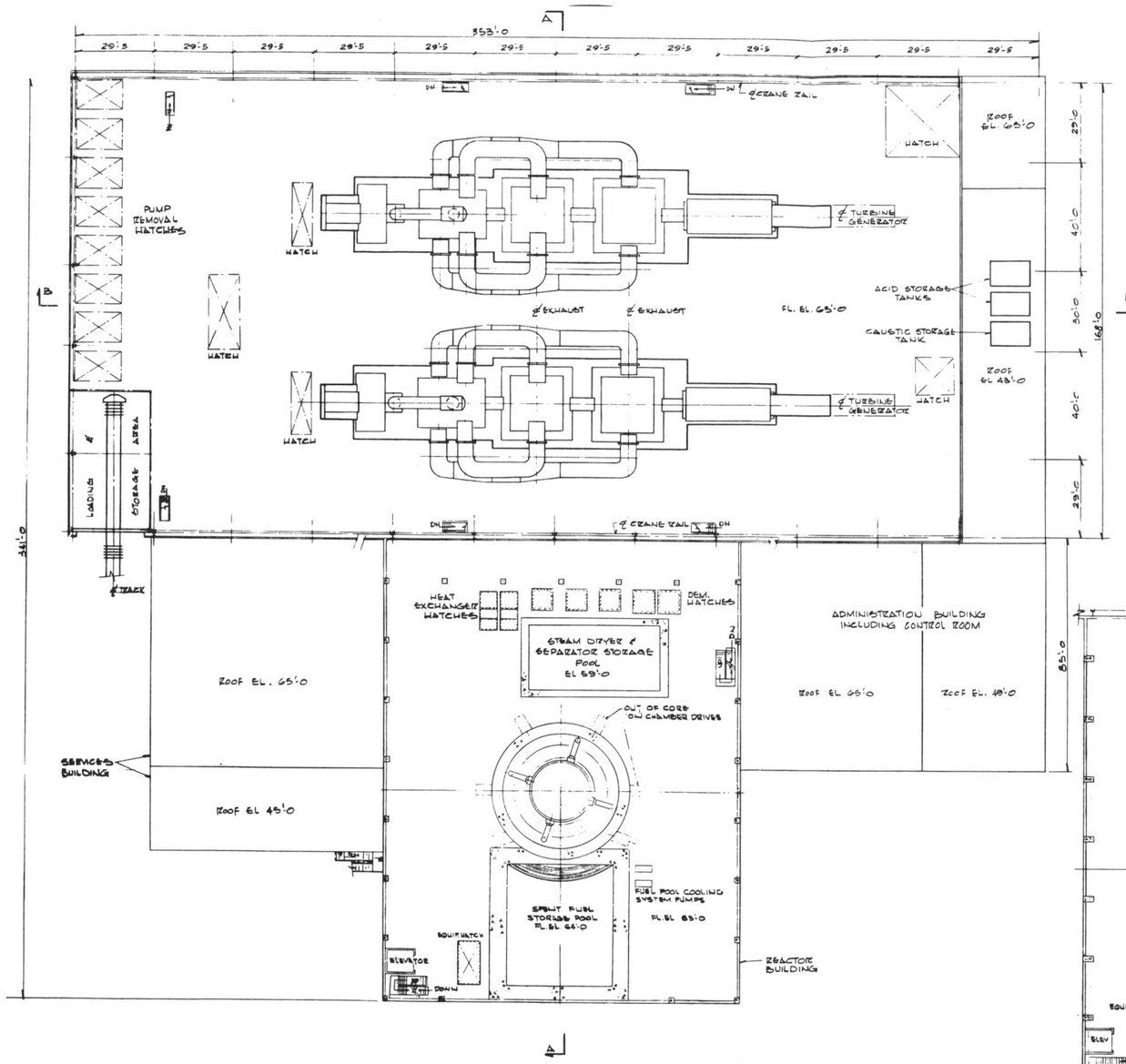
ENGINEERS AND CONTRACTORS



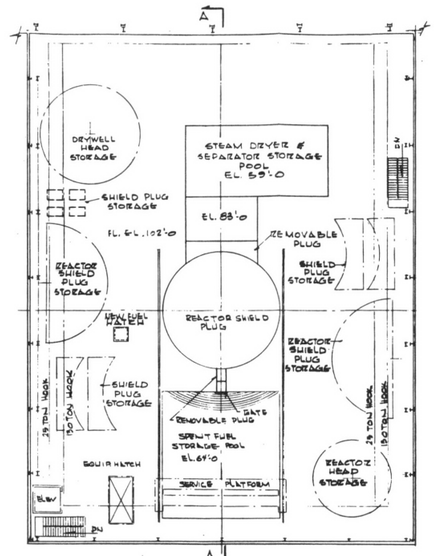

NOTE:
PLOT PLAN FOR ARRANGEMENT "D" IDENTICAL TO ARRANGEMENT "C" EXCEPT FOR DIFFERENCES IN TURBINE BUILDING DIMENSIONS.

**PLOT PLAN – ARRANGEMENT C**

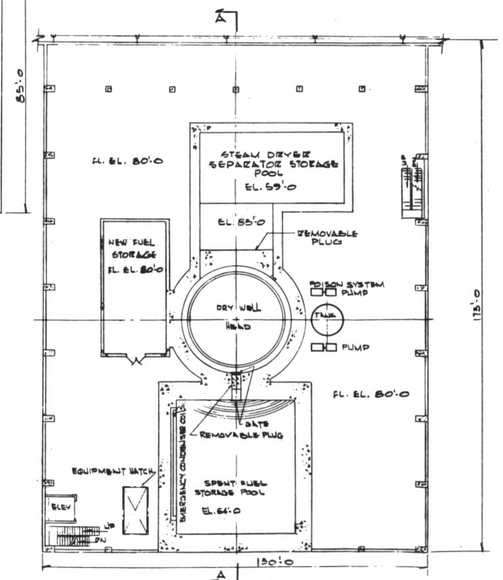
SIMILAR TO DRAWING NO. 3250–M–41
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
**UNITED STATES ATOMIC ENERGY COMMISSION**
◈ ◈ ◈
**EBASCO SERVICES INCORPORATED**
ENGINEERS AND CONTRACTORS




TURBINE BUILDING - OPERATING FLOOR PLAN & ABOVE




PLAN FL. EL. 102




PLAN FL. EL. 80'-0

CONCEPTUAL DESIGN DRAWING DO NOT TO BE USED FOR CONSTRUCT

4

**GENERAL ARRANGEMENT OPERATING FLOOR PLAN – ARRANGEMENT A**

SIMILAR TO DRAWING NO. 3250–M–3

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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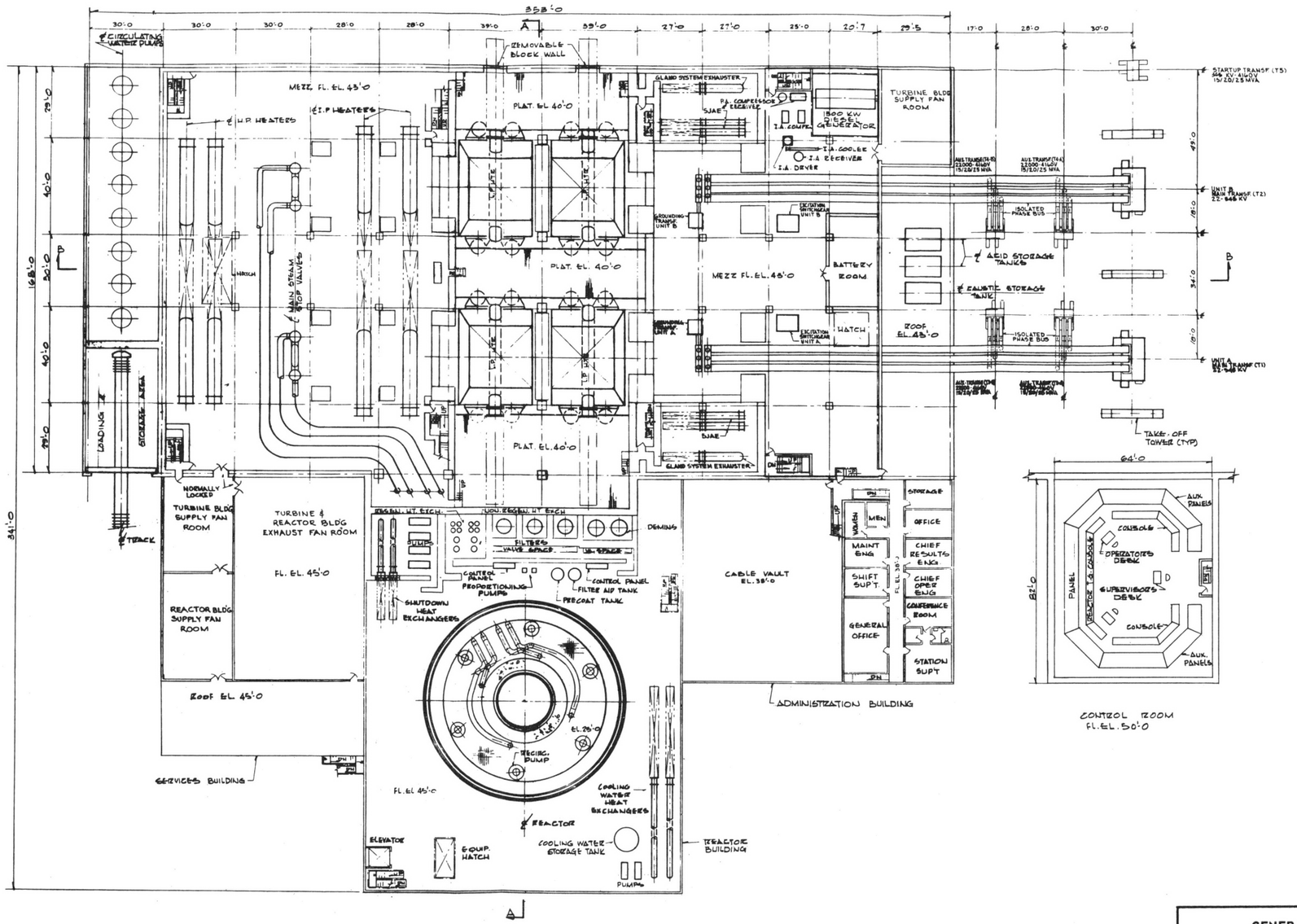


TURBINE BUILDING - MEZZANINE FLOOR PLAN

**GENERAL ARRANGEMENT**
**MEZZ. FLOOR PLAN – ARRANGEMENT A**
SIMILAR TO DRAWING NO. 3250–M–4
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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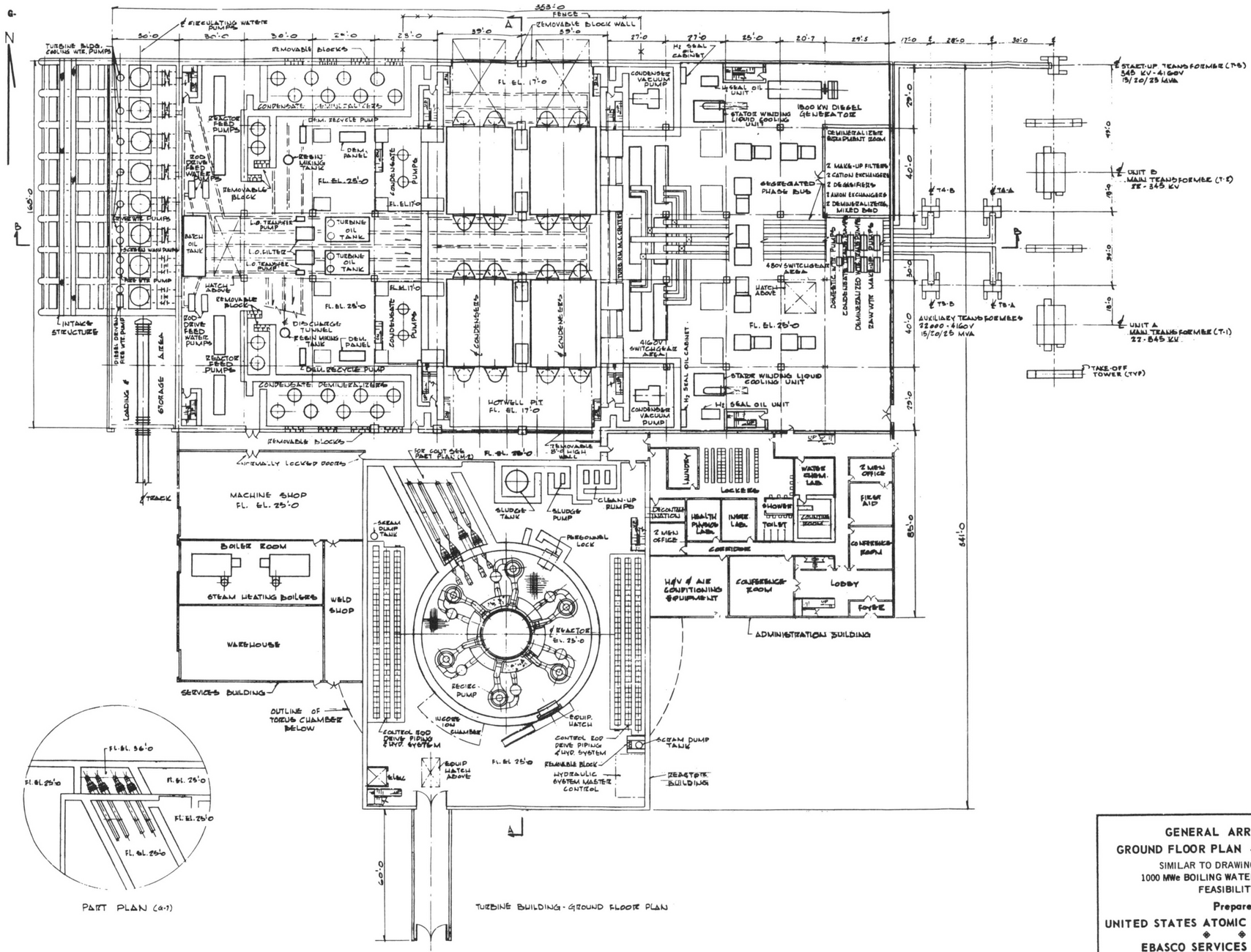


**GENERAL ARRANGEMENT**
**GROUND FLOOR PLAN – ARRANGEMENT A**
SIMILAR TO DRAWING NO. 3250–M–5
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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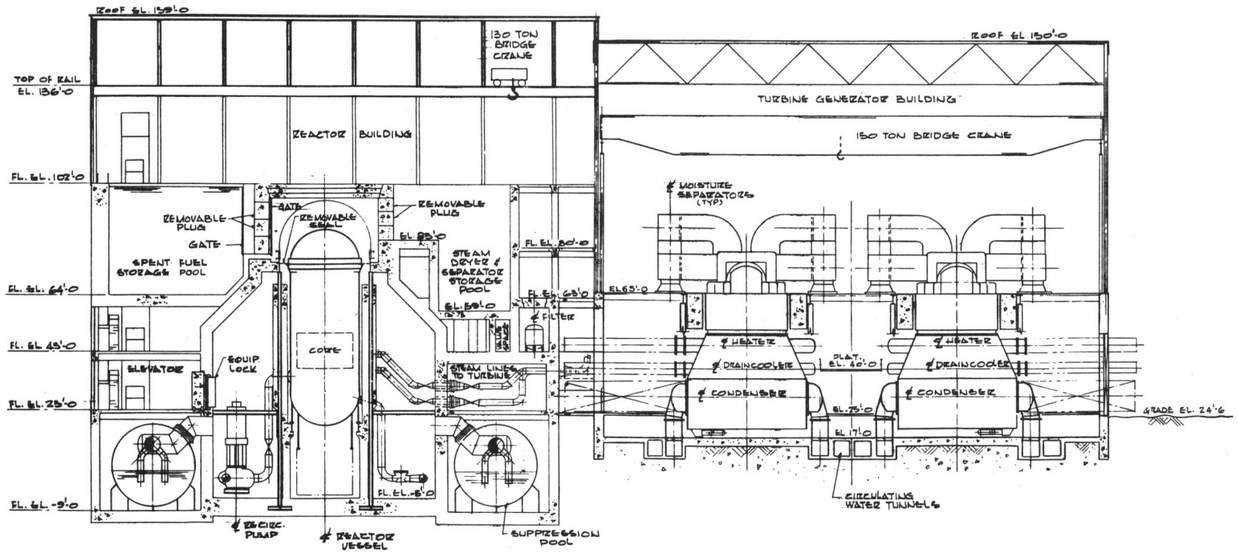


SECTION A-A

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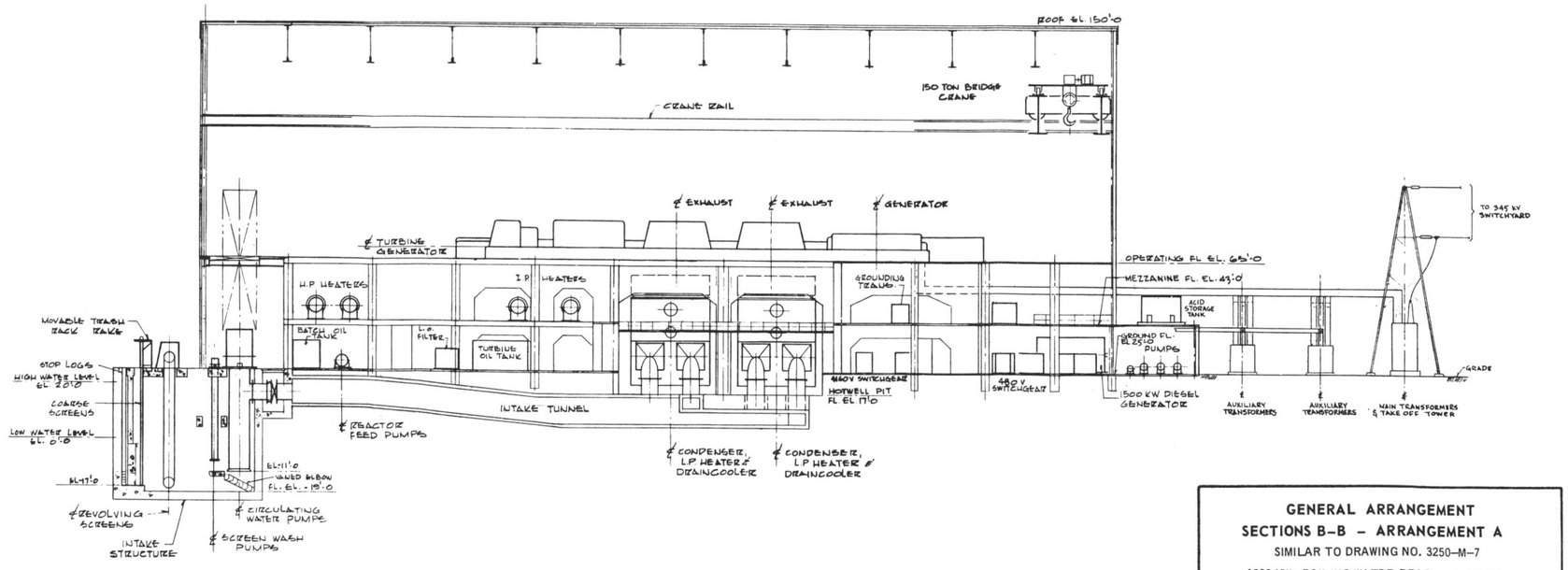


SECTION B-B

**GENERAL ARRANGEMENT**
**SECTIONS B–B – ARRANGEMENT A**
SIMILAR TO DRAWING NO. 3250–M–7
**1000 MWe BOILING WATER REACTOR PLANT**
**FEASIBILITY STUDY**
**Prepared For**
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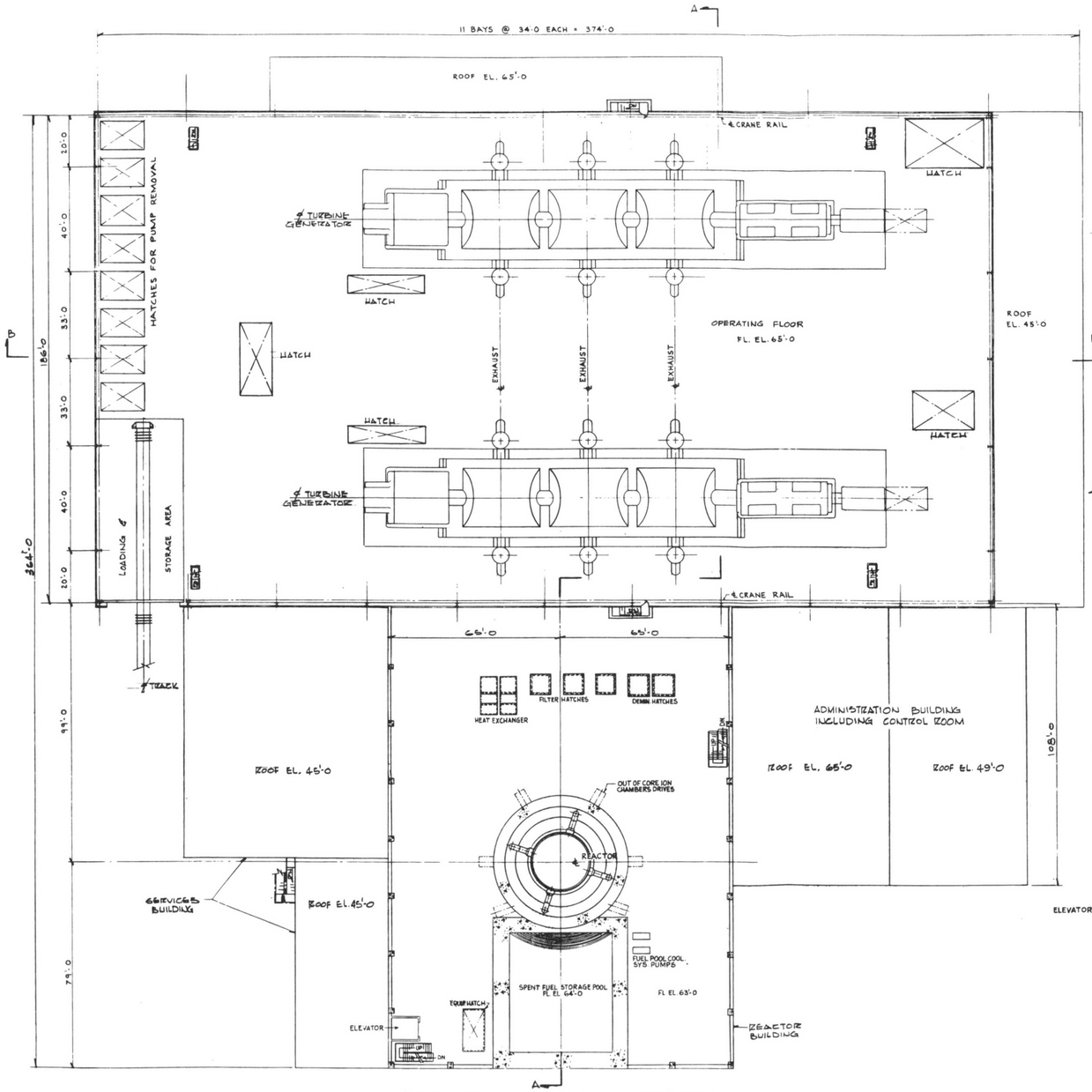
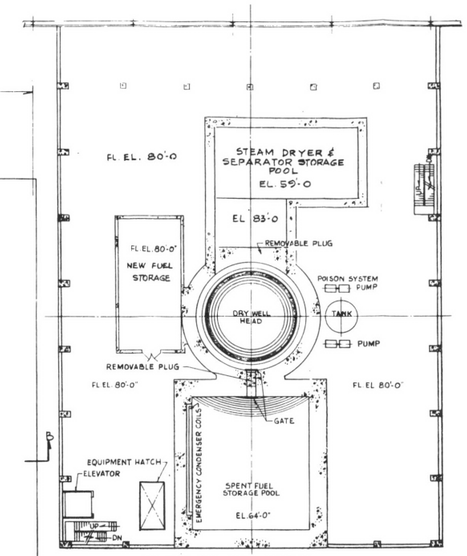
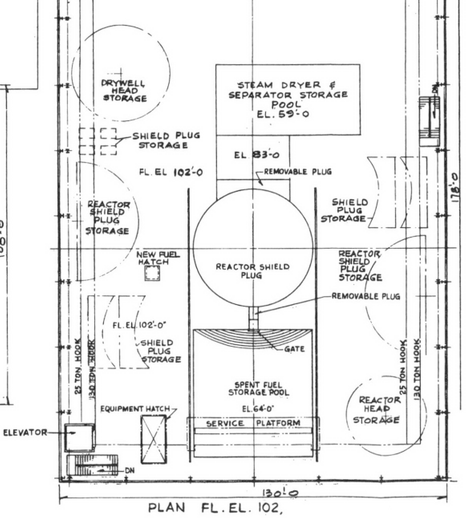



TURBINE BUILDING - OPERATING FLOOR PLAN & ABOVE




PLAN FL. EL. 80'-0




PLAN FL. EL. 102'

**GENERAL ARRANGEMENT OPERATING FLOOR PLAN – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 3250–M–23

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

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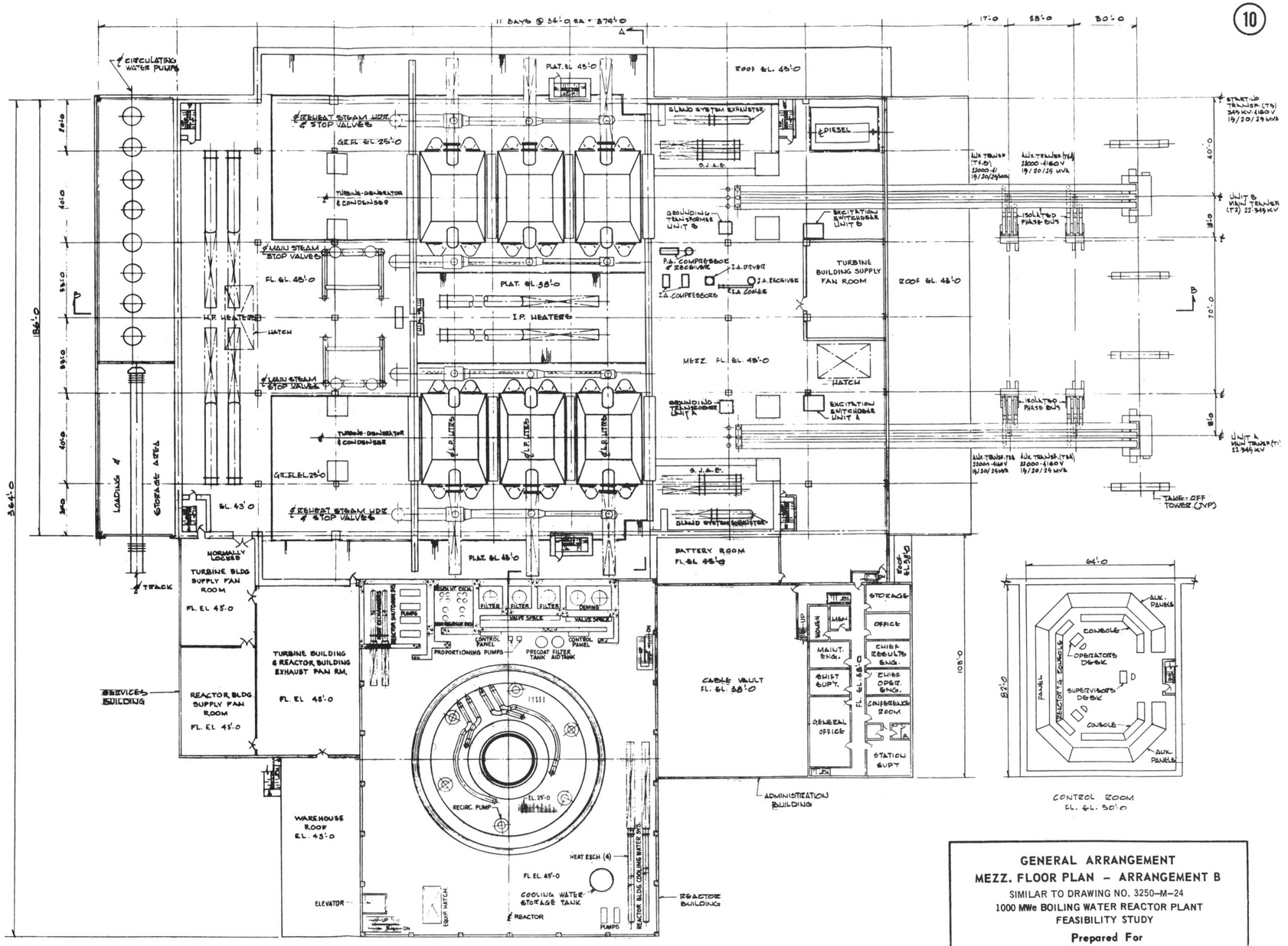


TURBINE BUILDING-MEZZANINE FLOOR PLAN

CONTROL ROOM
FL. EL. 50'-0

**GENERAL ARRANGEMENT**
**MEZZ. FLOOR PLAN – ARRANGEMENT B**
SIMILAR TO DRAWING NO. 3250–M–24
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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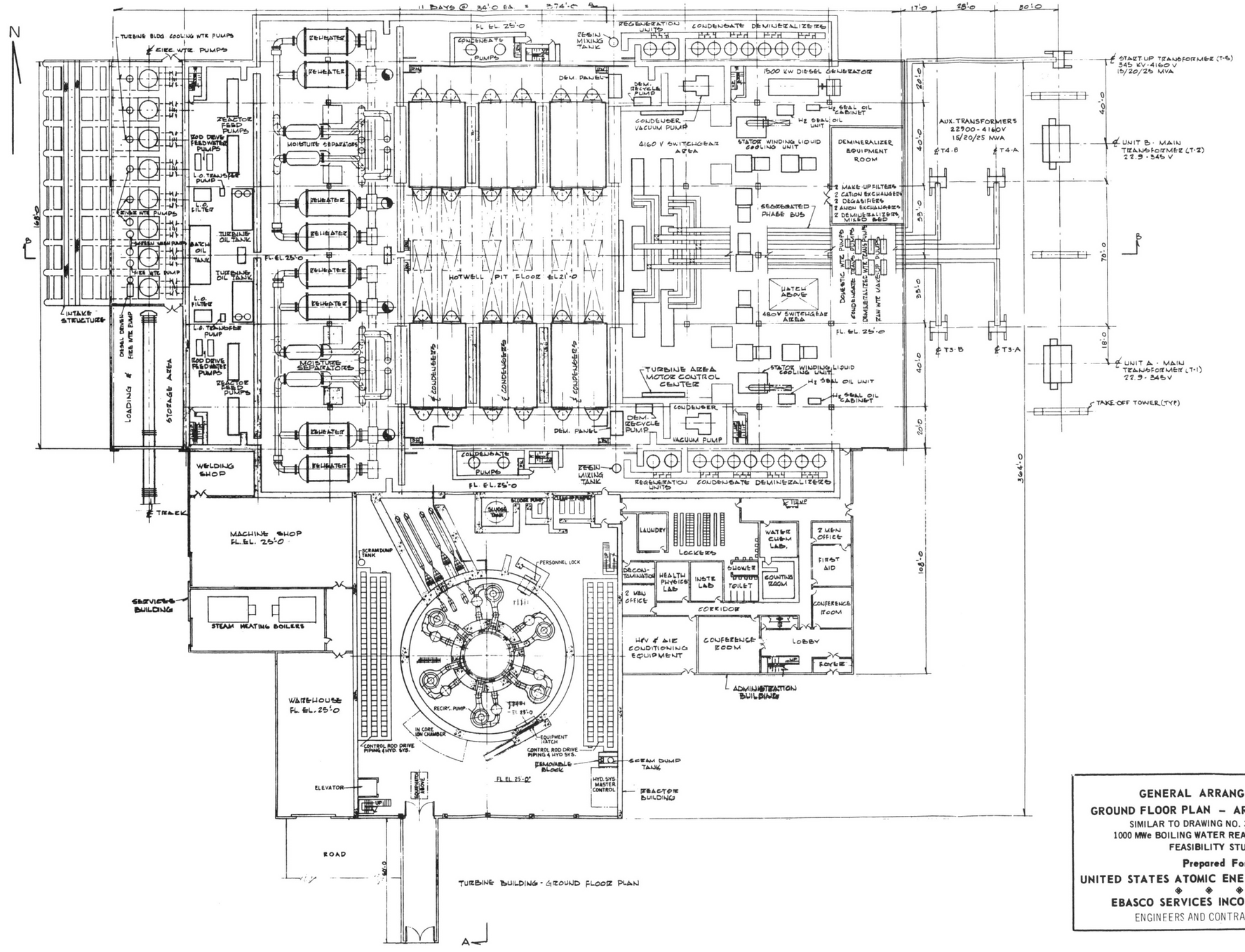


TURBINE BUILDING - GROUND FLOOR PLAN

**GENERAL ARRANGEMENT**
**GROUND FLOOR PLAN – ARRANGEMENT B**
SIMILAR TO DRAWING NO. 3250–M–25
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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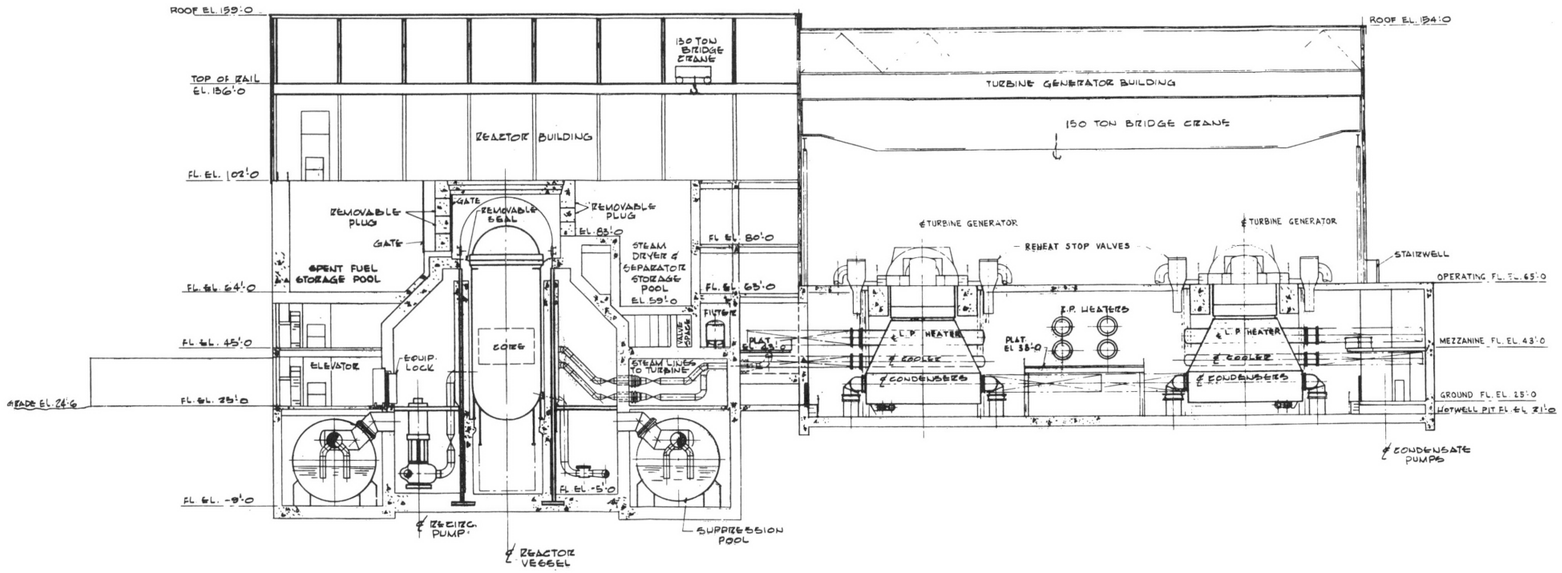


SECTION A-A

**GENERAL ARRANGEMENT**
**SECTIONS A–A – ARRANGEMENT B**
SIMILA R TO DRAWING NO. 3250–M–26
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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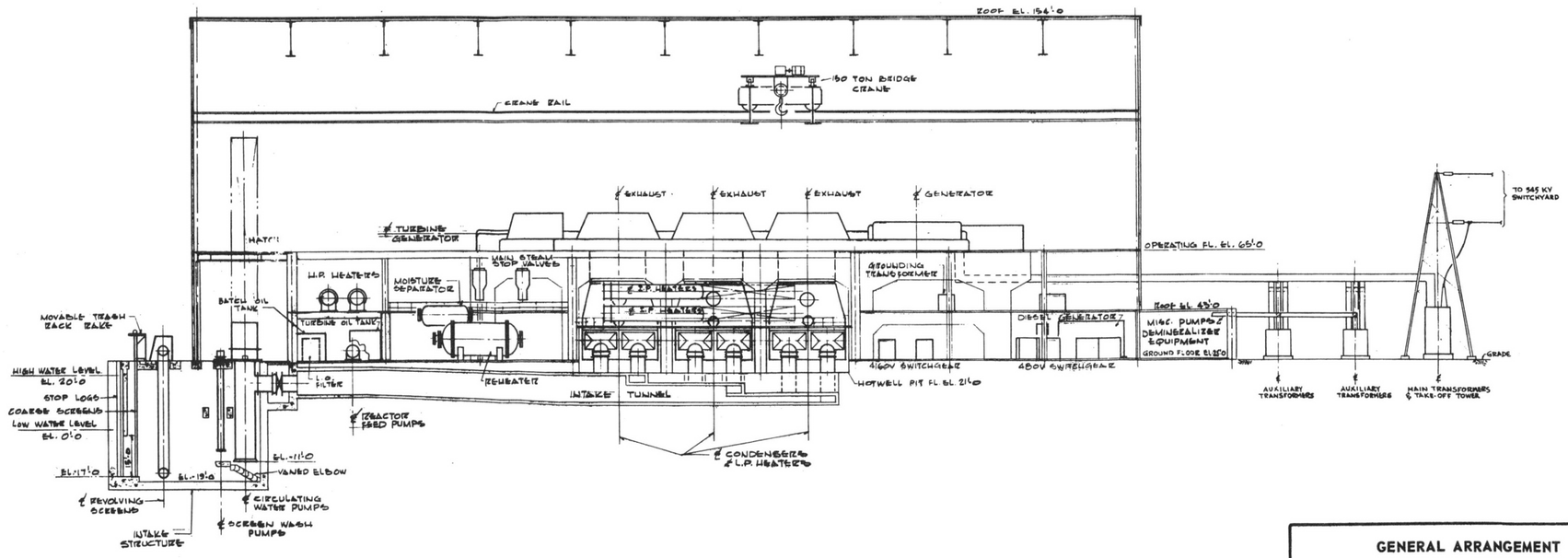


SECTION B-B

**GENERAL ARRANGEMENT**
**SECTIONS B–B – ARRANGEMENT B**
SIMILAR TO DRAWING NO. 3250–M–27
**1000 MWe BOILING WATER REACTOR PLANT**
**FEASIBILITY STUDY**
**Prepared For**
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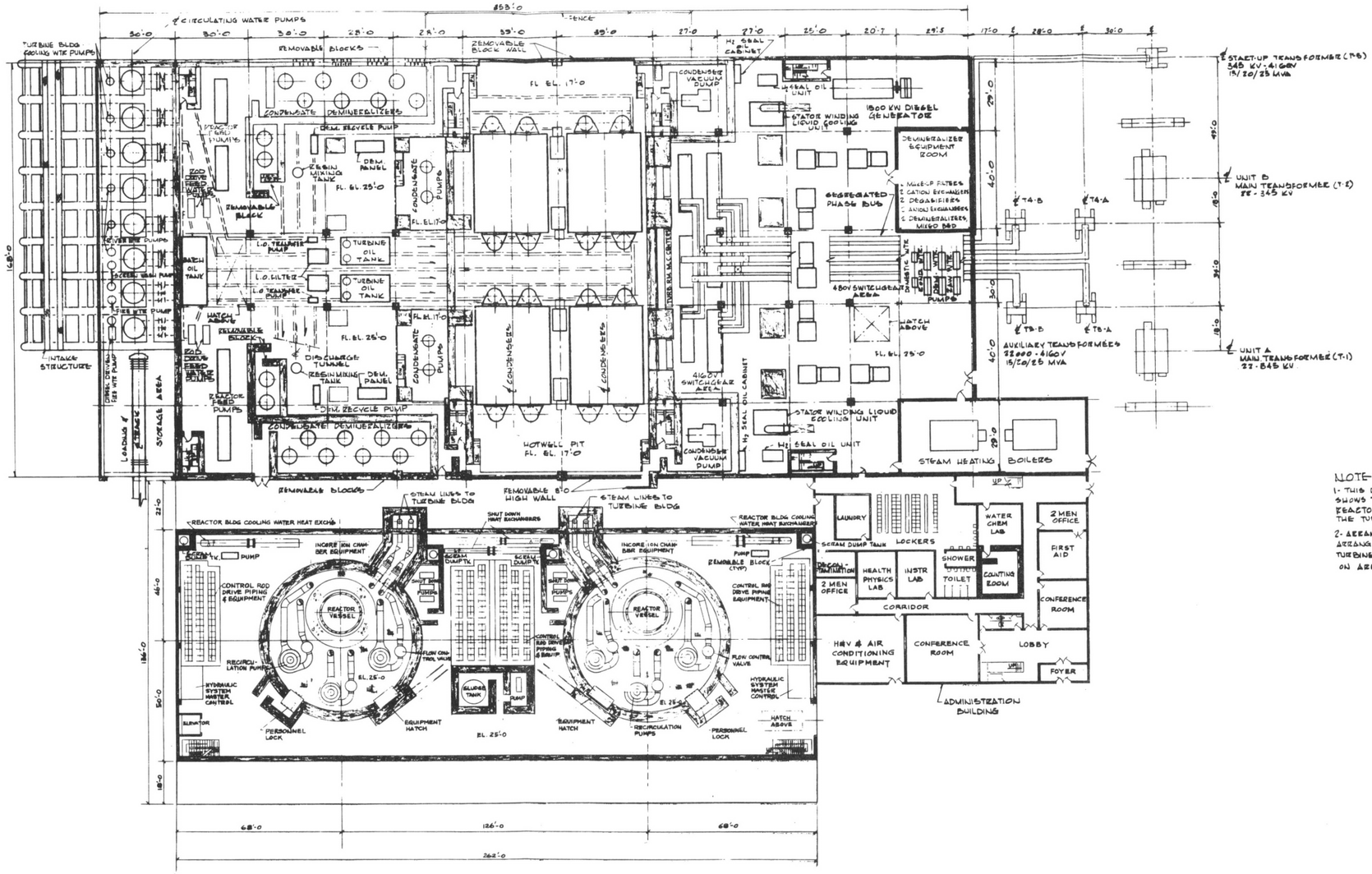


TURBINE BUILDING - GROUND FLOOR PLAN

NOTE
1- THIS DRAWING OF ARRANGEMENT(C) SHOWS THE LOCATION OF THE TWIN REACTOR BUILDINGS WITH RESPECT TO THE TURBINE BUILDING OF ARRGT.(A).

2- ARRANGEMENT(D) IS SIMILAR TO ARRANGEMENT(C) EXCEPT FOR THE TURBINE BUILDING WHICH IS AS SHOWN ON ARRANGEMENT(B).

**GENERAL ARRANGEMENT**
**GROUND FLOOR PLAN – ARRANGEMENT C**
SIMILAR TO DRAWING NO. 3250–M–42
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
**Prepared For**
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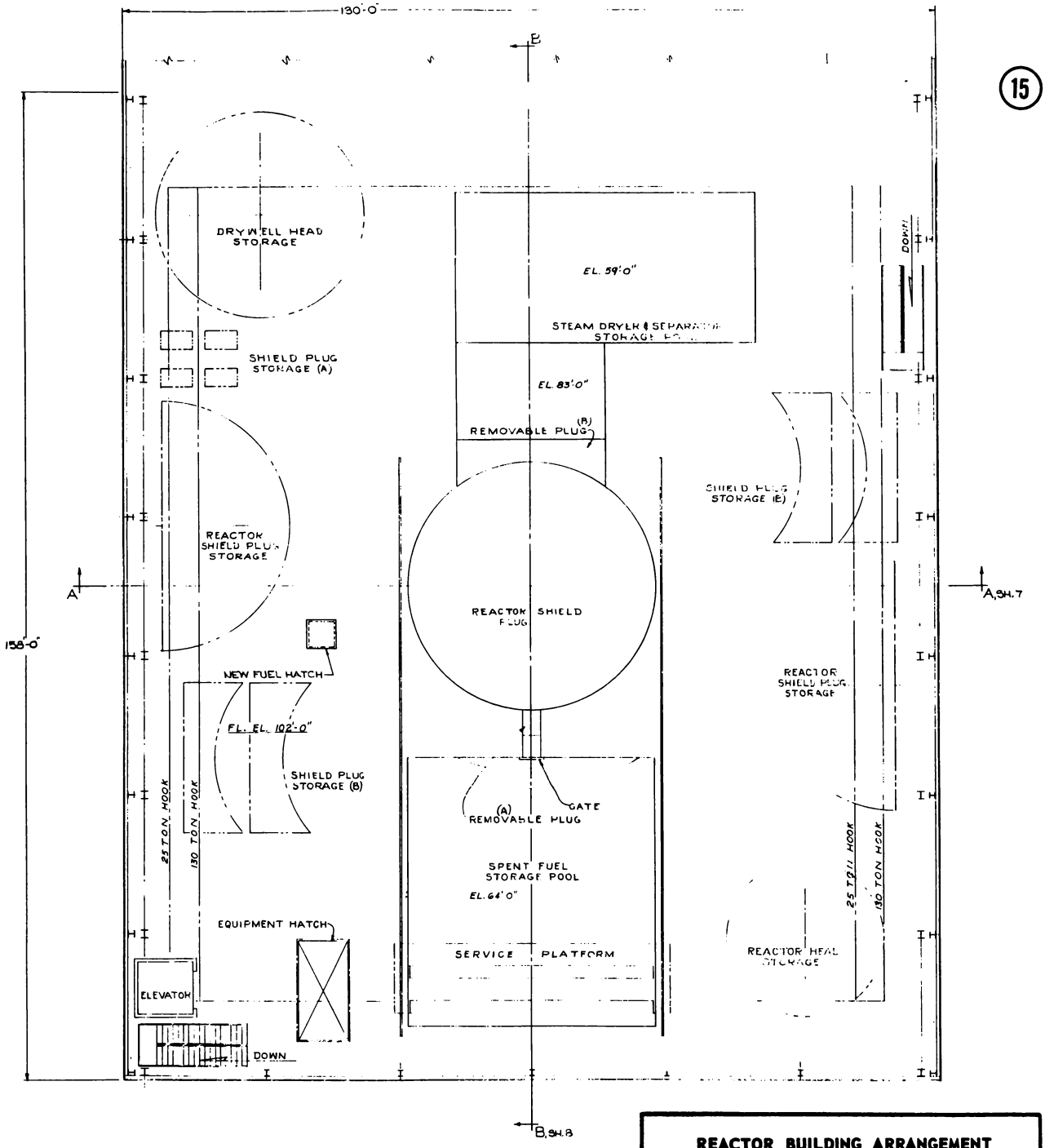


PLAN No. 1

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 1

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

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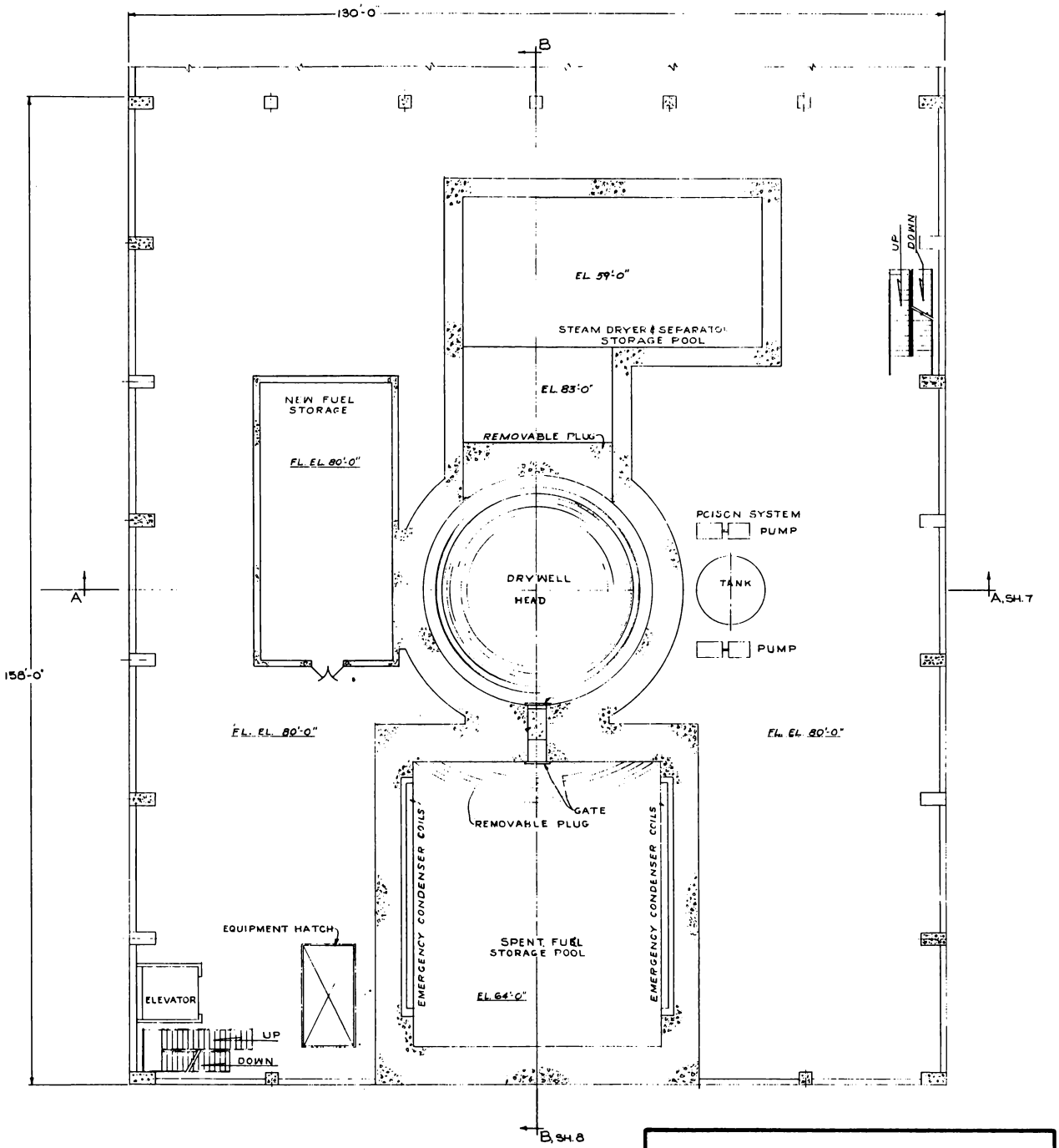
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT




PLAN No 2

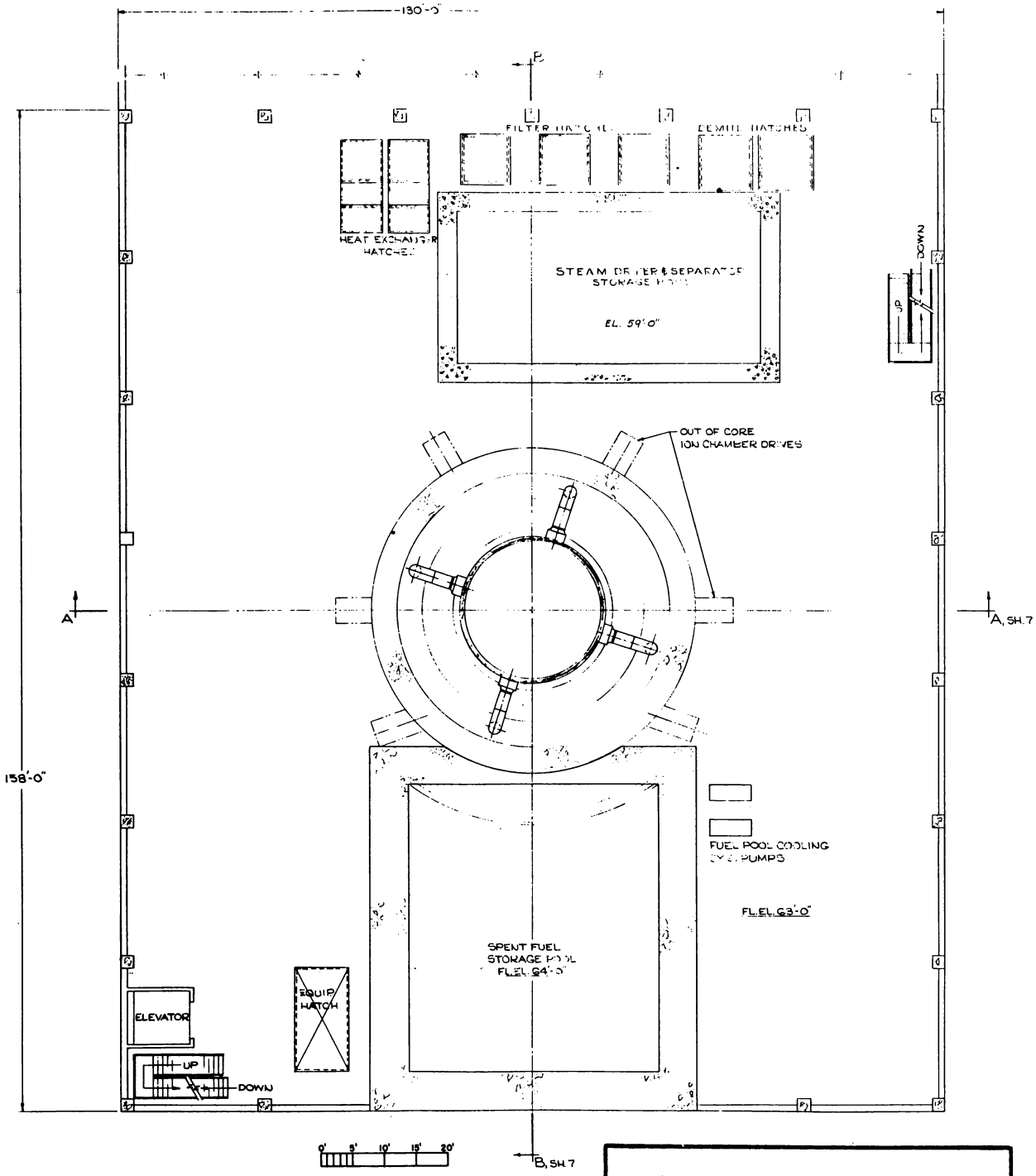
**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 2

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




PLAN NO. 3

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 3

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For

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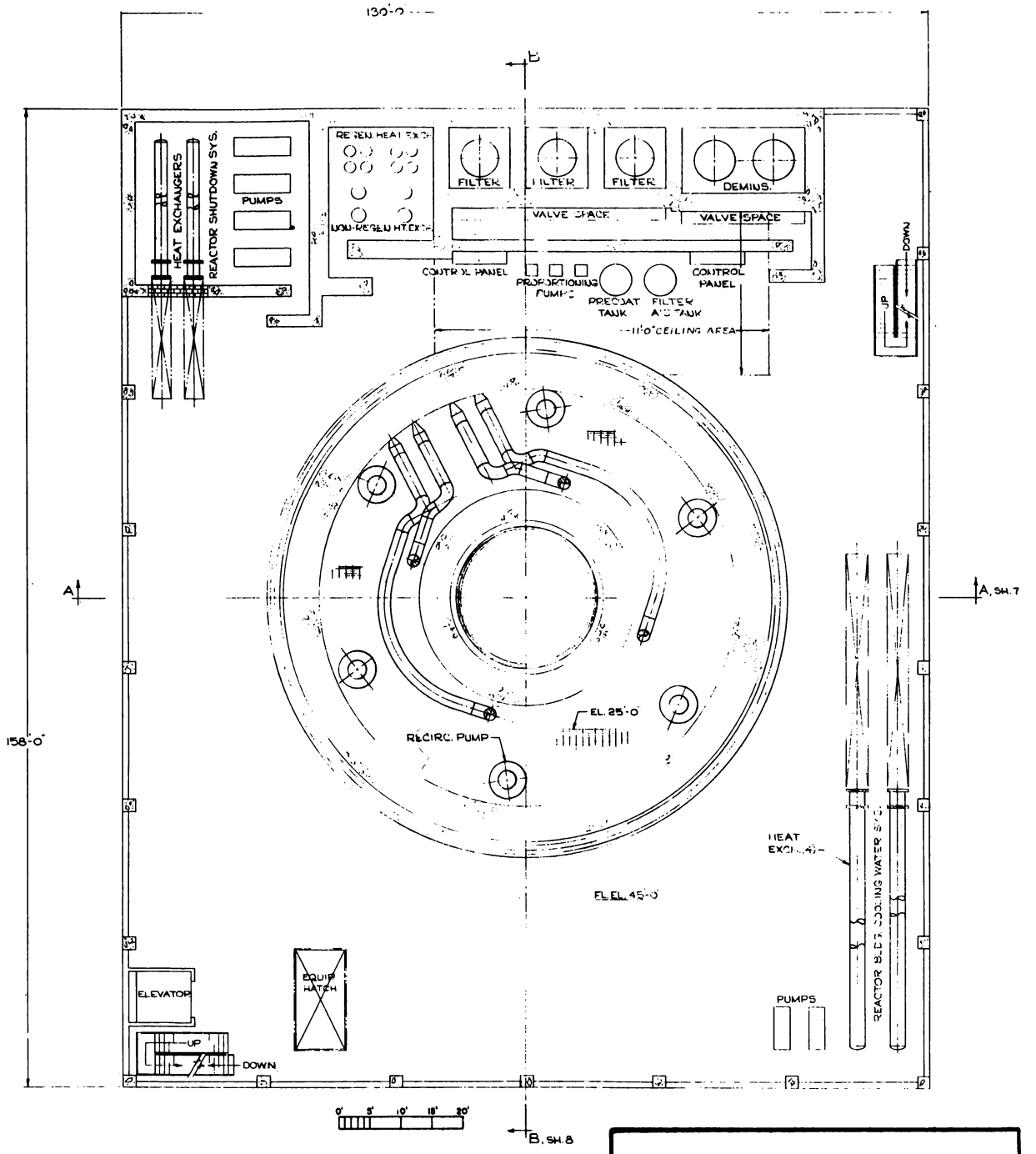
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

GPO 812-316-2




PLAN NO. 4

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 4

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For

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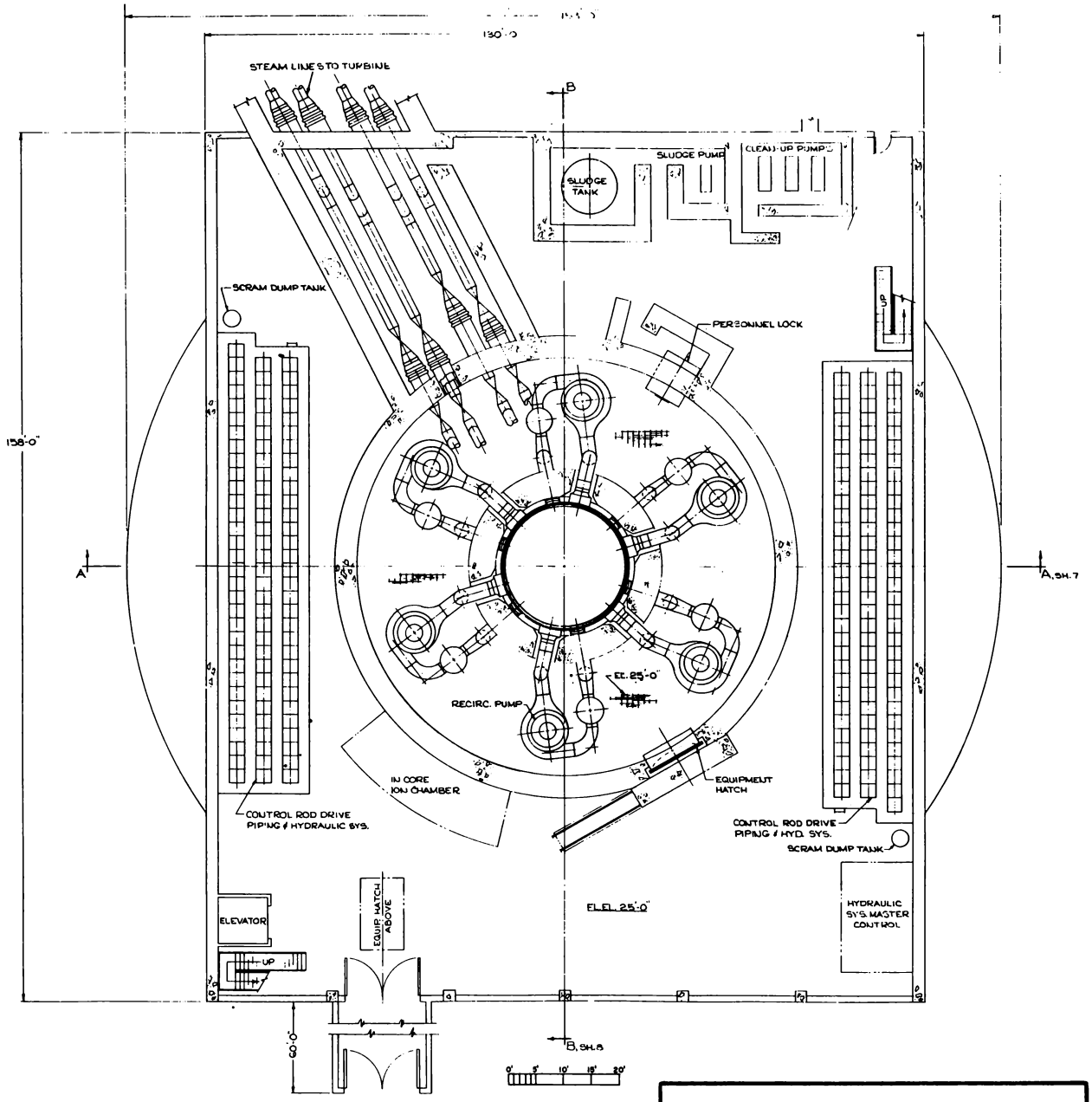
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT




PLAN NO. 5

**REACTOR BUILDING ARRANGEMENT**

**SIMILAR TO DRAWING NO. 148F210 SHEET 5**

**1000 MWe BOILING WATER REACTOR PLANT**
**FEASIBILITY STUDY**

**Prepared For**
**UNITED STATES ATOMIC ENERGY COMMISSION**

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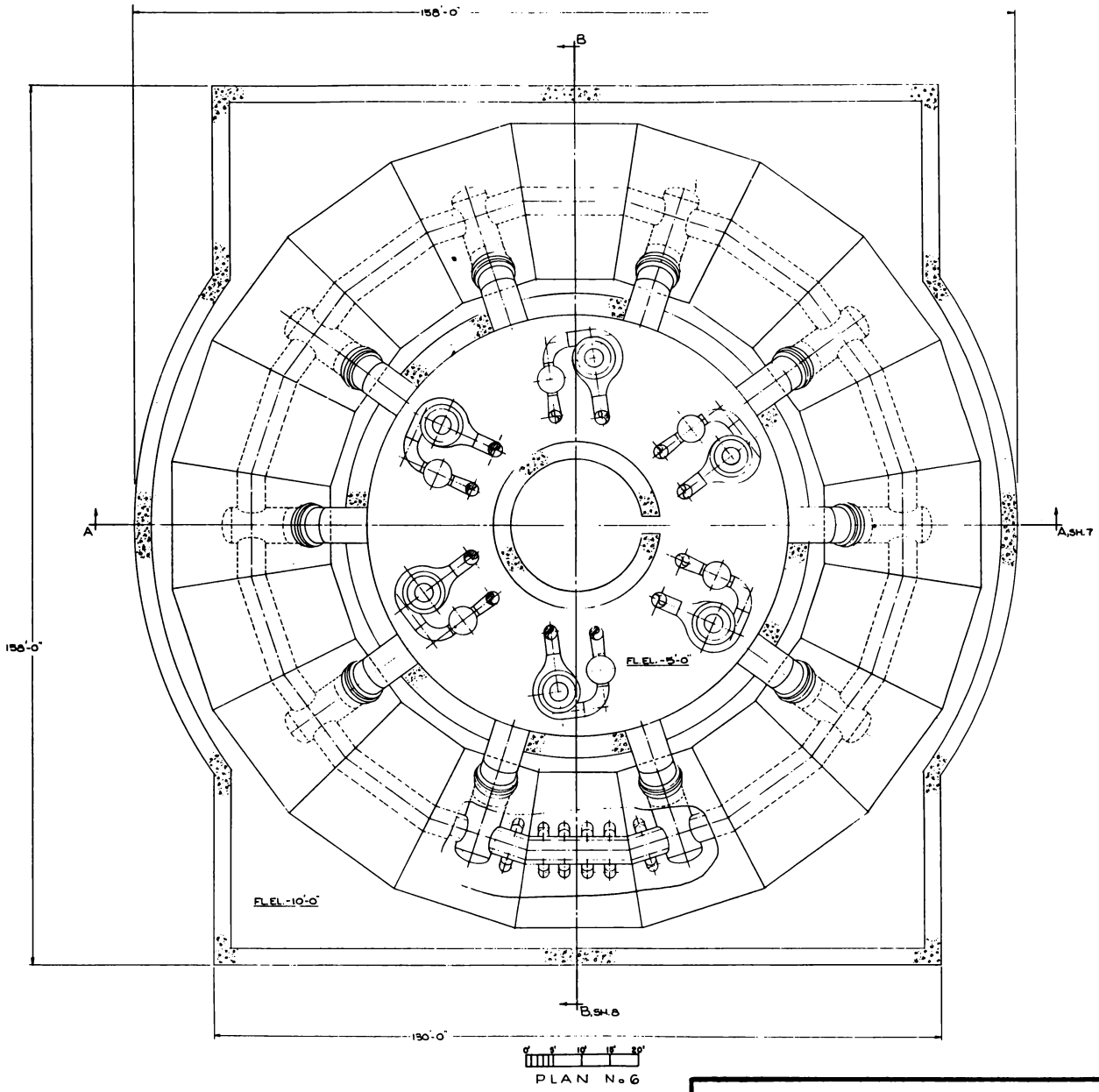
**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**




PLAN No. 6

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 6

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

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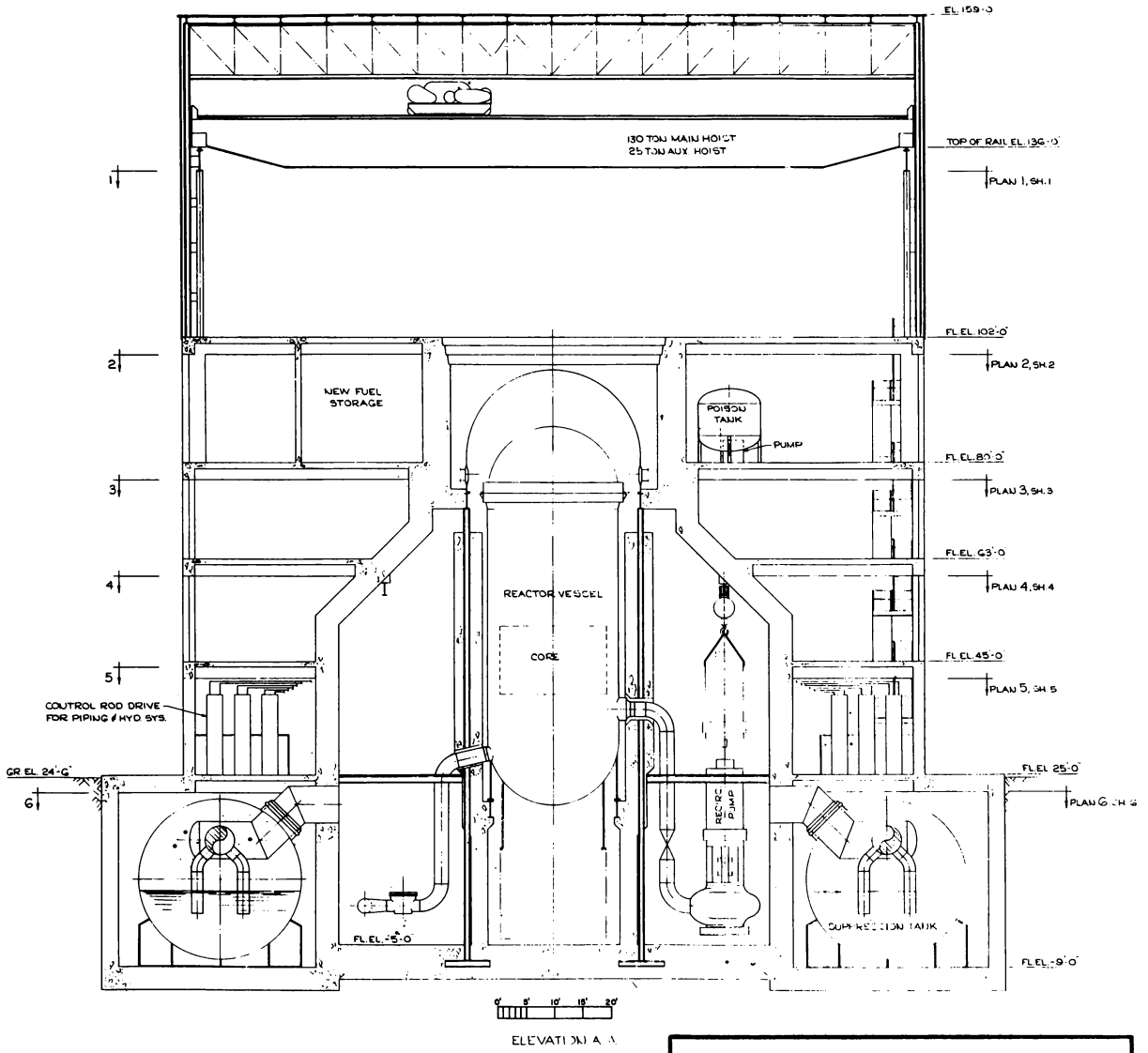
GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 7

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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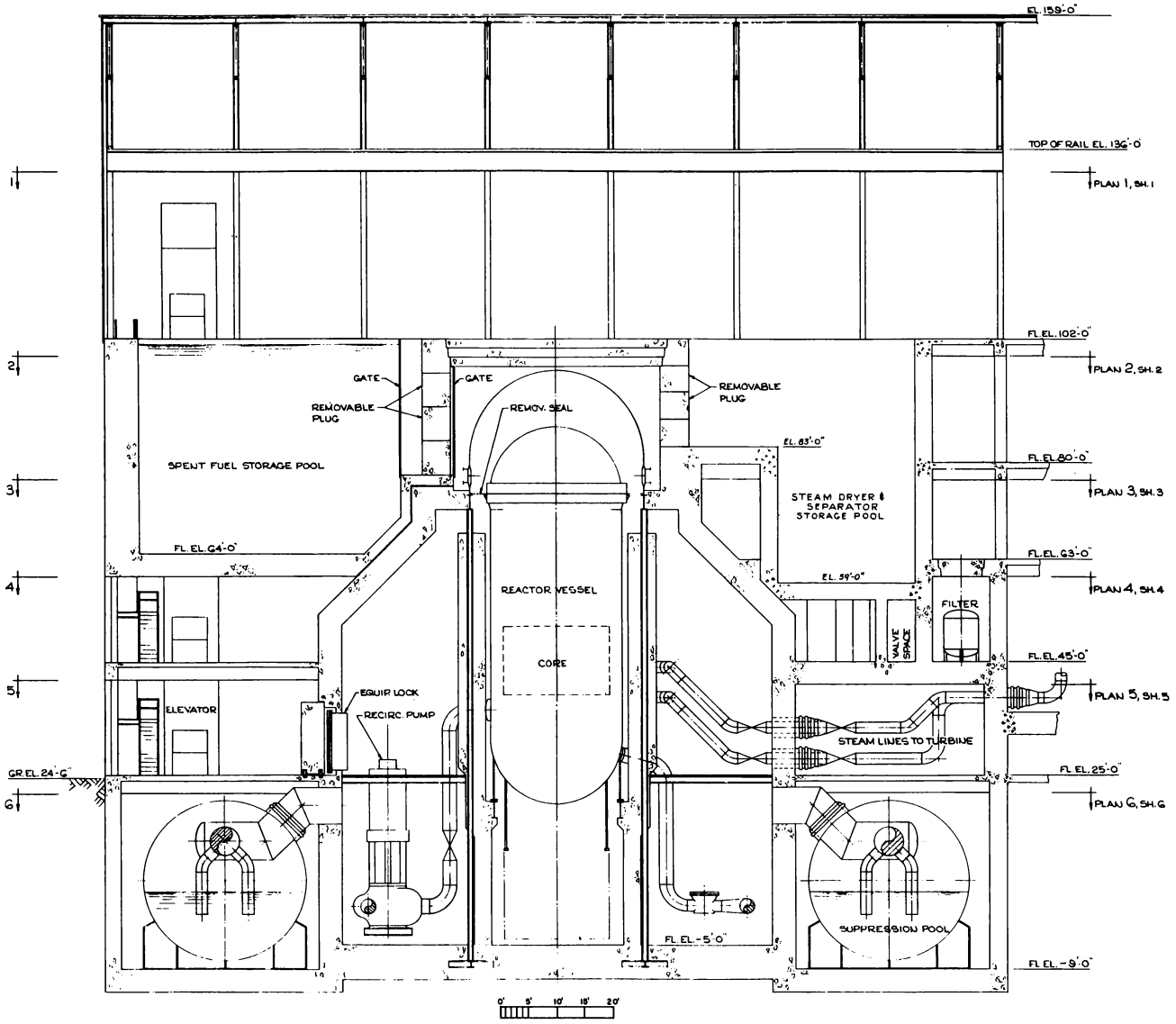
**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT




ELEVATION B-B

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F210 SHEET 8

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For

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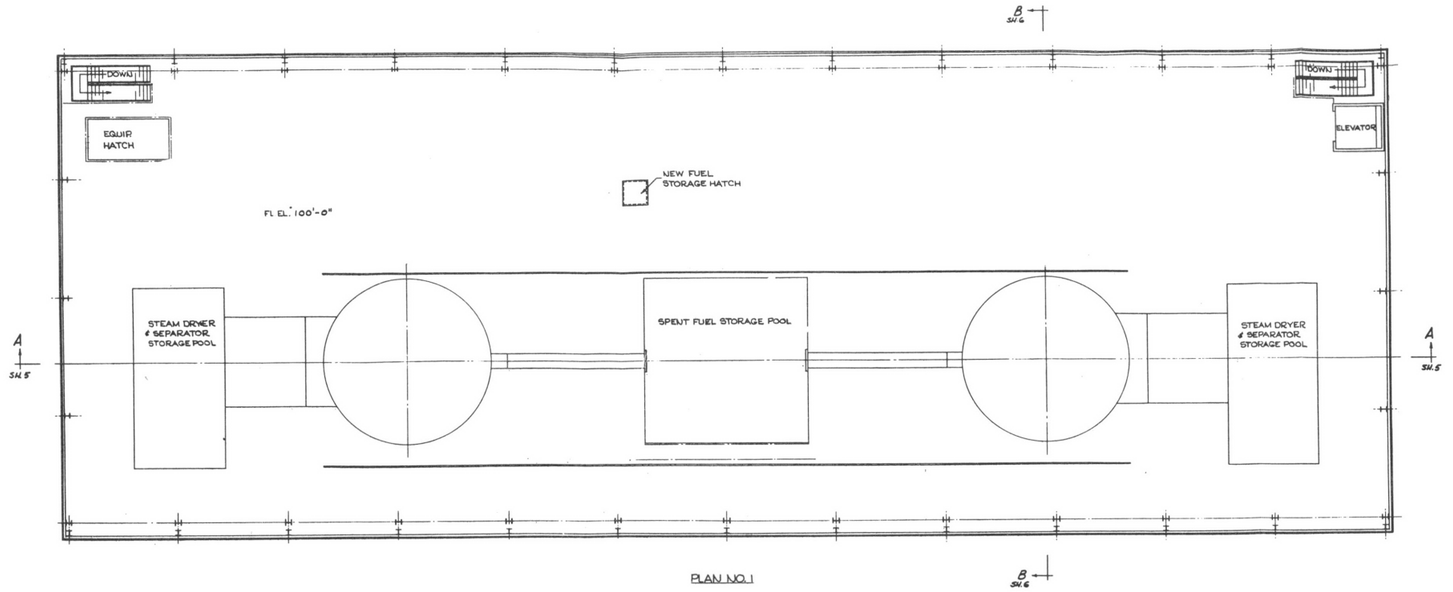
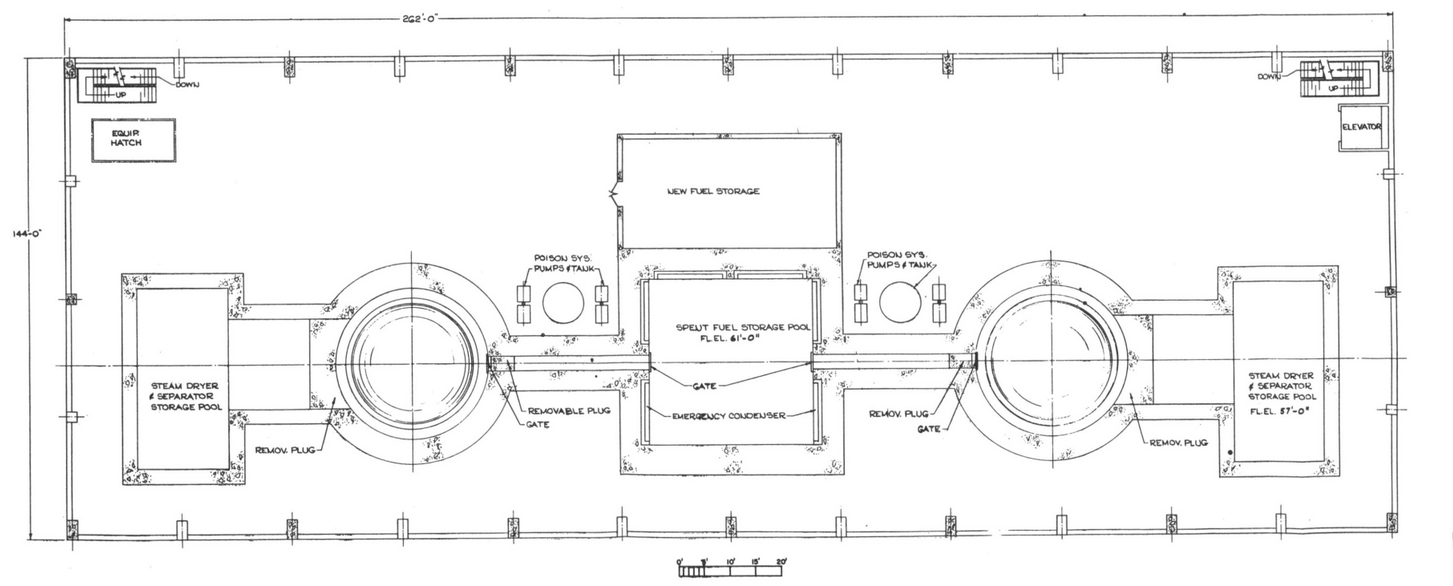
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT




PLAN NO. 1




PLAN NO. 2

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E294 SHEET 1

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

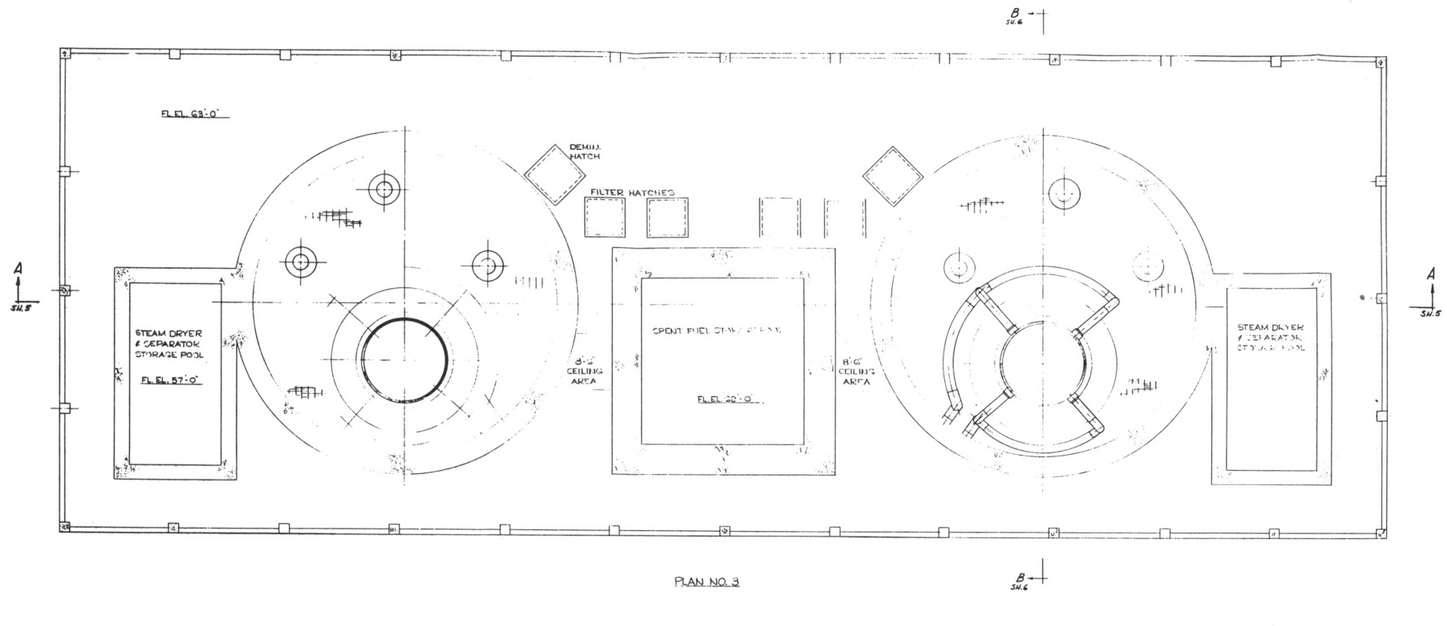
Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

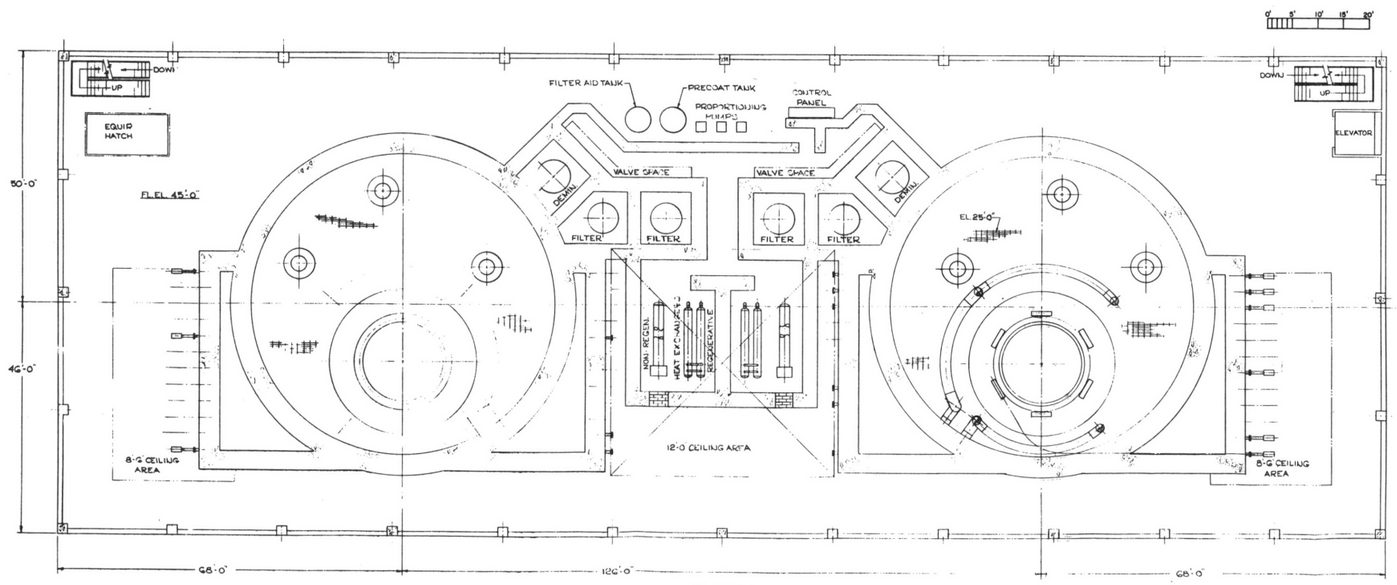
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

GPO 812–316–8




PLAN NO. 3




PLAN NO. 4

0 5 10 15 20 25
GRAPHIC SCALE IN FEET

**REACTOR BUILDING ARRANGEMENT**
SIMILAR TO DRAWING NO. 237E294 SHEET 2

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY
Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

B SH. 6

B SH. 6

PERSONNEL LOCK

EQUIPMENT HATCH

EL. 25'-0"

HATCH ABOVE

HYDRAULIC SYSTEM MASTER CONTROL

RECIRCULATION PUMPS

FLOW CONTROL VALVE

CONTROL ROD DRIVE PIPING & EQUIPMENT

REACTOR VESSEL

PUMP

SLUDGE TANK

SHUT DOWN PUMPS

SCRAM DUMP TANK

SHUT DOWN HEAT EXCHANGERS

INCORE ION CHAMBER EQUIPT.

STEAM LINES TO TURBINE BLDG.

ELEVATOR

PUMP

REACTOR BLDG. COOLING WATER HEAT EXCHANGER

136'-0"

62'-0"

A SH.5

A SH.5

68'-0"

126'-0"

68'-0"

262'-0"

B SH. 6

B SH. 6

0' 5' 10' 15' 20'

PLAN No. 5

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E294 SHEET 3

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

Prepared For

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GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

B SH. 6

B SH. 6

EL. -9'-0"

EL. -9'-0"

FLOW CONTROL VALVES

RECIRCULATION PUMPS

EL. -5'-0"

REACTOR VESSEL

27'-0" DIA. TORUS

136'-0"

68'-0"

A SH. 5

68'-0"

126'-0"

68'-0"

262'-0"

B SH. 6

B SH. 6

0' 5' 10' 15' 20'

PLAN No. 6

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E294 SHEET 4

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
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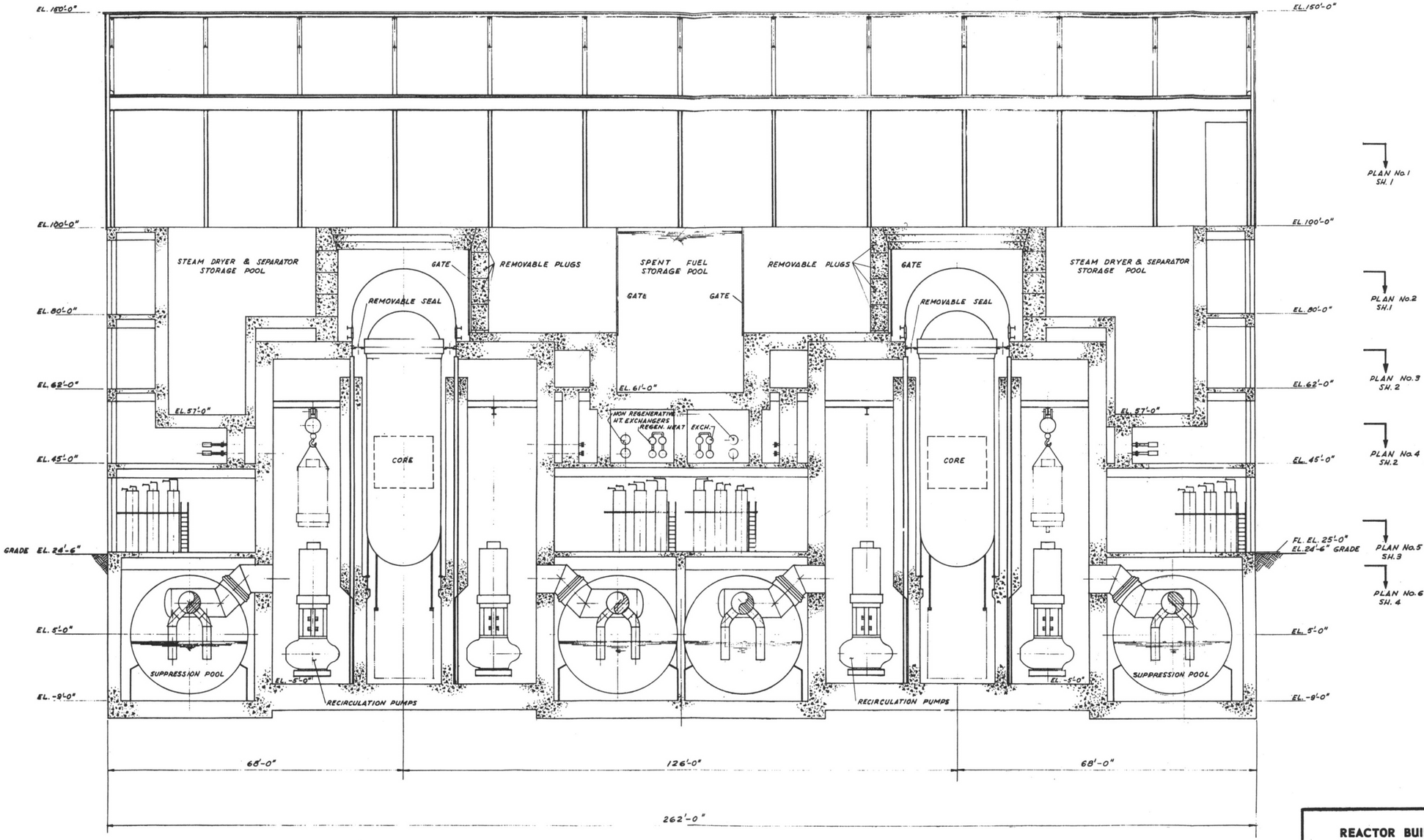
GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT




ELEVATION A-A

GPO 812-316-9

**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E294 SHEET 5

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
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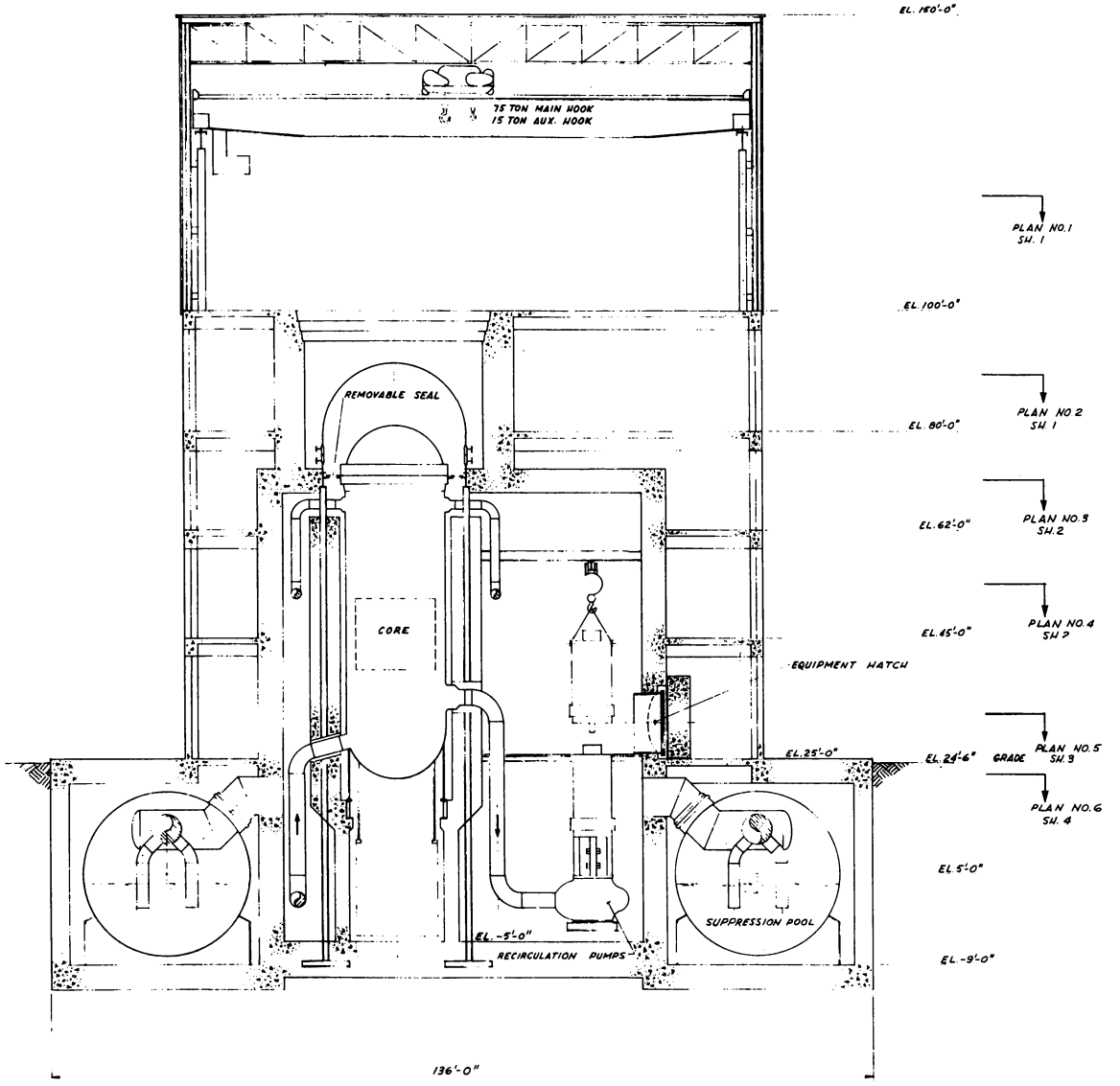
GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




ELEVATION B-B

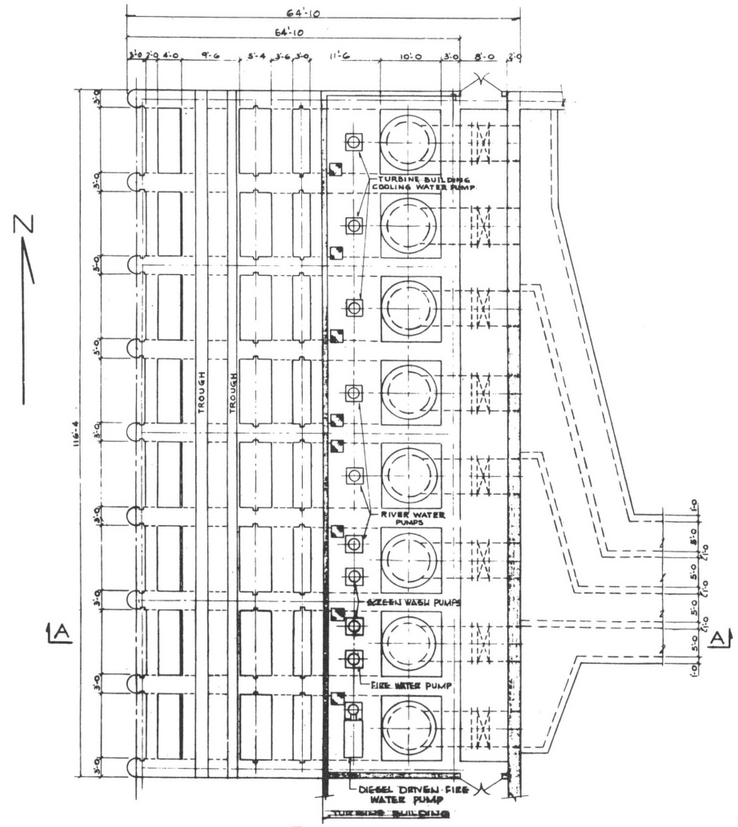
**REACTOR BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E294 SHEET 6

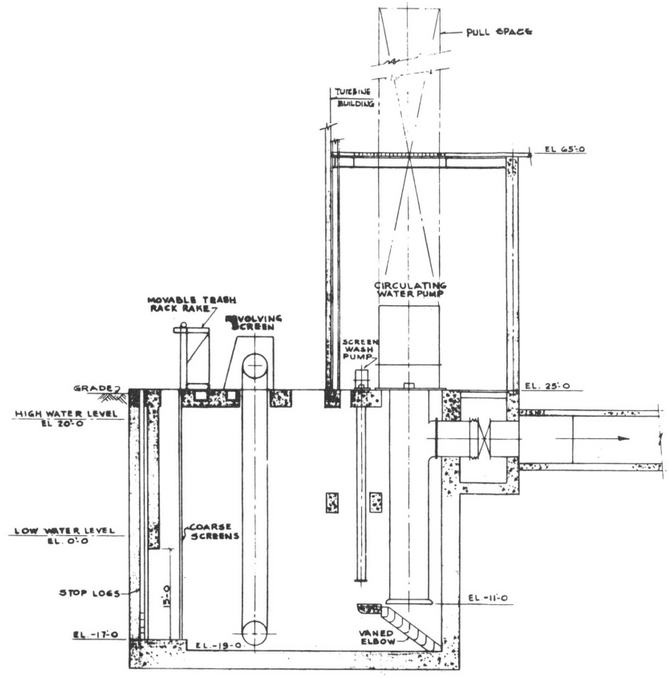
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




PLAN




SECTION A-A

**INTAKE STRUCTURE**

SIMILAR TO DRAWING NO. 3250–S–1

**1000 MWe BOILING WATER REACTOR PLANT**
**FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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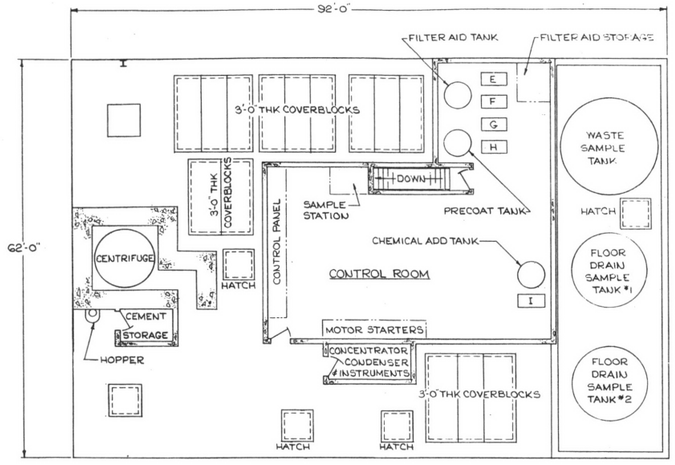
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS

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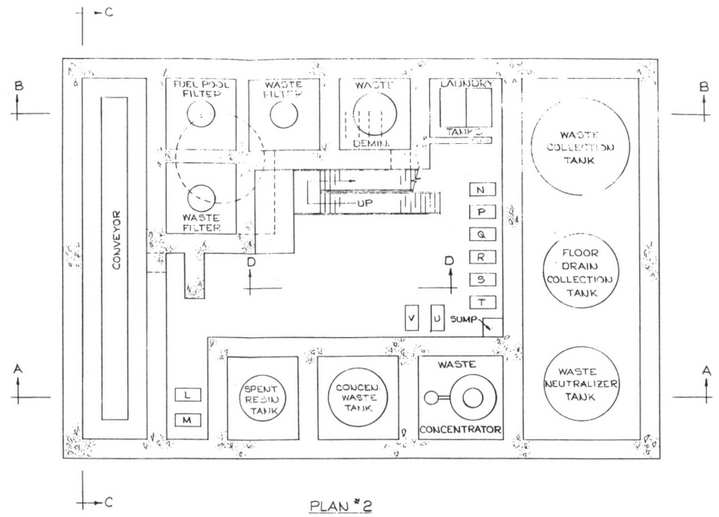


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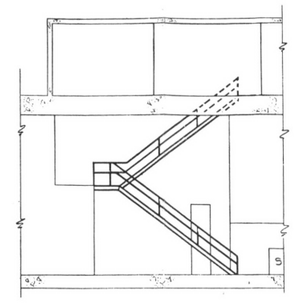
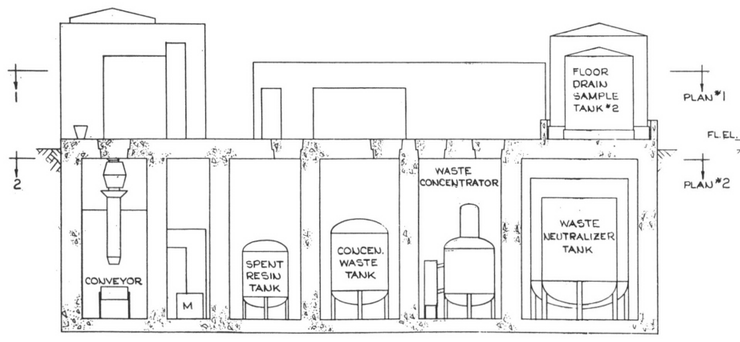


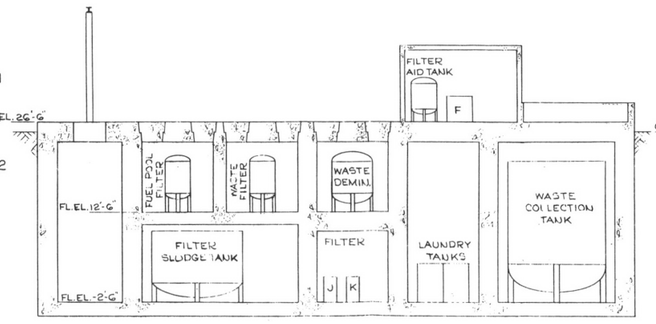

ELEV. D-D

| PUMP SCHEDULE | | | |
|---|---|---|---|
| E | FILTER AID PROPORTIONING PUMP #1 | Q | CLEAN WASTE TRANSFER PUMP |
| F | FILTER AID PROP PUMP #2 | R | FLOOR DRAIN TRANSFER PUMP |
| G | FILTER AID PROP PUMP #3 | S | WASTE NEUTRALIZER TRANSFER PUMP |
| H | FILTER PRECOAT PUMP | T | WASTE CONCENTRATOR FEED PUMP |
| I | CHEMICAL ADD TANK PROP PUMP | U | WASTE DISCHARGE PUMP #1 |
| J | FILTER SLUDGE PUMP #1 | V | WASTE DISCHARGE PUMP #2 |
| K | FILTER SLUDGE PUMP #2 | | |
| L | CONCENTRATED WASTE TRANSFER PUMP | | |
| M | RESIN TRANSFER PUMP | | |
| N | LAUNDRY WASTE TRANSFER PUMP | | |
| P | WASTE TRANSFER PUMP | | |

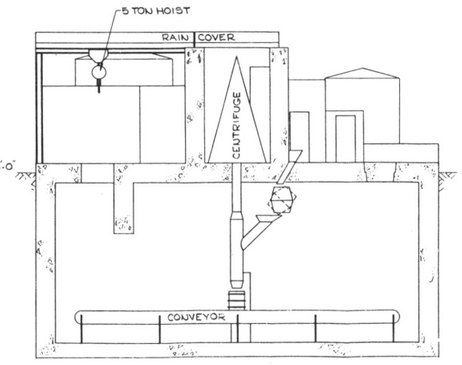
0 5 10 15 20 25
GRAPHIC SCALE IN FEET




ELEV. A-A




ELEV. B-B




ELEV. C-C

REVISIONS

**RADWASTE BUILDING ARRANGEMENT**

SIMILAR TO DRAWING NO. 148F230

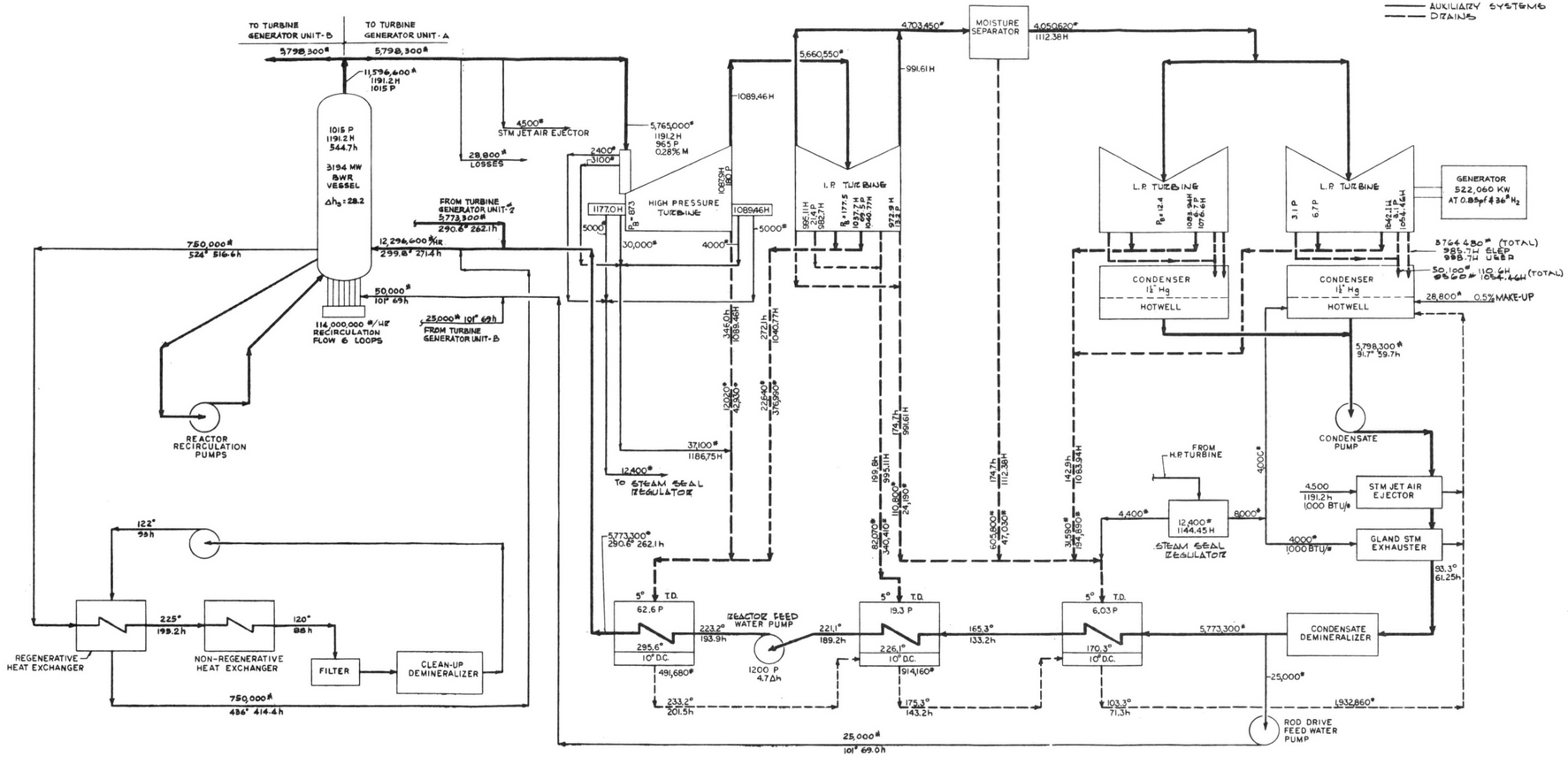
**1000 MWe BOILING WATER REACTOR PLANT**
**FEASIBILITY STUDY**

**Prepared For**
**UNITED STATES ATOMIC ENERGY COMMISSION**

◆ ◆ ◆

**GENERAL ELECTRIC**
**ATOMIC POWER EQUIPMENT DEPARTMENT**

LEGEND
- # FLOW-LBS/HR
- P FREEDURE - 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



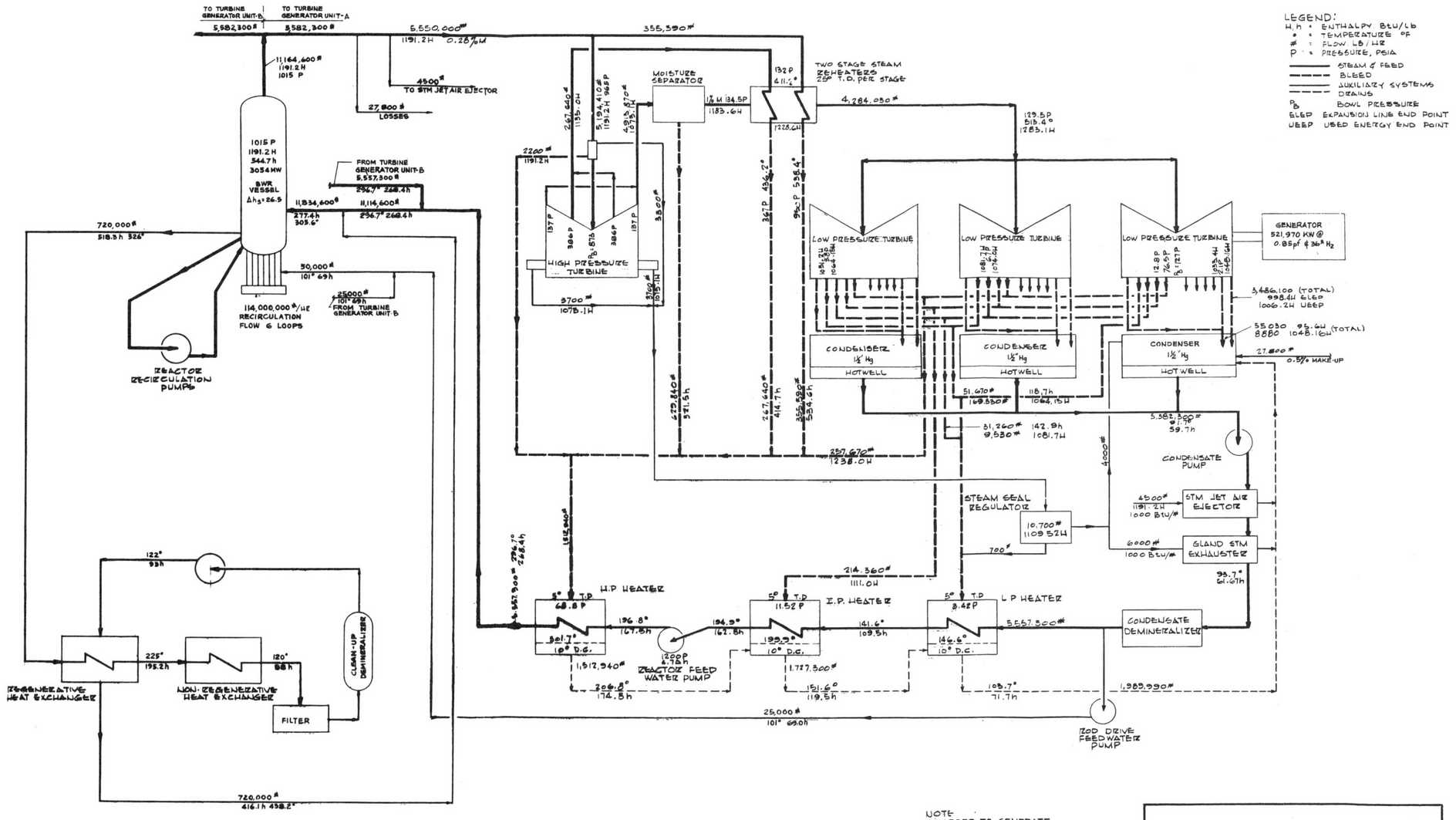

ESTIMATED GROSS PLANT HEAT RATE = $\frac{3194\ MW \times 3.413 \times 10^6\ BTU/MW\text{-}HR}{1,044,120\ KW} = 10,440\ BTU/KW\text{-}HR$

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

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




ESTIMATED GROSS PLANT HEAT RATE = $\frac{3054 \text{ MW} \times 3.413 \times 10^6 \text{ BTU/MW-HR}}{1,043,940 \text{ KW}}$ = 9,984 BTU/KW-HR

ESTIMATED NET PLANT HEAT RATE = $\frac{3054 \text{ MW} \times 3.413 \times 10^6 \text{ BTU/MW-HR}}{1,000,000 \text{ KW}}$ = 10,420 BTU/KW-HR

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

**HEAT BALANCE – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 3250–M–28

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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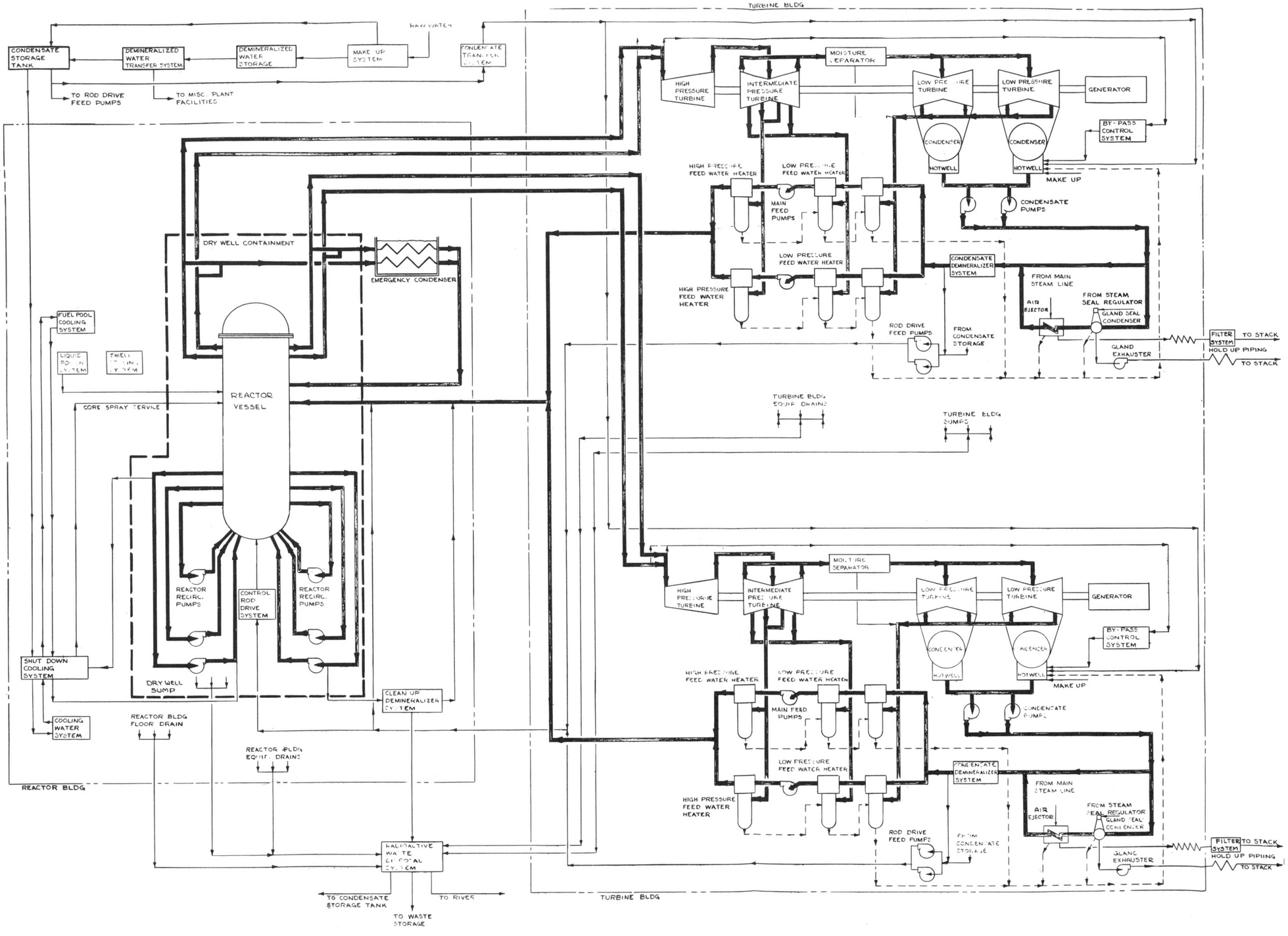
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS

GPO 812–316–10




**PLANT FLOW DIAGRAM – ARRANGEMENT A**

SIMILAR TO DRAWING NO. 237E285

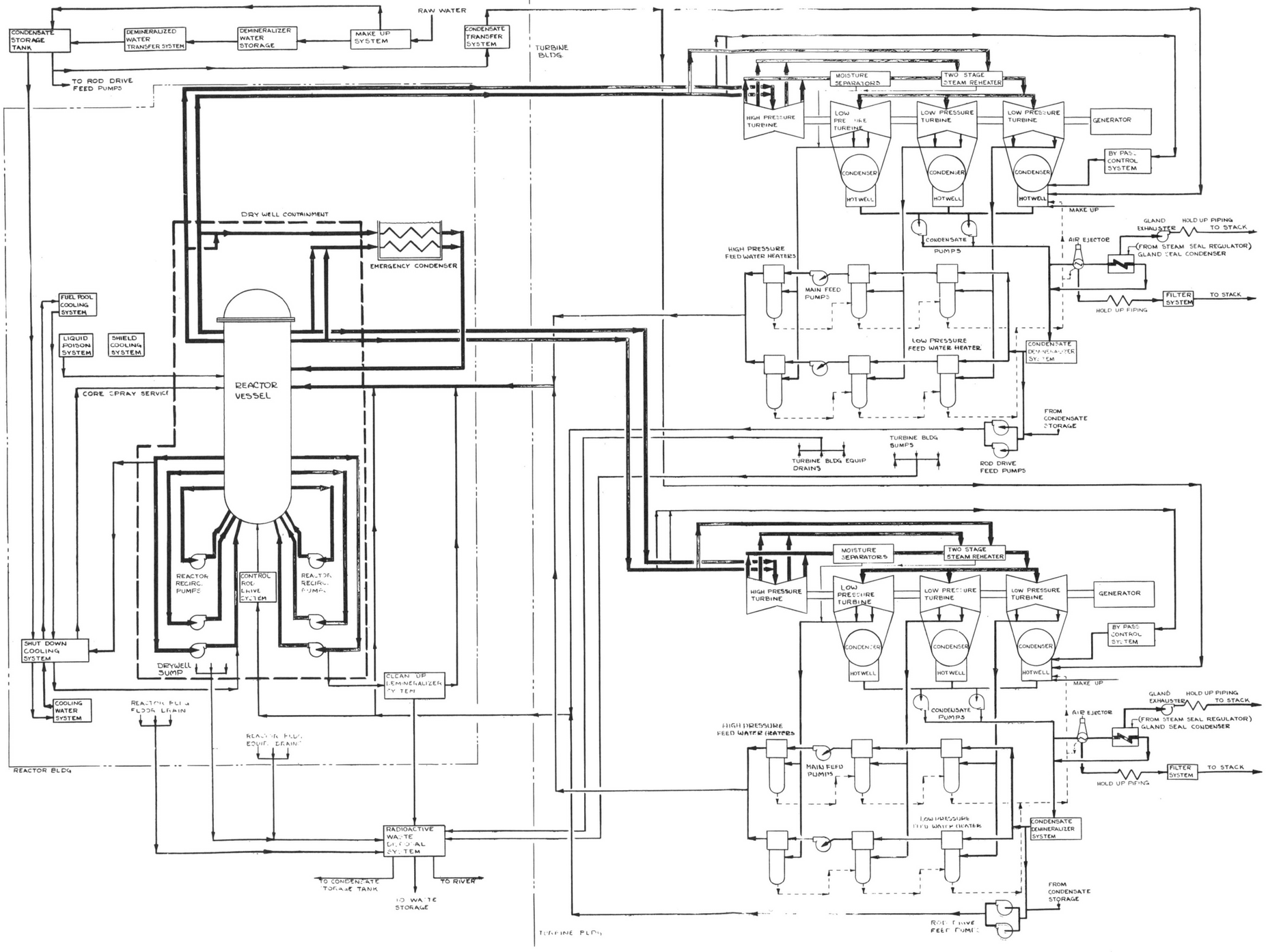
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT



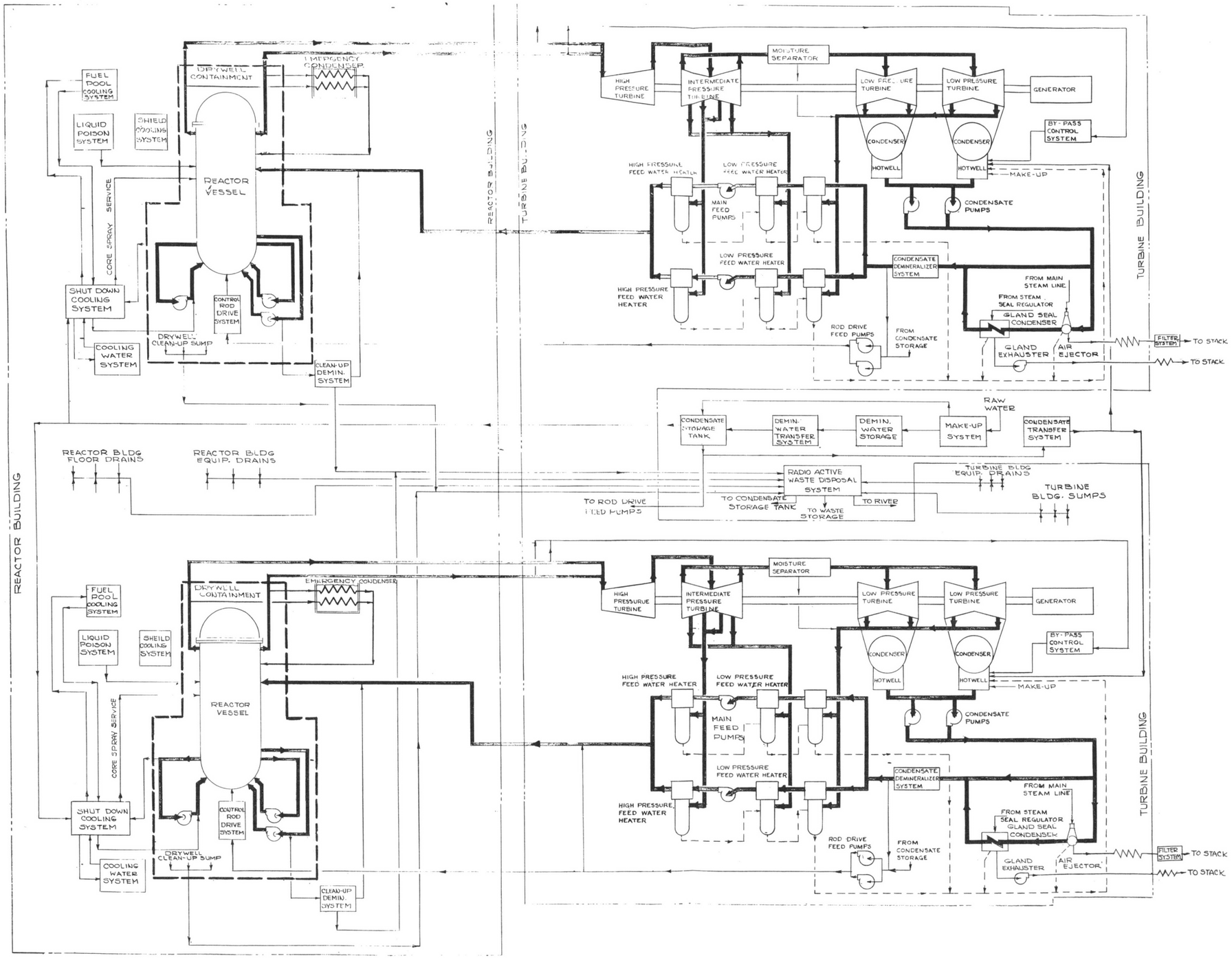

**PLANT FLOW DIAGRAM – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 237E288

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




**PLANT FLOW DIAGRAM – ARRANGEMENT C**

SIMILAR TO DRAWING NO. 237E287

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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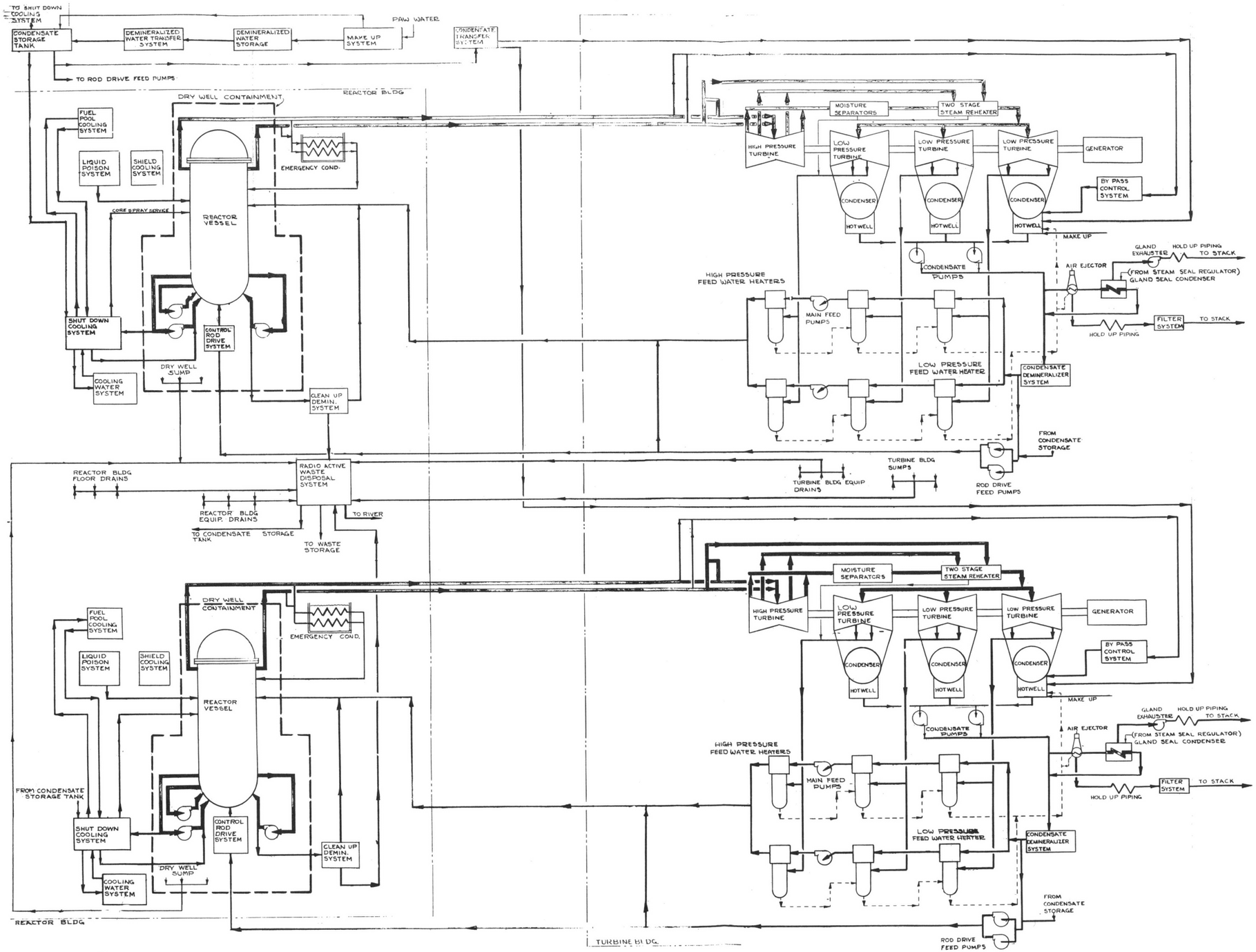
**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT




GPO 812-316-11

**PLANT FLOW DIAGRAM – ARRANGEMENT D**

SIMILAR TO DRAWING NO. 237E289

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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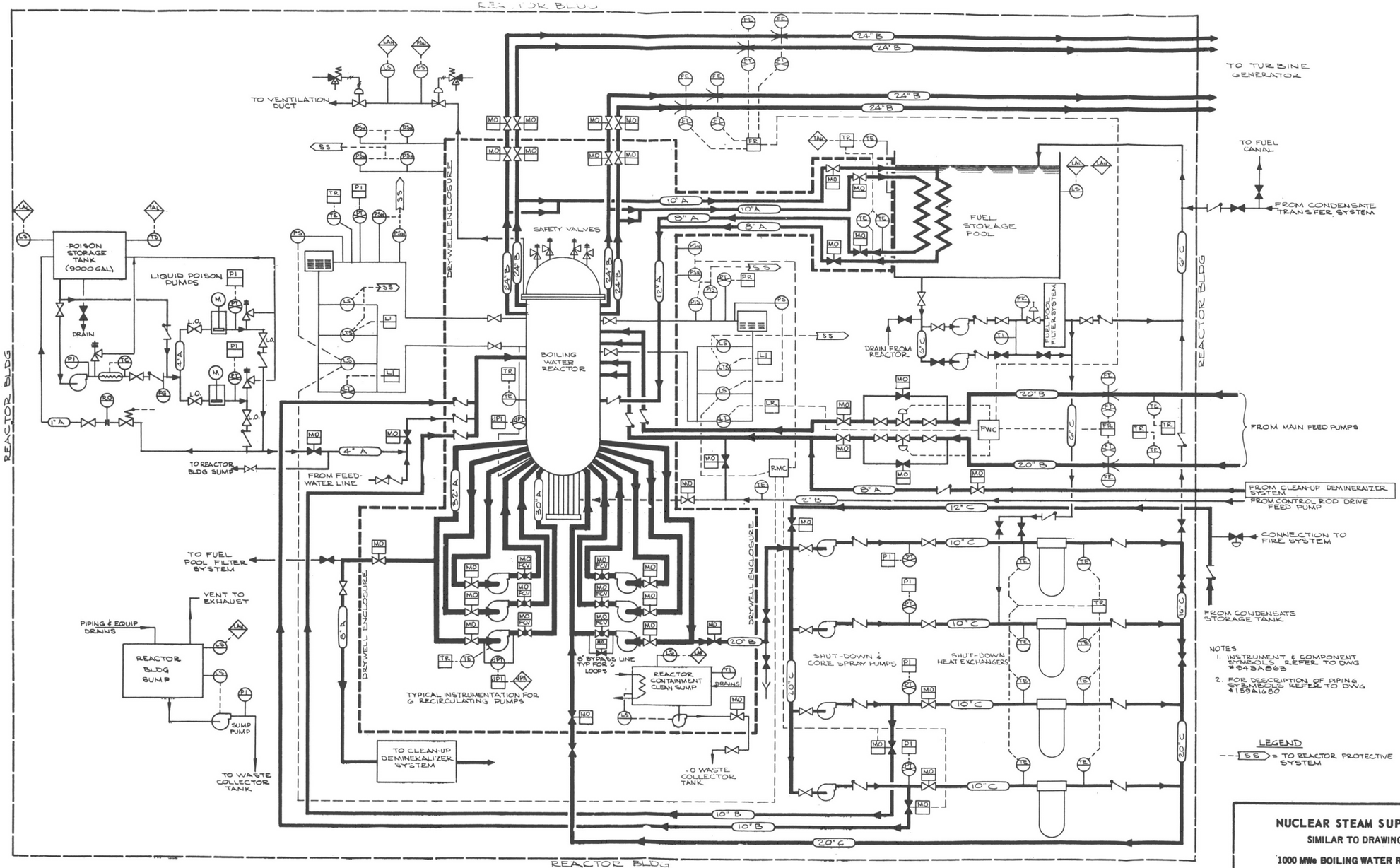
**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT




NOTES
1. INSTRUMENT & COMPONENT SYMBOLS REFER TO DWG #934A863
2. FOR DESCRIPTION OF PIPING SYMBOLS REFER TO DWG #159A1680

LEGEND
--SS-- = TO REACTOR PROTECTIVE SYSTEM

PRELIMINARY

**NUCLEAR STEAM SUPPLY SYSTEM**

SIMILAR TO DRAWING NO. 237E283

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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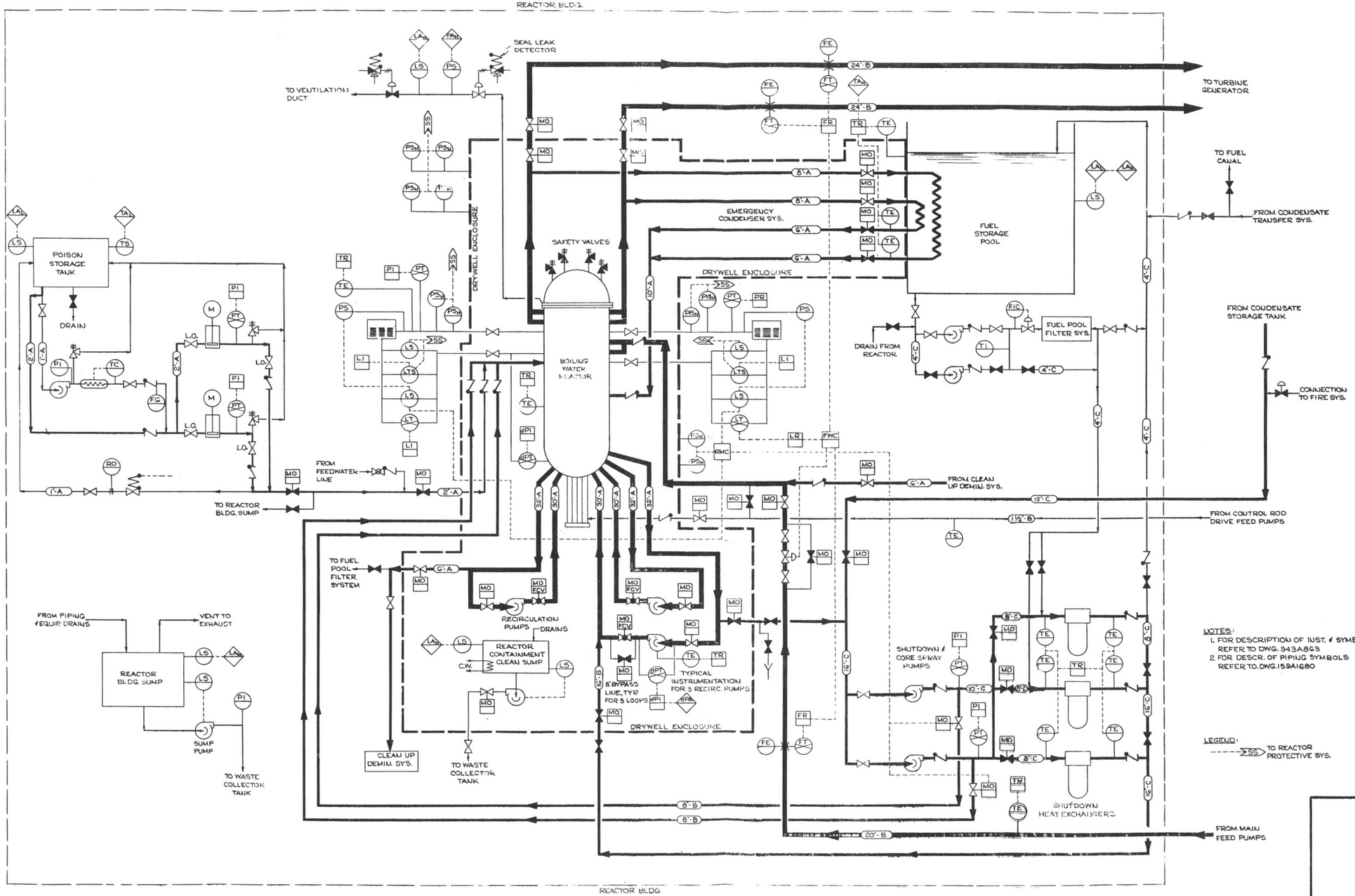
**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**




NOTES:
1. FOR DESCRIPTION OF INST. # SYMBOLS REFER TO DWG. 943A863
2. FOR DESCR. OF PIPING SYMBOLS REFER TO DWG. 159A1680

LEGEND:
SS — TO REACTOR PROTECTIVE SYS.

NOTE: IN ORDER TO PRODUCE 1000 MWe, ARRANGEMENT 'C' WILL PROVIDE FOR TWO IDENTICAL UNITS OF THE PLANT SHOWN ABOVE.

**P & I DIAGRAM – NSSS, (C & D)**

SIMILAR TO DRAWING NO. 237E284

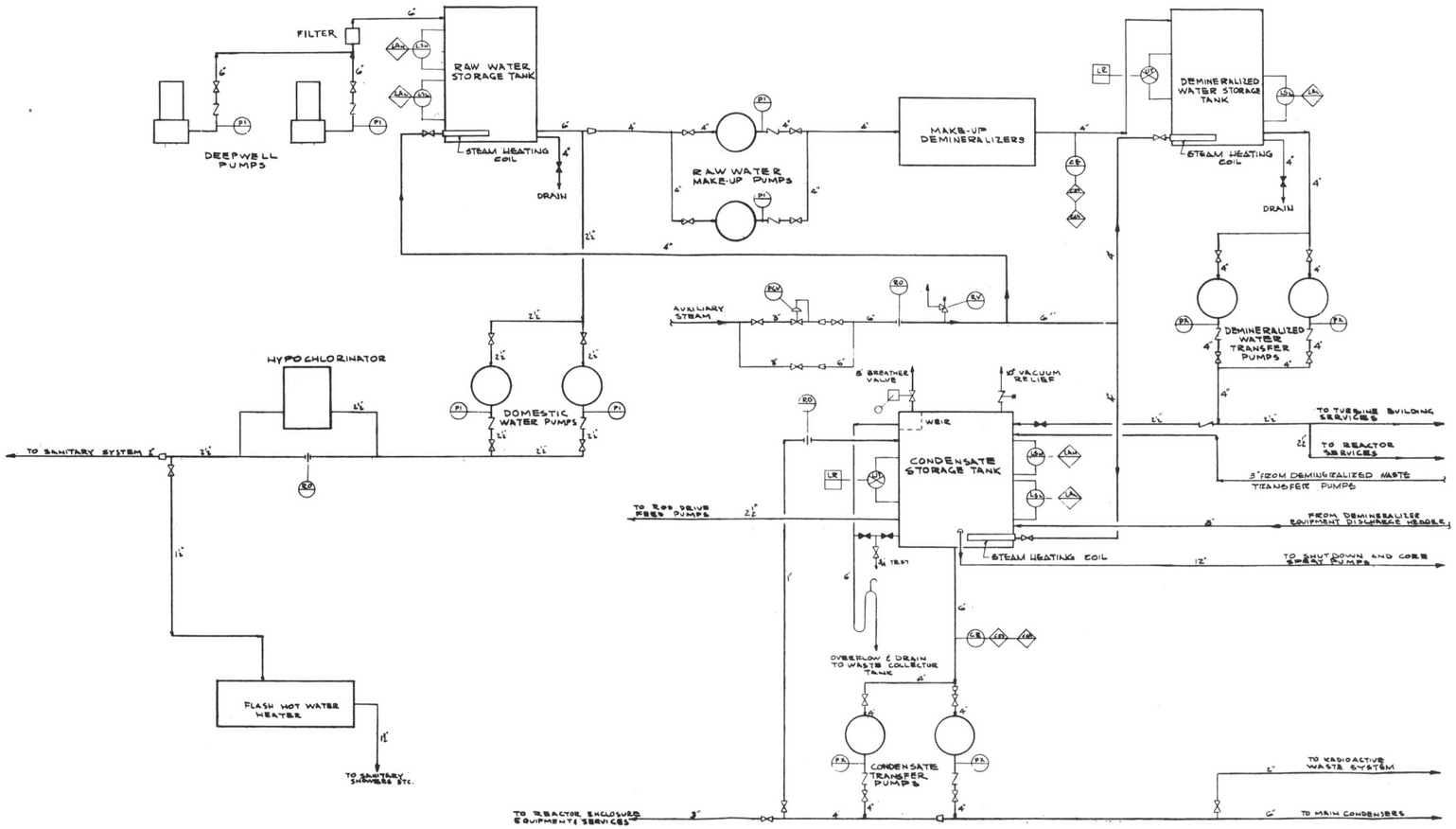
**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT




GENERAL NOTE
ALL PIPING SCH 40 - 150# RATING

**FLOW DIAGRAM MAKE-UP SYSTEM & WATER STORAGE**

SIMILAR TO DRAWING NO. 3250–M–11
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**
**UNITED STATES ATOMIC ENERGY COMMISSION**

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**EBASCO SERVICES INCORPORATED**
ENGINEERS AND CONTRACTORS

MAKE UP CONDENSATE FROM COND EVAPOR STORAGE TANK

DEMINERALIZED WATER FROM TRANSFER PUMPS

REACTOR BUILDING COOLING WATER SURGE TANK

TO EDDT

REACTOR BUILDING COOLING WATER PUMPS

SAMPLE COOLERS

NON REGENERATIVE HEAT EXCHANGER

REACTOR BUILDING COOLING WATER HEAT EXCHANGERS

REACTOR SHUTDOWN HEAT EXCHANGERS

REACTOR RECIRCULATING PUMP COOLERS

TO REACTOR CONTAINMENT CLEAN SUMP

REACTOR SHIELD COOLER

CLEAN SUMP COOLER

TO REACTOR CONTAINMENT CLEAN SUMP

DRY WELL COOLER

DRYWELL CONTAINMENT

FUEL POOL COOLER

CONCENTRATED WASTE STORAGE TANK COOLER

WASTE CONC. CONDENSER

BEARINGS COOLER

SHUT DOWN PUMP COOLERS

TO TURBINE B COOLERS

TURBINE A LUBE OIL COOLERS

CONDENSER A VACUUM PUMP

REACTOR FEED PUMPS (4 UNITS TOTAL)

TO UNIT B REACTOR FEED PUMPS

FROM UNIT B REACTOR FEED PUMPS

FROM TURBINE B COOLERS

INSTRUMENT AIR COMPRESSOR AFTER COOLER

INSTRUMENT AIR COMPRESSOR

TO SAMPLE COOLERS

GENERATOR A HYDROGEN COOLERS

GENERATOR A STATOR WATER COOLERS

PLANT AIR COMPRESSOR AFTER COOLER

PLANT AIR COMPRESSOR

DIESEL ENGINE COOLERS

H2 SEAL OIL COOLER

TO GENERATOR B COOLERS

FROM GENERATOR B COOLERS

FROM UNIT B EQUIPMENT

TO RADWASTE BUILDING

TO S.C. TO CIRCULATING WATER PUMPS

DISCHARGE TUNNELS FOR UNIT NO.2

DISCHARGE TUNNELS FOR UNIT NO.1

DISCHARGE TO RIVER

LUB WATER + STUFFING BOX DR.

TO AND FROM MAIN CONDENSER UNIT 'B'

TO AND FROM MAIN CONDENSER UNIT 'B'

MAIN CONDENSER UNIT 'A'

MAIN CONDENSER UNIT 'A'

INSTRUMENTATION TYPICAL FOR MAIN CONDENSERS

SUCTION TUNNELS

SUCTION TUNNELS

CONDENSER CIRCULATING WATER PUMPS

INTAKE SCREENS

FOR INTAKE DESIGN

FOR INTAKE DESIGN

START & STOP SCREENS

TYPICAL FOR EIGHT SCREENS

SCREEN WASH PUMPS

TO SCREEN WASH

TURBINE BUILDING COOLING WATER PUMPS

RIVER WATER PUMPS

FIRE WATER PUMP

DIESEL DRIVEN FIRE WATER PUMP

TO FIRE SYSTEM

NOTE
DIAGRAM AS SHOWN FOR ARRANGEMENT (A). REQUIREMENTS FOR UNIT (B) IDENTICAL EXCEPT FOR THIS REQUIREMENT OF TWO ADDITIONAL CONDENSERS.

GPO 812-316-12

**FLOW DIAGRAM SERVICE & COOLING WATER SYSTEMS**

SIMILAR TO DRAWING NO. 3250-M-12

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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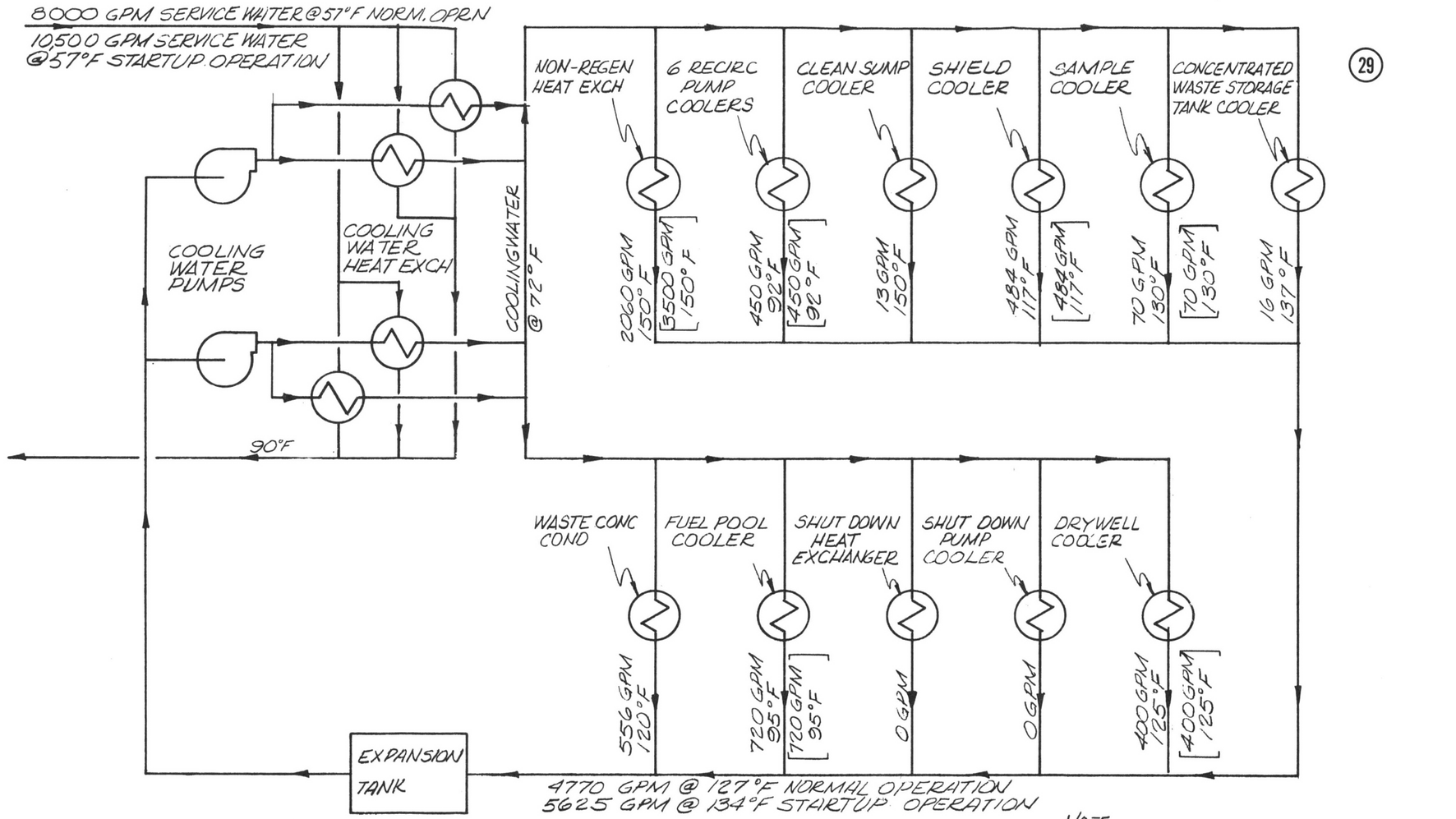
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS




NOTE
BRACKETS INDICATE FLOW REQUIREMENTS DURING STARTUP OPERATION

**NSSS COOLING WATER DIAGRAM**

SIMILAR TO DRAWING NO. 107C4488

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT

SERVICE CONDITIONS @ 57°F
4000 GPM NORMAL
5240 GPM START UP
3920 GPM SHUT DOWN

COOLING WATER PUMPS

COOLING WATER HEAT EXCH

90°F

EXPANSION TANK

2420/127°
2836/134°
4650/100°

COOLING WATER @ 72°F

NON-REGEN. HEAT EXCH.
3 RECIRC. PUMP COOLERS
CLEAN SUMP COOLER
SHIELD COOLER
SAMPLE COOLER
CONCENTRATED WASTE STORAGE TANK COOLER

| | | | | | |
|---|---|---|---|---|---|
| 1030/150° | 225/92° | 13/150° | 244/117° | 57/130° | 10/137° |
| 1750/150° | 225/92° | o/– | 244/117° | 57/130° | o/– |
| -500/112° | o/– | 13/150° | 244/117° | 57/130° | 10/137° |

SERVICE:
NORMAL
START UP
SHUT DOWN

WASTE CONC. CONDENSER
FUEL POOL COOLER
SHUT DOWN HEAT EXCHANGER
SHUT DOWN PUMP COOLER
DRY WELL COOLER

| | | | | |
|---|---|---|---|---|
| 280/120° | 360/95° | o/– | o/– | 200/125° |
| o/– | 360/95° | o/– | o/– | 200/125° |
| o/– | o/– | 3600/95° | 45/112° | 200/125° |

SERVICE:
NORMAL
START UP
SHUT DOWN

LEGEND
GPM/TEMP°F

NOTE:
THIS SYSTEM SERVES ONE REACTOR (500 MW) OF A TWO REACTOR SYSTEM

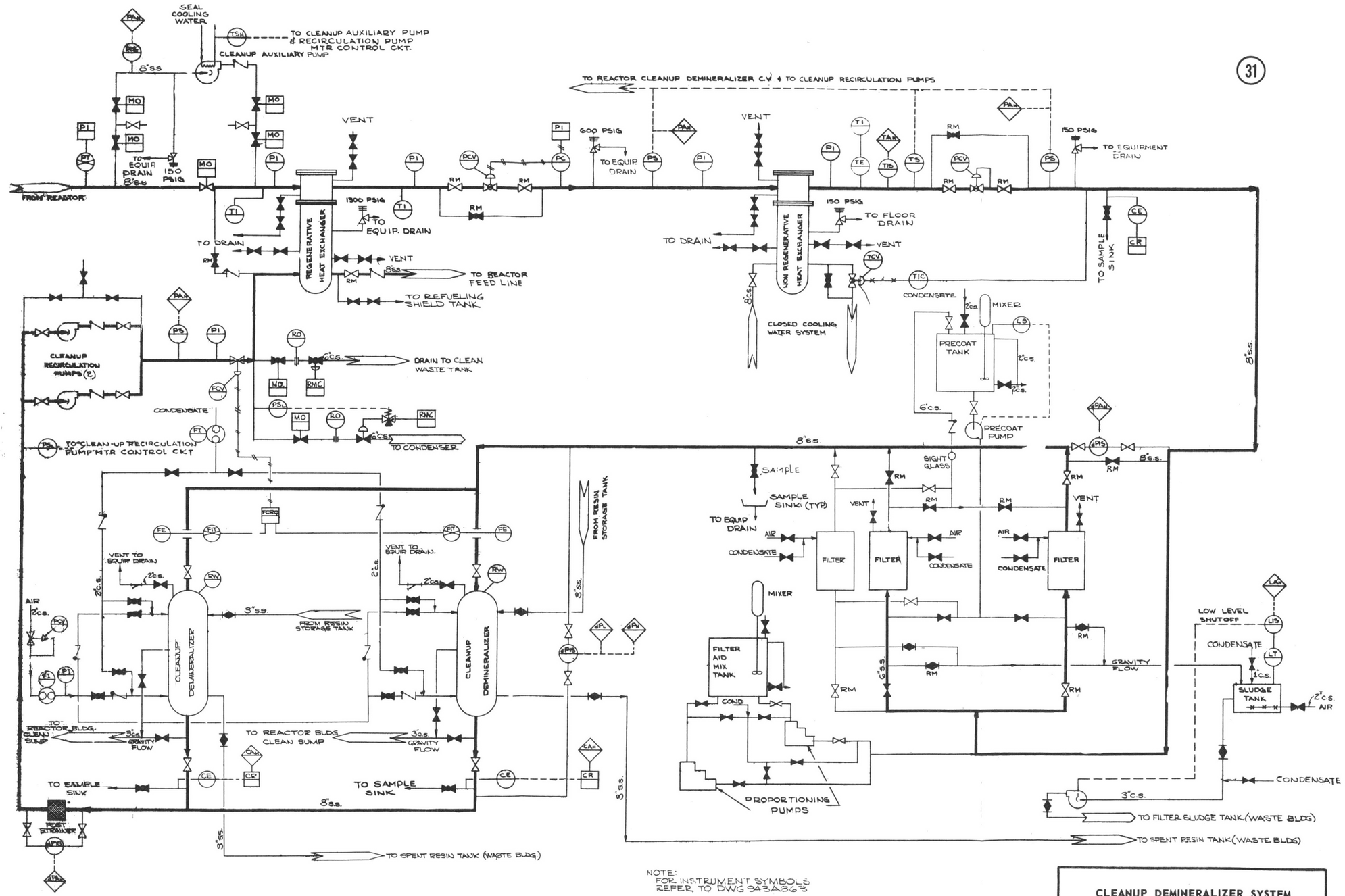
**NSSS COOLING WATER DIAGRAM**

SIMILAR TO DRAWING NO. 107C4619

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




NOTE:
FOR INSTRUMENT SYMBOLS REFER TO DWG 943A863

**CLEANUP DEMINERALIZER SYSTEM**

SIMILAR TO DRAWING NO. 148F226

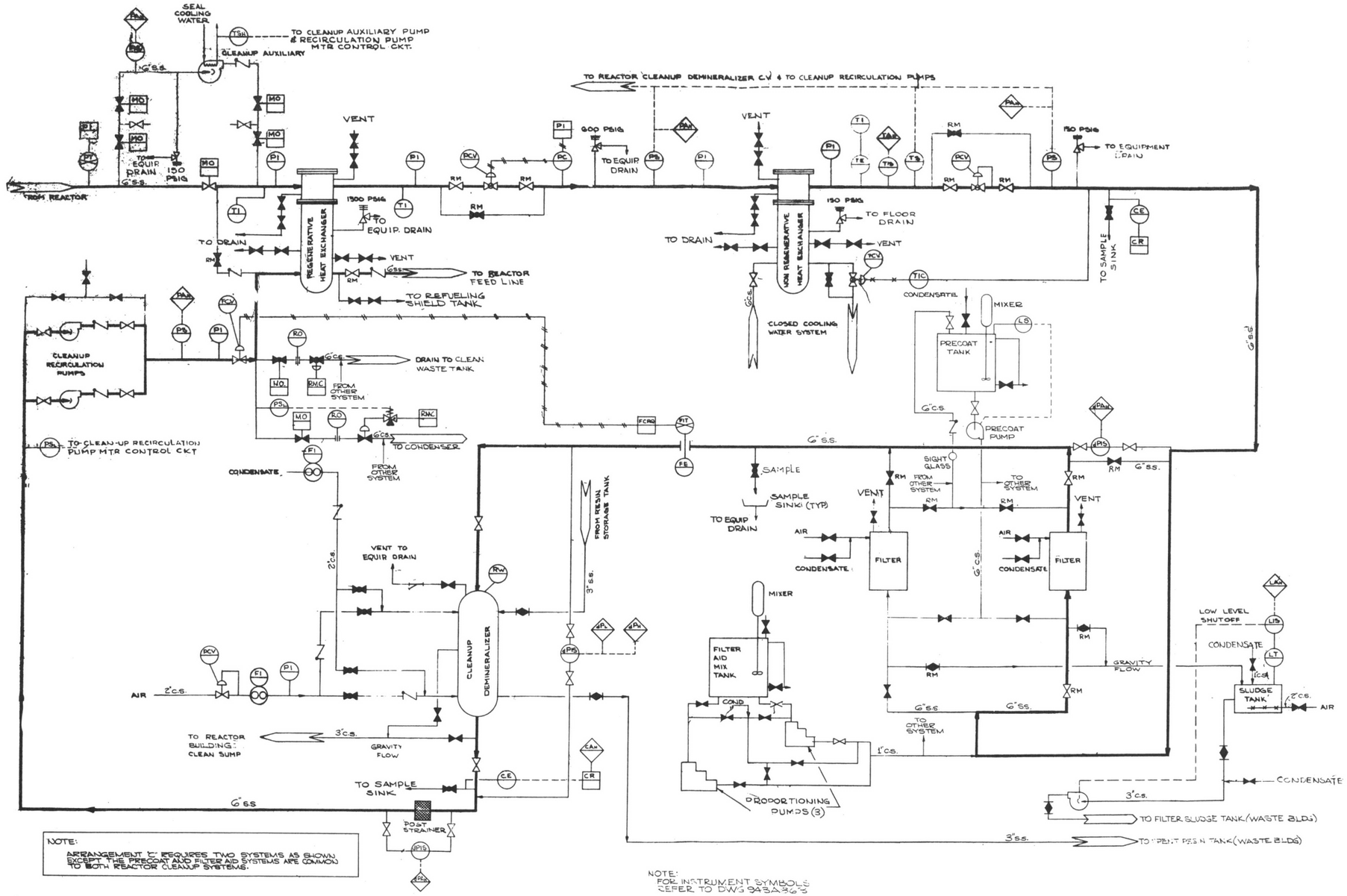
1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT




NOTE:
ARRANGEMENT 'C' REQUIRES TWO SYSTEMS AS SHOWN EXCEPT THE PRECOAT AND FILTER AID SYSTEMS ARE COMMON TO BOTH REACTOR CLEANUP SYSTEMS.

NOTE:
FOR INSTRUMENT SYMBOLS REFER TO DWG 943A363

GPO 812-316-13

**CLEANUP DEMINERALIZER SYSTEM**

SIMILAR TO DRAWING NO. 148F227

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**

FROM CONCENTRATED ACID STORAGE
2"c.s.
REGENERANT PROPORTIONING PUMPS
FROM CONCENTRATED CAUSTIC STORAGE
2"c.s.
2"s.s.
2"c.s.
2"c.s.
FI
RO
FI
RO
CI
CI
TO CLEAN-UP DEMIN.
TO WASTE DEMIN.
2"c.s. TO RESIN DISPOSAL
3"s.s.
3"s.s.
3"s.s.
REGENERATED RESIN. STORAGE TANK
AIR
FI
2"c.s.
FI
3"c.s.
3"c.s.
2"c.s.
2"c.s.
WATER
2"c.s.
ANION RESIN REGENERATION TANK
AIR
FI
2"c.s.
RO
3"c.s.
FI
3"s.s.
WASTE
FROM WASTE DEMIN.
3"s.s.
FROM CLEAN-UP DEMIN.
3"s.s.
RESIN
3"s.s.
2"s.s.
FROM CONDENSATE PUMPS
TIA
QRA
FRO
VENT
2"c.s.
2"c.s.
10"c.s.
AIR
FI
TYPICAL- FOR 8
DEMINERALIZER
dPI
RESIN
WATER
TO RESIN DISPOSAL
10"c.s.
CR
STRAINER
dPI
RECYCLE PUMP
RECYCLE
TO REACTOR FEED PUMPS
EFFLUENT
TO RADIOACTIVE WASTE 3"c.s.
WATER
CATION RESIN REGENERATION TANK
3"s.s.
RO
FI
3"s.s.
2"c.s.
FI
AIR
3"c.s.
3"s.s.
3"s.s.
2"c.s.
CR
WATER 2"c.s.
FROM CONDENSATE STORAGE
3"c.s.
FI
3"s.s.
3"s.s.
WASTE (REGENERANTS)
WASTE (RINSE & SLUICE)
2"c.s.

TWO SYSTEMS AS SHOWN ARE REQUIRED FOR EACH ARRANGEMENT

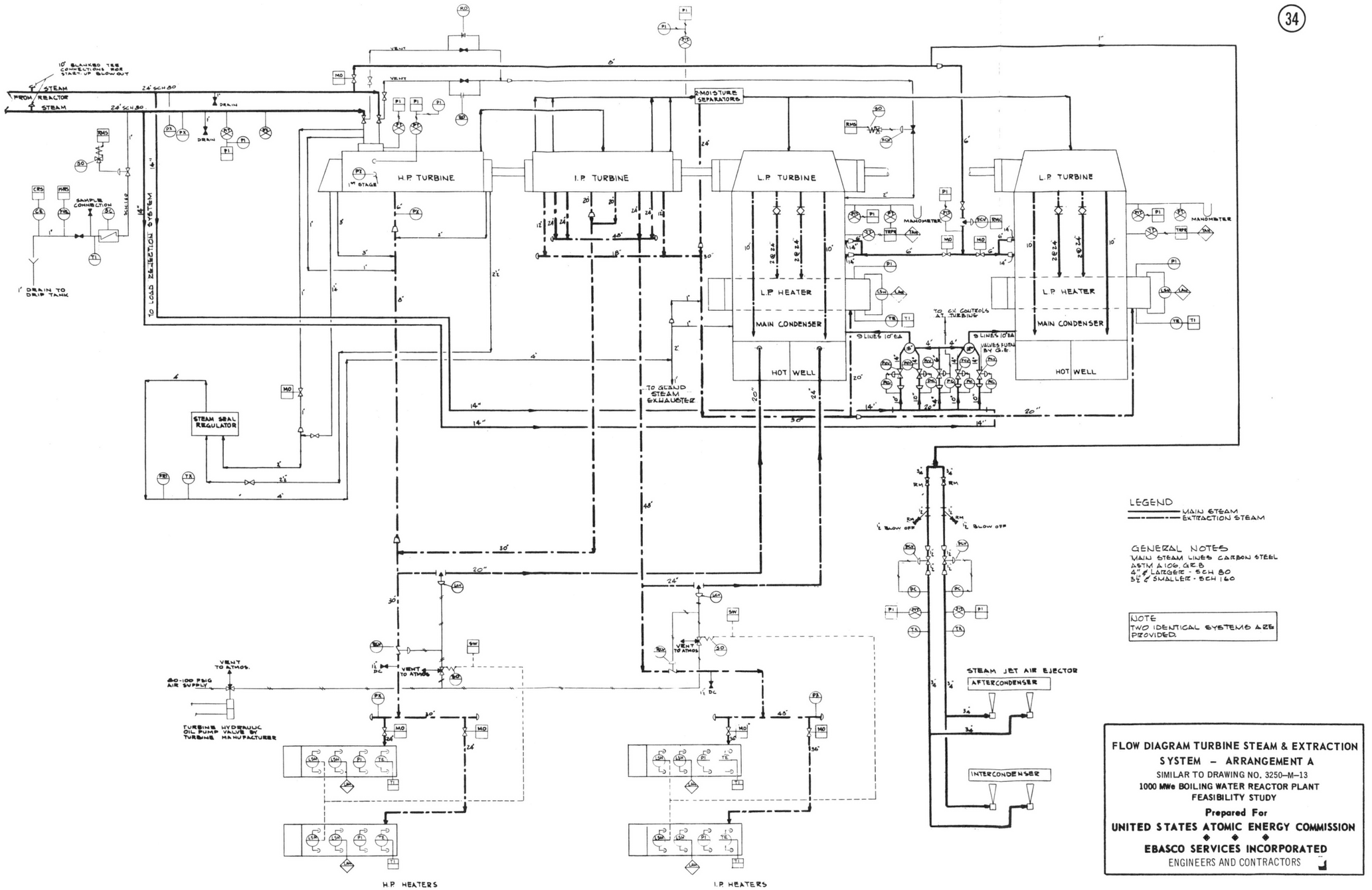
**FLOW DIAGRAM CONDENSATE DEMINERALIZER**

SIMILAR TO DRAWING NO. 798D628

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




LEGEND

MAIN STEAM

EXTRACTION STEAM

GENERAL NOTES

MAIN STEAM LINES CARBON STEEL
ASTM A 106 GR B
4" ϕ & LARGER - SCH 80
3½" ϕ & SMALLER - SCH 160

NOTE

TWO IDENTICAL SYSTEMS ARE PROVIDED.

**FLOW DIAGRAM TURBINE STEAM & EXTRACTION SYSTEM – ARRANGEMENT A**

SIMILAR TO DRAWING NO. 3250–M–13

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS

10" BLANKED TEE CONNECTION & BOIL STRET-UP BLOWOUT
FROM REACTOR SYSTEM
STEAM
24" SCH 80
VENT
2-MOISTURE SEPARATORS
2-TWO STAGE STEAM REHEATERS
H.P. TURBINE
1ST STAGE
L.P. TURBINE
MANOMETER
L.P. HEATER
MAIN CONDENSER
HOT WELL
TO C.W. CONTROL & TURBINE
18" HEADER
20" HEADER
2½" TO GLAND STEAM EXHAUSTER
TO LOAD REJECTION SYSTEM
SAMPLE CONNECTION
DRAIN TO DRIP TANK
STEAM SEAL REGULATOR
VENT TO ATMOS
80-100 PSIG AIR SUPPLY
TURBINE HYDRAULIC OIL PUMP VALVE BY TURBINE MANUFACTURER
H.P. HEATERS
I.P. HEATERS
STEAM JET AIR EJECTOR
AFTER CONDENSER
INTER CONDENSER
BLOW OFF

LEGEND
- MAIN STEAM
- EXTRACTION STEAM
- REHEAT STEAM

GENERAL NOTES:
- MAIN STEAM LINES: C.S. ASTM A 106, GR B
  4" & LARGER - SCH 80
  3" & SMALLER - SCH 160

NOTE
TWO IDENTICAL SYSTEMS ARE PROVIDED

# FLOW DIAGRAM TURBINE STEAM & EXTRACTION SYSTEM – ARRANGEMENT B

SIMILAR TO DRAWING NO. 3250–M–33

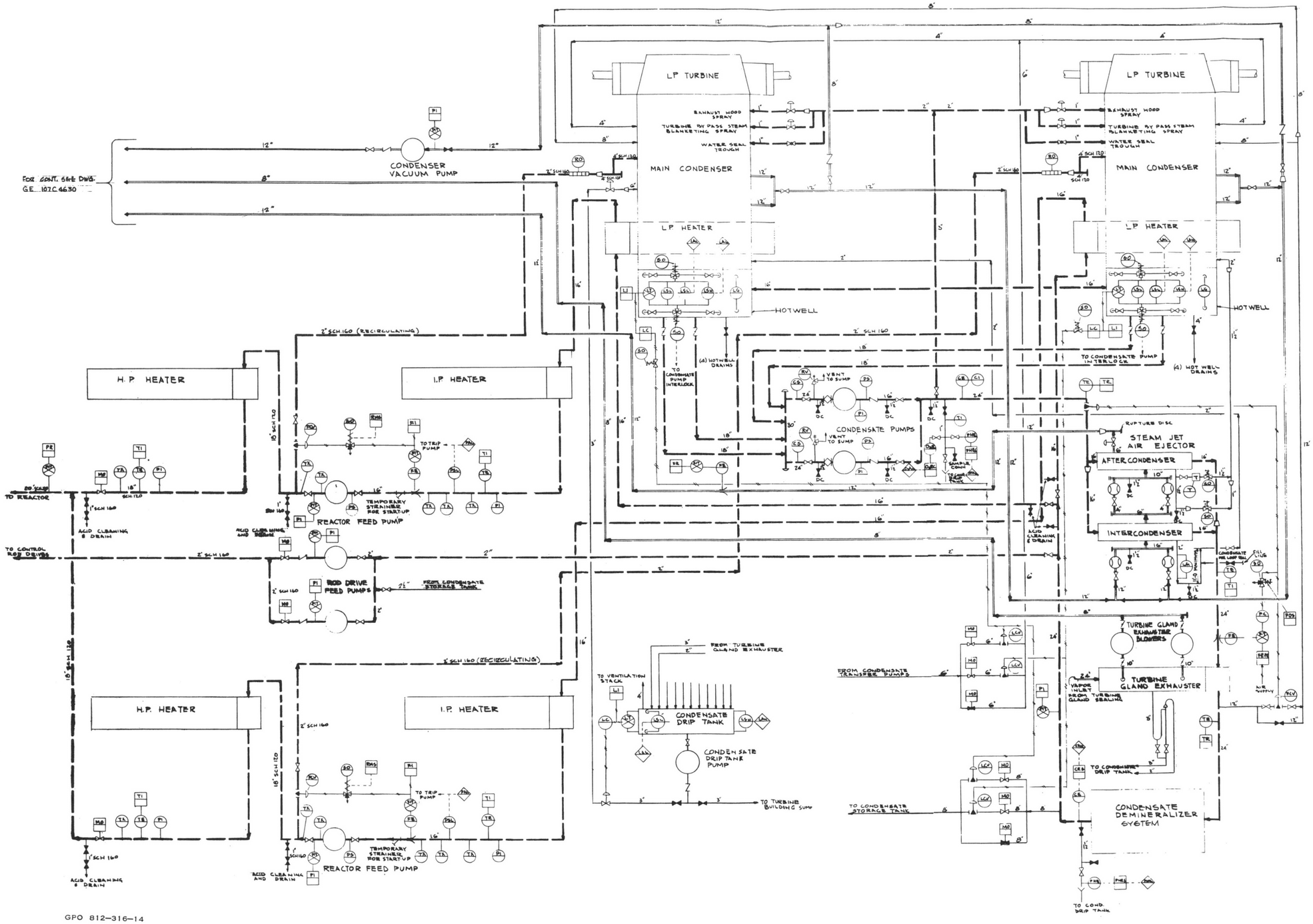
1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS




LEGEND
CONDENSATE
AIR EVACUATION

GENERAL NOTES:
ALL PIPING TO BE OF THE FOLLOWING SCHEDULE AND RATING UNLESS OTHERWISE NOTED:
- UP STREAM OF ROD DRIVE FEED PUMPS SCH 40 C.STL AND 150# RATING
- UP STREAM OF REACTOR FEED PUMPS SCH 40 C.STL AND 150# RATING
- DOWN STREAM OF ROD DRIVE FEED PUMPS SCH 120 AND OR SCH 160 AND 900# RATING
- DOWN STREAM OF REACTOR FEED PUMPS SCH 120 AND OR SCH 160 AND 600# RATING

NOTE
TWO IDENTICAL SYSTEMS ARE PROVIDED.

**FLOW DIAGRAM CONDENSATE & FEEDWATER SYSTEM – ARRANGEMENT A**

SIMILAR TO DRAWING NO. 3250–M–14

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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**EBASCO SERVICES INCORPORATED**

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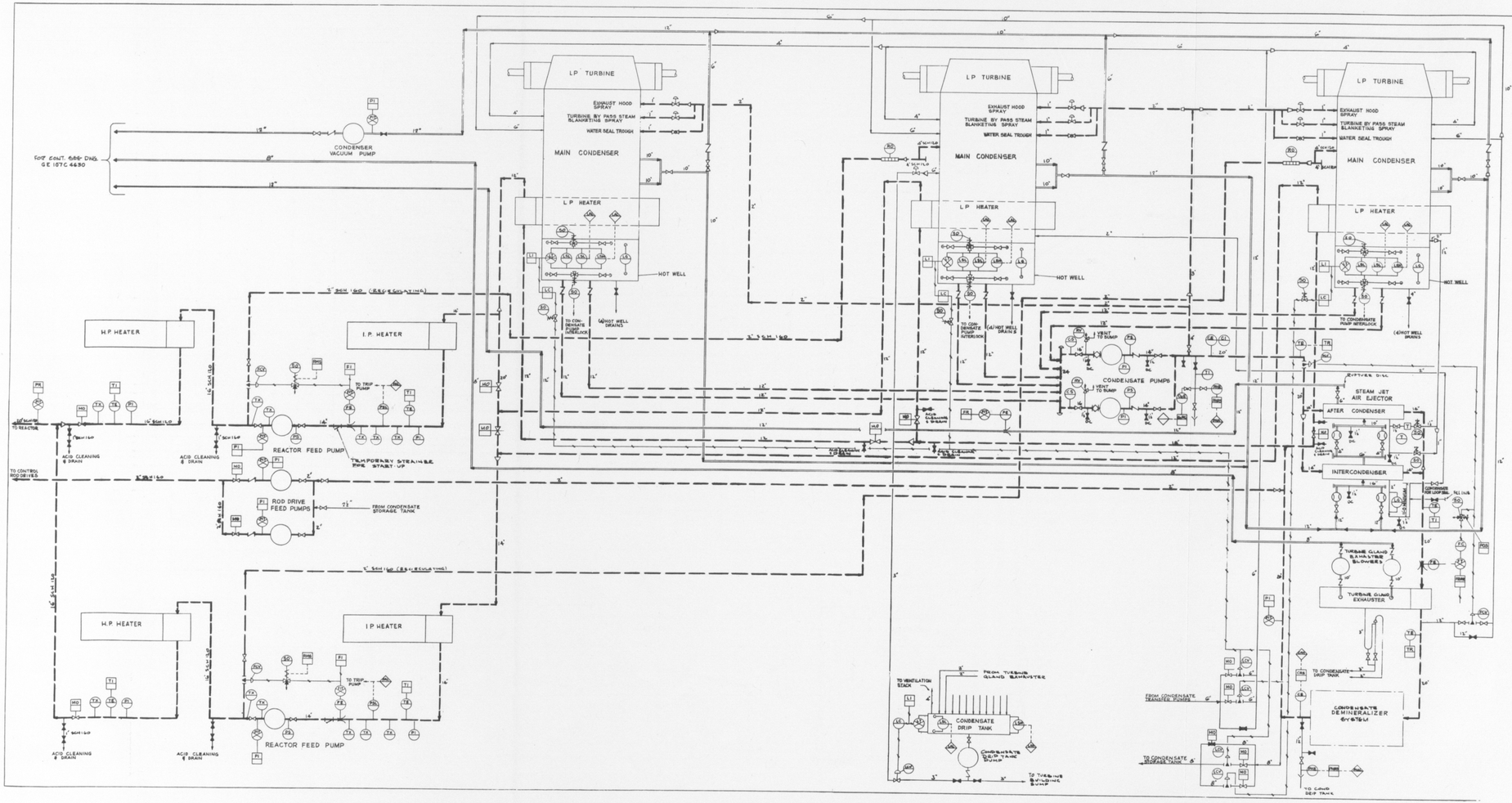
GPO 812–316–14




LEGEND
CONDENSATE
AIR EVACUATION

GENERAL NOTES
ALL PIPING TO BE OF THE FOLLOWING SCHEDULE AND RATING UNLESS OTHERWISE NOTED
- HOT STREAM OF FEED PUMPS UP TO AND INCL. 4" AND 150# RATING
- HOT STREAM OF REACTOR FEED PUMPS SCH 40 CARBON STEEL
- DOWN STREAM OF REACTOR FEED PUMPS SCH 160 AND 300# RATING
- DOWN STREAM OF REACTOR FEED PUMPS SCH 160 AND 600# RATING

NOTE:
TWO IDENTICAL SYSTEMS ARE PROVIDED.

**FLOW DIAGRAM CONDENSATE & FEEDWATER SYSTEM – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 3250–M–34
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

*Prepared For*

**UNITED STATES ATOMIC ENERGY COMMISSION**

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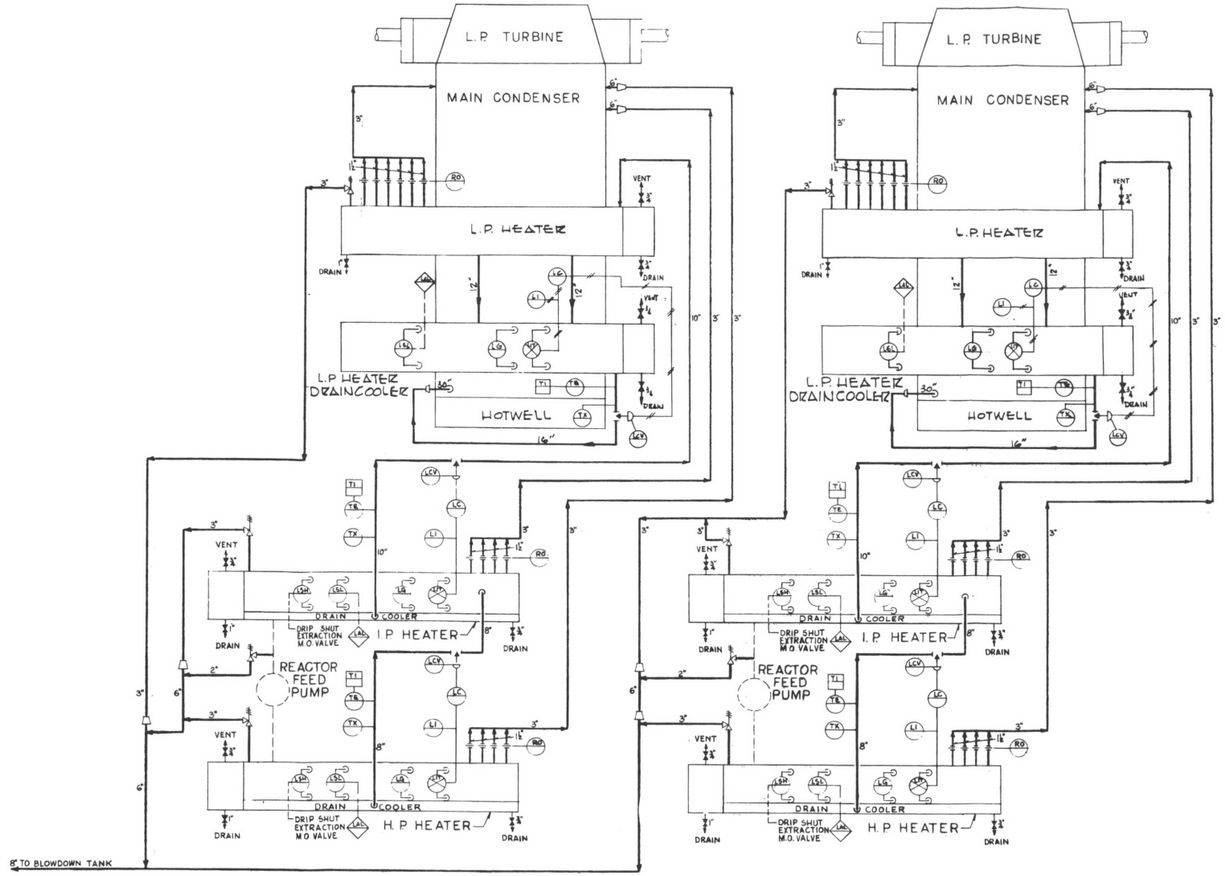
**EBASCO SERVICES INCORPORATED**
NUCLEAR ENGINEERING DEPARTMENT




NOTE:
ALL HEATER & DRAIN COOLER VENTS CONNECT TO EXHAUST NECK. ALL HEATER & DRAIN COOLERS VENT TO HOT WELL.

**FLOW DIAGRAM HEATER DRIPS, DRAINS & VENTS – ARRANGEMENT A**

SIMILAR TO DRAWING NO. 3250–M–16

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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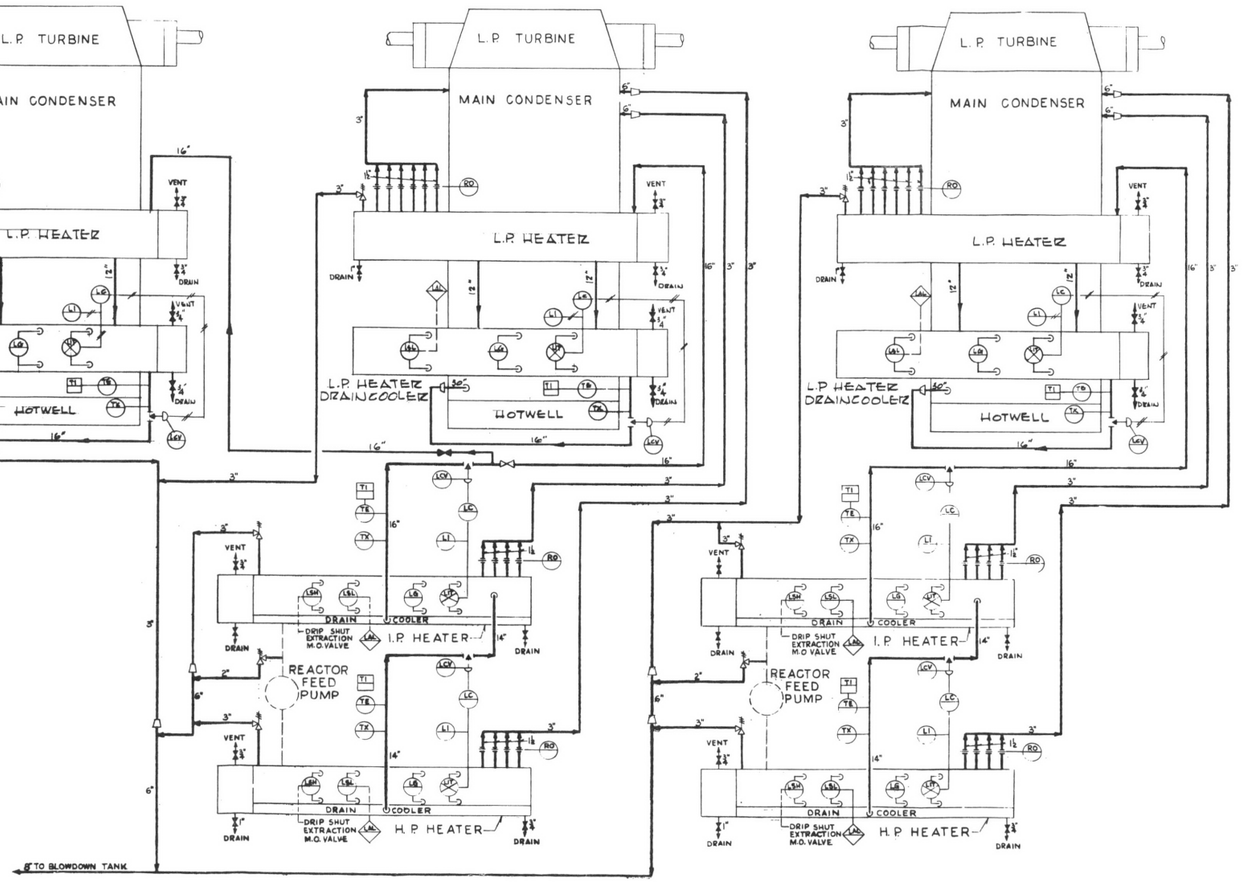
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS




NOTE:
ALL HEATER & DRAIN COOLER VENTS CONNECT TO EXHAUST NECK. ALL HEATER & DRAINCOOLERS VENT TO HOT WELL.

**FLOW DIAGRAM HEATER DRIPS, DRAINS & VENTS – ARRANGEMENT B**

SIMILAR TO DRAWING NO. 3250–M–36

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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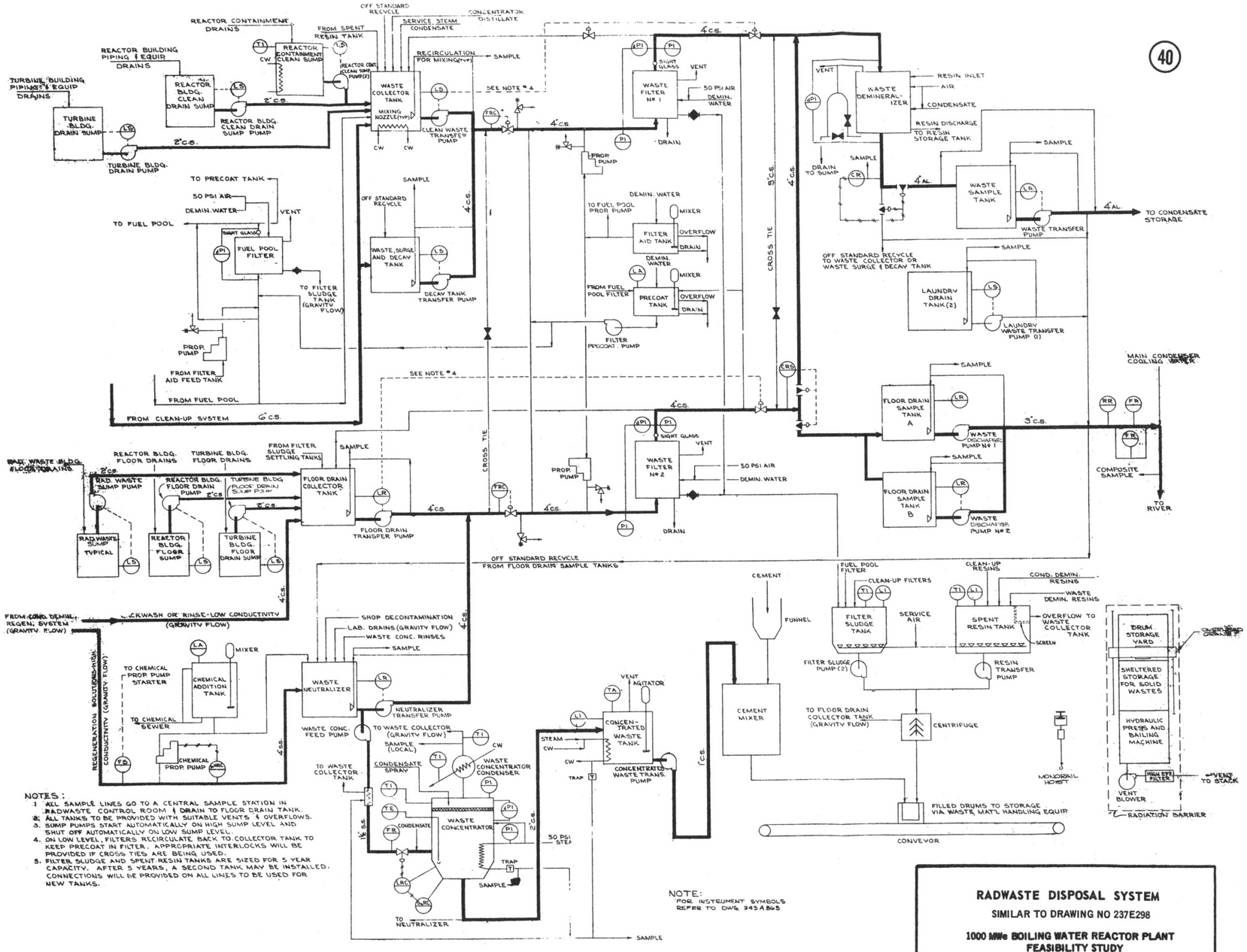
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS




NOTES:

1. ALL SAMPLE LINES GO TO A CENTRAL SAMPLE STATION IN RADWASTE CONTROL ROOM & DRAIN TO FLOOR DRAIN TANK.
2. ALL TANKS TO BE PROVIDED WITH SUITABLE VENTS & OVERFLOWS.
3. SUMP PUMPS START AUTOMATICALLY ON HIGH SUMP LEVEL AND SHUT OFF AUTOMATICALLY ON LOW SUMP LEVEL.
4. ON LOW LEVEL, FILTERS RECIRCULATE BACK TO COLLECTOR TANK TO KEEP PRECOAT IN FILTER. APPROPRIATE INTERLOCKS WILL BE PROVIDED IF CROSS TIES ARE BEING USED.
5. FILTER SLUDGE AND SPENT RESIN TANKS ARE SIZED FOR 5 YEAR CAPACITY. AFTER 5 YEARS, A SECOND TANK MAY BE INSTALLED. CONNECTIONS WILL BE PROVIDED ON ALL LINES TO BE USED FOR NEW TANKS.

NOTE:
FOR INSTRUMENT SYMBOLS REFER TO DWG 943A865

**RADWASTE DISPOSAL SYSTEM**

SIMILAR TO DRAWING NO 237E298

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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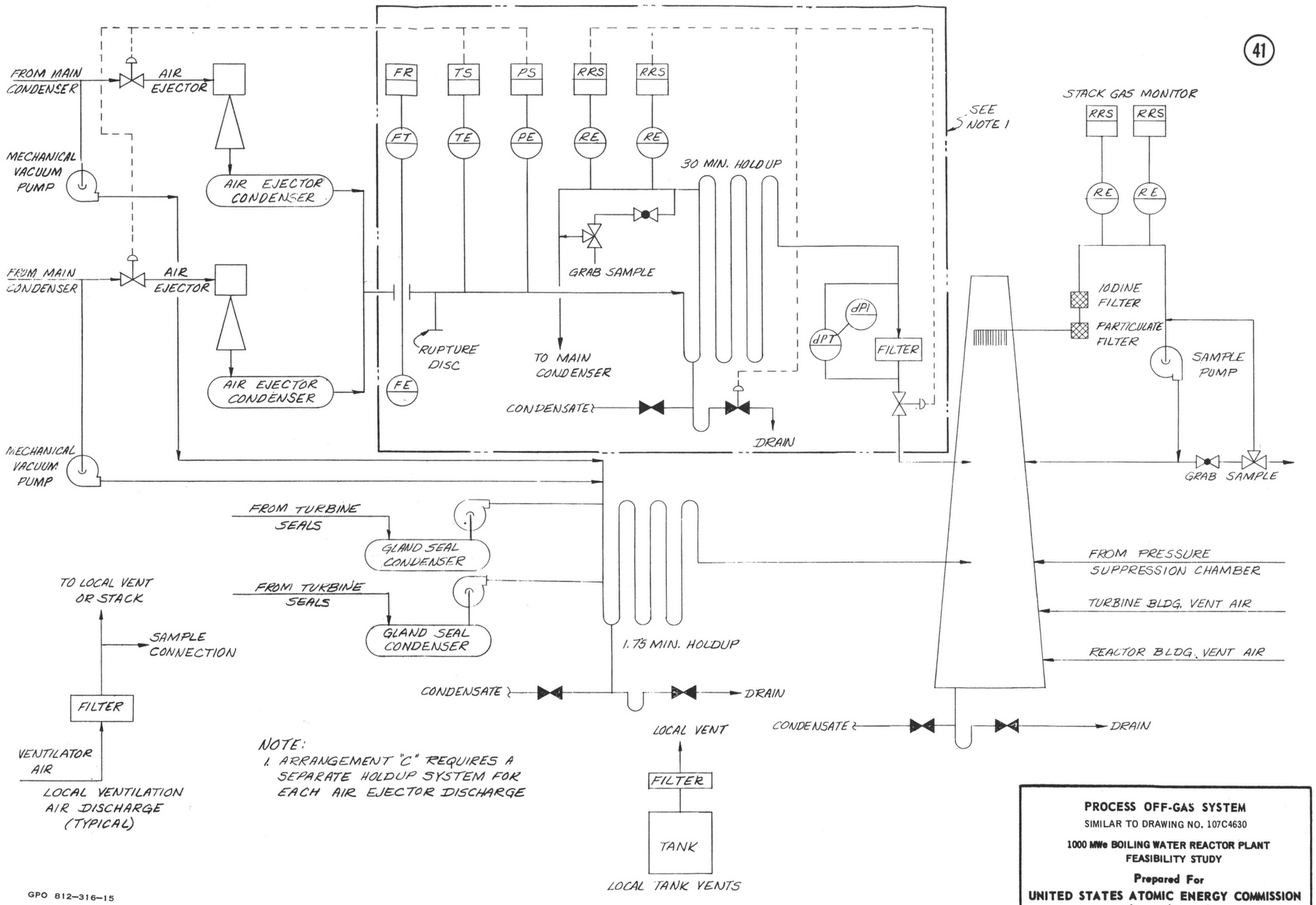
**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**




NOTE:
1. ARRANGEMENT "C" REQUIRES A SEPARATE HOLDUP SYSTEM FOR EACH AIR EJECTOR DISCHARGE

GPO 812-316-15

**PROCESS OFF-GAS SYSTEM**

SIMILAR TO DRAWING NO. 107C4630

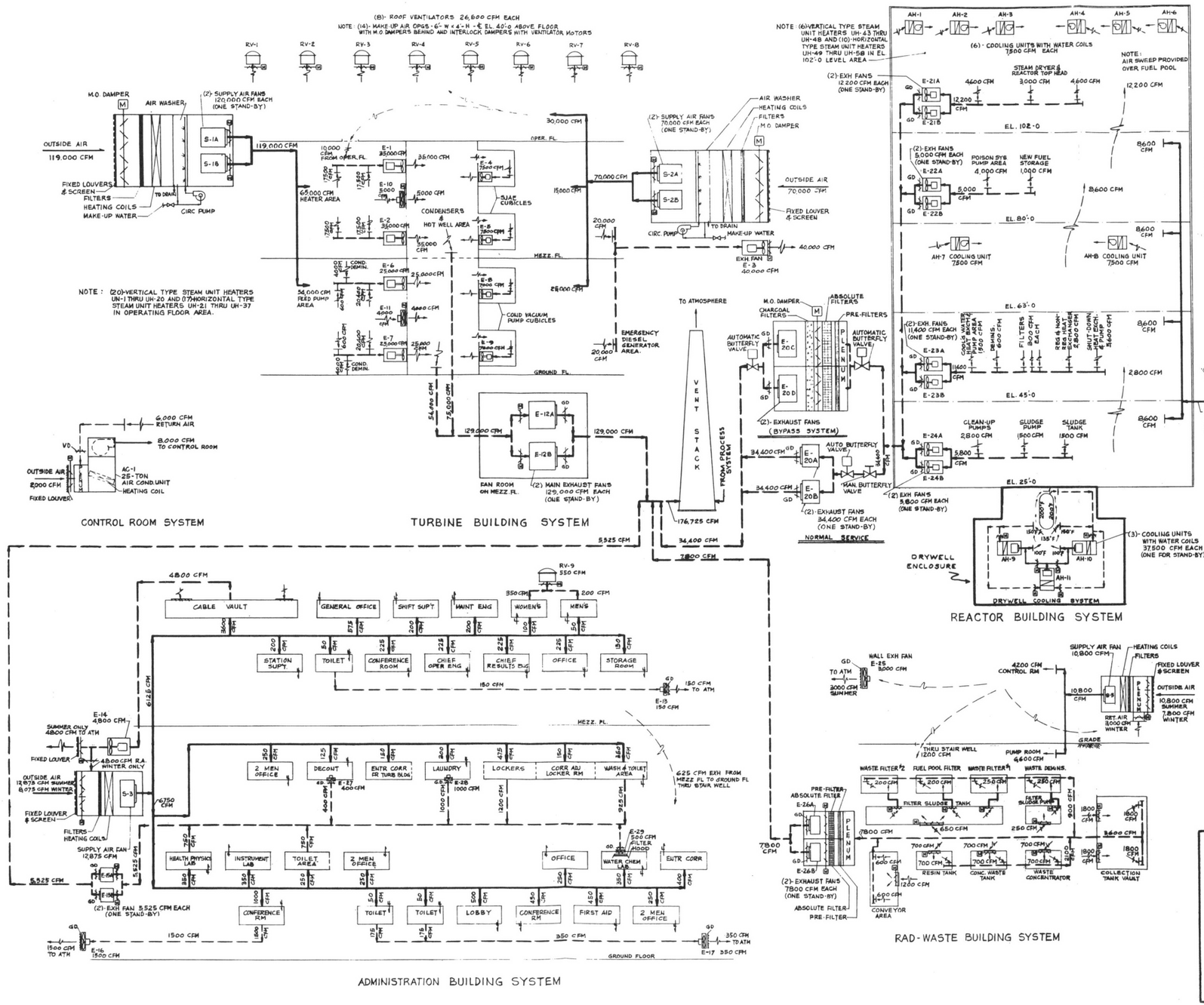
**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**




NOTE: THIS SCHEMATIC DIAGRAM IS PREPARED FOR ARRANGEMENT (A) AS SHOWN, ARRANGEMENT (B) SIMILAR.

**FLOW DIAGRAM HEATING, VENT. & AIR CONDITION SYSTEMS**

SIMILAR TO DRAWING NO. 3250-H&V-1

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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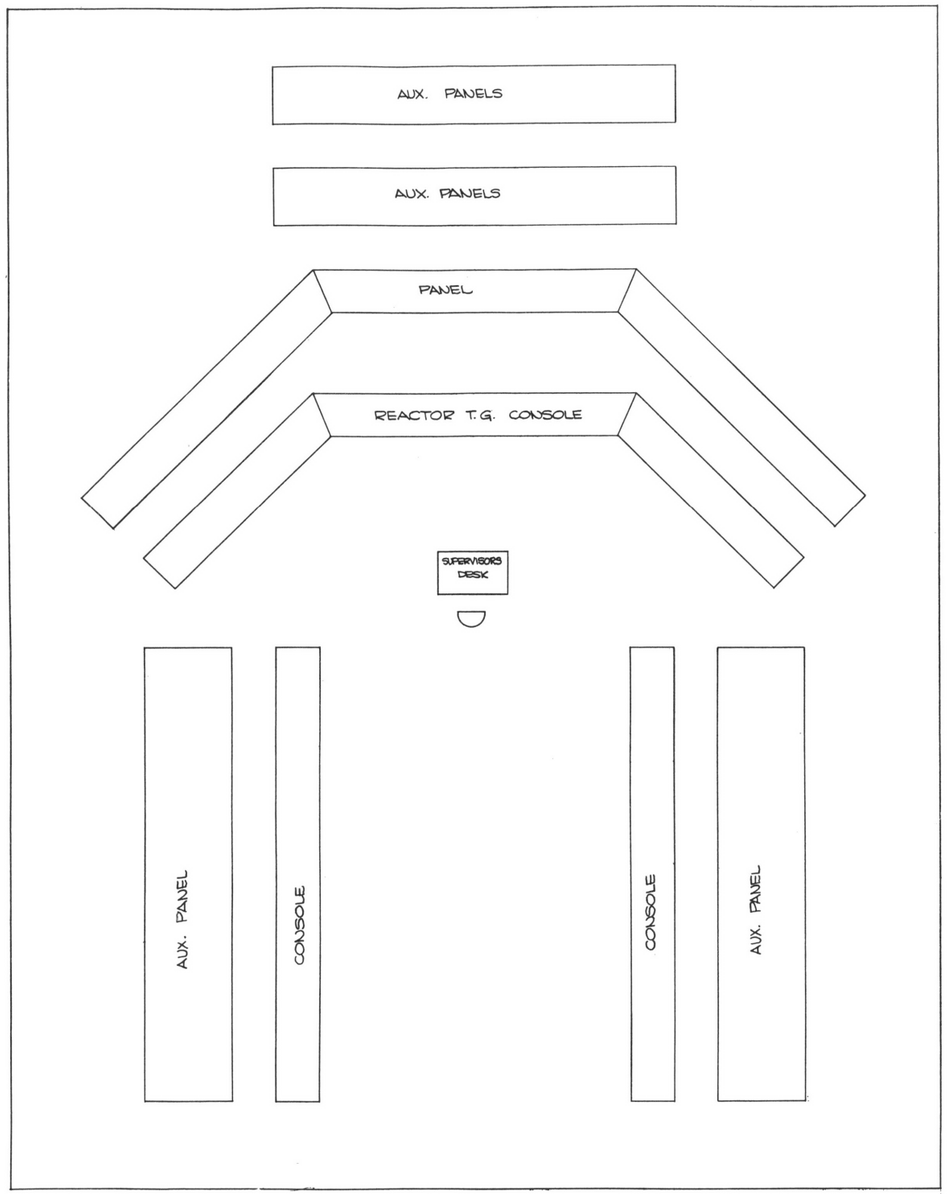
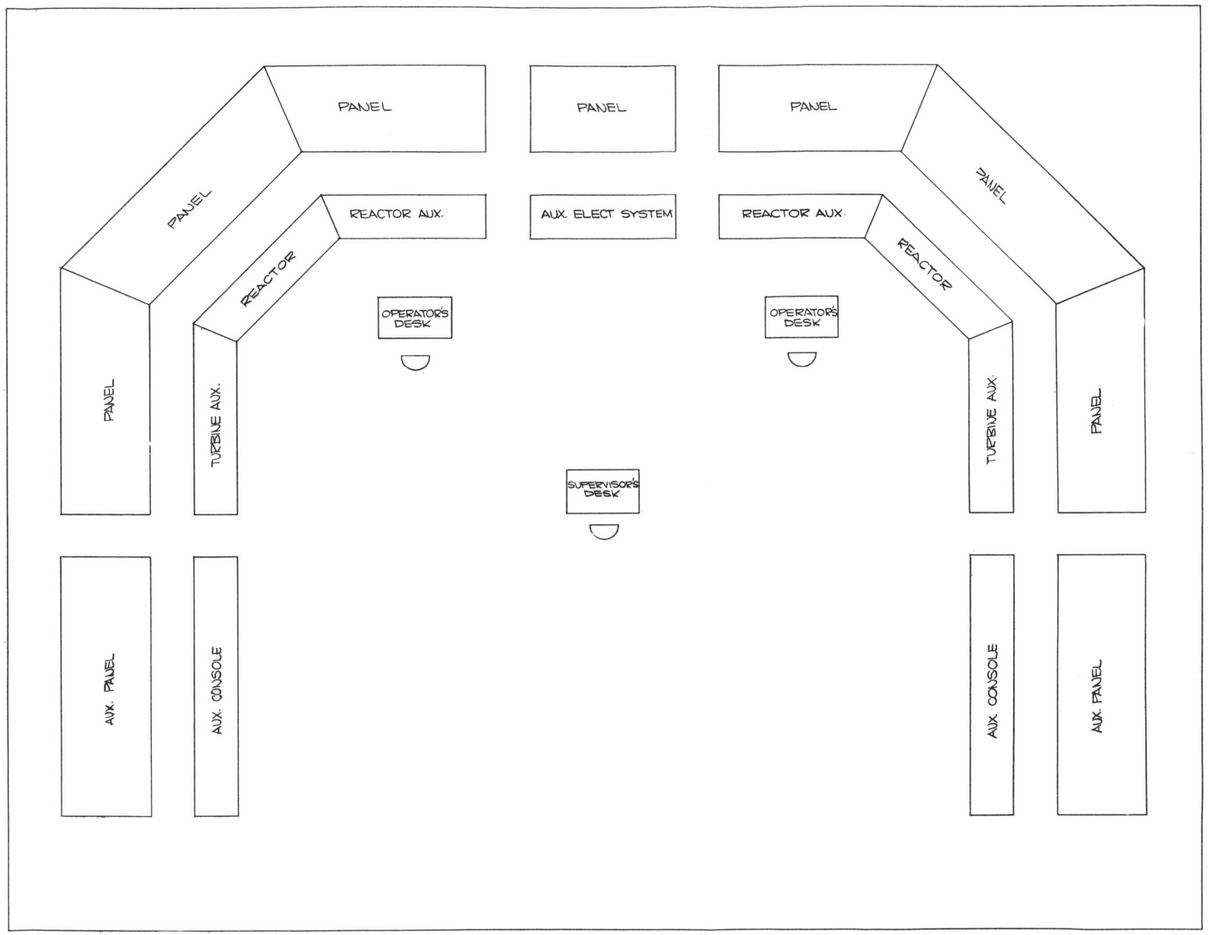
**EBASCO SERVICES INCORPORATED**

ENGINEERS AND CONTRACTORS




ARRANGEMENT "A"




ARRANGEMENT "C"

**CONTROL ROOM PANEL ARRANGEMENT**

SIMILAR TO DRAWING NO. 237E136

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

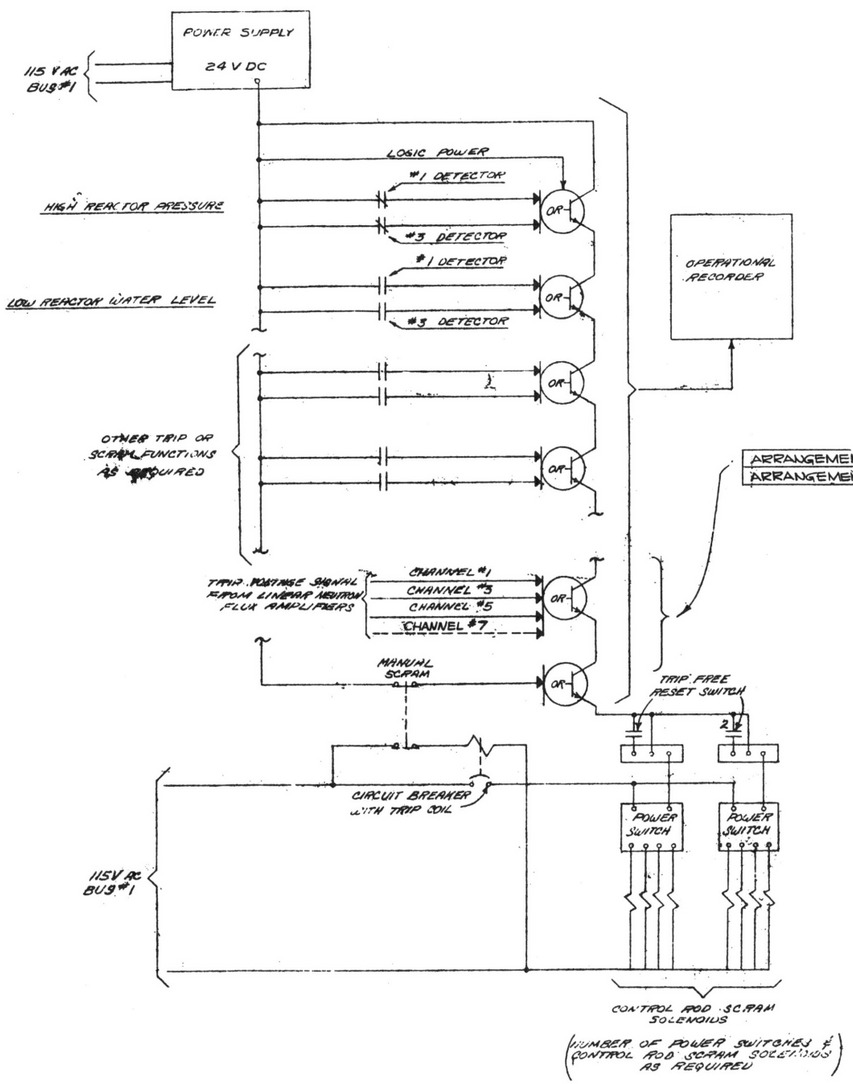
Prepared For

UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC

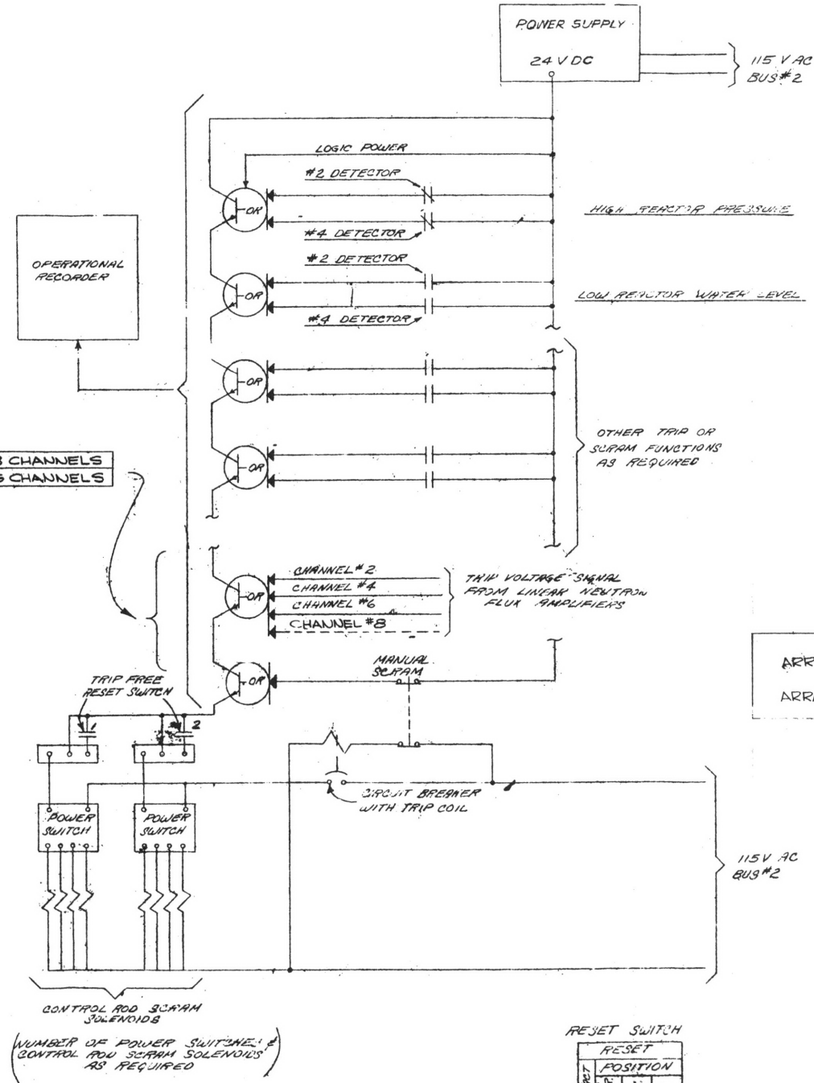
ATOMIC POWER EQUIPMENT DEPARTMENT

REACTOR PROTECTION CHANNEL #1




| | |
|---|---|
| ARRANGEMENT "A" | 8 CHANNELS |
| ARRANGEMENT "C" | 6 CHANNELS |

REACTOR PROTECTION CHANNEL #2




DEFINITION OF LOGIC ELEMENTS

"OR" LOGIC - A TRIP OUTPUT SIGNAL IS DERIVED FROM THE PRESENCE OF A TRIP AT EITHER INPUT OR AT BOTH INPUTS.

* TRIP SIGNAL IS INDICATED BY ZERO VOLTAGE

ARRANGEMENT "A" REQUIRES ONE SYSTEM AS SHOWN.

ARRANGEMENT "C" REQUIRES TWO SYSTEMS AS SHOWN.

RESET SWITCH

| | RESET POSITION | |
|---|---|---|
| CHANNEL | 1 | 2 |
| 1 | X | |
| 2 | | X |

**REACTOR PROTECTION SYSTEM**

SIMILAR TO DRAWING NO. 762D755

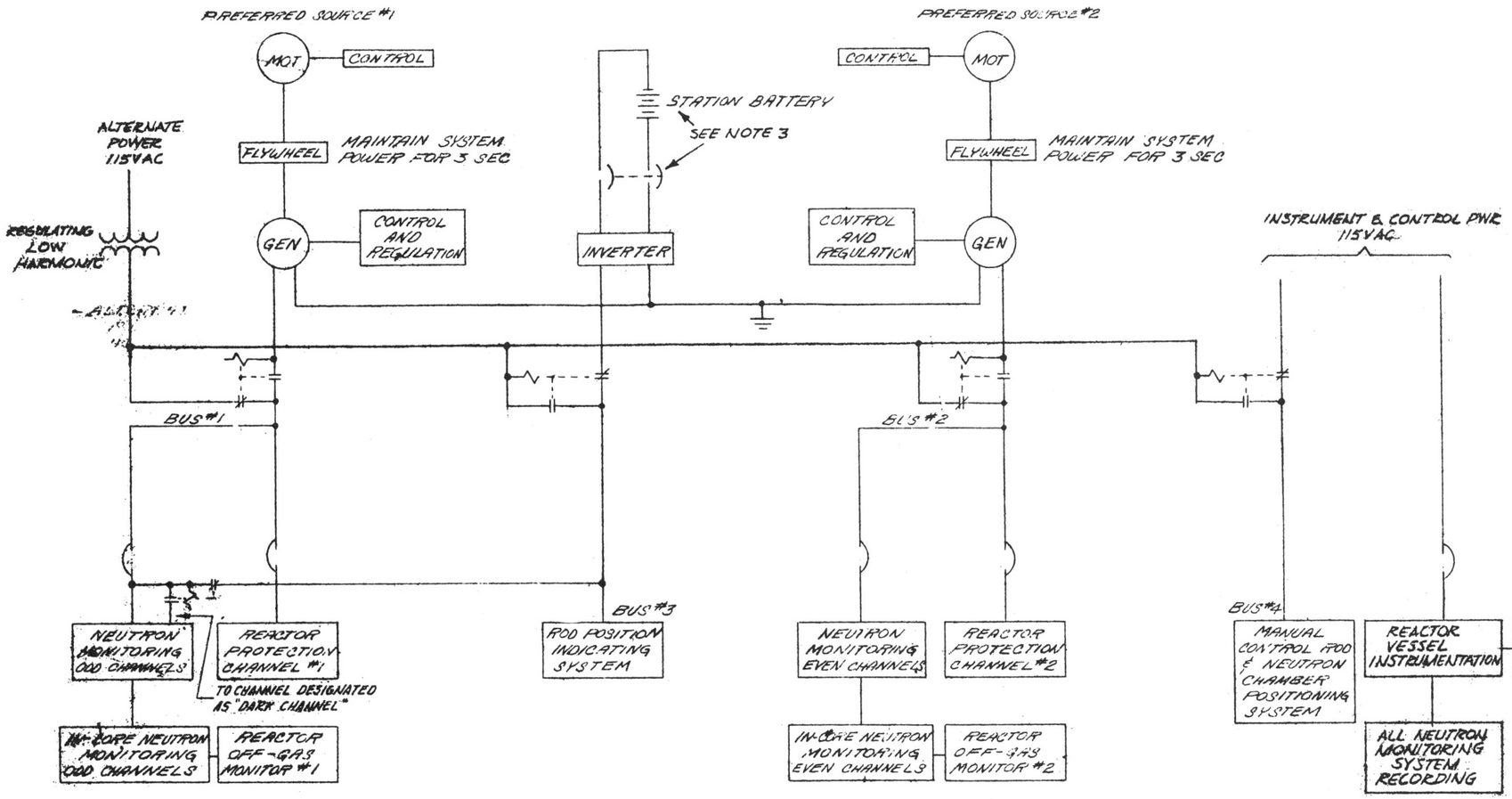
**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

**GENERAL ELECTRIC**

**ATOMIC POWER EQUIPMENT DEPARTMENT**




NOTES:

1. BUSES 1,2,3 & 4 AND INST. & CONTROL CIRCUITS SHOWN ARE RESERVED SOLELY FOR THE INDICATED SYSTEMS.
2. BUSES 1,2 & 3 SHALL BE PROVIDED WITH ONE SYSTEM GROUND. THIS GROUND SHALL BE COMMON WITH GROUNDING POINTS OF ALL PANELS AND THE STATION GROUND.
3. NUCLEAR STEAM SUPPLY INCLUDES ALL DEVICES AND SYSTEM SHOWN EXCEPT STATION BATTERY AND ASSOCIATED BREAKER.

ARRANGEMENT "A" REQUIRES ONE SYSTEM AS SHOWN
ARRANGEMENT "C" REQUIRES TWO SYSTEMS AS SHOWN.

GPO 812–316–16

**PROTECTION CONTROL SYSTEM – POWER SUPPLY**

SIMILAR TO DRAWING NO. 107C4627
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

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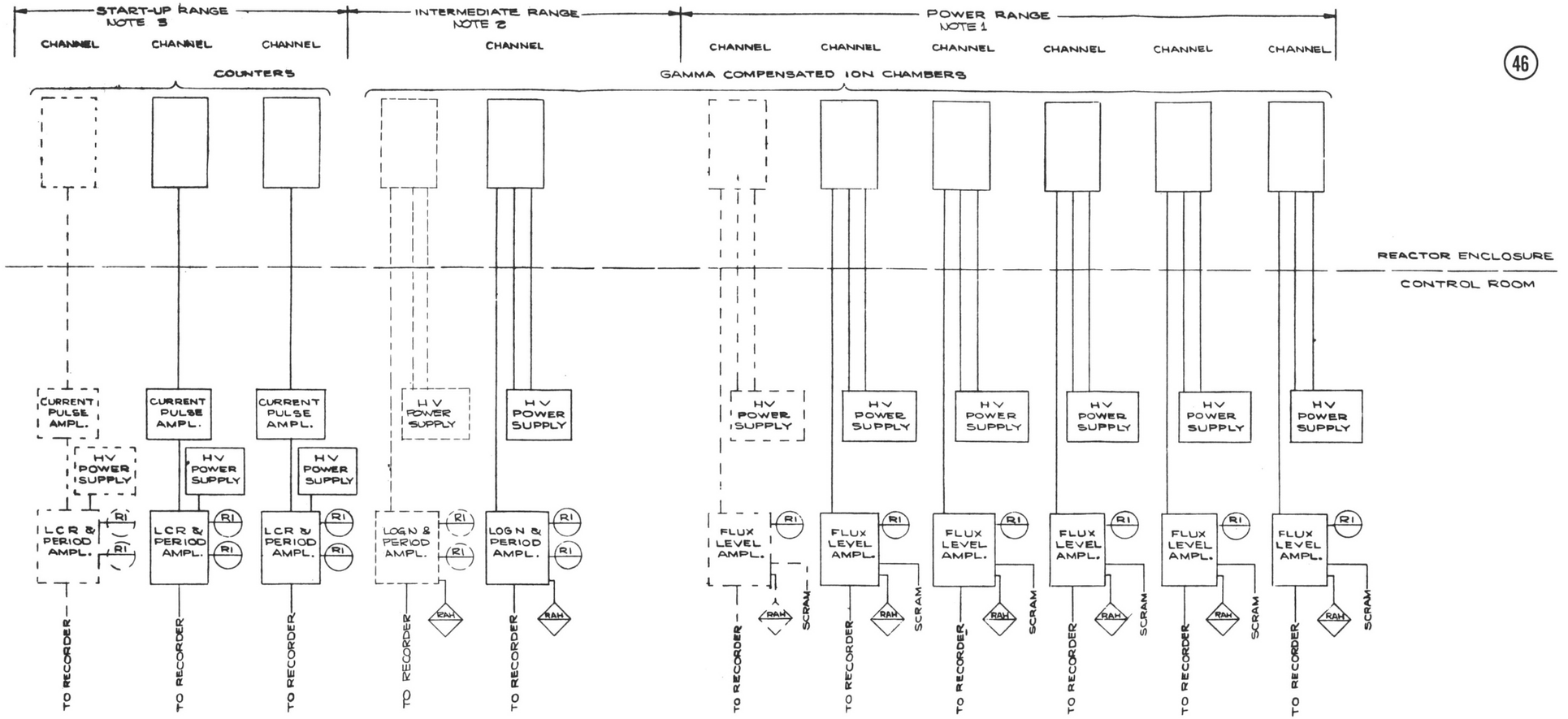
GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




SYMBOLS:

RI = RADIATION INDICATOR
RAH = RADIATION ALARM ON HIGH LEVEL

NOTES:

1. THESE CHANNELS COVER THE POWER RANGE ALARM ON EXCESSIVE NEUTRON FLUX AND SCRAM WHEN ONE ODD NUMBERED CHANNEL AND ONE EVEN NUMBERED CHANNEL INDICATES AN OVER FLUX.
2. THESE CHANNELS ARE FOR INFORMATION ONLY TO SHOW REACTOR POWER DURING START-UP AND TO MONITOR REACTOR PERIOD.
3. THESE CHANNELS COVER THE START-UP RANGE AND ALARM ON APPROACH TO A SHORT PERIOD.

| | NUMBER OF CHANNELS PER REACTOR | | |
|---|---|---|---|
| | START-UP RANGE | INTERMEDIATE RANGE | POWER RANGE |
| ARRANGEMENT "A" | 6 | 2 | 8 |
| ARRANGEMENT "C" | 4 | 1 | 6 |

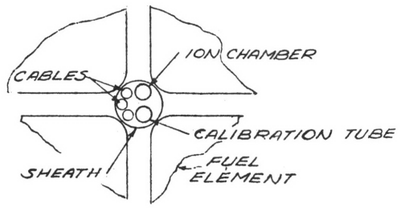
**NEUTRON MONITOR SYSTEM**

SIMILAR TO DRAWING NO. 762D750

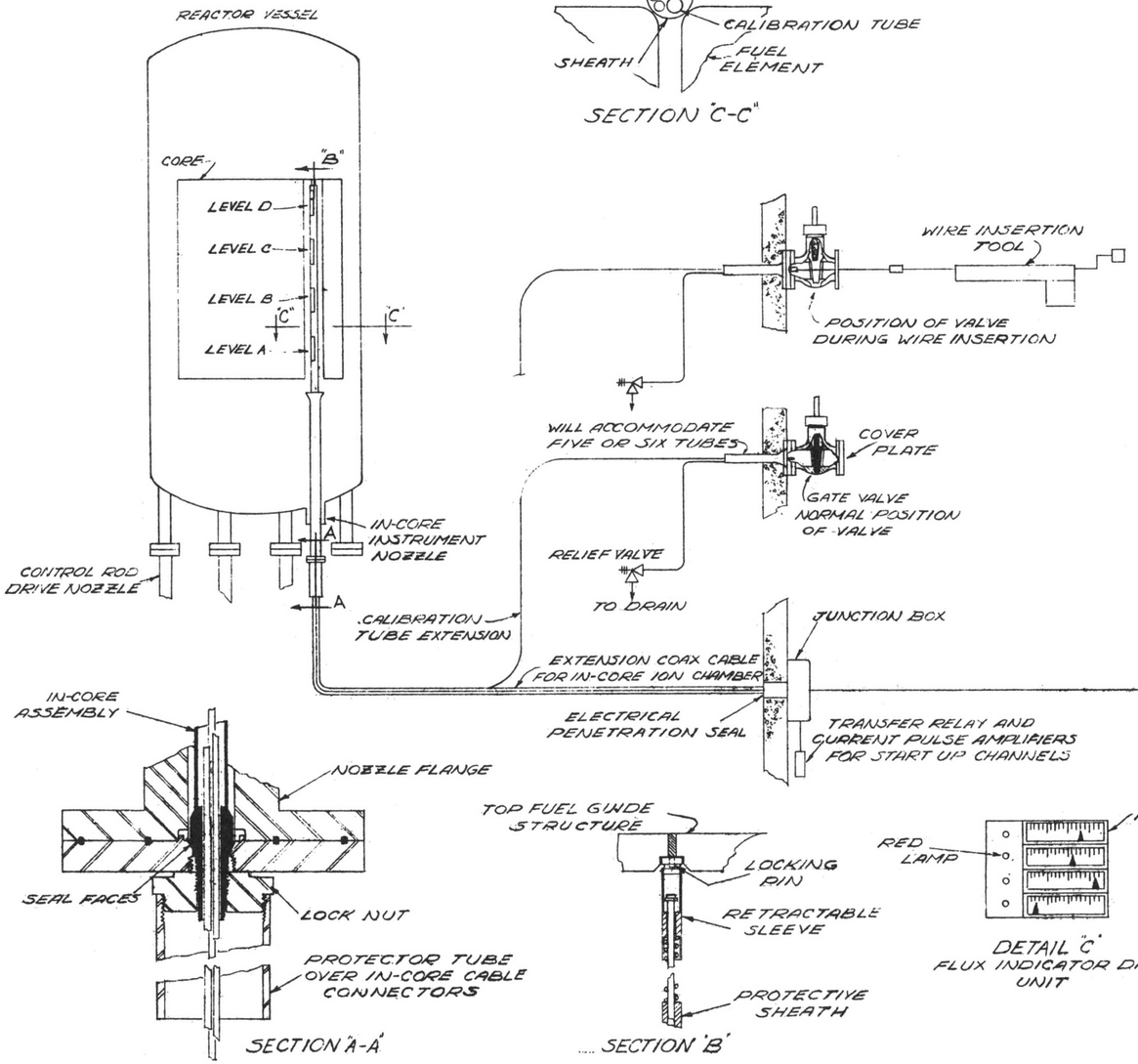
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT




SECTION "C-C"



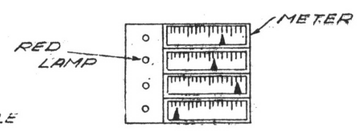




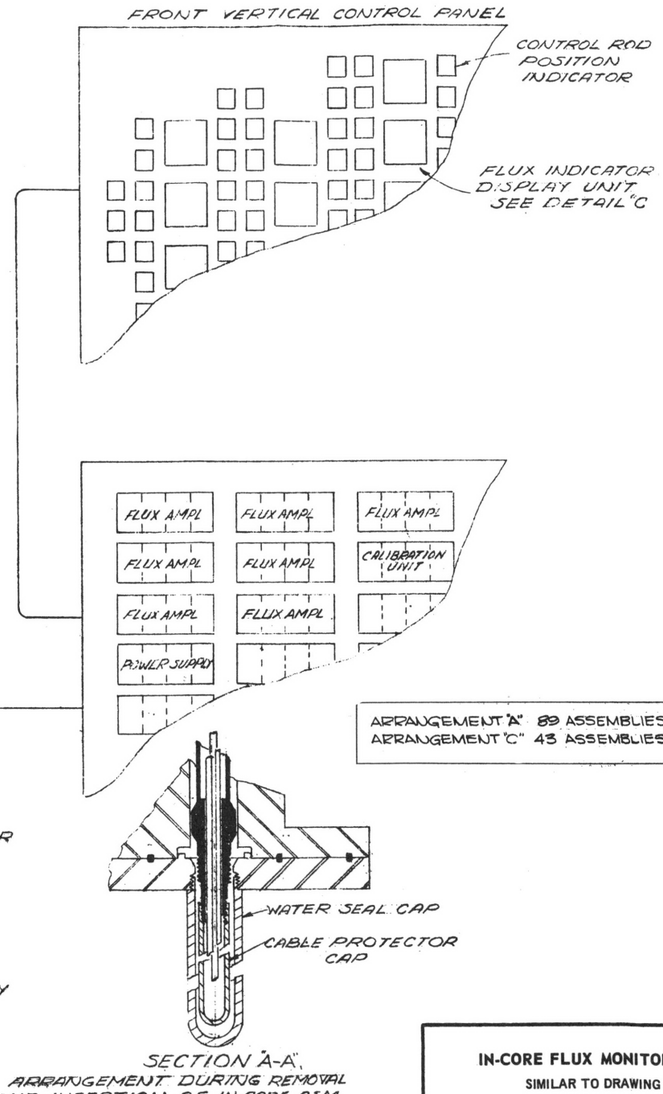
SECTION "A-A"




SECTION "B"




DETAIL "C"
FLUX INDICATOR DISPLAY UNIT




ARRANGEMENT "A" 89 ASSEMBLIES AT 4 LEVELS
ARRANGEMENT "C" 43 ASSEMBLIES AT 4 LEVELS




SECTION "A-A"
ARRANGEMENT DURING REMOVAL AND INSERTION OF IN-CORE ASM.

**IN-CORE FLUX MONITORING SYSTEM**

SIMILAR TO DRAWING NO. 798D862

**1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY**

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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**GENERAL ELECTRIC**

ATOMIC POWER EQUIPMENT DEPARTMENT

| NUMBER OF CHANNELS PER REACTOR | |
|---|---|
| ARRANGEMENT "A" | 8 |
| ARRANGEMENT "C" | 4 |

GROSS FISSION PRODUCT RELEASE MONITOR

| NUMBER OF CHANNELS PER REACTOR | |
|---|---|
| ARRANGEMENT "A" | 2 |
| ARRANGEMENT "C" | 2 |

REACTOR BUILDING VENTILATION EXHAUST MONITOR

| NUMBER OF CHANNELS PER REACTOR | |
|---|---|
| ARRANGEMENT "A" | 2 |
| ARRANGEMENT "C" | 2 |

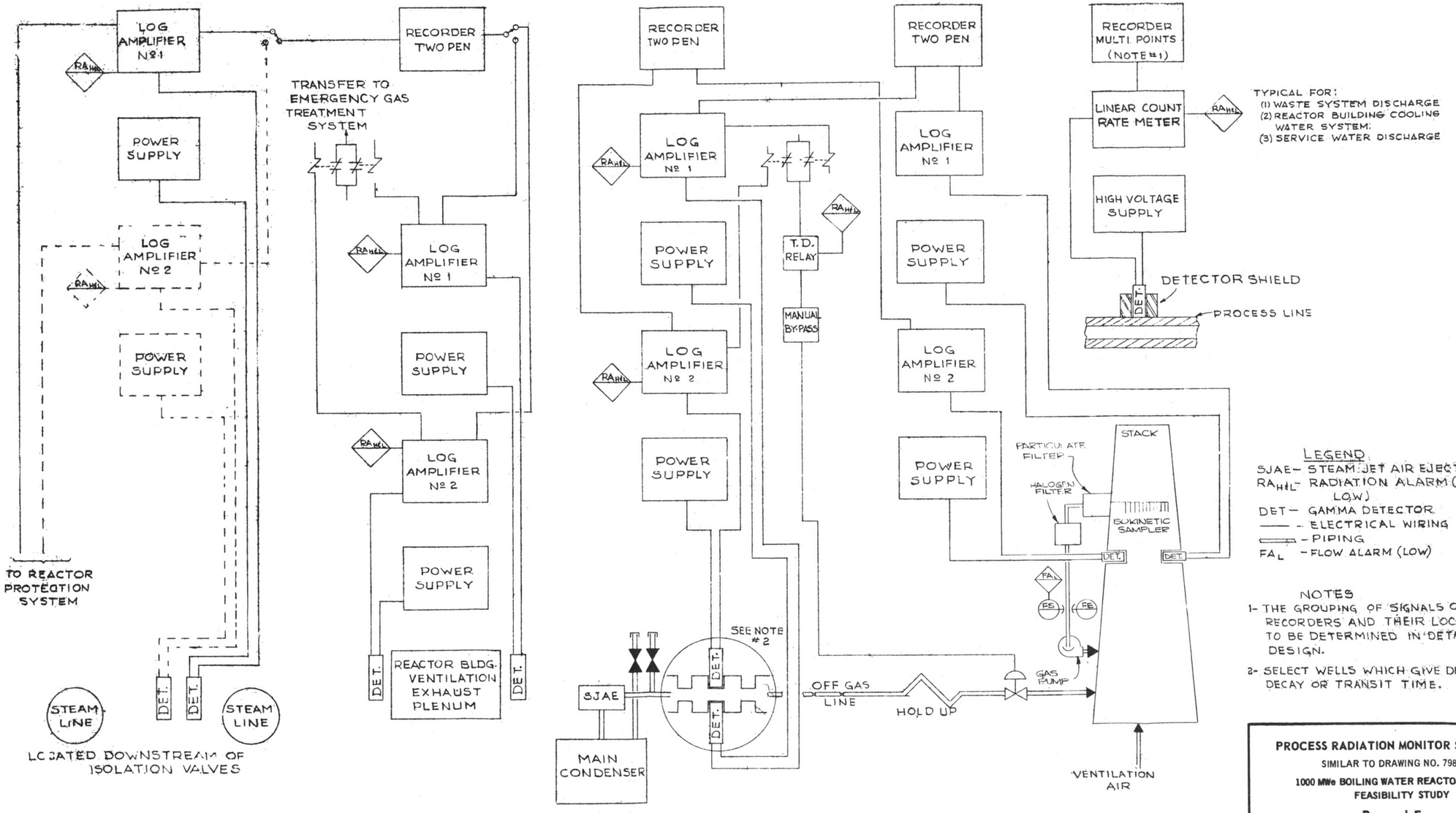
AIR EJECTOR MONITOR Nº 1 AND Nº 2

| NUMBER OF CHANNELS PER PLANT | |
|---|---|
| ARRANGEMENT "A" | 2 |
| ARRANGEMENT "C" | 2 |

STACK MONITOR Nº 1 AND Nº 2

| NUMBER OF CHANNELS PER PLANT | |
|---|---|
| ARRANGEMENT "A" | 6 |
| ARRANGEMENT "C" | 6 |

LIQUID PROCESS MONITORS




LEGEND

SJAE – STEAM JET AIR EJECTOR
$RA_{HL}$ – RADIATION ALARM (HIGH LOW)
DET – GAMMA DETECTOR
—— – ELECTRICAL WIRING
⟺ – PIPING
$FA_L$ – FLOW ALARM (LOW)

NOTES

1- THE GROUPING OF SIGNALS ON RECORDERS AND THEIR LOCATIO TO BE DETERMINED IN DETAIL DESIGN.

2- SELECT WELLS WHICH GIVE DESIRED DECAY OR TRANSIT TIME.

**PROCESS RADIATION MONITOR SYSTEM**

SIMILAR TO DRAWING NO. 798D863

1000 MWe BOILING WATER REACTOR PLANT FEASIBILITY STUDY

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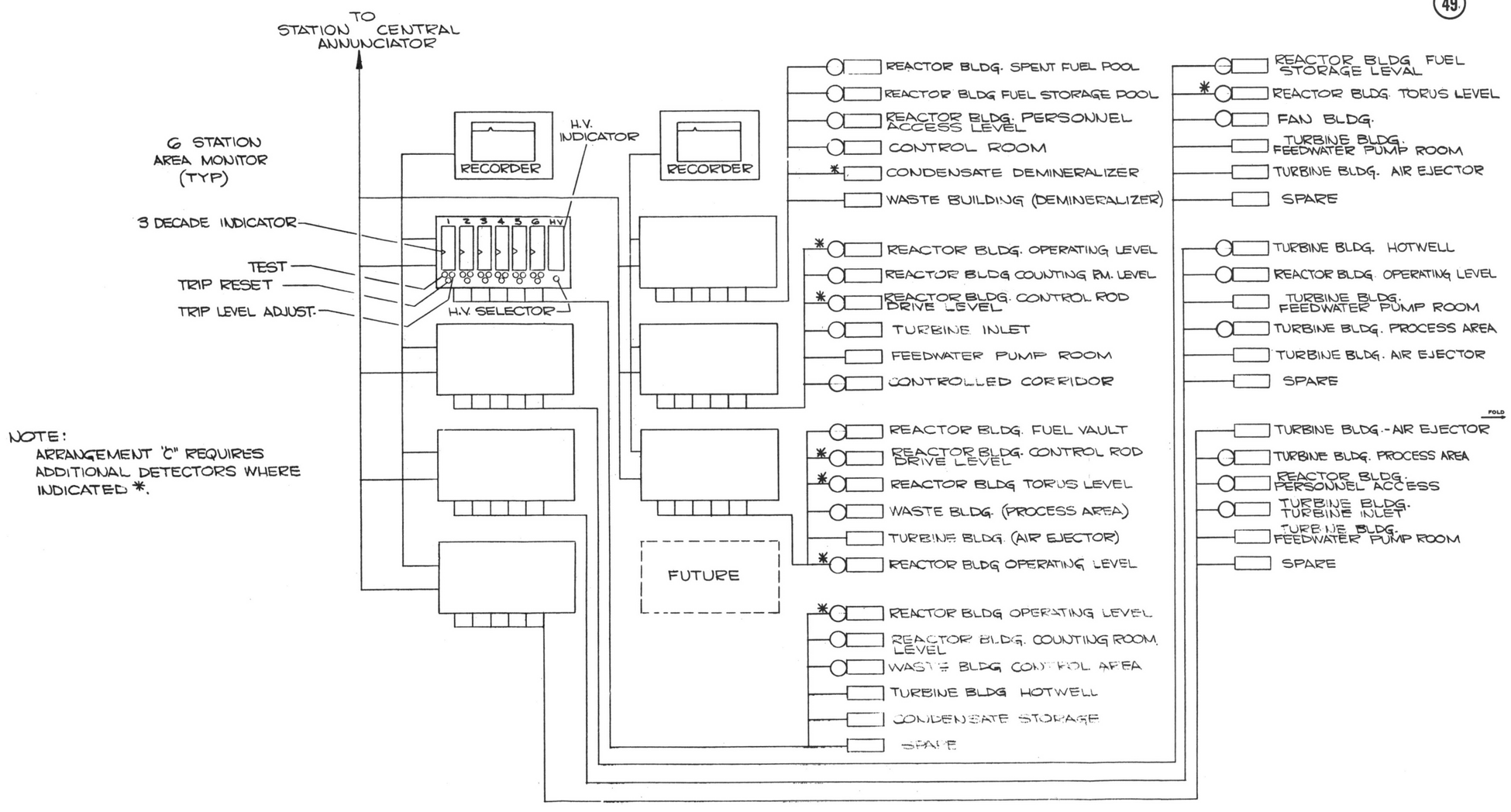
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC

ATOMIC POWER EQUIPMENT DEPARTMENT




NOTE:
ARRANGEMENT "C" REQUIRES ADDITIONAL DETECTORS WHERE INDICATED *.

| | DETECTORS | AREA MONITORS | RECORDERS |
|---|---|---|---|
| ARRANGEMENT "A" | 42 | 7 | 2 |
| ARRANGEMENT "C" | 54 | 9 | 3 |

GPO 812-316-17

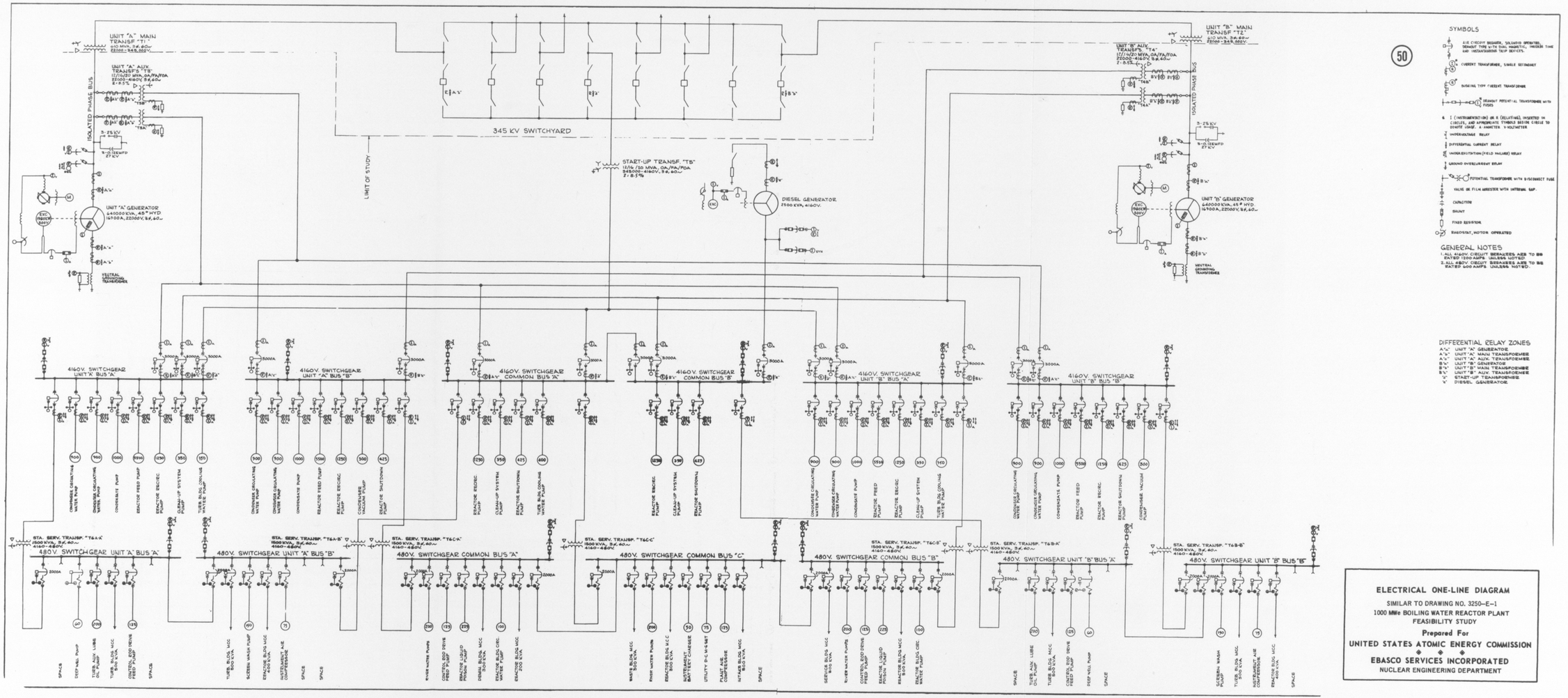
**AREA RADIATION MONITOR SYSTEM**

SIMILAR TO DRAWING NO. 107C4628

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

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SYMBOLS

- 3-Ø CIRCUIT BREAKER, SOLENOID OPERATED, DRAWOUT TYPE WITH THAL MAGNETIC, INVERSE TIME AND INSTANTANEOUS TRIP DEVICES.
- CURRENT TRANSFORMER, SINGLE SECONDARY
- BUSHING TYPE CURRENT TRANSFORMER
- DRAWOUT POTENTIAL TRANSFORMER WITH FUSES
- I (INSTRUMENTATION) OR R (RELAYING) INSERTED IN CIRCLES, AND APPROPRIATE SYMBOLS BESIDE CIRCLE TO DENOTE USAGE. A AMMETER V VOLTMETER
- UNDERVOLTAGE RELAY
- DIFFERENTIAL CURRENT RELAY
- UNDER EXCITATION (FIELD FAILURE) RELAY
- GROUND OVERCURRENT RELAY
- POTENTIAL TRANSFORMER WITH DISCONNECT FUSE
- VALVE OR FLOW ARRESTER WITH INTERNAL GAP.
- CAPACITOR
- SHUNT
- FIXED RESISTOR
- RHEOSTAT, MOTOR OPERATED

GENERAL NOTES

1. ALL 4160V. CIRCUIT BREAKERS ARE TO BE RATED 1200 AMPS. UNLESS NOTED.
2. ALL 480V CIRCUIT BREAKERS ARE TO BE RATED 600 AMPS. UNLESS NOTED.

DIFFERENTIAL RELAY ZONES

- A"x": UNIT "A" GENERATOR
- A"x": UNIT "A" MAIN TRANSFORMER
- A"x": UNIT "A" AUX. TRANSFORMER
- B"x": UNIT "B" GENERATOR
- B"x": UNIT "B" MAIN TRANSFORMER
- B"x": UNIT "B" AUX. TRANSFORMER
- V: START-UP TRANSFORMER
- V: DIESEL GENERATOR

**ELECTRICAL ONE-LINE DIAGRAM**

SIMILAR TO DRAWING NO. 3250-E-1
1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

**Prepared For**

**UNITED STATES ATOMIC ENERGY COMMISSION**

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**EBASCO SERVICES INCORPORATED**
NUCLEAR ENGINEERING DEPARTMENT

| INSTRUMENT FUNCTION / MEASURED VARIABLE | | CONTROLLING | | | | MEASURING | | | | | | | | | SWITCH | | | ALARMS | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | RECORDING | INDICATING | BLIND | CONTROL VALVE | RECORDING | INDICATING | OBSERVATION GLASS | PRIMARY ELEMENT | TEST POINT | TRANSMITTER | INTEGRATOR | AMPLIFIER | SAMPLER | | INDICATING | BLIND | ALARM | ALARM LOW | ALARM HIGH | POWER SUPPLY |
| | | -RC | -IC | -C | -CV | -R | -I | -G | -E | -X | -T | -Q | Am | $S_m$ | | -IS | -S | -A | $A_L$ | $A_H$ | |
| AIR | A | | | | | | | | | | | | | $AS_m$ | | | | | | | |
| CONDUCTIVITY | C | CRC | CIC | | CCV | CR | CI | | CE | CX | | | | $CS_m$ | | CIS | CS | CA | $CA_L$ | $CA_H$ | |
| DENSITY | D | DRC | DIC | DC | DCV | DR | DI | | | DX | | | | | | DIS | DS | DA | $DA_L$ | $DA_H$ | |
| DIFF. PRESS | dP | dPRC | dPIC | dPC | dPCV | dPR | dPI | | | | dPT | | | | | dPIS | dPS | dPA | $dPA_L$ | $dPA_H$ | |
| FLOW | F | FRC | FIC | FC | FCV | FR | FI | FG | FE | FX | FT | | | | | FIS | FS | FA | $FA_L$ | $FA_H$ | |
| HUMIDITY | H | HRC | HIC | | | HR | HI | | HE | | | | | | | | | | | | |
| pH | pH | pHRC | pHIC | pHC | pHCV | pHR | pHI | | pHE | pHX | | | $pHA_m$ | $pHS_m$ | | | | | | | |
| LEVEL | L | LRC | LIC | LC | LCV | LR | LI | LG | | | LT | | | | | LIS | LS | LA | $LA_L$ | $LA_H$ | |
| OXYGEN | $O_2$ | | | | | $O_2R$ | $O_2I$ | | $O_2E$ | | | | | | | | | | | | |
| PRESSURE | P | PRC | PIC | PC | PCV | PR | PI | | | PX | PT | | | | | PIS | PS | PA | $PA_L$ | $PA_H$ | |
| POSITION | Po | | | | | PoR | PoI | | | | PoT | | | | | | PoS | | | | |
| RADIATION | R | | | | | RR | RI | | RE | RX | | | $RA_m$ | $RS_m$ | | | RS | RA | | | RPw |
| SPECIFIC GRAVITY | Sg | | | | | | | | | | | | | | | | | | | | |
| SPEED | Sp | SpRC | SpIC | SpC | | SpR | SpI | | SpE | | | | | | | | | | | | |
| TEMPERATURE | T | TRC | TIC | TC | TCV | TR | TI | | TE | TX | TT | | | | | TIS | TS | TA | TA | TA | |
| TURBIDITY | Tu | | | | | | | | | | | | | | | | | | | | |
| VISCOSITY | V | | | | | | | | | | | | | | | | | | | | |
| VIBRATION | Vb | | | | | | | | | | | | | | | | VbS | VbA | | | |
| WEIGHT | W | | | | | | | | | | | | | | | | | | | | |
| WEIGHT FACTOR | Wf | WfRC | WfIC | | WfV | | | | | | | | | | Suffix sub H or L may be added to designate High or Low. | | | | | | |
| WATER | Wa | | | | | | | | | | | | | | | | | | | | |

## TYPICAL INSTRUMENT SYMBOLS

LOCAL PANEL MOUNTED — CONTROL ROOM PANEL MOUNTED
TBR-1 — PANEL NO. — P-7
P1 / 123 — SERVICE — PRC
INSTRUMENT NO — 2-46
CASE NO (For multiple functions only)
SINGLE SERVICE

SC
SAMPLE COOLER

PNEUMATIC — RMC
ELECTRICAL — RMS
REMOTE MANUAL CONTROLLER

TBR-18 PS 12 PAL — TBR-16 TS 19 TAH — P-22 FRS 123 FAL — P-7 A
MULTIPLE PURPOSE ANNUNCIATOR

### LINE SYMBOLS

PRIMARY FLOW
SECONDARY FLOW
ASME BOILER CODE PIPING
INSTRUMENT PROCESS PIPING
INSTRUMENT AIR LINES
INSTRUMENT ELECTRICAL LEADS
INSTRUMENT CAPILLARY TUBING
FLEXIBLE HOSE

### TEMPERATURE ELEMENTS

TE — THERMO-COUPLE
TE — RESISTANCE THERMOMETER
TX — THERMAL WELL
TI — THERMOMETER
TI — THERMOMETER WITH CAPILLARY

### FLOW DEVICES

FE — NOZZLE
FE — ORIFICE
RO — RESTRICTING ORIFICE
FI — IN THE LINE FLOW INDICATOR
FG — SIGHT FLOW INDICATOR

### ALARMS

(PANEL NO.) P-8 — A — VISIBLE & AUDIBLE ALARM (ANNUNCIATOR)
P-17 — PILOT LIGHT
TYPICAL — LAH — LEVEL ALARM (HIGH) (LOW SIMILAR)

### TRANSMITTERS

T — LOCAL MOUNTED
IT — INDICATING

### MISCELLANEOUS

LOCATION 123F419 / COORDINATE 2-D — REFERENCE TO DRAWING NO. SHOWN
FURNISHED WITH ASSOC. EQUIPMENT
OPEN DRAIN
1925-6"-D — LINE NO. / LINE SIZE / LINE SPECIFICATION — LINE

NOTES:
1. FIRST LETTER OF INSTRUMENT SYMBOL SHALL BE PROCESS VARIABLE.
2. A SWITCH IS A DEVICE CAPABLE OF OFF - ON ACTION.
3. AN ALARM IS THE ANNUNCIATOR, SIGNAL HORN OR OTHER DEVICE, ACTUATED BY SWITCH CONTACTS.

### OPERATORS

MOTOR: MO — ELECTRIC; AO — AIR; HO — HYDRAULIC; MO — DAMPER WITH OPERATOR
OTHER: DIAPHRAM; SOLENOID; FLOAT; MANUAL TRIP & RESET
DIAPHRAM WITH POSITIONER

### INSTRUMENT POWER SUPPLY

115 VAC ELECTRICAL SUPPLY — 30 PSI INST. AIR AIR SUPPLY
P/E PNEUMATIC ELECT — E/P ELECT PNEUMATIC CONVERTERS

### INSTRUMENT AIR SET

PCV — 30 PSIG — 100 PSIG INST AIR

### VALVES

ANGLE — GATE — GLOBE — CHECK — SAFETY OR RELIEF — RM. (REMOTE MANUAL)
NORMALLY CLOSED END — ACTUATED CLOSED END — 3-WAY — 4-WAY
OPENS — CONTROL — CLOSES
REGULATING & PRESSURE RED. VALVE (ON AIR FAILURE)
PdI — PCV — AIR LOCK — A.L. — 2"-26 — PdS — COMBINATIONS
OPEN - LOCKED - CLOSED L.O. L.C.
OPEN - NORMALLY - CLOSED POSITIONS
BUTTERFLY — PLUG VALVE

**INSTRUMENT SYMBOLS**

SIMILAR TO DRAWING NO. 943A863

1000 MWe BOILING WATER REACTOR PLANT
FEASIBILITY STUDY

Prepared For
UNITED STATES ATOMIC ENERGY COMMISSION

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

GPO 812—316—4